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DAY THREE

1989 October 25
BR2 Research Reactor Modifications:

Experience gained from the BR2 Beryllium Matrix Replacement and Second Matrix Surveillance Programme.

E. KOONEN

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ABSTRACT

BR2 Research Reactor Modifications:
Experience gained from the BR2 Beryllium Matrix Replacement and Second Matrix Surveillance Programme.

The unloading and replacement of the first beryllium matrix of the BR2 reactor took place in 1979-80. By this time the maximum fast fluence in the hottest channel had reached about $8 \times 10^{22} \frac{n(>1\text{ MeV})}{\text{cm}^2}$. Dimensional stability and swelling of the beryllium matrix have been investigated. The swelling, mainly due to the formation of gas atoms, was found a nearly linear function of the fast fluence up to a value of $\sim 6.4 \times 10^{22} \frac{n(>1\text{ MeV})}{\text{cm}^2}$ at the temperature of $\sim 50$ °C normally existing in the matrix. For higher values of the fast fluence, several observations showed an accelerated increase of the swelling. Consequently the maximum allowed fast fluence for the second beryllium matrix has been limited to $6.4 \times 10^{22} \frac{n(>1\text{ MeV})}{\text{cm}^2}$.

The surveillance programme of the second BR2 beryllium matrix mainly concerns direct observations and measurements on the beryllium matrix itself. Dimensional measurements allow comparison of the relative swelling in axial and radial directions with the dilatation coefficients obtained for the first matrix. Visual inspections are performed on the inner surfaces of the reactor channels in order to record the beginning and the evolution of cracks and to measure the total length of cracks in function of the fast fluence. Irradiations and measurements are also performed on test samples coming from the heats which served for the manufacturing of the second matrix. Isochronal pre- and post-annealing swelling of some samples has been measured. These measurements are in good accordance with the model of a threshold temperature for accelerated helium bubble induced swelling of beryllium, as a linearly decreasing function of the fast fluence.
1. INTRODUCTION

Beryllium is used as a neutron moderator and reflector under the form of a reactor core matrix in the materials testing reactor BR2 at CEN/SCK, Mol.

Beryllium has been chosen because of its excellent neutron thermalisation (provided by low absorption and high scattering capabilities). It also combines a low density with high specific heat and thermal conductivity. The major drawback seems to be the embrittlement and swelling under irradiation.

2. THE BR2 BERYLLIUM MATRIX

2.1. Properties of the beryllium under use at BR2

The beryllium used for the matrix has been manufactured by vacuum hot-pressing of impact attritioned powder of 100 mesh at least. The average grain size is \( \leq 25 \mu m \). The minimum bulk density is 99 % of theoretical. As far as nuclear requirements are concerned, the total danger sum does not exceed 75 % for chemical analysis. This sum is defined as follows:

\[
\text{Total danger} = \sum W(i) \times K_s(i)
\]

where : \( W(i) \) = % weight concentration of impurity element i
\( K_s(i) \) = ratio of the macroscopic thermal absorption cross-section of impurity element i to that of beryllium.

2.2. Description of the beryllium matrix

The central region of the reactor pressure vessel contains a matrix of beryllium metal which acts to position all fuel elements, control rods, beryllium plugs and experiments. Its general shape is a cylinder, 1100 mm diameter and 914 mm high (fig. 1).

The matrix is mainly an assembly of a great number of prismatic irregular hexagonal bars, each with a cylindrical bore which forms the channel. There are 79 assemblies : sixty-four 85 mm standard channels, ten 50 mm reflector channels and five 200 mm channels.
The central 200 mm channel is vertical. All other channels are inclined and form a hyperboloidal arrangement around the central one. This feature combines a very compact core with the requirement of sufficient space for individual access to all channels through penetrations in the top cover of the reactor vessel. The maximum angle of a channel with the vertical is 10°24'.

The beryllium bars are supported and maintained in position by means of stainless steel extension pieces, rigidly keyed to the upper and lower ends of the beryllium bars (fig. 2). The extension pieces are essentially cylindrical tubes, completed at 2 levels with hexagonal pieces which fit a honeycomb pattern and grant the correct spacing between the beryllium bars. The extreme spacing pieces are in contact with the pressure vessel and assure radial stability of the matrix.

The assemblies made of beryllium bars and their extension pieces are supported by stainless steel tubes resting on a stainless steel grid which is rigidly connected to the lower part of the reactor vessel. These supporting tubes assure the vertical positioning of the beryllium matrix.

Between the upper end of the upper extension pieces and the individual penetrations in the reactor top cover, stainless steel guide tubes are installed in order to complete the reactor channels. Springs located in the reactor top cover assure axial stability of the matrix.

3. THE MATRIX REPLACEMENT

The BR2 reactor was put into service with an experimental load in January 1963.

In 1974, a visual inspection of the inner surfaces of the reactor channels showed signs of cracks, not observed three years earlier during the previous examination.

A safety evaluation concluded that further operation was possible without undue risk to the operators and the public. It was therefore decided to continue reactor operation while proceeding with a surveillance programme and preparing the matrix replacement operation.
The surveillance programme mainly concerned regular visual inspections and analysis of all operational incidents for a possible implication of the beryllium matrix. The visual observations showed a continual evolution of the number and the length of cracks (fig. 3), but this matrix degradation never affected the normal operation of the reactor.

Another important issue was the status of the reactor pressure vessel. As an inspection of the inner surface is only possible when the matrix is unloaded, the question arised whether or not a replacement of the pressure vessel had to be foreseen. Based on the opinions and advices of several experts, it was decided no to do so. In accordance with the National Safety Authority, an in-service inspection programme, to be carried out after the unloading of the first matrix, was established.

Planning and logistic preparation of the replacement operation were undertaken. New constituent parts of the reactor channels were ordered; all steps of the fabrication process up to delivery were accompanied by an extensive quality control programme.

After delivery, the channels have been assembled on site and loaded in a mock-up of the BR2 pressure vessel:

- first by direct manual access in order to verify geometrical compatibility and to test the operational sequences.
- later on by remote handling in order to qualify tools, remote handling equipment and working procedures, and to train the operators.

The reactor was finally shutdown by the end of 1978. At this stage the matrix was heavily cracked and the "hottest" channel had reached a fast fluence of $7.9 \times 10^{22} \frac{n(> 1 \text{MeV})}{\text{cm}^2}$.

Before unloading, extensive dimensional measurements were carried out in the old matrix. The analysis of the results not only gave a better understanding of the way global deformation of the matrix had proceeded, but also helped to establish surveillance criteria for the new matrix.

Most of the replacement work was performed by BR2 operation staff.

Due to extensive preparation, this task has been carried out with only minor radiation exposure of the operators: the total dose by all the personnel involved was only 1.4 times the dose received during an equivalent routine reactor operation period. No problems of beryllium contamination has been encountered.
The main steps of the replacement operation were the following [3]:

- unloading of the guide tubes and removing of the top cover of the pressure vessel.

- unloading of the channels: serious difficulties were encountered. The embrittlement of the beryllium caused loss of material during unloading. In the central region of the matrix, some channels couldn't be removed as an entity because the beryllium hexagons fell to pieces when they were withdrawn. In addition some highly active particles, trapped during reactor operation by the interstices in the matrix, were released. This led to high radioactivity in the sub-pile room and contamination of the reactor pool. In order to cope with these difficulties, the working procedures had to be modified, unloading tools adapted and new equipment prepared.

- cleaning of the pressure vessel: in particular the dropped beryllium pieces and other particles had to be removed.

- inspection of the pressure vessel (Al 5052): remote visual examinations, dimensional controls and an ultrasonic inspection were carried out. The general conclusion was that the pressure vessel was in an excellent state of conservation and had not to be replaced.

- loading of the new reactor channels, re-installation of other reactor internals and the pressure vessel top cover: extensive dimensional measurements and controls took place in the new matrix, in order to establish the initial conditions for the surveillance programme.

In addition, maintenance and modernisation work have been executed to improve the performances and the safety of the installation, for instance:

- an in-service inspection of the primary circuit.

- modernisation of the reactor control instrumentation.

- maintenance and improvement of the control rods.

The BR2 Safety Report has been revised in order to obtain the license for further operation.

Routine operation of the reactor was resumed in July 1980.
4. EXPERIENCE GAINED WITH THE FIRST BERYLLIUM MATRIX

4.1. Nuclear characteristics

Under the influence of irradiation, beryllium undergoes several reactions which lead to the formation of gas atoms. The most important one is the 
(n,2n) reaction above a threshold energy of about 2.7 MeV:

\[ {}_4^{9}\text{Be} + {}_0^1\text{n} \rightarrow {}_4^8\text{Be} + {}_0^2\text{n} \]

with the subsequent decay of half-life 2.10^{-15}s to:

\[ {}_4^8\text{Be} \rightarrow {}_2^4\text{He} + 94 \text{keV} \]

The second reaction of importance is the (n,a) reaction with a threshold energy of about 0.7 MeV and it gives rise to the following sequence:

\[ {}_4^9\text{Be} + {}_0^1\text{n} \rightarrow {}_2^4\text{He} + {}_2^6\text{He} \]

\[ {}_2^6\text{He} \rightarrow {}_2^4\text{He} + {}_1^6\text{Li} \]

\[ {}_3^6\text{Li} + {}_0^1\text{n}_{\text{thermal}} \rightarrow {}_2^4\text{He} + {}_1^1\text{H} \]

\[ {}_1^3\text{H} \rightarrow {}_2^4\text{He} \]

\[ {}_2^3\text{He} + {}_0^1\text{n}_{\text{thermal}} \rightarrow {}_1^1\text{H} + {}_1^1\text{H} \]

Again this reaction sequence leads to the production of two helium atoms in addition to one tritium atom and one proton.

A third reaction occurs with photons of energy higher than 1.67 MeV:

\[ {}_4^9\text{Be} + \gamma \rightarrow {}_2^4\text{He} + {}_0^1\text{n} \]

This reaction provides the start-up source.
The neutron absorbing atoms formed ($^6\text{Li}$, $\sigma_a = 940$ b; $^3\text{He}$, $\sigma_a = 5300$ b) lead to a neutron poisoning. The $^6\text{Li}$-poisoning is saturated after about one year of operation (~200 days at nominal power), leading to a reactivity loss of about 1.6 $. The $^3\text{He}$-poisoning increases at each reactor shutdown: when the reactor is under operation, $^3\text{He}$ is transformed back to $^3\text{H}$, but at the same time more tritium is produced. Due to the long half-life of tritium, saturation is not reached during the lifetime of the matrix and the $^3\text{He}$-poisoning increases proportionally to the shutdown duration, the slope being a linear function of the fast neutron fluence. Experimentally the following relation was established:

$$\frac{d\rho}{dt}(3\text{Re}) \sim -8.15 \times 10^{-7} \times E \ [\$/d]$$

(1)

with: $\rho =$ reactivity in $\$ \text{ due to the } ^3\text{He-effect}$

$E =$ energy produced in MWd

This linearly growing anti-reactivity limits the duration of the shutdown and the maximum allowed shutdown duration diminishes with the lifetime of the beryllium matrix.

4.2. Swelling characteristics

The swelling, essentially due to the formation of the gas atoms particularly helium, which do nearly not diffuse at the temperatures normally realized in the matrix (40-50 °C), is also a nearly linear function of the fast neutron fluence. At higher fluences ($> 6.4 \times 10^{22} \text{ } n(> 1 \text{ MeV}) \text{ cm}^{-2}$), however, the diffusion seems to accelerate. The gas atoms then migrate to lattice imperfections and grain boundaries where they precipitate to form bubbles, causing a swelling faster than proportional to the fast fluence.

Several observations made on the first matrix support this qualitative model of two distinct stages in the swelling process [1,2,4]:

- the increase of the inner diameters of the reactor channels has been found a linear function of the fast fluence up to $~6.4 \times 10^{22} \text{ } n(> 1 \text{ MeV}) \text{ cm}^{-2}$

Above this fluence, the diameters increase more rapidly (fig. 4).
the elongation of the beryllium hexagons shows a similar linear dependence until \(6.4 \times 10^{22} \frac{n(>1\text{ MeV})}{\text{cm}^2}\), and a quicker increase above.

the evolution of the length of cracks, observed at the inner surface of the channels from about \(4 \times 10^{22} \frac{n(>1\text{ MeV})}{\text{cm}^2}\), is also fairly linear with the fast fluence, even beyond \(6.4 \times 10^{22} \text{n/cm}^2\) (fig. 3).

the tritium concentration in the primary water began to increase at a time when the maximum fast fluence in the matrix reached \(~6.5 \times 10^{22} \frac{n(>1\text{ MeV})}{\text{cm}^2}\).

Information gathered from the literature [5-17] confirms that helium bubble swelling is the predominant mechanism resulting in dimensional changes of irradiated beryllium and that the point at which the helium bubbles start to agglomerate is related to a swelling threshold temperature which decreases when the fast fluence (and the corresponding helium concentration) increases. This threshold temperature also varies with the type of material (hot pressed, extruded, pressureless sintered with various densities, oxide contents and chemical impurities).

As a consequence of all these observations and information, the maximum admissible fast fluence for the second beryllium matrix has been limited to \(6.4 \times 10^{22} \frac{n(>1\text{ MeV})}{\text{cm}^2}\), in accordance with the National Safety Authority.

5. SURVEILLANCE PROGRAMME OF THE SECOND BERYLLIUM MATRIX

The main objective of this programme is to verify that the second matrix behaves like the first one, so that its lifetime can be extended up to a fast fluence of \(6.4 \times 10^{22} \frac{n(>1\text{ MeV})}{\text{cm}^2}\).

This programme is based on:
- inspections and measurements on the second matrix.
- irradiation of test samples (see section 6).

Direct measurements on the matrix itself mainly concern dimensional measurements (comparison of the relative swelling in axial and radial directions with the dilatation coefficients obtained for the first matrix). Figures 5 and 6 give, as an example, the results of diametrical measurements in channel B0.
Visual inspections are performed on the inner surfaces of the reactor channels in order to record the beginning and the evolution of cracks, to measure the total length of cracks as a function of the fast fluence and to detect possible loss of material.

In addition, tritium and beryllium concentrations in the primary water are regularly determined and operational incidents are carefully analysed for a possible implication of the beryllium matrix.

Up to the present time, the results are in good accordance with what is expected.

6. PRE- AND POST-ANNEALING SWELLING OF IRRADIATED BERYLLIUM SAMPLES

The main objective is to compare the swelling of the samples with that of the reactor channels. By doing so, one can determine to what extent the swelling of the matrix is hindered or prevented by the mechanical constraints of the reactor core structure. Hindered swelling can lead to an increase of stresses and cracks. Visual inspections of the samples are also done to compare the moment of appearance and the evolution of the cracks. Another objective is to get a better understanding of the swelling process.

The material for the samples has been taken from the beryllium cores removed from the beryllium hexagons, while manufacturing the reactor channels.

Twenty cylinders (diameter $\approx 15$ mm, length $\approx 100$ mm) have been fabricated from each of five heats used for different types of reactor channels.

Thirty-six of these samples have been irradiated in baskets in the axis of standard six-plate fuel elements loaded in the central crown of the reactor core.

The loading scheme alternates samples and activation monitors which measure the integrated equivalent fission flux. The axial flux distribution has been determined by means of calorimetric probes. Visual inspections, dimensional measurements and replacement of the activation monitors took place about every $0.8 \times 10^{22} \frac{n(>1\text{ MeV})}{\text{cm}^2}$.

In 1982, some samples have not been reloaded in order to keep some witnesses of the fast fluence reached ($\sim 0.8 \times 10^{22} \frac{n(>1\text{ MeV})}{\text{cm}^2}$).

Unfortunately, this has not been done for $1.6$ and $2.4 \times 10^{22} \frac{n(>1\text{ MeV})}{\text{cm}^2}$.
In 1988, two baskets containing eight samples have been unloaded at about
2.8 \times 10^{22} \frac{n(>1 \text{ MeV})}{\text{cm}^2}. Four samples were used in the annealing programme, the
other four will serve as witnesses of the fast fluence reached at this
point.

Isochronal annealing programmes at 400 °C and 600 °C for 24 h have been
performed on two groups of four samples. Each group comprised two samples
irradiated to about 0.8 \times 10^{22} \frac{n(>1 \text{ MeV})}{\text{cm}^2} and two samples which had accu-
mulated about 2.4 \times 10^{22} \frac{n(>1 \text{ MeV})}{\text{cm}^2}.

Figure 7 gives the results of the elongation measurements (pre- and
post-annealing):
- Regression analysis of the data before annealing is in good accordance
with the elongation measurements on the reactor channels (up to 
3.0 \times 10^{22} n(>1 \text{ MeV})/\text{cm}^2).
  The relation found is (for irradiation at about 50 °C):

\[ \frac{\Delta L}{L} = 0.00185x(\varnothing t) \] (2)

where \( \varnothing t = \) fast fluence (\( > 1 \text{ MeV} \)) in \( 10^{22} n/\text{cm}^2 \).

- Annealing at 400 °C for 24 h showed a slight increase (35 %) in swelling
for the samples, irradiated up to 0.8 \times 10^{22} \frac{n(>1 \text{ MeV})}{\text{cm}^2}, and an increase of
\sim 70 % for the samples irradiated up to 2.8 \times 10^{22} \frac{n(>1 \text{ MeV})}{\text{cm}^2}.
- Annealing at 600 °C for 24 h gave an increase in swelling of \sim 150 % for
the first group of samples and \sim 270 % for the second group.

Figure 8 gives the results of the diametrical measurements (pre- and post-
annealing):
- Regression analysis of the data before annealing is in discordance with
the inner diameter measurements in the reactor channels. The relation
found for the samples is (irradiation temperature \sim 50 °C):

\[ \frac{\Delta D}{D} = 0.0034x(\varnothing t) \] (3)

 whereas the regression for the reactor channels is:

\[ \frac{\Delta D}{D} = 0.0014x(\varnothing t) \] (4)

where \( \varnothing t = \) fast fluence (\( > 1 \text{ MeV} \)) in \( 10^{22} n/\text{cm}^2 \).
In the reactor matrix, diametrical swelling is of course hindered by the core structure. In this way, stresses are induced in the beryllium which can lead to cracks earlier than expected when only considering bubble swelling. It should be noted that the diametrical swelling curves show a slight non-linear behaviour compared to the elongation curves.

- Annealing at 400 °C for 24 h gave no significant increase in swelling. Only for sample 24b, with the highest fluence \(2.87 \times 10^{22}\), a 25 % increase was obtained.
- Annealing at 600 °C for 24 h caused a much more important increase: ~90 % for the two samples with a fluence of ~2.8x10^{22} and ~60 % for the two other samples.

It is very interesting to note that these results are in good accordance with the model of two distinctive stages in the swelling behavior of beryllium, separated by a threshold temperature which decreases in function of the fast fluence [2,6,9]. Several data by Rich [5], Tromp [6] and Beeston [9] have been used to establish this model. The curve in fig. 9 is drawn on the assumption that the critical volumetric swelling degree is ~3%, and that the threshold swelling temperature decreases about linearly with the fast fluence. One must note that our data concern post-annealing swelling, but nevertheless they are in good accordance with the model. Indeed, the samples irradiated up to ~2.8x10^{22} \(\frac{n(>1 \text{ MeV})}{\text{cm}^2}\) show an important increase in swelling when annealed at 600 °C, as compared to the behaviour after the 400 °C annealing during the same duration of 24 h. For this fast fluence, fig. 9 gives a swelling threshold temperature of ~400 °C.

Concerning the samples irradiated to ~0.8x10^{22} \(\frac{n(>1 \text{ MeV})}{\text{cm}^2}\), annealing at 400 °C gave a small increase, whereas annealing at 600 °C nearly doubled the swelling. The threshold temperature at this fast fluence is expected to be ~630 °C.

These considerations give an additional justification of the maximum fast fluence value, up to which the second beryllium matrix may be irradiated.

N.B.: To our knowledge, the highest fast fluence for irradiation in the high temperature region was obtained at EBR-II with hot-pressed beryllium irradiated at ~470 °C up to a fluence of \(1.2 \times 10^{22} \frac{n(>1 \text{ MeV})}{\text{cm}^2}\) [13].
7. CONCLUSIONS

The replacement of the first BR2 beryllium matrix was carried out in 1979-80. This matrix had been irradiated up to a fast fluence of $7.9 \times 10^{22} \frac{n(>1 \text{ MeV})}{\text{cm}^2}$. Investigation of the dimensional stability and the swelling of the first matrix led to the definition of a maximum allowed fast fluence for the second matrix. Agreement has been reached with the National Safety Authority concerning this value: $6.4 \times 10^{22} \frac{n(>1 \text{ MeV})}{\text{cm}^2}$.

The surveillance programme of the second beryllium matrix is executed on a regular basis in order to allow operation up to this maximum allowed fast fluence.

The study programme on swelling under irradiation of beryllium samples is helpful in getting a better understanding of the physical processes involved. In this way an additional justification of the maximum allowed fast fluence up to which the beryllium matrix may be irradiated, is obtained.

Acknowledgments:
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FIG. 1 General view of the BR2 reactor

FIG. 2 BR2 STANDARD CHANNEL $\phi$4
CRACKED CHANNELS IN THE FIRST Be-MATRIX.  

**Fig. 3**

![Graph showing the relationship between length of cracks and number of cracked channels.](image)

**Fig. 4**

![Graph showing the relationship between fast fluence and mean diametrical swelling.](image)
DIAMETERS IN CHANNEL B60, AXIAL DISTRIBUTION
AT 0 DEGREES, FROM CYCLE 09/80 UP TO 03/89

DIAMETERS IN CHANNEL B60, RADIAL DISTRIBUTION
AT LEVEL +0, FROM CYCLE 09/80 UP TO 03/89
PRE- & POST-ANNEALING SELLING OF IRRADIATED Be-SAMPLES

Figure 7

Figure 8
SWELLING THRESHOLD TEMPERATURE OF Be AS A FUNCTION OF FAST FLUENCE.

Pre-Post-anneal: volum.swel.

- ■ 24c 0.46% 0.54%
- ● 8c 0.52% 0.58%
- ● 25a 2.39% 2.75%
- ● 24b 2.45% 3.30%
- ○ 34c 0.54% 1.06%
- ○ 7c 0.62% 1.12%
- + 10a 2.26% 5.2%
- ● 9b 2.6% 6.0%

BR2 1st matrix

Fast Fluence (>1MeV) 10E22 n/cm2

Temperature °C

Pre-Post-anneal: volum.swel.

- ■ 24c 0.46% 0.54%
- ● 8c 0.52% 0.58%
- ● 25a 2.39% 2.75%
- ● 24b 2.45% 3.30%
- ○ 34c 0.54% 1.06%
- ○ 7c 0.62% 1.12%
- + 10a 2.26% 5.2%
- ● 9b 2.6% 6.0%

BR2 1st matrix
CORROSION PROBLEM IN THE C.R.E.N.K.

TRIGA MARK II RESEARCH REACTOR.

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CORROSION PROBLEM IN THE C.R.E.N.K.*
TRIGA MARK II RESEARCH REACTOR.

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1. ABSTRACT

In August 1987, a routine underwater optical inspection of the aluminum tank housing the core of the CRENK Triga Mark reactor, carried out to update safety condition of the reactor, revealed pitting corrosion attacks on the 8 mm thick aluminum tank bottom. The paper discusses the work carried out by the reactor staff to dismantle the reactor in order to allow a more precise investigation of the corrosion problem, to repair the aluminum tank bottom, and to enhance the reactor overall safety condition.

2. INTRODUCTION

The Centre Régional d'Etude Nucléaires de Kinshasa (C.R.E.N.K.) Triga Mark II reactor, located in Kinshasa (Zaïre), went critical on March 24, 1972. It is a 1 Mw(th) machine with a pulsing capabilities up to 1,600 MW(th). The reactor core is located in a 8 mm thick, 7 m height cylindrical aluminum tank with a diameter of 2 m, [1].

In August 1987, a routine underwater optical inspection of the aluminum tank housing the reactor core, carried out to update safety condition of the reactor, revealed pitting corrosion attacks on the 8 mm thick aluminum vessel bottom.

Twelve pits were identified using an above water telescope. The corrosion problem seemed serious, one of the pits having an apparent depth of 5 mm, as could be estimated using long distance measurements from the top of the reactor, [2].

Concerning pitting corrosion one must know something of the probability of pitting, the rate of penetration, the shape of the pit depth distribution curve in order to tell whether repair is necessary, or worth while, or whether general failure is impending, [3].

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In order to gain these knowledges and to identify the causes of the pitting corrosion it was necessary to dismantle the reactor in order to investigate more fully the corrosion problem.

2. DISMANTLING THE REACTOR CORE

To dismantle a reactor core is always a difficult, and at times, dangerous task. It was carried out in two months time by a 5 men local team, fortunately without serious incident but with some difficulties.

Two serious problems were encountered. The first one was the presence of a cocked fuel element which was very difficult to disengage. The team had to lower the water level to approximately 1 m above the core to have a better grip on the fuel element.

The second problem was related to the rotary rack assembly, which turned out to be hotter than anticipated, posing a serious radiological hazard to the staff.

3. STATE OF THE ALUMINUM REACTOR TANK

If the reactor has been in service since 1972, the tank housing the core was in fact manufactured in 1965 in Austria using 6061-T67 aluminum. The tank was coated on the outside with a layer of pitch followed by a layer of tar paper. It was then embedded in an above ground high density concrete structure, 7.7 m height. It contains about 22 cubic meter of demineralized water, \[ 1 \text{J} \]. Figure 1 gives the CRENK Triga Mark II reactor cross section.

The inspection of the aluminum tank carried out after the removal of the core structure showed a less serious corrosion problem than was anticipated. Only 8 pits were identified as resulting from the corrosion of the aluminum plate. The most advanced corroded spot had a depth of 2.5 mm. The range of the depths of the 8 pits was between 1 mm and 2.5 mm. The pit shape vary somewhat but the mouth of a pit tended to be circular, while the cross section was roughly hemispherical. Figure 2 gives the location of the 8 pits that were identified, relative to the ground position of some elements of the core structure.

To ascertain that no corrosion was taking place from the exterior of the tar coated aluminum tank, its thickness was determined ultrasonically from the bottom up to a height of 2 m above the core structure. The result of the ultrasonic survey indicated that the area
Fig. 1 Vertical cross section of the C.R.E.N.-K Reactor.
(Figure in mm)
covered was free of any corrosion problem. It was discovered however that the area covered by the thermal column has a thickness a bit less than the standard 8 mm.

4. EVALUATION OF THE CAUSES OF THE CORROSION PROBLEM

To really enhance the safety of the reactor it is important to know the causes of the corrosion of the aluminum tank bottom.

Many factors can be at the origin of pitting corrosion of aluminum alloy. It is difficult to distinguish their respective importance in a given case, such as the one under investigation.

One has to take into consideration the structure of the metal, water chemistry, absorption of impurities or gases to explain the propagation process of the pitting. However the initiation of the process is normally due to one particular reason, \[ 3 \].

As far as the initiation of pitting corrosion of aluminum is concerned the factors to be considered fall into three categories, \[ 3 \], \[ 4 \]:

a) Chemical factors: Chloride, Calcium Bicarbonate, Copper, Mercury, Chromium, Lead, Oxygen;

b) Metallurgical factors: wrong thermal treatment; intermetallic compounds such as (Fe Al3), (Al Cu2), (Al Fe Si) which create cathodic areas with respect to pure aluminum, while the compound of aluminum with Zinc or Magnesium produce anodic area with respect to pure aluminum; difference in reaction rate between crystals of different orientation.

c) Microbiological factors: bacteria, such as desulfovibric desulfuricans acting in the presence of cations or anions as cathodic depolarising agents to reduce sulphate to sulphide.

As far as the corrosion of the CRENK aluminum tank is concerned one has to take into account the fact that the aluminum plate used to fabricate the bottom of the reactor tank is different from those used to manufacture the reactor tank wall. Indeed, during the installation of the aluminum tank it was discovered that its bottom has a small crack. The original bottom of the tank was thus cut out and replaced by a new one fabricated in Belgium using what was supposed to be the same quality of aluminum, \[ 5 \].

Since all the corrosion pits that were identified are located on the tank bottom, it is likely that the initiation of the corrosion process is due essentially
to metallurgical factors; either:

a) to impurities in the original aluminum plate used to make the bottom of the tank; or

b) to impurities incrusted in the plate during the machining process; or

c) to impurities dropped into the tank bottom during the eighteen years of operation of the reactor.

Taking into account the fact that all the pits are located outside the area covered by the core, that is outside the reflector, (see 'fig.' 3), it is likely that the corrosion process was initiated by galvanic couples, with impurities dropped into the aluminum tank during the operation of the reactor playing the cathodic role. If the pit created survives the initiation phase, it propagates by galvanic reaction with the aluminum being anodic and the impurities being cathodic.

5. RATE OF PENETRATION OF PITS IN ALUMINUM

It is important to evaluate the rate of penetration of pits in aluminum. It allows one to determine if the repair of the aluminum tank bottom is necessary.

One can use the following formula to evaluate the rate of penetration, ( /3/, p. 60):

\[
\frac{1}{3} d = K(t) \quad (1)
\]

where:

\[ d = \text{maximum pit depth}; \ t = \text{time}; \ K = \text{constant that depends on the alloy and the environment}. \]

Since 17 years separate the time of the manufacture of the tank bottom and the time that the maximum pit depth of 2,5 mm were measured, it follows from relation 1 that in the case of the CRENK aluminum tank the constant K is in the range:

\[
K = 0.981 < K < K = 2.5 \quad (2)
\]

If the same rate of penetration holds for the future the deepest pit will go through the aluminum tank bottom at a time, t, such that:
\[
\frac{8 - d'}{K} < t < \frac{8 - d'}{K}
\]  
(3)

with \(d' = 2.5\text{mm}\); that is:

10 years < \(t\) < 177 years.  
(4)

Tanking into account the lower value of, \(t\), one can wait for 10 years before carrying out the repair of the aluminum tank bottom. We thought it advisable, however, to repair the tank bottom right away.

6. THE REPAIR OF THE CORROSION DAMAGE

To remedy the corrosion problem one has to fill the cavities created by the pits. Three solutions can be considered.

The first solution is to fill the pits with aluminum by welding procedure. The second solution is to use concresive epoxy resins. The third solution is to use silicone rubber.

The filling by welding procedure was deemed unfit because one has to heat the aluminum tank to such a high temperature as to cause local defects. The filling of the pits with epoxy resins was discarded because it was felt that under hard and intensive radiations epoxy will not remain stable in the long run. Before using the third solution, that is filling the cavities with silicone, a sample made of aluminum pieces binded together with silicone was irradiated at the BR2 Material Testing Nuclear Reactor in Mol (Belgium) to an integrated flux equivalent to a 1 Mw(th) reactor working full time for 20 years. No damage of the sample was recorded. The repair was thus carried out using silicone covered by small pieces of 1 mm aluminum to act as a protective barrier, ('Fig.' 3).

7. SAFETY UPGRADING OF THE CRENK NUCLEAR REACTOR

After the repair work the reactor was brought in full working condition without any problem.

To avoid the recurrence of the corrosion problem, the following measures were taken:

a) The top of the aluminum tank was sealed more tightly than before to avoid the drop of foreign objects into the reactor:
Fig. 2 Approximate location of pits in the tank bottom. Figures in bracket give the coordinates of pits relative to the centre. (in cm.)

Fig. 3 Filling of pits
b) to improve the water chemistry, and to reduce bacteriological growth in the water, frequent and intensive stirring of the water in the reactor tank is carried out using the primary circuit water pumps;

c) More precise monitors of the pH and resistivity of the demineralized water was installed;

d) A careful control of Chlorine in the water in the tank is now carried out on routine basis;

e) Since corrosion of aluminium may be noticed by peak values of some elements, such as Fe, Cu, SO2, in the water, chemical analysis of the water is now carried out every week;

f) An underwater telescope using a flexible endoscope is being manufactured to monitor constantly the bottom of the aluminum tank, particularly the area under the core which is not visible from the top of the reactor.

8. CONCLUSION

It is our conclusion that the corrosion process of the CRENK reactor aluminum tank bottom was due to galvanic couples initiated by the drop into the reactor tank of materials being cathodic to the aluminum, such as Iron, Titanium, Vanadium, Nickel or Copper. It was not possible to determine the exact galvanic couple but the most likely candidate is "Iron-Aluminum".

Although care was taken to insure that the pH of the water in the reactor tank was always close to neutral as possible, it should be mentioned that the tap water in Kinshasa is strongly acidic in nature, which make it a good electrolyte for the galvanic couple. At all events other factors than metallurgical one, such as bacteria, can intervene to make a pit propagates once it has survived the initiation stage, \( T \).

With the most advanced corroded pits having a depth of only 2,5 mm out a possible maximum of 8 mm the safety of the reactor was certainly not a short term issue. Besides, since the rate of penetration in pitting corrosion usually diminishes with time, perforation may not occur for a considerable time if the metal thickness is adequate, as is the case for the CRENK aluminum tank. Thus we could have waited, possibly for the remaining life of the CRENK reactor. However we thought it worthwhile to meet the challenge and dismantle the reactor in order to gain a better insight of the causes of the corrosion problem. Besides it is a good exercise to dismantle and re-assemble a nuclear reactor, in preparation for the final shutdown of the reactor at the end of its life cycle.
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SAFETY ASPECTS ON ENTIRE LIFTING OLD HWRR TANK

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ABSTRACT

The replacement of Heavy Water Research Reactor (HWRR) tank is a major part of the HWRR reconstruction. The entire lifting up, transportation and burial of the reactor tank must be safely carried out in disassembling reactor complex. For this purpose some safety-related matters were analyzed.

The correct evaluation of radiation damage of the HWRR graphite reflector, in which the reactor tank is located, is a key problem for entire lifting old reactor tank. The analyses on the graphite behavior induced by irradiation, specially on the dimension change, the release of storage energy of graphite reflector as latent heat were given in this paper. The mechanical performance of the old reactor tank and the radiation protection in the technological process of replacing the reactor tank were also related here.

The analyses show that the dimension growth of the graphite reflector after twenty year's irradiation is still not a problem and the reactor tank had ductility enough. Based on these analyses it was decided that the old reactor tank can be lifted out of the reactor complex in one piece in stead of small ones.

1. INTRODUCTION

The heavy water research reactor (HWRR) built in the Institute of Atomic Energy, Beijing, China, is a multipurpose research reactor with low enriched uranium as fuel and with heavy water as moderator and coolant. The old HWRR tank had to be replaced after twenty year's reactor operation, because the coolant bypass flow between fuel channel base and bottom grid plate of reactor tank was as high as 40%.

The extent to disassemble reactor complex depends on safety and smoothly hoisting the reactor tank. It was decided entirely to lift the old reactor tank out of the reactor body in one piece in stead of small ones as a basis
1. The old reactor tank was unlikely stuck in the cylindrical graphite reflector due to a not severe dimension growth induced by irradiation; 2. This tank had ductility and percentage elongation still enough; 3. There is a enough hoisting height for crane in reactor hall; 4. The old reactor tank with intensive radioactivity can be reliably lifted out of the reactor body and directly put it into a underground burial well built in the reactor hall by remote manipulation. The safety analyses and necessary safety measures related to entire lifting the old reactor tank were given and put stress on the analyses of the graphite reflector behaviour induced by irradiation in this paper.

2. ANALYSES ON IRRADIATION BEHAVIOUR OF GRAPHITE REFLECTOR

The graphite reflector of HWRR is constructed by trapezoid graphite bricks arranged in 15 tiers. The bottom of graphite reflector consists of the lowest 3 tiers, the rest 12 tiers pile up the cylindrical graphite reflector with inner diameter 1430 mm and outer diameter 2630 mm, in which the reactor tank is installed. The reactor tank and graphite reflector are located in a steel vessel, as shown in Fig.1. The original radial gap between the reactor tank and the cylindrical graphite reflector is 4.5 mm, and there is a gap of 11.8 mm between the cylindrical graphite reflector and the steel vessel. The dimension growth, accumulation and release of storage energy, oxidation and compressive strength of the irradiated graphite reflector have a direct bearing on entire lifting old reactor tank.

2.1. Dimension growth

Behaviour of dimension changes of graphite bricks induced by irradiation is quite complicated, graphite bricks stretch out in some cases and contract in the other conditions. Dimension changes of graphite bricks strongly depend on irradiation temperature, integral neutron flux and spectrum, technological process of graphite brick manufacture and orientation of graphite bricks in graphite reflector.

The neutron flux distribution in the HWRR graphite reflector was calculated by a 2-D 4-Group code for reactor physics and a 1-D 3-Group code for reactor shielding. Combining the theoretical results with the measured thermal neutron flux and fast spectrum in the experimental holes of the graphite reflector, the integral fast neutron flux distribution in the graphite reflector was determined.

The radial temperature distribution in the HWRR graphite reflector was
given by the statistical data of the reactor operation history and the measured graphite temperature at various reactor power. The axial temperature distribution was roughly considered to be in accordance with the measured axial neutron flux distribution in the graphite reflector.

The graphite brick was radially divided into 5 small pieces for calculation, the radial dimension change is the sum of changes of 5 pieces and the results are listed in Table 1. The axial and peripheral dimension changes calculated in the same way are given in Table 2 and 3, respectively.

The analyses indicate that the graphite bricks used in the HWRR stretch in three directions. The radial maximum dimension growth is 4.40 mm, even if his growth is only directed to the tank, it is still less than the original radial gap of 4.5 mm between the reactor tank and the graphite reflector. Therefore the reactor tank is located in the graphite reflector without binding force. After lifting the old tank out of the reactor body, we made a survey of the graphite reflector, and it shows that the analyses on irradiation behaviour of graphite reflector are correct.

2.2. Accumulation and release of storage energy

The storage energy in the graphite bricks induced by neutron radiation were estimated and the highest accumulation of storage energy occurred in the inner surface of graphite reflector. Generally speaking, the accumulation of storage energy in the HWRR graphite reflector is not too high. The calculated storage energy in the inner surface are listed in Table 4, in which the discontinuous operation and the annealing effect of early operation with lower power were neglected. As for the configuration of the HWRR-complex, it has not a big heat source which results in a sudden temperature raise over the radiation temperature. Therefore, there is no possibility of storage energy release. Even then hoisting the reactor tank out of the reactor body, the storage energy can not be released for lack of friction heat.

2.3. Oxidation and strength

The highest temperature of graphite reflector, which is in a nitrogen atmosphere, is about 250°C at rated power of 10 MW. Generally speaking, it is unnecessary to consider oxidation of the graphite reflector; therefore, the strength of graphite bricks could not worsen in the absence of graphite oxidation.

Regarding strength of graphite, the graphite bricks in the HWRR bear an
axial pressure due to dead weight of them. The axial maximum dimension growth is 24.68 mm, which is far less than the original gap of 250 mm between the graphite reflector top and the steel vessel. It shows that the graphite bricks do not suffer additional extruding stress. When the peripheral dimension growth is over the gap between graphite bricks, they are extruded each other. Therefore, only compressive strength should be considered here. If the graphite strain is less than $0.8 \times 10^{-6}$, the graphite stress lies in a permit extent for the HWRR graphite. From table 3 it can be seen, the strain of graphite bricks are in the permit extent in principle, except somewhere.

3. MECHANICAL PERFORMANCE CHANGE OF THE IRRADIATED REACTOR TANK

The tem temperate of the reactor tank is less than 110°C at the reactor power of 10 MW. The integral fast neutron exposure (above 1 Mev) of the old reactor tank is only about $2.6 \times 10^{16}$ neutrons / cm² after twenty year’s operation. This exposure can not result in an obvious neutron radiation damage of the reactor tank made of Aluminum. In this case there was no big change of the mechanical break strength of the tank, however, the plasticity lowered.

The inner surface of the reactor tank was inspected by an underwater TV system. Some corrosion products and incrustation were deposited on the surface of the tank, specially near by the boundary line between heavy water and helium gas above it. On the boundary line and welding seam there were some point-corrosions, however, there were no clear crystal lattice corrosion and crackle.

In view of the analyses and inspection mentioned above, the lower part of the tank can not fall away due to its dead weight, and it can not break by an unexpected collision in hoist process.

4. RADIOACTIVITY INTENSITY OF THE REACTOR TANK AND RADIATION PROTECTION

4.1. Radioactivity intensity of the reactor tank

The estimation of the radioactivity intensity of the old reactor tank was done for the radiation protection regarding the hoist of this tank. The radioactive sources of the old reactor tank are, fission products due to fuel failure, activation of impurities in coolant and tank materials, and specially, activation of corrosion products, wear and tear remains stemed from the primary cooling system and deposited on the bottom of the tank. Among the number, the cobalt of the bearings material Stellite-Alloy of the
heavy water pumps is the major radioactive source. The composition and content of the samples taken from the bottom of the tank were determined by neutron activation analyses. Assuming that all of wear and tear remains and corrosion products were deposited on the bottom of the tank, the estimated radioactivity intensity is about 2700 curies, which is overconservative.

4.2. Radiation Protection

Based on the estimation mentioned above, to lift and shift the old tank had to be done by remote manipulation. The radiation protection regarding the lift and shift of the old tank was actually treated like safety of means and facilities of lift and shift.

As for radiation protection safety of lifting and shifting the old tank, we must take account of these factors, gamma ray exposure, tritium escape and radioactive dusts. The corresponding protective measures were taken in the process of lifting and shifting the old tank, using special hoisting tool with shielding body for reduction of personal radiation dosage; washing the old tank before dismantling it for reduction of tritium escape; and when the old tank was lifted out of the reactor body, a white dustcoat was automatically worn over the old tank in order to mitigate the spread and contamination of radioactive dusts and airsol.

4.3. Monitoring

The radioactivity intensity of the lifted tank was measured by Ge(Li) gamma spectrometer, the major nuclides are $^{60}$Co and $^{65}$Zn, the total radioactivity intensity is 30.9 curies, which is far less than the estimation. The resulted radiation dose rate at different distance from the central line of the tank was measured by various dosimeters and ion-chambers, as shown in Fig. 2.

5. CONCLUSION

The entire lifting up, transportation and burial of the old reactor tank were completed safely and smoothly, which was for the first time in China.

After lift of the old tank, the inspection and some measurements on the graphite reflector show that the analyses on irradiation behaviour of graphite bricks are correct and the analysis method can meet the engineering requirements to dismantle reactor body like HWRR.

The radioactivity intensity of the old tank is far less than the estimation, the reason lies in irregular distribution of the wear and tear
remains and corrosion products deposited on the bottom of the tank, which results in difficulty of uniformly sampling. However, the radiation protection measures are effective and correct.

REFERENCE

Fig. 1. Vertical Section of HWRR

Fig. 2. Measured Dose Rate
### Table I. Radial Dimension Changes of The Graphite Reflector [mm]

<table>
<thead>
<tr>
<th>Elevation [mm]</th>
<th>Radial divided sections of graphite reflector</th>
<th>0-85</th>
<th>85-195</th>
<th>195-325</th>
<th>325-455</th>
<th>455-600</th>
<th>Total change [mm]</th>
</tr>
</thead>
<tbody>
<tr>
<td>1700</td>
<td></td>
<td>0.503</td>
<td>0.432</td>
<td>0.320</td>
<td>0.228</td>
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<td>1.695</td>
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<td>0.606</td>
<td>0.707</td>
<td>0.455</td>
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<tr>
<td>1300</td>
<td></td>
<td>1.700</td>
<td>1.391</td>
<td>0.766</td>
<td>0.365</td>
<td>0.265</td>
<td>4.397</td>
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<td>0.334</td>
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<td>0.670</td>
<td>0.437</td>
<td>0.264</td>
<td>0.232</td>
<td>2.434</td>
</tr>
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</table>

### Table II. Axial Dimension Changes of The Graphite Reflector [mm]

<table>
<thead>
<tr>
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<th>Radial divided sections of graphite reflector</th>
<th>0</th>
<th>85</th>
<th>195</th>
<th>325</th>
<th>455</th>
<th>600</th>
</tr>
</thead>
<tbody>
<tr>
<td>1700</td>
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<td>1.51</td>
<td>1.07</td>
<td>0.76</td>
<td>0.60</td>
<td>0.36</td>
<td>0.08</td>
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<tr>
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<td>2.57</td>
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<td>0.98</td>
<td>0.67</td>
<td>0.57</td>
<td>0.16</td>
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<td>6.17</td>
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<tr>
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<td>1.54</td>
<td>0.95</td>
<td>2.33</td>
<td>1.00</td>
<td>0.56</td>
<td>0.49</td>
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<td>0.95</td>
<td>0.66</td>
<td>0.57</td>
<td>0.15</td>
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<td>Total change</td>
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<td>24.68</td>
<td>18.93</td>
<td>13.38</td>
<td>7.64</td>
<td>5.01</td>
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Table III. Peripheral Dimension Changes Of The Graphite Reflector [mm]

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<th>Elevation [mm]</th>
<th>Radial divided sections of graphite reflector</th>
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<tr>
<td></td>
<td>0</td>
</tr>
<tr>
<td>1700</td>
<td>0.746</td>
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<td>1.225</td>
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<td>1300</td>
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<td>0.820</td>
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<td>2.860</td>
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<tr>
<td>100</td>
<td>1.156</td>
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Table IV. Storage Energy At Inner Surface Of The Graphite Reflector

<table>
<thead>
<tr>
<th>Elevation [mm]</th>
<th>Temperature [°C]</th>
<th>Integral neutron flux [$10^{20}$ n/cm²]</th>
<th>Storage energy [Calorie/g]</th>
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</thead>
<tbody>
<tr>
<td></td>
<td>100</td>
<td>300</td>
<td>500</td>
</tr>
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<td>136</td>
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<td>6.95</td>
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<td>145</td>
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<td>145</td>
</tr>
<tr>
<td>1100</td>
<td>240</td>
<td>232</td>
<td>240</td>
</tr>
<tr>
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<td>500</td>
<td>500</td>
<td>300</td>
</tr>
<tr>
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<td></td>
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</tbody>
</table>
DISASSEMBLING AND MODIFICATION OF RA-3

RUBEN D. TARIZZO

COMISION NACIONAL DE ENERGIA ATOMICA

BUENOS AIRES

ARGENTINA
DISASSEMBLING AND MODIFICATION OF RA-3

SUMMARY

The objective of this paper is to describe the partial disassembling and modification of RA-3 called Modernization project and it comprises all the technical and administrative steps directly related with this task.

All these requirements have been revised and approved by the Argentine regulatory authority called CALIN (Consejo Asesor de Licenciamiento de Instalaciones Nucleares).

The improvement of RA-3 is a result of the lack of 90% enriched uranium obliging a change over to 20% enriched uranium. This brought about design modifications both in fuel elements and the reactor.

The presentation of documents to the licencing authority as well as the chronogram are detailed separately.

The modernization project was divided in 25 tasks:

1) Changing fuel element support table.
2) Changing heat exchanger.
3) Repairing of cooling towers.
4) Repairing of primary circuit valves.
5) Repairing of irradiation channels.
6) Constructions of a new sampler.
7) Changing tangential channel.
8) Cleaning and disassembling of reactor (inside).
9) Changing continuous demineralizer (ion interchanging column).
10) Detection of failure in fuel elements.
11) Modifications of nuclear instrumentation.
12) Modifications of conventional instrumentation.
13) Modifications of electrical system.
14) Changing telemanipulators.
15) Constructions of mechanism bridge.
16) Changing a primary circuit valve when the heat exchanger is changed too.
17) Painting ground floor, hall floor, and pump room floor with epoxy resin levelling.
18) Installation of fire alarm system.
19) Radioactive liquid discharge.
20) Modifications of secondary circuit. This task involves:
    A) Installation of a third secondary pump.
    B) Extension of this piping.
    C) Installation of two 12 inch valves to the present cooling towers pools independent.
    D) Installation of filtering system.
21) Optimization hot water bed.
22) Changing detector support table.
23) Removal, decontamination and reinstallation of shielding.
25) Changing pneumatic system.

The task of cleaning the tank is part of the RA-3 modernization project and it comprises the disassembling and moving of fuel elements, graphites, control rods, grid plugs, neutron detectors, temperature sensors, pressure measuring equipment, handling baskets, sample boxes, irradiation channels, shieldings elements found in the bottom, cone suction plate (clapeta).

Besides it comprises the emptying, decontamination and tank inspection.

Some examples are given referring to operations with number of persons, exposure and dose involved in actions corresponding to 1988.
1. DESCRIPTION OF FORMER REACTOR

RA-3 is a research production reactor which operated at a normal power of 3 MW reaching a maximum power of 5 MW with a neutronic flux $1.8 \times 10^{13} \text{n/cm}^2.\text{sec}$ using 90% enriched uranium as fuel.

It is an open vertical tank reactor refrigerated by forced circulating demineralized water.

The finality of the reactor is the production of radioisotopes at a commercial scale and experimentation in a wide range of issues joined to basic research and technological applications.

The core consists of 16 fuel elements for the minimum critical conditions, three sides reflected with graphite.

For normal operation and full power, the core configuration could reach 30 elements.

Each fuel element consists of 19 plates prepared in parallel manufactured with 90% enriched uranium and covered by aluminium. Each plate contains about 10.5 grs. of uranium 235 90%. The reactor is controlled lifting and lowering four neutron absorbing elements called control rods which are made of cadmium lined with stainless steel.

The group of electronic measuring devices allows controlling from low power to maximum power.

The devices are divided into: A) two measuring channels supplied with fission chambers. These detectors move vertically to increase the range extension.

B) A measuring channel using a compensated ionization chamber from intermediate range to maximum power with logarithmic amplifier, a measuring chamber with linear scale and compensated ionization chamber.

C) Three measuring channels with non-compensated ionization chamber which supply signals used by the safety system to limit the maximum power value.

The safety actions are:
Above 10% operation power: alarm
Above 20% operation power: automatic lowering of control rods.
Above 30% operation power: scram

The new 20% enriched uranium fuel element should contain 290.7 +/- grs. of the uranium 235 isotope. The fuel plate contains the combustible nucleus which consists of a homogeneous fine dispersion of uranium trioxide inside an aluminium matrix. The contents of 235 uranium of each combustible nucleus should be of 15.3 +/- 3 grs.
The reactivity control plate is made of neutron absorber material (silver cadmium indium alloy) and a stainless steel hermetic cover.

2. **OPERATION OF DISASSEMBLING**

2.1. Moving of fuel elements prepared for control rod to radioactive wastes.

The previous radiological valuation took into account figures from files.

**Operation:**

The container was placed beside the decay pool. It is provided with a rope with a hook at one end, which is used for hooking and hoisting fuel elements that are on a special support stored in the decay pool bottom. The fuel element was lifted slowly up to 1 m. below the water surface. Exposure rate was measured just above water surface and if this rate was less than 100 mR/h, it was possible to lift the fuel element and then insert it into the container.

This operation was performed with a remote controlled tool. The exposure rate values obtained for each Fuel Element and Fuel Element for control rod are pointed out as follows (specifying channel number on radioactive wastes, contact exposure etc.).

Average Dose: Less Than 0.01 mSv.
<table>
<thead>
<tr>
<th>Fuel Elem. N°</th>
<th>Channel N° Dep.</th>
<th>Exposure 1m water mR/h</th>
<th>Contact exposure container mR/h</th>
<th>Exposure 2m (Air) R/h</th>
</tr>
</thead>
<tbody>
<tr>
<td>278</td>
<td>158</td>
<td>10</td>
<td>2,8</td>
<td>8,5</td>
</tr>
<tr>
<td>277</td>
<td>159</td>
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<td>276</td>
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<td>3,0</td>
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<tr>
<td>281</td>
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<td>BRF2</td>
<td>98</td>
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</tbody>
</table>

**TABLE I**
2.2. Disassembling, moving, and disposal of graphites and fuel elements of the former core.

The previous radiological valuation of the graphites as well as the decay pool indicated operation was radiologically manageable. Values of exposure measured in graphites were oscillating between 500 and 700 mR/h.

Previous radiological valuation to the task "moving of fuel elements" took into account figures from files, which contain information of spent fuel elements with greater time decay.

For the moving of fuel elements prepared for control rod it was necessary to install a shielding on the hot cell ceiling to protect operators in the place.

The exposures measured to each fuel element were the following. (Table II).

<table>
<thead>
<tr>
<th>Hot Cell</th>
<th>Top Tank</th>
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<tbody>
<tr>
<td>1 and 2</td>
<td>MEASURED POINTS</td>
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</tbody>
</table>

<table>
<thead>
<tr>
<th>Hot Cell Roof</th>
<th>Hole to DP</th>
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</thead>
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<tr>
<td>3, 4 and 5</td>
<td>MEASURED POINTS</td>
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</table>

<table>
<thead>
<tr>
<th>Decaying Pool</th>
<th>Waiting zone</th>
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<tr>
<td>6 and 7</td>
<td>MEASURED POINTS</td>
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<tr>
<td>Fuel Elem. No</td>
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<tr>
<td>--------------</td>
<td>---</td>
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<tr>
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<tr>
<td>S130</td>
<td>50</td>
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</table>

MEASUREMENTS TAKEN (mR/h) IN POINTS
TABLE II
2.3. Withdrawing of temperature sensor.

It was temporarily placed in hot cell until it was moved to adequate disposal.
Contact exposure was 1200 mR/h
Personnel involved: two operators and one radiation protection officer.
Average incorporated dose: 0.05 mSv.

2.4. Withdrawing of chambers of equipments A, B y C

Which were moved to loop room with a maximum exposure contact of 300 mR/h.
Personnel involved: two operators and one radiation protection officer.

2.5. Moving of ionization chamber shieldings to the disposal bunker (cementery).

The shielding was hoisted with a gantry crane passing it through a trap door to the ground floor room.
In this sector a shielded lorry was used to protect the operator who moved the shielding to a bunker specially designed to dispose of active materials.
The contact exposure of the shielding was 2 R/h.
The contact exposure of the bunker was 15 mR/h.
Personnel involved: five operators and two radiation protection officers with an average dose of 0.040 mGy/h.

2.6. Disassembling and moving of pressure difference meter

Previous measurements pointed out that the lower sector of the pipe, called plug, was highly activated (± 500 R/h). Besides, because of the pipe shape, manipulation was very difficult.
Previous Radiological Valuation was carried out as follows: the pipe was tied and then slowly hoisted with a gantry crane. It was monitored continuously, the values being:

![Diagram](attachment:fig2.png)
Due to the high exposure value measured in point two, it was decided to perform the measurement at the lowest sector pipe by sinking a telescopic gage detector protected hermetically with an adequate sheath. The contact exposure of the lowest sector was 500 R/h. Then, the pipe was again sunk into the water, and it was decided to cut it off at the lowest angle (point two).

Dose involved:
- Radiation protection officer: 0.1 mSv
- Operators (Average): 0.05 mSv

2.6.1. Operation Description

It was decided to perform a cutting test with the cold sector pipe.

Scissors "to cut metals" and a hack saw were used, measuring cutting time and at the same time testing the cold operation.

Then four cuts were performed, with a hack saw, which was found to be the best tool for this purpose (less duration: 15"").

A shielding was necessarily installed to protect operators as shown in the drawing:

Once the shielding test had been finished and the procedure had been approved, the operation was begun.

The cutting was performed by an operator, from inside the hot cell, while the exposure rate and time were being
measured by a radiation protection officer. The cutting task was performed in 20" (Twenty seconds). This time was longer than testing time, because during the cutting the saw jammed. Once the cutting was finished, the activated section was left sunk in the water bound with a rope and a hook. The other section of the pipe was moved to the disposal bunker, giving an exposure rate below 1 mR/h.

2.6.2. Moving to Decay Pool

The shieldings placed over the hot cell roof were lifted and then put on both sides of the gap as shown below:

Then, the activated section pipe was hoisted from the hot cell roof through the duct that connects the hot cell with the tank. Once the pipe was placed inside hot cell, it was centralized (by means of telemManipulators) in the duct leading to the decay pool, and lowered (permanently bound with a steel cable). Finally, the pipe was unhooked and stored in the decay pool.

Personnel and dose involved:

<table>
<thead>
<tr>
<th>Operator Nº</th>
<th>Body Dose</th>
<th>Hands Dose</th>
</tr>
</thead>
<tbody>
<tr>
<td>1</td>
<td>0.1 mSv</td>
<td>0.7 mSv</td>
</tr>
<tr>
<td>2</td>
<td>0.08 mSv</td>
<td></td>
</tr>
<tr>
<td>3</td>
<td>0.06 mSv</td>
<td></td>
</tr>
<tr>
<td>4</td>
<td>0.06 mSv</td>
<td></td>
</tr>
<tr>
<td>R.P. Officer</td>
<td>0.1 mSv</td>
<td>0.7 mSv</td>
</tr>
<tr>
<td>2</td>
<td>0.1 mSv</td>
<td></td>
</tr>
</tbody>
</table>
2.7. Disassembling and moving of irradiation channel number 4.

Once disconnected the flange which join the channel to the tank wall, the channel was hoisted slowly, continously measuring the exposure until it was 80 cm. below the water surface where exposure was 40 mR/h. It was measured with a telescopic gage detector protected hermetically by and adequate sheath.

The figure was 22 R/h. Then, using the gantry crane and operators on hot cell ceiling the channel was hoisted with hooks and ropes. Then it was hoisted until the necessary height to move it to the trap door and immediate lowering to the ground floor room an later disposal into the decay pool. This element will wait until its disposal in radioactive residues. There is a report about "Transfer of fuel elements to radioactive residues" which is available.

Personnel involved in the last task: Three operators and one radiation protection officer with a maximum dose of 0.02 mSv (to only one).
Fig 5 MOVING OF IRRADIATION CHANNEL

1 Operators Place X 15 mR/h
2 Irradiation Channel
Contact Exposure 22 R/h
2.8. Partial draining and decontamination of the tank
Previous conditions were:
   a) Cisterns number 3 and 5 empty and clean.
   b) Piping tank to cisterns in good condition.
   c) Fuel elements, graphites, control rods and other
      objects that must be withdrawn with full tank.
The tank was drained up to water difusing height without
difficulty.
Measurements taken from cistern samples (with tank
water) pointed out that it was possible to discharge to
environment without risk.
2.8.1. Tank decontamination
The previous radiological valuation was carried out
by monitorig inside the tank and on the water surface with
a digital dosimeter and an ionization chamber called "Baby
Line". The figures were:
   Exposure rate: 40 mR/h
   Dose equivalent: 0.054 mSv
These measurements were performed using a container
fitted with detectors inside.
It took the container 5' to move from tank top to
water surface, where it remained for 5' took another 5' to
move back to tank top level again. Total time: 15'.
Afterwards, a number of sweep tests were carried out
every 60 cm. to determine tank wall contamination,
begining on tank top (former water level) as shown in the
drawing:

![Diagram of tank with water level mark](image)
A cage was especially made to transport operators, it hung from the gantry crane.

Once this task was finished, the papers were measured giving the following values: Table N°3.

Personnel involved: Dose equivalent

Operator N°1 0.023 mSv
Operator N°2 0.005 mSv
Operator N°3 0.005 mSv
Rad. Prot. Officer 0.011 mSv

The former table shown relatively low level of contamination.

However, the test performed pointed out that the contamination was washed down.

Spectrometric analysis detected Cs 137 and Co 60.

Due to the appearance of operative problems in carrying out the sweep test with the cage, poor stability, sudden movements, and defective support, it was decided to construct a platform slightly smaller than tank circumference that could slide in vertically.

During the operation the walls were sprayed with water from a hose, rubbed with soft brushes, and rinsed with abundant water, without chemical products.

Thus, all the tank walls were decontaminated, up to 1 m. below the "step" (half tank).

At that point, water level was approximately 1 m. below the platform.

Exposure on platform floor: 4 mR/h.
<table>
<thead>
<tr>
<th>Sample N°</th>
<th>Number imp/sec</th>
<th>Factor Probe limp/seg = 1,2x10^{-5} \mu Ci/cm²</th>
<th>Activity \mu Ci/cm²</th>
<th>Paper Surface cm²</th>
<th>Tot. Activity \mu Ci</th>
<th>scraped surface (estimate) cm²</th>
<th>Contamination level 10^{-4} \mu Ci/cm²</th>
</tr>
</thead>
<tbody>
<tr>
<td>7</td>
<td>30</td>
<td>&quot;</td>
<td>3,6x10^{-4}</td>
<td>30</td>
<td>0,0108</td>
<td>100</td>
<td>1,08</td>
</tr>
<tr>
<td>6</td>
<td>30</td>
<td>&quot;</td>
<td>3,6x10^{-4}</td>
<td>30</td>
<td>0,0108</td>
<td>200</td>
<td>0,54</td>
</tr>
<tr>
<td>4</td>
<td>70</td>
<td>&quot;</td>
<td>8,4x10^{-4}</td>
<td>30</td>
<td>0,0252</td>
<td>400</td>
<td>0,63</td>
</tr>
<tr>
<td>3</td>
<td>40</td>
<td>&quot;</td>
<td>4,8x10^{-4}</td>
<td>30</td>
<td>0,0144</td>
<td>50</td>
<td>2,88</td>
</tr>
<tr>
<td>2</td>
<td>70</td>
<td>&quot;</td>
<td>8,4x10^{-4}</td>
<td>30</td>
<td>0,0252</td>
<td>160</td>
<td>1,58</td>
</tr>
<tr>
<td>8</td>
<td>400</td>
<td>&quot;</td>
<td>4,8x10^{-3}</td>
<td>30</td>
<td>0,144</td>
<td>900</td>
<td>1,60</td>
</tr>
<tr>
<td>1</td>
<td>400</td>
<td>&quot;</td>
<td>4,8x10^{-3}</td>
<td>30</td>
<td>0,144</td>
<td>250</td>
<td>5,76</td>
</tr>
</tbody>
</table>

**TABLE III**
JOYO MODIFICATION PROGRAM FOR DEMONSTRATION TESTS OF 
FBR INNOVATIVE TECHNOLOGY DEVELOPMENT

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JOYO MODIFICATION PROGRAM FOR DEMONSTRATION TESTS OF FBR INNOVATIVE TECHNOLOGY DEVELOPMENT

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ABSTRACT

A planning is under way at PNC to modify the experimental fast reactor JOYO. The project is called "Mark-III (MK-III) program". The purpose of MK-III is to expand the function of JOYO, and to make it possible to receive demonstration tests of new or high level technologies for the FBR development.

The MK-III program consists of two main modifications; a modification for improvement to highly efficient irradiation facility and a modification for demonstration test of new technologies and concepts that have a high potential to reduce FBR plant construction cost, to elevate plant reliability and to improve plant safety.

These modifications are scheduled to start in 1991.
1. Introduction

The experimental fast reactor JOYO was constructed as the first sodium cooled fast neutron reactor in 1974 and has made significant contributions to the development of FBR in Japan.

JOYO, first of all, provided the technical experience required for the design of the prototype FBR MONJU (280MWe) through operation of the Mark-I core (50-75MWt) which was designed as a miniature of a standard FBR core with a blanket, operated from 1977 to 1981. Secondly, the JOYO plant served as an irradiation facility to develop FBR fuels and materials needed for advanced reactors, by modifying the core configuration from the MK-I core to the MK-II core (100MWt), replacing blankets with reflectors and using higher power fuel assemblies, to increase the fast neutron flux, in 1982.

A planning is under way for a third phase of JOYO, which will develop the innovative technologies required for FBR commercialization. This plan is named "MK-III program" and consists of two main modifications.

The first is an improvement in irradiation capability, to conduct R&D on high performance and high burn-up fuels of commercial FBRs. Modifications to the reactor core, the heat transport system and the fuel handling system are necessary to realize this improvement. As a result, JOYO is expected to increase its reactor thermal rating from 100MWt to about 140MWt.

Another mission is to develop new technologies and concepts that have a high potential to reduce FBR plant construction cost, to elevate plant reliability and to improve plant safety. To achieve these objectives,
many programs such as testings of newly developed materials, a
demonstration testing of secondary sodium loop elimination including
development of a bellows expansion joint and a steam generator with
double-walled tubes and so on are under development.

The plan of the MK-III program is scheduled to start in 1991.

2. Description of JOYO

2.1 Outline of JOYO and its operational history

JOYO is a fast experimental reactor which uses plutonium-uranium mixed
oxide and sodium as the fuel and the coolant, respectively. The project began
in 1964 with preliminary design, followed by R&D of sodium technology.
Site construction began at O-arai, Ibaraki-ken on January 1970, and the reactor
attained its first criticality on 24th April 1977 with breeder core (MK-I).
Conversion from the breeder core to the irradiation core (MK-II) was
successfully completed by handling about 600 subassemblies in 10 months on
schedule as planned and criticality with the MK-II core was attained in
November 1982. The complete operating history of JOYO is illustrated in
Fig.1.

Both 50MWt and 75MWt power level were achieved with the MK-I
breeder core. The MK-I core operation covered the period from initial
criticality in 1977 through 1981. There were 260 reactor startups during MK-I
operation resulting from many kinds of reactor tests such as low power
reactor physics tests, reactor dynamics tests, power ascent tests and transient
tests. The maximum burn-up attained on a MK-I driver fuel assembly was
40,500 MWd/t which was close to the design limit of 42,000MWd/t.
The MK-I operation was completed at the end of 1981 with a scram to natural circulation test from 75MWt. The cooling system of JOYO consists of two main cooling systems and an auxiliary cooling system which removes decay heat of the core if heat removal by the main cooling systems were impossible (Fig. 2). Each of the main cooling systems and the auxiliary cooling system, then consists of the primary system, which transmits the heat from the core to the intermediate heat exchanger, and the secondary system, which transmits the heat from the intermediate heat exchanger (IHX) to the air-cooled dump heat exchanger (DHX), in normal operation.

Replacement of the whole core components from MK-I to MK-II was conducted during the year 1982. The comparison of core configuration of MK-I and MK-II is shown in Fig. 3. Their main core parameters are listed on Table 1. The MK-II core has the characteristics of higher fast neutron flux to enable accelerated irradiation of fuels and materials, higher core power density with advanced fuel subassemblies of higher linear heat rate. To obtain such features, the MK-II core is composed of the core fuel assemblies surrounded by the stainless steel reflectors, and is operated at 100MWt rated power.

The 100MWt power ascent program consisted of low power core characterization tests and high power tests. As for the low power tests, reactivity coefficient, loop pressure drop and core flow rate distribution were measured, and the performance of the reactivity control systems and the core cooling capability were confirmed prior to power ascent. Power ascent was conducted in a step-by-step manner with 25MWt power increments. At each power step, safety, stability and heat balance of the cooling systems together with its control margin were confirmed.
At the full power operation, the reactor output temperature reached 500°C with 370 °C at the inlet. The highest sodium temperature at the fuel subassembly outlet was 554 °C, which agrees with the prediction based on the measured flow rate. Also, it was confirmed that all four DHXs were able to remove the heat with an air flow margin of 20%.

Twelve normal operation cycles at 100MWt had finished by 1986, where one normal cycle consisted of 45 days operation. Maximum burn-up of the fuel pins had reached about 48,000MWd/t. Until now, JOYO has experienced neither fuel pin failure nor serious trouble on the components of the reactor system.

In order to perform the irradiation test more efficiently and also to utilize the driver fuels more effectively, the core has been modified since the thirteenth duty cycle operation which started at the beginning of September, 1987, by using modified driver fuels named J2 fuel with new license.

Comparison of the specification of the conventional J1 driver fuels and the newly licensed J2 fuels is shown in Table 2. \(^{235}\text{U}\) enrichment of J2 fuel is increased from 12\(^{\circ}\)/o to 18\(^{\circ}\)/o to get higher reactivity. Special feature of J2 fuel is that the fissile Pu content \((\frac{^{239}\text{Pu} + ^{241}\text{Pu}}{\text{Pu} + \text{U}})\) is specified, in contrast with the Pu oxide content \((\frac{\text{PuO}_2}{\text{PuO}_2 + \text{UO}_2})\) which was adopted in the specification of J1 fuel. The reason to introduce the new specification is to maintain constant nuclear property of driver fuels against the changes in fissile Pu content of reprocessed Pu.

The number of J2 fuel in the core has been gradually increased as shown in Table 3. The operation period of a cycle was, then, gradually extended from 45 days corresponding to the number of J2 fuel in the core. The average burn-up of the core was also increased at the same time.
Consequently, 70 days operation was attained at the fifteenth duty cycle which finished its operation in May, 1988, where the maximum burn-up of a driver fuel reached over 70,000MWd/t.

2.2 Irradiation facility of JOYO

JOYO provides some kinds of instrumented test assembly, together with various uninstrumented irradiation subassemblies (UNIS).

UNISs have the same outer shape as the core driver fuel subassembly. Thus, they are able to be transferred through the fuel handling facility of the reactor into and out of the core. In addition, they can be exchanged for driver fuel subassemblies, namely, they are able to be irradiated at any position in the core. The UNISs provide certain monitors which inform of typical irradiation conditions through post-irradiation examinations. The UNISs which are utilized at present are classified as type A, B and C, as shown in Fig. 4.

The instrumented test assembly (INTA) has been developed as a test rig capable of monitoring the behavior of fuels and materials during irradiation period. The INTAs provide some on-line instruments which inform of many irradiation conditions and describe the behavior of fuels and materials.

The informations from these instruments are logged continuously during the irradiation, i.e., during the reactor operation. The signals are sent to the data acquisition system by the instrument leads which are penetrating the boundary of the reactor vessel through its upper internals. Thus, the INTAs are to be very long test rigs which are loaded from the top of the reactor vessel and occupy the long space extending from the above of the reactor vessel to the reactor core.

The schematic diagram of INTA is shown in Fig. 5.
3. Outline of MK-III program

3.1 Background

Approximately for forty years after EBR-I's first criticality, in which the concept of FBR was demonstrated and gave impetus to the prospect of a long term reliance on nuclear fuel, the FBR technologies have been steadily developed in each of the countries in which FBRs have been or are being built.

Also in Japan, the R&D for FBR is being proceeded, aiming at the commercialization around 2030 through construction of several FBRs with a step-by-step improvement of technologies and economies. The prototype FBR MONJU which stands between the experimental reactor and the demonstration reactor in the development program of FBR and aims to attain technological advancement and economic prospect towards the establishment of commercial viability of future FBR plants is being built.

In general, the FBR development is in the planning stage to realize a demonstration plant with technologies and economies which is comparable with those of commercial LWR.

In this respect, it is requested to clarify a long-term strategy for FBR development, based on the result of R&D including the experience of plant design, construction, operation and maintenance and further innovative technologies or new concepts for FBR plant system.

As mentioned above, the target of the development for commercialization is expected to attain before 2030 by reducing the construction cost and the fuel cycle cost, namely, by reducing the power generating cost of FBR less than that of LWR, with the same level of safety and maintainability.
From this point of view, the following items are requested to test or demonstrate;

(1) Steady and unsteady irradiation test for the development of high performance and high burn-up fuel.
(2) Evaluation of inherent safety features based on ATWS related test.
(3) Demonstration test of new system concept and components.

3.2 Objectives

JOYO modification program is planned for following utilization; (Fig. 6, Fig. 7 and Fig. 8)

(1) Utilization as a highly efficient irradiation facility

Irradiation capability of JOYO is to be improved by core modification and power up rating. Fast neutron fluence per a year is to be increased to approximately $9 \times 10^{22} \text{n/cm}^2$ which is equivalent to twice of present value, in order to demonstrate FBR fuel burn-up of 150~200GWD/t (average in assembly) by about 2000, at which year the determination of basic specifications of commercial plant seems to be scheduled.

(2) Utilization as a demonstration facility of new system concept and components

The innovative plant systems, components and new material such as Self Actuated Shut Down System, Expansion Joint (Bellows) System, Component Integral System, Secondary Sodium Loop Elimination System, In-Core Anomaly Diagnostic System, Advanced Sodium Pump, Structural and Shielding Materials, etc., are to be
developed based on demonstration tests in JOYO by taking necessary modifications.

3.3 Modification to a highly efficient irradiation facility

The planned modification is to increase the fast neutron flux by 30%~40% and to increase the plant availability by 50%.

It means that the irradiation time for the necessary fuel burn-up is expected to reduce by a half.

The modifications on reactor core, heat transport system and fuel handling system are mainly necessary to realize this improvement.

3.3.1 Core Configuration

- Expansion of core fuel zone; The maximum number of core fuel subassemblies are increased from 67 to 85 and the two zoned core is designed for flattening power distribution.

The core configuration and main parameters are shown in Fig. 3 and Table 1, respectively and the modification procedure of the core configuration from MK-II to MK-III is shown in Fig. 9.

- Change of control rod allocation; Two control rods are moved to the fifth row from the third row.

- Application of radial shielding assemblies; Corresponding with the increase of fast neutron flux, the most outside stainless steel reflectors are replaced with the B\textsubscript{4}C neutron shielding assemblies which are to provide neutron and gamma shielding for the reactor structure materials.

- Active core length; The active core length is decreased to 48cm from 55cm in order to get a higher neutron flux.
3.3.2 Fuel design

The geometrical specifications of driver fuel is the same as current Mark-II fuel. However, at the evaluation of maximum allowable linear heat rate (more than 500W/cm), the growth effect of central void in fresh fuel pellets during power ascent is to be considered.

3.3.3 Heat transport system

As the result of the core design to improve fast neutron flux, JOYO is expected to increase its reactor thermal rating from 100MWt to about 140MWt.

For the increase of thermal rating, the temperature difference between core inlet and outlet is to be changed from 130°C to 150°C, coolant flow rate is increased by about 20% and the heat transfer area of DHX is increased by 40%.

3.3.4 Improvement of plant availability

- Improvement of fuel handling system; For the transfer rotor which transfers the fuel between inside and outside of containment vessel, another function as a storage facility of spent fuels (about 15 subassemblies per one refueling) is to be added to decrease the refueling time by about 30%. The comparison of refueling time of MK-II and MK-III is shown in Fig. 10.
- Reduction of annual inspection period; The period of annual inspection is decreased to about two months from 4~5 months by utilization of remote inspection devices.
- Improvement of irradiation technology; The loading time of INTA is decreased to about two days from one month by the utilization of in-
sodium connector which eliminates the cutting of instrumental lead in
the core region.

3.4 Development programs of new technologies and concepts

(1) New materials testing

The possible candidate materials of structure, shielding and fuel
are planned to test.

(2) Self-actuated shutdown system (SASS)

A self-actuated shutdown system with a temperature sensitive
electromagnet, which is expected to operate by temperature increase of
the coolant without any control of the existing protection systems, is
scheduled to be installed for the demonstration test.

(3) Demonstration test of bellows expansion joint

The demonstration test of a bellows expansion joint is planned in
the secondary system of JOYO at the time of the modification of the
heat transport system.

(4) Development of in-core anomaly diagnostic system

The demonstration test of an in-core neutron flux monitor and
acoustic monitor and the functional tests of in-core anomaly diagnostic
system are planned to improve FBR plant safety.

(5) Development of fuel failure diagnostic system

Demonstration tests are planned at the MK-III core on a fast
failed fuel detection (FFD) & location system (FFDL) and an on-line
plant contamination monitor, which improve operational reliability and
optimize plant design and safety logics.
(6) Development of automatic reactor operation system

The demonstration test of the automatic reactor operation system is planned for the reduction of operator's load and the improvement of operational reliability.

(7) Development of new concept for the heat transport system

The demonstration test program on secondary sodium loop elimination system which has a high potential to reduce FBR plant construction cost (down 10-15%) is under development. In order to realize this concept, the technical feasibility studies are under way for the development of highly reliable steam generator and the refinement of safety logics. A 70MW steam generator of double-walled tube type is planned to be installed in the primary heat transport system. The layout of secondary sodium loop elimination system is shown in Fig. 11.

3.5 Procedure and schedule of modification

The modification to highly efficient irradiation facility is to start with a licensing work on changing the location of control rods and the expansion of core size from 67 of fuel assemblies to 85 in 1991.

The modifications of the fuel handling system and the heat transport system are planned to complete by 1993 and 1996, respectively.

The demonstration tests of new technologies and concepts is planned to proceed step by step from 1993 to 1999.

The demonstration test of the secondary sodium loop elimination system including development of a bellows expansion joint and a steam generator with double-walled tubes is planned to start with the preparations for the establishment of safety logics and the development of highly reliable steam generator.
4. Conclusion

The modification planning is under way for the third phase of FBR development utilizing JOYO, which will develop the innovative technologies required for FBR commercialization.

JOYO has many re-licensing experiences such as modification to the MK-II core, change of the MK-II driver fuel specifications and fission products release test and so on in the past.

The MK-III program will be also proceeded step by step, along with the licensing process, the reassessment of current plant conditions and the results of related R&Ds.
Table 1  Main Core Parameters of JOYO

<table>
<thead>
<tr>
<th>Item</th>
<th>Core (Fuel)</th>
<th>MK-I</th>
<th>MK-II</th>
<th>MK-III</th>
</tr>
</thead>
<tbody>
<tr>
<td></td>
<td></td>
<td>First</td>
<td>Second</td>
<td></td>
</tr>
<tr>
<td>Reactor Output</td>
<td>MWt</td>
<td>50</td>
<td>75</td>
<td>100</td>
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<tr>
<td>Primary Coolant Flow Rate</td>
<td>lb/</td>
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<td>2,200</td>
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<td>Reactor Inlet Temperature</td>
<td>°C</td>
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<td>370</td>
<td>370</td>
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<td>Reactor Outlet Temperature</td>
<td>°C</td>
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<td>470</td>
<td>500</td>
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<td>Core Stack Length</td>
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<td>60</td>
<td>60</td>
<td>55</td>
</tr>
<tr>
<td>Core Volume (max.)</td>
<td>t</td>
<td>294</td>
<td>304</td>
<td>250</td>
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<tr>
<td>Linear Heat Rate (max.)</td>
<td>W/cm</td>
<td>210</td>
<td>320</td>
<td>400</td>
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<tr>
<td>Fuel Pin Diameter</td>
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<td>6.3</td>
<td>5.5</td>
</tr>
<tr>
<td>PuO2/(PuO2+UO2) w/o</td>
<td>%</td>
<td>18</td>
<td>18</td>
<td>-30</td>
</tr>
<tr>
<td>ZrU Enrichment</td>
<td>w/o</td>
<td>23</td>
<td>23</td>
<td>-12(11)</td>
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<tr>
<td>Neutron Flux (max.) (u/m² cm²/sec)</td>
<td></td>
<td>2.1x10⁵</td>
<td>3.0x10⁵</td>
<td>4.2x10⁵</td>
</tr>
<tr>
<td>Neutron Flux (Core av.) (u/m² cm²/sec)</td>
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<td>1.2x10⁵</td>
<td>1.9x10⁵</td>
<td>3.1x10⁵</td>
</tr>
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<td>Max. Excess Reactivity (%)</td>
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<td>-4.5</td>
<td>-4.5</td>
<td>-5.5</td>
</tr>
<tr>
<td>Control Rod Worth</td>
<td>%/k/k</td>
<td>Safety Rod =5.6 Regenerating Rod =2.8 Safety Rod =5.6 Regenerating Rod =2.8</td>
<td>-9</td>
<td>T.B.D.</td>
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<td>Max. Burn-up (pin av.)</td>
<td>MW/d</td>
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<td>75,000</td>
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<td>Operation Cycle</td>
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<td>45 days</td>
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<td></td>
<td></td>
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</tr>
<tr>
<td></td>
<td></td>
<td>15 days</td>
<td>15 days</td>
<td>23 days</td>
</tr>
<tr>
<td></td>
<td></td>
<td>Outage</td>
<td>Outage</td>
<td>Outage</td>
</tr>
</tbody>
</table>

Table 2  Main Parameters of MK-II and MK-III Driver Fuel

<table>
<thead>
<tr>
<th>Item</th>
<th>Core (Fuel)</th>
<th>MK-II</th>
<th>MK-III</th>
</tr>
</thead>
<tbody>
<tr>
<td></td>
<td></td>
<td>J1</td>
<td>J2</td>
</tr>
<tr>
<td></td>
<td></td>
<td>Inner Core</td>
<td>Outer Core</td>
</tr>
<tr>
<td>Cladding Outer Diameter (mm)</td>
<td></td>
<td>5.5</td>
<td>5.5</td>
</tr>
<tr>
<td>Cladding Inner Diameter (mm)</td>
<td></td>
<td>4.8</td>
<td>4.8</td>
</tr>
<tr>
<td>Fuel Pellet Outer Diameter (mm)</td>
<td></td>
<td>4.63</td>
<td>4.63</td>
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<tr>
<td>Fuel Pellet Form</td>
<td></td>
<td>Solid</td>
<td>Solid</td>
</tr>
<tr>
<td>Fuel Pellet Density (% T.D.)</td>
<td></td>
<td>93</td>
<td>94</td>
</tr>
<tr>
<td>235U Enrichment (w/%)</td>
<td></td>
<td>-12</td>
<td>-18</td>
</tr>
<tr>
<td>Pu Fissile Content (w/%)</td>
<td></td>
<td>22</td>
<td>20</td>
</tr>
<tr>
<td>Core Height (cm)</td>
<td></td>
<td>55</td>
<td>48</td>
</tr>
<tr>
<td>No. of Fuel Pins</td>
<td></td>
<td>127</td>
<td>127</td>
</tr>
<tr>
<td>No. of Core Fuel Assemblies (#)</td>
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<td>67</td>
<td>25</td>
</tr>
</tbody>
</table>

Table 3  Extension of Operation Period and Core Average Burn-up

<table>
<thead>
<tr>
<th>Operation Cycle No.</th>
<th>12</th>
<th>13</th>
<th>14</th>
<th>15</th>
<th>16</th>
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<th>18</th>
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<tbody>
<tr>
<td>Operation Days</td>
<td>45</td>
<td>55</td>
<td>60</td>
<td>70</td>
<td>70</td>
<td>70</td>
<td>70</td>
</tr>
<tr>
<td>Component of Fuel Assemblies</td>
<td></td>
<td></td>
<td></td>
<td></td>
<td></td>
<td></td>
<td></td>
</tr>
<tr>
<td>J1</td>
<td>65</td>
<td>60</td>
<td>55</td>
<td>42</td>
<td>39</td>
<td>29</td>
<td>21</td>
</tr>
<tr>
<td>J2</td>
<td>0</td>
<td>5</td>
<td>12</td>
<td>22</td>
<td>24</td>
<td>33</td>
<td>41</td>
</tr>
</tbody>
</table>
| Core Average Burn-up (x10^1 MWd/)
|                      | 3.5|
|                      | 3.0|
|                      | 2.5|
|                      | 2.0|
|                      | 1.5|
|------|------|------|------|------|------|------|------|------|------|------|------|------|------|
| MK-I (Breeder core) | | | | | | | | | | | | | |
| Low Power Test | 50MW Performance Test | 50MW Operation | 75MW | 75MW Operation | Core Replacement | MK-II (Irradiation core) | | | | | | |
| | | | | | | | | | | | | |
| Key Items | Inspection | Operation | Core Replacement | Performance Test | Operation | | | | | | | |
| First Criticality 50MW 1977.4.24 | 50MW | 50MW | 75MW | 75MW | 75MW | 100MW | 100MW | | | | | |

Fig. 1 Experimental Fast Reactor JOYO Operational History

The reactor cooling system consists of two loops of main cooling and one loop of auxiliary cooling. The heat generated in the core is dissipated to the atmosphere through primary sodium and secondary sodium.

Fig. 2 Reactor Cooling System
Fig. 3 Core Configuration

Fig. 4 Uninstrumented Irradiation Subassemblies
Fig. 5 The schematic diagram of Instrumented Test Assembly (INTA-S)

Fig. 6 Outline of MK-III program
**Fig. 7 Schedule of MK-III Program**

**Fig. 8 Summary of MK-III Program**
Fig. 9 Modification Procedure of Core Configuration from MK-II to MK-III

Fig. 10 Comparison of refueling time of MK-II and MK-III

Legend
- CORE
- Fuel
- Fuel(inner)
- Control rod
- Reflection

<table>
<thead>
<tr>
<th>Reactor Power (MW)</th>
<th>Neutron Flux</th>
<th>Maximum Linear Heat Rate (W/cm²)</th>
<th>Number of Fuels</th>
<th>Core Stack Length (cm)</th>
<th>Pin Diameter (mm)</th>
<th>Number of Pins</th>
<th>Fuel Pellet Density (%TD)</th>
<th>Control Rod</th>
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</thead>
<tbody>
<tr>
<td>MK-II</td>
<td>100</td>
<td>400</td>
<td>67</td>
<td>55</td>
<td>5.5</td>
<td>127</td>
<td>94</td>
<td>6 at 3rd Row</td>
</tr>
<tr>
<td>MK-III (Step-1)</td>
<td>100</td>
<td>~400</td>
<td>67</td>
<td>55</td>
<td>5.5</td>
<td>127</td>
<td>94</td>
<td>5 at 3rd Row, 1 at 5th Row</td>
</tr>
<tr>
<td>MK-III (Step-2)</td>
<td>~100</td>
<td>~400</td>
<td>(inner) 55</td>
<td>(inner) 55</td>
<td>5.5</td>
<td>127</td>
<td>94</td>
<td>4 at 3rd Row, 2 at 5th Row</td>
</tr>
</tbody>
</table>

Ref. N/S : New fuel
S/F : Spent fuel
Pot-A : Pot for Cooling
Pot-B : Pot for Transport
Newly-constructed SG-building which consists of Concrete C/V is newly constructed at the west side of Reactor Auxiliary Building.

**CONFIGURATION**

**DIAGRAM**

Fig. 11 Layout of Secondary Sodium Loop Elimination System
THE CONVERSION OF NRU FROM HEU TO LEU FUEL

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Chalk River Nuclear Laboratories
Chalk River, Ontario, KOJ 1JO
THE CONVERSION OF NRU FROM HEU TO LEU FUEL

IAEA-SM-310/73

ABSTRACT

The program at Chalk River Nuclear Laboratories (CRNL) to develop and test low-enriched uranium fuel (LEU, <20% U-235) is reviewed, and the status of the conversion of the NRU reactor from highly enriched uranium (HEU, 93% U-235) to LEU fuel is discussed. The replacement LEU fuels developed and tested at CRNL contain high-density uranium silicide particles dispersed in aluminum, in cylindrical rods. The silicides tested include U₃Si, U₃SiAl, U₃Si₁₀Al and U₃Si₁₂ (U-3.96 wt% Si; U-3.5 wt% Si-1.5 wt% Al; U-3.2 wt% Si-3 wt% Al; U-7.3 wt% Si, respectively). Fuel elements were fabricated with uranium loadings suitable for NRU, 3.15 gU/cm³, and for NRX, 4.5 gU/cm³, and were irradiated under normal fuel-operating conditions. Eight experimental irradiations involving 100 mini-elements and 84 full-length elements (7x12-element rods) were completed to qualify the LEU fuel and the fabrication technology. Post-irradiation examinations confirmed that the performance of the LEU fuel, and that of a medium-enrichment uranium (MEU, 45% U-235) alloy fuel tested as a back-up, was comparable to the HEU fuel. The uranium silicide dispersion fuel swelling was approximately linear up to burnups exceeding NRU's design terminal burnup (80 at%). NRU was partially converted to LEU fuel when the first 31 prototype fuel rods manufactured with industrial-scale production equipment were installed in the reactor. The rods were loaded in NRU at a fuelling rate of about two rods per week over the period 1988 September to December. This partial LEU core (one third of a full NRU core) has allowed the reactor engineers and physicists to evaluate the bulk effects of the LEU conversion on NRU operations. As expected, the irradiation is proceeding without incident.
1. INTRODUCTION

The NRU reactor at CRNL is a multipurpose, tank-type thermal research reactor, using D_2O moderator and coolant and highly enriched uranium (HEU, 93% U-235) U-Al alloy driver fuel. Various kinds of rod assemblies are suspended in the tank on a 7 1/4 in (19.7 mm) hexagonal lattice. The reactor produces radioisotopes, provides neutron beams for research, and has special facilities for testing metallurgical specimens and advanced power reactor fuels, and for performing fuel damage (blowdown) tests.

The NRU reactor achieved criticality in 1957 and ran at 220 MW (Th) using driver fuel rods containing flat plates of natural uranium metal. It was converted to highly enriched, rod-type driver fuel in 1964, and currently runs at about 130 MW (Th). The linear-fissile loading of each 2.7 m long fuel element (twelve per rod) has risen from 0.3 g U-235/cm to the present 1.8 g U-235/cm. Roughly 90 of the 227 reactor sites are occupied by driver fuel, and these are taken to approximately 80% burnup in 11 months.

As part of an international effort to reduce the use of HEU, and the risk of nuclear proliferation, a development program was undertaken at CRNL to produce low enrichment uranium (LEU, <20% U-235) driver fuel for NRU. CRNL's progress has been reported at annual meetings sponsored by the U.S. Reduced Enrichment for Research and Test Reactors (RERTR) program [1-4]. The salient features of the LEU fuel development and testing program, and the current status of the NRU conversion program are reviewed in this paper. The LEU fuel burnup analysis has recently been completed using high-precision liquid chromatography, and where appropriate, the analyzed burnup values will be given.

2. FUEL DEVELOPMENT AND FABRICATION

The NRU fuel-rod design which consists of twelve elements, each containing an HEU-Al alloy core with finned aluminum cladding and aluminum end plugs (see Figure 1), has proven to be extremely durable and reliable. The geometry, dimensions and linear fissile loading of this 12-element fuel rod were maintained in the LEU conversion program for licensing simplicity. CRNL physics studies showed very little effect on reactor operations from using LEU driver rods, apart from the need to fuel 6 more reactor sites or reduce exit burnup by about 10%. The problem thus came down to the development of a replacement fuel with five times the HEU fuel uranium density, which would be stable to high burnup.

The replacement LEU fuel elements consist of a core, containing high-density uranium silicide particles dispersed in an aluminum matrix, with the same finned aluminum cladding and end plugs. The fuel is made by melting uranium and silicon in a high-frequency induction furnace. Cast billets are heat-treated to transform the as-cast structure to U_3Si, then the billets are reduced to powders via a series of comminution processes. Uranium silicide powder is mixed with aluminum powder and hot-extruded into cores. The finned aluminum cladding is extruded onto the cores in a semi-continuous process, with the cores acting as a floating mandrel. The cladding is welded to the end-plugs to hermetically seal the elements.
The high-density silicides tested were $U_3Si$, $U_3Si$ alloyed with 1.5 wt% Al and 3 wt% Al, and $U_3Si_2$ ($U$-3.96 wt% Si; $U$-3.5 wt% Si-1.5 wt% Al; $U$-3.2 wt% Si-3 wt% Al, and $U$-7.3 wt% Si, respectively). Fuel elements were fabricated with uranium loadings suitable for NRU, 3.15 gU/cm$^3$, and for NRX, 4.5 gU/cm$^3$. Following the decision to shut down NRX, testing of NRX-type fuel was discontinued.

Industrial-scale fuel-production equipment has been installed in temporary facilities at CRNL, and is currently being used to manufacture prototype fuel rods to familiarize the operators with the manufacturing process, and to confirm that the production rate is high enough to meet CRNL's annual fuel requirements. To date, more than 90 kg of LEU has been processed into NRU rods (see section 4) and the production line is running satisfactorily. Most of the process variables and parameters established during the development program have been used in the production line and no problems have resulted from scaling up.

A new building has been constructed for LEU fuel fabrication. The manufacturing equipment is scheduled for installation and re-commissioning in late 1989. It is anticipated that full-scale production of NRU LEU fuel will commence early in 1990 in the new building, and the first production fuel rods will be loaded into NRU later that year. NRU should be completely converted to LEU fuel within a year of the first production of LEU fuel rods in the new building.

3. LEU SILICIDE DISPERSION FUEL TESTING

3.1. Test Elements

The test vehicle for irradiating silicide dispersion fuel in most tests has been the mini-element. The mini-element fuel-core diameter (5.5 mm) and clad wall thickness (0.76 mm) are the same as in full-size NRU elements. Mini-elements are, however, only 184 mm long compared with 2.9 m for NRU elements. The mini-elements also resemble NRU elements in that they have six cooling fins at 60° intervals around the cladding, the fin width being 0.76 mm and fin height 0.96 mm. To date, approximately 100 mini-elements have been successfully irradiated to burnups in the range 56-93% U-235 depletion in the NRU and NRX reactors.

3.2. Irradiation Conditions

The LEU silicide dispersion fuels were irradiated in NRU. A medium-enrichment uranium (MEU, 45% U-235) fuel tested as a backup was irradiated in NRX. The mini-elements were irradiated in a fuel carriage made from an aluminum cylinder with four holes bored axially through it at 90° intervals. An aluminum liner containing a string of four mini-elements was inserted into each hole or flow channel. The mini-elements were located centrally in the flow channels by four-pronged anodized spiders located on the end spigots of the mini-elements. The assembly could be loaded in any of the normal driver fuel positions in NRU.

The mini-elements were irradiated at linear powers representative of a typical NRU driver fuel rod, ranging from 40 to 112 kW/m. The typical
neutron flux was approximately $1.1 \times 10^{18}$ n/m$^2$/s, the heavy-water coolant flow approximately 7.28 L/s, and coolant velocity approximately 10.9 m/s. The coolant inlet temperature ranged between 30 to 37°C and the coolant outlet temperature between 40 to 45°C during the mini-element irradiations.

3.3. **Mini-Element Test Results**

The fuel test matrix is shown in Table 1. The detailed results were reported elsewhere [1-4], therefore only the salient features of the irradiations will be reported here.

3.3.1. **Comparison of HEU-Al and LEU Silicide Dispersion Fuel Performance**

In Exp-FZZ-905, mini-elements containing Al-21 wt% U alloy fuel (93% enriched U), Al-37 wt% U alloy fuel (45% enriched U) and Al-61.5 wt% USiAl (20% enriched U) silicide dispersion fuel, with 0.63 gU-235/cm$^3$, the fissile loading suitable for use in NRU (and for the new MAPLE-X reactors), were irradiated to 57% U-235 burnup (analyzed). The elements were in excellent condition after irradiation; they appeared as they did in the as-fabricated condition except for a dull oxide layer on the cladding. Post-irradiation examinations (PIE) showed that the performance of the uranium silicide dispersion and the MEU alloy fuel was comparable to the HEU alloy fuel. All element diametral changes were less than 1.6% and length changes were within 0.2% after 57 at% burnup. As shown in Figure 2, the Al-HEU fuel microstructure remained essentially unchanged after irradiation but the LEU silicide particles reacted with the aluminum matrix material, forming a thin interfacial layer, probably UAl$_3$ with dissolved Si, around the fuel particles. Fission-gas bubbles ranging in size up to 5 μm were contained in the kernels of the fuel particles. Considerably less fission-gas bubbles were retained in the interfacial layers. Fuel-core swelling ranged between 3.4 to 4.5 vol% at the exit burnup. From the results it appears that all three materials swell at roughly the same rate, up to 57 at% burnup.

3.3.2. **High-Burnup Confirmation**

In the Exp-FZZ-909B irradiation, mini-elements containing the USiAl and USi*Al dispersions, with 3.15 gU/cm$^3$, were irradiated up to 89 at% burnup (analyzed). Immersion density measurement indicated that the cores had swollen by 5.92-7.63 vol% after 77 at% burnup and by 6.57-7.76 vol% after 89 at% burnup, see Figure 3. These results showed swelling was approximately linear right up to 89 at% burnup, and confirmed that NRU-composition silicide dispersion fuels could exceed the design terminal burnup (80 at%) without exceeding the threshold of breakaway swelling.

PIE showed that both dispersions behaved similarly, i.e., the uranium silicide reacted with the aluminum matrix and fission-gas bubbles formed in the fuel particles. The interfacial layers were thinner near the fuel-core periphery and their edges were more sharply defined than at the fuel-core centre. The fission-gas bubbles were about the same diameter (up to 5 μm) in both locations but thicker interfacial layers and more particle coalescence had occurred at the fuel-core centre.
3.3.3. Dispersions with High U Loading and Fine Particles

In the FZZ-909A experiment, Al-USiAl and Al-USi*Al dispersions containing the higher loading required for NRX (4.5 gU/cm$^3$) were tested. Fuel performance was good and the materials behaved similar to the dispersions containing 3.15 gU/cm$^3$; however, swelling was marginally over 1 vol% per 10 at% burnup, at the terminal burnup of 74 at%.

In the Exp-FZZ-910 experiment, dispersions with a high loading of fine particles showed greater swelling compared to fuel with coarser particles at similar burnup. Mini-elements containing USiAl and USi*Al dispersions with fine particle-size distributions and 4.5 gU/cm$^3$ loading exceeded the swelling envelope of approximately 1 vol% per 10 at% burnup of the previous mini-element irradiations. The results suggest that particle size must be closely controlled to ensure good performance at high U loadings.

3.3.4. In-Reactor Corrosion

In-reactor corrosion behaviour of uranium silicide dispersion fuels has been investigated in the FZZ-911 and the FZZ-915 irradiations. The mini-elements contained Al-USiAl and Al-USi*Al, 3.15 gU/cm$^3$. In the FZZ-911A experiment, four mini-elements had 1.2 mm diameter holes drilled in the cladding mid-section, and were irradiated in the linear power range 60-87 kW/m in NRU. The first and second mini-elements were removed from the reactor after reaching 19 and 32 at% burnup, respectively, and the remaining two after 48 at% burnup (analyzed).

Post-irradiated metallography and neutron radiography revealed that ellipsoidal cavities had developed beneath the holes in the cladding. The cavity size increased with increasing burnup. These cavities correspond to 1.1 mg and 3.2 mg of U-235 lost to the coolant after 19 and 32 at% burnup, respectively, and 9.8-48.0 mg U-235 after 48 at% burnup. These results indicate that the corrosion rate of the purposely defected fuel elements is acceptably low.

The FZZ-915 experiment was similar to the FZZ-911 experiment, except that the mini-elements were pre-irradiated to burnups in the range of 23 to 79 at% (analyzed) before the 1.2 mm diameter holes were drilled in the cladding. These elements were further irradiated in the NRU reactor to evaluate the performance of the defective fuel, and during the additional 38 full-power days no increase in activity in the coolant above the normal background was detected. Neutron radiography and metallographic examinations revealed that the cavities were typically 0.7 mm deep by 1.3 mm across, i.e., only marginally larger than the original cavity made by the drill tip. These results indicate that the corrosion resistance of the LEU fuel is possibly increased by previous burnup.

3.3.5. Fuel-Core Surface Imperfections

The objective of the FZZ-911B experiment was to evaluate the performance of intact mini-elements having slight as-fabricated or deliberately introduced imperfections (machined grooves) in the core surface. However, the mini-elements contained the same fine particles that caused enhanced
swelling in Exp-FZZ-910. Therefore, examinations were also carried out to
determine the effects of the fine particle size on the core swelling when
the loading is 3.15 gU/cm$^3$.

Post-irradiation examinations revealed that the aluminum cladding had
flowed into and filled the surface defects. More importantly, the core
volume of the FZZ-911B mini-elements had only increased by approximately
4.9% after approximately 60 at% burnup compared with 7.0 to 17.5% swelling
in the FZZ-910 cores at about the same burnup. Swelling was 7.1 vol% af
after 80 at% burnup. These results indicate that core surface defects had
no detrimental effect on fuel performance, and fuel swelling was
acceptable at the lower silicide loading required for NRU, even when fine
particles were used.

3.3.6. Effect of Particle Size on Core Swelling

In the Exp-FZZ-918 experiment, 16 mini-elements containing Al-61.4 wt%
uranium silicide were irradiated in NRU to help establish limits on the
particle-size distribution to be used in the manufacturing specifications.
The mini-elements were divided into 4 groups, each group containing
progressively lower fractions of fines (particles less than 44 μm in
size).

Fuel swelling was linear with burnup, and to a first approximation
proportional to the percentage of fines contained. After 89 at% burnup
(analyzed) the mini-elements containing a high proportion of fines swelled
by 6.6 to 6.8 vol% while the mini-element with a low fraction of fines
swelled by 5.8 vol%.

3.3.7. Al-U3Si2 Dispersion Fuel

We have recently expanded the program to include U3Si2 dispersions to
complement the U3Si line. Twelve Al-U3Si2 mini-elements (3.15 gU/cm$^3$) were
fabricated with a variety of particle-size distributions and installed in
NRU in 1988 June. The assembly was removed for interim post-irradiation
examinations after reaching approximately 60 and 80 at% burnup, then it
was returned to the reactor to continue the irradiation to 93 at% burnup.
Metallographic examinations showed that the U3Si2 fuel behaved similarly
to the U3Si dispersions. An interfacial layer formed around the fuel
carbons, and fission-gas bubbles, ranging in diameter up to 10 μm, could
be seen in the fuel particles. The U3Si2 was not heat treated, and
contained 4 wt% free uranium, the highest level expected from local non-
uniformity in full-size castings. This appeared to have no detrimental
effect on fuel performance. However, no swelling dependence on particle-
size distribution was observed, and the Al-U3Si2 dispersion fuel swelling
was lower than Al-U3Si at similar burnup. The results are shown in Figure
4, and are compared with data from Al-U3Si mini-elements (Exp-FZZ-918).

3.4. Thermal Ramp Tests: Post-Irradiation Heating Tests

Thermal ramping tests were conducted in hot cells to determine the effects
of temperature excursions on the dimensional stability and fission-product
activity release from previously irradiated silicide dispersion fuel.
Whole mini-elements and short segments of mini-elements with the fuel meat
exposed were chosen, having fuel burnups of either 23 or 93 at%. Half the samples contained Al-61.5 w/o USiAl and half contained Al-62.4 w/o USi*Al.

The test conditions were: Argon gas flow rate - 1.66 mL/s; Heating rate - 0.2 and 0.4°C/s; Temperature - in the range 530 to 720°C; Holding time - temperature held constant (±4°C) for an hour.

In the thermal ramp tests, a whole mini-element irradiated to 93 at% burnup developed small localized blisters, some with pinhole cracks releasing fission products ($^{85}$Kr and $^{137}$Cs) after 0.25 h at 530°C. This behaviour prevented gross pillowing or ballooning. A mini-element irradiated to 93 at% burnup and ramped to 640°C developed radial cracks, which tore the cladding and released fission products from the core. Even at this high temperature the element maintained considerable structural integrity. This behaviour is interpreted to mean that coolant channels would not become blocked even if the fuel was subjected to some hypothetical abnormal event with the potential to cause overheating of the core, e.g. 530°C compared with the normal operating maximum of 200°C.

3.5. Full-Size NRU Fuel Test Results

The excellent performance of mini-elements containing the USiAl and USi*Al dispersion fuel led naturally to the fabrication and testing of full-size, 12-element NRU fuel assemblies. In experiment FZZ-913, seven assemblies were irradiated in NRU, 3 containing Al-62.4 wt% USi*Al and 4 containing Al-61.0 wt% U$_3$Si (identified as FL-001 to FL-007). The LEU assemblies were installed in NRU during 1984, replacing the currently used HEU alloy fuel in NRU, and were irradiated at typical driver-fuel operating conditions. The electrical conductivity of the coolant ranged between 0.25 and 0.7 μmho/cm during irradiation. The pH was not routinely measured, but ranged between 5.5 and 7. The coolant inlet temperature ranged between 30-37°C and the outlet between 60-70°C. Inlet pressure was approximately 80 psi (579 KPa) and outlet approximately 30 psi (207 KPa). Each rod occupies an average of 5 core positions during its lifetime (~340 d residence time @ 70% efficiency) as it is moved from the outside of the core (low-flux site) to the centre (high-flux site) and back to the outside. Average element linear power ranged from 40-50 kW/m, with the maximum being approximately 80 kW/m.

Burnup analysis showed that the rods were irradiated to 67 to 84 at% burnup (peak). Visual examinations showed that the fuel elements were in good condition; they were identical in appearance to the HEU elements with the normal aluminum oxide layer coating the surface. Dimensional measurements indicated that elongation during irradiation was negligible; the USi*Al and the U$_3$Si dispersion fuel elements were within 0.5% of their original length.

Post-irradiation metallography revealed that, as expected, the full-length elements' high burnup behaviour was similar to that of the mini-elements. The fuel particles had reacted with the matrix aluminum forming the normal aluminide interfacial layer (UA$_3$ with dissolved Si). The interfacial layer which formed around the U$_3$Si particles was considerably thinner than that in the USi*Al dispersion. Small fission-gas bubbles were contained in the kernels of the fuel particles and ranged in size up
to 10 μm in diameter. Considerably fewer fission-gas bubbles had been retained in the interfacial layers.

Evidence of the axial burnup gradient was observed in the full-length elements. In the high-burnup sections (mid-length) more particle coalescence had occurred than at the lower burnup sections (ends). There was less evidence of particle coalescence in the U₃Si dispersion. However, there was no evidence of the linking-up of fission-gas bubbles in any of the fuels examined, indicating that the fuels were well away from the onset of breakaway swelling. Immersion-density swelling measurements could not be made on the full-length elements, so estimates based on the dimensional changes from the underwater and metallographic examinations have been calculated. Core diameter increases of up to 3% and 4% have been measured at the ends and at the middle of the fuel elements, respectively. These give conservative estimates of the core's swelling by less than 1 vol% per 10 at% burnup at the terminal burnup of 84 at%.

It is clear that the high-burnup performance of USi*Al and U₃Si dispersion fuel containing 3.15 gU/cm³—the loading required for the research reactors at CRNL—is acceptable. Factors contributing to the good performance were the suitable particle-size distribution and the superior restraint provided by the thick-walled cladding.

4. STATUS OF NRU CONVERSION

NRU was partially converted to LEU when the first 31 LEU fuel rods manufactured using industrial-scale equipment were installed during 1988 September to December. This partial (one third) LEU core allowed the reactor engineers and physicists to evaluate the bulk effects of LEU conversion on reactor operations. Table 2 shows the burnup profiles of selected Al-U₃Si dispersion fuel rods from the first campaign, as of 1989 July. As expected, the irradiation is proceeding without incident and the reactor engineers have seen no difference in fuel-rod behaviour or handling compared to HEU fuel. In a given neutron flux the reactivity worth of the LEU rods is indistinguishable from HEU rods at comparable burnup. Bulk parameters such as coolant/moderator chemistry and overall reactivity have not changed perceptibly with the partial LEU core.

In 1989 July, the first of the rods reached their exit burnup. Two of these rods were cut apart in the NRU bays for PIE. Under-water visual examinations revealed clean, straight elements with only the light oxide coating on the cladding that is normally also seen on the HEU elements. No detailed metallographic examinations were planned since the rods behaved satisfactorily during irradiation, as expected. One of the elements which was inadvertently bent during handling demonstrated good ductility after full irradiation. By the end of 1989 all of the prototype LEU fuel rods will have completed their irradiation in NRU.

Fuel production in the new facility is expected to begin in 1990. It is expected that the full conversion of NRU will commence when a stable fuel production rate exceeding the fuel-usage rate is achieved. The projected date for this milestone is 1990 September.
5. CONCLUSIONS

1. Suitable LEU silicide dispersion fuels have been developed and tested at CRNL for the conversion of NRU from HEU to LEU. Under NRU fuel operating conditions, the fuels are stable up to burnups exceeding the design terminal burnup (80 at%).

2. NRU has been partially converted to LEU; 31 prototype LEU rods containing AL-61 wt% U3Si fuel (one third of a NRU core) were installed during 1988 September to December. The irradiation is continuing without incident, as expected. Post-irradiation examinations of the first rods to be discharged confirmed that the fuel achieved the design burnup in good condition.

3. Construction of a new fuel-fabrication facility is essentially completed and LEU fuel production is expected to begin in 1990. The complete conversion of NRU is expected to begin in 1990 September.

6. ACKNOWLEDGEMENTS

Many people, some of whom have since left CRNL, have contributed to the success of the LEU fuel development and conversion program. They are too numerous to properly acknowledge here. However, we would especially like to thank J.C. Wood, M.T. Foo, L.C. Berthiaume, L.N. Herbert, J.D. Schaefer, J.G. Goudreau, F.C. Iglesias, R.G. Barrand, J. Mitchell, J.R. Kelm, R.J. Chenier and the CRNL hot-cell metallography group, and E.J. McKee and the rod-shop group, for their contributions.

7. REFERENCES


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<th>TEST OBJECTIVES</th>
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<tr>
<td>FZZ-905</td>
<td>8</td>
<td>Al-61.5% USiAl</td>
<td>Compare LEU dispersions (3.15 MgU/m³) with U-Al alloys</td>
<td>LEU fuel performance comparable to MEU and HEU alloys</td>
<td>57</td>
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<tr>
<td></td>
<td>4</td>
<td>Al-21% U</td>
<td></td>
<td></td>
<td></td>
</tr>
<tr>
<td></td>
<td>4</td>
<td>Al-37% U</td>
<td></td>
<td></td>
<td></td>
</tr>
<tr>
<td>FZZ-909A</td>
<td>6</td>
<td>Al-72.4% USiAl</td>
<td>Test dispersions containing 4.5 MgU/m³</td>
<td>Fuel swelling marginally above 1 vol% per 10 at% burnup</td>
<td>73</td>
</tr>
<tr>
<td></td>
<td>6</td>
<td>Al-73.4% USi*Al</td>
<td></td>
<td></td>
<td></td>
</tr>
<tr>
<td>FZZ-909B</td>
<td>6</td>
<td>Al-61.5% USiAl</td>
<td>High burnup confirmation</td>
<td>Fuel swelling &lt;1 vol% per 10 at% burnup and linear up to the terminal burnup</td>
<td>89</td>
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<tr>
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<td>6</td>
<td>Al-62.4% USi*Al</td>
<td></td>
<td></td>
<td></td>
</tr>
<tr>
<td>FZZ-910</td>
<td>8</td>
<td>Al-72.4% USiAl</td>
<td>Test dispersions with fines</td>
<td>Particle size needs to be controlled to minimum swelling</td>
<td>51</td>
</tr>
<tr>
<td></td>
<td>8</td>
<td>Al-73.4% USi*Al</td>
<td></td>
<td></td>
<td></td>
</tr>
<tr>
<td>FZZ-911</td>
<td>4</td>
<td>Al-61.5% USiAl</td>
<td>Drilled defects in cladding</td>
<td>Corrosion resistance acceptable</td>
<td>18-48</td>
</tr>
<tr>
<td></td>
<td>8</td>
<td>Al-62.4% USi*Al</td>
<td>Fuel core surface imperfections</td>
<td>Surface defects have no detrimental effect Swelling 5.7 to 6.9 vol%</td>
<td>88</td>
</tr>
<tr>
<td>FZZ-913(^a)</td>
<td>36</td>
<td>Al-62.4% USi*Al</td>
<td>Full-size assembly irradiation</td>
<td>Both dispersions suitable for use in NRU. U₃Si chosen as reference</td>
<td>67-84</td>
</tr>
<tr>
<td></td>
<td>48</td>
<td>Al-61.0% U₃Si</td>
<td></td>
<td></td>
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<td>FZZ-915</td>
<td>6</td>
<td>Al-61.5% USiAl</td>
<td>In-reactor corrosion of pre-irradiated dispersions</td>
<td>Prior irradiation enhances corrosion resistance</td>
<td>23-79</td>
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<td>6</td>
<td>Al-61.5% USiAl</td>
<td></td>
<td></td>
<td></td>
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<tr>
<td>FZZ-918</td>
<td>16</td>
<td>Al-61.4% U₃Si</td>
<td>Define optimum particle size distributions</td>
<td>Swelling proportional to percentage of fines (5.8 to 6.8 vol% after 89 at% BU)</td>
<td>89</td>
</tr>
<tr>
<td>FZZ-921</td>
<td>12</td>
<td>Al-64% U₃Si₂</td>
<td>Compare to U₃Si and evaluate effect of particle size</td>
<td>Behaviour similar to U₃Si dispersions but less swelling</td>
<td></td>
</tr>
</tbody>
</table>

\(^a\) Full-length NRU 12-element assemblies
# Table 2

**Status of Select LEU Fuel Irradiations (As of 1989 July 31)**

<table>
<thead>
<tr>
<th>ROD ID</th>
<th>INSERTION DATE</th>
<th>REMOVAL DATE</th>
<th>INITIAL U-235 (g)</th>
<th>BURNUP (X U-235)</th>
<th>POWER MWe</th>
</tr>
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<tbody>
<tr>
<td>FLO08</td>
<td>88 00 29</td>
<td>89 07 30</td>
<td>495.7</td>
<td>80</td>
<td>320.1</td>
</tr>
<tr>
<td>FLO09</td>
<td>88 09 30</td>
<td>...</td>
<td>494.1</td>
<td>79</td>
<td>314.2</td>
</tr>
<tr>
<td>FLO10</td>
<td>88 10 01</td>
<td>...</td>
<td>493.9</td>
<td>79</td>
<td>313.1</td>
</tr>
<tr>
<td>FLO11</td>
<td>88 10 04</td>
<td>89 07 30</td>
<td>497.0</td>
<td>80</td>
<td>321.3</td>
</tr>
<tr>
<td>FLO12</td>
<td>88 10 06</td>
<td>89 05 31</td>
<td>494.0</td>
<td>61</td>
<td>244.9</td>
</tr>
<tr>
<td>FLO15</td>
<td>88 10 13</td>
<td>...</td>
<td>492.6</td>
<td>77</td>
<td>304.9</td>
</tr>
<tr>
<td>FLO20</td>
<td>88 11 07</td>
<td>...</td>
<td>493.3</td>
<td>72</td>
<td>285.3</td>
</tr>
<tr>
<td>FLO28</td>
<td>88 12 12</td>
<td>...</td>
<td>492.1</td>
<td>71</td>
<td>282.5</td>
</tr>
<tr>
<td>FLO38</td>
<td>88 12 28</td>
<td>...</td>
<td>494.0</td>
<td>54</td>
<td>215.8</td>
</tr>
</tbody>
</table>

![Diagram](image-url)

**Figure 1**

NRU Fuel Rod
Fig. 2. Microstructure of: (a) Al-21 wt% U fuel after 57 at% burnup, UA14 - dark grey, Al - light grey; (b) Al-61.5 wt% USiAL after 57 at% burnup, U3Si - dark grey, UA13 - light grey, Al - white.
Fig. 3 Swelling of LEU fuel core containing Al-61.5 wt% USiAl and Al-62.4 wt% USi*Al.

Fig. 4 Swelling of LEU fuel core containing Al-61 wt% U3Si and Al-64 wt% U3Si2.
Upgrading activities for the HFR Petten

J. Ahlf (Joint Research Centre, Petten, the Netherlands)

W.L. Zijp (Netherlands Energy Research Foundation, Petten, the Netherlands)
Upgrading activities for the HFR Petten

Abstract

The HFR in Petten, the Netherlands, is a water cooled and moderated research reactor. It has been in continuous and successful operation during more than 25 years. The reactor is utilized as a multi-purpose research reactor with a predominance of materials testing for fission and fusion energy. Other programmes include nuclear physics and solid state physics at the beam tubes and general services such as radioisotope production, activation analysis and neutron radiography. It has been continuous policy to keep the installation up to date by implementing technical developments and by refurbishing or replacing all components and equipment which approached the end of their useful life. In addition the facilities and the ancillary experimental equipment was continuously adapted and kept versatile in view of changing requirements from the experimental programmes. Performance upgrading comprised increase of the initial power of 20 MW in two steps to 30 MW and now 45 MW, accompanied by improving the core loading pattern in order to provide an increasing number of high flux irradiation positions. These improvements were rendered possible because of the achievements in fuel element design and manufacture. In the mid 70's it became apparent that embrittlement of the reactor vessel material would become a licensing problem in the foreseeable future. A decision was taken to replace the old vessel by a new one which then could take into account recent experience with respect to experimental requirements. After the vessel replacement a programme was started to replace other ageing components. The primary heat exchangers and the pool heat exchanger have been replaced recently; replacement of the beryllium reflector is nearly finished. All the nuclear instrumentation channels have been replaced. Repair or refurbishment of peripheral equipment such as the outlet-line for the secondary cooling, the guaranteed power supply for the reactor and the fire prevention system is under way. A full upgrade of the control room is under preparation. Because all upgrading actions were carefully planned well in advance of actual component failures, unanticipated outages could be avoided.

1. INTRODUCTION

The High Flux Reactor (HFR) at Petten, the Netherlands, belongs to the Institute for Advanced Materials of the Joint Research Centre (JRC) of the European Communities. It is operated under contract by the Netherlands Energy Research Foundation (ECN).

The HFR has been designed according to the principles of the well-known Oak Ridge Research Reactor. It is light water cooled and moderated. The core is situated in a closed-top aluminium tank, situated at the bottom of a water-filled pool, 9 meter deep. The reactor is exploited as a multi-purpose research reactor with a predominance of materials testing for fission and fusion energy deployment, but also intense use of the beam tubes for research with neutrons.
Since a long period the irradiation positions inside and outside the core and at the beam tubes are occupied to a very high extent. The favourable exploitation record is achieved by maintaining the following principles:
- the operation schedule is planned far in advance and kept as strictly as possible in order to provide for a high availability and dependable irradiation conditions;
- technical progress in fuel technology, instrumentation and experimental techniques is adopted and implemented in order to upgrade performance with respect to operation schedule and irradiation conditions;
- refurbishing actions are carefully planned well in advance of actual component failures in order to avoid unanticipated outages;
- changes in exploitation pattern are anticipated in close contact with major clients and development activities for appropriate experimental equipment are started at an early stage.

2. UPGRADEING OPERATIONAL PERFORMANCE

Design of the HFR commenced in 1958 and first criticality was reached in November 1961. The major milestones of HFR's history are contained in Table I. The maximum power was raised from initially 20 MW to 30 MW in 1966 and to 45 MW in 1970. The power increases were possible without changing the cooling system. Later refurbishing actions (see below) were always made with the option to increase power to 60 MW. But as yet the exploitation programme did not urgently call for going beyond 45 MW.

Table I: HFR Petten, History

<table>
<thead>
<tr>
<th>Year</th>
<th>Event</th>
</tr>
</thead>
<tbody>
<tr>
<td>1958 - 1961</td>
<td>Design and construction</td>
</tr>
<tr>
<td>1961</td>
<td>First criticality of HFR (November 9)</td>
</tr>
<tr>
<td></td>
<td>Maximum power 20 MW</td>
</tr>
<tr>
<td>1962</td>
<td>Transfer from RCN to EURATOM (October 31)</td>
</tr>
<tr>
<td>1966</td>
<td>Power increase to 30 MW (May 8)</td>
</tr>
<tr>
<td>1970</td>
<td>Power increase to 45 MW (February 20)</td>
</tr>
<tr>
<td>Mid 1972</td>
<td>Introduction of burnable poison</td>
</tr>
<tr>
<td>1974 - 1977</td>
<td>Feasibility study for replacement of reactor vessel</td>
</tr>
<tr>
<td>1978</td>
<td>Decision to replace reactor vessel</td>
</tr>
<tr>
<td>1980 - 1981</td>
<td>Design of new reactor vessel</td>
</tr>
<tr>
<td>Nov.'83 - Feb.'85</td>
<td>Period of shut-down for reactor vessel replacement</td>
</tr>
<tr>
<td>Jan. 1985</td>
<td>First criticality after vessel replacement</td>
</tr>
<tr>
<td>1987</td>
<td>Replacement of primary heat exchangers</td>
</tr>
<tr>
<td>1988/1989</td>
<td>Replacement of beryllium reflector elements</td>
</tr>
<tr>
<td>1989</td>
<td>Replacement of pool heat exchangers</td>
</tr>
</tbody>
</table>

From the beginning it was general policy to operate the reactor with a standard core configuration, in order to provide for reproducible irradiation conditions. Nevertheless the standard core configuration was
stepwise improved mainly by adopting developments in fuel technology to provide for an increasing number of in-core positions with high neutron flux densities. The development of the HFR core configuration and number and flux characteristics of the in-core positions is shown in Fig. 1.

As stated, upgrading of performance was strongly related to the implementation of fuel technology improvements. In quite a number of steps the number of plates per fuel assembly and per control assembly as well as the mass of U-235 per assembly was increased. A further major step was the introduction of B-10 as a burnable poison. The history of fuel improvement is shown in Table II, whereas the changes of operating schedule and fuel management is given in Table III. The present reference operating schedule per year comprises 11 operation cycles of 25 days followed by a 3-days shut-down period for core reloading and experiment reshuffling. Two periods of about 25 days in spring and summer are provided for major maintenance and refurbishing activities. It is mentioned that because of uncertainties and disturbances in the fuel cycle the operation cycle was reduced to 21 days mid 1988. It is anticipated that full operation is resumed with the beginning of 1990.

Table II: HFR Petten, History of Fuel Type

<table>
<thead>
<tr>
<th>Date</th>
<th>Fuel assemblies</th>
<th>Control assemblies</th>
</tr>
</thead>
<tbody>
<tr>
<td></td>
<td>Number of plates</td>
<td>Mass of U-235 [g]</td>
</tr>
<tr>
<td></td>
<td></td>
<td></td>
</tr>
<tr>
<td>Nov. 1961</td>
<td>19</td>
<td>120</td>
</tr>
<tr>
<td>Feb. 1963</td>
<td>19</td>
<td>160</td>
</tr>
<tr>
<td>Nov. 1964</td>
<td>19</td>
<td>180</td>
</tr>
<tr>
<td>Mar. 1969</td>
<td>23</td>
<td>200</td>
</tr>
<tr>
<td>Sep. 1969</td>
<td></td>
<td></td>
</tr>
<tr>
<td>Oct. 1969</td>
<td>23</td>
<td>217</td>
</tr>
<tr>
<td>Nov. 1969</td>
<td>23</td>
<td>240</td>
</tr>
<tr>
<td>Feb. 1970</td>
<td></td>
<td></td>
</tr>
<tr>
<td>Aug. 1972</td>
<td>23</td>
<td>345</td>
</tr>
<tr>
<td>May 1973</td>
<td>23</td>
<td>345</td>
</tr>
<tr>
<td>Jul. 1973</td>
<td></td>
<td></td>
</tr>
<tr>
<td>Feb. 1974</td>
<td>23</td>
<td>390</td>
</tr>
<tr>
<td>Mar. 1975</td>
<td>23</td>
<td>390</td>
</tr>
<tr>
<td>Mar. 1978</td>
<td>23</td>
<td>390</td>
</tr>
<tr>
<td>May 1981</td>
<td></td>
<td></td>
</tr>
<tr>
<td>Oct. 1981</td>
<td>23</td>
<td>405</td>
</tr>
<tr>
<td>Mar. 1985</td>
<td>23</td>
<td>420</td>
</tr>
</tbody>
</table>

*) B-10 in the fuel meat
**Table III: HFR Petten, Characteristics of Operation Cycles**

<table>
<thead>
<tr>
<th>Year Period</th>
<th>Duration [days]</th>
<th>Initial mass of U-235 [g]</th>
<th>Fresh elements per cycle</th>
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</thead>
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<tr>
<td>1962 - 1966</td>
<td>15-18</td>
<td>± 5000-6000</td>
<td></td>
</tr>
<tr>
<td>1966 - 1970</td>
<td>15-18</td>
<td>± 6200</td>
<td></td>
</tr>
<tr>
<td>1970 - 1973</td>
<td>12</td>
<td>± 6700</td>
<td>7 * 240 g fresh element</td>
</tr>
<tr>
<td>1972 Cycle 8</td>
<td>18</td>
<td>± 9700</td>
<td>6 * 345/1000</td>
</tr>
<tr>
<td>1973 Cycle 1</td>
<td>36</td>
<td>± 10500</td>
<td>8 * 345/1200</td>
</tr>
<tr>
<td>1973 Cycle 6</td>
<td>36</td>
<td>± 11000</td>
<td>8 * 390/1200</td>
</tr>
<tr>
<td>1974 Cycle 3</td>
<td>36</td>
<td></td>
<td>10 * 390/1000 meat</td>
</tr>
<tr>
<td>1974 Cycle 8</td>
<td>36</td>
<td></td>
<td>10 * 390/1200 side plate</td>
</tr>
<tr>
<td>1975 Cycle 6</td>
<td>36</td>
<td></td>
<td></td>
</tr>
<tr>
<td>1976 Cycle 5</td>
<td>26</td>
<td>± 11500</td>
<td>6 * 390/1200</td>
</tr>
<tr>
<td>1978 Cycle Mar.</td>
<td>26</td>
<td>± 11000</td>
<td>6 * 390/1000</td>
</tr>
<tr>
<td>1981 Cycle Oct.</td>
<td>26</td>
<td>± 11000</td>
<td>6 * 405/1000</td>
</tr>
<tr>
<td>1985 Cycle Mar.</td>
<td>26</td>
<td>± 11500</td>
<td>6 * 420/1000</td>
</tr>
<tr>
<td>1988 Cycle Jan.</td>
<td>26</td>
<td>± 12000</td>
<td>7 * 420/1000</td>
</tr>
</tbody>
</table>

The reactor is operated with five shifts of operators. In addition there are three maintenance groups within the reactor department, a mechanical maintenance group, an electrotechnical group and a group for instrumentation and informatics. Apart from maintenance work these service groups play an active and important role in design, planning and co-ordinating refurbishment and upgrading activities.

3. **EXPERIMENTAL FACILITIES**

The HFR being designed as a multi-purpose research reactor, has been used over more than 20 years for all kinds of materials testing, for radioisotope production at large scale and other processes making use of the transmutation of nuclei by neutron reactions, as well as for research with neutrons at the beam tubes. Also neutron radiography and more recently neutron capture therapy have been addressed in the programmes [2].

Whenever advisable in view of the anticipated programme, experimental equipment and ancillary and supporting facilities were standardized and closely connected to reactor operation. The major items are briefly addressed:
- in-pile sections for in-core positions and poolside facilities are composed of standardized components which are held in stock;
- for special experimental requirements a variety of instrumentation options is held available;
- dedicated devices for all kinds of radioisotope production via the fission and neutron capture routes have been developed and steadily upgraded;
- a water make-up and supply system to serve a large number of rigs for LWR related experiments is in service;
- a versatile gas supply system (pressurized helium, neon and nitrogen manyfold) for temperature control of irradiation rigs is installed in the reactor hall;
- a multi-channel sweep loop system for studies of fission product release from HTR fuel and studies of tritium release from fusion blanket breeder materials is in operation;
- neutron radiography facilities comprising a pool and a beam tube device have been built and are under continuous development;
- a centralized data acquisition system serves the experimental groups as well as the reactor operators.

Detailed information on the HFR and its experimental facilities is given in Ref. [1].

For large materials testing programmes a research reactor is not selfstanding but needs the embedment into the infrastructure of a large nuclear research centre. In this respect the close and fruitful co-operation between JRC and ECN is worth noting. This remark refers to neutron metrology, reactor physics calculations, and the on site availability of the well equipped ECN hot laboratories.

4. VESSEL REPLACEMENT

In the mid 70's it was realized that embrittlement of the vessel material would become a licensing problem in the future. So it was decided to install a new vessel. The vessel replacement was carefully planned and prepared. The whole action of removing the old vessel and replacing it by a new one took place in a shut-down period not longer than 15 months. Comprehensive reports on this unique action have been given recently (see ref. [3] and [4]).

The design of a new vessel offered the opportunity to introduce major improvements with respect to experimental utilization (see Fig. 2). The accessibility of the pool side facility was improved considerably. A second pool side facility was provided on the opposite side of the first one. The old thermal column was removed and replaced by two large cross-section beam tubes which are in future intended to be used for neutron capture therapy.

It is noteworthy that the radiation dose to employees of the reactor department decreased drastically after the introduction of the new reactor vessel. This decrease is among others due to a restraint structure and special jets to influence the coolant flow near the reactor vessel which reduces the instant release of N-16.

5. MAJOR REFURBISHING PROJECTS

With the vessel replacement a period of more than 20 years ended in which worldwide achievements in operating and exploiting research reactors were adopted to upgrade the HFR. On the other hand the vessel replacement was the first and of course most important step to refurbish worn-out or outdated components and equipment. A number of important refurbishment actions followed in recent years.
In 1988 the original heat exchangers for the primary circuit which were originally designed for 20 MW and had served for more than 25 years at power levels up to 45 MW were replaced by new ones of enlarged heat removal capacity. Vibrations of the inner tubes and a loose baffle plate indicated degrading performance and secondly increased heat removal capacity was necessary in view of the option to increase the reactor power further.

A next step was the replacement of the pool cooling heat exchanger, comprising enlargement of the heat transfer capacity of the pool system again in view of a future power increase. The replacement was performed in the March stop of this year.

Also the Beryllium reflector elements at the periphery of the HFR core region served for a long period. Replacement of the original elements became necessary, since they are affected by mechanical damage and wear-out as well as by local deformations due to a combination of frequent handling and of irradiation induced dimensional changes and embrittlement. All 28 elements are exchanged against new ones in 1989.

Another upgrading and refurbishing project is the extension of the reactor power safety protection system. The objective is to provide redundancy and diversification for the present 3-channel flux protection system. Through the presence of two such systems, both working in a two-out-of-three mode, the vulnerability against failures is significantly decreased and likewise other spurious causes of safety system interventions. This project is also nearly finished.

An important step in the direction of further improving the control and evaluation of experimental and reactor operation data is the upgrading of the central data acquisition system DACOS. Both the data logger system and the front-end computers have been renewed in summer 1988 [4]. The modified system is much faster in response than the old system. The number of read-out channels has been doubled to 2048 and the speed of the sensor scanning improved drastically. Implementation of the system is in progress.

The former rotating uninterrupted power supplies have been replaced by two static no-break sets of 40 kVA. These units are connected in parallel in a redundant circuit, so that each unit can take the full load if the other unit fails to function.

Whereas parts of the initial instrumentation have been replaced years ago, a complete renewal of the HFR control room is now under preparation. In the course of the modernization outdated components will be replaced and the whole equipment will be designed according to modern ergonomic principles. This comprises also modernized equipment for information access and display. The whole project will be executed stepwise over a period of several years during normal maintenance stops in order to avoid additional outage.

Measures are under investigation to renew the outlet-line of the secondary cooling circuit which extends from the HFR building via the beach to the North Sea and which has deteriorated seriously under the action of wind and corrosive environment.
On request of the licensing authorities a complete reappraisal of the safety analysis is in progress which will replace the old HFR hazard report written roughly 30 years ago. This safety analysis will be the basis for a new safety report which is asked for in view of a renewal of the present HFR operating license.

6. SUMMARY AND CONCLUSIONS

The multi-purpose research reactor HFR Petten has been continuously upgraded and modernized since its first criticality in 1961. After a number of important refurbishment actions in the last 5 years starting with the vessel replacement in 1984 the reactor can be regarded as a modern and up-to-date research tool even after nearly 30 years of full utilization. The sound basis of managing the reactor that way is underlined by impressive records with respect to high availability (Fig. 3), high utilization (constantly between 70 and 80% of the technical limit) and low dose equivalent to operating staff.

7. REFERENCES


Fig. 1 Development of HFR Core Configuration
Penetrations for in-core irradiation capsules

Pool side facility II

Primary coolant inlet

Core box

Horizontal beam tubes

Pool side facility I

Primary coolant outlet

Control rod drive mechanisms

Fig. 2 New Reactor Vessel of HFR
Fig. 3 Availability of the HFR from 1961 to 1988
ABSTRACT

The work reported in this paper presents the tasks completed or currently under completion for the renewal of the Nuclear Instrumentation and Control System, Radiation Protection System and Process Instrumentation System for Egypt's first research reactor (ETRR-1). The mentioned tasks started in 1980. The work reported includes the procurement and installation procedures and gives also a historical background which introduces ETRR-1 and its operating history together with the need for and philosophy behind the renovation of the above mentioned systems which were first put in operation in 1961.

I.1 Historical Background

ETRR-1 is a 2 MW thermal heterogenous research reactor of the type WWR-C supplied by the Soviet Union. It went critical for the first time in the fall of 1961. The reactor is designed for Neutron Beam Experiments, Isotope Production and Biological Experiments. It has been operating on regular intermittent basis except for shut down periods for major maintenance and repair.

In the early seventies the nuclear power program which was initiated in 1964 was reactivated. It was thought, then, to extend the use of ETRR-1 as an experimental & training facility for the implementation of the national nuclear power program. To meet these new requirements two alternative approaches were considered:-

a) Upgrading ETRR-1 and boosting its rated power to meet the new material/fuel testing, loop experiments, systems performance analysis in order to cope with the national nuclear power program.

b) Construction of a new experimental/prototype reactor facility (in the 20-30 MW range) to meet the requirements of the national nuclear power program keeping ETRR-1 in its existing conditions. Due to the relatively large investments needed for the construction of a new experimental reactor facility it was decided to start with the first alternative to upgrade ETRR-1 and extend its operating lifetime. Before proceeding with the upgrading tasks an inspection was carried out on ETRR-1 internals which revealed the following:-

- Reactor tank and mechanical components (pumps, heat exchangers, control drives-etc) can withstand normal operating schedule for at least 10-15 years.
- Stored fresh fuel in stock is enough to operate the reactor at its rated power level according to current operating schedule for the same period.
- Reactor instrumentation for safety, control and process systems should be renewed in order to support the operation of the reactor for such extended period.

Thus an extensive program was initiated in the late seventies in order to reconstruct and modernise the instrumentation systems of ETRR-1 keeping it at the same design rated power (2 MW.). Building of the new experimental prototype reactor facility has thus been delayed for 10-15 years until enough funds can be allocated for such big task.

I.2 Renewal Philosophy

The need for renewal of the reactor instrumentation systems in the early eighties has a significant importance at that time, because it will probably be exploited during the rest of the reactor life time. Partial renewal of the reactor instrumentation systems was not recommended since the mixing of old and new parts and components could not lead to the needed sufficient and long lasting improvements. The cost of total renewal of the reactor instrumentation systems amounts only to a small fraction of the cost of a new reactor facility. Thus there was a clear economic advantage to renew the existing ETRR-1 instrumentation systems and some other significant parts of it until the new experimental/prototype reactor facility project sees the light.

The need for complete renewal of all ETRR-1 instrumentation systems was based on the following propositions.
1) Aging and obslence of the old systems which was designed in the mid fifties and was operating for more than 20 years. The difficulty in securing the spare parts for the old designed electronic systems made it difficult to operate the reactor without prolonged maintenance periods. This imposed large burden on the reactor operation & maintenance staff.

2) The design of the old system was non modular with insufficient reliability and need for long repair time.

3) Rapid development in the electronic industry made it unfeasible to make partial renewal of the instrumentation systems due to incompatibility of the old and new electronic components.

II- RENOVATION OF INSTRUMENTATION OF ETRR-1

This section outlines the approach followed in the renovation process of the three categories of systems, namely
1- Instrumentation and control (I & C) system.
2- Radiation Protection System (RPS).
3- Process Instrumentation System (PIS).

For each category the following items are presented:-
- General Description of the new system with emphasis on new design philosophy.
- Procurement and Installation tasks.
- Renovation effect on improving safety features.
II.1 Nuclear Instrumentation and Control (I & C) System

II.1.1 General Description

The new I & C system is composed of the following subsystems:
- Safety channels.
- Logarithmic Channels.
- Linear Switchable Channel.

II.1.1.1 Safety Channels

The safety channels consist of three independent channels which monitor the reactor power. These are connected to the emergency circuit through a 2 out of 3 logic. Each safety channel has three (3) ranges (10⁻⁶ amp., 10⁻⁵ amp. & 10⁻⁴ amp.) which ensure safe operation of the reactor at different power levels.

Fig. 1 presents a schematic of the safety channels showing its modular electronic components.

The old safety system was composed of only one channel in the start-up range and two channels in the power range with one common powersupply. There was no buffer amplifiers or recorders on these channels.

II.1.1.2 Logarithmic Channels

Three independent channels are used to measure the reactor period during the start-up of the reactor. These are connected to the emergency circuit through 2 out of 3 logic.

Fig. 2 presents a schematic of the logarithmic channels showing its modular electronic components.

In the old system there were no any logarithmic channels instead the doubling time was measured by the operator using a stop watch.

II.1.1.3 Linear Switchable Channel

This channel measures the reactor flux from 10⁻¹² amp to 10⁻⁴ amp. in 16 ranges which allows the precise change in reactor power to the detected. This channel is connected to the automatic rod drive system through a control unit (Motric 96 E) to control the reactor power in the automatic mode of operation.

Fig. 3 presents a schematic of the linear switchable channel showing its modular electronic components. The Motric 96-E is an on/off switch operated from the input signal from the linear amplifier. It connects the voltage of the automatic control rod to the motor drive if the difference between the measured reactor power and the reference value exceeds ± 1%.

In the old system an amplidyne was used to amplify the difference voltage which operate the motor drive of the automatic control rod. A comparator using d-c supply from a dry cell was used to produce the reference voltage. The resulting transient response was not as good as in the new system.

From the above description of the new I & C system it is clear that it included the following improvements compared to the old system:-
A- Independency: Through the use of independent ionization chamber, power supply and high voltage supply for each channel.

B- Reliability: Through the use of 3 channels for each system with a 2 out of 3 logic unit which enables on line testing.

C- Availability: Through the use of standard modular units and identical components in all units. This enables the operator to interchange the components in the three units without the need of too many spare components. Modular design and identical components made the maintenance a lot easier.

The safety of the reactor is greatly enhanced by introducing the above improvements.

II.1.2 Procurement and Installation

Through a bilateral agreement with KFA Julich, in F.R.G., the design of the new system was jointly developed through 1980. Hartman & Brown company in F.R.G. was selected as the supplier of the new system. Installation was carried out by ETRR-1 operation and control staff with the help of some experts from KFA Julich. Installation was completed in 1984.

II.2 Radiation Protection System (PRS)

II.2.1 General Description

The RPS is designed to cover the monitoring of radiation levels at selected areas in the reactor building. The system is composed of 30 channel gamma radiation assembly, together with an air monitoring and water activity controlling equipment. The distribution of these detectors in the reactor building is shown in Fig. 4. Twenty five channels of the RPS utilize silicium semi conductor detectors covering the different ranges of radiation exposure from 0 to 3 x 10^4 uSV/h in five steps. The other five channels utilizes Geiger-Muller detectors as gamma indicators for measuring the activity concentration in the sampled air. Each channel has an alarm output with adjustable level. Monitored radiation levels are continuously compared with alarm set points for each detector channel. As the radiation level approaches the set points, alarm is initiated at the central unit. At the same time alarm signal appears at the detector itself which is equipped with alarm unit. The central unit of the RPS includes a series of electronic channels mounted on 10 racks. These racks contain alarm indicators, linear rate meters of the 30 channels, printer, switches of air monitors etc... Eight of the 30 signals are relayed to the operator in the reactor control room which is located one floor above the location of the central unit.

II.2.2 Procurement and Installation

The system was supplied by Hungary through IAEA technical assistance (project No EGY/9/015). Installation was carried out by the operation reactor staff together with the Hungarian experts. First the old system was dismantelled and the wiring completely changed to install the new system which was first operated in 1988. After operation of one year, recalibration and adjustment were under taken in march 1989.
The new system has a lot of advantages over the old system which can be summarized in the following:
- Standardization: the new system contains standard units so it can be replaced by any other similar units.
- Independency: each channel has its own detector, amplifier, rate meter and alarm system.
- Availability: the channel use identical units which enable the change of one unit by another unit from other channels.

The new system has greatly enhanced the radiological safety measure in the reactor building.

II.3. Process Instrumentation System (PIS)

II.3.1 General Description

The PIS consists of 30 channels to measure the process parameters. The sensors in the different locations transfers the parameters to current which is amplified and send a volt signal to 3 central units in the control room, where it is displayed and recorded. The parameters measured are:
- Primary coolant circuit: pressure, flow, temp., temp. difference, flow in filter, pump bearing temp.
- Dearator: water level, flow, air flow, outlet temp. depression.
- Secondary coolant circuit pressure, flow, temp., temp. difference.
- Levels: central tank, shield tank, distillate water tank, spent fuel storage, wast storage tanks 1 & 2.
- Air depression: above reactor, under reactor, in spent fuel tank in hot cells, pumproom, before ventilators.
- Thermal power indicator.
- Water conductivity in distillate water tanks.

The 3 central units contain the amplifiers, displayers and alarm signals for each channel as shown in Figures 5,6,7.

II.3.2 Procurement and Installation

The PIS system was supplied from Hungary through IAEA technical assistance (project o. EGY/4/028), in 1989. It is anticipated to be installed by the Hungarian experts and ETRR-1 operation and control staff during the last quarter of 1989.

The new system is a modular system having the same advantages mentioned before for the other two systems namely standardization, availability, reliability and independency.

III- CONCLUSIONS

The tasks undertaken for the renewal of the instrumentation systems of ETRR-1 was cost effective and unique for a developing country. The work highlighted the importance of IAEA technical assistance programs in helping a developing country in the implementation of rehabilitation programs for its nuclear facilities independently from the reactor supplier country.

ETRR-1 was operating since 1961 and the decision to renovate its instrumentation systems was taken in 1980 with Egypt's new open door policy after the 1973 war. The completed work indicated the importance of self dependance in performing the design and implementation of rehabilitation tasks and securing the proper
financing channels for these tasks.

The experience gained in performing such an important job justifies the effort and money spent on it.

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Fig (4-a) Location of RPS detectors in 1st Floor

Fig (4-b) Location of RPS detectors on ground floor
Fig(5) Rack1 of PIS
The Modification of the Rossendorf Research Reactor
Technical Performance and Responsibility of the Control and Licence Authorities

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The Modification of the Rossendorf Research Reactor, Technical Performance and Responsibility of the Control and Licence Authorities

Abstract

The Rossendorf Research Reactor of the WWR-SM type is a heterogeneous water moderated and cooled tank reactor with a thermal power of 10 MW, which had been in operation from 1957 to 1986. It has been shut down 1987 for a comprehensive modification to enlarge its nuclear safety and to improve the efficiency for irradiation and experimental works. The modification will be implemented in two steps, the first one to be finished in 1989 comprises

i. the replacement of the reactor tank and its components, of the reactor cooling system, of the ventilation and of the electric power installation,

ii. the construction of a new reactor control room and of filtering equipments,

iii. the renewal of the process instrumentation and control engineering equipment for reactor operation, of the equipment for radiation protection monitoring, and of the reactor operation and safety documentation.

The second step to be implemented in the nineties is to comprise

i. the enlargement of the capacity for the storage of spent fuel,

ii. the modernization of the reactor operation by computer aided control,

iii. the installation of an automated measuring system for accidental and environmental monitoring.

Two objects of the modification, the replacement of the reactor tank and the design of a new and safer one as well as the enlargement of the redundancy of the core emergency cooling system are described in detail. For the tank replacement the exposure data are also given.

Furthermore, the licencing procedure based on national ordinances and standards as well as on international standards and recommendations and the mutual responsibilities and activities of the licencing authority and of the reactor manager are presented. Finally, the present state
of the modification and the schedule up to the reactor recommissioning and test operation at full power is outlined.

1. Introduction to the reactor RFR

The Rossendorf Research Reactor (RFR) of the WWR-SM type is a heterogeneous water moderated and cooled tank reactor. It was put into operation in 1957. The original nominal thermal power of 2 MW was increased in two steps, to 5 MW in 1965 and to 10 MW in 1979 /1/. The fuel is an aluminium clad Al-U-dispersion enriched in U-235 by 36%. The material of the reactor tank and of some important components like separator, core holder and irradiation channels is an aluminium alloy.

The reactor has been mainly used for:
- basic and applied research in nuclear physics,
- applied research in nuclear energetics,
- developing and testing new technical methods of reactor diagnostics,
- radioactive isotopes and neutron transmutation doped silicon,
- short term irradiation for activation and structural analyses.

For that 9 horizontal and 35 vertical irradiation channels are installed at the reactor.

In 1973 a modernization of the nuclear instrumentation system /2/ was carried out.

The averaged annual operation time at nominal power was in the eighties about 4000 hours.

The RFR has been shut down for a comprehensive modification in December, 31 1986.

2. Basis and aim of the modification

The preparation of the RFR modification started in 1977 with a study on possible improvements of neutron irradiation and experimental works and of their efficiency by further increasing the thermal power and on the related costs. In the following years the study splitted up for two variants, first to retain the 10 MW power but to improve the irradiation and experimental facilities and second to enlarge the core and the thermal power to 15 MW. Resulting from these investigations and from studies on safety topics of the reactor tank and some of its important components a modification programme for the RFR was elaborated. It was based on a comparison of the given
state and the "shall" concerning the requirements in national ordinances on nuclear safety and radiation protection /3-5/, in standards and guides on the technical, organizational and personnel prerequisites for the safe operation of nuclear facilities /6/ as well as the special requirements of the National Board on Atomic Safety and Radiation Protection (SAAS) /7/ and the corresponding international recommendations of the IAEA /8/ and the CMEA /9/. Furthermore it was based on the documents:

i. conception for the improvement of the prerequisites for neutron irradiation and experimental work at the RFR from 1984,

ii. conception for the various tasks in overhauling and backfitting the RFR from 1984,

iii. fundamental requirements of nuclear safety and radiation protection to the various tasks of the RFR overhauling from 1984.

The modification didn't concern the thermal power, it should be retained at 10 MW.

The modification programme was confirmed by the SAAS. The start of the modification was the very beginning of 1987. The technical and technological projects for the overhauling were elaborated by or together with the industrial contractors from 1985 to 1986 and implemented from 1987 to 1989.

The aims of the modification were first to increase the nuclear safety of the RFR by replacing main components by new designed and constructed ones and second to achieve more efficiency in irradiation and in experimental work.

3. The state of the main components of the reactor system before the modification

Most of the components of the reactor facility, i.e. the reactor tank, the cooling circuits and the control room have been in operation since 1957, of course under periodic inspections and with a lot of backfittings /10-14/. The design and construction of the reactor itself enabled all important inspections but those of the tank bottom, the lower welding seams and the outer tank surface.

The secondary cooling circuit was operated in an off-line regime with all the disadvantages of such a system, e.g. the demand for a continuous drinking water supply (about 50% of the institute's total drinking water demand) and the oversalting of the circuit components due to vaporization (about 2 m³ per MWh).
In the stack the effluents were monitored on aerosols and iodies only by continuous sampling and discontinuous measuring. The noble gases were monitored continuously.

The control room became over the years overburdened by all the instrumental backfittings and didn't respond to modern ergonomic requirements. There was also no independent control table for a reactor emergency shut down outside the control room.

4. Objects of modification

The programme of the RFR modification comprised the following objects.

i dismantling and intermediate storing of the old reactor tank and some of its components.

ii design, construction and mounting of a new reactor tank and of new components.

iii renewal of the cooling system, i.e. coolant pumps, heat exchangers, cooling towers, coolant pipes, armatures, valves and enlargement of the primary circuit's refining system.

iv improvement of the core emergency cooling system by using the water content of the secondary circuit.

v renewal of the process instrumentation and control engineering equipment for reactor operation.

vi construction of a new and ergonomically styled control room by enlargement of the reactor building and installation of a second, independent control table for reactor emergency shut down outside the control room.

vii replacement of the complete electric supply line by a new one and enlargement of the battery based independent power supply for devices most important to nuclear safety and radiation protection.

viii backfitting of the radiation protection monitoring by installation of a computer aided stack effluent monitor, by renewal of the room air monitoring equipment, by replacement of the gate contamination monitors and by installation of high range gamma dose rate monitors inside and outside the reactor hall.
ix renewal of the ventilation system by replacement of the old one, by construction of a filter house and installation of aerosol and iodine filters and by installation of an automated system switching for reduced ventilation and for filtering the effluents under accident conditions

x improvement of the irradiation capacity by modification of the tenth horizontal channel and some further vertical channels

xi modification of the complete operating, safety and emergency documents.

In the following the implementation of two of the modification objects, the replacement of the tank and the improvement of the emergency core cooling system, are described in detail. Information on the other objects are given in /15/.

4.1. Replacement of the tank

The reactor tank was to replace by a newly designed one, and its new components should fulfil the requirements for higher nuclear safety, for better inspection, for lower probability of LOCA and for better irradiation conditions. In detail this concerned the following changes in the design:

i Design of the new tank with the old outer dimensions but without an insert for the thermal column and with a reduction of welding seams in the vicinity of the core

ii Liquidation of the so-called central tank with the separator inside by a new designed separator with a very low leakage rate between the suction and pressure pipes of the primary cooling circuit

iii Shortening of the given thermal column and modification of the tenth horizontal channel for experiments

iv Protection of the head ends of the ten horizontal channels by protection plate segments against any heavy object falling down from the reactor top.

The new components were designed to enable in future dismantling and changing those with the highest neutron fluence if required. The separator, for example, is fixed to the bottom by a central screw and tightened against the tank by a metallic packing. This design guarantees both the replacement if required and keeping the core water covered in case of a leak in the tank or in the pressure
part of the primary cooling circuit, thus preventing a core meltdown.

The planning and technical preparation of the tank dismantling, of the storage basins for the tank and for its components, and of the transport to and the deposition in the basins were carried out with strict consideration of the radiation protection aspects. Therefore, calculations of the activation and the resulting dose rate fields and some experimental evaluations were performed as well as taking into consideration the measured values from the hungarian KFKI reactor modification/16/.

The collective dose from dismantling and storing should not exceed 50 % of the total value for the complete modification and the maximum individual dose should not exceed 20 % of the annual limit. The individual steps of this task were guided by radiation protection directives and supervised by experienced health physicists.

The dismantling work was started after an about 4 month's decay time and after complete decontamination of all surfaces accessible. The components core holder and beryllium reflector were dismantled first, decontaminated by remote techniques and unloaded from the tank by the reactor hall crane, transported to a spent fuel pond in the hall's floor and deposited in the water.

The tank itself with the remaining components central tank and separator was then fixed to a traverse at the reactor hall crane and lifted out of the cavity in the biological shield. Then in free-hanging position it was sprayed with a black coloured varnish to fix the outer contamination. After that it was transported by the crane over a shielded area in the reactor hall and layed down on the hall floor by some manoeuvres with the crane. Then, with a new fixing to the traverse the tank was again lifted up and put down on a truck especially prepared for this transport by a frame for the tank and a shielding between the tank and the driver cabine. The tank was transported by the truck to the nearby located open air basin for the intermediate storage. It was there lifted up, transported to and deposited in the basin by a truck crane. The basin was finally covered by concrete slabs.

All these handlings were trained many times in advance to make all things clear and safe, and resulting from that, the complete operation was carried out without any difficulty. The only not-foreseen incident was a contamination with C-14, limited in area, which resulted from the first graphit plate of the thermal column that was crumbled by the effects of long term irradiation.
The tank and its components will be stored for about 5 years, the programme of material investigations and of final deposition or recycling is to be elaborated after the completed modification.

The collective dose equivalent to the 66 workers employed in that work amounted to 67 mSv. The highest individual dose was smaller than 4 mSv. The real value of the collective dose was by 80% smaller than the calculated one.

The new tank and the new components were ordered in 1984 at the combinat Chemieanlagenbau Leipzig-Grimma (CLG), GDR. The design was drafted by the staff of the Rossendorf Central Institute for Nuclear Research (IfK) and was completed in cooperative work. The most important topic besides the already above mentioned ones is the dividing of the 3 circular plates in the tank in circular segments to reduce the efforts of future dismantling or mounting work. Treatments of the segments are possible at the reactor workshop and by it a contamination of the tank's bottom by splinters is avoided. Another reason for the segmentation was to get accessibility to the tank bottom for visual and technical inspection. The segments of the two lower circular plates, the steadying sieve and the protection plate, will be fixed together by only clamps and without screwing. This method was tested at the Dresden Technical University and its applicability was confirmed. The new tank and its components consist of the aluminium alloy AlMg3.

The tank manufacturer handed to the reactor staff in March 1986 samples of welding seams for neutron irradiation at high flux positions to get irradiated samples for technical tension and notch tests. The samples were irradiated with a fast neutron fluence of $2 \times 10^{20}$ n/cm$^2$ calculated from the known fast flux distribution at the position of irradiation and at 10 MW thermal power. This fluence at the position of the tank welding seams nearest to the core corresponds to an operation time of about 10 years at 60000 MWh/a.

The analysis of these samples don’t show any significant changes in material parameters, i.e. all parameters like tensile strength, tensile yield strength, breaking elongation and notch energy, remained within permissible limits. One of the prerequisites for the design, the manufacturing and the mounting of the new tank was the elaboration of a quality assurance programme by the reactor staff and the manufacturer and its confirmation by SAAS. One of its crucial points was the accessibility of the welding seams for future inspection and the inspection method. The accessibility had to be proofed at the design.
On February 1988 the tank was delivered to CINR and in the reactor hall outside the biological shield vertically fixed in a frame. Then the segments of the steadying sieve and the protection plate were mounted in the tank and after that the tank was put by the crane into the cavity in the biological shield. Then the separator was mounted and thereafter the base plate for the core.

The first test of tightness of the separator packing failed. After a finishing dealing of the flange and the packing the test resulted positive, the leakage rate for a pressure difference of about 20 kPa was 0.94 l/min.

Thereafter the dismantling and mounting of the steadying sieve, located below the head ends of the horizontal channels, was tested performing remote technology. The tests proofed successfully.

4.2. Improvement of the emergency core cooling

New systems were installed to make the cooling in the case of LOCA safer than before. The sprinkling device consists of 6 showers installed over the core at the level of the upper mounting plate and is connected to the 3 individual water supplying systems. The first one consists of three tanks with a total of 18 m³ volume, installed high over the reactor top so that the water flows to the showers by the hydrostatical pressure. The valves are threefold redundant, two of them are motor driven, the third must be opened manually. The second system i.e. the new one makes use of the 100 m³ desalinated water of the secondary cooling circuit. This water can be pumped to the tanks after exhausting of the first system. Both systems guarantee an uninterrupted emergency cooling of at least 12 hours with a flow rate of 8-10 m³/h. The third system is based on the drinking water supply for the total institute, the water can be injected in the tanks of the first system after exhausting of the systems 1 and 2 or after failing of system 2.

These measures guarantee a higher degree of safety for the new reactor design in the event of LOCA.

The capacity of the complete emergency cooling system is sufficient for removing the residual heat from the fuel after LOCA and to prevent core melting. Therefore, LOCA’s of the RFR can be classified as hypothetical accidents because a simultaneous failure of the 3 cooling systems has to be supposed additionally to the probability of LOCA. That results finally in the calculated probability for the RFR’s LOCA of lower than 10⁻⁶/a.
5. Sequence of second step modification measures

In the nineties the modification will be continued by the following measures

i) Enlargement of the given capacity for intermediate storage of spent fuel by construction of a facility for long term dry storage. It has to be implemented up to 1993, for the given capacity of the spent fuel ponds in the reactor hall will be exhausted in 1994 due to the planned operation with 60 000 MWh/a.

ii) Modernization of the core's protection and safety system (SUS) up to 1994 in connection with the step-wise introduction of computerized methods in control and regulation of reactor operation, in registration of process parameters, in storing operational data and in performing accident analyses.

iii) Installation of an automated system for monitoring radioactive releases from the RFR facility for normal and accidental conditions as well as airborne radioactivity from accidents at other nuclear facilities. Computer aided processing of radiological and meteorological data for assessment of the exposure to the environment and for initiating the required measures according to the plant programme of emergency preparedness.

6. The licencing procedure

The modification of the RFR was object of a comprehensive licencing procedure between the SAAS and the RFR manager with the two main aims

i) to prove, that current regulations have been regarded and that after finishing the modification nuclear safety and radiation protection of the RFR facility have been improved and are in conformity with the current requirements

ii) to prove, that all the work of modification has been prepared and performed taking into account the ALARA principle.

The fundamental regulations that have to be regarded were /3-9/.

The responsibility of elaborating all the documents for licencing and for preparing the individual modification tasks under special regard of radiation protection and nuclear safety rests with the RFR manager. He and his
staff have been advised and controlled by the radiation protection and nuclear safety officers of the CINR. The officers had to give comments to the documents before their handing-over to the SAAS. Only with positive comments the document could get an approval by the SAAS, but in most cases the approval was given not before the fulfilment of additional requirements.

All the requirements, both of the SAAS and of the CINR officers for nuclear safety and radiation protection were computer aided recorded and controlled in fulfilment.

For all the modification tasks with exposure to radiation the following preparations were required

i assessments of the individual and collective exposure by estimation of the dose rate at the working places by measurement or calculation, of the handlings and movements of the workers and of the duration of the work

ii working instructions for all the steps in a given work concerning the quick and safe performance and

iii check lists for the proper implementation of a given work

The licencing procedure itself is divided into the part of the approval to design, fabrication and construction and in the part of the approval to the commissioning. In course of the first part for example the following documents had to be elaborated /4/:

i a principle outline of the total modification and the individual changes and improvements provided for on the base of an intercomparison of the given state of the facility and the requirements in current regulations

ii an accident analysis (determination of the basic design accidents)

iii a programme for providing nuclear safety at all equipment relevant to safety

iv a programme for providing radiation protection supervision and monitoring at the facility

v projects and implementing programmes for all the individual parts of the modification

vi quality assurance programmes for manufacturing and construction

vii planned methods for dismantling, mounting, transporting as well as handling of radioactive waste
measures of nuclear material safeguarding and physical protecting during the course of modification

ix safety report of the facility.

The particularity of the licencing procedure for the RFR was given in the fact, that the SAAS didn't require the simultaneous submission of the projects and documents of all parts of modification for approval. Outgoing from the approval to the principle outline of the total modification the practical implementation of components could be started stepwise after receiving the approval to the special documents. This procedure of licencing allowed a flexible implementation.

During the modification work the second part of documents had to be elaborated as a preparation for the approval to the commissioning. It comprised for example /4/:

i the conditions and limits for safe operation of the facility

ii instructions for acceptance tests and test operations as well as forms to record the test results

iii programmes for recurrent tests and inspections to components relevant to safety

iv programmes for the critical experiment, the commissioning and the test operation at nominal power

v a programme for emergency planning and preparedness

vi working instructions for maintenance and service to components relevant to nuclear safety

vii working instructions for maintenance and service to the facility in general

viii a programme for the training of the operational staff.

A special attention was devoted to the preparation and implementation of functional and acceptance tests for the components of the reactor facility.

About 500 various documents and comments had been elaborated and more than 150 tests had been carried out and recorded so far.
7. State after two and a half years of modification

All the objects of the first modification step but the heat exchangers of the coolant system were finished at the beginning of 1989. These objects were put to taking-over tests by the contractors and the reactor staff in the first half of 1989 and will be put as a system to the official and final acceptance test for SAAS confirmation and licencing in November 1989.

The critical experiment and the measurements of the physical parameters of the modified reactor facility are planned at December 1989. The commissioning for the reactor test operation at 10 MW thermal power shall be performed in January 1990. This reactor test operation will take a time of about one year.

In carrying out the modification we gained much experience that could partly feedbacked into the modification itself, for example

i planning and managing of modification objects

ii design, manufacturing and assembling of reactor components

iii methods for dismantling of activated and contaminated components

iv special tools for dismantling, mounting and assembling

v elimination of vibrations from components and pipes

vi instructions for elaborating all the documents for approval and licencing.

All the operational documents, the safety report and the emergency plan have been revised taking into account the current national ordinances, standards and directives as well as the recommendations and guides of the IAEA and the CMEA.

The aim of the modification, the increase of nuclear safety of the facility and of the efficiency in future irradiations and experiments could be achieved. However, we cannot quantify the results before recommissioning the reactor and having measured data from full power operation for interpretation.
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MODIFICACION DEL REACTOR IAN-R1

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ABSTRACT

The IAN-R1 reactor is the only nuclear reactor operating in Colombia; it is installed at the Institute of Nuclear Affairs (IAN) in Bogotá, which is an official body coming under the Ministry of Mining and Energy. This reactor started operation in January 1965 with a rated power of 10 kW and was modified a year later to operate at 20 kW, which has been its rated power up to the present. Given its importance for the application of nuclear technology in Colombia for various purposes, principally in the areas of neutron activation analysis, determination of uranium content in minerals using the delayed neutron counting method, production of certain radioisotopes such as $^{198}$Au and $^{82}$Br for engineering applications, and production of radioactive material for teaching and research purposes, research has been in progress for some years into ways of increasing its power. The study on experimental requirements and on the demand for locally produced radioisotopes came to the conclusion that its power should be increased to 1000 kW, which would allow the facility to remain on the same site. The modification includes conversion of the core to low-enriched fuel, operation up to 1 MW, modification of the shielding, renovation of instrumentation and installation of a radioisotope processing plant. When the reactor is modified we will be able to produce other radioisotopes for applications in nuclear medicine, industry and engineering; at the same time, the safety of the facility will be optimized and the experimental facilities improved.
1. RESUMEN

El Reactor IAN-R1 es el único reactor nuclear en operación en Colombia y está instalado en Bogotá en el Instituto de Asuntos Nucleares, entidad oficial dependiente del Ministerio de Minas y Energía. Este reactor inició su operación en enero de 1965 con una potencia nominal de 10 Kw y un año más tarde fue modificado para operarlo a 20 Kw, potencia de trabajo nominal hasta la fecha.

Dada su importancia en la aplicación de tecnología nuclear para distintos propósitos en Colombia, principalmente en las áreas de análisis por activación neutrónica, determinaciones de contenidos de uranio en minerales por el método de contaje de neutrones retardados, producción de algunos radioisótopos como el Au-198 y Br-82 para aplicaciones en trabajos de ingeniería y producción de material radioactivo para docencia e investigación; desde hace algunos años se iniciaron los estudios para el aumento de su potencia.

El estudio de necesidades experimentales y de producción local de radioisotopos llevaron a la conclusión de aumentar la potencia hasta 1000 Kw, pudiéndose mantener la instalación en el mismo sitio.

La modificación comprende el cambio del núcleo a combustible de bajo enriquecimiento, la operatividad hasta 1 Mw, la modificación del blindaje, la renovación de la instrumentación y la instalación de una planta de procesamiento de radioisotopos.

Con el reactor modificado, se podrán producir otros radioisotopos para aplicaciones en medicina nuclear, industria e ingeniería y, a la vez, se optimizará la seguridad de la instalación y se mejorarán las facilidades experimentales.
2. LA FACILIDAD ACTUAL

El Reactor IAN-R1, en operación en el Instituto de Asuntos Nucleares, entidad ésta oficial dependiente del Ministerio de Minas y Energía en Bogotá, Colombia, es el único reactor existente en el país hasta la fecha. Inició su operación en 1965 y opera en la actualidad a 20 kilovatios. Es un reactor tipo piscina, con uranio enriquecido al 90% en el isótopo 235, moderado y refrigerado por agua liviana, reflejado por grafito y con tres barras para control y seguridad, así: dos en acero inoxidable al boro, unidas mediante embragues electromagnéticos a mecanismos que les permiten su desplazamiento vertical dentro del núcleo y una tercera, reguladora, construida en acero inoxidable y unida a un servomecanismo de movimiento vertical, para hacer ajustes finos de la potencia.

El núcleo, con los elementos reflectores, las tres barras de control y seguridad (figura 1), tres cámaras de ionización y una cámara de fisión se encuentran inmersas en un tanque cilíndrico de acero al carbón que contiene el agua liviana. Alrededor del tanque, una masa de concreto monolítico se levanta sobre una base octogonada hacia arriba en forma escalonada para terminar en el tercer nivel como una conformación cilíndrica que sirve de blindaje biológico a todo el sistema (figura 2).

La refrigeración del reactor es por convección natural. El agua cumple funciones de refrigerante, moderador y blindaje biológico; es purificada mediante un pequeño sistema de procesamiento, el cual está compuesto por una bomba, un filtro, un intercambiador iónico y los correspondientes indicadores de temperatura, flujo de agua y conductividad. El sistema anterior tiene también un intercambiador de calor adecuado para 10 kilovatios. El agua circula en circuito cerrado entre la piscina y este sistema de procesamiento.

La instrumentación nuclear para el control del reactor consiste de tres canales de operación y uno de seguridad.

El canal de arranque lo componen: una cámara de fisión como detector, un preamplificador, un amplificador de pulsos, un escalímetro, un medidor logarítmico de rata de contaje y un graficador con rango de $10^3$ c.p.s.

El canal de período y log-n está compuesto por: una cámara de ionización compensada como detector, un amplificador de período y un medidor de potencia logarítmica con un graficador de nueve décadas que va desde un milivatio hasta cien kilovatios.

El canal lineal de potencia está compuesto por: una cámara de ionización compensada, un micro-microamperímetro lineal y un graficador que va desde $10^{-4}$ hasta $10^3$ vatios.

El canal de seguridad, está compuesto por una cámara de ionización circular de placas paralelas conectada a un amplificador de seguridad.
Complementando esta instrumentación en la consola de control, desde donde se opera el reactor, se dispone de indicadores analógicos y digitales de información de posición de las barras, tres medidores de nivel de radiación correspondientes a zonas críticas en la boca del tanque, en el sistema de procesamiento y en la sala de control y el sistema de caída de barras por detección de anormalidades en el canal lineal, en el logarítmico y de periodo, en las fuentes de alimentación de los detectores, por alto nivel de radiación en las zonas de control, por alto nivel de potencia, por movimientos sísmicos además de la opción manual de emergencia. La tabla 1 muestra las características técnicas principales del reactor actual.

3. OBRAS CIVILES REQUERIDAS PARA LAS MODIFICACIONES DEL REACTOR IAN-R1

Las obras civiles requeridas para la modificación del Reactor Nuclear IAN-R1, son las siguientes:

1. Extensión de la estructura de la piscina con el propósito de instalar el nuevo Reactor MAPLE de 1 megavatio.

2. Ampliación de los blindajes de concreto alrededor de la instalación y fortalecimiento de las fundaciones para soportar la carga adicional.

3. Construcción de un edificio adicional para la instalación de una planta de producción de radioisótopos y del actual reactor, que sería relocalizado en una nueva piscina de mayor tamaño con el propósito de disponer finalmente de dos reactores. La nueva piscina tendrá un tanque de acero inoxidable instalado y que sobresalga 1 metro sobre la superficie.

4. Adaptación de algunas oficinas y laboratorios cercanos para la instalación de equipo nuevo complementario para el suministro electrónico.

5. Instalación de nuevas estructuras metálicas que permitan el acceso a los distintos sitios de la instalación.

La figura 3 muestra las actuales y futuras instalaciones cercanas al Reactor; el actual Reactor IAN-R1 será relocalizado en un sitio cercano al Reactor MAPLE y a la Planta de Producción de Radioisótopos. Las ventanas actuales en la instalación del Reactor serán removidas para establecer un recinto cerrado con las correspondientes condiciones de ventilación y aire acondicionado, nuevo sistema de protección contra incendio y nuevo sistema de recolección de desechos radiactivos. A la nueva instalación del Reactor IAN-R1, se trasladarán todos sus componentes y en la piscina existente se instalarán todos los correspondientes al nuevo Reactor MAPLE.
4. UTILIZACION DE LA FACILIDAD MODIFICADA
DENTRO DEL PROGRAMA NUCLEAR COLOMBIANO

Teniendo en cuenta que el actual reactor es el centro de las actividades nucleares del Instituto, el aumento de su potencia permitirá extender favorablemente sus servicios a las diferentes áreas tecnológicas que lo utilizan. Así, los análisis por activación neutónica se extenderán a otros elementos que en la actualidad presentan dificultad de activación por el bajo flujo neutróncico de la instalación. Los programas de investigación con las universidades se verán favorecidos al poder utilizar los dos tubos de haces neutrónicos en investigación de materiales, efecto de radiaciones sobre los mismos y desarrollos de utilización industrial. Los programas en física de reactores e ingeniería nuclear tendrán oportunidad de evaluar los diferentes parámetros con un instrumento de mayor potencia donde los mismos pueden ser fácilmente estudiados. Sin embargo, uno de los principales propósitos de la modificación es la producción de radioisótopos para la medicina nuclear, la ingeniería colombiana, la industria y la enseñanza.

Los procedimientos radioisotópicos en medicina fueron introducidos en Colombia a principios de la década de 1950; el primer laboratorio organizado, dedicado a la práctica de la Medicina Nuclear, se constituyó en el Instituto Nacional de Cancerología y comenzó a funcionar en julio de 1955. Ya en la década de 1960 existían en el país tres centros de radioisótopos y hacia 1974 el número había aumentado a siete. Con el concurso de varios factores, entre ellos el desarrollo del Programa Nacional de Control de Cáncer, la facilidad de obtención de material radiactivo a través del Instituto de Asuntos Nucleares y el aumento de la docencia y divulgación de los métodos radioisotópicos entre el cuerpo médico, este número asciende hoy a 26, de los cuales ocho se encuentran en la capital de la República. La Sociedad Colombiana de Medicina Nuclear se funda en 1969 y en la fecha cuenta con unos sesenta miembros en sus distintas categorías y es una de las más importantes de América Latina. La especialidad lleva a cabo unos 50.000 procedimientos "in vivo" al año en el país, lo que demuestra ampliamente el reconocimiento de su utilidad en la práctica de la medicina actual.

El Instituto de Asuntos Nucleares importa el material radiactivo de varios productores internacionales y lo distribuye a unos 150 usuarios en Colombia; la mayoría de ellos utiliza sistemas "in vitro", para determinaciones hormonales y de factores inmunológicos, especialmente. Los principales isótopos importados son el I-131 (cerca de 36 curios por año) y generadores de Mo-Tc99m (unos 550 por año), que junto con el resto de materiales suman un total de cerca de 50.000 dólares mensuales en la actualidad, después de un
aumento de cerca del 80% en los últimos meses. Por otra parte, el IAN produce y distribuye unos 1800 estuches (kits de 5 viales multidosis, cada uno) para marcación con Tc-99m, a costos mucho más bajos que los importados, manteniendo su alta calidad, lo cual beneficia notoriamente la práctica de muchos procedimientos (gammagrafía hepática con coloide y fitato, biliar con derivados del IDA, renales con DMSA, glucoheptonato y DTPA, óseas con MDP, compartimentos vasculares y miocardio con pirofosfatos y otros diversos estudios). Aquí es conveniente anotar que muchos de los centros de Medicina Nuclear son oficiales, es decir, pertenecen a hospitales universitarios del Gobierno, y por lo tanto atienden un significativo grupo de pacientes del sector socioeconómico más débil; esto implica que la Medicina Nuclear cumple una misión social de importancia en la Nación.

En la actualidad, los centros de Medicina Nuclear cuentan con especialistas colombianos idóneos y con equipos modernos, incluyendo varios con SPECT que, junto con una adecuada provisión de material radiactivo, les permite llevar a cabo todos los procedimientos que se requieran en la práctica médica diaria y en la necesaria investigación clínica, de especial importancia en la docencia y en el desarrollo tecnológico y radiofarmacológico de la especialidad. Hay una franca tendencia al aumento de los centros de Medicina Nuclear; sin embargo, el constante incremento en el costo de los radioisótopos, debido en especial a la devaluación monetaria, hacen que los procedimientos ocasionen erogaciones presupuestales impotantes y que muchas veces no queden al alcance del grueso público. Como ejemplo podemos mencionar que un generador de Mo-Tc con actividad inicial de 600 mCi en Tc-99m cuesta en promedio US$525 y que el examen más costoso practicado con este isótopo, excluyendo los estudios cardiológicos, tiene un valor promedio de US$60. El salario mínimo mensual, que devenga cerca del 75% de la población trabajadora del país, es apenas de unos US$100.

Con la adquisición de un nuevo reactor nuclear, capaz de sostener una adecuada producción de isótopos, especialmente I-131 y Mo-99, y con una planta de producción y montaje de generadores de Mo-Tc, se calcula que el ahorro en estos costos de material radiactivo sería del orden del 30%. Esto permitiría incrementar el número de generadores utilizados por los centros de Medicina Nuclear, abaratando así los costos de los estudios diagnósticos para los pacientes y aumentando la población que se beneficiaría con dichos exámenes. Por otra parte, con los precios competitivos podría iniciarse una activa exportación de generadores y radionúclidos a los países vecinos que tienen una activa práctica de Medicina Nuclear y cuyos costos son igualmente onerosos que los actuales nuestros.

La producción inicial esperada de Mo-99 será cerca de 1.000 GBq semanalmente a partir de la irradiación de blancos de molibdeno
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natural, con el propósito de atender principalmente los requerimientos de Bogotá y zonas cercanas; se contempla la eventual producción
de 3000 BGq semanalmente en el futuro. También, la facilidad producirá 100 GBq de 1-131 semanalmente a partir de blancos de óxido
de teluro y se contempla también la producción futura de 0,3 BGq
de 1-125 semanalmente a partir de blancos de xenón enriquecido.
También se espera poder producir F-18, Cr-51, P-32, Fe-59, Hg-197
y Au-19S para uso médico.
La producción de radioisótopos para aplicaciones en agricultura,
industria e ingeniería, permitirá comprometer al Instituto en actividades de mayor alcance, limitadas en la actualidad por las bajas
actividades específicas logradas.
Además de todo lo anterior, la nueva facilidad estará de acuerdo
con la moderna filosofía de seguridad aplicable a los reactores
nucleares de investigación, a la producción de material radiactivo
y al manejo técnico de los deshechos.
La Planta de Producción de Radioisótopos, dispondrá de facilidades
para Mo99, Tc99m, 1-131, Tl-201, P-32, Ir-192 y Xe-133.
Tres lugares en el núcleo y diversas opciones en el reflector de
agua pesada permitirá la irradiación de materiales. Los dos tubos
de haces estarán habilitados para la investigación y la radiografía.
Las figuras 4 y 5 muestran los detalles técnicos del nuevo núcleo
y del nuevo reactor.


LIDSTONE, R.F. The MAPLE Upgrade of the IAN-R1 Research Facility. August 1, 1939, Bogotá.

TABLA I  
CARACTERISTICAS TECNICAS DEL REACTOR IAN-R1  

Tipo: piscina  

Moderador: Agua liviana  

Refrigerante: Agua liviana (refrigeración por convección natural).  

Potencia de operación: 20 Kw térmicos.  

Elementos reflectores: 20 elementos de grafito recubiertos con una resina poliestérica.  

Barras de control: 3 barras de control (2 de seguridad construidas en acero al boro, y 1 de control construida en acero inoxidable).  

Facilidades para irradiación de muestras:  
- Dos tubos de haces neutrónicos con un flujo de $4.8 \times 10^{10}$ n.m. s. w. térmico y $1.1 \times 10^{10}$ n.m. s. w. rápido.  
- Seis penetraciones en los elementos reflectores con un flujo neutrónico máximo de $3.5 \times 10^5$ n.m. s.  
- Un sistema neumático con un flujo neutrónico máximo de $2.5 \times 10^5$ n.m. s.  

Combustible: Uranio metálico enriquecido al 90% en el isótopo U-235 aleado y recubierto con aluminio tipo 1100.  

Carga nominal normal:  
- 2.175 Kg., distribuidos en 16 elementos formando un arreglo de cuatro por cuatro.  
- 11 estándar, con 10 placas cada uno.  
- 3 de control, con 6 placas cada uno.  
- 1 elemento con 9 placas y una simulada.  
- 1 elemento con 8 placas con combustible y dos simuladas.  

Instrumentación: Una consola de control con tres canales de operación y uno de seguridad.  

Canales de operación:  

1. Canal de arranque, constituido por una cámara de fisión como detector, un preamplificador, un amplificador de pulsos, un escálmetro, un medidor de tasa de contaje y un graficador con rango de $1$ a $10 \times 10^3$ c.p.s.
2. Canal de período y log-n, constituido por una cámara de ionización compensada como detector, un amplificador de período, un medidor de potencia logarítmica con graficador de nueve décadas, de 1 mw a 100 kw., y un graficador para el período.

3. Canal lineal de potencia, constituido por una cámara de ionización compensada, un micro-microamperímetro lineal y un graficador con escalas de $10 \times 10^{-4}$ a $10 \times 10^{3}$ w.

4. Canal de seguridad, constituido por una cámara de ionización circular de placas paralelas conectadas a un amplificador de seguridad.
Fig. 1. Núcleo del Reactor IAN-R1
Soporte de mecanismos y detectores

Guías para desplazamiento de barras

Tubo del sistema neumático

Blindaje biológico

Cámara de ionización

Tubos de haces

Recinto de la fuente

Soporte del núcleo

Núcleo

Fig. 2. Sección vertical del Reactor IAN-R1
Fig. 3. Proyecto de ampliación de las instalaciones del Reactor IAN-R1 - Primera planta
Fig. 4 Corte Horizontal Nucleo MAPLE Colombia
**Fig. 5 Reactor MAPLE**
ИССЛЕДОВАТЕЛЬСКИЙ РЕАКТОР ВВР–СМ–20

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THE WWR-SM-20 RESEARCH REACTOR

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ABSTRACT

In this paper the design features and experimental capabilities of the WWR-SM-20 20 MW research reactor are described. The reactor uses fuel assemblies consisting of six coaxial fuel tubes with a square cross-section. IRT-3M fuel assemblies can be used with both 90% enriched and 36% enriched uranium. The main characteristics of the IRT-3M fuel assemblies are given, as are the technical and physical parameters of the WWR-SM-20 reactor. The core can hold up to ten ampoule-type channels with a diameter of up to 68 mm. For irradiation purposes, up to 22 26-mm-diameter channels in the fuel assemblies, and up to 48 42-mm-diameter channels in the beryllium blocks of the reflector can be used. In the graphite blanket between the horizontal channels, channels with a diameter of up to 130 mm can be used. The thermal neutron flux density has a maximum value of $1.5 \times 10^{18} \text{ m}^{-2} \cdot \text{s}^{-1}$ in the core and $2.3 \times 10^{18} \text{ m}^{-2} \cdot \text{s}^{-1}$ in the reflector, and the fast neutron flux density ($cE > 0.821 \text{ MeV}$) a maximum of $1.9 \times 10^{18} \text{ m}^{-2} \cdot \text{s}^{-1}$. A number of design features have been incorporated in the WWR-SM-20 reactor to make it effectively safe.
ИССЛЕДОВАТЕЛЬСКИЙ РЕАКТОР ВВР-СМ-20

АННОТАЦИЯ

Описываются особенности конструкции и экспериментальные возможности исследовательского реактора ВВР-СМ-20 мощностью 20 МВт. В реакторе используются тепловыделяющие сборки (ТВС), состоящие из шести коаксиальных трубчатых тепловыделяющих элементов (твэлов) квадратного сечения. Могут использоваться ТВС ИРТ-3М как с ураном 90%-ного, так и 36%-ного обогащения. Приводятся основные характеристики ТВС ИРТ-3М, технические и физические параметры реактора ВВР-СМ-20. В активной зоне могут размещаться до 10 каналов ампульного типа диаметром до 68 мм. Для облучений могут использоваться до 22 каналов диаметром 26 мм в ТВС, до 48 каналов диаметром 42 мм в бериллиевых блоках отражателя. В графитовой кладке между горизонтальными каналами могут использоваться каналы диаметром до 130 мм. Плотность потока тепловых нейтронов в активной зоне - до $1,5 \times 10^{18} \text{ м}^{-2}.\text{с}^{-1}$, в отражателе – до $2,3 \times 10^{18} \text{ м}^{-2}.\text{с}^{-1}$, плотность потока быстрых нейтронов (с $E > 0,621 \text{ МэВ}$) – до $1,9 \times 10^{18} \text{ м}^{-2}.\text{с}^{-1}$.

В реакторе ВВР-СМ-20 реализован ряд конструктивных решений, делающих его практически безопасным.

I. ВВЕДЕНИЕ

Проект реактора ВВР-СМ-20 разработан с учетом возможности использования здания и других сооружений исследовательского реактора ВВР-С Института ядерной физики АН Узбекской ССР в Ташкенте. Реактор ВВР-С мощностью 2 МВт был введен в эксплуатацию в 1960 г. В активной зоне реактора использовались кассеты с тепловыделяющими элементами (твэлами) стержневого типа ( ЭК-10). С целью расширения экспериментальных возможностей реактора ВВР-С в 1971 г. в ИАЭ им. И.В.Курчатова был разработан проект его реконструкции, основанный на использовании в активной зоне вместо кассет...
с твэлами ЭК-10 тепловыделяющих сборок (ТВС) с трубчатыми твэлами типа ИРТ-2М [1]. Проведенная в конце 1971 г. по этому проекту реконструкция реактора ВВР-С позволила повысить его мощность с 2 до 10 МВт и обеспечить возможность размещения в активной зоне петлевого канала диаметром 130 мм и 4-х каналов диаметром до 67 мм для испытания материалов, а в бериллиевом отражателе — до 22 каналов диаметром 45 мм для производства изотопов и активационного анализа. Плотность потока тепловых нейтронов в активной зоне и отражателе была повышена до \( \sim 10^{18} \text{ м}^{-2} \cdot \text{с}^{-1} \). В центральной ловушке нейтронов плотность потока тепловых нейтронов составила \( \sim 3 \cdot 10^{18} \text{ м}^{-2} \cdot \text{с}^{-1} \).

Разработка проекта нового реактора ВВР-СМ-20 была вызвана необходимостью:
- замены бака реактора и внутрибаковых узлов, проработавших более 29 лет;
- увеличения количества ампульных и петлевых устройств, размещаемых в активной зоне, и каналов в отражателе для облучения изотопных мишеней и повышения в них плотности потока нейтронов с целью обеспечения проведения исследований на более высоком уровне;
- повышения безопасности эксплуатации реактора в связи с тем, что в районе его расположения при землетрясениях интенсивность воздействия может достигать 8 баллов по шкале Меркалли.

Проект реактора ВВР-СМ-20 разработан на основе использования в активной зоне тепловыделяющих сборок типа ИРТ-3М, обладающих более развитой поверхностью теплоотдачи, чем ТВС типа ИРТ-2М. Могут использоваться ТВС ИРТ-3М как с ураном 90%-ного обогащения, так и с ураном 36%-ного обогащения.

2. КОНСТРУКЦИЯ РЕАКТОРА

Продольный и поперечный разрезы реактора ВВР-СМ-20 показаны на рис. 1 и 2. Бак реактора ВВР-СМ-20 представляет собой сварной сосуд цилиндрической формы из алюминиевого сплава внутренним диаметром 2268 мм и высотой 5900 мм.
Рис. 1. Продольный разрез реактора Р-3-01-60

1 - бак реактора, 2 - канал вертикальный экспериментальный, 3 - короб распределительный, 4 - канал горизонтальный, 5 - теплопроводящая сборка (ТС), 6 - корпус реактора, 7 - трубопровод вспомогательный, 8 - трубопровод вспомогательный, 9 - верхняя опорная решетка, 10 - решетка опорная, 11 - кан. 12 - корпус реактора, 13 - канал, 14 - трубопровод конденсатный, 15 - канал с рабочей основной (стальной), 16 - верхняя технологическая защита, 17 - кронштейны для крепления каналов ОП.
Рис. 2. Поперечный разрез реактора ВВР-СМ-20
I - бак реактора, 2 - корпус реактора, 3 - бериллиевые блоки, 4 - канал горизонтальный, 5 - короб распределительный напорного трубопровода, 6 - щит свинцовый, 7 - каналы вертикальные экспериментальные, 8 - блок графитовый, 9 - каналы блоков детектирования СУЗ, 10 - ТВС шеститрубная, 11 - ТВС шеститрубная с рабочим органом СУЗ, 12 - каналы вертикальные экспериментальные.
Он установлен в существующей шахте в биологической защите реактора ВВР-С на закладную опорную плиту с зазором 30 мм по отношению к тепловому экрану из чугунных колец и бетону, т.е. также как был установлен и бак реактора ВВР-С. Надежность такого конструктивного решения подтверждена эксплуатацией в течение 25–29 лет 5 реакторов этого типа в Советском Союзе и 6 реакторов за рубежом.

В днище бака вварены только 2 патрубка: для подсоединения отводящего и подводящего (напорного) трубопроводов диаметром 400 мм первого контура системы охлаждения. Наклонный патрубок для подсоединения трубы, предназначенной для перегрузки отработавших ТВС из бака реактора в бассейн-хранилище, вварен в верхней части бака. Эта же труба используется и в качестве переливной в случае повышения уровня воды в баке выше нормального. Сверху бак реактора имеет разборную съемную крышу, на которой крепятся кронштейны каналов рабочих органов (стержней) системы управления и защиты (СУЗ), стойки для роликов тросовой системы от стержней СУЗ к приводам, кронштейны для вертикальных экспериментальных каналов и каналы блоков детектирования СУЗ. Центральная часть крышки открыта для обеспечения доступа к активной зоне при проведении перегрузочных работ.

Для вывода пучков нейтронов от реактора в физический зал в пределах бака, между корпусом реактора и стенкой бака, используются полые алюминиевые вытеснители. Такое конструктивное решение исключает необходимость иметь отверстия в стенке бака реактора и, таким образом, предотвращает возможность возникновения аварийных ситуаций при разрушении горизонтального канала в пределах бака реактора.

Корпус реактора приварен к днищу бака. Он предназначен для размещения ТВС и бериллиевых блоков отражателя. Корпус реактора, так же как и бак, — из алюминиевого сплава. Внутренний диаметр корпуса — 725 мм. В корпусе реактора имеется опорная решетка из алюминиевого сплава толщиной 100 мм, на которую устанавливаются ТВС, бериллиевые блоки,
вытеснители для экспериментальных каналов. Опорная решетка в случае необходимости может быть заменена новой.

Под опорной решеткой в корпусе реактора размещена вставка, состоящая из цилиндрической обечайки, двух решеток и расположенных между ними труб внутренним диаметром 26 мм. Десять из них являются продолжением каналов, в которых размещены рабочие органы (стержни) СУЗ. В остальных между решетками размещены поглотители. Верхняя решетка вставки повторяет конфигурацию опорной решетки. Вставка исключает возможность перемещения ТВС в случае гипотетического разрушения опорной решетки. Наличие поглотителей под опорной решеткой исключает образование критической массы в случае оплавления ТВС и попадания топлива во вставку.

Корпус реактора соединен с камерой, через которую вода поступает к отводящему трубопроводу первого контура после выхода из активной зоны. Камера разделена на две половины перегородкой, верхний край которой расположен на одном уровне с верхним торцем корпуса реактора. В верхней части камеры имеются отверстия для срыва сифона. Благодаря этому, в случае опорожнения бака реактора при разрыве отводящего или подводящего трубопроводов первого контура часть корпуса реактора, в которой расположена активная зона, остается заполненной водой. На уровне верхнего торца активной зоны в камере предусмотрены три клапана, обеспечивающие возможность возникновения естественной циркуляции воды через активную зону и камере после прекращения принудительной.

В случае разрыва отводящего или подводящего трубопровода первого контура уровень воды в баке реактора снижается только до верхнего торца корпуса реактора, поскольку отверстия для выхода воды из распределительных коробов, соединенных с подводящим трубопроводом через специальную напорную камеру, расположены выше торца корпуса.

Для восполнения потерь воды из корпуса реактора за счет испарения после разогрева ее до кипения, если разрыв
трубопроводов произойдет при работе реактора на мощности, предусмотрена система подпитки, включающая баки общей емкостью 40 м³ и два независимых трубопровода подачи воды в верхнюю часть корпуса реактора.

Проект реактора ВВР-СМ-20 предусматривает использование в его активной зоне ТВС типа ИРТ-ЗМ [2]. Каждая ТВС состоит из 6 коаксиальных твэлов квадратного сечения (рис. 3). Толщина стенки твэла – 1,4 мм. Оболочки твэла – из алюминиевого сплава, сердечник – из диоксида урана в алюминиевой матрице. ТВС размещаются на опорной решетке реактора с шагом 71,5 мм. Во внутреннюю полость ТВС устанавливается канал с рабочим органом СУЗ или канал для облучения блок-контейнеров. В реакторе могут использоваться как ТВС с топливом 90%-ного, так и ТВС с топливом 36%-ного обогащения. Основные параметры 6-трубных ТВС типа ИРТ-ЗМ даны в таблице I.

<table>
<thead>
<tr>
<th>Наименование параметра</th>
<th>Значение параметра</th>
</tr>
</thead>
<tbody>
<tr>
<td>Обогащение урана, %</td>
<td></td>
</tr>
<tr>
<td>90</td>
<td>0,264</td>
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<tr>
<td>36</td>
<td>0,309</td>
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<tr>
<td>Толщина твэлов, мм</td>
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<tr>
<td>Ширина зазора между твэлами, мм</td>
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<tr>
<td>Длина сердечника, м</td>
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<td>Объемная доля воды (в центре ТВС канал с вытеснителем)</td>
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<td>Удельная поверхность теплоотдачи, м²/л</td>
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<tr>
<td>Поверхность теплоотдачи ТВС, м²</td>
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<tr>
<td>Содержание урана-235 в ТВС, кг</td>
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</tr>
<tr>
<td>Удельная загрузка урана-235, кг/м³</td>
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</tr>
<tr>
<td>Отношение ядерных концентраций водорода и урана-235</td>
<td>160</td>
</tr>
</tbody>
</table>
Рис. 3. Поперечные сечения ТВС типа ИРТ-ЭМ:
а) шеститрубная ТВС со стержнем СУЗ,
б) шеститрубная ТВС с экспериментальным каналом для облучения блок-контейнеров,
1 - твэлы, 2 - канал, 3 - стержень СУЗ, 4 - блок-контейнер.
Максимальное количество ТВС в рабочих загрузках активной зоны - 32. В остальных 10 ячейках активной зоны (из 42, в которых могут устанавливаться ТВС) размещаются алюминиевые вытеснители для вертикальных экспериментальных каналов (рис. 2). При отсутствии каналов в вытеснители устанавливаются алюминиевые или бериллиевые пробки. Вытеснители исключают повреждение оболочек твэлов при загрузке экспериментальных каналов в реактор или их выгрузке.

Пространство между корпусом реактора и ТВС заполнено 30 сменными бериллиевыми блоками, образующими боковой отражатель. Из них 18 блоков квадратного сечения с наружным размером (под ключ) - 69 мм и 12 блоков с поперечным сечением, обеспечивающим заполнение пространства между корпусом реактора и блоками квадратного сечения.

Снаружи корпус реактора на две трети своего периметра окружен бериллиевыми блоками. Толщина блоков в радиальном направлении - 70 мм. Конфигурация блоков такова, что они образуют отверстия для размещения торцев горизонтальных каналов, доходящих до корпуса реактора.

Пространство между двумя взаимно перпендикулярными горизонтальными каналами заполнено кладкой из сменных графитовых (в алюминиевых оболочках) блоков. Междубериллиевыми блоками снаружи корпуса реактора и графитовыми блоками установлен свинцовый экран (щит). Между стенкой бака и графитовыми блоками установлены алюминиевые блоки с отверстиями, в которых размещаются каналы с блоками детектирования системы управления и защиты реактора.

В качестве рабочих органов СУЗ используются ИИ стержней-поглотителей нейтронов. По своим функциям стержни подразделяются следующим образом: 2 стержня аварийной защиты (АО), 8 компенсирующих стержней (КС), стержень автоматического регулирования (АР). В качестве поглотителя в стержнях СУЗ используются таблетки карбида бора. Они размещены в трубках из коррозионностойкой стали наружным диаметром 23 мм. Ниже стержня поглотителя все органы, кроме АР,
имеют вытеснители из алюминия. Каждый стержень АЗ и стержень АР имеют индивидуальные приводы. Каждые два компенсирующих стержня имеют один привод. Приводы стержней АЗ и КС имеют электромагнитные муфты. Стержни размещаются в каналах и соединяются с приводами с помощью тросов. Приводы размещены на двух специальных площадках на периферии бетонного массива биологической защиты и ограждены защитными экранами для предотвращения случайного воздействия на них при работах по транспортировке узлов в зале мостовым краном.

3. ЭКСПЕРИМЕНТАЛЬНЫЕ ВОЗМОЖНОСТИ РЕАКТОРА

В зависимости от требований экспериментальной программы рабочие загрузки реактора могут в процессе эксплуатации реактора изменяться. Конструкция реактора обеспечивает возможность таких изменений. Конфигурации двух из возможных рабочих загрузок активной зоны реактора даны на рис. 4. Основные параметры этих загрузок как для случая использования в реакторе ТВС с топливом 90%-ного обогащения, так и при использовании ТВС с топливом 36%-ного обогащения, даны в таблице П. Как видно из таблицы, запасы реактивности загрузок реактора "свежими" ТВС в случае использования ТВС с топливом 36%-ного обогащения не ниже, чем при использовании ТВС с топливом 90%-ного обогащения. При использовании ТВС с топливом 36%-ного обогащения плотность потоков тепло-вых нейтронов в экспериментальных каналах, размещаемых в активной зоне, ниже на ~15%, в отражателе — на ~10%. Плотности потоков быстрых нейтронов (с Е > 0,821 МэВ) в активной зоне — ниже на ~7%. Неравномерность энерговыделения в активной зоне слабо зависит от степени обогащения топлива в ТВС.

В представленных на рис. 4 рабочих загрузках в активной зоне могут размещаться до 10 каналов ампульного типа наружным диаметром до 68 мм для облучения образцов топлива или конструкционных материалов. Плотности потока быстрых нейтронов (с Е > 0,821 МэВ) в ампульных каналах до 1,5·10^{18}
Рис. 4. Возможные варианты загрузок реактора ВВР-СМ-20.
1—ТВС шеститрубная, 2—канал со стержнем А3, 3—канал со стержнем КС, 4—канал со стержнем АР, 5—каналы вертикальные экспериментальные в активной зоне, 6—каналы в бериллиевых блоках отражателя, 7—вытеснитель для экспериментального канала, 8—канал в шеститрубной ТВС для облучения блок-контейнеров, 9—бериллиевые блоки отражателя.
Таблица П
Основные параметры загрузок ВВР-СМ-20

<table>
<thead>
<tr>
<th>Параметр</th>
<th>Варианты загрузок реактора</th>
</tr>
</thead>
<tbody>
<tr>
<td></td>
<td>рис. 4а</td>
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<tr>
<td>Мощность реактора, МВт</td>
<td>20</td>
</tr>
<tr>
<td>Количество ТВС в загрузке</td>
<td>32</td>
</tr>
<tr>
<td>Объем активной зоны, л</td>
<td>95</td>
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<tr>
<td>Обогащение урана, %</td>
<td>90</td>
</tr>
<tr>
<td>Масса урана-235 в загрузке со &quot;свежими&quot; ТВС, кг</td>
<td>8,44</td>
</tr>
<tr>
<td>Запас реактивности загрузки со &quot;свежими&quot; ТВС, % Δ К/К</td>
<td>18,8</td>
</tr>
<tr>
<td>Максимальная объемная мощность, МВт/л</td>
<td>0,40</td>
</tr>
<tr>
<td>Максимальная плотность теплового потока, МВт/м²</td>
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</tr>
<tr>
<td>Неравномерность энерговыделения в активной зоне</td>
<td>1,9</td>
</tr>
<tr>
<td>Расход воды в первом контуре, кг/с</td>
<td>458</td>
</tr>
<tr>
<td>Потери напора на активной зоне, кПа</td>
<td>39</td>
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<tr>
<td>Температура воды на входе в ТВС, °С</td>
<td>45</td>
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<tr>
<td>Максимальная расчетная температура стенки твэла, °C</td>
<td>95</td>
</tr>
<tr>
<td>Температура начала кипения воды в наиболее напряженной точке, °C</td>
<td>122</td>
</tr>
<tr>
<td>Максимальная плотность потока нейтронов x10⁻¹⁸ м⁻²·с⁻¹:</td>
<td></td>
</tr>
<tr>
<td>- тепловых:</td>
<td></td>
</tr>
<tr>
<td>- в активной зоне</td>
<td>1,5</td>
</tr>
<tr>
<td>- в отражателе</td>
<td>2,0</td>
</tr>
<tr>
<td>- быстрых (E &gt; 0,821 МэВ) в активной зоне</td>
<td>1,9</td>
</tr>
</tbody>
</table>
В активной зоне внутри шеститрубных ТБС могут быть использованы 22 вертикальных канала внутренним диаметром 26 мм для облучения блок-контейнеров с образцами или изотопными мишеньми. Плотности потока быстрых нейтронов в этих каналах – до $1.9 \times 10^{18} \text{ м}^{-2} \text{с}^{-1}$, тепловых – до $1.5 \times 10^{18} \text{ м}^{-2} \text{с}^{-1}$.

В бериллиевых блоках отражателя внутри корпуса реактора в имеющиеся в них центральные отверстия диаметром 48 мм могут устанавливаться до 30 вертикальных каналов внутренним диаметром 42 мм, которые могут использоваться для облучений блок-контейнеров с изотопными мишеньми и другими образцами. Плотности потока тепловых нейтронов в этих каналах – $(2.3-1.2) \times 10^{18} \text{ м}^{-2} \text{с}^{-1}$. В бериллиевых блоках вне корпуса реактора могут устанавливаться до 18 вертикальных каналов внутренним диаметром 42 мм. Шесть из них устанавливаются над горизонтальными каналами, остальные – между каналами. Эти каналы могут использоваться для облучения блок-контейнеров с изотопными мишеньми. Плотности потока тепловых нейтронов в этих каналах $(0.9-0.6) \times 10^{18} \text{ м}^{-2} \text{с}^{-1}$.

В кладке из графитовых блоков за свинцовым экраном могут устанавливаться вертикальные каналы диаметром до 130 мм (вместо 4-х блоков) и диаметром до 68 мм вместо одиночных блоков. Диапазон изменения плотности потока тепловых нейтронов в каналах, установленных в кладке – $(1.8-0.4) \times 10^{17} \text{ м}^{-2} \text{с}^{-1}$. Каналы в кладке можно использовать для установки пневмопоект для активационного анализа, поскольку наличие свинцового экрана существенно снизит нагрев образцов, для легирования кремния и других экспериментов.

Для вывода пучков нейтронов реактор имеет 6 горизонтальных экспериментальных каналов диаметром 100 мм (рис. 2). В пределах бака реактора каналы представляют собой полые алюминиевые трубы, устанавливаемые между корпусом реактора и стенкой бака соосно с уже имеющимися каналами в бетонном массиве защиты реактора (шиберах). Конструкцией каналов
обеспечивается плотный контакт торцев канала с корпусом реактора и стенкой бака. Плотность потока тепловых нейтронов на торцах каналов - $(0,85-0,6).10^{18}\text{ м}^{-2}\cdot\text{с}^{-1}$.

Три горизонтальных канала диаметром 60 мм, имеющиеся в бетонном массиве, доходит только до стенки бака. Благодаря тому, что в этой части бака пространство между стенкой бака и корпусом реактора заполнено графитовыми блоками, плотность потока тепловых нейтронов на стенке бака $3.10^{14}\text{ м}^{-2}\cdot\text{с}^{-1}$. Эти три горизонтальных канала могут быть использованы, например, для нейтронной радиографии или активационного анализа.

4. ЛИТЕРАТУРА


MODIFICATIONS OF RESEARCH REACTOR LWR-15

AND EXPERIMENTAL REACTOR LR-0

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MODIFICATIONS OF RESEARCH REACTOR LWR-15 AND EXPERIMENTAL REACTOR LR-0

The new research reactor LWR-15 has resulted from a fundamental reconstruction of the research reactor WWR-S which had been operated in Nuclear Research Institute (NRI), Řež, Czechoslovakia since 1957. The original tank-type WWR-S research reactor was designed for thermal power of 2 MW and its power had been gradually increased during the first few years of operation up to 4 MW(th) by a modification of fuel assemblies. After a partial reconstruction of the technological circuits of the WWR-S reactor (1974-1976), its power reached 10 MW(th). There are a description of the last version of WWR-S research reactor briefly reminded and principal differences in the design and equipment of LWR-15 given in the paper. The experimental reactor LR-0 has its origin in the experimental reactor TR-0 which was put into operation in NRI in 1972, as a heavy water moderated, natural uranium fueled zero power reactor. After finishing the original research programme of TR-0 in 1979, the reactor was reconstructed and since 1983 has been operated as the fullscale experimental reactor LR-0 with light water moderated and slightly enriched uranium fuelled cores. The experience of more then 5 year operation in the framework of a very broad R & D programme, which has been asking very often to modify (at least partially) all the reactor systems, is given in the paper with a brief description of the experimental possibilities and special equipment of the LR-0 reactor.
1. INTRODUCTION

Research and experimental reactors have been playing an important role in the frame of great majority of national nuclear programmes. In the case of Czechoslovakia which besides the usual level of utilization of different forms of atomic energy (corresponding to the country's economy demands and possibilities) has an ambitious nuclear power programme, this is the more valid. Czechoslovakia is developing its nuclear power complex not only for covering its own electricity demands but also as an important part of its engineering exports. At least briefly, the current status of the Czechoslovak nuclear programme can be characterized by two main features:

1) the country with its 8 units of PWR (each of them having 440 MW_e power output) is producing nearly 27% of total electricity production by nuclear power and having additional 4 units of the same type and 5 units of PWR with 1000 MW_e each under construction is going to reach 50% share of nuclear power by the year 2000.

2) the Czechoslovak industry is manufacturing (except nuclear fuel) all main components of nuclear power plants by own capacity.

Meeting the requirements that arise from such a programme of nuclear power utilization needs adequate research and development programme including besides others research and experimental reactors of suitable capabilities. Because of the original Czechoslovak research reactor basis being established during the 15 year period between 1957-1972 was found not fulfilling all the demands a certain modernization programme
of this basis has been adopted to our plans for 80s.

The brief story of the mentioned "childhood" of the czechoslovak research reactor basis has been following: The first research reactor (and the first nuclear reactor facility at the territory of Czechoslovakia at all) was light-water-moderated and cooled reactor called WWR-S with high enriched U235 fuel and thermal power output of 2 $\text{MW}_{\text{th}}$. This reactor reached first time criticality in September 1957. Later on (1964), there was a modification of the reactor's fuel assembly arrangement realized and the reactor power increased up to 4 $\text{MW}_{\text{th}}$. During the years 1974-76, after a partial reconstruction of technological circuits, the thermal power of the WWR-S reactor had been increased up to 10 $\text{MW}_{\text{th}}$.

In the beginning of 70s the czechoslovak research reactor basis had been broadened significantly and 2 experimental zero power reactors were put into operation in the framework of the project of the first czechoslovak nuclear power plant called A-1 which was equipped by a heavy-water-moderated, natural uranium-fueled and gas (CO$_2$) - cooled reactor with the power of 150 $\text{MW}_{\text{el}}$. In 1971, there was a swimming-pool type experimental reactor SR-0 with the power of 2 $\text{kW}_{\text{th}}$ put into operation in the research and development basis of the czechoslovak manufacturer of heavy machinery including nuclear power devices - the Skoda Works comp. at the city of Pilsen. In the middle of 1972, there was another experimental zero power reactor TR-0 put into operation in the Nuclear Research Institute in Rez near Prague, being full-scale physical model of the core of the power reactor of A-1 station.
Using those experimental reactors and research reactor WWR-S, all research and experimental works were carried-out which were necessary for checking of the selected units of the A-1 NPP, its start-up and safe operation during the first period of its utilization including training of the operators and other staff. After a principal decision to base the czechoslovak nuclear power in the industrial scale upon the power reactors of PWR-type a series of experimental works was realized by use of inserting light-water moderated and slightly enriched U235 fueled zones in the heavy water moderated experimental reactor TR-0 during the period between 1976 and 1978.

Simultaneously, there was a decision adopted for the following decade of 1980s which supposed a realization of totaly five items in the frame of the czechoslovak research reactor basis modernization. Two of them being planned by NRI can be briefly characterized as follows:

- fundamental reconstruction of the research reactor WWR-S onto the research reactor LWR-15 with thermal power output of 15 MW(th).

- reconstruction of the TR-O reactor onto an experimental zero-power light water-moderated reactor LR-O with fuel assemblies being shortened but in their cross-section identical with those of power PWRs realized in the frame of the czechoslovak nuclear power programme (those being the reactors of WWER-type designed in Soviet Union).

Besides these two items which have been already realized by NRI and will be described in this paper, three additional items were incorporated in the czechoslovak research reactor basis modernization programme for the 1980s:
reconstruction of the experimental reactor SR-0 in the Skoda Works comp. in Pilsen;

construction of a training zero-power reactor VR-1P on the Faculty of Nuclear and Physical Engineering at the Czech Technical University in Prague (which has been described in details by the poster paper IAEA-SM-310/13P being presented in Poster Session I Research Reactor Programmes and Experience of this Symposium);

Construction of a training reactor VR-1B on the Electrotechnical Faculty of the Slovak Technical University in Bratislava, which has been started in May 1989.

2. DESCRIPTION OF THE RESEARCH REACTORS WWR-S AND LWR-15

The WWR-S research reactor facility in the version as it was operated until the end of 1987 could be briefly described as follows:

2.1. WWR-S Research Reactor

The research reactor WWR-S [1] is the light-water moderated and cooled, light-water and beryllium reflected tank type nuclear reactor. The reactor core contains highly enriched (80%) IRT-M-type fuel elements. Nominal reactor power is 10 MW(th), operating thermal neutron flux is $10^{17} \text{n/m}^2\cdot\text{s}.\text{MW}$, average coolant temperature in the core is 50°C. Critical mass of water reflected core with four irradiation channels inside the core is 3083.93 g U 235. Maximum built-in reactivity (cold, clean) is 12 dollars.

The reactor core consists of IRT-M-type fuel elements, Be reflector elements and structural components made from aluminium based alloy. The square lattice
pitch is 71.5 mm, the core has 52 working positions, 28-36 of them is designed for fuel assemblies, 10-15 for Be reflector, the rest are occupied by vertical irradiation channels, pneumatic tube conveyor, loops etc.

The fuel assembly consists of four or three square concentric tubes, 69x69 mm of outer one (the four tube assembly has one inner removable fuel element). The active fuel length is 580 mm. Fuel material is Al-U alloy with 37 solid % of uranium, average fuel enrichment is 80% in U 235. Fuel element wall thickness is 2 mm with 0.4 mm of fuel and 0.8 mm of Al based alloy both side cladding. Maximum burnup is 50 MWD per assembly, i.e. 40%. The control and safety rods are placed in the central positions of three tube fuel assemblies having the inner fuel element replaced by Al based alloy channel.

The control rods are 23 mm in outer diameter, 1 mm wall thickness, 21 mm absorber in diameter. The 8 control and 3 safety rods are operating inside fuel assemblies, the rod used for automatic control is located in Be reflector. The active length of all control rods is 600 mm. The bottom parts of the rods are joined with aluminium rods of 513 mm length and 23 mm in diameter. These Al parts compensate the local increasing of neutron thermal flux density. The total reactivity weight of all control and safety rods is 25 dollars.

The reactor core placed in aluminium cylinder (660 mm in diameter, 900 mm height) is cooled by water with continuous forced circulation (4 main pumps, 1 auxiliary pump for shut-down heat removal in the first circuit). There are 2 heat exchangers (water to water) between the first and second circuits and 3 exchangers (w/w) between the second and third circuits. Coolant flow rate at nominal power is 1600 m$^3$/hour, cross
sectional area for coolant flow 0.12 m², coolant velocity 3.5 m/s. First cooling circuit works under atmospheric pressure. Temperature difference of coolant between inlet and outlet of the reactor is 8.5°C (at nominal power). Average specific power density in the core is 12.3 kW/MW.l, average specific power for a weight unit of fuel is 230 kW/MW.kg U 235. Heat transfer area is 20.8 m², maximum heat flux is 718 kW/M², average heat flux is 481.3 kW/M². Maximum fuel element cladding temperature is 100°C.

Reactor vessel has cylindrical form (2.3 m in diameter, 5 m height), made from aluminium based alloy. Dimensions with shielding are 3.8 m in diameter and 6.8 m height. Radial shielding consists of water (80 cm), cast-iron (20 cm) and iron-concrete (160 cm), axial shielding to the top consists of water (350 cm), air (155 cm), steel-cast-iron (80 cm) and axial shielding to the bottom consists of water (115 cm).

2.2. LWR-15 Research Facility

In terms of the fuel, moderator, reflector, coolant, and core layout the reactor is identical to the WWR-S reactor. Also unchanged remains the reactor shielding, as well as the overall reactor building structure.

The main elements being reconstructed are as follows:

- replacement of the existing aluminium vessel with a stainless steel vessel;
- reconstruction of the primary and the purification circuits to assure their reliable function up to a reactor output of 20 MW(th);
- reconstruction of the safety and control systems to keep all nuclear safety regulations valid in Czecho-slovakia.
In addition the air ducts, hot chambers, electrical auxiliaries, and health control system have been reconstructed. A central information system DASOR will be added to provide and record all necessary reactor operation data. Starting 1991 the 80% enriched fuel will be replaced by 36% enriched IRT-M type fuel elements.

Following the reconstruction as stated above the reactor LWR-15 will be able to work at 15 MW(th) power with increased neutron flux densities maintaining the principal physical parameters of the WWR-S reactor core and keeping all nuclear safety regulations required.

The reactor safety and control system has undergone a principal modification in both logical and safety circuits as well as control and safety rods. In the case of WWR-S reactor all the control and safety functions were derived from the measuring devices which had three identical channels (consisting always of one logarithmic and one linear measuring channel). The logical and safety circuits did not allow the safety regulations to be applied to full extent.

In the LWR-15 system the measuring units with both analogue and binary output will be used for logical and safety circuits. The equipment for measuring neutron flux densities will consist of:

- three identical independent measuring devices each of them consisting of one wide-band start-up channel, one logarithmic channel, one linear channel and one special channel monitoring excess of maximum power of the reactor,
- reactivity meter.

Neutron flux density measuring system in connection with logical and emergency circuits will provide for:

- implementation of safety regulations required,
- proper handling of signals from measuring devices in manual and automatic operation regimes,
- availability of relevant information, warning and emergency signals,
- correct function of control and safety rods including the rod mode selection (shut-down, compensation, control),
- correct intervention order during the start-up and reactor operation.

There are also substantial changes in construction of the system of control and safety rods. In the case of WWR-S reactor the rods were suspended on strings and wound electromechanically in an off-reactor position. This system has been replaced by a system of autonomous control monoblocks with the rods rigidly connected to the driving unit (stepping motor) via rack gearing. The whole electromechanically controlled monoblock works inside the reactor. Whereas the old system enabled only three of the total eleven absorbing rods to be used as shut-down rods, the monoblock design makes it possible to use all twelve rods for this purpose.

The changes in the safety and control system required a deep safety analysis in which the PSA methods have been adopted. Other safety systems, to which the PSA method might be applied, are the primary cooling circuit of the LWR-15 reactor, the emergency cooling system, the emergency shower system used in the case of cooling medium leakage in the primary cooling circuit [2].
3. EXPERIMENTAL REACTOR LR-0

After the decision (in early 1970s) to base the nuclear power in Czechoslovakia upon PWRs and after finishing the research programme on the neutronics of heavy water moderated and natural uranium fueled reactor lattices (in the later 1970s) the experimental reactor TR-0 was reconstructed for the purpose of a full-scale experimental research of light water moderated and slightly enriched uranium fueled cores. The new version of the experimental zero-power reactor in NRI has been called LR-0. The LR-0 reactor reached the critical state for the first time on 19 December 1982 and after a half-year trial operation was put into a regular operation in June 1983. More then five years of the utilization of LR-0 for a large variety of experiments allowing not only the studies of the neutronics of the mentioned type of reactor cores but also a number of some additional applied tasks connected with the safe and reliable operation of PWRs in Czechoslovakia (space-energy distribution of neutron flux impinging on reactor pressure vessel/exposure/for evaluation of its radiation damage, neutronics of spent fuel storage etc.) has given a good deal of experience with frequent modifications of such a flexible reactor facility like the LR-0 experimental reactor has been.

3.1. Description of the LR-0

The LR-0 reactor makes it possible to perform experiments simulating neutronical processes (including kinetics) and owing to verify calculated characteristics of the WWER-type reactor core. The multipurpose design of the LR-0 technological equipment allows accomplishment of experiments with both WWER-440 and WWER-1000 type fuel assemblies in either symetric or non-symetric arrangements with standard or variable pitch in the
triangular core lattice. Measurements may be performed with experimental absorption clusters in WWER-1000 type assemblies fixed in desired heights. Moreover, one selected experimental cluster is movable and may be shifted even during the critical state of the reactor.

The fission chain reaction in the core is controlled mainly by changing moderator level in the reactor vessel. Thus, clean cores without absorbers may be simulated. The moderator may contain a certain concentration of the boric acid ($H_3BO_3$) between 0 up to 12 g per liter $H_2O$ and may be heated up to $70^\circ C$. The thermal power of the reactor is limited to 5 kW for 1 hour of the operation and the heat is dissipated into the moderator. The thermal neutron flux density in the centre of the core is $10^{13} m^{-2}s^{-1}$.

The technological equipment of the LR-0 includes a reactor vessel, a core assembly, moderator circuits and a reactor control system. The design of the reactor makes possible an easy rearrangement of the reactor core and a modification of an operational regime according to the requirements of an experiment with adhering to a high level of nuclear safety.

3.1.1 LR-0 Vessel

The aluminium reactor vessel is situated in a concrete shield and covered with mobile shielding platforms. The lower cylindrical part of the vessel has a diameter of 3,5 m and a height of 6,5 m, the upper part of the vessel with a square cross-section 6x6 m and height of 1,5 m is covered by a rotary circular cover with a system of enclosable openings, which provide accessibility of any site in the vessel. The bottom of the vessel is equipped by 3 necks of I.D. 0,2 m; two of them are assigned for the release of moderator, the central neck serves for the moderator inlet. Besides this, there are 12 dry horizontal
channels of I.D. 0.07 m for detectors of a control device, which provide neutron flux density monitoring in the reactor, and 1 dry vertical channel, which passes through the inlet neck serving for inserting of a neutron source closely to the heel of the central fuel assembly. Further necks in the reactor vessel cylindrical wall enable positioning of measuring heads of moderator level indicators or serve for experimental purposes. The mantle of the vessel is covered with 1 mm of Cd-sheet and removable thermal shielding plates.

3.1.2 LR-0 core

The cores of experimental reactor LR-0 are created in principle from shortened dismountable experimental fuel assemblies of the WWER-440 and WWER-1000 types. All those fuel assemblies are loaded with unified fuel elements. The fuel element is represented by a coating tube (cladding) of O.D. 9.15x0.72 mm made of zirconium with 1% niobium which is filled with sintered UO₂ pellets of O.D. 7.53 mm with central hole of 1.4 mm in diameter. Several fuel elements may be opened at both ends and the pellets may be removed. The height of the fuel filling in the element is 1.25 m the total length of the element is 1.357 m. The fuel enrichment varies from 1.6 to 4.4 % U235.

The LR-0/440 assembly has 126 fuel elements in a triangular lattice with a pitch of 12.2 mm. The spacer grids used are either standard stainless steel spot-welded honeycomb-shaped ones or made of drilled plates. The hexagonal aluminium coating tube of the assembly is 2 mm thick and its outer size is 144 mm nut. There is a zirconium-1% niobium tube of O.D. 10.3x0.65 mm going through the centre of the assembly. Under experimental conditions it is also possible to simulate the influence of a partial insertion of a control element assembly in the core.
The control element assembly is represented here by a storeyed assembly whose absorption part has exchangeable segments.

The hexagonal coatless fuel assembly LR-0/1000 is formed by a skeleton which is filled in by the fuel elements. The stainless-steel skeleton of the assembly is composed of the head, the heel and 18 tubes of O.D. 12,6x0,8 mm which serve, besides their supporting function, as a guide thimbles of absorption elements of an absorption cluster which is a model of an analogous control element of the corresponding power reactor WWER-1000. The size of the assembly head is 0,235 m nut. There are 312 fuel elements in the assembly being arranged in a triangular lattice with a pitch of 12,75 mm. There is a zirconium - 1% niobium tube again of O.D. 10,3x0,65 mm passes through the centre of each assembly for in-pile instrumentation. The distribution of fuel elements, guiding tubes and a central tube is provided by means of five stainless-steel spot-welded honeycomb-shaped spacer grids. Their number and position along the height of the assembly may be modified according to experimental requirements.

The absorption cluster is formed by a bundle of 18 elements fixed to the stainless-steel cluster head. The absorption elements are shifted into the guide thimbles of the fuel assembly skeleton, their distribution may be either regular (i.e. in a regular triangular lattice) or standard (identical with the slightly irregular arrangement in the WWER-1000 power reactor assembly). The absorption elements with the coating stainless-steel tube of O.D. 8,2x0,6 mm are filled with B₄C pellets. Instead of the absorption elements it is possible to use elements of burnable absorber.
There is a driving unit of the cluster attached to the assembly head which allows to lift and drop the cluster at a velocity of 25 mm/s or to fix the cluster in the range of 1300 mm with steps of 2 mm. The gravitational fall of the cluster from the height of 1300 mm takes less then 3 s. The clusters are suspended by help of a free roller from a stranded wire. The wire is wound onto a drum driven by a step motor.

3.1.3. LR-0 Moderator Circuits

The experimental reactor LR-0 has the following moderator circuits: a main, an auxiliary and a preparation circuit. The main moderator circuit serves for filling the reactor vessel with moderator and its pouring back into the storage tank. For safety reasons, the main circuit is separated from the other circuits of the moderator during operation. Before filling into the reactor, the moderator is stored in a 25 m³ storage tank which is below the level of the reactor vessel bottom. The filling of moderator into the reactor vessel is provided by means of two glandless pumps, which drive the moderator through a tubing system and two pairs of control and closing valves into the reactor vessel. The main inlet closing valve may operate in a dose regime with an adjustable time interval length of sampling. From the vessel, the moderator can be released by a pair of control and closing valves or by two safety valves with an inner diameter of 200 mm.

The moderator preparation circuit is disconnected from the main circuit during operation. Demineralized water supplied into the circuit is stored in three storage tanks with a total volume of 18 m³.

Concentrated $\text{H}_3\text{BO}_3$ solution (up to 40 g $\text{H}_3\text{BO}_3$/l $\text{H}_2\text{O}$) is prepared in a vessel with a stirrer and an electric heater and stored in a storage tank of the
concentrate. The volume of the dissolving tank as well as the storage tank of the H$_3$BO$_3$ concentrate is 3 m$^3$. Demineralized water and concentrated H$_3$BO$_3$ solution can be mixed and stored in a mixing tank (25 m$^3$). There are also other three storage tanks, each of them with a volume of 6 m$^3$, in the preparation circuit.

3.1.4. LR-O Control system

The system of the reactor control consists of equipment for reactor state determination, for data treatment and contact with the staff and for control of reactor conditions.

The means for reactor state determination include measuring equipments of power and relative power rate as well as of the moderator level in the reactor vessel and the height setting of the experimental absorption clusters.

The power measuring equipment is divided into a startup apparatus, an operation apparatus and an override protection. Each apparatus has three independent channels. The startup and operation apparatuses measure power and relative power rate at all stages of reactor operation. The linear branch of the operation apparatus can work in two operational regimes (following and fixation of power) and offers deviation signals. The protection from exceeding the power above a preset limit is provided by the override protection apparatus. Compensated current chambers serve as detectors of the operation and override protection apparatuses. Each channel of the startup apparatus is equipped with a fission chamber. Neutron detectors of all the measuring apparatuses are situated in twelve horizontal dry channels under the supporting plate of the reactor core or may be inserted into vertical channels around the core.
The operation moderator level indicators in three independent channels provide for measuring the height of the moderator level in the reactor vessel with the accuracy of \( \pm 1 \text{ mm} \).

The means for data treatment and contact with the staff include circuits of automatic test, safety and logic circuits. The test circuit controls the initial state of important technological equipments and of the control system including functional checking of the means for state determination. The system of operation moderator level indicators and the power measuring system compare measured values with preset levels and transform the result to logic signals. These logic signals are treated in safety circuits according to the selection principle "two out of three". Safety circuits also treat simple brakedown signals, e.g. pressure drop of air in distributions for pneumatic valves, disconnecting the main circuit of moderator from the preparation circuit, pushing the scram button, etc. Logic circuits organize work regimes of the control system, perform an initial test, control the reactor startup procedure, supervise allowability of handling instructions of the staff and provide for mutual coordination of functions and signalization.

The contact of the staff with the control system is facilitated by measuring, signalization and handling elements. These elements make parts of the control system.

The control of the fission chain reaction is provided by changing the moderator level in the reactor core. The action units are represented by pumps and valves of the main moderator circuit. Two pneumatic safety valves are the basic action units of the reactor safety system. As a complementary action unit the
system of absorption clusters is used. From all the absorption clusters used in the reactor core, at least 6 are in operation in the upper end position. The remaining clusters in the core serve for experimental purposes.

4. CONCLUSIONS

The czechoslovak research reactor basis had been found by the end of 1970s being non-adequate to the requirements of a relatively ambitious national nuclear power programme based upon power reactors of the PWR-type and has been therefore gradually modernized during 1980s by realization of a five item research reactor basis modernization programme. Two partial goals of this programme - fundamental reconstruction of the research reactor WWR-S onto the research reactor LWR-15 and modification of the experimental reactor TR-0 onto the experimental reactor LR-0 have already been reached by the middle of the year 1989. The stages of physical and power startup of the LWR-15 as well as the more then five year regular operation of LR-0 verified a quality of the solutions of all engineering problems connected with such a pretentious programme of the modernization of the research reactor basis. This, besides others, gives a good hope to contribute significantly to an improvement of economical efficiency, reliability and safety of the national nuclear power programme of the country.
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REPLACEMENT OF THE ADVANCED TEST REACTOR CONTROL ROOM

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REPLACEMENT OF THE ADVANCED TEST REACTOR CONTROL ROOM

ABSTRACT

The control room for the Advanced Test Reactor has been replaced to provide modern equipment utilizing current standards and meeting the current human factors requirements. The control room was designed in the early 1960 era and had not been significantly upgraded since the initial installation. The replacement did not change any of the safety circuits or equipment but did result in replacement of some of the recorders that display information from the safety systems. The replacement was completed in concert with the replacement of the control room simulator which provided important feedback on the design. The design successfully incorporates computer-based systems into the display of the plant variables. This improved design provides the operator with more information in a more usable form than was provided by the original design. The replacement was successfully completed within the scheduled time thereby minimizing the down time for the reactor.

1. INTRODUCTION

The Advanced Test Reactor (ATR) is a 250 MWt research reactor located at the Idaho National Engineering Laboratory (INEL). The reactor provides high neutron fluxes for material testing in support of the U.S. Department of Energy programs. The reactor has been in operation since 1969 and until recently retained the original control room design. Although safety circuits and associated instrumentation have been upgraded to maintain modern standards, most of the operator-machine interface remained the same as the original installation until early 1989. A modern control room has recently been installed to provide an improved operator-machine interface and to provide easier access to more operating information. This paper describes the changes made, the reasons for them, the approach to control the modification, and the experience resulting from the modification.

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2. FACILITY DESCRIPTION

The ATR is designed for irradiation of a number of experiments simultaneously in flux traps formed by fuel elements arranged in the serpentine geometry shown in Fig. 1. The reactor is operated at relatively low temperature and moderate pressure. It is fueled with highly-enriched U-235 contained in plate-type fuel elements constructed of aluminum. It is water moderated and utilizes beryllium as the reflector. The control of the reactor is accomplished by adjustment of hafnium absorbers located in the central cruciform housing and in the reflector. The safety rods are hafnium absorbers located in six of the flux traps that also include irradiation facilities.

The reactor is controlled from two separate control rooms. One is the process control room which contains the controls to adjust the primary coolant system parameters such as pressure, flow, and temperature. The second is the main control room which contains the reactivity controls. The main control room contains indications and alarms that are available in the process control room, but it contains no active controls for those parameters. The plant operation is directed from this room by the shift manager and appropriate staff. It necessarily contains the appropriate communication facilities to assure that all needed interfaces within the plant and the control areas are established. The control of experiment parameters is accomplished at control consoles located in basement areas of the building in the vicinity of the supporting equipment for the experiments.

The design of the control room was completed around 1960 and has remained basically unchanged since that time. A significant upgrade of the plant protective system (PPS) was completed in 1978, but the upgrade did not change the appearance of the control room or the manner in which the data is displayed to the operator.

A computer-based data acquisition system is included to provide operating history and some manipulation of data for the use of the operator. Several modifications to this system have been completed since the original design to take advantage of the changing capability of computers. The current system uses the VAX-11/750 and includes two machines for reliability.

There has been a simulator facility for the reactor control room since the beginning of operation. This facility provided adequate simulation of the system response, but it did not duplicate all features of the control room.

3. CONSIDERATIONS FOR THE DESIGN CHANGE

The experience of the failure at the Three Mile Island nuclear plant indicated that the operator-machine interface has considerable importance to the performance of operators in control of accident sequences. This experience resulted in industry activities to improve the interface and provide the operator with added tools to follow the
Fig. 1. Horizontal cross section of ATR core.
progression of both normal operation and accident sequences. This experience was a factor in initiating an updated design for the ATR control room. The features of the ATR control room did not include any serious deficiencies requiring immediate changes; however, the evaluations as a result of the industry concerns indicated that improvements could be obtained. Improvements in grouping of displays and controls to provide better conformance to established Human Factors Engineering were design requirements.

In addition to the improvements considered for the reactivity control interface, other features of the control room were evaluated for improvement at the same time. The recorders in use in the control room were designed and built in the 1960’s and were becoming difficult to maintain. Replacement parts often were not being manufactured and had to be specially fabricated or salvaged from out of service recorders from other facilities. Rather than replace the equipment in a piecemeal fashion, replacement of all recorders with available modern recorder systems was chosen.

4. DESCRIPTION OF THE CHANGES

The change of the control room design takes advantage of the power of computer-based systems that can provide flexibility to display data in several forms and to manipulate it to provide more information than the original design. However, such designs are a considerable departure from the original design. Therefore, to assure a successful transition to a new design, most features of the original display were maintained. The original design displayed essentially all of the system parameters and alarms on an upright panel immediately in front of the operator console. The upright panel and the displays were maintained by installing modern recorders. The recorders were relocated to establish appropriate grouping of parameters.

The major change to the control room is in the operating console. The control switches and position indicators for the reactivity control elements are replaced with modern equivalents. Rotary switches have been replaced with matrixed push button switches for control element selection with rotary switches retained for actual element motion control, and all switches have been relocated for better access. The location of the position indications are also changed to provide easier viewing consistent with the recommendations of human factors studies. Additionally, the modified console contains five video displays for the operators use. A sixth video display is installed at the shift managers station which is away from the operating console but still within the control room. These displays have the capability to show current status and recent history of several categories of parameters.

The currently available information from these video displays is listed in Table I. As more experience is obtained with the system, others will be developed. As may be noted from the table, there are displays that provide the operator with information on the control logic (interlocks and controls) including the current logic status. The system has the potential to provide listings of procedures and drawings that may be of use for particular operational evolutions or particular accident sequences. This application remains to be developed.
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<tbody>
<tr>
<td>Reactor Quadrants</td>
<td>Reactor process parameters by core quadrant (flow, pressure, temperature, and others for the primary coolant)</td>
</tr>
<tr>
<td>Reactor Total</td>
<td>Power from the thermal monitors for reactor total and each core quadrant</td>
</tr>
<tr>
<td>Water Power</td>
<td>Recorder icons for the thermal power system</td>
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<tr>
<td>Fission Break</td>
<td>Recorder icons for the fission break monitors in the primary coolant system</td>
</tr>
<tr>
<td>Wide Range</td>
<td>Recorder icons for the rate of change monitors (neutron flux)</td>
</tr>
<tr>
<td>PPS Part I</td>
<td>Recorder icons for pressure, temperature, neutron level, and radiation level (Plant Protective System)</td>
</tr>
<tr>
<td>Log CR and Log N</td>
<td>Recorder icons for the log count rate and the log N systems (neutron flux parameters)</td>
</tr>
<tr>
<td>Power Distribution</td>
<td>Block diagram of the fraction of full power for the N-16 recorders (power distribution)</td>
</tr>
<tr>
<td>Neutron Level</td>
<td>Recorder icons for the neutron level system</td>
</tr>
<tr>
<td>Power Split .</td>
<td>Relative and absolute power generation for each lobe of core</td>
</tr>
<tr>
<td>Rod Panel</td>
<td>Position of reactivity control elements and safety rods</td>
</tr>
<tr>
<td>Procedure Panel</td>
<td>Diagrams of control system logic and status (control elements and safety rods)</td>
</tr>
<tr>
<td>Message Window</td>
<td>Status of selected items that are soft alarms (notification)</td>
</tr>
<tr>
<td>TTY Window</td>
<td>Allows request of data not a part of a particular display</td>
</tr>
<tr>
<td>Building Alarms</td>
<td>Provides indication of building door status</td>
</tr>
<tr>
<td>SER Report</td>
<td>Sequence of Event Report (sequence for multiple alarm conditions)</td>
</tr>
<tr>
<td>History Menu</td>
<td>History of selected parameters at requested frequency (trending)</td>
</tr>
<tr>
<td>DAC/DAN</td>
<td>Selection of the data network to provide the data (two available)</td>
</tr>
</tbody>
</table>
The video displays are driven by six standard Hewlett Packard series 9000 work stations. These work stations receive plant parameter information from the plant data system which accumulates the information. The work stations are programed to arrange the parameters into the desired displays for the operators use. The displays utilize colors to provide the operator with information about the status of variables to ensure that they are maintained within the required ranges. Variables within normal ranges are displayed in green, if an alarm level is reached, the display of the variable becomes orange, and if a trip level is reached, red is displayed. The displays also provide trend data for selected parameters, such as outlet temperature, neutron level, and primary coolant flow to assist in the interpretation of both normal operation and off-normal conditions.

In addition to the changes to the operating console and the replacement of the existing recorders, the annunciator panels and the communications systems between the control rooms and the experiment console areas were replaced. These replacements are essentially like for like in function, but current technology is used in the new panels and the communications system. The annunciator panels utilize fiber optic technology and are computer based allowing additional flexibility.

As part of the over all facility upgrade a new reactor control room simulator was provided. The simulated control room is built to provide a high fidelity simulation in physical appearance, operational characteristics, and system response of the reactor control room. This permitted evaluation of all aspects of the operator reactor interface including human factors considerations prior to finalizing design for the reactor control room. All programming for the video display units was developed and tested in the simulator. The total control room simulation was utilized for nearly a year for reactor operator training prior to start up of the reactor with the new control room. Operating concerns identified during this training period were considered and changes made in the design as appropriate. As a result, the design benefitted from the hands-on experience during training.

5. IMPLEMENTING THE CHANGE

Formal procedures for ATR modifications are in place to assure that modifications are carefully reviewed and meet the requirements for a nuclear plant. These procedures include formal reviews of the design requirements and of the design relative to the requirements. Significant changes such as the control room replacement are also accompanied by a readiness review which is conducted by an independent committee to assure that the design has been properly completed and that the necessary training and equipment are available for the modification. The design of the control room replacement was completed well in advance of the actual installation allowing considerable time for planning and review of the activities.
Since the modification would remove normal safety systems from service, it was necessary to evaluate the plant status during the modification to assure safety. The modification was completed without fuel in the reactor which reduced the number of systems required for safety. A temporary panel containing necessary systems such as the area evacuation system, manual control of the confinement isolation with associated heating and ventilating functions, and security communication equipment was installed. This panel provided the required plant protection during the modification.

Prior to beginning the installation, considerable effort was devoted to planning and scheduling the activities to minimize errors and help assure that the modification was completed in a minimum time. This planning also coordinated other plant activities, such as experiment loop modifications and general plant preventive maintenance in progress at the same time. The planning of the work prior to beginning the modification was provided to the craftsmen in the form of work instructions that helped assure the final product.

Following the control room replacement, each system affected by the modifications was tested to assure that the modifications were consistent with the design. Some of the installation required that safety systems were disconnected from certain panels involved in the modification and then connected to new recorders without modification to the function of the safety system. Accurate completion of such activities was clearly important; therefore, quality assurance procedures were utilized extensively in the work activity. The systems were tested using formal procedures that were reviewed by a committee that reviews all ATR procedures. The results of the testing were finally reviewed by a readiness review committee and all control room systems and control room simulator systems were placed under configuration control prior to returning the facility to operational status. The readiness reviews conducted prior to the beginning and at the end of the replacement were completed by independent committees from both the U. S. Department of Energy and the operating contractor.

6. RESULTS OF THE REPLACEMENT

The schedule for the replacement of the control room included 60 days for installation and testing. As a result of the detailed planning and scheduling in advance of the activity, the work was completed consistent with the schedule. The testing detected only a few errors demonstrating the importance of the detailed preparations.

Experience to date indicates that the control room design has been easy for the operating staff to use and has provided a reliable long term supportable facility with no serious operational concerns. Administrative requirements are in place to assure that selection of displays, which is at operator discretion, includes certain parameters that are important to proper operation of the reactor. Administrative requirements are also in place to assure that the information displayed in the upright panels (the original configuration) is appropriately considered in control actions.
The operating experience has indicated that the existing data acquisition system does not update the information transmitted to the workstations as often as is needed in certain circumstances. This delay in the transmission of data is being corrected to provide fast data updates consistent with operational requirements and human factors concerns.

7. SUMMARY

The reactor control room for the ATR has been replaced with modern equipment including an operating console utilizing a computer-based system to provide parameter and procedural information to the operator. The replacement is based on current human factors principles to assure that the information provided to the operator can be utilized with a minimum of concern for errors in interpretation. The displays also provide additional trending information important to operation during normal as well as off-normal conditions. The modification successfully met the design requirements and was completed within the scheduled time which minimized the unavailability of the facility.
RESULTS FROM POST-MORTEM TESTS WITH MATERIAL FROM THE OLD CORE-BOX OF THE HIGH FLUX REACTOR (HFR) AT PETTEN

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RESULTS FROM POST-MORTEM TESTS WITH MATERIAL FROM THE OLD CORE-BOX OF THE HIGH FLUX REACTOR (HFR) AT PETTEN

ABSTRACT

Results are reported from hardness measurements, tensile tests and fracture mechanics experiments (fatigue crack growth and fracture toughness) on 5154 aluminium specimens, fabricated from remnants of the old HFR core box. The specimen material was exposed to a maximum thermal neutron fluence of $7.5 \times 10^{26} \text{n/m}^2 (E < 0.4 \text{eV})$.

Test results for this fluence (ratio of the thermal to fast neutron flux density is 1.17) are: hardness 63HR15N, 0.2 - yield strength 525 MPa and total elongation 2.2% strain. Material which was exposed to a lower thermal fluence of $5.6 \times 10^{26} \text{n/m}^2$, but with a thermal to fast neutron ratio of about 4, shows more radiation hardening: 67HR15N, 0.2 - yield strength 580 MPa and 1.5% total elongation.

Fatigue crack growth rates range from $5 \times 10^{-5} \text{mm/cycle}$ to $10^{-3} \text{mm/cycle}$ for $\Delta K$ ranging from 8 to 20 MPa\(\sqrt{\text{m}}\). The most highly exposed ($7.5 \times 10^{26} \text{n/m}^2$) material shows accelerated fatigue crack growth due to unstable crack extension at $\Delta K$ of about 15 Mpa\(\sqrt{\text{m}}\). The lowermost meaningful measure of plane strain fracture toughness is 18 MPa\(\sqrt{\text{m}}\).

Except for the fracture toughness which is a factor of about 3 higher the results show reasonable agreement with the expected mechanical properties estimated in the "safe end-of-life" assessment of the old HFR vessel.
INTRODUCTION

The HFR reactor vessel was replaced in 1984 after more than 20 years of operation because doubts had arisen over the condition of the aluminium alloy construction material (ASTM 5154). Data from destructive testing of the old vessel material are used in support of the design analysis for the new vessel which is constructed from the same material.

EXPERIMENTAL

Material, original condition.

The construction material of the old HFR vessel was a modified British aluminium-magnesium alloy, designated BS1477 NP5/6M (modified : 3.5-3.9% Mg). The chemical analysis of the fabrication cast is given in Table 1. The composition is close to the specification for type 5154 aluminium alloy.

Table 1. Chemical composition of the vessel material (wt%)

<table>
<thead>
<tr>
<th></th>
<th>Mg</th>
<th>Si</th>
<th>Fe</th>
<th>Mn</th>
<th>Cu</th>
<th>Zn</th>
</tr>
</thead>
<tbody>
<tr>
<td>old vessel</td>
<td>3.7</td>
<td>0.14</td>
<td>0.38</td>
<td>0.32</td>
<td>0.02</td>
<td>0.03</td>
</tr>
<tr>
<td>5154 alloy</td>
<td>3.1-3.9</td>
<td>- 0.45 -</td>
<td>0.10</td>
<td>0.10</td>
<td>0.20</td>
<td></td>
</tr>
</tbody>
</table>

Room temperature (300 K) tensile properties are : 0.2 - yield stress 100 MPa, UTS 250 MPa, uniform elongation 20% strain and total elongation 25% strain. The data illustrate the fully annealed "Soft" condition of the material, similar to the 5154-0 condition.

Material, end-of-life condition.

The most highly exposed segments of the old reactor vessel, the centre parts of the core box, have been selected for post-mortem testing.

The walls have been exposed to neutron radiation for a period of about $4.24 \times 10^8$ s, ending November, 1983. The accumulated neutron fluences, calculated for the mid-centre positions, are listed in Table 2.
Table 2. Estimated neutron fluences of the core-box walls, calculated for the mid-centre positions (ref. 3).

<table>
<thead>
<tr>
<th>fluence, $10^{26}$ n/m$^2$</th>
<th>Northwall</th>
<th>Eastwall</th>
<th>Southwall</th>
<th>Westwall</th>
</tr>
</thead>
<tbody>
<tr>
<td>thermal ($E &lt; 0.414$ eV)</td>
<td>5.6</td>
<td>3.2</td>
<td>5.0</td>
<td>7.5</td>
</tr>
<tr>
<td>fast ($E &gt; 0.1$ MeV)</td>
<td>1.7</td>
<td>0.8</td>
<td>1.9</td>
<td>6.9</td>
</tr>
<tr>
<td>ratio thermal/fast</td>
<td>3.3</td>
<td>4.0</td>
<td>2.6</td>
<td>1.1</td>
</tr>
</tbody>
</table>

After the destructive removal of the old vessel, the core box walls were preserved for further segmentation and preparation of tensile specimens and compact tension specimens. Further material remains available for bend test specimens and fracture toughness specimens.

All specimens were machined from mid-thickness ($\frac{3}{4}T$) material by equally reducing both surfaces, except for the tensile samples from the West wall which have been produced from quarter thickness ($\frac{1}{4}T$, $\frac{3}{4}T$) material.

Testing.
The post-mortem testing encompassed Rockwell hardness measurements, tension testing, fatigue crack growth testing and fracture resistance measurements. All tests were performed in air at room temperature.

Superficial Rockwell hardness measurements were made on the mill-cut side-surfaces of the blocks, prior to the final machining of the specimens. The tests were performed according to the ASTM standard method E 18 using the 15N scale with 150N total load applied by the diamond cone indentor and a load removal time of 8s.

Tensile tests were performed according to the ASTM standard test method E 8. The specimens were pin-loaded and tested at constant strain rate of $5 \times 10^{-4}$ s$^{-1}$.

The fatigue crack growth tests were performed under constant-amplitude cyclic loading (triangular wave form) with R-ratio of 0.1 (ratio of the peak values of the applied loading cycle). The mechanical notch (crack starter) length
was 12.5 mm ($a/W = 0.25$). The pre-crack extension length was 2 mm.

Pre-crack fatigue loading was performed at a cyclic frequency of 20 Hz, crack growth testing was performed at 10 Hz. The target final crack length was 32 mm ($a/W = 0.64$). The $\Delta K$-values ranged from about 8 MPa$\sqrt{m}$ to 25 MPa$\sqrt{m}$. The tests were performed according to the ASTM test method E 647 for constant-load-amplitude fatigue crack growth rates above $10^{-8}$m/cycle. Crack extension was measured by means of the direct current potential drop technique.

After termination of the fatigue crack growth tests the specimens were fracture-loaded by means of displacement controlled monotonic loading until separation. The applied displacement and the corresponding load were recorded to derive a measure of the fracture toughness of the specimens. This fracture-loading was performed after having achieved a certain fatigue crack length which was less or equal to the target final fatigue crack length of 32 mm. The fracture resistance measurements were performed basically according to the ASTM E 399 standard practice for plane-strain fracture toughness ($K_{IC}$) measurements, however the results were analyzed according to the standard practice ASTM B645 for fracture toughness testing of aluminium alloys.

Post-test examinations consisted of optical microscopy, fractography, transmission electron microscopy and microprobe X-ray analyses to measure the Si-content of selected samples. The silicon is primarily created by the nuclear reactions of the thermal neutrons with the aluminium. The thermal neutron exposure of the samples is calculated from the Si-measurements, taking into account the original Si-content of the non-exposed material.

RESULTS AND DISCUSSION

In this paper the accent is placed on the results from the fracture resistance measurements. The data, obtained from twenty pre-fatigue-loaded specimens, are reported in terms of the stress-intensity factor $K_1$, the linear-elastic fracture parameter being used in the recent defect analyses of the new HFR vessel (1). The data are combined with related information from the other experimental measurements. The full extent of all experimental results will be reported together with the metallographic observations at an appropriate symposium next year (2).
Two versions of the Compact-Tension type specimen were originally intended to be applied for the post-mortem testing of the core-box material. The dimensions of the specimens for the fatigue crack growth experiments are 62.5 mm width and 50.0 mm height. The thickness of these specimens is 10.0 mm or 12.5 mm. Because the dimensions of specimens for fracture toughness experiments can only be justified after completion of the experiment, a specimen with dimensions of 50.0 mm width and 48.0 mm height with the conservative thickness dimension of 17.0 mm was considered to be favourable to measure valid $K_{IC}$-values. However, due to budgetary restrictions the fatigue crack growth specimens were also used for the fracture resistance measurements. The specimens did meet the E399 thickness (B) requirement

$$B > 2.5 \frac{K_Q}{\sigma_y}\sqrt{t}$$

with B being 10.0 mm or 12.5 mm, $K_Q$ the conditional fracture toughness and $\sigma_y$ the tensile yield strength. One specimen from the East wall, with the lowest thermal neutron exposure and the highest $K_Q$-value, should have a thickness of about 13.0 mm, if a yield stress value of 420 MPa is assumed.

The twenty specimens were almost equally divided over the four walls of the old core-box. The selected positions represent about 50 to 90 % of the maximum neutron exposure of each wall. The measured Si-content ranges from 0.55 to 1.85 weight percentages. The thermal fluences, calculated from the Si-measurements, range from $1.8 \times 10^{26}$ n/m$^2$ to $7.5 \times 10^{26}$ n/m$^2$. The estimated values of the thermal-to-fast-neutron flux density ratio range from the minimum value of 1.0 for a specimen from the west wall to a maximum value of 6.3 for a specimen from the South wall (3).

At shut-down of the old HFR reactor vessel in 1983, the maximum thermal neutron fluence at the center of the West wall has been estimated at $7.5 \times 10^{26}$ n/m$^2$ (4). From the Si-measurements this maximum thermal fluence is calculated to be $8.3 \times 10^{26}$ n/m$^2$ for material from the mid-center positions of the West wall. The underestimation of about 10 % is quite reasonable having regard for the uncertainties in the original calculations which had to take into account neutron history over a period of more than 20 years. Using the maximum thermal fluence of $8.3 \times 10^{26}$ n/m$^2$ as a reference, the twenty specimens cover a range of 20 % to 90 % of the maximum thermal neutron exposure.
Hardness measurements and tension tests showed irradiation hardening and an associated reduction of ductility. The Superficial Rockwell hardness number increased by a maximum of about 35 points HR$_{15N}$. The measured hardness numbers ranged from 51 to 67 HR$_{15N}$.

The tension tests showed yield strength values ranging from 380 MPa to 590 MPa with corresponding Ultimate Tensile Strength (UTS) data ranging from 430 MPa to 610 MPa and total elongation values of 4.1% strain to 1.5% strain respectively. The data for the West wall (thermal to fast flux ratio between 1.0 and 1.5) fit very well with data from accelerated irradiations in the period 1970-1981 (same ratio of about 1.0) for the surveillance program of the old HFR vessel, reported by Lybrink (5).

In addition to the hardening effect from the thermal neutrons, the thermal-to-fast neutron flux density ratio plays a role. This effect is demonstrated in Table 3 where data are listed for West wall and North wall locations. There is an equal Si-content of about 1.37 weight percentage (thermal neutron fluence of $5.5 \times 10^{26}$ n/m$^2$) but with thermal to fast flux ratios of 1.0 and 4.8 respectively.

Table 3. Effect of neutron flux density ratio (th/f-ratio) on radiation hardening

<table>
<thead>
<tr>
<th>Specimen location</th>
<th>Si-wt-%</th>
<th>th/f-ratio</th>
<th>hardness HR$_{15N}$</th>
<th>yield strength MPa</th>
<th>UTS MPa</th>
<th>uniform % elongation</th>
<th>total % elongation</th>
</tr>
</thead>
<tbody>
<tr>
<td>West wall</td>
<td>1.36</td>
<td>1.0</td>
<td>62.6</td>
<td>506</td>
<td>526</td>
<td>1.2</td>
<td>1.8</td>
</tr>
<tr>
<td>North wall</td>
<td>1.38</td>
<td>4.8</td>
<td>66.6</td>
<td>573</td>
<td>592</td>
<td>0.5</td>
<td>1.5</td>
</tr>
</tbody>
</table>

The influence is clear for the hardness and the tensile strength values. Due to the low ductility values, the thermal-to-fast flux ratio effect is less obvious on total elongation but uniform elongation is clearly affected. Increased strengthening with a higher thermal-to-fast flux ratio has been reported previously by Farrell (6). The measured effect is larger than predicted by Lybrink in ref. 4.
The results from the fracture toughness experiments are listed in Table 4. The table gives the maximum failure load ($P_m$), the pop-in load ($P_{pop}$), the load corresponding to an effective crack extension of 2 percent (load at 5% deviation from linearity, $P_5$) and the conditional fracture toughness load ($P_Q$) together with the calculated value of the conditional fracture toughness ($K_{IQ}$). Further the qualification of the $K_{IQ}$-values according to E399 or B645 is given in this table.

Table 4. Results from fracture toughness experiments

<table>
<thead>
<tr>
<th>No.*</th>
<th>$P_m$</th>
<th>$P_{pop}$</th>
<th>$P_5$</th>
<th>$P_Q$</th>
<th>$P_m/P_Q$ ratio</th>
<th>$K_{IQ}$</th>
<th>Qualification</th>
</tr>
</thead>
<tbody>
<tr>
<td>N11</td>
<td>4.10</td>
<td>3.95</td>
<td>3.95</td>
<td>3.95</td>
<td>1.04</td>
<td>25.3</td>
<td>M</td>
</tr>
<tr>
<td>N12</td>
<td>4.50</td>
<td>4.30</td>
<td>4.30</td>
<td>4.30</td>
<td>1.05</td>
<td>22.0</td>
<td>V</td>
</tr>
<tr>
<td>N13</td>
<td>4.30</td>
<td>4.20</td>
<td>4.20</td>
<td>4.20</td>
<td>1.02</td>
<td>24.9</td>
<td>M</td>
</tr>
<tr>
<td>N14</td>
<td>10.75</td>
<td>9.80</td>
<td>9.70</td>
<td>9.80</td>
<td>1.10</td>
<td>25.3</td>
<td>unvalid : a/w = 0.31</td>
</tr>
<tr>
<td>N15</td>
<td>3.90</td>
<td>3.70</td>
<td>3.70</td>
<td>3.70</td>
<td>1.05</td>
<td>23.9</td>
<td>M</td>
</tr>
<tr>
<td>N18</td>
<td>5.55</td>
<td>5.25</td>
<td>5.20</td>
<td>5.25</td>
<td>1.06</td>
<td>24.0</td>
<td>V</td>
</tr>
<tr>
<td>E22</td>
<td>4.70</td>
<td>4.25</td>
<td>4.20</td>
<td>4.25</td>
<td>1.10</td>
<td>27.0</td>
<td>M</td>
</tr>
<tr>
<td>E23</td>
<td>5.45</td>
<td>4.55</td>
<td>4.50</td>
<td>4.55</td>
<td>1.20</td>
<td>28.9</td>
<td>M</td>
</tr>
<tr>
<td>E25</td>
<td>4.85</td>
<td>4.35</td>
<td>4.40</td>
<td>4.40</td>
<td>1.10</td>
<td>27.8</td>
<td>M</td>
</tr>
<tr>
<td>E27</td>
<td>6.15</td>
<td>4.75</td>
<td>4.75</td>
<td>4.75</td>
<td>1.29</td>
<td>30.3</td>
<td>unvalid : B&lt;13 mm, $P_m/P_Q$=1.29</td>
</tr>
<tr>
<td>S28</td>
<td>3.60</td>
<td>3.50</td>
<td>3.50</td>
<td>3.50</td>
<td>1.03</td>
<td>23.0</td>
<td>M</td>
</tr>
<tr>
<td>S30</td>
<td>3.80</td>
<td>3.65</td>
<td>3.60</td>
<td>3.65</td>
<td>1.04</td>
<td>24.2</td>
<td>M</td>
</tr>
<tr>
<td>S31</td>
<td>3.7est.</td>
<td>3.65</td>
<td>3.65</td>
<td>3.65</td>
<td>1.01</td>
<td>24.3</td>
<td>M</td>
</tr>
<tr>
<td>S33</td>
<td>4.30</td>
<td>4.05</td>
<td>4.00</td>
<td>4.05</td>
<td>1.06</td>
<td>26.6</td>
<td>M</td>
</tr>
<tr>
<td>S34</td>
<td>11.55</td>
<td>10.55</td>
<td>10.45</td>
<td>10.55</td>
<td>1.09</td>
<td>26.5</td>
<td>unvalid : a/w = 0.30</td>
</tr>
<tr>
<td>S2-4</td>
<td>6.80</td>
<td>6.60</td>
<td>6.65</td>
<td>6.65</td>
<td>1.02</td>
<td>24.4</td>
<td>V</td>
</tr>
<tr>
<td>W3</td>
<td>5.35</td>
<td>5.30</td>
<td>5.30</td>
<td>5.30</td>
<td>1.01</td>
<td>22.3</td>
<td>M</td>
</tr>
<tr>
<td>W4</td>
<td>5.25</td>
<td>5.20</td>
<td>5.20</td>
<td>5.20</td>
<td>1.01</td>
<td>17.7</td>
<td>M</td>
</tr>
<tr>
<td>W7</td>
<td>3.85</td>
<td>3.85</td>
<td>-</td>
<td>3.85</td>
<td>1.01</td>
<td>16.5</td>
<td>unvalid, $K_{fat}/K_Q$ = 1.1</td>
</tr>
<tr>
<td>W10</td>
<td>6.10</td>
<td>5.85</td>
<td>5.80</td>
<td>5.85</td>
<td>1.04</td>
<td>25.7</td>
<td>M</td>
</tr>
</tbody>
</table>

* N = North wall, E = East wall, S = South wall, W = West wall

** V = valid $K_{IC}$ according to ASTM E399

M = meaningful "$K_{IC}$" according to ASTM B645

est = estimated due to early interruption of test record
The $K_Q$-values range from 16.5 MPa$\sqrt{m}$ to 30.3 MPa$\sqrt{m}$ with Si-content ranging from 0.54 wt-percentage to 1.85 wt-percentage (thermal fluence $1.8 \times 10^{26} \text{ n/m}^2$ to $7.5 \times 10^{26} \text{ n/m}^2$). The data showed a consistent trend of decreasing $K_Q$-values with increasing Si-content, an effect of the thermal-to-fast flux ratio could not be identified.

Most of the specimens did not fulfill the E399 validity requirement on pre-crack length. The ASTM standard practice B645 for fracture toughness testing of aluminium alloys augments the basic test method E399 in the areas of the requirements for valid test results in terms of specimen size and fatigue pre-cracking conditions. If the E399 crack length requirement is not met, B645 requires $0.4 < a/w < 0.6$ to classify the conditional fracture toughness $K_Q$ as a meaningful measure of critical fracture toughness ("$K_{IC}$"). This meaningful measure of the critical fracture toughness is considered to be within 5 to 10 percent of the value of the critical stress intensity factor $K_{IC}$ which would have been obtained if all criteria had been met. The specimens approached the B645 requirement. Two specimens had a very short crack length of about $a/w = 0.3$, nevertheless the $K_{IQ}$ values of 25.4 MPa$\sqrt{m}$ and 26.5 MPa$\sqrt{m}$ fall within the range of the meaningful critical toughness values according B645.

Two experiments did not fulfill the E399 load ratio requirement $P_m/P_Q \leq 1.10$. One did meet the relaxed B645 requirement $P_m/P_Q \leq 1.2$, the $P_m/P_Q$-ratio of the other was 1.29. The latter is the test with the specimen which did also not fulfill the thickness requirement. The lowest $K_Q$-value of 16.5 MPa$\sqrt{m}$ is obtained from an experiment which did not meet the final fatigue loading condition due to the high stress intensity factor range ($\Delta K$) during the final fatigue loading. The $K$-ratio ($K_{max}/K_Q$) was about 1. This specimen showed crack growth instability during the final fatigue crack growth. $K_Q$ is obtained from the crack growth data at instability.

Three results give valid $K_{IC}$ values ranging from 22.0 MPa$\sqrt{m}$ to 24.4 MPa$\sqrt{m}$. Four results are judged invalid, although $K_Q$-values fit with the trend and the band-width of the other results. The remaining 13 results are considered to give meaningful "$K_{IC}$" values, ranging from 17.7 to 28.9 MPa$\sqrt{m}$. The lowermost meaningful "$K_{IC}$" value of 17.7 MPa$\sqrt{m}$ is about 3 times the value of 6 MPa$\sqrt{m}$ which was estimated at shut-down of the old reactor vessel on the basis of the surveillance tensile test results (7). This factor of 3 implies critical crack length values 9 times greater than in the "fitness-for-purpose" assessments of the new HFR vessel compared with the crack length values based on the $K_{IC}$-value of 6 MPa$\sqrt{m}$. 
SUMMARY AND CONCLUSIONS

After more than 20 years neutron exposure of the HFR core-box walls:
- the Rockwell Superficial hardness number increased by up to 35 points $HR_{15N}$
- the 0.2-yield strength increased by up to 500 Mpa
- the ductility is reduced to a minimum value of about 1.5% total elongation with 0.6% uniform elongation
- the irradiation effect on the fatigue crack growth rate is minor for applied $\Delta K$-ranges between 8 MPa$\sqrt{m}$ to 15 MPa$\sqrt{m}$
- the lowermost meaningful measure of the critical plane-strain fracture toughness $K_{IC}$ is 17.7 MPa$\sqrt{m}$
- a maximum of 1.9 weight percentage Si is created by reactions of the thermal neutrons with the aluminium (max. thermal fluence of $8.3 \times 10^{26} n/m^2$, $E < 0.414$ eV)
- the greatest hardening is observed for the North wall
- the thermal to fast neutron flux density ratio has to be taken into account for neutron damage predictions of aluminium-magnesium alloys (5000 series)

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CONVERSION DEL NUCLEO DEL REACTOR TRIGA MARK III
DEL CENTRO NUCLEAR DE MEXICO.

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DIVISION DE INVESTIGACION Y DESARROLLO
GERENCIA DE SISTEMAS NUCLEARES.
KM. 36.5 CARRETERA MEXICO TOLUCA.
SALAZAR, EDO. DE MEXICO.
MEXICO
CONVERSION OF THE CORE OF THE TRIGA MARK III REACTOR
AT THE MEXICAN NUCLEAR CENTRE

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ABSTRACT

It was decided to convert the core of the TRIGA Mark III reactor at the Mexican Nuclear Centre run by the National Nuclear Research Institute because of problems detected during the operation, such as a lack of excess reactivity for operation at nominal power over long periods and difficulties in the maintenance and calibration of the control panel. In order to compensate for the lack of excess reactivity the fuel elements taken to the highest burnup were replaced by fresh fuel elements acquired for this purpose. The latter, however, had a different enrichment, and this necessitated a detailed analysis of the neutronic and thermohydraulic behaviour of the reactor with a view to determining a mixed core configuration which would meet safe operation requirements. In conducting the thermohydraulic analysis, a natural convection coolant flow model was developed to determine coolant velocity and pressure drop patterns within the core. The heat transfer equations were solved and it was found that the hottest fuel element did not attain critical heat flux conditions. In loading this core it was also necessary to analyse procedures and to consider the possible effects of reaching criticality with fuel elements having different enrichments. The loading procedure is described, as is the measurement system and the results obtained. In order to resolve the calibration and maintenance problems, a new, more advanced control panel was designed with conventional and nuclear detection systems and modern components.
CONVERSION DEL NÚCLEO DEL REACTOR TRIGA MARK III
DEL CENTRO NUCLEAR DE MÉXICO.

RESUMEN.

La conversión del núcleo del reactor TRIGA Mark III instalado en el Centro Nuclear de México del Instituto Nacional de Investigaciones Nucleares, es un proceso resultante de problemas detectados durante la utilización del mismo, tales como falta de exceso de reactividad para operación a potencia nominal durante periodos prolongados y dificultades en el mantenimiento y calibración de la consola de control. Para compensar la falta de exceso de reactividad, los elementos combustibles más consumidos fueron sustituidos por elementos combustibles frescos adquiridos con ese propósito, que son sin embargo, de diferente enriquecimiento, lo que hizo necesario un análisis detallado del comportamiento neutrónico y termohidráulico del reactor para proponer una configuración de núcleo mixto que satisficiera necesidades y condiciones de operación segura. Para el análisis termohidráulico se desarrolló un modelo de flujo de refrigerante por convección natural, en el cual se determinaron patrones de velocidad y de caldas de presión del refrigerante en el núcleo. Fueron resueltas las ecuaciones de transferencia de calor y se determinó que el elemento combustible más calente no alcanza condiciones de flujo crítico de calor. Para la carga de este núcleo, también fue necesario analizar el procedimiento y considerar los posibles efectos al alcanzar criticidad con elementos combustibles de diferente enriquecimiento. De este proceso de carga, se describió el método utilizado, así como el sistema de medición y los resultados obtenidos. Para resolver los problemas de calibración y mantenimiento, se consideró oportuno construir una nueva consola de control con sistemas de detección convencionales y nucleares con componentes modernos y diseño de concepto más avanzado.
1. INTRODUCCIÓN.

El reactor TRIGA Mark III del Centro Nuclear de México, es un reactor de experimentación tipo alberca de 1 MW de potencia, que alcanzó criticalidad por primera vez en noviembre de 1968.

El núcleo inicial del reactor estaba formado por elementos combustibles denominados estándar, los cuales contenían un 8.5% en peso de uranio enriquecido al 20% en U235 en una matriz de hidruro de zirconio.

Debido al largo tiempo de utilización, se presentaron problemas de falta de exceso de reactividad para operarlo a potencia nominal durante períodos prolongados, por lo que fue necesaria la sustitución de algunos elementos combustibles. Para resolver este problema se adquirieron elementos combustibles tipo FLIP, que son de alto enriquecimiento, 70% en U235, sin tenerse en número suficiente para un núcleo completo, por lo que fue necesario la formación de un núcleo mixto.

Esto también causó problemas de calibración y mantenimiento de la consola de control, por lo que se consideró necesario diseñar y construir una nueva consola de control con sistemas modernos más confiables de fácil calibración y mantenimiento.

Especiales consideraciones fueron tomadas para la carga de este núcleo mixto por la diferencia en enriquecimiento de elementos combustibles operando en una misma configuración. Se describe el método utilizado, así como el sistema de medición diseñado para la misma.

2. FÍSICA DE REACTORES.

Para el análisis neutrónico del núcleo del reactor se hicieron dos estudios en los que se obtuvieron excesos de reactividad, distribuciones de flujos, distribuciones de potencia y variaciones del factor de multiplicación efectiva en función del quemado. En estos estudios se aplicaron dos métodos de análisis de los parámetros mencionados. En el primero de ellos se usó el código WIMS que usa teoría de transporte con características que se mencionan en seguida. En el otro estudio se utilizó el código EXTERMINATOR-II que aplica teoría de difusión y el cual se describe también a continuación.

De algunas configuraciones mixtas consideradas, fueron analizadas dos de ellas, además de una configuración de un núcleo estándar, con una distribución de elementos combustibles como sigue:
**CONFIGURACIÓN 1:**

<table>
<thead>
<tr>
<th>Anillo</th>
<th>Estándar</th>
<th>Flip</th>
</tr>
</thead>
<tbody>
<tr>
<td>B</td>
<td>6</td>
<td>0</td>
</tr>
<tr>
<td>C</td>
<td>12</td>
<td>0</td>
</tr>
<tr>
<td>D</td>
<td>18</td>
<td>0</td>
</tr>
<tr>
<td>E</td>
<td>24</td>
<td>0</td>
</tr>
<tr>
<td>F</td>
<td>30</td>
<td>0</td>
</tr>
</tbody>
</table>

**Anillo C:**
2 Barras de control, una de ellas con seguidor de combustible

**Anillo D:**
2 Barras de control con seguidor de combustible. (Ver figura 1).

---

**FIG. 1 NÚCLEO ESTANDAR CONFIGURACIÓN 1**
CONFIGURACIÓN 8:

<table>
<thead>
<tr>
<th>Anillo</th>
<th>Elementos Combustibles Estándar</th>
<th>Elementos Combustibles FLIP</th>
</tr>
</thead>
<tbody>
<tr>
<td>Anillo B</td>
<td>6</td>
<td></td>
</tr>
<tr>
<td>Anillo C</td>
<td></td>
<td>10</td>
</tr>
<tr>
<td>Anillo D</td>
<td></td>
<td>16</td>
</tr>
<tr>
<td>Anillo E</td>
<td></td>
<td>4</td>
</tr>
<tr>
<td>Anillo F</td>
<td></td>
<td>30</td>
</tr>
</tbody>
</table>

Anillo C:
2 Barras de control, una de ellas con seguidor de combustible FLIP

Anillo D:
2 Barras de control con seguidor de combustible FLIP. (Ver figura 2).

FIG. 2 NUCLEO MIXTO. CONFIGURACIÓN 8
CONFIGURACIÓN 9:

ELEMENTOS COMBUSTIBLES
ESTANDAR

Anillo B: 6
Anillo C: 10
Anillo D: 16
Anillo E: 24
Anillo F: 30

Anillo C:
2 Barras de control, una de ellas con seguidor de combustible FLIP

Anillo D:
2 Barras de control con seguidor de combustible FLIP.
Datos de composición y dimensiones de los elementos combustibles fueron tomados de la documentación suministrada por el fabricante.

2.1. **Teoría de transporte.** [1]

El código WIMS es utilizado para calcular el núcleo de un reactor como un haz de combustible utilizando teoría de transporte. Es posible incluir efectos de productos de fisión en la operación del reactor, y determinar el quemado de los elementos combustibles. La biblioteca de secciones efícales que se utilizó es de 69 grupos de energía en la cual se encuentran secciones eficaces del Er-167, Er-166 y del hidrógeno de la molécula de hidruro de zirconio, componentes característicos del combustible tipo TRIGA.

Las bibliotecas de datos utilizadas para los cálculos provienen del ENDF/B-IV, proporcionada por el Instituto Joseph Stefan de Lubliana, Yugoslavia utilizadas para los cálculos de reactores tipo TRIGA.

De las opciones que ofrece el código, se usó la opción PIJ que resuelve la ecuación de transporte en multigrupos en dos dimensiones (R, O) por el método de probabilidades de colisión, para obtener distribuciones de flujos y de potencia. Para los cálculos de quemado se usó la opción PERSEUS en la que los anillos del núcleo son homogeneizados y la ecuación de transporte es resuelta en una dimensión, también por el método de probabilidades de colisión. Además se usó la opción DSN que homogeneiza los anillos del núcleo y resuelve la ecuación de transporte por el método de ordenadas discretas para la estimación de fugas de neutrones en las direcciones radial y axial. En todos los cálculos se impuso una condición a la frontera de flujo nulo a 42 cm del centro del reactor, es decir después de 16 cm de agua más 4 cm de reflector (anillo G, compuesto por elementos de grafito). El buckling se utilizó como un parámetro de ajuste que depende de la opción de cálculo utilizada, por lo que se usó el método estándar de cálculo de escapes por teoría de difusión corregida por transporte diagonal. Resultados de cálculos de exceso de reactividad, distribuciones de flujo a potencia y de quemado son presentados en las Tablas I, II, y III.

En las figuras 4 y 5 se muestran los perfiles de flujo obtenidos y en las figuras 5 y 6, la k-efectiva en función del quemado.
### TABLA I. - EXCESO DE REACTIVIDAD DEL NÚCLEO. COMPARACION EXPERIMENTO-CÁLCULO.

<table>
<thead>
<tr>
<th></th>
<th>$\Delta k/k$ (%)</th>
<th>Error (%) ($E-C)/E$</th>
</tr>
</thead>
<tbody>
<tr>
<td>Experimento</td>
<td>4.9</td>
<td>12.2</td>
</tr>
<tr>
<td>Cálculo</td>
<td>5.5</td>
<td></td>
</tr>
</tbody>
</table>

### TABLA II. - FLUJO TÉRMICO RELATIVO POR ANILLO, EXPERIMENTAL Y TEORICO.

<table>
<thead>
<tr>
<th>ANILLO</th>
<th>A</th>
<th>B</th>
<th>C</th>
<th>D</th>
<th>E</th>
<th>F</th>
</tr>
</thead>
<tbody>
<tr>
<td>Experimento</td>
<td>1.0</td>
<td>0.798</td>
<td>0.686</td>
<td>0.627</td>
<td>0.532</td>
<td>0.420</td>
</tr>
<tr>
<td>Teoría</td>
<td>1.0</td>
<td>0.757</td>
<td>0.704</td>
<td>0.621</td>
<td>0.540</td>
<td>0.489</td>
</tr>
</tbody>
</table>

### TABLA III. - EXCESO DE REACTIVIDAD. CONFIGURACIONES 8 Y 9.

<table>
<thead>
<tr>
<th>Config.</th>
<th>K-efectiva</th>
<th>$\Delta k/k$ (%)</th>
</tr>
</thead>
<tbody>
<tr>
<td>8</td>
<td>1.05176</td>
<td>4.92</td>
</tr>
<tr>
<td>9</td>
<td>1.05226</td>
<td>4.97</td>
</tr>
</tbody>
</table>
FIG. 4 FLUJO TÉRMICO RELATIVO POR ANILLO
LA CONFIGURACIÓN 1 ES DE UN NÚCLEO ESTANDAR

• CONFIGURACIÓN 1
• CONFIGURACIÓN 8
X CONFIGURACIÓN 9

FIG. 5 FLUJO RÁPIDO RELATIVO POR ANILLO
LA CONFIGURACIÓN 1 ES DE UN NÚCLEO ESTANDAR

• CONFIGURACIÓN 1
• CONFIGURACIÓN 8
X CONFIGURACIÓN 9
FIG. 6 K-EFECTIVA EN FUNCIÓN DEL QUEMADO

CONFIGURACIÓN 8

FIG. 7 K-EFECTIVA EN FUNCIÓN DEL QUEMADO

CONFIGURACIÓN 9
2.2. **Teoría de difusión.** [2]

El código de cómputo EXTERMINATOR-11 fue utilizado para simular el núcleo mixto en geometría R-Z y obtener el factor de multiplicación efectivo, dato necesario para determinar el buckling geométrico axial, a su vez usado para la simulación detallada del núcleo en geometría R-0. Se utilizó para la simulación una biblioteca de secciones eficaces microscópicas en 7 grupos de energía a 23°C para simulación en frío y a 200°C para simulación a potencia nominal, proporcionada por General Atomics.

El contenido isotópico de los elementos estándar del núcleo mixto, fue calculado con el programa MACTRI.\[1\]

Los resultados obtenidos para las 2 configuraciones analizadas son presentados en las Tablas IV, V, VI, VII, VIII, IX y X a continuación.

<table>
<thead>
<tr>
<th>Tabla IV. - Configuración 8 simulación geometría R-0.</th>
<th>Cálculo de buckling geométrico axial.</th>
</tr>
</thead>
<tbody>
<tr>
<td><strong>Configuración núcleo mixto</strong></td>
<td><strong>Buckling Geométrico Axial calculado (k_{-}\text{efectiva} (\text{cm}^{-2}))</strong></td>
</tr>
<tr>
<td>Combustibles estándar y FLIP nuevos</td>
<td>4.0E-03</td>
</tr>
<tr>
<td>Combustibles estándar quemados y FLIP nuevos</td>
<td>4.0E-03</td>
</tr>
</tbody>
</table>
**TABLA V.- CONFIGURACION 9 SIMULACION GEOMETRIA R-Θ.**
**CALCULO DE BUCKLING GEOMETRICO AXIAL.**

<table>
<thead>
<tr>
<th>CONFIGURACION</th>
<th>NUCLEO MIXTO</th>
<th>BUCKLING GEOMETRICO AXIAL CALCULADO</th>
<th>K-efectiva</th>
<th>Δk/k (%)</th>
</tr>
</thead>
<tbody>
<tr>
<td></td>
<td></td>
<td>(SPG²)</td>
<td>4.07E-03</td>
<td>1.0531</td>
</tr>
<tr>
<td></td>
<td></td>
<td>K-efectiva</td>
<td>4.07E-03</td>
<td>1.0430</td>
</tr>
<tr>
<td>Combustibles</td>
<td></td>
<td>quemados y FLIP nuevos</td>
<td></td>
<td></td>
</tr>
<tr>
<td>estanar y</td>
<td></td>
<td></td>
<td></td>
<td></td>
</tr>
<tr>
<td>FLIP nuevos</td>
<td></td>
<td></td>
<td></td>
<td></td>
</tr>
<tr>
<td></td>
<td></td>
<td></td>
<td></td>
<td></td>
</tr>
</tbody>
</table>

**TABLA VI.- CONFIGURACION 8 Y 9 EXCESOS DE REACTIVIDAD.**

<table>
<thead>
<tr>
<th>NUCLEO MIXTO</th>
<th>CONFIGURACION</th>
<th>k-efectiva</th>
<th>Δk/k (%)</th>
</tr>
</thead>
<tbody>
<tr>
<td>Frio</td>
<td>8</td>
<td>1.0421</td>
<td>4.039</td>
</tr>
<tr>
<td>Caliente</td>
<td>8</td>
<td>1.0223</td>
<td>2.181</td>
</tr>
<tr>
<td>Frio</td>
<td>9</td>
<td>1.0430</td>
<td>4.122</td>
</tr>
<tr>
<td>Caliente</td>
<td>9</td>
<td>1.0216</td>
<td>2.114</td>
</tr>
<tr>
<td>NÚCLEO MIXTO</td>
<td>ARREGLO</td>
<td>k-efectiva</td>
<td>$\Delta k/k$ (%)</td>
</tr>
<tr>
<td>-------------</td>
<td>--------------------------------------</td>
<td>------------</td>
<td>-----------------</td>
</tr>
<tr>
<td>CALIENTE</td>
<td>Venenos en combustible estándar</td>
<td>1.0162</td>
<td>1.594</td>
</tr>
<tr>
<td>CALIENTE</td>
<td>Venenos en combustible FLIP</td>
<td>1.0073</td>
<td>0.7247</td>
</tr>
<tr>
<td>CALIENTE</td>
<td>Venenos en combustibles estándar y en FLIP</td>
<td>1.0039</td>
<td>0.3900</td>
</tr>
</tbody>
</table>

<table>
<thead>
<tr>
<th>NÚCLEO MIXTO</th>
<th>ARREGLO</th>
<th>k-efectiva</th>
<th>$\Delta k/k$ (%)</th>
</tr>
</thead>
<tbody>
<tr>
<td>CALIENTE</td>
<td>Venenos en combustibles estándar</td>
<td>1.0109</td>
<td>1.078</td>
</tr>
<tr>
<td>CALIENTE</td>
<td>Venenos en combustibles FLIP</td>
<td>1.00866</td>
<td>0.8585</td>
</tr>
<tr>
<td>CALIENTE</td>
<td>Venenos en combustibles estándar y en FLIP</td>
<td>1.0000</td>
<td>-----</td>
</tr>
</tbody>
</table>
Puede observarse de los resultados anteriores, que para la configuración 8 el exceso de reactividad disponible al pasar del núcleo frío a potencia nominal con venenos en equilibrio es de 0.390 K/K% y para la configuración 9 no se tiene exceso de reactividad disponible, teniéndose solamente un reactor crítico.

2.3. Teoría de transporte/teoría de difusión. [3]

Con apoyo del Organismo Internacional de Energía Atómica se hizo una revisión de los estudios anteriores, con la asesoría de la Dra. Irena Mele del Instituto Joseph Stefan de Yugoslavia. En esta revisión se utilizaron los códigos WIMS de transporte y TRIGAP de difusión, para generación de constantes de celdas simulación del reactor. Partiendo de los datos técnicos del reactor e información de los elementos combustibles, se determinaron sus propiedades físicas y nucleares.

El núcleo es representado por regiones axiales y radiales homogéneas, dividido en 7 anillos concéntricos. Cada región radial es representada por un anillo y cada posición en el anillo representa una celda unitaria, siendo todas las celdas de igual volumen.

Se calcularon las secciones eficaces macroscópicas usando el código WIMS-D4 para celdas unitarias y para cada uno de los elementos combustibles estándar, FLIP, seguidor estándar y seguidor FLIP, y para las posiciones del núcleo que no contienen material fisionable. En este caso se utilizó la opción de supercelada, que considera una región central que no contiene material fisionable rodeada por seis elementos combustibles y agua.

En ambos casos el cálculo de transporte se hizo para 18 grupos de energía. A continuación se hicieron las correcciones a las secciones eficaces macroscópicas por efecto de temperatura, Xenón y Samario en equilibrio para la celda unitaria y por temperatura solamente para las superceldas. Se procedió entonces a efectuar cálculos de K-efectiva, distribución de flujos y de potencia, y de quemado para configuraciones de núcleo estándar, usando el código TRIGAP. Se analizó el primer núcleo crítico y el primer núcleo de operación del reactor TRIGA Mark III. Dei análisis del núcleo crítico y el primer núcleo de operación del reactor TRIGA Mark III. Del análisis del núcleo crítico se efectuó el ajuste de las fugas axiales, obteniendo el valor del buckling geométrico. Este valor del buckling axial es utilizado para todos los cálculos posteriores. El análisis del primer núcleo de operación, fue realizado para núcleo frío, núcleo a potencia nominal considerando el quemado, sin venenos en equilibrio y núcleo a potencia nominal considerando el quemado con venenos en equilibrio. El valor obtenido para el buckling geométrico axial es de $5.525 \times 10^{-3}$ cm$^{-2}$. Este valor fue obtenido iterativamente de la simulación del primer núcleo crítico.
La Tabla IX muestra los resultados de la simulación del primer núcleo en operación en frío y a potencia nominal.

**TABLA IX.- SIMULACIÓN PRIMER NÚCLEO DE OPERACIÓN EN FRIO Y A POTENCIA NOMINAL.**

<table>
<thead>
<tr>
<th>NÚCLEO EN OPERACIÓN</th>
<th>Kef</th>
<th>EXCESO DE REACTIVIDAD*</th>
<th>POR POTENCIA*</th>
<th>%ΔK/K</th>
</tr>
</thead>
<tbody>
<tr>
<td>FRIO</td>
<td>1.044171</td>
<td>6.043</td>
<td></td>
<td></td>
</tr>
<tr>
<td>CALIENTE</td>
<td>1.019722</td>
<td>2.763</td>
<td>3.280</td>
<td>2.2961</td>
</tr>
</tbody>
</table>

Los resultados obtenidos tienen un error entre 6 y un 10% con respecto a valores experimentales.

En las Tablas X y XI se muestra los resultados de la simulación del núcleo considerando el quemado sin venenos en equilibrio y con venenos en equilibrio respectivamente.

**TABLA X.- CONFIGURACION 1 QUEMADO SIN XENON EN EQUILIBRIO.**

<table>
<thead>
<tr>
<th>PASO DE QUEMADO MWH</th>
<th>Kef</th>
<th>REACTIVIDAD ($)</th>
<th>REACTIVIDAD %ΔK/K</th>
<th>($$)</th>
<th>(%ΔK/K)</th>
</tr>
</thead>
<tbody>
<tr>
<td>0</td>
<td>1.0197223</td>
<td>2.763</td>
<td>1.934085</td>
<td>.-</td>
<td>.-</td>
</tr>
<tr>
<td>50</td>
<td>1.0195900</td>
<td>2.7448</td>
<td>1.92136</td>
<td>0.0182</td>
<td>0.012725</td>
</tr>
<tr>
<td>172.6325</td>
<td>1.0180465</td>
<td>2.5323</td>
<td>1.77266</td>
<td>0.2307</td>
<td>0.0161426</td>
</tr>
<tr>
<td>295.2650</td>
<td>1.0166445</td>
<td>2.3388</td>
<td>1.637199</td>
<td>0.4242</td>
<td>0.296886</td>
</tr>
<tr>
<td>417.8976</td>
<td>1.0152426</td>
<td>2.1448</td>
<td>1.501375</td>
<td>0.6182</td>
<td>0.432711</td>
</tr>
</tbody>
</table>
TABLA XI. CONFIGURACIÓN 1 QUEMADO CON XENON EN EQUILIBRIO.

<table>
<thead>
<tr>
<th>PASO DE QUEMADO MWH</th>
<th>KEF</th>
<th>Reactividad</th>
<th>Reactividad % Δ/K</th>
<th>($)</th>
<th>(%ΔK/K)</th>
</tr>
</thead>
<tbody>
<tr>
<td>0</td>
<td>1.0197223</td>
<td>2.763</td>
<td>1.934085</td>
<td>--</td>
<td>--</td>
</tr>
<tr>
<td>50</td>
<td>1.0002940</td>
<td>0.0419876</td>
<td>0.02939</td>
<td>2.721</td>
<td>1.904</td>
</tr>
<tr>
<td>172.6325</td>
<td>0.998794</td>
<td>-0.1749374</td>
<td>-0.120745</td>
<td>--</td>
<td>--</td>
</tr>
<tr>
<td>295.2650</td>
<td>0.9974329</td>
<td>-0.367672</td>
<td>-0.257370</td>
<td>--</td>
<td>--</td>
</tr>
<tr>
<td>417.8976</td>
<td>0.996071</td>
<td>-0.56339891</td>
<td>-0.394379</td>
<td>--</td>
<td>--</td>
</tr>
</tbody>
</table>

En las figuras 7 Y 8 se muestra el comportamiento del exceso de reactividad en función del quemado de combustible sin venenos y con venenos en equilibrio respectivamente, para la configuración 1.
Actualmente se simula la configuración 9 utilizando este método. Se espera tener resultados próximamente, para proceder a hacer un análisis comparativo de los diferentes métodos utilizados así como de mediciones experimentales.

3. TERMOHIDRAULICA.

En el reactor TRIGA la remoción del calor generado en las barras de combustible se logra por convección natural del agua de la alberca como único modo de refrigeración del núcleo. Si bien existen programas de computadora con los cuales puede efectuarse el análisis termohidráulico del núcleo, se decidió desarrollar un modelo específico para efectuar una estimación de los campos de velocidades, temperaturas y presiones establecidos por convección natural, del cual se hace una descripción y se presentan resultados, para la configuración 8.
3.1. **Modelo.**

Tomando como base el trabajo descrito en la referencia [5] el modelo es un esquema de volúmenes o "nodos" y "uniones".

En estos esquemas el volumen del problema se subdivide en volúmenes pequeños para los cuales se plantean y resuelven las ecuaciones de balance de masa y energía. Los volúmenes están interconectados entre si por uniones que son ductos en los cuales se plantean y resuelven las ecuaciones de balance de momento. Con estos elementos es posible simular redes hidráulicas complejas, incluyendo los efectos de convección natural.

3.2. **Ecuaciones generales.**

Para los volúmenes o uniones el balance de masa en condiciones estacionarias queda expresado como:

\[
\sum_{k} \rho_{k} V_{k} A_{k} - \sum_{k} \rho_{k} V_{k} A_{k} = 0 \quad i = 1, \ldots, N
\]

\(k(i_{e})\) \(k(i_{s})\)

en donde:

- \(k\) : índice de sumatoria sobre las uniones.
- \(i_{e}\) : uniones que conducen fluido hacia el nodo.
- \(i_{s}\) : uniones que extraen fluido del nodo.

De manera similar para el balance de energía tenemos:

\[
Q_{i} + \sum_{k} \rho_{k} V_{k} A_{k} h_{k} - \sum_{k} \rho_{k} V_{k} A_{k} h_{k} = 0
\]

\(k(i_{e})\) \(k(i_{s})\)

en donde:

- \(Q_{i}\) : calor generado en el volumen \(i\)
- \(h_{k}\) : entalpía de la unión.
La ecuación de momento planteada para cada unión que conecta a los volúmenes (I) y (J):

\[ p_I - p_J = \left( \rho_k (K + f - \frac{L}{D} - \frac{\rho_k}{\rho_I}) \right)^2 + \frac{\rho_k}{\rho_j} \left( \frac{A_k}{A_j} \right)^2 \alpha_k^2 - \rho_k g (Z_j - Z_i) \]

en donde: 
- \( p_I \): Presión del nodo inicial.
- \( p_J \): Presión del nodo final.
- \( f \): Factor de fricción.
- \( L \): Longitud del ducto.
- \( D \): Diámetro equivalente.
- \( K \): Factor de pérdida.
- \( Z_I \): Altura del nodo i.
- \( Z_J \): Altura del nodo j.
- \( A_I \): Sección transversal del nodo i.
- \( A_J \): Sección transversal del nodo j.
- \( \alpha_k \): Velocidad del fluido.

Para resolver el sistema de ecuaciones derivado de los balances de masa, momento y energía, se emplea un método iterativo partiendo de condiciones iniciales propuestas para luego modificárlas sucesivamente hasta satisfacer todas las ecuaciones.

3.3. Parámetros del modelo del reactor TRIGA.

Existen diferentes posibilidades para discretizar el volumen del reactor TRIGA en nodos y uniones. Sin embargo, para reducir el número de incógnitas del problema, algunas hipótesis de simplificación fueron introducidas.

a).- Las variables serán función de las posiciones radial y axial: esto es se despreciarán las variaciones en la coordenada azimutal.
b).- El coeficiente de pérdida por fricción que aparece en la ecuación de momento se toma como una constante de valor \( f = 0.002 \)

c).- Se desprecian los efectos de flujo a dos fases.

d).- Las barras de combustible se suponen todas de un mismo radio \( R_b \); dispuestas en 6 anillos concéntricos donde en cada uno hay 6, 12, 18, 24, 30 y 36 barras respectivamente y en cada anillo se supone que la colocación de las barras es simétrica.

Con estas hipótesis el modelo hidráulico del reactor puede suponerse compuesto por 7 canales verticales; cada canal es el volumen anular comprendido entre dos anillos consecutivos de barras de combustible (esto resulta de la hipótesis a), y el área transversal será aquella comprendida entre las dos poligonales que unen los centros de las barras en cada anillo, menos el área comprendida por las secciones de las mismas barras.

3.4. Calor generado en los canales.

Se supone la generación axial de potencia con un perfil de coseno truncado de tal suerte que el ángulo de truncamiento \( \Theta_c \) es tal que se cumple con la propiedad.

\[
\frac{P_{\text{max}}}{P_{\text{prom}}} = 1.25
\]

\( P_{\text{max}} \): Amplitud máxima del coseno.

\( P_{\text{prom}} \): Valor promedio

El perfil axial de potencia se expresa entonces como:

\[
P(\Theta) = P_{\text{max}} \cos(\Theta)
\]

\[
\frac{\Theta_c}{\text{sen} \Theta_c} = 1.25
\]

De esta manera \( \Theta \) tiene un valor de: \( \Theta_c = 1.1311025 \)

Y la distribución radial de potencia se incluye en forma de un factor de potencia radial promedio por anillo.
3.5. Resultados del análisis para el reactor TRIGA.

3.5.1. Datos empleados en la solución del modelo propuesto para la configuración 8.

Radio del dedal central: 1.68 cm
Radio de las barras de comb.: 1.8669 cm
Radio de la coraza: 27.0665 cm
Número de anillos de comb.: 6
Area del orificio superior en cada barra: 6.6515 cm²

<table>
<thead>
<tr>
<th>Anillo</th>
<th>No. Barras</th>
<th>Radio del Centro de las barras al centro del dedal central</th>
<th>Factor de Potencia Radial</th>
</tr>
</thead>
<tbody>
<tr>
<td>B</td>
<td>6</td>
<td>4.0513 cm</td>
<td>0.097655</td>
</tr>
<tr>
<td>C</td>
<td>12</td>
<td>7.9807 cm</td>
<td>0.188547</td>
</tr>
<tr>
<td>D</td>
<td>18</td>
<td>12.0142 cm</td>
<td>0.243903</td>
</tr>
<tr>
<td>E</td>
<td>24</td>
<td>15.9156 cm</td>
<td>0.253492</td>
</tr>
<tr>
<td>F</td>
<td>30</td>
<td>19.8882 cm</td>
<td>0.216404</td>
</tr>
<tr>
<td>G</td>
<td>36</td>
<td>23.8608 cm</td>
<td>0.000000</td>
</tr>
</tbody>
</table>

Potencia Total: 1076.446 KW(th)

Profundidad de la tapa superior del núcleo a la superficie de la alberca: 6.3332 m

Como condición de frontera se ha supuesto que el agua de la alberca está a una temperatura de 25°C.

Los resultados obtenidos con el programa de computadora FLUJ3 se muestran en las figuras 10, 11 y 12 para las temperaturas, velocidades verticales y presiones respectivamente.

Se hace la anotación de que a las presiones mostradas en la figura 11 hay que sumarles el valor que tenga la presión atmosférica.
FIG. 10 DISTRIBUCIÓN DE TEMPERATURAS [°C]

<table>
<thead>
<tr>
<th>53.43</th>
<th>52.86</th>
<th>52.30</th>
<th>51.99</th>
<th>51.55</th>
<th>51.22</th>
<th>51.09</th>
</tr>
</thead>
<tbody>
<tr>
<td>53.34</td>
<td>52.60</td>
<td>52.11</td>
<td>51.85</td>
<td>51.23</td>
<td>50.85</td>
<td>50.62</td>
</tr>
<tr>
<td>53.30</td>
<td>52.79</td>
<td>52.06</td>
<td>51.84</td>
<td>51.06</td>
<td>50.67</td>
<td>50.56</td>
</tr>
<tr>
<td>51.33</td>
<td>50.83</td>
<td>50.05</td>
<td>49.86</td>
<td>49.03</td>
<td>48.51</td>
<td>48.41</td>
</tr>
<tr>
<td>48.47</td>
<td>47.96</td>
<td>47.09</td>
<td>46.92</td>
<td>46.04</td>
<td>45.42</td>
<td>45.40</td>
</tr>
<tr>
<td>44.95</td>
<td>44.46</td>
<td>43.54</td>
<td>43.31</td>
<td>42.45</td>
<td>41.51</td>
<td>41.28</td>
</tr>
<tr>
<td>40.97</td>
<td>40.54</td>
<td>39.68</td>
<td>39.37</td>
<td>38.51</td>
<td>37.82</td>
<td>37.62</td>
</tr>
<tr>
<td>36.80</td>
<td>36.44</td>
<td>35.68</td>
<td>35.36</td>
<td>34.35</td>
<td>33.97</td>
<td>33.71</td>
</tr>
<tr>
<td>32.78</td>
<td>32.50</td>
<td>31.93</td>
<td>31.59</td>
<td>30.61</td>
<td>29.45</td>
<td>29.49</td>
</tr>
<tr>
<td>29.18</td>
<td>28.99</td>
<td>28.64</td>
<td>28.41</td>
<td>27.72</td>
<td>26.83</td>
<td>26.05</td>
</tr>
<tr>
<td>26.16</td>
<td>26.04</td>
<td>25.93</td>
<td>25.84</td>
<td>25.75</td>
<td>25.68</td>
<td>25.59</td>
</tr>
<tr>
<td>26.16</td>
<td>26.03</td>
<td>25.91</td>
<td>25.78</td>
<td>25.66</td>
<td>25.53</td>
<td>25.41</td>
</tr>
</tbody>
</table>

FIG. 11 DISTRIBUCIÓN DE VELOCIDADES [m/s]

<table>
<thead>
<tr>
<th>0.128</th>
<th>0.101</th>
<th>0.091</th>
<th>0.076</th>
<th>0.064</th>
<th>0.047</th>
</tr>
</thead>
<tbody>
<tr>
<td>0.263</td>
<td>0.184</td>
<td>0.175</td>
<td>0.157</td>
<td>0.132</td>
<td>0.068</td>
</tr>
<tr>
<td>0.269</td>
<td>0.235</td>
<td>0.220</td>
<td>0.200</td>
<td>0.160</td>
<td>0.078</td>
</tr>
<tr>
<td>0.282</td>
<td>0.262</td>
<td>0.237</td>
<td>0.210</td>
<td>0.147</td>
<td>0.059</td>
</tr>
<tr>
<td>0.280</td>
<td>0.270</td>
<td>0.234</td>
<td>0.206</td>
<td>0.138</td>
<td>0.072</td>
</tr>
<tr>
<td>0.282</td>
<td>0.285</td>
<td>0.244</td>
<td>0.218</td>
<td>0.160</td>
<td>0.069</td>
</tr>
<tr>
<td>0.273</td>
<td>0.293</td>
<td>0.248</td>
<td>0.225</td>
<td>0.163</td>
<td>0.062</td>
</tr>
<tr>
<td>0.256</td>
<td>0.289</td>
<td>0.241</td>
<td>0.220</td>
<td>0.146</td>
<td>0.052</td>
</tr>
<tr>
<td>0.245</td>
<td>0.291</td>
<td>0.245</td>
<td>0.219</td>
<td>0.142</td>
<td>0.062</td>
</tr>
<tr>
<td>0.217</td>
<td>0.299</td>
<td>0.249</td>
<td>0.216</td>
<td>0.136</td>
<td>0.053</td>
</tr>
<tr>
<td>0.169</td>
<td>0.263</td>
<td>0.216</td>
<td>0.187</td>
<td>0.130</td>
<td>0.063</td>
</tr>
<tr>
<td>0.090</td>
<td>0.137</td>
<td>0.127</td>
<td>0.141</td>
<td>0.134</td>
<td>0.134</td>
</tr>
</tbody>
</table>
Finalmente se toma la barra de combustible que genera más potencia para calcular la temperatura sobre la superficie de la camisa y el cociente de flujo crítico de calor. Las suposiciones que se han hecho para tal efecto son las siguientes:

Potencia en la barra: 22.1150 KW

Posición de la barra: Anillo C

Se utilizan las correlaciones de McAdams y Bernath [7] para los cálculos de temperatura de superficie y flujo crítico de calor respectivamente.

<table>
<thead>
<tr>
<th>FIG. 12 DISTRIBUCIÓN DE PRESIONES [KPa]</th>
</tr>
</thead>
<tbody>
<tr>
<td>61.94665</td>
</tr>
<tr>
<td>63.49561 63.49561 63.49561 63.49561 63.49561 63.49561 63.49561</td>
</tr>
<tr>
<td>63.95734 63.95734 63.95734 63.95734 63.95734 63.95734 63.95734</td>
</tr>
<tr>
<td>64.41960 64.41960 64.41960 64.41960 64.41960 64.41960 64.41960</td>
</tr>
<tr>
<td>64.88256 64.88256 64.88256 64.88256 64.88256 64.88256 64.88256</td>
</tr>
<tr>
<td>65.34622 65.34622 65.34622 65.34622 65.34622 65.34622 65.34623</td>
</tr>
<tr>
<td>65.81055 65.81055 65.81055 65.81055 65.81055 65.81055 65.81055</td>
</tr>
<tr>
<td>66.27551 66.27551 66.27551 66.27551 66.27551 66.27551 66.27551</td>
</tr>
<tr>
<td>67.34099 67.34099 67.34099 67.34099 67.34099 67.34099 67.34099</td>
</tr>
<tr>
<td>67.80629 68.08629 68.08629 68.08629 68.08629 68.08629 68.08629</td>
</tr>
<tr>
<td>67.36303</td>
</tr>
<tr>
<td>68.10249</td>
</tr>
</tbody>
</table>
Los resultados son mostrados en la siguiente tabla.

<table>
<thead>
<tr>
<th>SEGMENTO (de abajo hacia arriba)</th>
<th>FLUJO DE CALOR (W/cm²)</th>
<th>T (°C)</th>
<th>q&quot;crit/q&quot; (Bernath)</th>
</tr>
</thead>
<tbody>
<tr>
<td>1</td>
<td>0.0</td>
<td>0.0</td>
<td>–</td>
</tr>
<tr>
<td>2</td>
<td>0.0</td>
<td>0.0</td>
<td>–</td>
</tr>
<tr>
<td>3</td>
<td>33.84</td>
<td>21.92</td>
<td>13.57</td>
</tr>
<tr>
<td>4</td>
<td>46.87</td>
<td>23.85</td>
<td>9.515</td>
</tr>
<tr>
<td>5</td>
<td>56.12</td>
<td>24.99</td>
<td>7.679</td>
</tr>
<tr>
<td>6</td>
<td>61.03</td>
<td>25.54</td>
<td>6.8031</td>
</tr>
<tr>
<td>7</td>
<td>61.03</td>
<td>25.54</td>
<td>6.557</td>
</tr>
<tr>
<td>8</td>
<td>56.12</td>
<td>24.99</td>
<td>6.892</td>
</tr>
<tr>
<td>9</td>
<td>46.87</td>
<td>23.85</td>
<td>8.017</td>
</tr>
<tr>
<td>10</td>
<td>33.84</td>
<td>21.92</td>
<td>10.883</td>
</tr>
<tr>
<td>11</td>
<td>0.0</td>
<td>0.0</td>
<td>–</td>
</tr>
<tr>
<td>12</td>
<td>0.0</td>
<td>0.0</td>
<td>–</td>
</tr>
</tbody>
</table>

Usando el valor de referencia para condiciones de ebullición en alberca; el valor mínimo para el cociente de flujo crítico de calor es de 2.772

De estos cálculos se concluye que en condiciones de operación estacionarias existe suficiente margen, sin que se presente la condición de flujo crítico de calor.

4. INSTRUMENTACIÓN Y CONTROL.

Desde que el reactor entró en operación en noviembre de 1968, hasta la fecha no se ha modificado ninguno de sus componentes presentándose ya un grave problema en la calibración y sustitución de partes de respuesto debido a que en su mayoría son de importación y algunas de ellas ya han sido descontinuadas, esto provoca que haya poca disponibilidad del reactor.

4.1. Problemática de la instrumentación original.

La problemática de la instrumentación del TRIGA se reduce en los siguientes puntos:

1.- Componentes descontinuados.
2.- Alto costo de importación de partes y servicio.
3.- Paros frecuentes por fallas en la instrumentación.
4.- Inestabilidad creciente de los sistemas.
5.- Dificultad para realizar la calibración.
Al desarrollarse nuevos sistemas de instrumentación y control para el reactor TRIGA, los puntos anteriores fueron superados y a su vez se obtuvo experiencia en el diseño y construcción de los mismos.

4.2. Descripción de la instrumentación.

La instrumentación del reactor TRIGA se puede dividir en grupos de acuerdo a su función. Así tenemos:

4.2.1. Sistemas nucleares.

Son los encargados de monitorear directamente el nivel de flujo neutrónico dentro del núcleo del reactor y a su vez generar las señales de control y seguridad para casos de incrementos abruptos de potencia que pudieran poner en riesgo la integridad del reactor y del personal.

La instrumentación nuclear diseñada y construida está compuesta por los siguientes sistemas:

- Sistema de relación de conteo.
- El sistema de potencia lineal.
- Sistema de potencia logarítmica y derivador de periodo.
- Sistema de porcentaje de potencia.

4.2.2. Sistemas convencionales.

Los sistemas convencionales monitorean la conductividad y temperatura del moderador y la temperatura del combustible, para lo cual se usan celdas de conductividad y termopares tipo J para el moderador y termopares tipo K para el elemento combustible instrumentado.

4.2.3. Sistemas de control.

Estos sistemas determinan el estado del reactor, verifican las acciones del operador y en su caso inhibir alguna condición que ponga el riesgo la seguridad de la instalación y del personal.

4.2.4. Sistemas de respaldo.

Suministan y vigilan los diferentes voltajes de alimentación requeridos en el resto de la consola.

4.2.5. Sistemas auxiliares.
4.3. **Construcción de la consola.**

El proyecto consola de control del reactor TRIGA tiene los siguientes objetivos.

a) Diseño y construcción de todos los sistemas que integran la consola de control del reactor TRIGA.

b) Elaboración de la documentación necesaria y registro de todos los diseños generados para la instrumentación de la consola de control.

c) Ensamblado de la consola, realización de pruebas e instalación en el reactor TRIGA Mark III.

d) Licenciamiento.

De los cuales actualmente se encuentra en la fase de pruebas y se espera ponerla en operación durante 1990.

5. **CARGA DEL NÚCLEO.**

Se presenta a continuación una sinopsis de los factores que se tomaron en cuenta para la carga del núcleo y los pasos que se siguieron para lograrlo.

5.1. **Factores de seguridad.**

Dentro de los factores fundamentales para la seguridad, se consideró que los sistemas y dispositivos de control y seguridad han de estar en condiciones de reducir la reactividad con más rapidez de la que ésta pueda insertarse por operación normal; los sistemas de detección deberán ser suficientemente sensibles para detectar las alteraciones del flujo neutrónico en el núcleo y deberá iniciarse una acción rápida y eficaz por parte de los dispositivos de seguridad siempre que sea necesario, para evitar que el núcleo supere los límites de seguridad.

Para determinar la naturaleza y amplitud de las medidas a adoptar, se consideró la probabilidad de un incidente o evento insólito, así como sus posibles consecuencias.

5.2. **Instrumentación de carga.**

La instrumentación para la carga de elementos combustibles al núcleo del reactor, está colocada en el extremo del cuarto de exposición. Esta instrumentación es capaz de funcionar adecuadamente para bajos niveles de flujo neutrónico, de discriminar la radiación gamma y obtener una respuesta representativa del flujo neutrónico durante la carga. Una
cámara de ionización compensada está conectada a un canal lineal de lectura digital, éste canal está provisto de una señal de corte, la cual acciona las barras de control si durante el proceso de carga se generase un transitorio.

Una cámara de fisión conectada a un sistema de conteo y calibrada para determinar la potencia desde el nivel subcrítico hasta .5 KW, permite tener una referencia durante el proceso de carga. Este canal se utiliza también para seguir los transitorios generados durante la calibración de barras de control o determinación del valor en reactividad de elementos combustibles. Durante los transitorios este canal se conecta a una microcomputadora para captar, almacenar y procesar los datos. El resultado final de este proceso es el valor en reactividad de un combustible o la curva de calibración de una barra de control.

Tres detectores de trifluoruro de boro y su instrumentación asociada, permiten obtener la razón de conteo durante el proceso de carga. Un medidor de relación exhibe en todo momento el valor de la razón de conteo y proporciona un medio para determinar la estabilidad del flujo neutrónico. Además de este dispositivo se cuenta con un monitor que proporciona una señal audible durante todo el proceso de carga. Ver figuras 13 y 14.

Finalmente una microcomputadora proporciona el soporte para el procesamiento de datos durante la carga, análisis del comportamiento de los detectores y su sistema electrónico asociado, predicciones de masa crítica y el número de combustibles que deberán cargarse en cada etapa.

Una gráfica del inverso de la multiplicación (o de la intensidad de flujo por unidad de tiempo), es utilizada al realizarse la aproximación a crítica, y en ella se predice el punto en que ésta se alcanzará.

La gráfica es de especial interés en núcleos del tipo de reticulado a los cuales se añade combustible gradualmente en presencia de moderador y reflector. El factor de multiplicación del ensamble es mantenido en un valor seguro con las barras de control entre una adición y otra de combustible. En este caso, la curva mencionada es una ayuda muy útil para determinar la cantidad de combustible en que debe incrementarse el conjunto. En algunas instalaciones se sigue la norma de aumentar como máximo la mitad de la cantidad de combustible que según las predicciones se precisa para alcanzar la criticidad. Ver fig. 15.

5.3. Procedimiento de carga.

5.3.1. Objetivo.
FIG. 13 SISTEMA DE DETECCION CON BF3

FIG. 14 SISTEMA DE DETECCION CON CAMARA DE FISON
Seguimiento de la carga del núcleo del reactor TRIGA Mark III.

Fig. 15 Aproximación a criticidad del núcleo mixto.
Realizar la carga de un núcleo mixto para el reactor TRIGA Mark III, con elementos estándar irradiados y elementos combustible tipo FLIP nuevos.

Estimar la masa crítica, diseñar una configuración cuyo exceso pueda ser compensado con la barra reguladora, alcanzar criticidad y diseñar una nueva configuración cuyo exceso tenga un valor igual o un poco mayor que el de la barra reguladora.

Determinar el margen de apagado de los dispositivos de control.

5.3.2. Alcance.

El procedimiento es aplicable cuando el núcleo del reactor se encuentra en la posición de cuarto de exposición, para obtener una configuración crítica del núcleo mixto y un exceso de reactividad que no sea mayor de un 15% el valor de la barra reguladora.

5.3.3. Proceso de carga.

Se realizan pruebas e inspecciones en el reactor conforme a las listas de verificación diaria y semanal.

Se retiran los dispositivos y detectores de la parte superior del ensamble.

Se verifica la instrumentación de carga y se realizan conteos de fondo y fuente, colocando definitivamente la fuente en el centro geométrico del ensamble.

Se verifica el apagado seguro del reactor.

Los datos y resultados se registran en las bitácoras del grupo de operación y del grupo de cálculo y mediciones.

Se inició la carga, insertando uno a uno los seis combustibles correspondientes al anillo B.

Se extrajeron en las barras de control y se determinó la razón de conteo, se calculó el factor multiplicación y una vez terminado este paso se aplica el sistema de apagado seguro del reactor.

Se repitieron los pasos anteriores cargando un número predeterminado de combustibles cada vez, hasta completar el anillo D, vigilando siempre la razón de conteo.

En cada paso fueron graficados los valores del inverso de la multiplicación contra el número de combustibles en el núcleo.
Especial atención al análisis de los datos y valores de las predicciones, estos resultados se presentan en la gráfica de la fig. 15.

El proceso de carga con seguimiento de la multiplicación se dio por terminado para una multiplicación de 98.8 y 59 elementos combustibles en el núcleo del reactor.

En este punto se tomaron en consideración, los valores de los combustibles calculados previamente, los valores de barra, la posible interferencia de la instrumentación y la predicción de 68.2 combustibles para conseguir criticidad.

Se diseñó una configuración con 75 elementos combustibles en total, de los cuales 26 de ellos fueron elementos tipo FLIP, se colocó la instrumentación de operación, se realizaron las comprobaciones de los diferentes sistemas y dispositivos de control y seguridad.

Finalmente se procedió a la extracción de las barras de control una a la vez, la de transitorios, la de seguridad, y la fina sin conseguir criticidad. La parte final de este proceso correspondió a la extracción cuidadosa de la barra reguladora, limitándola a 50 posiciones por incremento y después de la posición 150 se observa el transitorio, la posición final de la barra reguladora, fue 135. Las limitaciones están relacionadas con el valor en reactividad que un operador puede manejar sin problemas, y que corresponde a periodos mayores que diez segundos.

6. CONCLUSIONES.

La metodología desarrollada hasta ahora para el análisis del núcleo mixto del reactor TRIGA Mark III, con el apoyo del OIEA, será utilizada para el diseño de nuevas herramientas para la operación del reactor y se analizará también la posibilidad de aplicarlas al diseño de otro tipo de reactores de investigación de concepto avanzado.

REFERENCIAS.


NEW IRRADIATION FACILITIES IN THE OAK RIDGE HIGH
FLUX ISOTOPE REACTOR (HFIR)

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NEW IRRADIATION FACILITIES IN THE OAK RIDGE HIGH
FLUX ISOPOE REACTOR (HFIR)\textsuperscript{a}

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ABSTRACT

Modifications have been made to the High Flux Isotope Reactor (HFIR) which permit the operation of instrumented irradiation capsules in the target region, and more and larger capsules in the removable beryllium region. As many as two instrumented target capsules can now be accommodated and positions for up to eight 46-mm-diam instrumented capsules are now available in the removable beryllium region. One instrumented target capsule has already been irradiated and new capsules have been prepared for irradiation in the removable beryllium region.

1. INTRODUCTION

The High Flux Isotope Reactor (HFIR) is a pressurized, light-water-cooled, beryllium-reflected, 85-MW reactor. The HFIR was designed for the production of isotopes, particularly transuranium isotopes. This production requires high thermal and epithermal fluxes; indeed, the HFIR target region (the cylindrical space inside the two concentric annular fuel elements) has the highest steady-state thermal neutron flux in the world. The relatively high reactor power and power density leads to a high fast neutron flux near the core, so that the HFIR is also used for materials irradiation experiments.

While the HFIR had outstanding neutronics characteristics for materials irradiations, some relatively minor aspects of its original mechanical design severely limited its usefulness for that purpose. In 1984, an ad hoc committee was established at the Oak Ridge National Laboratory (ORNL) to "... consider and recommend changes and improvements to the Laboratory's facilities for materials irradiation testing." The committee's report \cite{1} included recommendations for certain modifications to the HFIR that would significantly enhance the number and value of materials irradiation experiments that could be accommodated by the reactor.

The basic improvements that were needed to provide better materials irradiation facilities at the HFIR were in two areas. The highest flux positions in the target region could not be instrumented, and the removable beryllium (RB) positions were few and much smaller than those of general purpose reactors. These deficiencies have been remedied through the HFIR Irradiation Facilities Improvement (HIFI) Project which has provided two instrumented target region facilities and larger and additional RB irradiation positions with straight-line access penetrations through the pressure vessel.

\textsuperscript{a}Research sponsored by U.S. Department of Energy under contract No. DE-AC05-84OR21400 with Martin Marietta Energy Systems, Inc.
2. DESCRIPTION OF FACILITIES

A general arrangement of the new materials irradiation facilities with typical instrumented target and RB capsules in place is shown in Fig. 1. The characteristics of these facilities are presented in Table 1.

Providing instrumented target facilities required newly designed and fabricated components from the bottom to the top of the reactor "stack." These components included a fuel grid, target holder, target tower, target hole plug, quick-access hatch, rabbit facility U-bend, and several in-pool tools for removing and replacing components. The target tower extends upward from the target region to a quick-access hatch and target hole plug in the pressure vessel lid. The tower houses three guide tubes - one for the hydraulic rabbit facility and the other two for the instrumented target facilities.

With these components modified, at least two small target capsules of 16-mm-diam may be instrumented. The guide tubes in the target tower are large enough such that by occupying up to seven target positions, capsules of up to 25-mm-diam can be accommodated (Fig. 2).

The new RB facilities required a modified design for the reflector, replacing the four 37-mm-diam positions with eight holes, each with a 48-mm diam. This change increased the total experimental volume available within the removable beryllium by a factor of greater than 3. These new positions are referred to as the RB Star (RB*) facilities.

In addition, several components above the beryllium and the core were modified to provide straight-line access to all eight of the RB* positions. The straight-line access permits rotation and vertical relocation of irradiation capsules during the course of an experiment and facilitates experiment interchangeability.

Recording and control equipment is in place to operate two singly-contained capsules and two doubly-contained capsules, with space readily available to expand the equipment for the operation of a total of eight fully instrumented capsules.

3. TYPICAL EXPERIMENTS

Significant funding for the necessary modifications was provided by the Magnetic Fusion Energy (MFE) program. The first instrumented target capsule was the target temperature test (TTT) capsule. It was a part of the US/Japan fusion materials program and was irradiated to determine more accurately the probable temperature in the uninstrumented target capsules previously irradiated as part of that program. Two thermocouple array tubes (TCATs), each having seven thermocouple junctions, were used to measure the centerline temperature of mock specimens. The experiment performed well, and revealed (Fig. 3) that the gamma heating decreases much more rapidly at the ends of the capsule than had previously been thought. A general configuration of the TTT capsule is shown in Fig. 4.

The larger reflector positions permit spectral tailored experiments, similar to those previously performed in the Oak Ridge Research Reactor (ORR), to be performed in the HFIR where fluence can be achieved in about half the time. Indeed, MFE specimens irradiated in ORR spectral tailored capsules have been retrieved and are being reencapsulated for continued irradiation in HFIR RB* positions. Toward this end, series of experiments have been designed to
Fig. 1. The High Flux Isotope Reactor (HFIR).

NEW EXPERIMENTAL FACILITIES IN HFIR

- Typical Instrumented Target Capsule
- Typical RB* Capsule
- Quick Access Hatch
- Target Tower
- Fuel Element
- Removable Beryllium
- Shroud Flange
- Shroud and Upper Track Assembly
- Reactor Vessel
Table 1. Characteristics of primary HFIR materials irradiation facilities

<table>
<thead>
<tr>
<th>Characteristics</th>
<th>Irradiation position</th>
</tr>
</thead>
<tbody>
<tr>
<td></td>
<td>RB</td>
</tr>
<tr>
<td>Fast neutron flux, (E &gt; 0.1) MeV ((10^{18} \text{ m}^{-2}\text{s}^{-1}))</td>
<td>6</td>
</tr>
<tr>
<td>Thermal neutron flux ((10^{18} \text{ m}^{-2}\text{s}^{-1}))</td>
<td>13</td>
</tr>
<tr>
<td>Maximum displacements per atom per calendar year, stainless steel</td>
<td>8</td>
</tr>
<tr>
<td>Gamma heating ((\text{W/g SS}))</td>
<td>14</td>
</tr>
<tr>
<td>Typical capsule diameter (mm)</td>
<td>46</td>
</tr>
<tr>
<td>Typical capsule length (mm)</td>
<td>500</td>
</tr>
<tr>
<td>Number of available positions</td>
<td>8&lt;sup&gt;b&lt;/sup&gt;</td>
</tr>
<tr>
<td>Minimum specimen temperature (°C)</td>
<td>60</td>
</tr>
<tr>
<td>Instrumentation</td>
<td>Yes</td>
</tr>
<tr>
<td>Typical fuel cycle length (days)</td>
<td>25</td>
</tr>
</tbody>
</table>

<sup>a</sup>By occupying up to seven positions, 25-mm-diam can be accommodated.

<sup>b</sup>Plus four smaller positions, ~12-mm diam.

<sup>c</sup>A total of 37 target positions exist. The number available depends on the number being used for transuranium isotope production.

<sup>d</sup>Two target positions can accommodate instrumented capsules.
Fig. 2. Instrumented target positions illustrating capability of accommodating 25-mm capsules.
Fig. 3. Heat generation in HFIR target region (data is for 100 MW and should be multiplied by 0.85 to obtain present values).
Fig. 4. General configuration of TTT capsule.

- **TEMPERATURE (°C)**
  - 300
  - 400
  - 500

- **REGIONS**
  - 1
  - 2
  - 3
  - 4
  - 5
  - 6
  - 7
  - 8
  - 9
  - 10
  - 11

- **CONTAINS**
  - 1, 3, 5: Gamma Susceptors
  - 2, 9: Tensile Specimens
  - 4, 7, 8: Fatigue Specimens
  - 6: TEM Disks
  - 10, 11: JP = Tensile Specimen

- **NOTE**
  - Mock specimens are made from heavy wall 316 SS tubing for TCAT to go through.
irradiate up to 250 mechanical property specimens each at temperatures of 60, 200, 330, and 400°C [2]. Around each of these experiments will be a 4.2-mm thick Hafnium sleeve which will reduce the thermal neutron flux by about 85%, thus permitting the specimens to receive the same helium production-to-displacements per atom (He/dpa) ratio as is expected in the first wall of a MFE device. Horizontal and vertical cross sections through the 330°C capsule are shown in Fig. 5.

New RB* capsules have also been assembled for the High Temperature Gas-Cooled Reactor (HTGR) program. These will irradiate coated particle fuel compacts in a graphite fuel body. A horizontal cross section through a typical HTGR fuel capsule is shown in Fig. 6.

4. SUMMARY

These new HFIR facilities provide the materials irradiation community with very powerful tools with which to carry on its work. The HFIR should now be considered a world-class materials testing reactor; in the case of the instrumented target capsules, it surpasses any reactor for the magnitude of neutron flux available in instrumented irradiation experiments.

5. REFERENCES


Fig. 5. Lower half of the HFIR-MFE-330J-1 capsule.
Fig. 6. Horizontal section through irradiation capsule HRB-21.
THOR TRIGA FUEL CONVERSION POWER UPRATING
AND USE OF COMPUTER-BASED CONTROL SYSTEM

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THOR TRIGA FUEL CONVERSION POWER UPRATING
AND USE OF COMPUTER-BASED CONTROL SYSTEM

Abstract

The THOR - Tsing Hua Open Pool Reactor - went critical in April 1962 fueled with High Enriched Uranium (HEU) plate-type fuel. The core operated at 1 MW with a forced flow cooling system supplying 1500 GPM (340 M³/hr). The step-wise conversion to the use of TRIGA fuel (U-ZrH₁.₆) with low enriched uranium was begun in 1977. Initially, five clusters of TRIGA fuel were introduced into the outer portion of the core. The initial all-TRIGA core configuration was achieved in 1987. During 1988, the decision was made to upgrade the reactor power to 3 MW. The basic principles for the upgrade were to keep the existing 8-inch (203 mm) primary piping, purchase a new primary cooling system pump, motor and heat exchanger through GA, obtain all other components locally and all installation would be under the direction of the University. The 3 MW cooling system will have a flow rate of about 2500 GPM (570 M³/hr) and will not require any change from the present core configuration. Also in 1988, it was decided to upgrade the reactor instrumentation and control system to the computer-based design offered by General Atomics. This system incorporates the use of a multifunction NM-1000 microprocessor operated neutron monitor channel and the companion NP-1000 current mode safety channel. The combination of these two channels provides the redundant safety function of percent power with scram as well as wide range log power from below source level to full power. This computer-based control system design is licensed by the U.S. Nuclear Regulatory Commission for routine operational use on the GA TRIGA Mark I reactor in San Diego. The conversion is expected to be completed during 1990.

1. Introduction

The THOR - Tsing Hua Open Pool Reactor - went into initial operation with criticality being achieved in April 1962. It was fueled with High Enriched Uranium (HEU) plate-type fuel and subsequently operated at 1 MW with about 21 fuel elements surrounded by one row of graphite reflector elements. The system had a typical 7 x 9 grid plate, 4 control blades, a thermal column and eight beam ports. The forced flow cooling system operated at about 1500 GPM (340 M³/hr). The reactor is used mainly as a teaching tool for research in neutron physics, reactor engineering, chemistry, isotope production and training.
2. TRIGA Conversion

For the purposes of decreasing the fuel cycle costs and increasing the nuclear safety of the facility, the step-wise conversion to the use of TRIGA fuel (U-ZrH$_{1.6}$) with low enriched uranium was begun in 1977. Initially, five clusters of TRIGA fuel, each containing four fuel rods with a 20-inch fueled length, as shown in Fig. 1, were introduced into the outer portion of the core. The facility continued to operate with a mixed core, with an increasing fraction of TRIGA fuel, as several more fuel cluster additions were made over the following several years. The first nine clusters contained U-ZrH$_{1.6}$ fuel with 8.5 wt-% U (19.7% enriched) and then the following clusters contained higher loaded fuel with 20 wt-% U (19.7% enriched). The decreasing fraction of plate fuel was generally kept in the central part of the core. A typical mixed core configuration is presented in Fig. 2.

By 1986 over 25 TRIGA fuel clusters (with over 100 fuel rods) had been acquired and in the summer of 1987 the THOR reactor began operating with a complete TRIGA core. The initial all-TRIGA core configuration had a central dry tube irradiation facility (~2.5 inch [63.5 mm] I.D.) surrounded by 34 fuel rods with 8.5 wt-% U and two additional smaller irradiation facilities (~1.25 inch [38.1 mm] I.D.), consisting of one dry tube and one water-filled tube. The outer 64 fuel rods contained 20 wt-% U. There was a single row of graphite reflector elements adjacent to three sides of the core and 5 movable dry tubes for irradiations (~2.5 inch [63.5 mm] I.D.) were adjacent to the fourth side. The initial all-TRIGA core had an excess reactivity of about 3.5%, the maximum measured fuel temperature was about 270°C at 1 MW and the measured thermal neutron flux in the smaller dry tube near the center of the core was about $1.3 \times 10^{17}$ n/M$^2$-sec. The initial all TRIGA fuel core configuration is given in Fig. 3.
THOR TYPICAL MIXED-CORE CONFIGURATION

Fig. 2

THOR INITIAL ALL TRIGA CORE CONFIGURATION

Fig. 3
3. **Power Upgrade**

During 1988, the decision was made to upgrade the reactor power to 3 MW. A contract was awarded to General Atomics in 1989 for the engineering and evaluation of the primary and secondary cooling system pump, motor and heat exchanger. The basic principles for the upgrade were to retain the existing 8-inch (203 mm) primary piping, and that all other components (such as cooling towers, additional pumps and piping), and all installation would be provided locally and under the direction of the University. The 3 MW cooling system will have a primary loop flow rate of about 2500 GPM (570 M³/hr) and will not require any change from the present core configuration.

4. **Computer-Based Control System**

Also, in 1988 it was decided to upgrade the reactor instrumentation and control system (ICS) to the computer-based design offered by General Atomics as shown in Fig. 4. This system incorporates the use of a multifunction NM-1000 microprocessor operated neutron monitor channel and the companion NP-1000 current mode safety channel. The combination of these two channels provides the redundant safety function of percent power with scram as well as wide range log power from below source level to full power, period, and multi-range linear power from source level to full power. The control system logic is contained in a separate control system computer (CSC) with a color graphics display which is the interface between the operator and the reactor.

**OPERATING CONSOLE FOR GA COMPUTER-BASED INSTRUMENTATION AND CONTROL SYSTEM**

![Operating Console](image)
While information from the NM-1000 channel is processed and displayed by the CSC, the NM-1000 power channel is independent, has its own output displays and connects to the scram circuit. The NP-1000 is an independent channel which delivers power level data to the CSC for processing and display and also connects to the safety system scram circuit and hardwired indicators.

The NM-1000 digital neutron monitor channel was developed for the nuclear power industry and is fully qualified for use in the demanding and restrictive conditions of a nuclear power generating plant. Its design is based on a special GA-designed fission chamber, and low noise ultra-fast pulse amplifier. The NP-1000 was developed specifically for use with research reactor safety systems.

The CSC manages all control rod movements, accounting for such things as interlocks and choice of particular operating modes. It also processes and displays information on control rod position, power level, fuel and water temperature and can display pulse characteristics. The CSC also performs many other functions, such as monitoring reactor usage, and historical operating data can be saved for replay at a later date. A computer-based control system has many advantages over an analog system: speed, accuracy, reliability, and the ability for self-calibration, improved diagnostics, graphic displays and logging of vital information.

The NM-1000 nuclear channel has multifunction capability to provide neutron monitoring over a wide power range from a single detector. The selectable functions are any or all of the following: Percent power, wide-range log power, power rate of change and multi-range linear power.

For the THOR ICS all four of the above functions are used. The percent power channel with scram displays linear power level from 1 to 125%. The wide-range log power function is a digital version of the patented GA 10-decade log power system to cover the reactor power range from below source level to 150% power and provide a period signal. For the log power function, the chamber signal from startup (pulse counting) range through the Campbelling [root mean square (RMS) signal processing] range covers in excess of 10-decades of power level. The self-contained microprocessor combines these signals and derives the power rate of change (period) through the full range of power. The microprocessor automatically tests the system to ensure that the upper decades are operable while the reactor is operating in the lower decades and vice versa when the reactor is at high power.

For the multi-range function, the NM-1000 uses the same signal source as for the log function, however, instead of the microprocessor converting the signal into a log function, it converts it into 10 linear power ranges. This feature provides for a more precise reading of linear power level over the entire range of reactor power. The same self-checking features are included as for the log function. The multi-range function is either auto-range or slaved to a position switch on the operator's console via the control system computer.
The NM-1000 system is contained in two enclosures, one for the amplifier and one for the processor assemblies. Communication between the amplifier and processor assemblies is via two twisted-pair shielded cables. The amplifier/microprocessor circuit design employs the latest concepts in automatic on-line self diagnostics and calibration verification. Detection of unacceptable circuit performance is automatically alarmed. The system is automatically calibrated and checked (including the testing of trip levels) during the "Prestart" check function of the console. The checkout data is recorded for future use.

The neutron detector uses a standard 0.2 counts/per nV fission chamber. It has, however, been improved by additional shielding to provide a greater signal-to-noise ratio. The low noise construction of the chamber assembly allows the system to respond to a low reactor shutdown level which is subject to being masked by noise.

The NP-1000 Power Safety Channel is a complete linear percent power or pulse power monitoring system housed within one compact enclosure which contains current to voltage conversion signal conditioning, power supplies, trip circuits, isolation devices, and computer interface circuitry. The power level trip circuit is hardwired into the scram system and the isolated analog outputs are monitored by the CSC as well as being hardwired to a bargraph indicator. The detector is either a fission chamber or an ionization chamber.

The reactor control console contains the several components needed by the operator for reactor control. Included are the following: Reactor control panel, control system computer (CSC), two color graphics CRT monitors (one is high resolution), power and temperature meter panel, two disc drives and a graphics printer.

As previously mentioned, the power and period information from the NM-1000 channel and the NP-1000 channels are processed and displayed by the CSC. However, the NP-1000 safety system is independent, has its own output displays, and connects directly to the control system scram circuit, assuring the safe shutdown of the reactor even if the computer control system is shut down. Thus, both percent power channels, wide-range log power and period have their output displayed on meters as well as on the color CRT. This is also true of fuel temperature. Digital rod position indication is shown in inches.

The CSC provides all of the logic functions needed to control the reactor and augments the safety system by monitoring for undesirable operating characteristics. It displays reactor operational information in a color format on a high-resolution CRT monitor for ease of comprehension. A simplified control system logic diagram is shown in Fig. 5.

This computer-based control system is licensed by the U.S. Nuclear Regulatory Commission for routine operational use on the GA TRIGA Mark I reactor in San Diego.

It is expected that all conversion, power uprating and instrumentation and control system changes will be completed during 1990.
IMPROVEMENT OF PULSING OPERATION PERFORMANCE
IN THE NUCLEAR SAFETY RESEARCH REACTOR (NSRR)

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The Nuclear Safety Research Reactor (NSRR) is one of the TRIGA type research reactors widely used in the world, and has mainly been used for studying the reactor fuel behavior during postulated reactivity-initiated accidents (RIAs). Its limited pulsing operation capability, however, could produce only a power burst from low power level simulating an RIA event from essentially zero power level. A computerized automatic reactor control system was newly developed and installed in the NSRR to simulate a wide range of abnormal events in the nuclear power plants. This digitalized reactor control system requests no manipulation of the control rods by reactor operator during a course of the pulsing operation. Using this fully automated operation system, a variety of power transients such as power ramping, power burst from high power level, and so on were realized with excellent stability and safety. Present modification work in the NSRR and its fruitful results indicate a new possibility in the utilization of the TRIGA type research reactor.

1. OUTLINE OF THE NSRR

The Nuclear Safety Research Reactor (NSRR) at the Japan Atomic Energy Research Institute (JAERI) is a modified TRIGA-ACPR (Annular Core Pulse Reactor). It was built for investigating reactor fuel behavior under a reactivity-initiated accident (RIA) condition, and went to the initial criticality on May, 1975.

The reactor core is mounted on the bottom of 3.6m wide, 4.5m long and 9m deep open pool as shown in Fig. 1, and contains 149 driver fuel rods, 6 regulating rods, 2 safety rods and 3 transient rods in its standard configuration as il-
illustrated in Fig. 2. Regulating rods and safety rods are fuel-followed control rods and transient rods are air-followed ones.

The NSRR had a capability of pulsing operation with reactivity insertion of up to 4.67. It was realized by withdrawing 3 transient rods pneumatically, and the maximum peak reactor power of 21,000MW and the maximum burst energy of 117MWs were attained. The corresponding minimum reactor period and pulse width at half the peak power were 1.17ms and 4.4ms, respectively. This type of pulsing operation can simulate an RIA event from nearly zero power only. We call this operation mode "Natural Pulse: NP". The NSRR could also operate in the steady state (SS) mode of up to 300kW.

2. MODIFICATION OF THE NSRR

Recently, the need for improving the flexibility of pulsing operation mode in the NSRR was increased to cover a wide range of accidental situation including an RIA event from an elevated power level and various abnormal power transients. Based on this requirement, reactor control system of the NSRR including nuclear instrumentation and control rod drive mechanisms were modified to realize new pulsing operation modes called "Shaped Pulse: SP" and "Combined Pulse: CP". All of the past operation capabilities in the NSRR were retained through this modification work. Table 1 lists the operation modes in the NSRR after the modification.

SP is a prolonged power transient (from a few seconds to a few 10 seconds) of up to 10MW produced by fast movement of 6 regulating rods. CP is a combined power transient of a SP and a NP produced by fast movement of 6 regulating rods and quick withdrawal of 3 transient rods.

These works were performed as a part of the modifica-
Table 1 NSRR Operation Modes and Reactor Power Characteristics

<table>
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<th>Reactor Power Behavior</th>
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... of the experimental facilities in the NSRR preparing for the new test series with pre-irradiated fuels.

2.1 Automatic reactor control system for SP and CP operations

To realize SP and CP operation modes, very quick, accurate enough and complicated manipulation of the control rods is required to overcome large and prompt negative temperature feedback which is an unique feature of the TRIGA type reactor and to produce desired power shape. This type of operation is too dangerous to perform by manual control. The fully automated and computerized reactor control system was, therefore, newly developed and installed in the NSRR. Figure 3 shows basic idea of this control system.

The movement of regulating rods in bank is controlled in two ways. In one of the control methods called "open loop control", the regulating rods move according to position demand signal as a function of time from the control demand release circuit without any feedback. In the other way which is called "closed loop control", the control demand release circuit gives reactor power demand as a function of time to power regulation circuit and the movement of the regulating rods is controlled under usual feedback control with a reactor power as a feedback signal.

We can use any combination of the open and closed loop controls in a sequence of SP and CP operation. Timings of the changeover of control method of the regulating rods and transient rod fire in the CP mode are also controlled by the control demand release circuit.
One of the unique features of this control system is an introduction of digital control equipments. All of the reactor operation data are supplied to the computer in the control demand release circuit by floppy disk and no manipulation of the control rods by operator is requested during a SP or CP operation.

On-line data logging and display system was also introduced to assist the reactor operator for recognizing the operation status which is short and complicated enough to know anything from the usual recorder charts.

2.2 Other important modification items related with SP and CP operations

The nuclear instrumentation in the NSRR had been designed to monitor reactor power below 300kW in SS mode and that below 30,000MW in NP mode. SP and CP modes, however, request to monitor reactor power between 0.1 and 10MW exactly. For that purpose, two new channels (logarithmic and linear high power channels) were added to the nuclear instrumentation as shown in Fig. 4. The linear high power channel gives signal for "closed loop control" mode and equips with automatic range changeover mechanism to improve accuracy and response time.

The measuring range of the safety channels were extended to cover 10MW level for the protective actions regarding to SP and CP modes.

SP and CP modes also request to increase driving speed of the regulating rods in bank to compensate large and increasing negative temperature feedback.

Driving motors of the regulating rods were changed from AC-servo motors to stepping motors, and maximum driving speed in bank was increased from 0.72mm/s to 75mm/s. Stepping motors were adopted to improve accuracy in position control ("open loop control" mode) and to simplify
the structure of newly installed control system with digital equipments.

3. OPERATING CHARACTERISTICS OF SP AND CP MODES

Here we present some of the operating experiences with SP and CP modes.

3.1 SP mode

(1) Constant high power operation - Fig. 5(a)

Constant high power operation is a typical and basic operation pattern in SP mode and produces a trapezoidal power shape. Operation method of this pattern was as follows:

The regulating rods were withdrawn in bank under "open loop control" mode to insert reactivity of exactly +80 during 1 second and then stopped until 14 seconds. During this period, reactor power gradually increased from low power critical (15W) to about 1MW. After 14 seconds, the reactor power was controlled with "closed loop control mode. The reactor power was kept at 1 MW for 2 seconds, increased to 10MW for 2 seconds, kept at 10MW for 8 seconds, and then reactor was stopped by automatic insertion of the control rods (different from scram action).

Quick power increase from low power critical to high power level (order of MW) was essential to avoid the effect of large negative temperature feedback and to limit integral power during initial power increase (integral power
per one pulsing operation is limited to 110MW-s in the NSRR due to safety assessment).

Simulation study has shown that "closed loop control" mode such as linear power increase and power increase at constant period (from 15W to 1MW) was not stable if quick increase was desired. Initial power increase with "open loop control" mode was, therefore, selected considering the characteristics of the pulsing reactor, too.

(2) Constant high power operation followed by power decrease at constant rate - Fig. 5(b)

In this case, high power level of 3MW was attained with the same operation method as constant high power operation at 10MW. Power level of 3MW was kept for 5 seconds and then reactor power was decreased to 0.2MW at constant rate for 20 seconds. Reactor power was kept at 0.2MW for 20 seconds and then reactor was stopped. If wanted, this type of operation can be shifted to SS mode.

Power shapes of high power level in these operations exactly reproduced planned shapes except small overshooting (2~3%) at the moment of arrival to desired power level.

3.2 CP mode

(1) Constant high power operation followed by power burst - Fig. 6(a)

High power level of 3MW was continued for 5 seconds with the same operation procedure as that in SP mode. Then, reactivity insertion of $2.0$ by withdrawing transient rods was performed to produce a power burst of about 2,000MW. This type of power transient will be useful for the simulation of the reactivity-initiated event from the rated power level.

(2) Power burst followed by high run-out power - Fig. 6(b)
At time zero, reactivity insertion of $3.0$ by withdrawing transient rods was initiated to produce a power burst of about 7,000 MW. The movement of the regulation rods was started after 1 second from the initiation of the power burst to keep the reactor power at 4 MW for 3 seconds. The reactor power was increased to 9 MW for 2 seconds, kept at 9 MW for 4 seconds, and then reactor was stopped.

As shown in this chapter, newly installed automatic reactor control system worked very well for producing various types of power transients stably in SP and CP modes.

4. CONCLUDING REMARKS

(1) Modification of the NSRR was successfully completed and new pulsing modes (SP and CP) were realized with enough stability and safety.

(2) Newly developed automatic reactor control system with digital control equipments showed excellent and flexible capability of producing complicated power shape.

(3) This successful modification work brought new important application area for the TRIGA type pulse reactor such as the simulation of transient overpower (TOP) in FBR safety field and various abnormal power transients in the LWR safety field as well as that of reactivity-initiated events.
CRNL RESEARCH REACTOR RETROFIT EMERGENCY FILTRATION SYSTEM

by

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ABSTRACT

This paper presents a brief history of NRX and NRU research reactor effluent air treatment systems before describing the selection and design of an appropriate retrofit Emergency Filtration System (EFS) to serve these reactors and the future MX-10 isotope production reactor. The conceptual design of the EFS began in 1984. A standby concrete shielded filter-adsorber system, sized to serve the reactor with the largest exhaust flow, was selected. The standby system, bypassed under normal operating conditions, is equipped with normal exhaust stream shutoff and diversion valves to be activated manually when an emergency is anticipated, or automatically when emergency levels of gamma radiation are detected in the exhaust stream. The first phase of the EFS installation, that is the construction of the EFS and the connection of NRU to the system, was completed in 1987. The second phase of construction, which includes the connection of NRX and provisions for the future connection of MX-10, is to be completed in 1990.
1.0 INTRODUCTION

Conceptual design of the Emergency Filtration System (EFS) to serve the Chalk River Nuclear Laboratories (CRNL) NRX and NRU research reactors and the future MX-10 isotope production reactor began in 1984. Before describing the new system the following brief description of NRX and NRU and the development of their exhaust air (effluent) treatment systems will place the EFS system in historical context. NRX, commissioned in 1947, was originally designed to be exhausted directly through underground concrete ducts to an adjacent 61 m (200') stack without filtration, however, particulate filtration (Building 101X) was added soon after startup [Fig. 1]. In 1957 NRU was commissioned complete with its own underground steel ducts, particulate filter house (Building 162), a new multiple exhaust fan installation (Building 163) and stack [Fig. 1]. The fans and the stack (on a hill about 1 km from the reactor) were designed to be shared by both NRX and NRU. Both filter houses had 300 mm (12") thick shielding walls but little roof shielding. Filter changing was designed to be done through removable roof panels. The E, F and G exhaust fan system consisted of two 50% capacity fans and one 50% capacity standby fan. It is of interest that the NRX design preceded the advent of large positively sealed ventilation shut-off valves and designers therefore chose to use water traps to shut off ventilation flow paths and to isolate fan installations. These traps provided effective shut-off capability but were very slow to fill, required an active drainage system and had a high pressure drop.

2.0 PURPOSE AND CONCEPT DEVELOPMENT

The 1984 decision to retrofit an Emergency Filtration System (ESF) was based on the need to capture radioactive iodine in case of abnormal reactor operating conditions and severe reactor fuel failure resulting from a loss of coolant. A committee was formed to define the requirements and to review the various concepts developed by CRNL Plant Design Division to meet the requirements. Concepts presented varied from systems with 100% on-line shielded exhaust air treatment paralleled by a second identical standby system to relatively simple unshielded systems reusing as much of the existing fan and filtration equipment as possible. For operational, economic and site limitation reasons the committee selected a shared standby (bypass) EFS (Building 160) [Fig. 1] located upstream from the existing fan installation (Building 163), equipped with seismically designed concrete shielding and self-contained HEPA filter-adsorber trains [Fig. 2]. Its capacity was to accommodate NRX which has the largest exhaust flow requirement. The reactors were to continue the use of the existing filter, fan and stack installations for normal reactor operations. The new installation was to be shared and automatically engaged when abnormal fission products were detected in either the NRX-MX-10 or the NRU reactor exhaust stream.
3.0 BUILDING DESIGN

Design of the EFS began in 1986. Shielding calculations indicated requirements for 900 mm (36") normal concrete on all four sides and the top of the filter-adsorber room. The seismic design of this concrete shielding proved to be the major element in the design effort. The entire structure and the connecting underground ducts were designed using forces generated by a dynamic, computer simulated, soil model excited by a simulated earthquake using horizontal bedrock accelerations of 0.22 g and a frequency spectrum typical of the site. Extra concrete reinforcing steel and extensive rock pining was required to accommodate the seismic forces and site bedrock features.

4.0 FILTER-ADSORBER DESIGN

The exhaust stream filter-adsorber trains [Fig. 2] utilize 150 mm (6") pleated 45-55% roughing filters, 472 l/s (1000 cfm) self-contained, metal 99.97% efficient HEPA filters and CRNL designed, 99.95% efficient gasketless 50 mm (2") bed TEDA impregnated charcoal adsorber units. The filter and adsorber design efficiencies meet the requirements of Canadian standards [3]. A total of 16 parallel trains provide a maximum of 7550 l/s (16,000 cfm) filtration capacity. No special exhaust stream pre-treatment is provided. This decision is based on studies of various fuel failure and reactor loop incidents which indicated that high steam and humidity were unlikely to occur [1].

5.0 FILTER AND ADSORBER TESTING

All HEPA filters and adsorbers are equipped for in-place testing with all test points piped to convenient stations on the wall at the end of each train. Access to leak testing stations and to any filter or adsorber is facilitated by removable metal deck panels over the entire filter-adsorber area [Fig. 3]. In-place HEPA testing is done using cold, poly-dispersed D.O.P. and in-place adsorber testing is done using radioactive iodine. These are standard CRNL tests done routinely on all plant HEPA filters and adsorbers and required no new methods or retraining of technicians. Laboratory testing of adsorber carbon is required to prove the ability of the carbon to adsorb and retain the radioiodines released by a major reactor incident. These retention tests were described by J. Slade et al. in a paper given at the 19th Air Cleaning Conference in Seattle [2]. They can be done by taking a sample directly from an adsorber unit which is temporarily removed and then replaced or by using the bypass canisters installed on several trains. Experience to date indicates that carbon iodine retention data obtained by testing the canister carbon samples does not give reliable (representative) results. Carbon samples taken
directly from the beds indicate that the carbon is still within specifications after almost three years of standby service.

6.0 ISOLATION AND DIVERSION VALVES

The reactor exhaust stream diversion valving required by NRX and NRU to access the standby EFS consists of two valves in series to shut off (isolate) the normal exhaust line and two valves in parallel to open the exhaust diversion line. 900 mm (36") wafer-butterfly type valves are used for this service on the NRU connecting ducts. The leak rate of these valves must be consistent with the HEPA filter and carbon adsorber efficiency. That is, the percentage that leaks past the valves should not be greater than the percentage that leaks through the filter or adsorber. The speed of valve closure or opening must be high enough to divert the normal exhaust stream to the EFS before significant fission products are released to the environment through the normal exhaust treatment system. The speed of closing and opening for the EFS valves is under 10 seconds.

7.0 VALVE CONTROL

Exhaust stream diversion and isolation may be manually initiated by the reactor operator by means of an absolute switch in cases where an abnormal condition is anticipated or slowly developing. Diversion and isolation may also be automatically initiated by means of an absolute switch activated by an exhaust duct mounted gamma detector. The first reactor to access the EFS has sole use of the facility. There is also a conditional system switch option available to the reactor operator allowing diversion of the exhaust stream for extra filter-adsorber protection during non-routine reactor loop experiments. This conditional diversion can be instantly over-rid by an absolute manual or automatic gamma actuated switch in any of the reactors sharing the system.

8.0 CONSTRUCTION

Design and construction of the abnormal conditions filter system and the underground duct connection to NRU were completed in 1987 at a cost of less than 2 million dollars. In 1989 a second connection will be made to serve NRX and the future MX-10 isotope production reactor which will be located near it.

9.0 CONCLUSION

In conclusion, it can be said that the provision of this emergency filtration system provides a vital environmental
protection system required to extend the useful life of CRNL research reactors and to serve the future MX-10 reactor.

10.0 REFERENCES


FIG. 1 SITE LAYOUT
FIG. 2 EFS BUILDING PLAN
FIG. 3 EFS BUILDING SECTION
In-Service Inspection at the High Flux Reactor (HFR) Petten, Purpose, Methods and Evaluation

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In-Service Inspection at the High Flux Reactor (HFR) Petten, Purpose, Methods and Evaluation

Abstract

The HFR was out of operation during 1984 for replacement of the aluminium reactor vessel which had suffered irradiation damage after 20 years of operation. The design philosophy for the new vessel is that the reactor can operate provided that it can be demonstrated that the vessel contains no critical defects. This requires knowledge of the material properties together with information on likely defect sizes. As a consequence the ability to inspect in-service is an important factor in the vessel design. This is discussed further in the paper. The whole inspection programme includes ultrasonic, eddy current, dimensional and visual inspection. The inspection equipment and methods are described, concentrating mainly on the ultrasonic and eddy current techniques. The paper concludes with a consideration of the significance of the findings in relation to vessel life, having regard also for the degradation of the vessel material fracture properties.

1. INTRODUCTION

The HFR is a 45 MW, pool type materials testing reactor with light water cooling and moderation [1]. It is mainly used in support of the West German and Netherlands nuclear programmes.

In the late 1970's concern over the condition of the material of the reactor vessel (AISI 5154-0) and the need to improve experimental access led to the decision to replace the vessel [2].

At the end of 1983 the operating license was withdrawn because it had not been possible to demonstrate that the reactor vessel was free from critical defects. It had however been shown that failure of the vessel was unlikely to impair reactor safety, but the licensing authorities were of the opinion that the risk of such an incident was not acceptable. The concern expressed was that deformation of the vessel could prevent shut-down and could lead to relocation of the fuel in a non-coolable geometry.

The design philosophy for the new vessel, agreed with the licensing authorities, is that the reactor can operate provided that it can be demonstrated that the vessel contains no critical defect. Clearly this requires knowledge of the material properties [3] together with information on likely defect sizes. As a consequence the ability to inspect in service (ISI) is an important factor in the vessel design. In particular, major welds are positioned in accessible areas and the internal surface of the core box is machined flat. In the original vessel the walls were profiled to match the shape of the curved plate fuel elements.
2. REACTOR VESSEL

The reactor vessel (Fig. 1) is an all welded construction about 5 m high. The core box area is fabricated from 50 mm thick forged material joined by electron beam welding. Elsewhere plate material 40 mm thick is employed using MIG welds. The intent of the ASME III Boiler and Pressure Vessel code was adopted as the design and construction standard.

3. INSPECTION PROGRAMME

In establishing the scope of the inspections it was clearly not practicable to cover 100% of the vessel, so a programme was drawn up for ultrasonic examination of a selection of typical welds vital to the integrity of those parts of the vessel important for maintenance of core geometry and control rod alignment. Included in this list are the two walls of the core box most subject to irradiation damage in service. As a further check on the core box the eddy current technique is used for the detection of surface and sub-surface defects. The programme also includes dimensional surveys to check control rod alignment and for any gross distortion of the vessel, a general visual inspection using closed circuit television, and penetrant testing of the primary coolant inlet vanes located above the core.

For comparison purposes a full inspection was carried out after delivery of the vessel to Petten and prior to entering service. (This was in addition to the fabrication inspections performed by the manufacturer). Where possible the same personnel and techniques were used for pre-service (PSI) and in-service inspection. In-service inspection is carried out every three years. An abbreviated inspection was performed in 1986 in view of the findings on Weld 22 during pre-service inspection.

4. INSPECTION EQUIPMENT

4.1. P-Scan

The scanning system used for both Ultrasonic and Eddy Current inspection is the SVEJSECENTRALEN (SVC) P-Scan system [4],[5]. P-Scan stands for Projection image SCANing, where ultrasonic data is mapped on to two projection planes, one parallel and one normal to the test piece surface. Fig. 2 shows how a lack of fusion defect towards the bottom of a weld would be presented using this system. Presentation of both top and side views allows the visualisation of defect size and location in three dimensions.

The P-Scan system is shown in Fig. 3. The processor unit is the hub of the complete system co-ordinating peripheral operations as well as storing test data. Control of the processor is via a keypad, two-letter commands are used to perform most functions.

The Interface unit is responsible for the production and reception of ultrasonic signals, probe position, data acquisition and control of peripheral scanning equipment.
Real Time scan status is displayed on two monitors, Fig. 4 shows the display format. One monitor displays the side and top views, signal amplitude and signal threshold data. The other shows the digitised A-Scan and the portion of the scan volume that is currently being illuminated with the ultrasonic beam.

The top and side views are shown to scale. During analysis the echo amplitude in relation to the calibration reference level of any reflector can be obtained by adjusting the display level threshold until the echo just appears in the top and side views. The size of the reflector can then be established using the 6 dB drop technique by lowering the display threshold a further 6 dB and scaling from the display screen.

Test data are recorded on magnetic tape, such that all scan parameters and scan ultrasonic amplitude data may be retrieved and re-analysed (by adjusting the Display Level Threshold). A hard copy of displayed information is available using a video display capture unit.

Probe positional data (X, Y and twist) and certain hardware determined parameters such as Ultrasonic Probe beam angle, beam spread and test material properties are combined with the digitised A-Scan information to determine the position of any ultrasound reflector within the scanned volume.

**4.2. Probes**

The ultrasonic probes used for PSI were Krautkramer compression and shear wave probes. Nominal frequency 2 MHz, crystal size 20 x 22 mm rectangular. The immersion probes used during ISI's are Aerotech 2.25 MHz with 19 mm diameter crystals.

**4.3. Pre-Service Inspection Manual Weld Scanner (MWS)**

During PSI the probe positional feedback was derived from a Manual Weld Scanner manufactured by SVC. The MWS base was attached to the vessel wall (outside the required scan window) using suction clamps. Scanner arm radial and angular movements are converted to voltage signals using potentiometers, and passed back to the P-Scan interface together with probe twist signal. These signals are interpreted by P-Scan in terms of planar cartesian co-ordinates and ultrasonic beam direction.

Scan windows were 250 x 200 mm for volumetric inspections. Welds were scanned in 250 mm lengths. The operator performed a raster scan of the scan window, confirming full coverage by viewing the P-Scan data display.

**4.4. Immersed Automated Scanning System**

ISI inspections require an immersible, remote, probe deployment technique to avoid the problems associated with personnel contamination/irradiation when in close proximity to the activated reactor vessel.
Inspectorate-International designed and manufactured a probe deployment/scanning system as shown in Fig. 5. A cylindrical mast is suspended from a frame mounted on the reactor vessel flange. A carriage mounted on this mast can be winched up and down the mast by means of a DC motor at the very top of the mast. Connection of the carriage to the motor is via stainless steel chain. An optical encoder mounted on the final drive sprocket shaft provides the positional information sent back to the scanner control cabinet.

On the carriage is mounted a purpose built cartesian co-ordinate scanning frame that provides 360 x 200 mm movement of a general purpose mounting plate. X and Y movements are controlled by stepping motors, positional feedback is provided by optical encoders mounted on the stepping motor drives. Magnetic proximity detectors indicate limits of scanner travel.

The scanner control cabinet is connected to the scanning mechanism by a waterproof umbilical cable. Both the umbilical cable and the electrical components of the scanning mechanism are held at an air pressure slightly higher than the water pressure at whatever the depth of the scanning mechanism. This prevents ingress of water to the electronic compartments of the scanning mechanism. The control cabinet houses the scan raster controls, positional information displays, electronic interface to the P-Scan system, and pneumatic and hoist controls.

A number of probe arms can be mounted on the scanner mounting plate to position the ultrasonic probes in the locations appropriate for the required scan window. Fixed ultrasonic probe stand-off is achieved by means of fixed support arms which are brought into contact with the reactor vessel by the action of pneumatic cylinders. Coarse adjustment of the scanner relative to the mast is also provided using pneumatic cylinders.

The probe beam angle is achieved by placing the probe in holders angled to produce ultrasonic beams in the test material of 45, 60, 70, 10 and 0 degrees.

The relationship between all the manipulator extension pieces, positional encoder readings and position of ultrasonic beam entry into the vessel is complex and so a computer program was written to enable the optimum scanner configuration for a particular scan window to be obtained.

A closed circuit television camera system is employed during ISI which serves to give remote visual confirmation of correct scanner operation.

4.5. Personnel Qualifications

All technicians operating the equipment are qualified to level II standard in accordance with the American Society of Non-destructive Testing (ASNT) requirements. Interpretation personnel were registered ASNT level III technicians.
4.6. **Visual and Dimensional Inspections**

Visual inspections are performed using a closed circuit television camera mounted on a manipulator. Data is recorded on video tapes. Dimensional measurements are taken by aligning an optical telescope with well identified features on the vessel and recording data from an X-Y co-ordinate table.

5. **ULTRASONIC TECHNIQUE**

Ultrasonic testing utilises a beam of high frequency sound passing through the volume of the material to be inspected. Any discontinuities in the material reflect a proportion of the incident beam back to the probe, which converts the reflected sound energy into an electrical signal. The time taken until the echo is received indicates the distance of the reflector from the probe and the amplitude of the received signal relative to a calibration reference reflector can be used to assess reflector size.

Expertise is needed in order to interpret accurately the ultrasonic signals in terms of weld defect size and location. The P-Scan system described earlier is used to calculate and plot the position of all reflectors detected within a certain time gate. Each indication plotted by P-Scan is verified using manual ray tracing techniques and traditional ultrasonic equipment.

During PSI angled shear wave probes were used with a gel couplant to transmit sound between the probe and test material. During ISI an immersion technique is used where the probes are held at a fixed stand-off distance from the vessel surface. The reactor coolant (water) provides the sonic couplant.

The 6 dB drop sizing technique, as given in the ASME XI code [6], is used during inspections.

For calibration of the ultrasonic equipment the probe and probe mounts need to be disconnected from the scanning equipment. Calibration is carried out manually with the aid of a small immersion tank. The position of the probe is monitored with potentiometers and the signals from these and the probe are passed to P-Scan. Calibration blocks with side drilled holes conform to the ASME standards. These holes are used to produce Distance Amplitude Curves (DAC). Signals from reflectors within the reactor vessel material that are above the DAC threshold are "reportable", reflectors giving a signal of above 50% DAC are "recorded".

6. **EDDY CURRENT TECHNIQUE**

Eddy current testing involves the induction of currents into the test material surface by a coil, held in a probe, close to the material surface. When these eddy currents are disturbed by the presence of surface breaking defects a measurable change in impedance of the probe coil occurs. The amount of impedance change is related to the size and nature of the defect.
An eddy current technique is used to inspect the vessel core box wall and corners for surface and near surface defects. The eddy current probes used are of the absolute type. The probe bodies are profiled to suit the scanner mountings and inspection surface. The probes are held at a constant lift-off from the vessel surface (0.5 mm) by means of three roller bearings adjacent to the probe which are held in contact with the vessel surface by the scanning frame pneumatic system.

The test frequency used is 10 kHz. During calibration the eddy current analyser is adjusted to ensure that "lift-off" signals are displayed only in the X axis (analyser CRT display), whilst signals from defects or notches appear in both X and Y axes. Only the Y component of the defect signal is used for defect size analysis.

A summing amplifier is used to add the Y component of the eddy current analyser display to the Y signal derived from the scanner position. The combined X-YY signal is fed to an X-Y plotter. The resulting output shows both defect position and size. A typical display is shown in Fig. 6. In the absence of any eddy current signals due to defects the plotter trace would be a scaled down image of the probe raster.

Calibration of the eddy current system is carried out using a block of material of the same properties as that of the reactor vessel. Various EDM notches were machined in the block. A 20 mm long by 3 mm deep notch is regarded as the reporting level. Signals having an amplitude of 50% of the reporting level are recorded.

7. INSPECTION RESULTS

Full results of the ultrasonic and eddy current inspections are given in the reports [7].

Eddy current testing did not detect any surface defects above the reporting level. However two areas of response were noted: The first was due to lift-off effects caused by the probe not travelling smoothly over the core box surface, the second was due to the material property changes occurring at the ends of the scan and in the core box west face.

The 1988 ISI ultrasonic examinations revealed reportable indications in several areas. Most indications were due to geometric features of the vessel. Only three of the welds examined gave reportable indications due to non-geometric features (defects), these were in welds 1, 22 and 44. Fig. 1 shows the position of these welds on the reactor vessel. All three are double vee butt welds.

The indication in weld 1 arises from a surface gouge resulting from a weld repair at the time of fabrication. The indication in weld 4 is consistent with a point reflector (spherical inclusion or void). Neither indication is relevant to vessel integrity.

The indication in weld 22 is of greater significance. A typical P-Scan image showing the defect is shown in Fig. 7. This defect was also found in previous inspections where it was concluded that this indication
was due to an embedded lack of weld fusion approximately 36 mm long by 3 mm. The size and location of the defect had already been confirmed by X-radiography techniques during the PSI.

From the visual and dimensional inspections there were no significant findings.

7.1. Comparison of PSI and ISI results

A comparison of the results on weld 22, obtained during all three inspections is given in table I.

The differences between inspections are small, and in general are within the P-Scan resolution of ± 4 mm. The main differences being between the results of the manual weld scanning versus the automated immersion testing. This is due to the fact that when an operator is scanning the inspection material with a hand held probe he is encouraged to twist the probe relative to the direction of the weld to obtain maximum response from the reflector (when the reflector surface is normal to the ultrasonic beam). With the automated scanning technique the probe is held at constant angle with respect to the weld. This often leads to a reduced amplitude response and hence smaller reported defect sizes.

The maximum amplitude response between inspections of weld 22 were all within 1-2 dB, which is well within the expected error band for reproducability of results in an ultrasonic inspection system.

8. CONCLUSIONS

Routine in-service inspection has been performed on the HFR vessel in accordance with the operating licence requirements. Inspections have taken place during routine shut-down periods and have not reduced the availability of the reactor for experimental purposes. All the evidence from the inspection leads to the conclusion that there has been no change in the vessel since its installation in 1984.

The significance of a weld defect discovered during pre-service inspection has been extensively analysed [8] with the result that it is not critical and not expected to grow over the life of the vessel. This analysis took into account the irradiated properties of the vessel material [3]. The ability to detect this defect by remote inspection techniques and to confirm the prediction that it would not change in-service gives added confidence in the quality of the inspection techniques, and thus an assurance that the remainder of the vessel is defect free. Should for any reason a defect arise in-service then there is a high probability that it would be detected at an early non-critical stage, thus giving time for repair and avoiding a possible extended shut-down period for replacement.
9. REFERENCES


<table>
<thead>
<tr>
<th>PSI/ISI</th>
<th>Probe Angle</th>
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(*) Ultrasonic amplitude relative to DAC datum.

Table I Weld 22 defect. Comparison of PSI and ISI ultrasonic scan data from inspections made from inside the vessel (upper table) and outside (lower table). Dimensions in mm.

ISI 1 is 1986 inspection; ISI 2 is 1988 inspection.
Fig. 1 HFR Reactor Vessel showing positions of welds with ultrasonic indications.

Fig. 2 P-scan principle. The defect indications shown correspond to a defect in the centre of the weld and near the bottom of the plate. The "Main Zone" M is the width of the volume examined. T is the thickness of the weld.

Fig. 3 P-scan system.
The P-scan system is equipped with an A-scan display on a TV-monitor. The A-scan is displayed when the system is in execution mode.

The A-scan displays a living picture: The initial pulse, the time gate and the beam markers move in accordance with the probe movement, whereas the main zone is fixed.

**Fig. 4 P-scan display**
Fig. 7 P-Scan display of weld 22 defect
NEUTRONIC CALCULATIONS FOR MODIFICATION OF KINKI UNIVERSITY REACTOR FROM HEU TO LEU FUELS

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NEUTRONIC CALCULATIONS FOR MODIFICATION OF KINKI UNIVERSITY REACTOR FROM HEU TO LEU FUELS

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ABSTRACT

The Kinki University Reactor (UTR-KINKI) is a modified Argonaut type, light water-moderated zero power reactor with highly enriched (90 wt.%) uranium-aluminum alloy flat MTR plate type fuel. Neutronic calculations were performed on UTR-KINKI to examine the feasibility of replacing the current HEU fuels with low enriched (19.75 wt.%) fuels without changing the main core dimensions and configurations other than the fuel element. The primary reason for studying the conversion is the concern with safeguard associated with the HEU plate fuel. The second reason for converting is related to a desire to introduce some spectrum modifying material around the experimental cavity at the center of the internal graphite reflector to establish a "modified spectrum neutron field" for radiation biological researches. The effect of reducing fuel enrichment on the nuclear characteristics including the neutron spectrum and the neutron flux distribution around the experimental cavity of UTR-KINKI are investigated. It is concluded from the present neutronic calculation that the LEU fuel is feasible for UTR-KINKI without any significant reduction or changes in the reactor performance.

1. INTRODUCTION

The Kinki University Reactor, UTR-KINKI \[1\], is a modified Argonaut type, light water-moderated and graphite-reflected, zero power (1W) reactor. The reactor is hetero-
geneous in design and consists of twelve fuel boxes in two slab arrangement separated by 46 cm internal graphite reflector. The overall dimensions of the reactor core, placement of the fuel boxes relative to other regions, along with the experimental cavity in the internal graphite reflector are indicated on the horizontal cross-sectional view of the core of UTR-KINKI in Fig. 1.

![Horizontal cross sectional view of UTR-KINKI core](image)

Fig. 1 Horizontal cross sectional view of UTR-KINKI core

The UTR-KINKI currently employs a highly-enriched (90 wt.%) uranium-aluminum alloy, flat MTR-type plate fuel. Investigation were performed to examine the feasibility of replacing the HEU fuel of current UTR-KINKI with low-enriched (nominal 20 wt.%) silicide or aluminide fuel without changing the main core configurations other than the fuel element. The primary reason for studying conversion of the UTR-KINKI to the LEU fuel is the concern with safeguards associated with the HEU plate fuel. A second reason for converting to the LEU fuel is related a desire to introduce some neutron spectrum modifying materials around the experimental cavity at the center of internal graphite reflector to establish a "modified spectrum neutron field" for radiation biological researches. Such a change would be difficult with the current HEU fuel because the introduction of substantial amount of spectrum modifying material into the core causes the reactor subcritical and we have only limited amount of
HEU fuel.

This paper represents the results of detailed neutronic investigations performed for UTR-KINKI conversion from HEU to LEU fuel. The study is concerned with the use of both LEU silicide (\(U_3Si_2\)-Al) and aluminide (UA1\(_x\)-Al) dispersion fuels from the point of view of the present commercial fuel fabrication bases. For uranium enrichment, 19.75\% was selected and for uranium loading in the fuel matrix, 3.8g-U/cc for silicide and 2.0g-U/cc in the case of aluminide, were selected as conservative values, respectively.

The neutronic investigations was done under the following restrictions: (1) Main core dimensions and configuration including the size and thickness of each fuel plate of UTR-KINKI except the water gaps between the fuel plates, i.e. the number of fuel plates per element, must be exactly same as in the current HEU core. (2) The change of neutron spectrum around the experimental cavity in the center of the internal graphite reflector should be minimum.

2. METHOD OF CALCULATION

Neutronic calculation was performed using SRAC Code System\(^{[2]}\). The system, JEARI Thermal Reactor Standard Code System for Reactor Design and Analysis, was developed at Japan Atomic Energy Research Institute and its adaptability to complex core configuration has been demonstrated through many benchmark experiments. SRAC Code System consists of neutron cross section libraries and auxiliary processing codes, neutron spectrum routines, a variety of transport and diffusion routines, dynamic parameters and cell burn-up routines, etc.

The fundamental group constant library was produced mainly from ENDF/B-IV nuclear data file with the energy structure of 107-group (48-group for thermal and 74-group for fast energy ranges, respectively, with 15 overlapping groups). The transport cross-sections for the \(P_0\) transport calculation were calculated by the \(B_1\) approximation, and the diffusion coefficients were obtained assuming \(D = 1/(\Sigma \tau)\). The resonance absorption for heavy nuclides was calculated by a table look-up method for the neutron energy above 130.04 eV and a collision probability method using ultra-fine energy points of 4600 for neutron energy between 130.07 and 0.68256 eV. The user library was constructed with a energy group structure of 50 groups (23 fast groups and 27 thermal groups).

The unit cell calculation for multi-group constants
was performed by collision probability method in 1-D slab geometry for fuel meat region composed of fuel meat ($U_3Si_2$-Al for silicide dispersion fuel and UA1x-Al for aluminide dispersion fuel), aluminum cladding and light-water. Multi-group (fast 31 and thermal 10 groups or fast 5 and thermal 5 groups) core calculation was performed by 2-D diffusion code (CITATION) in X-Y geometry.

3. RESULTS OF NEUTRONIC CALCULATIONS

The results of preliminary calculations on the minimum critical masses of LEU with reducing water gaps between the fuel plates of each fuel element are shown on Table I. In general, the minimum critical masses of LEU are several per cent larger in the case of aluminide dispersion fuels than silicide fuels.

Table I. Minimum Critical Masses for LEU fuel cores

<table>
<thead>
<tr>
<th>Water Gap</th>
<th>Critical Mass (LEU 19.75%)</th>
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</thead>
<tbody>
<tr>
<td></td>
<td>Silicide ($U_3Si_2$-Al)</td>
</tr>
<tr>
<td>1.016cm (12 plates per element)</td>
<td>3.420g U-235</td>
</tr>
<tr>
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<td>17.330g U</td>
</tr>
<tr>
<td>0.660cm (17 plates per element)</td>
<td>3.500g U-235</td>
</tr>
<tr>
<td></td>
<td>17.620g U</td>
</tr>
<tr>
<td>0.500cm (21 plates per element)</td>
<td>3.850g U-235</td>
</tr>
<tr>
<td></td>
<td>19.220g U</td>
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</table>

The effects of reducing fuel enrichment on the nuclear characteristics including neutron spectrum and thermal neutron flux distribution around the experimental cavity at the center of internal graphite reflector of UTR-KINKI were also investigated. The results show that there are no significant change in the neutron spectrum and the thermal neutron flux levels in the fueled region of LEU core are about 10% lower than that of current HEU core. However, they are much more flat in the internal graphite reflector region than that of HEU core. In the current HEU core, neutron spectrum in the experimental cavity at the center of internal graphite reflector has an excellent 1/E neutron spectrum over wide energy range. The calculated
neutron spectra of this region are almost unchanged, both for silicide and aluminide LEU fuel core. Fig. 2 shows the neutron spectrum of this region in the HEU core and Fig. 3 shows the neutron spectra in the LEU silicide core.

**Fig. 2** Neutron spectrum in the experimental cavity of UTR-KINKI, HEU fueled core

**Fig. 3** Neutron spectrum in the experimental cavity of UTR-KINKI, LEU silicide fueled core
The modified neutron spectrum in the experimental cavity at the center of internal graphite reflector due to the introduction of spectrum modifying materials were also investigated. By replacing part of the graphite reflector with some spectrum modifying materials, such as bismuth, aluminum, lead or stainless steel blocks, etc., significant modification of neutron spectrum can be attained. But these spectrum modifying materials have large negative reactivity values, i.e. about \(-2.5\% \Delta k/k\) for 3.82cm thick stainless steel blocks and about \(-0.05\% \Delta k/k\) for bismuth blocks of same thickness. Nevertheless, by increasing the loading of fuel plates per element, these large negative reactivity can be compensated without serious troubles. Figs. 4 shows the "modified" neutron spectrum in the experimental cavity of UTR-KINKI when 3.82cm thick stainless steel blocks were inserted around the experimental cavity in center of the internal graphite reflector.

Fig. 4 Modified neutron spectrum in the experimental cavity with stainless steel modifier, LEU core

Some of the results of core calculations other than the results mentioned above are as follows: (1) The temperature effect of reactivity in the LEU core is more negative than that in HEU core, both for silicide and aluminide fuels. (2) The value of \(\beta/\lambda\) in LEU core is approximately equivalent to that in HEU core.
4. CONCLUSION

The results of the present neutronic investigation show that the conversion of UTR-KINKI from HEU fuel to LEU fuel is quite feasible without changing any main core dimensions or configurations except the number of fuel plates per element. The effects of reducing fuel enrichment on the main nuclear characteristics of UTR-KINKI including the neutron spectrum in the experimental cavity and neutron flux distribution in the internal graphite reflector are very small and provide no problem in the application of UTR-KINKI.

REFERENCES

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CHALK RIVER, ONTARIO, CANADA 23-27 October 1989

SECURITY DEVICES AND EXPERIMENT FACILITIES AT ENEA TRIGA RC-1 REACTOR.

MODIFICATIONS OF EQUIPMENTS.

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ABSTRACT

RC-1 TRIGA operating exercise staff has realized some auxiliary security devices as follows identified:

a) - Neutron source automatic handling device.
b) - Irradiated samples "Rabbit" connection rotating rack
c) - Auxiliary equipment hot-fuel elements transferring

a) - Neutron source automatic handling device.

The reactor electronic control instrumentation system comprehends various instrumentation channels, which operating capability must be verified by licensee as per Italian regulations.

In order to obtain automatic and repeatable operations, TEMAV designed and constructed an equipment with remote drive to transfer neutron sources, on basis of requirements, performances and technical data requested by ENEA-TIB dept.

b) - Irradiated samples "Rabbit" connection rotating rack

Pneumatic radiating system for short lived materials allows an extraction of radiated samples in a time no longer as 4 seconds.
For system optimizing, both sides operability and sanitary protection, has been realized a specific rotating rack for connection of irradiated samples with pneumatic transfer (RABBIT).

c) - Auxiliary equipment hot-fuel elements transferring

Actually to permit 1 MW hot fuel elements storage into pits it is necessary to remove hot 100 KW fuel elements and transfer them to re-treatment plant.

The feasibility studies brought evidence the impossibility of use heavy truck inside Reactor Hall.

To avoid all problems it results the opportunity to leave truck outside Reactor Hall and to move only PEGASO container with a special equipment rail running. Rail-truck translation is assured by electromotor driving pull device and security cable.
1. FOREWORD

The ENEA 1 MW REACTOR RC-1 is a modified TRIGA MARK II reactor realized by the Gulf General Atomic (San Diego - California, USA). It is an etherogeneous-homogeneous type, totally reflected and water cooled. Reactor itself is placed at the bottom of an aluminium vessel, of cylindrical shape, open on the top, about 7 m high. This vessel is placed inside the biological shield.

The fuel elements which constitute the reactor core, are cylindrical and made of a ternary alloy of H2r-U, containing H and Zr in the ratio 1.7 to 1.

This alloy contains 8.5 wt % of U enriched to 20 % in U235. These fuel elements have small graphite cylinders at the top and the bottom, and are placed inside a stainless steel sheet.

The reactor incorporates various facilities for a basic nuclear research and training as well neutron and γ radiation studies, isotopes production and sample activation.

RC-1 TRIGA operating exercise staff has realized some auxiliary security devices to permit safe, accurate and quick reactor facilities utilization.

This paper includes a description of three of those security auxiliary devices as follows identified:

- Neutron source automatic handling device.
- Irradiated samples "Rabbit" connection rotating rack.
- Auxiliary equipment hot-fuel elements transferring.

The design of equipments herein described has been based according to specific regulations and standards for nuclear reactor plants as well italian general security dispositions and technical requirements.

All activities relevant to security were executed in compliance with ENEA - RC-1 TRIGA Nuclear Quality Assurance Program Manual and specific policies and procedures to be applied for such design and realization.

Enclosed figures show mechanical features, main assemblies and installation of above mentioned devices.
2. **NEUTRON SOURCES AUTOMATIC HANDLING DEVICE**

2.1 - **Basic problem outlines**

Reactor electronic control instrumentation system comprehends, as following detailed, a set of instruments for neutron flux measurement and control, a channel for start-up, two linear channels with a large range of survey, a logarithmic channel and also a safety channel.

The licensee, on basis of thechnical italian exercise regulations, is obliged to verify the operating capability of above mentioned channels before start-up and reactor running conditions.

To this purpose, in control schedule, before start-up it is foreseen to admit a signal the nearest as possible to the sensing elements of channel for verifying operating capability itself.

In the past this operation was practically executed fishing with a special electromechanical equipment the start-up source (Am-Be of 5 Ci) settled in a hole of reactor core, and positioning subsequently itself near to the above channel counters. Troubles with such operations were essentially:

a) Big difficulties in positioning neutron source in front of various detectors. Different measurements, effected in different times, with same detector, resulted difficult to compare.
b) System, too much hard-working, because of source coupling up and disconnecting.

c) Caution to be observed in all handling situations, for maintenance and reparation, because of high activity of electromechanical device itself.

d) The possibility to take up radiological doses because of operating on top of reactor and consequently, in a position to touch demineralized water of nuclear process (operation made by protected staff).

e) The waste of time referred to the above mentioned problems.

In order to obtain automatic and repeatable operations, TEMAV designed and constructed an equipment with remote driving to transfer neutron sources, on basis of requirements, performances and technical data requested by ENEA-TIB Dept.

The device so realized is admitting of important developments and with any modifications, may be identified like a standard equipment necessary to all TRIGA type reactors erected in different research institutes.
2.2 - Components

2.2.1 - Electrical supply board and control panel

Electrical supply is low voltage type (24 Vdc) and allows an independent driving of linear actuator. Both electrical equipment fixtures are located in Control Room and permit operation from main Control Panel Board (Console).

2.2.2 - Derivation box

This box is located in top reactor area and feeds the linear actuator separately.

2.2.3 - Supports with linear actuators

Aluminium support assembly are fastened to steel beams, installed at top of reactor tank itself. A single support allows a single installation of source, driven by a single linear actuator by relevant source support rod. The linear actuator, commercial type, consists of a motor device with reduction gear, worm type, with a helical gear ball bearing screw, which allows, with a high grade of precision and repeatability, linear movements.

Limit switches with magnetic proximity sensors allow predetermined operation selection (translation).
2.2.4 - Source support rods

It consists of a 7 m straight aluminium tube of $\varnothing=22$ mm composed by single tubes, end screwed and pin fixed, with box ending who contains the neutronic source (0.5 Ci) of Am-Be. Adjustable locking shaft connects tubes to actuators to allow source correct positioning.

2.2.5 - Detector tube containers

The aluminium ion chamber container structure is realized by two tubes $\varnothing=120$ mm and $\varnothing=40$ mm diameter with 1200 mm length and 90 mm center distance fastened to reflector structure. Each tube respectively permits ion chamber location and rod source driving during translation operation, all without modification of mutual distance itself.

2.3 - Operating features

Limit switches set-up is performed referring to neutron source location relevant to max sensibility of detector area.

All reading instrumentation operation, control and comparison, are standard performed to verify daily values validity.
Start-up channel log and power level 1 and 2 channel reading instrumentation are Control Panel Board (Console) performed with relevant meters located in Control Room. The device Control Panel Board contains all information for system operability with up-down level yellow flashing, red and green light indicating source translation and also top-down position for each instrumentation channel.
Fig. 2.4

NEUTRON SOURCE AUTOMATIC HANDLING DEVICE

- ELECTROMOTOR
- LINEAR ACTUATOR
- SUPPORT STRUCTURE
- TOP BEAM
- REACTOR TOP LEVEL
- WATER SURFACE
- SOURCE SUPPORT RODS
- SOURCE (5Cl)
- REFLECTOR
- DETECTOR TUBE CONTAINER
3. **IRRADIATED SAMPLES RABBIT CONNECTION ROTATING RACK**

3.1 - *Basic problem outlines*

Triga type reactor involves a research plant with some in pile radiation facilities such as a pneumatic radiating system for short lived materials and the pneumatic transfer system (RABBIT) itself allows an extraction of radiated samples in a time no longer as 4 seconds. Sample positioning into reactor core is obtained by a tubing system from blower to shielded cell located in the plant Radio-chemical Laboratory, where the "sample introduction box" finds housing, and from here runs up to the hole of grid peripheral ring of reactor core.

For system optimizing, both operability and health physic protection, has been realized a specific rotating rack for irradiated samples connection with pneumatic transfer.

3.2 - *Components*

3.2.1 - Shielded cell

Cell assembly consists of a lead shielded box of overall dimensions about 1000x1000x1500 mm which contains in-out rack connection samples.
3.2.2 - 10-positions shielded rack

The lead shielded rack with rotating top plate is assembled on bearing supports and anticorodal aluminium housing and elements.

It contains 10-position ring plate opening for samples capsule holders located on peripheral circumference, also totally shielded, each opening accessible by single hole on rotating top plate who contains 10-positions indicator.

A removable plug completes top plate.

3.2.3 - Basic structure

Basic structure consists of a step by step rotating electromotor driven circular plate. Empty rack is manually moved to loading circular plate operating position. Rotation is controlled by electromechanical and electronic system which working conditions are indicated on Panel Board and, in case, provokes rotation block in situation of simultaneous two samples insert, with relevant alarm in Control Room.

Rack system slides on appropriate guides allowing left position, all over rotating plate for loading, and right position for unloading or decaying.

Empty rack slides from guides derivation to rotating plate with box housing alignement.
3.2.4 - Control Panel Board

Control Panel Board of 24 Vdc electrical supply is located in Radio-chemical Room with remoting control in Reactor Control Room.

3.3 - Operating features

Rabbit connection rotating rack allows short lived isotopes irradiated samples use with automatic handling and operability.

Reactor operator checks all operations because of remote control in Control Room.

Automation device consists of filling sequence selection possibility for different samples to irradiate and automatic sequence execution.

All switches and control lights of main supply and equipment Control Board, located in Radio-chemical Room, have three lights indicating main operational situations:
- rack arrival missing in samples connection box;
- sample request indications in assigned opening;
- rack full and rack empty changing request indication.

Filling sequence is indicated on Control Panel Board by digital displays.
Fig. 3.4

IRRADIATED SAMPLES "RABBIT" CONNECTION

ROTATING RACK

- CONTROL PANEL BOARD
- 40-POSITION SHIELDED RACK
- REMOVABLE PLUG
- ROTATING PLATE
- BASIC STRUCTURE
- SYSTEM SLIDING GUIDES
- HAND LEVER TO ROTATING PLATE
- INSIDE SHIELDED CELL IS LOCATED THE STEP BY STEP ROTATING ELECTRODRIVE DRIVEN CIRCULAR PLATE
4. **AUXILIARY EQUIPMENT HOT FUEL ELEMENTS TRANSFERRING**

4.1 - **Basic problem outlines**

The ENEA RC-1 TRIGA reactor is a more than 20 years old installation originally of 100 KW, actually 1 MW power after structure modification.

Because of power upgrading 100 KW fuel elements were stored in pits located inside Reactor Hall. Fuel transfer from tank to pits is by "Coffin" realized.

Actually, to permit 1 MW hot fuel elements storage into pits, it is necessary to remove previous hot 100 KW fuel elements and transfer them to re-treatment plant. Special PEGASO container of 20000 Kg weight, loaded on special truck, performs transfer itself.

The RC-1 plant is a research installation not allowing industrial fuel handling as usually by PEGASO container performed.

The feasibility studies brought evidence to:
- RC-1 floor slab structurally not designed for heavy loads
- access door and Reactor Hall not designed for big dimension equipment
- bridge-crane working area not covering access door zone
- any case impossibility to engage main access door and run-way because of operational long time and security of TRIGA itself.
To avoid all above mentioned problems it results the opportunity to leave road-truck outside Reactor Hall and to move only PEGASO container with a special equipment rail running. Rail-truck translation is assured by electromotor driving pull device and security cable.

4.2 - Components

4.2.1 Rail

Rail-way 14 m lenght runs 8 m inside Reactor Hall connecting bridge-crane working area to access door, and extends outside Hall others 6 m.

Rails are installed on transversal steel mainframes of different hight, to obtain min 2% slope for assuring constant cables tension in pulling and security. Anchor bolts, heavy type, assure frames to slab foundation.

Rails structure section inside Reactor Hall, in access door zone, is allowed to rotation till vertical position by shaft electromechanical motor driving, connected with rotating section itself. This facility permits free access to Reactor door during all fuel transfer operations.

Pull device is constituted by electromechanical equipment and gear box at low revolution number. Security cable is also installed and hand operated by winch. Both devices are fastened on special headframe located near Reactor tank.
4.2.2 - Rail-truck

Grid of steel shapes, with loading surface to compensate rail slope, constitutes the rail-truck structure restoring original PEGASO road-truck fastening, to road truck. On front transversal beam, both pull and security hooks are installed.

4-axis (8 steel wheels), able to transfer total weight to railway structure, support the system and assure PEGASO container verticality to connect "Coffin" hot fuel elements with perfect security operation.

4.3 - Operating features

To perform transfer of hot fuel elements from pits to PEGASO rail-truck system is operating as following:
- transfer of PEGASO from road-truck to rail-truck outside Reactor Hall
- rail-truck with PEGASO pulling in Reactor Hall at Reactor tank proximity and fastening in operating position
- vertically rail section rotation assuring access to Reactor Hall door
- Reactor Hall door closing to assure negative pressure in ambience.
- hot fuel elements transfer by "Coffin" and bridge-crane, and cyclic operation till end of hot fuel transferring itself
- equipment assembly outside building sliding after railway restoration
- PEGASO road-truck loading.
Fig. 4.4

- Auxiliary equipment hot fuel elements transferring

Inside Reactor Hall

- Security Hooks
- Pegaso Container
- Rail-Truck Driving
- Head-Frame

Outside Reactor Hall

- Rail Section Rotating
- Access Door
- Rails Connection
- Rails
REACONDICIONAMIENTO DEL REACTOR EXPERIMENTAL DE LO AGUIRRE
(The Lo Aguirre Research Reactor Refurbishment)

GONZALO TORRES-OVIEDO
COMISION CHILENA DE ENERGIA NUCLEAR
CHILE
A description is given of the main work which had to be performed on the experimental reactor of the Lo Aguirre nuclear power plant (RECH-2) and following which it recently came into operation. In particular, an outline is given of the main changes and improvements made with regard to reactor physics calculations, the systems and components in the facility, and repair of existing fuel elements. Special importance was attached to the definition, application and meeting of nuclear safety requirements and the implementation of a consistent quality assurance programme. Certain aspects of the work performed, by virtue of the scope and importance of the tasks involved, resulted in clear improvements to and modernization of the facility — for example, the construction of a new control room, the construction of a computerized radiation protection and surveillance control room, the reconstruction of the primary coolant circuit, the complete refitting of reactor instrumentation to incorporate a computerized data acquisition system, the redesign and construction of reactor water treatment plants, improvements in experimental devices and the design and construction of new experimental devices. The reactor, construction of which was resumed in 1986, attained criticality on 6 September last using the HEU fuel available. We are now at the stage of characterizing the reactor by measuring process and nuclear parameters prior to commencing power operation.
REACONDICIONAMIENTO DEL REACTOR EXPERIMENTAL DE LO AGUIRRE

RESUMEN

Se hace una descripción de los principales trabajos que debieron realizarse en el reactor experimental del CEN Lo Aguirre (RECH-2) y que han conducido a su reciente puesta en operación.

En particular, se mencionan los principales cambios, modificaciones y mejoras incorporadas, tanto en el aspecto de cálculos de física del reactor, como en los sistemas y componentes de la instalación, y en la reparación de los elementos combustibles existentes. Especialmente importante ha sido también la definición, aplicación y cumplimiento de requerimientos de seguridad nuclear, acompañados de un consistente programa de garantía de calidad.

Entre las obras realizadas, se destacan algunas que, por su envergadura e importancia, demuestran un evidente progreso y modernización de la instalación: construcción de una nueva sala de control; construcción de una sala de control computarizada de vigilancia y protección radiológica; reconstrucción del circuito de refrigeración primaria; reacondicionamiento completo de la instrumentación del reactor, incorporando un sistema computarizado de adquisición de datos; redeño y construcción de plantas de tratamiento de aguas del reactor, mejoras en los dispositivos experimentales y diseño y construcción de otros.

El reactor, cuya construcción se había reanudado en 1986, alcanzó criticidad el día 06 de Setiembre recién pasado, con el combustible HEU disponible.
Ahora, se enfrenta la etapa de caracterización del reactor, a través de las mediciones de parámetros nucleares y de proceso, como paso a su operación a potencia.

I. INTRODUCCIÓN

En 1973, en virtud de un acuerdo de cooperación con la JEN (Junta de Energía Nuclear de España) se dio origen al proyecto de construir el reactor experimental de 10 MW, con combustible HEU, tipo MTR, en el Centro Nuclear de Lo Aguirre, a 30 km de Santiago de Chile. Los trabajos de diseño se llevaron a cabo en España, con excepción de la parte de edificios, que se prepararon en Chile. Las obras, contratos y suministros se realizaron desde 1974, conduciendo a la primera criticidad del reactor en Febrero de 1977.

Las consiguientes pruebas demostraron que algunos de los sistemas montados del reactor adolecían de defectos (en la instrumentación, tratamiento de aguas, refrigeración), además de existir otros sistemas aún sin su diseño completo y estar, por lo tanto, sin ser instalados (ventilación, refrigeración secundaria, control automático, etc.)

A partir de esa fecha, se procedió a revisar en forma parcial el diseño del reactor y se paralizó toda obra física en él. En paralelo, la asesoría española se interrumpió, razón por la cual el estudio se hizo más lento y dificultoso, por la carencia de cierta información básica. Contribuyó a eso también, la aparición de restricciones en la obtención de combustible HEU y a que se hiciese necesario estudiar, por lo tanto, la conversión del núcleo a combustible LEU.

Luego de diversos intentos de obtener la asesoría extranjera pudo realizarse en conjunto con la JEN (1982-83) una
revisión completa de la ingeniería básica del reactor, obteniéndose un importante avance para la prosecución de los trabajos que se habían ido determinando como necesarios. Incluso, respecto de su conversión a combustible LEU, se logró determinar la real posibilidad de efectuarlo en forma acceptable. Luego de esta etapa, unas nuevas evaluaciones se realizaron respecto de la factibilidad de terminar el reactor.

Vistos los avances, progresos y posibilidades concretas de materializar la obra, en 1985 se resolvió la terminación del reactor para que operase a potencia nominal de 10 MW, con el máximo aprovechamiento posible de los equipos existentes, muchos de los cuales habían provenido de España en el período anterior a 1977. A ese momento (1985) la colaboración de la JEN se había ya extinguido del todo.

Para llevar a cabo las obras de ingeniería, construcción, montaje y pruebas, la Comisión Chilena de Energía Nuclear (CCHEN) dispuso la asignación correspondiente de los recursos humanos y financieros, en base a sus capacidades y a las de la ingeniería nacional, y así se abordó la tarea.

Cuadro 1. Características técnicas del reactor

- Tipo de reactor : MTR, piscina abierta.
- Potencia, MW : 10
- Combustible : Placas planas, UAI\textsubscript{x} - Al
- Moderador, refrigerante : H\textsubscript{2}O
- Refrigeración : régimen convección forzada, flujo descendente.
  Caudal, m\textsuperscript{3}/h : 1.200
  régimen convección natural, hasta 100 kW.
- Paso grilla, cm x cm : 7,76 x 7,76  
- Geometría núcleo : 9 x 9 posiciones  
- Volúmen activo núcleo, l : 102,8 posiciones  
- Reflector radial : Grafito  
  axial : H₂O  
- Absorbente : Boral, 4 placas de control y un elemento de control fino.

- N° de elementos combustibles. : 29  
- Enriquecimiento U-235, % : 89,74  
- Densidad U, gr/cm³ : 0,39  
- N° placas/elemento : 18  
- Espesor placas, mm : 1,52  
- Espesor canal refrigeración mm. : 2,69  
- Espesor cladding, mm. : 0,445  
- U-235/placa, gr : 7,5  
- U-235/elemento, gr : 135

2.1 Cálculos del Núcleo

Fueron totalmente realizados de nuevo, para lo cual contribuyó en forma importante el que Chile haya participado en actividades del programa de reducción del enriquecimiento del combustible para los reactores experimentales.

Los cálculos de física del núcleo, estáticos y dinámicos, han sido hechos por personal de la CCHEN, contándose con colaboración de expertos OIEA para calificar su validez y fiabilidad. Lo mismo, en el caso de los cálculos termohidráulicos en estado estacionario y en análisis de accidentes.

Los códigos usados han sido, EREBUS, CITATION, WIMS-D, TWOTRAN, FINELM, PARET, en la parte neutrónica, y otros como, TRANS V2 para cálculos termohidráulicos. Con ellos, ha sido posible realizar cálculos fundamentales en 2 y 3 dimensiones, en especial para efectividad de placas de control y parámetros cinéticos. La principal limitante que aún permanece, es en la optimización de la obtención de las constantes nucleares. No obstante, la corroboración de los resultados se ha obtenido aprovechando la disponibilidad del reactor RECH-1, en operación en Chile desde 1976, y la comparación con resultados de cálculos benchmark. [1, 2, 3, 4, 5]

En la Fig. 1, se muestra un perfil de flujos neutrónicos en una línea central del núcleo, para una potencia de 2 MW.
2.2. Edificios del Reactor

Se componen del edificio de contención del reactor, que es una estructura de hormigón armado de forma cilíndrica con un domo casquete esférico, y de un edificio auxiliar, adosado al exterior, pero con una independencia estructural autosoportante (Fig. 2). Este, alberga en su planta cuarta, a la nueva sala de control, lugar en donde se encuentran las consolas de mando, en reemplazo de la antigua sala que estaba en el interior del edificio de contención. Ambos edificios están diseñados y construidos para resistir las solicitudes del SSE, y mantengan sus funciones de seguridad.

Poseen sistemas independientes de ventilación. El del edificio de contención es un circuito que en base a 3 parejas de ventiladores, opera en 4 modos, según la condición requerida: normal, emergencias radiológicas (2) y contraincendio. Crea una depresión de 15 mm.c.a.
Fig. 2. Edificio del Reactor
y produce 2 renovaciones/hora en el ambiente interior (Fig. 3).

La sala de control se mantiene a una leve depresión en especial, por el evento de un incendio en el interior del edificio de contención.

En este edificio auxiliar se habilitó 2 zonas de accesos controlados radiológicamente y con los recintos adecuados para cambio de vestuario y descontaminación eventual de personas.

**Protección Radiológica**

Es un recinto contiguo a la sala de control, se encuentra la central computarizada que recoge y procesa las señales para detecciones de gases y aerosoles desde una red de monitores distribuidos en el interior del edificio de contención, y del detector en la chimenea de salida del aire. Este último activa alarmas y provoca el aislamiento del edificio de contención en caso de sobrepasarse el nivel de radiactividad en el aire interior.

2.3. **Circuito Primario de Refrigeración**

Debido a defectos en el diseño hidráulico, fue necesario remover y reemplazar el circuito de tuberías y bombas de circulación, acordes con un diseño óptimo para las condicionantes provenientes del rediseño termohidráulico del reactor. Así, se determinó:

1) caudal de circulación en estado estacionario, de 1.200 m$^3$/h.

2) la evolución ante accidentes tipos LOFA y LOCA, y los niveles de refrigeración en régimen de convección natural.
Fig. 3.- Sistema Ventilación reactor RECH-2
El circuito quedó correctamente configurado en su parte mecánica, alimentación eléctrica normal y de emergencia, y de instrumentación, parte de la cual se vincula con el sistema de protección del reactor.

En la Fig. 4 se muestra un diagrama del sistema de refrigeración completo, consistente en 4 lazos en paralelo, de los cuales uno permanece en stand-by.

Las principales características termohidráulicas se muestran en el Cuadro II.

**Cuadro II. Características termohidráulicas RECH-2.**

- Caudal, $m^3/h$ : 1,200
- Potencia, MW : 2
- Temperatura agua entrada piscina, °C : 35
- Temperatura agua salida piscina, °C : 36,9
- Caudal efectivo por el núcleo, % : 72
- Potencia canal, kW : 3,7
- $ΔP$ núcleo, cm.c.a. : 140
- Temperatura máx. placa, °C : 44
- Caudal máx. C. secundario/lazo, $m^3/h$ : 900

**2.4 Instrumentación**

Este sistema sufrió un completo reacondicionamiento. De partida, todo el hardware existente fue removido, desde el núcleo, sistemas auxiliares, etc., y reinstallado de modo de responder a determinados criterios de seguridad, tales como redundancia, diversidad y de coincidencias, en los canales para la detección de variables nucleares y de protección.
Fig. 4.- Diagrama del Circuito Primario de Refrigeración del RECH-2.
En la Fig. 5 siguiente se muestra un resumen de la estructura básica con que la instrumentación del reactor quedó configurada. Además, en que el sistema de protección se aisló galvanicamente y se incorporó lógica digital en toda la instrumentación; en reemplazo de antiguos registradores, se incorporó un sistema computarizado de adquisición de datos, cuyo software, diseño y montaje fueron realizados por completo en la CCHEN, a partir de un IBM-PC. Con ello, se dispone de un apoyo relevante para la operación. Este sistema muestra periódicamente en forma gráfica y a través de una impresora, los estados de las variables de procesos no nucleares.

2.5 Componentes del Núcleo

Se aprovechó de realizar una completa revisión de los componentes interiores de la piscina del reactor y comprobar su adecuado diseño estructural. En particular, el conjunto de mecanismos de barras de control fue modificado, según el recálculo sísmico dinámico, para facilitar el acceso al núcleo en las maniobras de carga y descarga de elementos.

Otro cambio importante, se realizó en la incorporación de soportes en la cercanía del núcleo para 8 tubos secos que albergan, sendas cámaras de ionización para los canales instrumentales intermedios (escala logarítmica) y de potencia (escala lineal).

La determinación de una cantidad de tubos, forma de los soportes y ubicación, fueron determinados a través de detallados cálculos de flujos neutrónicos [6] y de acuerdo con la sensibilidad correspondiente de los elementos de detección.
Fig. 5. - DIAGRAMA DE INSTRUMENTACION DEL RECH-2

- CANALES DE ARRANQUE
  - CF
    - Instr. Pulsos
  - CANALES LOGARITMICOS
    - CIC
      - Ampl. Log.
  - CANALES LINEALES
    - CA
      - Ampl. Lineal
  - DETECTOR ROTURA DE VAJAS
    - BF3
      - Instr. Pulsos
  - POTENCIA POR N-16
    - C1
      - Amplificador
  - ΔP NUCLEO
    - Tr PESION
  - CAUDAL REFRIGERANTE
    - Tr PESION
  - TEMPERATURA ENTRADA NUCLEO
    - TERMOPAR
      - Transmisor
  - SALTO TERMICO NUCLEO
    - TERMOPARES
      - Transmisor
  - NIVEL DE AGUA
    - SONDA CAPAC
      - Transmisor
  - SENSOR

- LINEA DE PARADA DEL REACTOR
  - C/S
    - 1d
    - Reg 1 td
  - DRV
  - N-16
    - SAD
  - ΔP NUCLEO
    - SAD
  - Q
    - SAD
  - ΔT NUCLEO
    - SAD
  - L
    - SAD
También se aumentó la confiabilidad para el desarrollo del régimen de refrigeración por convección natural, duplicando las válvulas correspondientes existentes en el núcleo.

2.6 Dispositivos Experimentales

Se cuenta con 2 canales radiales de Ø 350 mm y 1 de Ø 600 mm, además de dos canales tangenciales de altos flujos de neutrones térmicos, también de Ø 350 mm.

Se han instalado 2 tubos secos para irradiación en el núcleo y además, 2 dispositivos neumáticos para irradiación in-core de corto tiempo. Estos últimos, están asociados a un laboratorio de análisis por activación neutrónica.

Para producción de radioisótopos se dispone de elementos portamuestras que son introducidos manualmente en el núcleo.

2.7 Documentación

Un aspecto al que se le asignó especial importancia y énfasis, fue el relacionado con la calidad y cantidad de los documentos que respaldan el diseño, montaje, pruebas y operación de los sistemas y componentes del reactor.

Habiéndose organizado la instalación en sistemas consistentes, a cada uno de ellos le corresponde un documento de síntesis llamado "Descripción de Sistema". Complementariamente, están los planos de especialidad asociados a cada sistema, los cálculos y memorias, catálogos de equipos, etc., que configuran el dossier del sistema.
Especial cuidado se puso en la calidad, objetividad y vigencia de la información contenida en esos documentos de ingeniería.

Junto a lo anterior, a cada componente importante para la seguridad para todos los sistemas del reactor se elaboraron "Procedimientos de Prueba", con el fin de verificar el grado de cumplimiento con sus especificaciones de diseño. Esto, dio origen a los respectivos informes de prueba.

De este modo toda la documentación del reactor está comprobadamente vigente y actualizada.

2.8. Análisis Sismológico

Con el fin de demostrar las capacidades resistentes de las estructuras con funciones de seguridad, se realizó un completo estudio geotécnico y análisis sismológico. Este último consistió en caracterizar los sismos de diseño y el espectro básico de respuesta, obtenido del estudio de riesgo sísmico [7].

En dicho estudio sismológico, se generaron cuatro sismos artificiales, eligiéndose aquel histórico, con las modificaciones necesarias, para contemplar los efectos más desfavorables sobre las estructuras a estudiar, como el SSE (Safety Shutdown Earthquake, o sismo de parada segura). Este sismo corresponde a uno de magnitud Ms = 8.5 y con intensidad en Santiago del orden de VIII en la escala de Mercalli. A partir de este SSE se analizaron las estructuras del edificio de contención y del bloque de la piscina, mediante 3 métodos diferentes, determinándose aceleraciones, desplazamientos y tensiones. El más representativo resultó, el método de análisis de respuesta por integración en el tiempo con un modelo de elementos finitos.
Así, resultaron los siguientes valores de aceleraciones de referencia para el diseño sísmico:

Edificio de contención

en la base : 0,3 g
en la cúpula : 0,6 g

Bloque Piscina

En cualquier parte: : 0,3 g

Los resultados de los distintos modelos de análisis concuerdan entre sí y demuestran que, tanto el edificio de contención como el bloque de piscina son suficientemente sismo-resistentes.

2.9 Reparación de los elementos combustibles

Debido a la detección de daños superficiales (rayaduras) y deformación parcial en algunas placas combustibles, se realizó un exhaustivo análisis y se desarmaron todos los elementos combustibles (31), efectuándose una calificación, y descartándose 36 placas cuyo daño superaba una profundidad de 100 µm. [8].

Como consecuencia de ello, se rearmaron sólo 29 elementos, con las placas aceptadas, y que constituyen la carga de combustible actualmente existente.
III. SEGURIDAD NUCLEAR Y GARANTÍA DE CALIDAD

3.1 Clasificación de Seguridad

El desarrollo de los trabajos se encuadró en formas y detallados aspectos y requerimientos de seguridad nuclear, para lo cual se implementó un programa de Garantía de Calidad (QA) extensivo a todos los componentes y sistemas del reactor, tuvieran o no asignadas funciones de seguridad. Una clasificación de seguridad y de garantía de calidad se aplicó para los elementos del reactor, la cual, determinó el alcance de las exigencias de diseño y de operación [9].

La clasificación aplicada se basó en: Sísmica, de Seguridad y de Calidad:

Como resultado, se obtiene un Código de Clasificación para los componentes y elementos del reactor:

<table>
<thead>
<tr>
<th>Código</th>
<th>Clase Sísmica</th>
<th>Clase G. de C.</th>
</tr>
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<tbody>
<tr>
<td>A1</td>
<td>1</td>
<td>1</td>
</tr>
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<td>B1</td>
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<td>B2</td>
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<tr>
<td>C1</td>
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<td>1</td>
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<tr>
<td>C2</td>
<td>2</td>
<td>1</td>
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<tr>
<td>E1 (Eléctrico)</td>
<td>1</td>
<td>1</td>
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<tr>
<td>E2 (Eléctrico)</td>
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<td>N</td>
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3.2 Programa de Calidad

El programa de Garantía de Calidad se basó en un Manual de G. de C. y en un conjunto de procedimientos para cada actividad relevante: elaboración de informes; pruebas de componentes y sistemas; archivo y tramitación de documentos; clasificación, identificación de planos,
etc. Además, se aplicó un sistema de inspecciones y auditorías periódicas, con todo lo cual la base documental y el resultado de la obra ha quedado cabalmente fundamentada.

3.3 Análisis de Accidentes

Está principalmente estructurado sobre la base de impedir la liberación importante de productos de fisión que ponga en riesgo al personal de operación y a la población, como resultado de una fusión parcial o total del núcleo. Por ello, se persigue la integridad de las 3 barreras de protección:

- vaina combustible
- frontera del refrigerante,
- edificio de contención.

Se ha aplicado, en general, el método determinístico para la identificación de los eventos iniciadores de accidentes, analizándose en detalle todos aquellos que carezcan de uno de mayor relevancia o envolvente.

Criterios conservadores se usaron en la determinación de los límites y condiciones de operación, como también en los de seguridad, de todas las variables asociadas al sistema de protección del reactor.

Los tipos de accidentes analizados se refieren a:

a) del tipo LOFA: corte de energía eléctrica, fallo de bombas, etc.

b) del tipo LOCA: rotura de cañerías del circuito primario (se impide que el núcleo quede al descubierto).
c) inserciones de reactividad: errores en carga de combustible, extracción incontrolada de una barra control, etc.

3.4. **Principales aspectos de seguridad**

Se resumen algunos de los aspectos o condiciones de diseño y operación más importantes, que apuntan a conseguir la seguridad del reactor:

- está impedido, por un sifón, que ante una rotura de las cañerías del circuito primario, el núcleo quede desnudo; es decir, no sumergido en el agua de la piscina.

- la extracción de las placas de control está impedida de hacerla con todas ellas simultáneamente.

- está impedido, a la partida, extraer placas de control si no hay un nivel de cuentas mínimo en canales de arranque.

- existe un selector de modo de operación (a potencia baja en régimen de convección natural, o a alta potencia en regimien de refrigeración por convección forzada) que inhabilita la extracción de barras si no hay correspondencia con la posición en el núcleo de las compuertas para convección natural.

- la temperatura del agua a la entrada del núcleo es constante, de modo de tener permanentemente y en toda condición térmica, un control sobre el margen DNB 27°C como límite operacional.

- en el cálculo termohidráulico se han empleado factores de incertidumbre, según método sum-estadístico. Se fijó como límite de seguridad inicial para la placa combustible, 85°C.
IV. RESULTADOS Y CONCLUSIONES

El 06 de Setiembre recién pasado, el reactor RECH-2 fue puesto nuevamente crítico, luego que todo el proceso de reacondicionamiento fuera terminado.

Se llevó a cabo un experimento de aproximación a crítico que, naturalmente tuvo en cuenta la información obtenida en el experimento de 1977, con el agregado que muchos datos fueron mejorados o cambiados en esta última etapa (por ejemplo, los nuevos materiales empleados en la reparación del combustible).

Fig. 6 CURVAS DE APROXIMACION A CRITICO RECH-2
(106-09-89)
A partir de ahora, se llevan a cabo todas las mediciones y caracterización del reactor crítico, como etapa previa a su próxima operación a potencia. Se ha diseñado un programa de incorporación de sondas para mediciones de los flujos neutrónicos en el núcleo, a partir de los cuales, se obtendrá la información para comprobar los resultados de los cálculos. De ese modo, a partir de 1990 el reactor del CEN Lo Aguirre (RECH-2) estará operando a regimen según sus características técnicas [10].
REFERENCIAS


OPERATIONAL SAFETY AND REACTOR LIFE IMPROVEMENTS
OF KYOTO UNIVERSITY REACTOR

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ABSTRACT

Recent important experiences on the operational safety and successful works for the operational life improvements of the reactor are described in the case of Kyoto University Reactor (KUR) of 25 years old 5 MW light water reactor provided by two kinds of thermal columns of graphite and heavy water as well as other various kinds of experimental facilities. In the graphite thermal column, noticeable amounts of neutron irradiation effects were accumulated in the graphite blocks near the core. Before the possible release of the stored energy, all the graphite blocks in the column were successfully replaced with new blocks at the chance of the installation of a liquid deuterium cold neutron source in the column. At the same time, special sealing mechanisms were also provided for the essential improvement on the problem of the radioactive argon produced in the column. In the other thermal column using heavy water, we have accomplished a successful repairing for a slow leakage of the heavy water through a thin instrumentation tube failure. The repairing work included the removals and reconstructions of the lead and graphite shielding layers and welding sealing of the instrumentation tube under radiation fields. Several mechanical components in the reactor cooling system were also exchanged to new components with improved designs and materials. On-line data loggings of almost all instrumentation signals are continuously performed with a high speed data analyses system for the diagnosis of the operational conditions of the reactor. Furthermore, through detailed investigations on critical components, the operational safety for further extended reactor life will be supported by well scheduled maintenance programs.
1. INTRODUCTION

The Kyoto University Reactor (KUR, a light water moderated 5 MW reactor built in 1964) is widely used by researchers from universities and institutes in the whole over Japan, through the semiannual research application system. The structure of the reactor is provided by a graphite thermal column, a heavy water facility and 8 beam and exposure tubes. Figure 1 shows the vertical section of the reactor after the recent completion of the modernization program, which includes the insertion of a liquid deuterium cold neutron source and constructions of 6 neutron guide tubes (2 for thermal neutrons, 3 for cold and 1 for very cold neutrons). The present report describes recent important experiences on the operational safety and successful works for the reactor life improvements. A more detailed description on the performances of the cold source will be given in another contribution in the present Symposium.

2. MODIFICATION OF THE GRAPHITE THERMAL COLUMN

The graphite thermal column has been used long time for neutron beam experiments (for examples, the spectrum measurements of neutrons emerging from the graphite, neutron wave propagation experiments and also several neutron scattering experiments) as well as many irradiation experiments in the well thermalized neutron field. However, there were growing demands for the modernization of the column, especially for providing cold neutron source for a recent trend of the neutron beam experiments. In the irradiation aspects also, the original thermal column necessitates the shutdown of the reactor for the insertion and extraction of an irradiation sample, and therefore, they needed a more convenient irradiation device with a pneumatic system in the position of the high cadmium ratio.
An additional motivation for the modification of the column comes from the possible effects due to the accumulated neutron irradiations for the many years in the graphite blocks near the reactor core.

In our project, the graphite blocks in the column were completely replaced to new ones, and at the same time, the column structure was completely improved for the purpose to insert a cold neutron source accompanying cold and very cold neutron beam holes and a pneumatic irradiation tube, as shown in Fig. 2. According to the safety considerations in the design of the modified column, the new graphite column consists of a fixed and a movable portions, the former constructed in a tunnel-like feature and the latter assembled in a place outside the reactor to cover the complicated in-pile components of the cold source in conjunction with the shielding door, as shown in Fig. 3, and thereafter inserted into the former structure. Further, all the graphite blocks are connected each other by using connecting pins and rods for maintaining sufficient mechanical strength and seismic resistivity as the whole. As the results of the present safety considerations and methods, the modified column was completed with the minimized radiation leakage after the cold source insertion accompanying increased number of beam tubes.

Another important improvement is the sealing mechanism for minimizing the radioactive argon leakage produced in the column. The radioactive argon from the thermal column was the major contribution in the stack gas as well as the reactor room gas monitorings. Providing the specially designed sealing mechanisms for all of the penetrations and the beam windows in the column, and further the use of several stages of decay tanks, the contributions were significantly diminished after the completion of the new thermal column provided by a low flow-rate argon sweep line.

3. INSPECTIONS AND REPAIRING OF THE HEAVY WATER FACILITY

The heavy water facility of KUR is used mainly for thermal neutron irradiations under low epithermal neutron
and gamma rays backgrounds, in biological researches and some medical studies such as neutron capture cancer therapy. In recent periodical inspections, however, a gradual increase of tritium contamination was noticed in the water of the sub-pool located opposite side to the heavy water facility in the reactor. After a number of detailed analyses of several records and investigations concerning the contamination distributions in the reactor room, we arrived the conclusion that the contamination should come from a slow leakage of the heavy water in the facility. Further efforts with the tritium monitoring following various changes of the heavy water level in the facility indicated the leakage spot to be a thin instrumentation tube for a thermocouple shown in Fig. 4.

A repairing program was decided to be four stage works consisting of i) removing the shielding components of the graphite and the lead layers in front of the heavy water tank, ii) cutting off the instrumentation flange at the middle portion of the nozzle and sealing the opening with a pin and welding, iii) decontaminations of the surrounding components, especially of the graphite layer containing a fraction of the escaped heavy water, and iv) reconstruction of the facility.

Before the start of these works, very careful estimations and several considerations for diminishing the radiation dose were carried out. One of the most effective ideas was the special preparation of an auxiliary shielding block of 15 cm thick iron completely fitting to the facility. The stainless steel bolts connecting the heavy water tank to the reactor core tank were also replaced to new ones before the welding works. Even after these precautions, the welding of the nozzle opening must be done under quite difficult conditions of very narrow space in considerable radiation field. Many times of rehearsals by the welder using a mock-up set of the same geometrical situations and the same materials led to the complete success of the welding passed the highest class examinations including penetration test and helium leak test.
The decontamination of the graphite layer was satisfactorily carried out for a long time baking process by enclosing the layer in a specially prepared air-tight vessel. Thereafter, the shielding components of the facility were completely reconstructed.

Temperature raising tests of the facility in the conditions of light water in the tank and thereafter of heavy water loading showed the complete tightness of the heavy water system. Further, repeated long time operations of the reactor at the full power level have now proven the final success of the repairing of the facility.

4. ADDITIONAL EXPERIENCES

In KUR, three heat exchangers are used for the cooling of the 5 MW thermal power. The heat exchanger in the original design consists of a soft steel shell and stainless steel tubes. After long time uses beyond 20 years, some kinds of corrosion phenomena brought a pin hole type failure in a narrow air-vent nozzle of soft steel portion containing the secondary water. Thereafter, these heat exchangers were replaced to those of all stainless steel type.

As one of the interesting phenomena accompanying continuous operations of the cooling system, apparent tendency of gradual enrichments of chlorine ions contained in the feed water is observed in the secondary coolant. Therefore, the concentration is carefully monitored before and after the periodical reactor operations.

In KUR, they have a many years research experiences of the reactor noise analyses related to the safety studies. Based on such a history, a computer system for the on-line data loggings of almost all reactor operation parameters such as neutron flux, control rods positions, coolant flow rates, coolant temperatures and so on have been accumulated, and these data are analysed with various kinds of statistical approaches.
5. CONCLUDING REMARKS

Through these recent experiences on the reactor safety and operational life improvements, we noticed again the importance of many years experiences of the reactor operation and maintenances including overhauls and components replacements as well as the accumulations of the research activities by the staff in the fields of reactor engineering and nuclear technology. A continuous efforts for the safety and the system improvements will be essential for the effective existences of the research reactor facility.

Acknowledgement

The most of the difficult works for the heavy water thermal column described in the present paper were carried out by the tight and direct collaborations of the staff of many related divisions in the Research Reactor Institute. The highest skill welding of the instrumentation tube was performed by the original manufacturer of the facility, Hitachi Manufacturing Company, Ltd.
FIGURE CAPTIONS

Fig. 1: Vertical section of KUR

Fig. 2: Vertical section and plan of the modernized graphite thermal column

Fig. 3: Movable graphite assembly including the cold neutron source and connected with the shielding door

Fig. 4: Heavy water tank of the facility showing the repairing portion
Experimental facilities of KUR (vertical cross section)

DC: Heavy Water Facility
TC(CNS): Graphite Thermal Column, Cold Neutron Source
Pn-1~3: Pneumatic Tubes
SI: Slant Exposure Tube
Hyd: Hydraulic Conveyor Tube
1: MOVABLE GRAPHITE ASSEMBLY
2: SHIELDING DOOR
3: SEISMIC-RESISTIVE TABLE
ПРОЕКТ РЕАКТОРА СФИНКС
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THE SPHINX REACTOR FOR ENGINEERING TESTS

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ABSTRACT

A research reactor known as SPHINX is under development in the USSR. The reactor will be used mainly to carry out tests on mock-up power reactor fuel assemblies under parameters close to normal parameters in experimental loop channels installed in the core and reflector of the reactor, as well as to test samples of structural materials in ampoule and loop channels. The SPHINX reactor is a channel-type reactor with light-water coolant and moderator. The maximum achievable neutron flux density in the experimental channels (cell composition 50% Fe, 50% H₂O) is 1.1 x 10¹⁵ neutrons/cm² · s for fast neutrons (E > 0.1 MeV) and 1.7 x 10¹⁵ for thermal neutrons at a reactor power of 200 MW. The design concepts used represent a further development of the technical features which have met with approval in the MR and MTR channel-type engineering test reactors currently in use in the USSR. The "in-pond channel" construction makes the facility flexible and eases the carrying out of experimental work while keeping discharges of radioactivity into the environment to a low level. The reactor and all associated buildings and constructions conform to modern radiation safety and environmental protection requirements.
ПРОЕКТ РЕАКТОРА СФИНКС ДЛЯ ИНЖЕНЕРНЫХ ИСПЫТАНИЙ

В СССР ведется разработка проекта исследовательского реактора СФИНКС. Основные направления использования реактора — проведение испытаний макетных ТВС энергетических реакторов с обеспечением параметров, близких к натурным, в экспериментальных петлевых каналах, установленных в активной зоне и отражателе реактора, а также испытания образцов конструкционных материалов в ампульных и петлевых каналах.

Реактор СФИНКС относится к классу канальных реакторов, теплоносителем и замедлителем в нем служит легкая вода. Максимальная плотность нейтронных потоков, достижимая в экспериментальных каналах (состав ячейки 50 % Fe, 50 % H₂O) до 1,1 \* 10⁵ нейтр./см² с по быстрым нейтронам (E > 0,1 МЭВ) и до 1,7 \* 10¹⁵ по тепловым нейтронам при мощности реактора 200 МВт.

Заложенные в проекте реактора концепции являются дальнейшим развитием технических решений, которые апробированы в действующих в СССР канальных реакторах для инженерных испытаний МР и МИР.

Устройство реактора по типу "каналы в бассейне" обеспечивает гибкость и удобство проведения экспериментальных работ и низкие выходы активности в окружающую среду.

Реактор, а также связанные с его эксплуатацией здания и сооружения выполняются с учетом современных требований по обеспечению безопасности и защиты окружающей среды.
ВВЕДЕНИЕ

В последние годы отмечается нарастание разрыва между потребностями в проведении материаловедческих экспериментальных работ и петлевых испытаний, связанных с разработкой топлива для энергетических ЯР, и возможностями, которые предоставляют действующие в СССР реакторы для материаловедческих и петлевых испытаний (МР, МИР, СМ-2).

Это явилось основным фактором, определяющим необходимость разработки исследовательского реактора СФИНКС.

Принято, что главными направлениями использования реактора будут проведение испытаний в экспериментальных петлевых каналах макетных ТЭС энергетических реакторов с обеспечением параметров, близких к натурным, а также испытания образцов конструкционных материалов в ампульных и петлевых каналах в условиях высоких нейтронных потоков. В то же время характеристики реактора и его конструкция дают возможность проведения на нем широкого круга экспериментальных и прикладных работ - производство радиоактивных изотопов, нейтронная радиография топливных элементов и сборок, нейтронно-активационный анализ, физические эксперименты на выведенных нейтронных пучках и т.п.

I. ФИЗИЧЕСКИЕ И КОНСТРУКТИВНЫЕ ОСНОВЫ РАЗРАБОТКИ РЕАКТОРА

По своему основному конструктивному решению реактор выполняется как многопетлевая установка, обеспечивающая одновременное ведение нескольких разнохарактерных экспериментов.
Исходя из условий удобства и безопасности работы реактора с петлевыми каналами, которые могут содержать теплоноситель высоких параметров, было принято решение активной зоны в канальном исполнении, при котором тепловыделяющие сборки (ТВС) размещаются в отдельных каналах, охлаждаемых водой под давлением.

Для обеспечения в экспериментальных устройствах реактора, размещаемых в активной зоне и отражателе, требуемых условий облучения, а также с целью достижения максимальной широты и разнообразия экспериментальных возможностей, в конструкции реактора реализуются следующие принципы:
- высокая удельная мощность, снимаемая с ТВС, за счет развитой поверхности теплосъема, охлаждения ТВС холодной водой под давлением, высокой скорости теплоносителя в активной зоне;
- возможность размещения экспериментальных каналов различного размера в любых ячейках решетки активной зоны, причем каждый экспериментальный канал окружен по крайней мере одним полным рядом рабочих каналов;
- снижение количества замедляющих нейтронов сред в активной зоне реактора за счет плотной упаковки рабочих каналов и установки в полости в"узлах"решетки между рабочими каналами алюминиевых вытеснителей;
- размещение в активной зоне распределенной системы устройств воздействия на локальные размножающие свойства (двухпозиционных поглощающих стержней вокруг каждого облучательного устройства и внутри любой ТВС);
- организация в отражателе двух рядов облучательных объемов различного диаметра.
В качестве ТВС, применяется конструкция из концентрических трубчатых теплоп проведителя элементов, аналогичная ТВС реакторов МИР [1] и МР [2], но с более развитой поверхностью теплосъема.

Размещение рабочих каналов в виде скрученного пучка (аналогично примененному в реакторе БР-2 [3]) обеспечивает близкую к максимально возможной для канальных реакторов плотность активной зоны и, одновременно, необходимую свободу в установке экспериментальных устройств и обслуживании оборудования реактора.

Принятая высота активной зоны реактора (800 мм) позволяет проводить достаточно корректное исследование длинномерных объектов облучения. При необходимости, для проведения специфических экспериментов, высота активной зоны реактора путем замены ТВС может быть увеличена до 1000 мм.

2. КОНСТРУКТИВНЫЕ ОСОБЕННОСТИ РЕАКТОРА

Реактор, включая активную зону, отражатель, систему подвода и отвода воды I контура, вертикальные экспериментальные каналы, размещены в бассейне с водой, который выполнен в бетонном массиве здания (рис.1,2).

Бассейн реактора и находящийся над ним реакторный зал окружен защитной герметичной оболочкой (контейнментом), который снабжен по тракту транспортировки облученных изделий шлюзовой камерой, обеспечивающей герметичность по воде. Ниже отметки реакторного зала контейнмент переходит в массив здания реактора, обеспечивающий необходимую прочность и герметичность.

Бассейн реактора соединен с заполненным водой транспортным коридором, который является начальным участком тракта в здании.
реактора для транспортировки отработанных ТВС, петлевых каналов и других облученных изделий на хранение, разделку и исследование.

Проем поворотного круга служит для обеспечения доступа к головкам экспериментальных каналов и проведения всех работ по обслуживанию реактора из реакторного зала.

Во время работы реактора проем поворотного круга и транспортный коридор закрыты стальными съемными защитными плитами.

В шахте бассейна под уровнем воды расположены камеры разъемов экспериментальных петлевых каналов, в которых размещены места соединения (разъемные или сварные) трубопроводов петлевых каналов со стационарными трубопроводами контуров экспериментальных петлевых установок.

Тепловыделяющие сборки (ТВС) представляют собой систему коаксиальных трубчатых твэлов, объединенных концевыми деталями. Каждый твэл представляет собой трубу, состоящую из топливного слоя, внешнего и внутреннего герметизирующего покрытия из алюминиевого сплава. Суммарная толщина наружного твэла - 1,5 мм, остальных - 1,25 мм. Минимальная толщина оболочки составляет соответственно 0,4 и 0,3 мм. В качестве материала топливного сердечника используется металллокерамическая топливная композиция (\( UO_2 + Al \)). Обогащение топлива по урану-235 - 90 %. Высота активной части ТВС 800 мм.

Для реактора разработаны ТВС двух типов: с десятью и с семью твэлами. ТВС первого типа имеют центральную полость, в которую может быть установлен вытеснитель диаметром 12,5 мм или стержень с выгорающим поглотителем. Содержание урана-235 в такой ТВС - 536 г. ТВС второго типа имеют центральную полость для установки экспериментального устройства диаметром 30 мм.
В качестве теплоносителя I контура используется вода с давлением на входе в активную зону 2,0 МПа (20 кгс/см²). Охлаждение ТВС осуществляется прямоточным движением теплоносителя сверху вниз. Температура воды на входе 45 °С, на выходе 90 °С, скорость на твэлах около 10 м/с.

Рабочие каналы в центральном сечении активной зоны размещаются по треугольной сетке с шагом 80 мм.

Для обеспечения в экспериментальных устройствах, размещенных в активной зоне, высокой плотности потока быстрых нейтронов замедляющая среда (вода) из пространства между рабочими и экспериментальными каналами в пределах активной зоны максимально удалена вытеснителями, изготовленными из слабопоглощающих материалов (бериллия и алюминиевых сплавов).

Рабочий канал состоит из трубы, выполненной из циркониевого сплава, с наружным диаметром 78 мм и толщиной стенки 1,5 мм, головки и хвостовика. Внутри канала завальцовано кольцо, служащее опорой для ТВС. Головка предназначена для центрирования, опоры и фиксации рабочего канала в расточках подводящего коллектора, хвостовиком рабочий канал центрируется в стояке отводящего коллектора.

Отверстия в верхней плите подводящего коллектора для установки рабочих каналов и ТВС закрыты съемными пробками.

Герметизация I контура обеспечивается кольцевыми резиновыми уплотнениями, установленными на пробках подводящего коллектора и на головках и хвостовиках рабочих каналов.

Конструкция подводящего коллектора обеспечивает установку в активную зону реактора широкого набора экспериментальных каналов (фильдовского типа) по диаметрам: в центральную ячейку — диаметром до 240 мм; в шесть периферийных ячеек — до 128 мм, или одновременную
установку до 19 экспериментальных каналов наружным диаметром 78 мм. При этом каждый экспериментальный канал окружен полным рядом ТВС. Необходимое пространство в активной зоне для установки экспериментальных каналов различного диаметра обеспечивается извлечением одного или группы рабочих каналов, при этом отверстия в нижней плите коллектора, образовавшиеся при извлечении группы рабочих каналов, закрываются специальными заглушками.

Предусмотрена возможность установки в периферийные ячейки активной зоны до шести прямоточных \( \mathcal{U} \)-образных экспериментальных каналов. В этом случае для вывода горизонтального участка отводящего трубопровода прямоточного канала один из рабочих каналов, окружающих каждый \( \mathcal{U} \)-образный канал, заменяется на вытеснитель.

Отводящий коллектор выполнен общим для всех рабочих каналов, однако наличие в коллекторе отдельных стояков (причем каждый стояк снабжен соплом и импульсными трубками для контроля расхода через рабочий канал, а также имеет отборы для отвода теплоносителя) позволяет обеспечить индивидуальный контроль расхода, температуры теплоносителя и контроль герметичности ТВС в каждом канале.

Расположение рабочих каналов в реакторе в форме "скрученного" пучка характеризуется следующими данными: минимальный шаг между каналами выполнен в центральной плоскости активной зоны и равен 80 мм (при наружном диаметре канала 78 мм); шаг в районе подводящего коллектора - 200 мм. При этом на верхнем и нижнем торцах активной зоны, на расстоянии 400 мм от ее центра, шаг между осями каналов составляет 81,7 мм.

Для выбранного пучка максимальный угол наклона рабочих каналов к вертикали составляет 11°, а для сменных блоков бериллиевого отражателя он равен 14°.
Рабочие органы системы управления и защиты (СУЗ) реактора размещаются в каналах СУЗ, устанавливаемых в рабочие каналы вместо ТВС. Рабочий орган СУЗ (РО СУЗ) состоит из двух частей - поглотителя и вытеснителя. Поглотитель представляет собой сборку в виде "беличьего колеса" из стержней диаметром 4,5 мм, содержащих карбид бора, материал покрытия - коррозионностойкая сталь толщиной 0,3 мм. Материал вытеснителя - алюминиевый сплав.

Для перемещения РО СУЗ используются индивидуальные шаговые электромагнитные приводы, которые устанавливаются на подводящем коллекторе реактора. Конструкция узла крепления привода СУЗ обеспечивает установку РО СУЗ с приводом в любую ячейку активной зоны.

Общее количество исполнительных органов СУЗ - до 25, из них до 21 - компенсирующих, три - аварийной защиты, один - автоматический регулятор.

В качестве дополнительных средств компенсации в реакторе предусмотрено использование двухпозиционных регуляторов, устанавливаемых внутрь ТВС в любом рабочем канале, а также поглотителей, которые могут устанавливаться в пространство между рабочими каналами вместо вытеснителей из алюминиевого сплава, или в вытеснители экспериментальных каналов.

Двухпозиционный регулятор в рабочем канале состоит из поглощающего стержня, устройства для перемещения стержня и головки. Головка двухпозиционного регулятора по типу уплотнения и крепления на подводящем коллекторе реактора аналогична съемной пробке рабочего канала. Перемещение стержней двухпозиционных регуляторов осуществляется дистанционно с помощью специального инструмента по мере выгорания активной зоны. Частота проведения этих операций - до 2 раз в сутки.
Детекторы СУЗ располагаются в воде за пределами обечайки отражателя напротив активной зоны и практически равномерно охватывают активную зону.

Активная зона реактора окружена боковым отражателем, состоящим из сменных бериллиевых и стационарных бериллиевых и графитовых блоков. Графит очехлован алюминием толщиной 4 мм. В блоках отражателя имеются ячейки для установки экспериментальных устройств с наружными диаметрами от 78 мм до 200 мм.

Для введения нейтронных пучков реактор оснащен тремя горизонтальными каналами (ГЭК) — одним тангенциальным и двумя радиальными. Плотность потока тепловых нейтронов в отражателе в районе расположения ГЭК до $9 \cdot 10^{14}$ н/см$^2$·с.

Охлаждение блоков отражателя, а также вытеснителей и поглотителей в межканальном пространстве активной зоны осуществляется водой бассейна, которая прокачивается через зазоры в под отражателем и по двум отводящим трубопроводам направляется в помещение контура бассейна реактора.

Заложенные в проекте реактора СФИНКС концепции в максимальной степени основаны на технических решениях, апробированных в ходе длительной эксплуатации действующих в СССР канальных реакторов для инженерных испытаний ИР и МЭР в необходимой мере являются их развитием с учетом повышения эффективности и безопасности эксплуатации реактора.
3. ОСНОВНЫЕ ФИЗИЧЕСКИЕ И ТЕХНИЧЕСКИЕ ХАРАКТЕРИСТИКИ РЕАКТОРА

Таблица I

Основные характеристики реактора

<table>
<thead>
<tr>
<th>Параметр</th>
<th>Значение</th>
</tr>
</thead>
<tbody>
<tr>
<td>Тип реактора</td>
<td>Канальный в бассейне</td>
</tr>
<tr>
<td>Режим работы</td>
<td>Стационарный, маневренный</td>
</tr>
<tr>
<td>Максимальная тепловая мощность, МВт</td>
<td>200</td>
</tr>
<tr>
<td>Топливо</td>
<td>$UO_2+Al$</td>
</tr>
<tr>
<td>Обогащение по урану-235, %</td>
<td>90</td>
</tr>
<tr>
<td>Тип твэла</td>
<td>Трубчатый</td>
</tr>
<tr>
<td>Загрузка урана-235, кг</td>
<td>22,5</td>
</tr>
<tr>
<td>Замедлитель и теплоноситель</td>
<td>Дистиллированная вода $Be$ + графит</td>
</tr>
<tr>
<td>Отражатель</td>
<td>0,8</td>
</tr>
<tr>
<td>Высота активной зоны, м</td>
<td>до 45</td>
</tr>
<tr>
<td>Количество ТВС</td>
<td>$1.7 \times 10^{15}$</td>
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<tr>
<td>Максимальная плотность потока тепловых нейтронов в активной зоне, $I/cm^2 \cdot s$</td>
<td>$1.1 \times 10^{15}$</td>
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<tr>
<td>Максимальная плотность потока быстрых нейтронов ($E &gt; 0.1$ МэВ) в активной зоне, $I/cm^2 \cdot s$</td>
<td>25</td>
</tr>
<tr>
<td>Продолжительность кампании, эф. сут</td>
<td>25</td>
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<tr>
<td>Среднее выгорание топлива в выгруженных ТВС, %</td>
<td>40</td>
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<tr>
<td>Количество рабочих органов СУЗ</td>
<td>до 25</td>
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Продолжение табл. I

<table>
<thead>
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<td>в том числе:</td>
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<tr>
<td>AR</td>
<td>I</td>
</tr>
<tr>
<td>КС</td>
<td>до 2I</td>
</tr>
<tr>
<td>ЛЗ</td>
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<td>Количество ионизационных камер</td>
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<td>Параметры контура охлаждения активной зоны:</td>
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<td>давление теплоносителя в подводящем коллекторе, МПа</td>
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<td>максимальный расход теплоносителя через активную зону, м³/ч</td>
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<td>скорость охлаждающей воды на твэлах, м/с</td>
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<td>температура теплоносителя на входе в ТВС, °C</td>
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<tr>
<td>температура теплоносителя на выходе из ТВС, °C</td>
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</tr>
<tr>
<td>Параметры контура охлаждения отражателя:</td>
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<td>расход теплоносителя, м³/ч</td>
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</tr>
<tr>
<td>температура теплоносителя на входе в отражатель, °C</td>
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</tr>
<tr>
<td>температура теплоносителя на выходе из отражателя, °C</td>
<td>50</td>
</tr>
</tbody>
</table>
4. ОСОБЕННОСТИ ФИЗИЧЕСКОГО И КОНСТРУКТИВНОГО УСТРОЙСТВА РЕАКТОРА, НАПРАВЛЕННЫЕ НА ПОВЫШЕНИЕ ЕГО БЕЗОПАСНОСТИ

Выбор конструктивной схемы реактора с погружением активной зоны и коммуникаций I контура в бассейн с водой определен вопросами обеспечения безопасности реактора при нормальной эксплуатации и в аварийных ситуациях.

Бассейн реактора со значительными запасами воды является не только средством радиационной защиты при перегрузках, но также эффективным теплоприемником и источником воды для охлаждения активной зоны в аварийных ситуациях.

Для обеспечения локализации радиоактивных продуктов деления в случае возникновения аварийной ситуации бассейн реактора и реакторный зал окружены защитной герметичной оболочкой.

Бассейн не имеет нижних выводов и окружен массивной бетонной стенкой, в связи с чем обеспечено охлаждение активной зоны водой в большинстве аварийных ситуаций.

Течь через горизонтальные каналы не представляет опасности для охлаждения активной зоны, в связи с наличием отдельного герметичного контура охлаждения ТЛС. Кроме того, предусмотрена система сбора и возврата протечек в бассейн.

Крепление подводящего коллектора рассчитано на вертикальное усилие, обусловленное тем, что рабочие каналы в месте своего уплотнения не воспринимают осевых усилий от рабочего давления. Такая конструкция исключает возникновение дополнительных усилий на подводящий коллектор в случае разрушения рабочих каналов.

Подводящие трубопроводы на входе в коллектор имеют обратные клапаны, предотвращающие потерю теплоносителя I контура в случае разрыва подводящего трубопровода.
I контур реактора имеет систему аварийного охлаждения активной зоны, включающую аварийные емкости с запасом воды, насосы аварийной циркуляции и насосы возврата протечек в контур.

Конструкция активной зоны и отражателя исключает непредусмотренные взаимные перемещения элементов, влияющих на реактивность (рабочих каналов с ТВС, блоков берилиевого отражателя, внутризонных вытеснителей, регулирующих и поглощающих стержней) в процессе нормальной эксплуатации.

Кроме того, конструкция рабочих каналов предусматривает фиксацию ТВС в осевом направлении, что исключает массовый подброс ТВС в рабочих каналах при разгерметизации напорного коллектора.

Помимо штатной СУЗ, в реакторе имеется возможность разместить разветвленную систему малогабаритных датчиков внутриреакторного контроля. Эта система, а также индивидуальный контроль температуры и расхода теплоносителя в каждом рабочем канале реактора обеспечивают надежную защиту ТВС от локального перегрева.

Достижению этой же цели помогает наличие физмодели реактора, на которой для каждой новой конфигурации загрузки активной зоны можно предварительно уточнить величину и характер неравномерности энерговыделения в активной зоне реактора.

Задача обеспечения безопасности реактора с учетом работы в нем петлевых каналов с теплоносителями высоких параметров была одним из основных факторов, определивших выбор принципиально-го конструктивного решения реактора, с выполнением активной зоны из отдельных рабочих каналов. Несущие трубы рабочих каналов обеспечивают достаточно надежную защиту ТВС реактора при многих аварийных ситуациях с петлевыми каналами.
В то же время, ряд мероприятий по обеспечению безопасности реактора будет принят при разработке петлевых каналов и петлевых контуров. Так, петлевые каналы предполагается оснастить защитными кожухами и трактами сброса, что позволит предотвратить неблагоприятные воздействия аварий в петлевых каналах на безопасность реакторной установки.

ЛИТЕРАТУРА

1 - Бак бассейна реактора, 2 - Канал рабочий, 3 - Отражатель, 4 - Коллектор подводящий, 5 - Коллектор отводящий, 6 - Исполнительный механизм СУЗ, 7 - Канал экспериментальный петлевой, 8 - Канал экспериментальный, 9 - Круг поворотный, 10 - Плита откатная, 11 - Камера петлевых разъемов.
РАЗРЕЗ ПО ГОРИЗОНТАЛЬНЫМ ЭКСПЕРИМЕНТАЛЬНЫМ КАНАЛАМ

РИС. 2

1 - Канал горизонтальный радиальный,
2 - Канал горизонтальный тангенциальный,
3 - Шибер.
ОСОБЕННОСТИ РЕКОНСТРУКЦИИ РЕАКТОРА ИРТ-2000 В СОФИИ (НРБ)

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Архангельский Н.В.

Государственный комитет по использованию атомной энергии СССР, г. Москва
DAY FOUR

1989 October 26
UPGRADING OF THE IRT-2000 REACTOR IN SOFIA (BULGARIA)

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ABSTRACT

In this paper the design features and experimental capabilities of the IRT-5000 pool-type research reactor (upgrading of the IRT-2000 reactor in Sofia) are reviewed; in this reactor, IRT-2M-type fuel assemblies containing 36% enriched uranium are used instead of cassettes with EhK-10 fuel elements. The main characteristics of the IRT-2M fuel assemblies are given, as are the technical and physical characteristics of the IRT-5000 reactor. Use of the IRT-2M-type fuel assemblies in the IRT-5000 will make possible operation of the reactor with a core volume of 60–83 litres at 7.5 MW, and will raise the maximum thermal neutron flux density to $1.5 \times 10^{18} \text{ m}^{-2} \cdot \text{s}^{-1}$. The design features employed will significantly improve the operational safety of the reactor.
ОСОБЕННОСТИ РЕКОНСТРУКЦИИ РЕАКТОРА ИРТ-2000 В СОФИИ (НРБ)

АННОТАЦИЯ

Рассматриваются конструктивные особенности и экспериментальные возможности исследовательского реактора бассейнового типа ИРТ-5000 (проект реконструкции реактора ИРТ-2000 в Софии), в котором вместо кассет с тепловыделяющими элементами (твэлами) ЭК-10 используются тепловыделяющие сборки (ТВС) типа ИРТ-2М с ураном 36%-ного обогащения. Приводятся основные характеристики ТВС ИРТ-2М, технические и физические характеристики реактора ИРТ-5000. Использование в ИРТ-5000 ТВС типа ИРТ-2М обеспечивает возможность его эксплуатации с объемом активной зоны 60–83 литра при мощности до 7,5 МВт и повышение максимальной плотности потока тепловых нейтронов до $1.5 \times 10^{18} \text{ м}^{-2} \text{с}^{-1}$. Использованные конструктивные решения существенно повысят безопасность эксплуатации реактора.

I. ВВЕДЕНИЕ

Исследовательский реактор ИРТ-2000 в Софии (НРБ) был введен в эксплуатацию в 1961 г. Реактор ИРТ-2000, как известно, является серийным реактором бассейнового типа, в котором обыкновенная вода используется в качестве замедлителя, теплоносителя и защиты. Он создан на основе проекта первого реактора этого типа, разработанного и сооруженного в Институте атомной энергии имени И.В.Курчатова в 1957 г. [1]. Реакторы этого типа сооружены в ряде Центров в СССР и за рубежом.

На реакторе проводятся научные исследования в области ядерной физики, химии, биологии и др. Реактор используется также и для производства ряда радиоизотопов. Облучение мишеньей для радиоизотопов на реакторе производится во всех имеющихся вертикальных каналах: центральном канале в активной зоне и 10 каналах вне корпуса активной зоны в отража-
В 1982 году была разработана программа реконструкции реактора [3], предусматривающая следующее:
- замену активной зоны реактора;
- установку нового бака реактора из коррозионностойкой стали без демонтажа существующего;
- реконструкцию технологических систем реактора.
В данной работе описывается часть разработок по реконструкции реактора ИРТ-2000, выполненная в ИАЭ им. И.В. Курчатова.

2. КОНСТРУКТИВНЫЕ ОСОБЕННОСТИ РЕАКТОРА ИРТ-5000

Продольный и поперечный разрезы реактора ИРТ-5000 даны на рис. 1 и 2. Активная зона реактора размещается в заполненном обычной обессоленной водой бассейне, облицовкой которого является бак из коррозионностойкой стали, установленный в старый алюминиевый бак. Глубина бассейна 7,785 м, длина 4,3 м, ширина 1,8 м. Зазор между баком из коррозионностойкой стали и старым баком из алюминия заполнен бетоном. В районе бокового теплового экрана биологической защиты в этом зазоре для охлаждения бетона установлен змеевик, подключенный к автономному контуру.

Корпус реактора установлен на промежуточном дне бассейна, поднятном на 400 мм от основного dna бака и образующем задерживающую емкость объемом 2,6 м³. Корпус реактора имеет одну дистанционирующую решетку — опорную, на которую устанавливаются ТВС, блоки отражателя, блок ловушки нейтронов, вытеснители различных типов и свинцовый щит. Вверху ТВС, блоки отражателя и другие устройства дистанционируются специальными выступами на их концевых деталях. Все узлы и устройства, устанавлиываемые в корпусе реактора, являются сменными. Предусмотрена также возможность замены корпуса реактора и опорной решетки.

Как видно из рис. 2, корпус реактора имеет 71 ячейку. В 8 периферийных ячейках, смежных с тепловой колонной, установлен свинцовый щит. В периферийных ячейках обычно уста-
Рис.1. Продольный разрез реактора ИРТ-5000
I-приводы стержней СУЗ, 2-вертикальные экспериментальные каналы (ЭК),
3-каналы со стержнями СУЗ, 4-стальные экраны, 5-опорная решетка,
6-тепловыделяющая сборка (ТВС), 7-корпус реактора, 8-промежуточное
(разделительное) дно, 9-бак реактора, 10-касательный горизонтальный
экспериментальный канал, 11-вертикальная перегородка, 12-эjectор,
13-воздушник.
Рис. 2. Поперечный разрез реактора ИРТ-5000

I-канал с неподвижной ИК, 2-свинцовый щит, 3-четырехтрубная ТВС, 4-горизонтальный экспериментальный канал, 5-стальной экран, 6-вертикальный экспериментальный канал, 7-трехтрубная ТВС, 8-бериллиевый блок отражателя, 9-блок ловушки нейтронов, 10-вытеснитель для ЭК, 11-задерживающая емкость, 12-перегрузочная труба для выгрузки ТВС из бассейна, 13-ячейки для временного хранения отработанных ТВС.
новлены бериллиевые блоки. В одном из них размещен канал со стержнем автоматического регулятора. В нескольких ячейках установлены вытеснители для размещения вертикальных экспериментальных каналов.

В реакторе используются 4-трубные и 3-трубные ТВС типа ИРТ-2М (рис. 3). В ТВС используются трубчатые твэлы квадратного сечения с толщиной стенки 2 мм. Толщина оболочек твэлов из алюминиевого сплава - 0,5 мм. Толщина сердечника твэла из диоксида урана в алюминиевой матрице - 1,0 мм. Ширина зазоров между твэлами - 4,5 мм. Основные характеристики этих ТВС приведены в таблице I. В 3-трубной ТВС устанавливается канал со стержнем системы управления и защиты (СУЗ) [4]. В 3-трубной ТВС может также устанавливаться экспериментальный канал для облучений наружным диаметром 28 мм.

Верхние концы каналов со стержнями СУЗ закреплены на специальном кронштейне. Нижние центрируются в решетке нижнего экрана, находящегося на дне бака под опорной решеткой. Каналы всех стержней СУЗ "мокрые". Стержни с помощью тросов связаны с приводами, которые размещены на специальной стойке.

В реакторе имеется 11 стержней СУЗ. Из них два используются для аварийной защиты (A3), 8 - для компенсации реактивности (КС) и один - для автоматического регулирования (АР). В качестве поглотителя в стержнях СУЗ используются таблетки карбида бора, заключенные в стальную оболочку наружным диаметром 23 мм. Ниже поглотителя каждый стержень, кроме АР, имеет вытеснитель из алюминия. В отличие от реактора ИРТ-2000 на ИРТ-5000 приводы стержней КС, так же как и стержней А3, имеют электромагнитные муфты, обеспечивающие быстрое погружение стержней КС в активную зону в случае появления аварийных сигналов.

Каналы с подвижными ионизационными камерами (ИК) СУЗ реактора размещены так же, как и на реакторе ИРТ-2000, т.е. в тепловой колонне. Неподвижные ИК размещены в "мокрых"
Рис. 3. Поперечные сечения ТВС типа ИРТ-2М
а) - четырехтрубная ТВС,
б) - трехтрубная ТВС со стержнем СУЗ.
1-твэлы, 2-канал, 3-стержень СУЗ, 4-вытеснитель.
каналах, устанавливаемых над бериллиевыми блоками, окружающими горизонтальный касательный экспериментальный канал. Каждая неподвижная ИК заключена в алюминиевый герметичный чехол, соединенный с резиновой трубкой, используемой для вывода кабелей от ИК к клеммнику.

Таблица I
Основные характеристики ТВС типа ИРТ-2М
с топливом 36%-ного обогащения

<table>
<thead>
<tr>
<th>Характеристика</th>
<th>4-трубные ТВС</th>
<th>3-трубные ТВС с каналом и стержнем СУЗ</th>
</tr>
</thead>
<tbody>
<tr>
<td>Доля объема активной зоны, занятая водой</td>
<td>0,725</td>
<td>0,649</td>
</tr>
<tr>
<td>Поверхность теплообмена в единице объема активной зоны, м²/л</td>
<td>0,265</td>
<td>0,228</td>
</tr>
<tr>
<td>Длина сердечника, м</td>
<td>0,58</td>
<td>0,58</td>
</tr>
<tr>
<td>Обогащение урана, %</td>
<td>36</td>
<td>36</td>
</tr>
<tr>
<td>Содержание урана=235 в ТВС, кг</td>
<td>0,230</td>
<td>0,198</td>
</tr>
<tr>
<td>Поверхность теплообмена в ТВС, м²</td>
<td>0,79</td>
<td>0,68</td>
</tr>
<tr>
<td>Концентрация урана=235 в активной зоне, кг/м³</td>
<td>77,6</td>
<td>66,8</td>
</tr>
<tr>
<td>Количество урана=235, приходящегося на 1 м² поверхности теплообмена, кг/м²</td>
<td>0,29</td>
<td>0,29</td>
</tr>
</tbody>
</table>

В связи с тем, что корпус тепловой колонны, как и вся облицовка бассейна, выполнен из коррозионностойкой стали, конструкция корпуса изменена для уменьшения возмущения потока нейтронов в касательном канале. Верхняя половина корпуса тепловой колонны сделана короче, чем нижняя. Пространство между корпусом реактора и торцевой стенкой верхней половины корпуса заполнено бериллием. Таким образом, касательный канал не проходит сквозь тепловую колонну, как в ИРТ-2000, а окружен бериллиевыми блоками.
В отличие от реактора ИРТ–2000 с традиционной конст рукцией горизонтальных экспериментальных каналов в пределах бассейна, при которой разрушение канала приводит к утечке воды из бассейна и оголению верхней части активной зоны с последующим возможным перегревом верхних концов твэлов до их разгерметизации или расплавления, для ИРТ–5000 разработана конструкция канала, позволяющая ис ключить такую аварийную ситуацию. Канал представляет собой полый вытеснитель из алюминия, устанавливаемый между стенкой корпуса реактора и стенкой бака. Для уменьшения потери плотности потока тепловых нейтронов в пучке, которая при стальной стенке бака реактора может достигать ~ 40% (при толщине – 8 мм), в стенке бака напротив вытеснителей могут быть установлены вставки из циркония.

Охлаждение реактора осуществляется водой бассейна, прокачиваемой через ТВС и блоки отражателя центробежными насосами и эжектором, работающими параллельно. Охлаждающая вода в активной зоне и отражателе движется сверху вниз и выводится в нижнюю полость бассейна, отделенную от его основного пространства промежуточным дном (рис. I).

Часть воды, прокачиваемая через активную зону и отражатель насосами, из нижней полости попадает в установленную в бассейне дополнительную задерживающую емкость. Из задерживающей емкости по трубопроводу вода поступает в насосы, затем проходит теплообменники и возвращается в бассейн через сопло эжектора.

Другая, большая часть прокачиваемой через активную зону воды, забирается из нижней полости эжектором, установленным, в отличие от использовавшихся ранее решений [I], вертикально внутри специального распределительного короба, расположенного в бассейне со стороны, противоположной активной зоне (рис. I). Эта часть воды поступает вместе с водой, выходящей из сопла эжектора, в бассейн через вертикальную перфорированную перегородку короба, которая обеспечивает направление основного потока воды в сторону активной
зоны. Другими преимуществами такого расположения эжектора в бассейне реактора являются:

- улучшение характеристик эжектора за счет увеличения его длины до оптимальной;
- уменьшение вибрации активной зоны, поскольку эжектор не соединен непосредственно с корпусом реактора;
- возможность демонтажа эжектора при необходимости его замены или последующей модернизации реактора без демонтажа активной зоны и горизонтальной перегородки.

Система охлаждения реактора с использованием эжектора обладает также следующим достоинством: не требуется системы аварийного охлаждения с активными элементами в случае полной лотерии электроснабжения, что существенно повышает безопасность реактора.

С точки зрения безопасности эксплуатации реактора важным является вывод всасывающего и напорного трубопроводов из бассейна в его верхней части, что исключает опорожнение бассейна более чем на один метр при любых разрывах трубопроводов и повреждениях оборудования первого контура. Для автоматического срыва сифона на всасывающем и напорном трубопроводах перед выходом их из бассейна предусмотрены отверстия (воздушники) с постоянным небольшим протоком воды.

Для слива воды из бассейна имеется специальная труба диаметром 60 мм, входящая в бассейн в его верхней части на том же уровне, где проходят трубопроводы первого контура, и опускающаяся до дна бассейна.

Для снижения нагрева боковой биологической защиты за счет поглощения гамма-излучения между новым баком и отражателем предусмотрен экран из стали общей толщиной 80 мм, охлаждаемый водой бассейна. Для снижения нагрева бетона под днищем бака между новым баком и активной зоной установлен экран из стали общей толщиной 125 мм, охлаждаемый водой, проходящей через активную зону.
3. ЭКСПЕРИМЕНТАЛЬНЫЕ ВОЗМОЖНОСТИ РЕАКТОРА ИРТ-5000

Конструкция реактора позволяет реализовывать различные варианты рабочих загрузок активной зоны и отражателя в зависимости от характера проводимых на реакторе экспериментов. Две из возможных рабочих загрузок реактора представлены на рис. 4: загрузка из 28 ТВС с центральной ловушкой нейтронов и компактная загрузка из 20 ТВС. Основные параметры этих загрузок даны в таблице П.

В корпусе реактора вместо любого бериллиевого блока могут быть установлены ТВС, вытеснитель, полностью заполненный водой, или вытеснитель с воздушной полостью (высотой 100 мм), расположенной на уровне горизонтальных экспериментальных каналов. Это обеспечивает в случае необходимости возможность изменения спектра нейтронов в том или ином экспериментальном канале. Возможна также установка бериллиевых блоков или вытеснителей вместо ТВС.

Запас реактивности представленных на рис. 4 загрузок активной зоны из свежих ТВС с бериллиевыми блоками в периферийных ячейках не может быть скомпенсирован 8 стержнями КС СУЗ. Однако восьми стержней КС достаточно для компенсации запаса реактивности, если при эксплуатации реактора производится не полная замена всех выгоревших ТВС в активной зоне на все свежие, а применяется режим частичных замен, который обеспечивает возможность достижения более глубокого выгорания топлива в выгружаемых ТВС. Поэтому нецелесообразно увеличивать количество компенсирующих стержней СУЗ из-за необходимости компенсации запаса реактивности реактора на первом этапе его эксплуатации. Можно начинать эксплуатацию реактора с меньшим, чем в рабочей загрузке, количеством ТВС в активной зоне или временно использовать дополнительные поглотители, устанавливаемые в ТВС так же, как и штатные стержни СУЗ, но без приводов. Переход от начальной к основной рабочей загрузке активной зоны должен проводиться по мере уменьшения запаса реактивности с выгоранием топлива.
Рис. 4. Возможные варианты рабочих загрузок реактора ИРТ-5000:
а) – загрузка из 28 ТВС с центральной ловушкой нейтронов,
б) – компактная загрузка из 20 ТВС,
1 – четырехтрубная ТВС, 2 – трехтрубная ТВС со стержнем КС, 3 – трехтрубная ТВС со стержнем АЗ, 4 – бериллиевый блок со стержнем АР,
5 – сменный бериллиевый блок отражателя (сплошной), 6 – сменный бериллиевый блок с пробкой, 7 – вытеснитель для экспериментального канала диаметром 60 мм, 8 – блок ловушки нейтронов, 9 – свинцовый щит.
Таблица II
Основные параметры загрузок ИРТ-5000 (рис. 4)

<table>
<thead>
<tr>
<th>Параметр</th>
<th>Рабочая</th>
<th>Компактная</th>
</tr>
</thead>
<tbody>
<tr>
<td></td>
<td>загрузка</td>
<td>с центральной загрузкой</td>
</tr>
<tr>
<td></td>
<td>рабочая</td>
<td>ловушкой нейтронов</td>
</tr>
<tr>
<td>Мощность, МВт</td>
<td>5</td>
<td>5</td>
</tr>
<tr>
<td>Количество ТВС в загрузке,</td>
<td>28</td>
<td>20</td>
</tr>
<tr>
<td>из них: 4-трубных</td>
<td>18</td>
<td>10</td>
</tr>
<tr>
<td>3-трубных со стержнями СУЗ</td>
<td>10</td>
<td>10</td>
</tr>
<tr>
<td>Объем активной зоны, л</td>
<td>83</td>
<td>59</td>
</tr>
<tr>
<td>Масса ураниума-235 в загрузке со &quot;свежими&quot; ТВС, кг</td>
<td>6,12</td>
<td>4,28</td>
</tr>
<tr>
<td>Запас реактивности загрузки со &quot;свежими&quot; ТВС, % ΔK/K</td>
<td>17,3</td>
<td>18,4</td>
</tr>
<tr>
<td>Максимальная объемная мощность, МВт/л</td>
<td>0,14</td>
<td>0,15</td>
</tr>
<tr>
<td>Максимальная плотность теплового потока, МВт/м²</td>
<td>0,56</td>
<td>0,60</td>
</tr>
<tr>
<td>Неравномерность энерговыделения в активной зоне</td>
<td>2,5</td>
<td>1,86</td>
</tr>
<tr>
<td>Расход воды первого контура через теплообменники, кг/с</td>
<td>66,7</td>
<td>66,7</td>
</tr>
<tr>
<td>Потери напора на активной зоне, кПа</td>
<td>15,4</td>
<td>16,8</td>
</tr>
<tr>
<td>Температура воды на входе в ТВС, °С</td>
<td>45</td>
<td>45</td>
</tr>
<tr>
<td>Максимальная расчетная температура стенки твэла, °С</td>
<td>85</td>
<td>86</td>
</tr>
<tr>
<td>Температура начала кипения воды в наиболее нагруженной точке, °С</td>
<td>130</td>
<td>130</td>
</tr>
<tr>
<td>Максимальная плотность потоков нейтронов $\times 10^{-18} \text{м}^{-2} \cdot \text{с}^{-1}$: тепловых:</td>
<td></td>
<td></td>
</tr>
<tr>
<td>в ловушке нейтронов</td>
<td>1,1</td>
<td>-</td>
</tr>
<tr>
<td>в каналах диаметром 60 мм в активной зоне</td>
<td>0,96</td>
<td>-</td>
</tr>
<tr>
<td>в сменных бериллиевых блоках</td>
<td>0,63</td>
<td>0,9</td>
</tr>
<tr>
<td>в каналах вне корпуса реактора</td>
<td>0,34</td>
<td>0,34</td>
</tr>
<tr>
<td>быстрых (с $E &gt; 0.821$ МэВ) в активной зоне</td>
<td>0,41</td>
<td>0,65</td>
</tr>
</tbody>
</table>
Как видно из таблицы П, при мощности реактора 5 МВт максимальная расчетная температура стенки твэла в рабочих загрузках равна 85–86°С.

Опыт эксплуатации реакторов ИРТ и ВВР-СМ, работающих в СССР с ТВС типа ИРТ-2М и ИРТ-3М, показал, что тепловой режим активной зоны безопасен, если максимальная расчетная температура стенки твэла не превышает ~105°С. Очевидно, что при эксплуатации реактора ИРТ-5000 ИЯИЯЭ его мощность может достигать 7,5 МВт.

Реактор ИРТ-5000, как и ИРТ-2000, имеет II горизонтальных каналов для вывода пучков нейтронов (рис. 2):
- 9 радиальных диаметром 100 мм;
- касательный диаметром 150 мм;
- радиальный, проходящий через тепловую (графитовую) колонну, диаметром 150 мм.

Расчетные значения плотностей потоков тепловых нейтронов на торцах горизонтальных каналов в рабочей загрузке из 28 ТВС при мощности реактора 5 МВт составляют
\[(1,3 \pm 3,0) \cdot 10^{-17} \text{ м}^{-2} \text{.с}^{-1} \].

Для облучения в пределах корпуса реактора в активной зоне и бериллиевом отражателе могут использоваться:
- каналы внутренним диаметром 26 мм в 3-трубных ТВС;
- каналы наружным диаметром 60 мм в специальных вытеснителях, размещаемых вместо ТВС (4 канала с плотностью потока тепловых нейтронов \((0,9 \div 1,0) \cdot 10^{-18} \text{ м}^{-2} \text{.с}^{-1}\) или за бериллиевыми блоками (6 каналов с плотностью потока тепловых нейтронов до \(0,3 \cdot 10^{18} \text{ м}^{-2} \text{.с}^{-1}\));
- канал наружным диаметром 45 мм в блоке ловушки нейтронов (плотность потока тепловых нейтронов \(1,1 \cdot 10^{18} \text{ м}^{-2} \text{.с}^{-1}\));
- канал наружным диаметром до 96 мм в блоке ловушки нейтронов (плотность потока тепловых нейтронов \(1,3 \cdot 10^{18} \text{ м}^{-2} \text{.с}^{-1}\));
- каналы или ампулы наружным диаметром 45 мм в бериллиевых блоках отражателя (6 каналов с плотностью потока...
тепловых нейтронов до 0,6.10^{18} \text{м}^{-2}\text{с}^{-1}).

Для облучений за пределами корпуса реактора в водяном отражателе могут использоваться:
- 7 вертикальных каналов наружным диаметром 54 мм между горизонтальными каналами с плотностью потока тепловых нейтронов \((I = 3).10^{17} \text{м}^{-2}\text{с}^{-1};
- 7 вертикальных каналов наружным диаметром 54 мм над горизонтальными каналами с плотностью потока тепловых нейтронов \((I = 3,4).10^{17} \text{м}^{-2}\text{с}^{-1};
- 2 вертикальных канала диаметром 180 мм;
- вертикальный канал диаметром 150 мм в тепловой колонне.

Таким образом, в результате реконструкции количество вертикальных каналов для облучения в пределах корпуса реактора увеличится с 2 до 17 и вне корпуса реактора - с 10 до 16.

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A Recommendation of the National Board for Atomic Safety and Radiation Protection for the Appointment of Nuclear Safety Control Officers for Research Reactors
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A Recommendation of the National Board for Atomic Safety and Radiation Protection for the Appointment of Nuclear Safety Control Officers for Research Reactors

Abstract

The Ordinance on the Implementation of Atomic Safety and Radiation Protection of the GDR requires that the managers of plants where nuclear facilities are operated appoint Control Officers for the fields of radiation protection, nuclear safety, physical protection, and accounting for and control of nuclear material. The Control Officers are staff members of the operating organization but their appointment is subject to approval by the National Board and requires adequate qualification. The main task of the Control Officers as specialists is to give advice to the plant manager who retains responsibility for the safety of nuclear facilities, and to verify on his behalf that all requirements within their competence are met by the operating group. For this reason the Control Officer has to be absolutely independent of the head of the operating group. To enable the Control Officers to accomplish all necessary control activities and to guarantee independence from the head of the operating group, the plant manager has to establish adequate regulations of operation. As a pattern for such regulations the National Board has issued a Recommendation for the Appointment of Nuclear Safety Control Officers for Research Reactors, which provides a comprehensive survey of the requisite qualification features as well as the duties and rights of these Control Officers. This recommendation will be dealt with in the presentation.

1. INTRODUCTION

In the German Democratic Republic (GDR) control officers for radiation protection have been employed for research reactors for more than two decades /1/. The new Ordinance on the Implementation of Atomic Safety and Radiation Protection of the GDR /2/ requires that from now on the managers of plants where research reactors are operated also appoint control officers for all subfields of atomic safety (see Fig. 1). This allows for a development that has led to an increasing specialization in the various fields of atomic safety and radiation protection.

To ensure an efficient activity of control officers, the plant managers are to elaborate instructions providing both
the qualification requirements and the rights and duties of control officers. As a pattern for these instructions, the National Board for Atomic Safety and Radiation Protection (the National Board) has issued a Recommendation for the Appointment of Nuclear Safety Control Officers for Research Reactors /3/. This recommendation will be presented in more details below.

2. PRINCIPLES FOR THE WORK OF CONTROL OFFICERS AND QUALIFICATION REQUIREMENTS ON CONTROL OFFICERS FOR NUCLEAR SAFETY OF RESEARCH REACTORS

The control officers are appointed by the plant manager and should be directly subordinated to him. They must never be disciplinarily subordinated to the head of the research reactor operating group. They perform their control activities on behalf of the plant manager and are both responsible and accountable to him. The appointment of control officers restricts neither the plant manager's responsibility nor that of the heads of operating groups.

For each control officer, the spatial and material spheres of responsibility have to be clearly established and, along with his main tasks and rights, laid down in his employment contract. If a control officer performs his control activities as a part-time job, this will take precedence over all his other tasks in the plant.

The appointment of a control officer requires approval by the National Board. This approval will be granted if the mentioned principles are met and the required qualifications are acquired. In accordance with a guideline of the National Board /4/ the appointment of control officers for nuclear safety of research reactors depends on the following qualification requirements:

- a university degree in relevant discipline;
- not less than 3 years' work with a research reactor;
- specific knowledge of the design, mode of operation and utilization of research reactors in their sphere of responsibility;
- knowledge of the legal and operational regulations for the implementation of atomic safety and radiation protection and of the provisions made in the license of the National Board, and
- a state certificate of qualifications.

Prerequisite for acquiring the state qualification certificate is the participation in special training courses conducted by the National Board as the national centre of further education in atomic safety and radiation protection. Major items of such training courses are:

- knowledge of the legal regulations and guidelines in the fields of atomic safety and radiation protection,
- fundamental knowledge of nuclear engineering, and
- special knowledge in the field of nuclear safety.

The state qualification certificate is granted by the National Board after successfully passing the examination. The control officers are bound to attend the retraining
courses of the National Board which run nearly every year.

3. RIGHTS AND DUTIES OF CONTROL OFFICERS FOR NUCLEAR SAFETY OF RESEARCH REACTORS

The main tasks of control officers consist in giving advice to the plant manager on all issues concerning the nuclear safety of research reactors and in supervising the heads of research reactor operating groups in the observance of their duties to ensure nuclear safety.

The controls have to be made regularly by the control officers in their respective spheres of responsibility. Type and extent of controls are to be laid down in a control plan.

To ensure effective control, each control officer has the right to
- enter all facilities and rooms of his sphere of responsibility at any time;
- demand information, assessments and reports associated with the nuclear safety of the research reactor from the head of the operating group or from members of operating personnel, and
- inspect all plant documents and records important to the research reactor's nuclear safety.

Each control officer has to control that the demands for ensuring nuclear safety provided in GDR legal regulations, state licences, guidelines of the National Board and plant instructions are observed. He has also to control that the measures provided for the occurrence of unusual events are taken.

Here it should be noted that, in accordance with the term usual in the GDR, an unusual event is understood to be every safety-relevant deviation from the intended operation or operational state. Unusual events are to be reported to the National Board. Depending on the urgency of notification, there are three stages of notification. All details of procedure in the case of unusual events have been regulated in a guideline of the National Board /5/. This guideline also contains the demand to consult, in cases of doubt, the control officers in establishing the appropriate notification stage.

Each control officer has to take part in operations relevant to the nuclear safety of research reactors in his sphere of responsibility and to evaluate the results. In particular, he has to be involved in
- checks of the limits and conditions of safe operation;
- critical experiments;
- training in nuclear accident prevention, and
- examinations of reactor operators.

Each control officer has to keep a control ledger where to enter all controls performed and their results, particularly detected deficiencies in ensuring nuclear safety and measures and deadlines set to correct these deficiencies.
Each control officer is bound to demand, from the head of the research reactor operating group, the correction of such deficiencies as, e.g., offences against safety-relevant provisions in legal regulations, licences, plant instructions or special requirements of the National Board. In every case it has to be checked whether the occurrence has to be considered as an unusual event in the sense of the guideline. Irrespective of the result of this check, the control officer has to lay down deadlines for correcting deficiencies and to communicate them to the head of the operating group. If these deadlines are not observed, the control officer has to inform the plant manager and the National Board.

In the case of imminent danger to persons, facilities or equipment the control officer has to bar rooms, facilities, equipment or experiments from further use unless this has already been done by the head of the operating group. If barring is not the most suitable measure, he has to demand immediate measures to restore nuclear safety to eliminate acute danger or to limit damage. In any case the plant manager and the National Board have to be informed without delay.

The control officers have to be involved in preparing changes of research reactors or in planning new nuclear facilities in their sphere of responsibility. Changes may concern the design or utilization of the research reactor as well as safety-relevant operation instructions.

In the GDR, it is a duty to classify changes of research reactors in two categories according to their relevance to nuclear safety. It generally holds that all facility changes requiring a change of limits and conditions of safe operation have to be submitted to the National Board for approval. All other changes are not subject to approval but have to be communicated to the National Board. The head of an operating group has to decide on whether a change is subject to approval or notification. Such a decision has to be reviewed by the control officer. In the case of changes subject to approval an application for change has to be submitted to the National Board together with a proposal for the required change of limits and conditions of safe operation. The proposal and the statement of reasons have also to be reviewed by the control officer. The results of review are summarized in a statement which is attached to the application for approval.

In periodic operation reports every head of an operating group has to account to the National Board for the safe operation of the reactor. In particular observance of established limits and conditions has to be proved. The deadlines for these reports and the exact specification of their content are laid down in the operation licence of every research reactor.

In addition to routine reports special reports to the National Board are necessary, particularly in the case of unusual events and fulfilment of conditions improved by the National Board. Any reporting to the National Board
requires review and assessment by the control officer.

In certain situations, say after major revisions, facility changes or unusual events, it can be necessary that, before start-up of the research reactor, the entire reactor or individual parts or systems are thoroughly checked. It depends on the result of this check, which is to be recorded in an acceptance protocol, whether the research reactor may be started up again. The recommendation of the National Board /3/ provides that control officers have to carry out acceptance tests after major revisions and unusual events of the lowest stage. Acceptance tests after safety-relevant facility changes and unusual events of the second stage can be made by control officers only if they have been entrusted with the job by the National Board. This decision will be made on a case-by-case basis. Acceptance tests after an unusual event of the highest notification stage are in any case performed by the National Board.

Each control officer is bound to regularly inform the plant manager of his control results and of problems related to nuclear safety in his sphere of responsibility. To this effect, the control officer has to prepare control reports that are also submitted to the National Board for information. Furthermore, each control officer is bound to submit, at the request of the National Board, assessments, expertises or comments on special safety problems in his area of control.

4. FIRST EXPERIENCE

The Ordinance on the Implementation of Atomic Safety and Radiation Protection of 11 October 1984 /2/ permits different forms of organization for appointing control officers for nuclear safety. The organizational structure can be adapted to individual plant features so that the necessary expenditure is kept low.

For small research reactors it is sufficient if the control activity is performed as a part-time job or if an officer carries out controls in several subsections of atomic safety and radiation protection simultaneously. These variants have been chosen for the small training reactors of the Dresden Technical University and the Zittau Engineering College.

For plants where several research reactors are operated, it is an advantage to additionally appoint a senior control officer for nuclear safety. He should supervise all measures that equally apply to the operation of several research reactors. A typical example are the measures for nuclear accident prevention. The appointment of a senior control officer enables the control officers for research reactors to participate themselves in the preparation of safety accounts without violating the prohibition of individual control. In such cases the demanded independent control is to be performed by the senior control officer. In this way the special knowledge of control officers can
In this way the special knowledge of control officers can be better utilized. This variant is used by the Central Institute of Nuclear Research of the Academy of Sciences of the GDR where three research reactors and a number of other nuclear facilities are operated.

On the whole, the first practical experience gained through the activity of control officers for nuclear safety at the five research reactors of the GDR allows to assess that an increased safety level has been attained. Of course, spectacular improvements could not be expected since, as early as before appointing control officers, the safety level of research reactors was high. The activity of control officers has improved the plant managers' insight into the issues of nuclear safety, so that these aspects are more allowed for in long-term planning. Furthermore it has led to a higher level in reporting to the National Board and in issuing applications for approval of facility changes. Thus the National Board was enabled to concentrate more than before on priorities of safety in connection with the national surveillance of research reactors.

5. REFERENCES


FIG. 1 CONTROL OFFICERS (C.O.) TO BE APPOINTED ACCORDING TO GDR REGULATIONS
LICENCIAMIENTO DEL REACTOR TRIGA MARK III
DEL CENTRO NUCLEAR DE MEXICO

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The Triga Mark III reactor at the Mexican Nuclear Centre went critical in 1968 and remained so until 1979 when the National Commission for Nuclear Safety and Safeguards (CNSNS), the Mexican regulatory authority, was set up. The reactor was therefore operating without a formal operating licence, and the CNSNS accordingly requested the ININ to license the reactor under the existing conditions and to ensure that any modification of the original design complied with Standards ANSI/ANS-15 and with the code of practice set out in IAEA Safety Series No. 35. The most relevant points in granting the operating licence were: (a) the preparation of the Safety Report; (b) the formulation and application of the Quality Assurance Programme; (c) the reconditioning of the following reactor systems: the cooling system; the ventilation and exhaust system; the monitoring system and control panel; (d) the training of the reactor operating staff at junior and senior levels; and (e) the formulation of procedures and instructions.

Once the provisional operating licence was obtained for the reactor it was considered necessary to modify the reactor core, which had been composed of 20% enriched standard fuel, to a mixed core based on a mixture of standard fuel and FLIP-type fuel with 70% $^{235}\text{U}$ enrichment. The CNSNS therefore requested that the mixed core be licensed and a technical report was accordingly annexed to the Safety Report, its contents including the following subjects: (a) neutron analysis of the proposed configuration; (b) reactor shutdown margins; (c) accident analysis; and (d) technical specifications. The licensing process was completed this year and we are now hoping to obtain the final operating licence.
RESUMEN.

LICENCIAMIENTO DEL REACTOR TRIGA MARK III DEL CENTRO NUCLEAR DE MEXICO.

El reactor Triga Mark III del Centro Nuclear de México alcanzó criticidad en 1968 y fue hasta 1979 cuando el organismo regulador, Comisión Nacional de Seguridad Nuclear y Salvaguardias (CNSNS) fue creado. Por tal motivo el reactor operaba sin licencia de operación formal por lo que la CNSNS pidió al ININ que el reactor fuera licenciado en las condiciones que se encontraba y que cualquier cambio al diseño original debía cumplir con las normas ANSI-ANS-15 y con el código de prácticas del OIEA Safety Series No. 35. Los tópicos más relevantes para el otorgamiento de la licencia de operación fueron: a) La elaboración del Informe de Seguridad, b) Elaboración y aplicación del Programa de Garantía de Calidad, c) Reacondicionamiento de los sistemas del reactor: Sistema de enfriamiento, Sistema de Ventilación y Extracción, Sistema de monitoreo, Consola de Control, d) Licenciamiento del personal de operación del reactor: operadores y operadores senior y e) Elaboración de procedimientos e instrucciones.

Una vez obtenido el permiso provisional de operación del reactor se vio la necesidad de modificar el núcleo del reactor que estaba compuesto de combustible estándar con 20% de enriquecimiento a núcleo mixto con una mezcla de combustible estándar y combustible tipo FLIP con 70% de enriquecimiento en U-235, esto ocasionó que la CNSNS solicitará que el núcleo mixto fuera licenciado por lo que se elaboró un reporte técnico como anexo al Informe de Seguridad que debería incluir entre otros temas: a) El análisis neutrónico de la configuración propuesta, b) Márgenes de parada del reactor, c) Análisis de Accidentes y e) Especificaciones Técnicas. El proceso de licenciamiento finalizó este año y nos encontramos en espera de la licencia de operación definitiva.

1. ANTECEDENTES.

El reactor TRIGA Mark III del Centro Nuclear de México, es un reactor crítico tipo alberca, enfriado y moderado con agua ligera que utiliza hiduro de zirconio mezclado con uranio enriquecido originalmente al 20% en U-235 como combustible moderador. Actualmente opera con núcleo mixto que utiliza elementos de combustible moderador de U-ZrH$_{1.7}$ con enriquecimientos de 20% y 70% en U-235.
Está diseñado para operar en régimen permanente con potencia térmica de 1MW y en régimen transitorio o pulsado a una potencia de 2,000 MW por 10 milésimas de segundo.

La construcción del reactor se inició en abril de 1964 y alcanzó criticidad por vez primera el 8 de noviembre de 1968. La energía total generada por el reactor al mes de mayo de 1989 es de 7,800,540.9 KW·H. A la fecha se han utilizado 120 elementos combustibles estándar, 26 elementos tipo FLIP, 3 barras de control con seguidor estándar, 3 barras de control con seguidor FLIP y una barra de control sin seguidor.

En 1979, por disposiciones del gobierno de México, el Instituto Nacional de Energía Nuclear (INEN) fue dividido en 3 organismos separados:

ININ : Instituto Nacional de Investigaciones Nucleares. Encargado de la investigación, promoción y desarrollo de la energía nuclear.

URAMEX : Uranio Mexicano. Encargado de la prospección, exploración y explotación de los yacimientos de uranio y la fabricación del combustible.

CNSNS : Comisión Nacional de Seguridad Nuclear y Salvaguardias. Encargado de la evaluación de la seguridad y salvaguardias de las instalaciones nucleares.

Previo a la división, la función reguladora estaba a cargo de un departamento dentro del INEN y sus funciones consistían principalmente en la revisión del Informe Preliminar de la planta nuclear de potencia de Laguna Verde. Una vez constituida la CNSNS y definidas las funciones que le otorgaba la Ley Nuclear, pidió al ININ que el reactor TRIGA obtuviera una licencia formal de operación quedando establecido que, antes de realizar cualquier modificación al reactor, éste debía ser licenciado como había sido construido esto debido al hecho de que el reactor presentaba problemas durante su operación en los últimos meses. Estos problemas consistían en: falta de exceso de reactividad, mal funcionamiento de la consola de control, operación deficiente del sistema de monitoreo de la sala del reactor, problemas con el sistema de refrigeración y con el sistema de ventilación y extracción de la sala del reactor.

La idea inicial fue entonces licenciar el reactor como estaba construido, pero dado que presentaba los problemas arriba mencionados la idea varió un poco. Se debía obtener la licencia de operación del reactor y las modificaciones que tuvieran lugar deberían ser licenciadas usando las normas ANSI-ANS-15 (1) y el código de prácticas Safety Series No. 35 (2) del Organismo Internacional de Energía Atómica.
Estos trabajos empezaron a realizarse en 1983 cuando se comenzó a elaborar el Informe de Seguridad del Reactor Triga Mark III del Centro Nuclear de México y a reacondicionar los sistemas del mismo. Mientras se licenciaba el reactor y se reacondicionaban los sistemas se podía seguir operando ya que desde la criticidad inicial lo había hecho sin contratiempo alguno.

Durante el proceso de obtención de la licencia de operación del reactor ante la CNSNS, se presentó un problema de corrosión en el recubrimiento de aluminio de la piscina del reactor, el cual fue descubierto en marzo de 1985 cuando se abrió el cuarto de exposición para ser inspeccionado. La fuga nunca representó un problema para el reactor ya que ésta alcanzó un máximo de 5 l/h disminuyendo hasta desaparecer completamente 2.5 meses después. Además el reactor cuenta con un sistema de reposición de agua pretratada de 50 l/h (3). A raíz de este incidente la CNSNS ordenó que las obras de reparación del recubrimiento de aluminio de la piscina del reactor así como los trabajos de reacondicionamiento de sus sistemas fueran realizados con estricto apego a las normas ANSI-ANS 15 y que el reactor volviera a operar hasta que obtuviera la licencia de operación cumpliendo con los requisitos antes impuestos.

Este proceso nos mantuvo fuera de operación aproximadamente 18 meses.

2.- LICÉNCIAMIENTO DEL REACTOR.

Los puntos más relevantes a ser cumplidos para obtener la licencia de operación fueron los siguientes:

a) Actualización del Informe de Seguridad.(4)
b) Instalación del Programa de Garantía de Calidad. (5)
c) Reacondicionamiento de los sistemas del reactor.(6)
d) Licenciamento de los operadores del reactor.
e) Elaboración de procesos administrativos, procedimientos de protección radiológica, operación y emergencias.

2.1 Actualización del Informe de Seguridad.

El primer Informe de Seguridad consistía en un documento basado en el manual de operación del Reactor Triga Torrey Pines de la Gulf General Atomics; éste se tuvo que actualizar para cumplir con los requisitos de la nueva reglamentación. El resultado es un Informe de Seguridad que actualmente cuenta con diez capítulos:

I.- Introducción.
II.- Sitio e Instalación.
III.- Descripción del Reactor.
IV.- Instrumentación y Control.
V.- Sistemas Auxiliares.
VI.- Especificaciones Técnicas.
VII.- Análisis de Accidentes.
VIII.- Plan de Seguridad.
IX.- Plan de Garantía de Calidad.
X.- Conducción de operaciones.

La versión actual incluye los cambios que se realizaron en el núcleo que utiliza dos tipos diferentes de enriquecimiento.

2.2 Programa de Garantía de Calidad.

Se elaboró de acuerdo a las normas ANSI-ANS 15-8 y al Código de Prácticas del OIEA Safety Series No. 50-C-QA. Consiste en 18 criterios de calidad y establece las políticas a seguir para cumplir con los requisitos que en cuanto a calidad existen para este tipo de reactor.

El Programa de Garantía de Calidad fue elaborado por el Grupo de Garantía de Calidad del ININ y aprobado por la CNSNS y ha estado en práctica durante la reparación del recubrimiento de la piscina y el reacondicionamiento de los sistemas del reactor. Actualmente se encuentra en evaluación la revisión tres.

2.3 Reacondicionamiento de los sistemas del reactor.

El reacondicionamiento de los sistemas del reactor estuvo sujeto al Programa de Garantía de Calidad por lo que fue necesario elaborar y autorizar procedimientos de reparación para cada una de las actividades relevantes para la seguridad.

2.3.1 Sistema de enfriamiento.

Se desincrustó el Intercambiador de calor del circuito primario de refrigeración, se detectaron fugas en dos tubos y se repararon.

2.3.2 Sistema de Ventilación y Extracción.

Se le dió mantenimiento mayor al sistema para poder cumplir con los requerimientos de recambio de aire en la sala del reactor estipulados en el diseño y mantener la presión diferencial entre la sala del reactor y el exterior en el intervalo de 15 pulgadas de agua.

2.3.3 Reacondicionamiento de los Sistemas de Detección y Monitoreo.

Se reacondicionaron los sistemas y se sustituyó el equipo que lo ameritaba.
2.3.4 Consola de Control.

Se llevó a cabo el mantenimiento de la consola de control del reactor el cual incluyó, entre otras actividades, el recableado completo, y la calibración y ajuste de los canales e indicadores. Actualmente se construye en el ININ una consola de control que sustituirá a la original.

2.4 Licenciamiento del personal de operación del reactor.

Para obtener la licencia de operadores senior del reactor se realizó un programa de entrenamiento basado en la norma ANSI-ANS-15.4. El programa de entrenamiento tuvo una duración de aproximadamente 250 horas y concluyó con el licenciamiento de los operadores, 3 de los cuales recibieron licencia de operador senior.

Recientemente, dado que las licencias otorgadas por la CNSNS son por dos años, se completó un programa de reentrenamiento: los operadores presentaron los exámenes correspondientes y obtuvieron la extensión de la licencia por dos años más.

2.5 Elaboración de procedimientos e instrucciones.

Como resultado del proceso de evaluación y de la aplicación del Programa de Garantía de Calidad, se elaboraron 76 procedimientos e instrucciones de operación, 8 procedimientos de protección radiológica, 5 procedimientos de emergencia y 15 procedimientos de control administrativo que se revisan periódicamente y se mantienen bajo control para asegurar que sea utilizada siempre la última versión.

Una vez realizadas estas actividades y concluida la reparación del recubrimiento de aluminio de la piscina del reactor, la CNSNS autorizó la recarga del combustible en el núcleo del reactor ya que este había sido descargado para facilitar la inspección y reparación del recubrimiento. Se otorgó entonces un permiso provisional de operación que nos permitió operar algunos meses.

3.- LICENCIAMIENTO DEL NÚCLEO MIXTO.

El reactor reanudó sus servicios de irradiación pero con un núcleo con poco exceso de reactividad por lo que se vió la necesidad de remplazar el núcleo a base de combustible tipo estándar (20% de enriquecimiento en U-235) por un núcleo formado por una mezcla de combustible estándar irradiado y combustible fresco tipo FLIP (Fuel Life Improvement Program) de General Atomics con un enriquecimiento en U-235 del 70%.
La principal razón de tener un núcleo mixto fue el hecho de que no se tenían suficientes combustibles tipo FLIP para conformar un núcleo completo, por lo que se estudió la mejor manera de utilizar el existente.

Se procedió a licenciar el nuevo núcleo del reactor, generándose un reporte técnico (7) como anexo al informe de Seguridad. Este reporte contiene el análisis de la nueva configuración y su impacto en la operación segura de la instalación, a) Análisis neutrónico del núcleo mixto, b) Cálculo del valor de barras de control y margen de apagado, c) Análisis del comportamiento térmico del núcleo mixto, d) Análisis de accidentes y e) Especificaciones técnicas.

Después del procedimiento normal de evaluación de la configuración propuesta y una vez contestadas a satisfacción las preguntas generadas por la CNSNS, ésta autorizó la carga del núcleo mixto del reactor que en la presente configuración opera con 59 combustibles estándar previamente irradiado, 26 combustibles tipo FLIP y 3 seguidores de combustible tipo FLIP de las barras de control. Después de haber cargado el núcleo mixto y realizado las mediciones de margen de apagado, y de comprobar que se opera dentro de los límites establecidos en las especificaciones técnicas, la CNSNS otorgó un permiso provisional de operación que será cambiado por la licencia definitiva de operación una vez que sea expedida por la Secretaría de Energía Minas e Industria Paraestatal quien tiene bajo su responsabilidad la expedición de dichas licencias.

4. CONCLUSIONES.

El proceso de licenciamiento ante la CNSNS fue de mutuo aprendizaje. La experiencia del organismo regulador fue obtenida principalmente de la evaluación de una planta nuclear de potencia, mientras que en nuestro caso era inexistente, por lo que resultó un tanto difícil unificar criterios de aceptabilidad y alcance de la normativa aplicable. Sin embargo, gracias a la buena disposición de ambas partes, se pudieron sortear dificultades y finalmente se establecieron los requisitos mínimos para la operación del reactor en condiciones de seguridad.

El licenciamiento del reactor del Centro Nuclear de México representó pues, un gran esfuerzo por parte de todas las personas involucradas, pero creemos que los beneficios fueron igualmente grandes. Se adquirió experiencia e; la implementación y aplicación del Programa de Garantía de Calidad, en la utilización de normas y estándares, en la elaboración de procedimientos e instrucciones y en general, se mejoraron las actividades de dirección y supervisión de la operación del reactor.
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THE REGULATION AND LICENSING OF RESEARCH
REACTORS AND ASSOCIATED FACILITIES
IN THE UNITED KINGDOM

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ABSTRACT

In the United Kingdom, the Nuclear Installations Inspectorate (NII) licenses nuclear facilities, including research reactors, on behalf of the Health and Safety Executive (HSE). The legislation, the regulatory organisations and the methods of operation that have been developed over the last 30 years result in a largely non-prescriptive form of control that is well suited to research reactors. The most important part of this regulatory system is the licence and the attachment of conditions which it permits. These conditions require the licensee to prepare arrangements to control the safety of the facility. In doing so the licensee is encouraged to develop a "safety culture" within its organisation. This is particularly important for research reactors which may have limited staff resources and where the ability, and at times the need, to have access to the core is much greater than for nuclear power plants. Present day issues such as the ageing of nuclear facilities, public access to the rationale behind regulatory decisions, and the emergence of more stringent safety requirements, which include a need for quantified safety criteria, have been addressed by the NII. This paper explores the relevance of such issues to the regulation of research reactors. In particular, it discusses some of the factors associated with research reactors that should be considered in developing criteria for the tolerability of risk from these nuclear facilities. From a consideration of these factors, it is the authors' view that the range of tolerable risk to the public from the operation of new research reactors may be expected to be more stringent than similar criteria for new nuclear power plants. Whereas the criteria for tolerable risk for research reactor workers are expected to be about the same as those for power reactor workers.
1. BACKGROUND

In the United Kingdom (UK) the main legislation governing the safety of nuclear installations, including research reactors and associated facilities, is the Health and Safety at Work etc Act 1974 (HSAW), [1], the relevant statutory provisions of the Nuclear Installations Act 1965 (NIA), [2], and the Ionising Radiation Regulations 1985 (IRRs), [3], made under the HSAW.

Under these Acts no site in the UK may be used for the purpose of installing or operating a research reactor, except for sites operated by the United Kingdom Atomic Energy Authority (UKAEA) or a government department, unless a nuclear site licence has been granted to a corporate body by the Health and Safety Executive (HSE) and is for the time being in force.

While at present the UKAEA research reactors do not operate on the basis of a licence granted by the HSE, it is intended that they will do so in the very near future. However, they have been required by a Ministerial Directive to have regard to any safety requirements ordinarily imposed on licensed operators. In addition, the UKAEA's establishments are regularly inspected by HSE inspectors under the general powers in the HSAW Act and to ensure compliance with the IRR's (see ref. [4]).

Similarly, government departments who may wish to operate a research reactor are obliged to have regard to the safety standards and requirements imposed on licensed facilities.

The UK regulatory system for research reactors is no different from that for other types of nuclear facilities in the UK; the same laws apply, the same regulations and regulators are involved and the same fundamental safety principles and licensing requirements are applied. However, as with all nuclear facilities, the extent and depth of the provisions necessary to ensure safety and of the safety analysis depends upon the hazard potential of the facility. The responsibility for the safety of the facility and for demonstrating the adequacy of the provisions and arrangements for safety rests with the licensee. Nevertheless, the regulators must ultimately make a judgement on the adequacy of these provisions and of the licensee's safety case.

The primary safety principle that the HSAW Act sets down is that the employer who controls an industrial plant must do whatever is reasonably practicable to ensure that the plant is safe and without risk to health. This has had the benefit of legal interpretation (see ref. [5]) as implying:
"... that a computation must be made in which the quantum of risk is placed in one scale and the sacrifice, whether in money, time or trouble, involved in the measures necessary to avert the risk is placed in the other; and that, if it be shown that there is a gross disproportion between them, the risk being insignificant in relation to the sacrifice, the person upon whom the duty is laid discharges the burden of proving that compliance was not reasonably practicable ... ".

2. REGULATORY ORGANISATIONS

The HSAW Act, as well as laying down general principles and requirements to control safety in the UK, provides the present basis for the regulatory organisation for safety. Two bodies were set up under this Act; the Health and Safety Commission (HSC) and the Health and Safety Executive (HSE).

The basic role of the HSC is to define the standards of safety to which industry is expected to operate. In performing this function it advises Ministers of requirements for legislation and oversees its implementation. The HSC consists of eight members representing Industry, the Trade Unions and the general public via Local Authority representatives. In addition, a chairman is appointed by the Secretary of State for Employment.

The Health and Safety Executive is responsible for implementing safety legislation, providing advice to the HSC and for promoting safety. It is a corporate body of three people who are appointed by the HSC and has some 3500 employees to assist it in its duties with an annual budget of around £110,000,000.

Under the HSAW Act the HSE has responsibilities for licensing nuclear facilities as required by the Nuclear Installations Act. The Nuclear Installations Inspectorate (NII) is that part of the HSE that is responsible, on HSE's behalf, for undertaking this licensing function and associated duties (eg inspections, assessments and advising on policy) for nuclear facilities.

The NII's organisation reflects these duties in that it has Branches whose main responsibilities cover either policy, assessment or inspection. At present it has some 140 professional staff in post to regulate all the licensed nuclear facilities and UKAEA nuclear sites in the UK. This is intended to rise to around 160 by early 1990. About 30% of NII professional staff are engaged on site inspection duties although assessment and, to a lesser extent, policy staff...
make site visits. Normally, for the smaller installations, such as research reactors, one designated NII inspector (the Site Inspector) would be responsible for enforcing the relevant law for several installations. However, he or she will be backed up by all the resources of the NII, and indeed, the HSE.

It can be deduced from the above that the organisations that have been set up to regulate nuclear facilities in the UK can form the basis for achieving social and industrial consensus and which are independent of government departments.

Thus, the Chief Inspector of Nuclear Installations can decide, on behalf of the HSE, to license or revoke a licence without interference from government departments or the nuclear industry. The regulators are answerable, nevertheless, for their actions to the representatives of the public via the HSC and ultimately parliament. Furthermore, by being required to give evidence at public inquiries on measures taken to regulate risks in respect of proposed developments of nuclear installations, the workings and decisions of the NII are exposed to public scrutiny.

As well as the above regulatory organisation for nuclear safety there are other bodies, the Authorising Departments, associated with controlling the activities of nuclear facilities. These bodies, which are parts of government departments, have the responsibility for authorising the disposal of radioactive waste from nuclear sites.

3. LICENSING OF RESEARCH REACTORS

3.1 The Licensing Process

The basis for licensing a research reactor is the same as that for other nuclear facilities in that the regulator must be satisfied that the prospective licensee has demonstrated that all reasonably practicable steps will be taken to eliminate accidents and to minimise their consequences should they occur. Similarly, in relation to radiation exposures from normal operation, it must be shown that doses will be kept as low as is reasonably practicable (ALARP) and below the permissible levels set down in the IRRs.

Licensing can be a long and complex process involving consideration of:

(a) need for a licence;
(b) competence of the applicant;
(c) siting (see ref. [6]);
(d) type of installation (especially any novel aspects);
(e) planning concerns;
(f) adequacy of the safety case;
(g) tolerability of the predicted risk to the workers and the public, etc.

Even when a licence is granted there will still be constraints on the licensee via the attachment of conditions to the licence, including a requirement to obtain the HSE's consent to operating the new facility.

Also, it may be that at some stage, prior to granting a licence, a public inquiry is called which then involves the public examination of the proposals. Objectors to the proposal are able to put forward their views and to question other witnesses who may be the applicant or, indeed, the regulators. The regulators position in these matters is one of neutrality being merely concerned with administering the law with respect to the safety of nuclear installations.

With regard to proposals to license existing facilities, the steps and principles are similar but account has to be taken of the state of the facilities as they exist, their operating history, design life limitations etc.

In either case, whether it be the licensing of a new facility or an existing one, the licensee will be obliged to develop adequate arrangements to address the various requirements of the nuclear site licence conditions.

3.2 Nuclear Site Licence Conditions

The Nuclear Installations Act provides powers to the HSE, which are delegated to the NII, to attach such conditions to a licence as it thinks fit in the interests of safety. These conditions may be added to, amended or revoked at any time during the period when a licence is in force. This provides a very powerful and flexible regime of regulatory control of nuclear safety without diluting the licensee's absolute responsibility for nuclear safety under the NIA. The conditions can be modified to take advantage of the experience and knowledge gained during the operation of the facility.

Furthermore, experience of the regulation of a wider range of nuclear facilities can be brought to bear on a narrower class of facilities such as research reactors. Indeed, it has been found from the wide experience of NII in regulating the whole range of nuclear installations (zero energy research reactors, isotope production facilities, fuel fabrication plants, reprocessing plants, nuclear power plants etc) that the same licence conditions, as developed over more than 30 years, can be applied to all different types of nuclear
facilities. Consequently, the NII is now moving to a position where the same licence conditions, embodied in what is called the "model licence", can be applied to all licensed sites.

The licence conditions apply to all stages of a research reactor project including design, construction, installation, commissioning, operation, maintenance, testing and inspection, modification and decommissioning. The conditions are essentially non-prescriptive. Generally, they require the licensee to prepare adequate arrangements, part of which the NII may require to be submitted to the NII for approval. These arrangements, amongst other things, cover such matters as emergency plans, operating rules and instructions, maintenance, radiological protection, nuclear safety committees, incidents on the site, safety documentation, training, construction of new plant, modifications and experiments on existing plant. Usually, the conditions include one or more of the following constraints:

(a) approvals - the licensee is required to obtain NII's approval of particular arrangements for controlling specific activities;

(b) consents - the licensee cannot carry out a particular activity, such as recommissioning the reactor until the NII has agreed to it; and

(c) directions - these enable the NII to require the licensee to take an action which the NII considers necessary (normally only used in extreme circumstances, e.g. to enforce a reactor shut down).

In specifying a requirement for arrangements for a certain activity, the NII does not dictate how to undertake that activity or how such a requirement can be met. In granting its approval to the licensee's arrangements the NII is stating that it has no objection to them and that it considers that they are adequate for the particular facility and circumstances under consideration at that time.

This non-prescriptive nature of the UK licensing and regulatory system for nuclear facilities allows the same licence conditions to be applied over a wide range of facilities and, moreover, maintains the position that the ultimate responsibility for safety rests with the licensee.

3.3 Benefits and Difficulties Associated with a Non-prescriptive Licensing Approach

The licensing system in the UK requires each licensee to
provide a rigorous proof of safe operation in the form of a safety case for the particular facility under its control. This may lead to different solutions to the same safety problem by different licensees. While this may appear wasteful in overall resource terms it can lead to a faster development of new and cost effective safety improvements. The regulator can act as a focal point in this development by encouraging licensees to think along new lines or adopt different practices.

However, the regulator in such circumstances must be careful not to dilute the commitment of the licensee by the transfer of some part of the licensee's responsibilities for ensuring safety to the regulator. The regulator must also avoid a confusion of function by not straying into the province of the designer or operator. He avoids these pitfalls by stating, as far as he can, the grounds on which he can be satisfied and by proceeding by careful questioning and then indicating his satisfaction or otherwise, rather than proposing specific courses of action.

These difficulties are implicit in all forms of safety regulation, not just that of licensing. However, the closeness of the interaction between licensee and regulator, and the complete dependence of the licensee in seeking permission to proceed with an activity, demands a very high level of management and technical competence on the regulator's part. These attributes are required, not least, to prevent excessive delay and unnecessary "ratchetting" of the safety requirement.

Despite these difficulties there are great benefits to be gained, in safety terms, from adopting a non-prescriptive licensing approach to the regulation of research reactors. These benefits relate to the development of a "safety culture" within the licensee's organisation with "management for safety" forming a prime objective of the operating organisation. This requirement is especially important for research reactor facilities which may have limited staff resources and where the ability, and at times the need, to have access to the core is much greater than for power reactors.

4. CURRENT AND FUTURE ISSUES

4.1 Public Access to the Rationale behind Regulatory Decisions

The public's perception of nuclear facilities has changed over the years to be much more sceptical and demanding in relation to safety requirements and demonstrating adequate
safety. This encourages regulators to provide more information on the background to their judgements.

In the UK this issue has been addressed at a fundamental level by the regulator providing:

(a) information on the principles that the NII use in assessing the safety of nuclear facilities [7,8];

(b) evidence to public inquiries on the proposed development of nuclear power facilities [9,10];

(c) a discussion document on the tolerability of risk from nuclear power stations for public comment [11], and a compilation of the comments received [12];

(d) information on the use of quantified risk assessment as an input to decision making [13];

(e) public information on the results of a major safety audit [14] and of NII assessments of a licensee's cases for continued operation of ageing plants [15,16]; and

(f) information on the regulatory basis for allowing the continued operation of some older research reactors [4].

These initiatives provide the means for extending the societal consensus for the regulatory response to the risks associated with UK nuclear facilities including research reactors. It is clear that there will be a continuing and expanding requirement to provide such information which can have particular relevance to the continued operation of ageing research reactors.

4.2 Ageing Facilities

4.2.1 General

Of the UK's operating research reactors the youngest is at least 17 years old and others are significantly older (see ref.[17]). These reactors are based on the technology and design concepts that were available in the 1960's. Present day safety requirements for a new research reactor are likely to be more stringent than those that existed over 30 years ago. This results from a general increase in standards and expectation, and from the development of the technology and methodology available to both the research reactor designer and safety assessor. These changes also influence the safety requirements for the continued operation of existing research reactors.
In the UK, the general approach of the regulator to such issues includes undertaking team audits of ageing facilities and requiring licensees to produce long term safety reviews of their facilities.

4.2.2 Long Term Safety Reviews (LTSRs)

This approach to re-assessing the safety of ageing nuclear facilities has been successfully applied to some of the oldest operating nuclear power plants in the world [15,16]. It can also be applied to research reactors. In the LTSR the licensee is expected:

(a) to confirm that the plant is adequately safe for continued operation;

(b) to identify and evaluate any factors which may limit the safe operation of the plant in the foreseeable future; and

(c) to assess the plants safety standards and practices and introduce any improvements which are reasonably practicable.

As well as reviewing the plant as it is now against the original safety standards, the LTSR requires the licensee to compare the existing levels of safety with modern standards and practices. By comparing with present day standards and practices shortcomings may be identified. Any improvements agreed to be reasonably practicable would be introduced to an agreed programme.

In considering the need to apply such reviews to research reactors, consideration has to be taken of the operational history of such facilities, their potential for harm to the public or workers and any public concern on the safety of these plants. Furthermore, the scope and necessity for such a review may be reduced by the periodic updating of the safety case for the facility.

4.2.3 Periodic Reviews of Safety Cases

A comprehensive safety case is an important starting point for ensuring safety. It forms the basis for licensing the site and other regulatory controls. It is one of the sources for the design, commissioning and operating constraints, arrangements and requirements. However, the safety case is not a once and for all time matter. It should be developed, reviewed and modified throughout the project life cycle.
In particular, experience from the operation of the research reactor can provide an important feedback into the safety case to test the original assumptions etc. Thus, the safety case will need to be reviewed periodically throughout the operating life of the plant to take account of this new information, new assessment techniques and developments in safety standards, and any significant plant modifications.

4.3 Safety Criteria and Tolerability of Risk

4.3.1 General Requirements

For many years the NII has had assessment principles \cite{7,8} relating to nuclear power plants and nuclear chemical plants which include quantified safety assessment criteria. The fundamental safety principles included in these publications are relevant to all nuclear facilities including research reactors.

More recently the HSE has produced a discussion document on guidelines for tolerable risk from nuclear power stations \cite{11}. "Tolerable" is by no means the same as "acceptable" in that it indicates the level of residual risk that society is prepared to live with in order to obtain certain benefits and in the confidence that the risk is being properly controlled. Whereas "acceptable" risk is generally regarded by those who are exposed to it as not worth worrying about.

While the actual bounding levels of tolerable risk derived in the discussion document for new nuclear power reactors \cite{11} may be subject to further development, as a consequence of the discussion and consultation process, the concepts can provide a starting point for considering what might be the tolerable risk from new research reactors. These include the concepts of different levels of risk which are, in decreasing magnitude of risk, : an "intolerable level of risk"; the "ALARP region"; and the lowest risk region - the "broadly acceptable region".

The "intolerable" level of risk cannot be justified on any grounds. Regulators would seek to ensure that no one is exposed to such levels of risk. This level is not immutable, however, being a matter for political decision based on society's perception of the hazard and contemporary realities. Below such levels exposure to the risk is only tolerated if it brings certain benefits and is reduced to as low as is reasonably practicable (ALARP). This is the "ALARP region. As the risk is reduced further the "broadly acceptable region" is entered where the risk becomes negligible in that, apart from taking usual precautions, it is wholly ignored.
Generally, tolerability of risk will be determined by the degree to which those exposed to it gain benefits from the activity and voluntarily incur the risk.

4.3.2 Tolerability of Individual Worker Risk from a Research Reactor

For a research reactor, it is the view of the authors of this paper that all those groups associated with the operation of the facility (eg plant operators, students and experimentalists) incur the risk from the facility voluntarily and gain benefits significantly greater than those arising to the general public from the operation of the facility. These benefits are similar to those that other workers in the nuclear industries gain from their employment. Thus, despite the different benefits to society that the public may perceive between a research reactor and a power reactor, the range of tolerable risk for workers is judged to be about the same in both cases. There may be differences in the level of safety that the regulator deems to be adequate in practice, however, amongst different facilities. This is a consequence of the application of the ALARP principle.

4.3.3 Tolerability of Risk to Members of the Public from a Research Reactor

Several factors have to be taken into account when applying "tolerability" concepts to the risk to the public from a specific new research reactor. These include:

(a) the proximity of the facility to large groups of the public (eg a research reactor situated in a city);

(b) the need to consider what other nuclear facilities are on the same site as that proposed for a new research reactor (it may be but one of ten such facilities on a large nuclear research site);

(c) the large range in potential hazards to the public that "research reactors", as a species of nuclear facilities, can pose as a consequence of the spread in power rating that the term "research reactor" covers, e.g. a factor of around 100,000 from zero energy to 100 MW facilities - see ref. [14], (assuming that potential hazard is proportional to power rating); and

(d) differences in the public perception of the benefits associated with a group of research reactors to those arising from a nuclear power plant programme.
From a consideration of these factors, it is apparent to the authors of this paper that it may be expected that the range of tolerable risk to individual members of the public from the operation of new research reactors is likely to be more stringent than that derived for new nuclear power reactors.

It is suggested that the application of the ALARP criteria below the "intolerable risk" region should provide the means of addressing the wide range of potential hazards associated with research reactors of vastly different radioactive inventories (see (c) above). As the inventory increases so the regulator will expect more design provisions to reduce the risk to a level that is ALARP.

Furthermore, in relation to societal risk, it is considered that the public's perception of the benefits associated with research reactors of a size which could give rise to significant off-site effects are likely to be considerably less than those ascribed to a major nuclear power plant programme. Therefore, it may be expected that the criteria for the societal risk associated with a research reactor programme could be much more stringent than those derived for a nuclear power plant programme.

5. CONCLUSIONS

This paper has illustrated how the flexible, but strict, non-prescriptive UK regulatory system is particularly relevant to the regulation of ageing research reactors operating in a changing social climate. With such a system regulators have the ability to respond to new issues such as the development of quantified safety criteria for research reactors.

(NOTE: This paper represents the views of the authors and does not necessarily represent those of the NII nor the HSE.)
6. REFERENCES

EXPERIENCE AND LESSONS LEARNED IN THE ASSESSMENT OF
SAFETY JUSTIFICATIONS FOR EXPERIMENTS MOUNTED IN
RESEARCH REACTORS

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ABSTRACT

Some experiments in research reactors are arguably a risky undertaking due to their uncertain outcome. The justifications for such experiments require careful assessment to validate their undertaking. The public, the operators and the installation itself must be safeguarded. Assessment of the potential risk is an acquired skill but in doing so the route can be eased by learning from the lessons experience can teach. This paper, essentially for the usage of safety managers, sets out some of the issues relating to the assessment process, gained from our experience over a few tens-of-years in the assessment of experiments. Many of the conclusions reached may appear all too obvious viewed in retrospect, but they were not necessarily clear at the time. Those organisations setting up assessment teams may find some of the conclusions of value such that their proposed management system can embrace methodologies for assessment that can avoid or lessen the impact of some of the pitfalls we have tried to identify. Failure to recognise some of these points may run the risk of delayed clearances, dilated timescales and cost over-runs. It is in the hope of reducing all these penalties that we offer our experiences.

1. BACKGROUND INTRODUCTION

There is little doubt that some experiments mounted in research reactors represent a potential threat to safety. Lack of knowledge has led to the experiment and this must reflect in a measure of uncertainty in the outcome. Furthermore the experiment will have a measure of human involvement resulting in the use of judgement, during the experiment. Whilst careful design, planning to meet unexpected events and recovery procedures will be available nevertheless rigs or loops or experiments do attract a significant measure of uncertainty, compounded with the risk of human error.
There is a need for safety both to protect the public, the reactor operators and the investment. It is important to demonstrate that experiments are acceptably safe.

The scene is set for the operator/scientist to compose a justification for the experiment. This justification will require validation by a process of assessment, probably by a safety organisation. This paper, written essentially for the benefit of emerging organisations, with a need for guidance into the processes of regulation, will discuss experience and lessons learned in the assessment of safety justifications. In retrospect much of the material in this paper may be viewed as obvious and common sense. In our experience involvement in the technical issues of assessment can blind the participants to some of the important organisational and management features of the assessment process.

The opinions in this paper are those of the author alone and do not necessarily reflect the policy of SRD or the UKAEA.

1.1 Assessors and Designers/Operators

Where does the safety assessment process start? It does not commence with the submission, to the safety organisation of a series of documents that describe and justify the safety of the experiment already designed and constructed by the operator scientist. Such an approach can only mean the Assessors are being presented with an attempted fait accompli. Whilst this may appear to be a very unlikely episode nevertheless attempts to complete a justification very late in the proceedings, for equipment already manufactured, have been experienced and are often an outcome of a history of changes or modifications to the original design made in the hope that the justification will be finally found acceptable.

Here then is one of the great problems underlying the production of a complicated experiment or rig. Whilst the initial concept is made clear and the basic principles laid out, the criteria for acceptability described and the safety assessors brought into early phases, nevertheless the clearance of the equipment at the feasibility stage can only be in principle. At this stage there is very little detail available for safety assessment since the design concept is based on broad ideas set up to show the experiment can be conducted within an apparent budget and timescale.
The project relies entirely on the skill and experience of the engineering designers and their ability to plan a programme for design, construction and loading of the experiment into the reactor. It is at this early stage in concept that the designer and safety authority must come together and thoroughly understand and appreciate exactly what phases and processes of assessment lie ahead. The designers should appreciate not merely the rather obvious aspects of safety such as the definition of acceptable codes of practice, criteria for the acceptability of risk etc etc, but should understand too how the assessor will analyse the justification. What methods will be used and how will they be applied?

The methodologies for safety assessment should be properly understood by the designers and interpreted to those designers by the safety engineers in documented notes that record the level of mutual appreciation.

If an analyst describes to a designer the tools he will use to assess the safety of a rig this must have a profound impact on not only the justification of the design but even on the design itself. It is easier to show a design may be found wanting, to a greater or lesser extent by an analyst, when the designer proceeds in his own manner, than when the designer has followed the procedures and methodologies of the analyst. In the latter case the stones to be turned by the analyst have already been turned by the designer in his justification. What then does the assessor want to see, how can he be clearly satisfied?

1.2 The Creation of the Safety Justification

Firstly, before answering the question let us consider, two possible approaches to the creation of safety justifications. The safety regulator can prescribe a large range of criteria and limits which must be met by the design. The designers task is to produce a design that fits within this prescribed set of rules. Here the onus for the degree of safety likely to be achieved is that of the regulator. It is his skill, experience and judgement that has led to a set of rules which govern the standard of safety required. There is not necessarily any fault in this arrangement since in an organisation with limited resources or where central direction is found to produce the best results then the process can be satisfactory.
The alternative is to place the onus for creating the safety case upon the operator, the regulatory body only giving the major principles and criteria for acceptability, leaving the margins, limits and rules for the operator to justify. This approach is based on the belief that it is better for the operator to understand and appreciate the full impact of safety issues since he will be managing the risk. The assessor will confirm in his analysis that sufficient work has been done to make out an adequate justification.

This strategy is satisfactory where the supplicant has the resources (skills, manpower and experience). Without these resources it may be better for smaller nuclear research endeavours to adopt the former approach. Our experience relates to the operator justifying the safety case and accepting the onus that the burden of proof lies on him.

It is vitally important, in our view that the designer/operator understands clearly how the justification will be analysed. This is not always the case. Experience has been obtained of shortcomings in the acceptance of a justification. For some equipments the designer/operators has arrived at the safety requirements by following time honoured routes based on a deterministic approach to the safety case. Such justifications have been assessed in a routine manner and found acceptable. Difficulties arise when the designer follows a traditional approach and the assessor, recognising that the safety issues may not be met merely by traditional deterministic routes, seeks other methodologies, such as the probabilistic approach, particularly where beyond design basis events are foreseen. Historically the point has been passed where a justification could be made out "incredible of failure" ie the chance of failure was inconceivable.

Nevertheless designers still need reminding that a full justification may be sought not only through deterministic methods but also via probabilistic safety analyses. Hence the vital importance of the designer/operator knowing from the outset exactly what assessment methods and uncertainty analyses are likely to be applied. A running dialogue from initiation of the project through the agreed assessment milestones can avoid misunderstandings, allegations of changes in standards of acceptability and other such recriminations. These latter elements result in delays, cost increases and resource crises as the programme dilates.
The first major conclusions therefore are:

(a) Make clear exactly what the criteria are for acceptable safety.

(b) Design/operator to discuss and identify the methodologies for safety (deterministic, probabilistic analysis).

(c) Declare the safety principles (typically redundancy, diversity, segregation, common cause etc).

(d) Define the design limits, maxima, minima etc in a specification.

(e) Document the above items in a memorandum of understanding, at the outset of project or undertaking.

Misunderstandings, misconceptions and divisions of opinion can be avoided if these precautions are adopted at the outset.

2. PROBLEMS ARISING AFTER THE IMPLEMENTATION OF THE INITIAL MEMORANDUM OF UNDERSTANDING

As a complex rig is progressed from the drawing board into reality so, as with most engineering designs some working compromises may be needed. It may transpire that combinations of elements such as pressure, stress and heat transfer require reoptimisation in order to stay within the envelope of the acceptance criteria. This is normal.

Change however can reflect on safety, and experience shows that this is not always properly appreciated, or noted too late after a milestone has been passed when turning back is seen as "impossible". This is potentially a very serious stage in the proceedings and every effort must be made to ensure the safety assessors are aware of the detail relating to the changes.

The response of the safety authority must be, in our view, clearly stated in terms of what extra requirements may result from the changes or modifications. Any design modifications, manufacturing errors, and deficiencies not within the original specification must be reported to the safety authority.

During the modification the designer must maintain full quality assurance for the data and their influence on the safety case. Whilst this precaution may appear to be elementary our experience is that full consultation on changes,
from the onset of change, is not necessarily undertaken. For example the assessor should be present when any trial is being planned to acquire data important to the justification so that its relevance can be confirmed.

Programme time must be allowed for not only producing safety documentation, so that it is available to the assessor but also time must be permitted for the inevitable secondary phase in an assessment when points arising require response from the designer/operator. These secondary questions are usually related to prime safety issues where reassurance is sought to satisfy the assessor.

This secondary phase is arguably the most important since doubts in the mind of the assessor are likely to be significant to safety and may require some effort to assuage. Programmes rarely if ever, have built into them time for the secondary phase to be resolved yet in our opinion it is this secondary phase that requires significant effort.

It is almost impossible to anticipate what particular technical issue may alert the assessor to a need for greater in-depth treatment than that supplied. However the major safety issues can be identified at an early stage. The designer and assessor should get together and debate these safety points so that when the justification is produced it is very likely to create only relatively minor secondary questions.

Here the liaison arrangement must be seen as a delicate and carefully balanced exercise; the assessor must not tell the designer literally what must be provided. Nor must the designer ask the assessor what it is exactly that is wanted.

We see these liaison meetings as a vital plank in the creation of a system for mutual respect between the designer/operator and assessor. This dialogue must be kept up throughout the safety campaign. Each party can guard its independence through firm but amicable dialogue, without collusion, and agree on plans to achieve the common goals. Agreement on resolved safety issues should always be documented.

Once this level of understanding is reached the safety justification can be more rapidly produced since the direction, scope and range of the document are settled. This liaison process is infinitely more productive than the confrontation
attitude ("this is what he will get" versus "this is what I want" with neither party prepared to readily give way).

The second set of conclusions are therefore:

(a) Involve the designers and assessors together with design changes, modifications, tests and repairs that influence safety. Categorise modifications if numerous with the highest class requiring clearance by the assessor (but notify the other categories too).

(b) Present safety information such that the assessor is given reasonable time to reflect on the material; do not expect an "instant assessment".

(c) Allow time for secondary questions from the assessor in the safety programme; these questions are likely to be the more significant ones.

(d) Plan to identify the safety issues of significance at an early stage together with the designer and the assessor and the likely means to resolve them in principle.

(e) Create a dialogue for the exchange of opinion, to create a middle ground, documented as a record of agreement and understanding.

3. THE REVIEW PROCESS

Common sense statements of the obvious would appear to be expected in a safety justification, yet surprisingly, this is by no means always the case. The assessor should find a clear statement of what the safety case is trying to prove. The proposer should define the aims and objectives of the case and state the criteria for acceptance, the methods used to formulate the safety arguments, the design specification reference and any special bounding rules. These statements should set out the stall for the detailed arguments and justifications to follow, confirming that the case is likely to be soundly based. Regrettably such statements are often absent from submissions, with the assessor left to form his own views!

Reviewing is not checking. It is expected that all calculations will have been checked by a QA system operated by the designer. A paper describing the QA system and its management could be an annex to the safety submission. The assessor will however review calculational methods such as computer codes and form a view as to their suitability and whether
the code is properly validated, particularly where the designer has developed a code specifically. Input data for use with codes should be pedigreed. In novel situations it may be necessary to carry out some independent calculations to verify the safety case. Production of independent calculational methods must be limited, at the discretion of the assessor, to important results however.

The assessor must satisfy himself that there are no omissions or deficiencies in the case. Evidence for the proper use of safety methodologies would enable an assessor to conclude that a rigorous approach had been adopted. Clear arguments supported by validated data, well recognised theory and qualified calculations methods again provide evidence of a robust approach.

The reviewer should ensure the criteria are acceptable, complete, have been properly applied and that an adequate margin to safety has been achieved. The omission of attention to a criterion should not arise at a late stage if the liaison between the designer and assessor has proceeded harmoniously. Raising issues relating to criteria at a late stage can only create delay to the clearance process. Where the margins are apparently small, then an uncertainty analysis may be called for as well as a sensitivity analysis. The underlying assumptions adopted may play a crucial part. We have found that uncertainty analyses have been omitted or limited or the data left for the assessor to carry out his own analysis! Uncertainty should always play a part in the analysis, in our view. It is essential that the designer/operator is absolutely honest and frank regarding the state of the design, its analysis and the physical condition of the actual hardware. Hardware quality will rest on the reports of inspections and tests carried out.

All deficiencies located during test and inspection must be reported to the safety organisation. Judgement will be applied as to whether a defect falls above or below a reportable level. The safety authority should be made aware that non-reportable defects exist and that they have been judgementally ignored. The safety authority may wish to call in such reports to confirm that judgement has been correctly applied. Our experience is that defects or deficiencies have occurred but have not been properly brought to the attention of the assessor on the grounds that it was a matter of no great concern, and therefore not documented.
Our view is that all information relating to deficiencies however apparently trivial must be recorded. The evidence should be judged by interested parties, as to its significance at the moment it arises but it must be documented, as part of the quality plan and it must be dealt with promptly. Later thoughts regarding the significance of a fault or defect, leading to the need to retest or repair may arise at an extremely inconvenient phase of the programme, adding cost and delay in excess of the immediate cost for retesting, or repair, at the outset.

Our experience is that the assessor is insufficiently brought into the confidence of the designer/operator early enough to defuse events. Why should this be so? Our view is that assessors can be too aggressive at the early stages of a design concept and overdo criticism when guidance and comment and encouragement or even sympathy would be the better approach.

Assessors can too easily forget that the amount of engineering compromise to be exercised in the formative stages of a safety study for a complex rig can leave the concept vulnerable to criticism. Heavy criticism at this stage is most inadvisable since the reaction of the designer will be to reveal details when they are more robust. The withdrawal phase to harden-up the design results in the exclusion of the assessor and a division can be created between the parties leading to risk of confrontation and fixed positions. Assessors should be more humble in the knowledge that their comments and criticisms carry great weight with the designers and an "enabling" attitude should prevail. This enabling role is the constructive side of the assessor's task; adverse criticism can tend to be destructive if insensitively utilised.

If an atmosphere of constructive criticism and enabling comment is to prevail then mutual respect and appreciation of the roles of the participants, designers, operators and assessors must be created.

Where the outcome of the justification for a safety topic has a major influence this fact should be flagged up as a critical item and receive special attention. The assessor should clearly state what, in principle, are his major concerns so that the designer/operator is in no doubt as to the points requiring special attention. Failure of the assessor to make his views known and indifference on the part of the designer to meet the viewpoint of the assessor can lead to an impasse.
Undoubtedly safety costs money and at the heart of the problem lies the thought that "too much effort is being spent on unnecessary studies to satisfy safety engineers who are unsure themselves as to when adequate margins are available". The dialogue between the assessor and the designer should establish frankly what the assessor sees as the deficiencies leading to inadequate margin. Whilst safety is not an item to be compromised it is certain that a measure of give and take where judgement is concerned is the hallmark that a satisfactory middle ground is being established.

Since nothing is ever likely to be absolutely safe, yielding by the assessor of a few points in favour of the designer is not likely to compromise safety to any significant extent, yet the effect on the team work needed to progress the safety case can be significant. Dogmatic insistence on every point at issue is not only aggravating but undoubtedly incorrect and symptomatic of lack of judgement.

The third set of conclusions are:

(a) The justification should make clear statements of what the safety case is trying to prove.

(b) Reviewing is not checking; it is a process for confirming that the science, data, methodologies, computer codes and their application are well recognised, valid and properly utilised and that the conclusions are sound and reasonable.

(c) A QA system that provides the evidence for the validity of the calculations, inspection data and test procedures should exist.

(d) Independent calculations by the assessor, utilising methods devised if necessary by himself, are an important aspect of review, particularly when applied to key safety issues (for which they should essentially be reserved).

(e) Identified criteria should be addressed at a very early stage in a dialogue of mutual understanding between the designer and the assessor.

(f) All relevant information relating to the design should be documented and declared to the safety authority. Complete frankness is essential.
(g) The designer/operator should bring the assessor into his confidence. Assessors should remember the sensitivity of designers to premature harsh criticism and always be constructive in their approach.

(h) The assessor should make it clear to the designer/operator exactly what his concerns are relating to major safety issues. The designer/operator is to be in no doubt regarding their importance.

(i) A climate of give and take on marginal issues should prevail with neither party adopting entrenched positions. The objective is to create an harmonious team relationship based on firm but fair and credible viewpoint.

(j) Be reasonable in mind and attitude.

4. GUIDELINES FOR ASSESSORS

Whilst checklists for assessors are undoubtedly of great value in ensuring nothing of importance is missed, it is not enough alone to ensure a satisfactory assessment of the justification. Assessors must exercise their own judgement as a skill, and not rely on check lists.

There is no check list that, if fully completed, will ensure the installation is adequately safe. The safety assessor should be experienced in design or operation or both. With experience the safety assessor will acquire an intuitive feel for critical issues.

Some of the methodologies adopted for analysis such as fault and event trees for example are powerful in that they represent a framework within which the discipline for identifying likely safety issues can be exercised and the hazards ranked as to importance. Such methods are of undoubted value and provide insights to safety matters or dependence between events that mere passing inspection cannot reveal. A fault tree analysis can unveil subtle links between events which on first examination may lie unsuspected. It is this ability to create a discipline for analysis that is the strength of the PSA methodologies.

The lesson emerging by analogy is that assessors too must have a personal methodology for assessment that is logical, comprehensive, based on extensive experience capable of flexible adaptation to novel
situations and soundly based on safety principles. Such a mental discipline must be generated; a haphazard approach will fail to identify weaknesses in a case.

Throughout the process of assessment the assessor will be asking himself questions such as "is this reasonable, is this valid, is this recognisable as sound". To answer the questions the assessor must use his own judgement. Case studies from previous experience backed up by sound technical training in science engineering and safety principles will be drawn on mentally when considering whether the issue in hand has an argument that comes up to expectations. This is obviously a very human process and like so many human activities is vulnerable to factors that can influence judgement such as stress, lack of confidence, sense of isolation.

Our experience is to use a small team of 2 or 3 assessors who meet together with the group leader and debate the issues, ventilating doubts, proposing provisional solutions, adopting a devil's advocate stance etc, until judgement can be applied on a reasonable basis to all the points of view, in some middle ground. The group leader acts as a chairman, raising issues, challenging opinions, seeking views, exploring ground until the real position emerges and final judgement can be made.

It is clear from the above that instant assessment mentioned earlier on in this paper is not a valid approach. In the above form of debate more thoughts can be raised and exercised. The forum creates an atmosphere of positive thinking and can reveal insights unlikely to be achieved by individuals. It is, perhaps, the human analogy to the fault tree, each member being a branch? The forum should be reserved for the more important issues since it can be costly to operate and consume the time of assessors. It has been found to be of enormous value in exploring all the likely facets of a safety issue and helps to eventually create a corpus of agreement which can be adopted as the definitive view. This is the strategy to adopt when individual assessors face uncertainty.

4.1 Final Set of Conclusions

In conclusion:

(a) Check lists and proforma and guidelines for assessors are useful as aide memoires but following these documents religiously does not constitute an assessment.
(b) Assessors should adopt a disciplined approach to the process of analysis, adopting logical, systematic methods which give insight into subtle safety issues and the relative importance of hazards and their interdependence.

(c) There is an overriding need for the assessor to apply his own judgement as to the adequacy and veracity of the safety case.

(d) Sound judgement is based on effective training, qualification and experience which give confidence to the assessor.

(e) Never carry out an instantaneous assessment. Understanding of where the real issues lie can only be achieved through critical thought processes; these take time and require due reflection.

(f) Team discussions with a senior assessor as Chairman can be adopted in order to understand and resolve complex safety issues. Uncertainty can be managed by this strategy.

Throughout this paper we have concentrated on management issues related to the assessment process. We have illustrated some of the problems with general examples taken from experience. No doubt, the lessons learned seem obvious when recalled after the event. They were not obvious at the time and a drift into less tractable positions can insidiously occur. The skill is in recognising when the drift into a confrontation mode is occurring. Perhaps at quarter-term an overall review is called for and conducted by a more senior manager, such as the group leader. Such an analysis may reveal any deterioration in either the assessment process or the production of the safety case. Management reviews by both the operator/owner and the assessor at the more senior management level could take place to discuss and debate the progress on the safety campaign, as currently perceived. We have not in fact conducted such a strategy fully as yet but feel that it has genuine potential merit, particularly for the review of the more complicated installations that have a multitude of issues presenting potentially serious safety problems. We intend in future to consider adopting this latter recourse, for completeness.
ОРГАНИЗАЦИЯ НАДЗОРА ЗА БЕЗОПАСНОСТЬЮ НА ЭТАПАХ ПРОЕКТИРОВАНИЯ, ЭКСПЛУАТАЦИИ И МОДЕРНИЗАЦИИ ИССЛЕДОВАТЕЛЬСКИХ РЕАКТОРОВ, ЭКСПЛУАТИРУЮЩИХСЯ В СССР.

Никольский Р.В.

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ORGANIZATION OF SAFETY SURVEILLANCE DURING THE DESIGN,
OPERATION AND MODERNIZATION OF RESEARCH
REACTORS IN THE USSR

R.V. Nikol'skij
USSR State Committee for the Supervision
of Nuclear Power Safety
Moscow, USSR

ABSTRACT

The tasks and functions performed by the USSR State Committee for the Supervision of Nuclear Power Safety in the area of safety surveillance of research reactors are described. The characteristic features of the technical status of research reactors with regard to safety criteria are reviewed. Examples are given of typical equipment failures and malfunctions which have caused deviations from normal operating regimes during recent years. The results are presented of an analysis of the safe operating status of research reactors monitored by the USSR State Committee for the Supervision of Nuclear Power Safety, together with information on their status after the implementation of measures to improve their safety. Future trends are examined with regard to the improvement of standards, regulations and procedures for surveillance of research reactors currently in operation, undergoing upgrading, and under construction.
Организация надзора за безопасностью на этапах проектирования, эксплуатации и модернизации исследовательских реакторов, эксплуатирующихся в СССР.

Abstract - Аннотация

Изложены задачи и функции, выполняемые Госатомэнергонадзором СССР по надзору за безопасностью исследовательских реакторов (ИР). Рассмотрены характерные особенности технического состояния ИР с точки зрения безопасности. Представлены характерные нарушения и отказы оборудования, вызывавшие отклонения от режимов нормальной эксплуатации за последние годы. Изложены результаты анализа состояния безопасной эксплуатации ИР подконтрольных Госатомэнергонадзору СССР и состояние по выполнению мероприятий по повышению их безопасности. Рассмотрены перспективные направления совершенствования норм и правил и процедур надзора применительно к эксплуатирующимся, реконструируемым и вновь сооружаемым ИР.

Введение

Госатомэнергонадзор СССР осуществляет государственный надзор за безопасным ведением работ на ИР, направленный на предупреждение аварий, выявление и анализ причин нарушений и проведение профилактических мероприятий по повышению надежности и безопасности их работы. Задачи и функции надзора определяются "Положением о Государственном комитете СССР по надзору за безопасным ведением работ" [1].

Подразделениями Госатомэнергонадзора СССР осуществляется государственный надзор за соблюдением министерствами, ведомствами, предприятиями и должностными лицами правил, норм и инструкций по безопасности при сооружении, вводе в эксплуатацию, эксплуатации и снятии с эксплуатации ИР, а также при хранении и транспортировании ядерного топлива и радиоактивных отходов в пределах территории предприятий-владельцев ИР.

В соответствии с изменением Закона СССР "О Совете Министров СССР", принятого в июле 1989 года, образован новый "Государственный Комитет СССР по надзору за безопасным ведением работ в промышленности и атомной энергетике" [2]. В настоящее время пересматривается структура и положение об органах надзора в атомной энергетике (включая ИР). Основные задачи надзора при этом сохраняются. Предстоит совершенствовать функции надзорных органов, основываясь на "Законе об использовании атомной энергии", проект которого также разрабатывается в соответствии с рекомендациями МАГАТЭ [3].

В докладе рассматривается состояние безопасности и организации надзора за безопасностью на этапах проектирования, эксплуатации и модернизации ИР, характеризуется политика надзорных органов в области безопасности применительно к вновь сооружаемым и реконструируемым реакторам.
1. ХАРАКТЕРИСТИКА ЭКСПЛУАТИРУЮЩИХСЯ ИР, ПОДКОНТРОЛЬНЫХ ГОСАТОМЭНЕРГОНАДЗОРУ СССР.

Под контролем Госатомэнергонадзора СССР эксплуатируется ИР бассейнового типа [4]: 5 типа ВВР и 4 типа ИРТ. Сооружаются 3 реактора. Основные характеристики реакторов представлены в таблице 1.

<table>
<thead>
<tr>
<th>№ п/п</th>
<th>тип реактора, место расположения</th>
<th>год ввода в эксплуатацию</th>
<th>мощность Мвт</th>
<th>примечание</th>
</tr>
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<tr>
<td>1</td>
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<td>2</td>
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<td>1967</td>
<td>10</td>
<td>остановлен для выполнения мероприятий по повышению сейсмостойкости</td>
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<tr>
<td>3</td>
<td>ВВР-СМ г. Ташкент</td>
<td>1959</td>
<td>10</td>
<td></td>
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<tr>
<td>4</td>
<td>ВВР-М г. Гатчина</td>
<td>1959</td>
<td>18</td>
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<td>5</td>
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<td>1960</td>
<td>10</td>
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<td>1962</td>
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<tr>
<td>8</td>
<td>ИРТ-М г. Тбилиси</td>
<td>1962</td>
<td>5</td>
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</tr>
<tr>
<td>9</td>
<td>ИРТ-Т г. Томск</td>
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<td>Аркус-21 г. Душанбе</td>
<td>-</td>
<td>0,05</td>
<td>строящийся</td>
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</tbody>
</table>
Эксплуатирующиеся реакторы сооружались в 50-60-е годы и эксплуатируются 22-30 лет. Из всех эксплуатирующихся реакторов только реактор ИРТ-Т в г. Томске был подвергнут кординальной реконструкции с заменой всего основного оборудования. На других реакторах проводились лишь частичные модернизации, связанные с расширением их экспериментальных возможностей и заменой оборудования, выработавшего ресурс.

На подконтрольных Госатомэнергонадзору СССР исследовательских реакторах серьезных нарушений пределов и условий безопасности, ядерных и технических аварий не было. Происходили следующие наиболее характерные исходные события с точки зрения возможного нарушения безопасности реакторов (в порядке убывания частоты событий):
- нарушение электроснабжения,
- землетрясения силой 3-7 баллов,
- неисправность экспериментальных устройств,
- отказ КИП,
- ошибки персонала при техническом обслуживании оборудования,
- ошибки персонала при переключениях,
- разуплотнение трубопроводов и теплообменников 1-го контура.

Во всех случаях системы безопасности реакторов или персонал воспрепятствовали развитию событий и их последствий сверх установленных пределов. Анализ этих событий и их последствий показывает, что вероятность тяжелых последствий каждого отдельного события мала.

В качестве иллюстрации по характеру и количеству отказов оборудования реакторов, приведших к срабатыванию аварийной защиты, можно привести данные за 1988 год:
- неисправность электроснабжения - 30
- отказы в электрических цепях СУЗ - 5
- отказы в системах КИП - 8
- неисправности в экспериментальных устройствах - 4
- ошибки персонала - 5

В 1985 году по предписанию Госатомэнергонадзора СССР владельцами реакторов при помощи специализированных организаций осуществлялась программа работ по повышению безопасности и надежности реакторов, которая предусматривала:
- приведение СУЗ реакторов в соответствие с требованиями правил ядерной безопасности (ПЯБ), т.е. устранение выявленных надзорными органами нарушений правил;
- усовершенствование устройств контроля исправности аппаратуры КИП, входящей в систему аварийной защиты;
- введение ведомственной стандартизированной поверки приборов СУЗ и КИП, каналов контроля мощности и аварийной защиты;
- анализ надежности аппаратуры СУЗ и ответов СУЗ на возможные неисправности;
- проведение поверочных расчетов на прочность основных узлов реакторов и оборудования 1-го контура;
- уточнение сроков службы основного оборудования, программ периодических обследований состояния этого оборудования.
В период после событий на ЧАЭС Госатомэнергонадзор СССР приступил к анализу наиболее тяжелых последствий возможных аварий, связанных с отказами оборудования и ошибками персонала. Рассматривались, как наиболее серьезные, следующие события:

- незапланированное введение значительной реактивности при потере контроля и управления цепной реакцией в результате грубых ошибок персонала при производстве перегрузочных работ;
- нарушение теплоотвода в активной зоне реактора при полном прекращении подачи электроэнергии и прекращении принудительной циркуляции теплоносителя через активную зону;
- нарушение теплоотвода в активной зоне реактора с осушением активной зоны вследствии разрыва трубопроводов первого контура или горизонтальных экспериментальных каналов.

После анализа состояния безопасности ИР [5], основывавшегося на результатах комиссионных проверок, в 1986-1988 г.г. были реализованы следующие мероприятия;

- смонтированы системы сбора и возврата теплоносителя на случай аварии с течью бака реактора или разрывом горизонтально-го канала,
- пересмотрены планы аварийных мероприятий, инструкции по эксплуатации и инструкции по ликвидации последствий аварий, усилены требования к квалификации и дисциплине персонала,
- откорректированы ТОБ'ы.

На реакторах, выполнивших мероприятия по повышению безопасности, количество случаев нарушения режимов нормальной эксплуатации снижено.

Госатомэнергонадзору СССР подконтрольно строительство трех ИР: высокопоточного пучкового исследовательского реактора ПиК в ЛИЯФ АН СССР (г.Гатчина) [6] и двух малогабаритных растворных реакторов "Аргус-21" стационарной мощностью несколько десятков киловатт. [7]

Поскольку проектирование эксплуатирующихся и строящихся ИР осуществлялось до выпуска современной нормативной документации, предприятиями - владельцами reactors выполняются мероприятия по переработке и корректировке проектно-конструкторской документации.

Эксплуатация и сооружение ИР не выполнявших в должном объеме мероприятия по повышению безопасности приостанавливалась органами надзора.

Под контролем Госатомэнергонадзора СССР осуществляются работы по снятию с эксплуатации reactor ИРТ-М ИЯЭ АН БССР (г. Минск).
В Госатомэнергонадзоре СССР принята двухуровневая структура надзора, состоящая из центрального аппарата Комитета со специализированным отделом, регионального Управления округом и инспекции, представительства которой расположены на трех реакторах, представляющих наибольший интерес с точки зрения надзора.

Основной деятельностью специализированного отдела является техническое и методическое руководство в области ИР Управлением регионального органа и инспекции, контроль за их деятельностью. Кроме этого, специализированный отдел вырабатывает единую техническую политику в области безопасности ИР, осуществляет связь с министерствами и ведомствами-владельцами ИР, определяет надзорные процедуры, осуществляет разработку методической и прочей руководящей документации, регламентирующей надзорную деятельность в области ИР.

Основной деятельностью Управления округа и его инспекции является контрольно-профилактическая работа, т.е. работа направленная на предупреждение нарушений требований нормативно-технической документации по безопасности ИР, осуществляет связь с министерствами и ведомствами-владельцами ИР, определяет надзорные процедуры, осуществляет разработку методической и прочей руководящей документации, регламентирующей надзорную деятельность в области ИР.

Инспекция осуществляет надзор за безопасностью ИР путем регулярного, с периодом в 6 месяцев их посещения, а также оперативного контроля за деятельностью реакторов, на которых расположены представительства инспекции.

В зависимости от важности работ, проводимых на ИР, Комитетом и его органами выдаются разрешения на этапы: проектирования, изготовления, строительства, монтажа, пуско-наладочных работ, эксплуатации, ремонта, снятия с эксплуатации. Процедуры выдачи указанных разрешений определены руководящими документами Комитета.

Чтобы учесть в полной мере опыт выполнения мероприятий по повышению безопасности Госатомэнергонадзор СССР установил порядок согласования проектной документации для разрабатываемых и реконструируемых ИР, в соответствии с которыми министерства и ведомства должны представлять:

- материалы, обосновывающие выбор площадки строительства, согласованные с другими органами надзора, а также с местными органами власти;
- технические задания на разработку и реконструкцию ИР, содержащие основные требования нормативно-технической документации (НТД) по безопасности в атомной энергетике с учетом рекомендаций МАГАТЭ;
- технический проект ИР, согласованный с другими органами надзора.

Проекты реконструируемых реакторов должны учитывать требования вновь разработанных "Общих положений по безопасности исследовательских реакторов", которые планируется ввести в 1989 г.
Исходя из того, что подготовка и высокая квалификация персонала, занимающегося эксплуатацией ИР, является наиболее важным фактором обеспечения безопасности, Госатомэнергонадзор СССР осуществляет надзор за подготовкой персонала на предприятиях-владельцах ИР, а также на курсах повышения квалификации персонала.

Надзор осуществляется:
участием представителей Госатомэнергонадзора СССР в проверках знаний правил, норм и инструкций, действующих в области эксплуатации ИР, проводимых комиссиями предприятия и его подразделения-владельца ИР;
выборочной проверкой знаний персонала, проводимой инспекторским составом при обследовании ИР;
проверкой на подконтрольных предприятиях соблюдения порядка подготовки персонала, установленного правилами и руководящими документами Госатомэнергонадзора СССР.

В последнем случае контролируются:
наличие и содержание на предприятии организационно-распорядительной документации по вопросам подготовки, повышения квалификации, допуска соответствующих категорий персонала к работе, аттестации и проверке знаний;
наличие учебных планов, программ подготовки, квалифицированных преподавателей и инструкторов, технических и наглядных средств обучения;
порядок проведения стажировок, дублирования и допуска к работе.
Программа подготовки персонала на специальных курсах повышения квалификации согласовывается Госатомэнергонадзором СССР и в дальнейшем им контролируется.

3. ПЕРСПЕКТИВНЫЕ НАПРАВЛЕНИЯ НАДЗОРНОЙ ДЕЯТЕЛЬНОСТИ.

3.1. Одной из основных проблем, помимо проблем, связанных с необходимостью технического совершенствования ИР, Госатомэнергонадзор СССР считает несовершенство организационных структур организаций и предприятий, осуществляющих ведение работ на ИР, владельцы которых находятся в девяти ведомствах, которые не имеют возможности в должной мере самостоятельно без помощи других ведомств обеспечить работы по всему циклу сооружения, эксплуатации и выводу из эксплуатации ИР.
Вопрос о совершенствовании организационных структур, включающих ИР, в настоящее время рассматривается ведомствами-владельцами и правительственные органами.

3.2. В 1989 году на основе анализа вновь разработанных ТОБ'ов и отступлений от современных требований НТД ведомства и предприятие-владельцы совместно со специализированными организациями согласуют с Госатомэнергонадзором СССР конкретные сроки эксплуатации каждого ИР, ресурс основного оборудования которых близок к исчерпанию. В первую очередь это касается реакторов, установленные сроки эксплуатации которых истекают в 1990-92 г.г.
3.3. В связи с вводом с 01.07.88г. "Норм проектирования сейсмостойких АС" и распространение их на ИР, планируется проведение сейсмического и тектонического исследования площадок всех эксплуатирующихся ИР с целью выполнения поверочных расчетов на прочность оборудования систем безопасности, а также строительных конструкций зданий. Госатомэнергонадзор СССР выказал мнение в адрес предприятий-владельцев реакторов о нецелесообразности проведения принципиальной реконструкции действующих и создания новых реакторов, расположенных в черте крупных городов, а также в районах повышенной сейсмической активности г.Алма-Ата и г.Ташкента.

3.4. Значительный технический опыт по выводу из эксплуатации ИР в настоящее время СССР не накоплен. Разработана процедура получения разрешения на демонтаж оборудования реактора ИРТ-М г. Минск, подлежащего реконструкции.

Документами, на основании которых выдается такое разрешение являются:

программа работ по консервации оборудования не подлежащего реконструкции и обеспечения безопасности транспортировки и хранения топлива;

проект демонтажа оборудования подлежащего реконструкции, выполненный специализированной организацией;

документация по организации захоронения радиоактивного оборудования.

3.5. В настоящее время Госатомэнергонадзор СССР пользуется в своей работе лишь двумя нормативными документами, посвященными ядерной и технической безопасности ИР, это "Правила ядерной безопасности исследовательских реакторов" и "Правила устройства и безопасной эксплуатации оборудования ядерных энергостанций, опытных и исследовательских ядерных реакторов и установок". Указанные правила разработаны в 1975 и 1972 г. г., содержат в себе требования к конструкции реакторов в соответствующей области безопасности и организации работ по эксплуатации ИР. В 1987 году рядом министерств и ведомств СССР, заинтересованных в выполнении программы развития атомной энергетики был утвержден "Сводный перечень и план разработки правил и норм в области атомной энергетики" (СППНАЭ-87), содержащий в себе и план разработки ряда новых и пересмотра действующих правил в области безопасности ИР. На сегодня разработана 3-я редакция "Общих положений обеспечения безопасности исследовательских реакторов при проектировании, сооружении и эксплуатации" (ОПБ ИР), 2-я редакция "Правил ядерной безопасности исследовательских реакторов" и окончательная редакция "Правил устройства и безопасной эксплуатации оборудования и трубопроводов атомных энергетических установок" (ПГА-01-85). В этих документах учтен многолетний опыт эксплуатации и надзора за эксплуатацией ИР в СССР, максимально учтен международный опыт.
В процессе работы над созданием трех вышеуказанных документов стало ясно, что нельзя обойтись лишь ими: обеспечение безопасности ИР должно охватывать весь его жизненный цикл — от разработки проекта до снятия с эксплуатации, поэтому сегодня в Госатомнадзоре СССР рассматривается проект нового, значительно более расширенного "Перечня норм и правил по безопасности исследовательских реакторов". Предполагаемый период разработки документов перечня — 3—5 лет.

ЛИТЕРАТУРА

ABSTRACT

Objectives and functions in research reactors (RR) safety supervision which being carried out by the USSR Gosatomenergonadzor have been presented here. Characteristic peculiarities of RR technical condition from the point of view of safety have been considered. Characteristic violations and equipment failures causing deviations from normal operation regimes during the period of fast years have been presented. The results of safe operation condition analysis of RR which are under Gosatomenergonadzor control are set out in this paper. Perspective directions in supervision norms, rules and procedures improvement as applied to RR which are being operated, modernized and designed have been considered.

INTRODUCTION

Gosatomenergonadzor of the USSR carries out the state supervision on safe work fulfilment on RR, oriented on accidents prevention, failures causes determination and analyses and preventive measures on the RR work reliability and safety enhance realization. Objectives and functions of supervision are determined in "Statement on the State Comettee of the USSR on safe fulfilment of work"[1].

The USSR Gosatomenergonadzor departments carry out the supervision on the observance by ministries, agencies, enterprises and functionaries of safety rules, norms and instructions during RR design, commissioning, operation and decommissioning as well as at storage and transportation of nuclear fuel and radioactive wastes within the territories of enterprises - owners of RR.

In accordance with the change of the USSR Law "About the Council of the USSR Ministers" adopted in July 1989, the new "State Committee of the USSR on supervision of nuclear power and industry safety" was organise [2]. Nowadays the structure and statement of supervisory bodies in nuclear power (including RR) are revising. The main objectives supervision with this will be preserved. We are going to improve supervisory bodies functions on the base of the "Low of Nuclear power use", draft version of which is being developed in accordance with IAEA recommendations [3].

This paper considers safety condition and supervision organization during design, operation and modernization of RR, characterizes the policy of supervisory bodies in the area of safety as applied to the reactors which are under design and are modernized now.
1. CHARACTERIZATION OF OPERATING RR WHICH ARE UNDER CONTROL OF THE USSR GOSATOMENERGONADZOR. Gosatomenergonadzor of the USSR carries out control over the following pool type RR [4]: 5 of the "BBP" type and 5 of "HPT" type. 3 reactor are under construction now. Main features of the reactors are presented in Table 1.

<table>
<thead>
<tr>
<th>item</th>
<th>reactor type, location</th>
<th>year of commission</th>
<th>capacity Mw</th>
<th>notes</th>
</tr>
</thead>
<tbody>
<tr>
<td>1</td>
<td>BBP-II Obninsk</td>
<td>1964</td>
<td>16</td>
<td></td>
</tr>
<tr>
<td>2</td>
<td>BBP-K Alma-Ata</td>
<td>1967</td>
<td>10</td>
<td>Shut down for fulfilment of measures on seismik improvement</td>
</tr>
<tr>
<td>3</td>
<td>BBP-CM Tashkent</td>
<td>1959</td>
<td>10</td>
<td></td>
</tr>
<tr>
<td>4</td>
<td>BBP-M Gatchina</td>
<td>1959</td>
<td>18</td>
<td></td>
</tr>
<tr>
<td>5</td>
<td>BBP-M Kiev</td>
<td>1960</td>
<td>10</td>
<td></td>
</tr>
<tr>
<td>6</td>
<td>ИРТ Moscow</td>
<td>1967</td>
<td>2,5</td>
<td></td>
</tr>
<tr>
<td>7</td>
<td>ИРТ-M Minsk</td>
<td>1962</td>
<td>5</td>
<td>Shut-down for modernization</td>
</tr>
<tr>
<td>8</td>
<td>ИРТ-M Tbilisi</td>
<td>1959</td>
<td>5</td>
<td>Shut-down for fulfilment of measures on safety improvement</td>
</tr>
<tr>
<td>9</td>
<td>ИРТ-T Tomsk</td>
<td>1967</td>
<td>6</td>
<td>modernized in 1984</td>
</tr>
<tr>
<td>10</td>
<td>ИРТ Riga</td>
<td>1961</td>
<td>5</td>
<td></td>
</tr>
<tr>
<td>11</td>
<td>ПИК Gatchina</td>
<td>-</td>
<td>100</td>
<td>building up</td>
</tr>
<tr>
<td>12</td>
<td>Argus-21 Ufa</td>
<td>-</td>
<td>0,05</td>
<td>building up</td>
</tr>
<tr>
<td>13</td>
<td>Argus-21 Dushanbe</td>
<td>-</td>
<td>0,05</td>
<td>building up</td>
</tr>
</tbody>
</table>
Operating reactor were designed in 1950-1960 and have been operated for 22-30 years already. From all operating reactor only HPT-T in Tomsk was subjected to cardinal modernization with major equipment replacememt. Other reactors had only partial modernizations connected with their experimental abilities expansion and replacement of the equipment having worn out its resource.

There were no serious safety limits and conditions violations, nuclear and technical accidents on RR which are under control of the USSR Gosatomenergonadzor. The following most characteristic from the point of view of possible reactor safety violation initial events took place (in the order of events frequency decrease):
- disruption of power supply,
- earthquakes of magnitude 3-7,
- malfunction of experimental devices,
- instrumentation failure,
- personnel errors during equipment maintenance,
- personnel errors during switches,
- leaks in primary circuit pipelines and heatexchangers,

In all cases reactor safety systems or personnel prevented development of the events and their consequences above prescribed limits. Analyses of these events and their consequences shows that probability of every separate event severe consequences is very low.

Data of 1988 may be given as illustration of a character and quantity of reactor equipment failures led to emergency protection operation:
- power supply malfunction - 30
- failures in electrical circuits of reactor control and protection systems - 5
- failures in control devices - 8
- malfunctions in experimental equipments - 4
- personnel errors - 5

In 1985 according to the USSR Gosatomenergonadzor proposal reactor owners with the help of specialized organizations fulfilled a programme on reactor safety and reliability increase, which included:
- adjustment of reactor control and protection system in conformance with requirements of nuclear safety rules("ПБЯ"), i.e. elimination of rules violations determined by regulatory body;
- introduction of agencyis standartized calibration of control and protection system and instrumentation devices, power control channels and emergency protection;
- improvement of instrumentation serviceability control devices which are included into emergency protection system;
- analysis of control and protection system devices reliability and control and protection system responses on possible malfunctions;
- fulfilment of colibration calculations on strength of primary circuit main reactor components and equipment;
- refinement of main equipment service life, of this equipment periodical examination programmes.
After Chernobyl accident Gosatomenergonadzor of the USSR began analysing the most severe consequences of possible accidents connected with equipment failures and personnel errors. The following events as the most serious were considered:

- unplanned introduction of significant reactivity at the loss of chain reaction control in the result of gross personnel errors during reloading activities;
- malfunction of heat removal in the reactor core with full loss of power supply and termination of forced coolant circulation through the core;
- malfunction of heat removal in the reactor core with loss of coolant accident as a result of primary circuit pipelines or horizontal experimental channel rupture.

After the analysis of reasearch reactors safety condition [5], based on the results of commissions checks, the following measures were realized in 1986-1988:

- systems of coolant gathering and return in the case of reactor tank leak or horizontal channel rupture were mounted;
- emergency measures plans, operation procedures and instructions on accident consequences elimination were revised, requirements to personnel qualification and discipline were strengthened;
- safety reports were corrected.

On reactors where safety improvement measures were fulfilled a number of normal operation regimes violations was decreased.

Construction of three research reactors: one high flux beam reactor "ПИМ" in Leningrad Nuclear Physics institute, town Gatchina [6], and two small-sized homogeneous reactors "АРГУС-21" with stationary capacity of several dozen of kw [7].

Since the design of research reactors which are operated under construction now were done before issuing of modern normative documentation, enterprises reactor owners fulfil now measures on revision and correction of design documentation.

Operation and building up RR which haven't fulfilled measures on safety increase in the needed volume are terminated by regulatory bodies.

Gosatomenergonadzor of the USSR carries out control on the decommissioning of reactor ИРТ-М of Nuclear energy institute within the Academy of Sciences of Byelarusian Soviet Socialist Republic (town of Minsk).
2. ORGANIZATION AND METHODOLOGY OF SUPERVISION ON SAFE WORK FULFILMENT ON RESEARCH REACTORS. Gosatomenergonadzor of the USSR adopted two-level structure of supervision consisting of central body with specialized department, regional office administration and inspections representation of which are located on three reactors which are of interest to supervision.

Main activity of specialized department is in technical management of regional office and inspections in the area of research reactors, control on their activity. Besides, the specialized department elaborates common technical policy in the area of research reactors safety, carries out connection with ministries and agencies owners of research reactors, determines supervisory procedures, carries out elaboration of methodical and other leading documents of Gosatomenergonadzor on the enterprises-owners of research reactor which are under Gosatomenergonadzor control.

Inspection fulfills supervision on research reactors safety by routine - with six-month interval between visits - and in-service inspection of the reactors where inspection representations are situated.

Depending on the importance of work being carried out on the research reactor the Committee and its regional offices issue permissions on the stages of design, manufacturing, construction, mounting, commissioning, operation, maintenance, decommissioning. Procedures of the mentioned permissions issue are determined by the leading documents of the Committee.

In order to take into account to full extent the experience on safety increase measures fulfilment Gosatomenergonadzor of the USSR established order of approval of design documentation for research reactors which are developed and modernized, in accordance with which ministries and agencies should present:

- materials which ground site choice for construction approved by other supervisory bodies and local authorities;
- technical assignment on development and modernization of a research reactor including main requirements of nuclear power safety normative documentation with the account of IAEA recommendations;
- technical design of a research reactor approved by other supervisory bodies.

Designs of the reactors which are to be modernized should take into account the requirements of newly developed "General statements on research reactors safety", which are planned to be introduced in 1989.

Hence training and high qualification of the personnel who deals with research reactors operation is the most important factor of safety assurance Gosatomenergonadzor of the USSR carries out supervision on personnel training on enterprises - owners of research reactor and also on the courses of personnel qualification level rasing.
The supervision is fulfilled by:

- Gosatomenergonadzor representatives participation in the check of knowledge of rules, norms and instructions currently in force in research reactors operation carried out by the commissions of enterprises - owners of research reactors;
- selective check of personnel knowledge conducted by inspector staff during research reactors examination;
- check on the enterprises under control of personnel training order observance prescribed by rules and Gosatomenergonadzor leading documents.

In the last case it is controlled:

- presence and keeping in the enterprises of administrative documentation on issues of personnel training, qualification level raising, permission for corresponding categories of personnel to work, certification and knowledge check;
- presence of training plans, programmes, qualified teachers and instructors, technical and visual aide of learning;
- order of probation period, "dubbling" and permission to work realization.

Programme of personnel training on special courses of qualification level raising is approved by Gosatomenergonadzor and controlled by it.

3. PERSPECTIVE DIRECTIONS IN SUPERVISORY ACTIVITY.

3.1 Gosatomenergonadzor of the USSR considers that one of the main problems connected with technical improvement of research reactors is imperfection of organizational structures of agencies and enterprises which carry out their work on research reactors owners of which can't independently, prairie activity through the whole cycle of construction, operation and decommissioning of research reactors without other agencies help.

Issues about perfection of organizational structures including research reactors are now considered by agencies-owners and governmental bodies.

3.2 In 1989 on the base of newly developed safety reports analysis and deviations come to an agreement with Gosatomenergonadzor of the USSR about the concrete deadlines of every research reactor operations, service life of main equipment of which is close to be worn out. In first place it concerns the reactors set operation deadlines of which and in 1990-1992.

3.3 In connection with the introduction in 01.07.88 of "Design norms of seismic prove atomic power plant" and spread of it on research reactors it is planned the fulfilment of seismic and tectonic investigation of all operating research reactors sites with the aim of calibration calculations fulfilment on safety systems equipment strength as well as of building constructions. Gosatomenergonadzor of the USSR adressed its opinion to enterprises-reactor owners about inexpediency operating reactors principal modernization carrying out and new reactors creation in the areas of large cities as well as in the areas of heightened seismic activity town of Alma-Ata and Tashkent.
3.4. Nowadays in the USSR there are no significant experience in research reactors decommissioning. It has been developed the procedure of permission reception for reactor ВПТ-М equipment dismantling (town of Minsk) which is going to be modernized.

Document on the base of which such permission is given are:
- work programme on conservation of the equipment which is not subjected to modernization and safety assurance programme on fuel transportation and storage;
- project of equipment dismantling subjected to modernization elaborated by specialized organization;
- documentation on radioactive equipment burial organization.

3.5. Nowadays Gosatomenergona
dzor of the USSR uses in the work only two normative documents devoted to nuclear and technical safety of research reactors. These are "Nuclear safety rules for research reactors" and "Rules of nuclear power plant equipment arrangement and safe operation". Mentioned rules were worked out in 1975 and 1972 and contain in themselves requirements to reactor design in the corresponding area of safety and to activity organization in research reactor operation. In 1987 a number of ministries and agencies of the USSR interested in the fulfilment of nuclear power development programme approved "Summary list and plan of norms and rules development in the area of nuclear power" (СПННАЭ-87), which contains in itself both a plan of new rules development and current rules revision in the area of research reactors safety. For today the third version of "General statements of research reactors safety assurance during designing, construction and operation" (ОПЕИР), the second version of "Research reactors nuclear safety rules" and final version of the "Rules of arrangement and safe operation of nuclear power facilities equipment and pipelines" (ПГА-01-89) have been developed. These documents take into account the experience of many years operation and supervision on research reactors operation in the USSR and in other countries.

In the process of the three mentioned above documents development it became obvious that it's impossible to do with them only - research reactors safety assurance should cover its whole life cycle - beginning with design development and ending with decommissioning. That's why Gosatomenergona
dzor of the USSR is considered nowadays the project of new, significantly broadened "List of norms and rules on research reactors safety". Supposed period of the list documents development - 3-5 years.
REFERENCES

2. Government formation is going to an end, Izvestnya, 1989, July, 14.
5. N. Archangelskiy, V. Dikarev, P. Yegorenkov, Ye. Ryazantzev, Research reactors safety increase, Atomnaya energiya, 64, issue 5, 1988.
DEVELOPMENT OF SMALL REACTOR SAFETY CRITERIA IN CANADA

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DEVELOPMENT OF SMALL REACTOR SAFETY CRITERIA IN CANADA

ABSTRACT

A number of new small reactor designs have been proposed in Canada over the last several years and some have reached the stage where licensing discussions have been initiated with the Atomic Energy Control Board (AECB). An inter-organizational Small Reactor Criteria (SRC) working group was formed in 1988 to propose safety and licensing criteria for these small reactors. Two levels of criteria are proposed. The first level forms a safety philosophy and the second is a set of criteria for specific reactor applications. The safety philosophy consists of three basic safety objectives together with evaluation criteria, and fourteen fundamental principles measured by specific criteria, which must be implemented to meet the safety objectives. Two of the fourteen principles are prime - defence in depth and safety culture; the other twelve principles can be seen as deriving from them. A benefit of this approach is that the concepts of defence in depth and safety culture become well-defined. The objectives and principles are presented in the paper and their criteria are summarized. The second level of criteria, under development, will form a safety application set and will provide small reactor criteria in a number of general areas, such as regulatory process and safety assessment, as well as for specific reactor life-cycle activities, from siting through to decommissioning. The criteria are largely deterministic. However, the frequencies and consequences of postulated accidents are assessed against numerical criteria to assist in judging the acceptability of plant design, operation, and proposed siting. All criteria proposed are designed to be testable in some evidentiary fashion, readily enabling an assessment of compliance for a given proposal.

RÉSUMÉ

Quelques nouveaux projets de petits réacteurs ont vu le jour au Canada depuis plusieurs années et certains d’entre eux en sont maintenant au point où les discussions ont été amorcées avec la Commission de contrôle de l’énergie atomique (CCEA) en vue de leur autorisation. Un groupe de travail sur les critères des petits réacteurs, regroupant des représentants de plusieurs organismes, a été créé en 1988 pour recommander des critères de sûreté et d’autorisation pour ces petits réacteurs. Deux niveaux de critères sont proposés. Le premier niveau établit les prémisses de sûreté, tandis que le second comprend une série de critères pour certaines applications particulières des réacteurs. Les prémisses de sûreté comprennent trois objectifs de sûreté fondamentaux et leurs critères d’évaluation, ainsi que quatorze principes fondamentaux mesurés par des critères précis, qui doivent tous être mis en vigueur pour atteindre les objectifs de sûreté. Deux des quatorze principes, la défense intégrée et le climat de sûreté, sont prépondérants; les douze autres principes en découlent plus ou moins. Cette approche a comme avantage de bien définir les notions de défense intégrée et de climat de sûreté. Le présent rapport traite de ces objectifs et de ces principes, et en résume les critères. Le
second niveau de critères, qui en est toujours au stade l'élaboration,
constituera une série d'applications sûres et fournira des critères pour
les petits réacteurs dans certains domaines généraux, comme le processus
réglementaire et les évaluations de sûreté, ainsi que pour les activités
précises tout au long de la vie du réacteur, de la sélection du site au
déclassement. Les critères sont surtout de nature déterministe.
Toutefois, la fréquence et les conséquences d'accidents postulés sont
évaluées par rapport à des critères numériques pour aider à juger de
l'acceptabilité de la conception de la centrale, de son exploitation et de
son site proposé. Tous les critères proposés sont conçus pour être
évalués de manière concluante, quelle qu'elle soit, permettant ainsi
d'évaluer facilement de la conformité d'un projet donné.

1. INTRODUCTION

Over the past several years, there has been renewed interest in small
reactors worldwide. A number of new small reactor designs have been
proposed in Canada and some have reached the stage where licensing
discussions have been initiated: [1-4]

- The SLOWPOKE Demonstration Reactor, a 2 MWt demonstration and test of
  a commercial district heating reactor, now operating at Whiteshell
  Nuclear Research Establishment in Pinawa, Manitoba.

- The SLOWPOKE Energy Systems 10 MWt (SES-10) commercial heating
  reactor. An application at the University of Saskatoon in
  Saskatchewan has been identified, and the reactor is now the subject
  of a joint feasibility study by the University and Atomic Energy of
  Canada Limited (AECL).

- The MAPLE-X10 isotope production reactor, a 10 MWt high-flux,
  multipurpose research reactor to be constructed at the Chalk River
  Nuclear Laboratories in Ontario.

- The Autonomous Marine Power Source (AMPS), a 1.5 MWt nuclear electric
  power plant for submarine applications, being marketed by Energy
  Conversion Systems Inc.

Regulatory criteria are gradually evolving for the specific reactors
undergoing licensing, but there is need for a unified approach to guide
designers, operators and regulators.

In response, an inter-organizational Small Reactor Criteria (SRC) working
group was formed in 1988 to propose criteria for small reactors. The group
comprises the four authors of this paper, from the principal organizations
in Canada responsible for reactor licensing or involved in development or
operation of small reactors. The criteria being developed are not
regulatory requirements but are meant primarily for consideration by the
sponsoring organizations. It is expected that these criteria will be
complemented by subsequent developments by various organizations in Canada, and by the IAEA efforts internationally.

This paper summarizes work done to date to develop a top-level set of criteria which form a safety philosophy and serve as a framework for more detailed developments. The work is now being presented in a public forum for two reasons:

- to obtain a wide peer review on the overall approach and on the basic safety philosophy; and
- to generate input for subsequent work on key issues.

A complete and detailed report, now in draft form, will be published in 1990.

2. **CRITERIA DEVELOPMENT**

These criteria are intended for land-based, pool-type, thermal reactors with coolant and pool temperatures not significantly higher than the boiling point at atmospheric pressure, having a maximum operating power level up to several tens of MW, and for which significant releases of radioactive fission products are possible. The intent is that the criteria could be extended, with caution, to reactors of different type and power levels. Some reactor types, however, may be able to achieve the intent of the safety criteria by special inherent features, making some of the criteria not fully applicable.

The process of developing these criteria has been described in detail in Reference [5]. As noted there, extensive use is being made of existing international and domestic work, for example references [6,7], and the results are expressed in a framework which is consistent with the approach of the International Atomic Energy Agency (IAEA). Considering the IAEA practice of defining four levels of documents, namely; "Safety Fundamentals", "Safety Standards", "Safety Guides", and "Safety Practices", the SRC group has completed a draft report, summarized in this paper, which is the equivalent of Safety Fundamentals; work is currently underway on developing more detailed Safety Standards.

We are pleased to acknowledge our debt to the work done by the IAEA, in particular the International Nuclear Safety Advisory Group (INSAG) which produced INSAG-3 [8], and various consultant/specialist groups which produced Safety Series 35 [9] for research reactors and the documents produced by the Nuclear Safety Standards program [10-14] for power reactors, all of which provided useful guidance. The INSAG report, INSAG-3, provides a logical framework for understanding the underlying objectives and principles of nuclear safety. The SRC work is based on the same general structure but has been modified to clarify and adapt the concepts for application to small reactors. New topics have been addressed and a coherent, hierarchical structure developed, with testable criteria.
The work is comprehensive, covering all factors of major importance to safety, for all phases of a reactor life-cycle.

The criteria are being developed as general, top tier requirements. Two levels of criteria are proposed, as shown in Figure 1; the first level forms a safety philosophy and the second a set of criteria for specific reactor applications. The safety philosophy gathers together primary concepts, consisting of a basic safety objective, three supporting objectives together with evaluation criteria, and fourteen fundamental principles measured by specific criteria; all elements need to be implemented to meet the safety objectives. The second level of criteria form a safety application set covering a number of general areas, such as regulatory process and safety assessment, and specific reactor life-cycle activities, from siting through decommissioning.

The criteria are, in the main, deterministic. However, probabilistic criteria are also proposed as targets. Criteria are designed to be testable in some evidentiary fashion, readily enabling an assessment of compliance for a given proposal.

3. **SAFETY OBJECTIVES**

The first level of the safety philosophy consists of a basic safety objective and three supporting objectives. These objectives are stated below, together with a summary discussion of the evaluation criteria. The relationship of objectives to the safety philosophy is shown in Figure 2.
3.1 **Basic Safety Objective**

The basic safety objective is to protect individuals, society and the environment by establishing and maintaining in small reactor facilities an effective defence against radiological hazard.

![Diagram showing Basic Safety Objective and Supporting Objectives](attachment:image.png)

3.2 **Radiation Protection Objective**

Radiation exposure within the facility and that due to any release of radioactive material from the facility shall be kept as low as reasonably achievable and below prescribed limits in all operational states; radiation exposures due to accidents shall be mitigated.

Radiation protection shall be provided in small reactor facilities for all operational states, including anticipated operational occurrences, and for accident conditions.
Radiation protection standards applicable to controlled circumstances have been developed by the International Commission on Radiological Protection (ICRP) to prevent harmful effects of ionizing radiation by keeping exposures sufficiently low that non-stochastic effects are precluded and the probability of stochastic effects is limited to prudent levels. Compliance with the Atomic Energy Control Regulations and the ICRP recommendations ensures appropriate radiation protection and provides a system of dose limitation which includes three basic and interrelated principles:

1. Justification of a practice.
2. Limitation of effective dose.
3. Optimization of radiation dose (ALARA).

The principle of justification of a practice in its broadest sense can be satisfied by the requirement of the proponent to justify the facility through the environmental review and assessment process.

The principle of dose limitation is embodied in the Atomic Energy Control Regulations [15] for limits on effective dose (effective dose equivalent) as given in Table 1. Inherent in these limits is the requirement to do a pathways analysis considering radiation dose due to ingestion, inhalation, and direct radiation.

**TABLE 1: Maximum Permissible Limits for Effective Dose for Operational States.**

<table>
<thead>
<tr>
<th>Dose Type</th>
<th>Atomic Radiation Workers</th>
<th>Others</th>
</tr>
</thead>
<tbody>
<tr>
<td>Whole body, gonads, bone marrow</td>
<td>30</td>
<td>5</td>
</tr>
<tr>
<td>Bone, skin, thyroid</td>
<td>150</td>
<td>30^1</td>
</tr>
<tr>
<td>Any tissue of hands, forearms, feet, and ankles</td>
<td>380</td>
<td>75</td>
</tr>
<tr>
<td>Lungs and other single organs or tissues</td>
<td>80</td>
<td>15</td>
</tr>
</tbody>
</table>

1. The dose to the thyroid of a person under the age of 16 years shall not exceed 15 mSv/year.
For most applications an achievement of one percent of the AECB regulatory limits demonstrates adequate optimization of radiation doses. Values above this level require analysis to show that they cannot be reduced at reasonable cost. Where dose reduction substantially below the one percent value can be readily identified and achieved at reasonable cost, this should be instituted.

The standards of dose limitation have been developed for conditions of exposure over which control is possible. Accident conditions may result in exposures which might not be controllable. For these conditions, alternative strategies are evaluated. Reactor safety features are provided to reduce the probabilities and releases associated with such events, and to provide intervention following an accident. Quantitative criteria are being developed to specify the level of protection for people, including dose levels for notification, sheltering and evacuation.

3.3 Risk Limitation Objective

The frequencies and radiological consequences of accidents in small reactor facilities shall be within acceptable bounds.

The purpose of this objective is to provide reasonable assurances that the public will be adequately protected against accidents at small reactor facilities. This is accomplished by ensuring that for all accidents taken into account in the design of the facility, radiological consequences, if any, would be minor; and that the likelihood of severe accidents with serious radiological consequences is very small.

To show that this objective is satisfied, criteria are presented which provide a quantitative framework in which to assess the frequencies and consequences of postulated accidents. The criteria serve both as design tools and as acceptance requirements for siting, design and operation. It is recognized that acceptability is not just a matter of satisfying numerical criteria; other principles in our framework also address how additional confidence can be achieved.

The frequencies and consequences of credible accidents are first evaluated during the design of a small reactor, and then compared to numerical acceptance values. The criteria are split into two portions: individual dose criteria, and collective dose criteria, as given in Tables 2 and 3. Each set of criteria is defined by three frequency ranges, and for each range, there is a dose band spanning about a factor of ten in dose. Predicted doses below the band are normally acceptable, while those above the band are not. Predicted doses within the band require justification as to why they cannot be reduced.
The three frequency ranges of Table 2 for individual risk limitation correspond broadly to:

- occurrences expected only a few times in the lifetime of the reactor;
- accidents terminated without core damage; and
- accidents which may involve fuel damage but are mitigated by the pool and buildings.

### TABLE 2: Dose to Most Exposed Individual.

<table>
<thead>
<tr>
<th>Frequency Range</th>
<th>Dose Band</th>
<th>Example</th>
</tr>
</thead>
<tbody>
<tr>
<td>$3 \times 10^{-1}$ to $3 \times 10^{-2}$/year</td>
<td>0.1 mSv to 0.5 mSv</td>
<td>Anticipated operational occurrence</td>
</tr>
<tr>
<td>$3 \times 10^{-2}$ to $10^{-4}$/year</td>
<td>0.5 mSv to 5.0 mSv</td>
<td>Accident terminated by safety system</td>
</tr>
<tr>
<td>$10^{-4}$ to $10^{-6}$/year</td>
<td>5 mSv to 100 mSv</td>
<td>Accident mitigated by pool/building</td>
</tr>
</tbody>
</table>

### TABLE 3: Collective Dose Criteria.

<table>
<thead>
<tr>
<th>Frequency Range</th>
<th>Dose Band</th>
</tr>
</thead>
<tbody>
<tr>
<td>$3 \times 10^{-1}$ to $3 \times 10^{-2}$/year</td>
<td>0.1 to 1.0 person-Sv</td>
</tr>
<tr>
<td>$3 \times 10^{-2}$ to $10^{-4}$/year</td>
<td>1.0 to 10 person-Sv</td>
</tr>
<tr>
<td>$10^{-4}$ to $10^{-6}$/year</td>
<td>10 to 100 person-Sv</td>
</tr>
</tbody>
</table>

In the first frequency range, the dose band is 1/50 to 1/10 of the legal limit for normal operation of 5 mSv. The desired result is that such occurrences have negligible impact on the surrounding community. Accidents terminated successfully by safety systems are an example of the second category; the dose band is between 1/10 of the legal limit for normal operation, and the limit. The dose band for the third category reflects a desire to push core damage accidents to less than once in 10,000 reactor
years. Doses in this band would not cause significant radiological harm to an individual; and lie between 1/50 and 2/5 of the power reactor dual failure\(^2\) dose limit used in Canada, 250 mSv.

It is implied by these criteria that severe accidents be shown to have a frequency less than \(10^{-6}\)/year, and if so then they need not affect the design. This recognizes that the disaster potential for small reactors is inherently limited by the fission product inventory [16], and therefore the effort spent in investigating very low-frequency accidents is not productive. Such accidents are addressed by having in place appropriate mitigation measures, such as on-site and, if necessary, off-site emergency plans. However, a brief investigation of the sensitivity of accident consequences to frequencies down to \(10^{-7}\) per year can be useful in assessing reactor safety, by determining if there is a sudden increase in consequences just below the cutoff frequency (a "cliff-edge").

The criteria for collective dose to the public surrounding the facility are shown in Table 3. These criteria are related to absolute potential harm, not potential harm relative to other sources of harm in the same population (such as latent fatalities unrelated to the reactor). The reason is that relative measures may be independent of population density and thus not valid criteria; for example, for a uniform nearby population growth, the numbers of latent fatalities both from the reactor and from other causes grow approximately in proportion. The collective dose limits chosen are about one percent of existing Canadian limits for power reactors.

### 3.4 Environmental Protection Objective

*There shall be no significant detrimental effects on the environment for all reactor operational states and those accidents taken into account in the reactor design; the impact resulting from accidents beyond the design basis shall be mitigated to the extent practicable.*

The purpose of this objective is to provide guidance in determining an acceptable level of environmental protection for a particular small reactor facility design at a specific site in Canada.

For normal operational states, releases from small reactor facilities would be very low and have no detrimental impact. Appropriate criteria under accident conditions present a greater challenge since little information

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\(^2\) The term "dual failure" as used in the Canadian context, is any combination of a serious process failure coupled with the assumed unavailability of any one safety system. A "serious process failure" is the failure of a process system or equipment that, in the absence of safety system action, could lead to fuel failure or the release of radioactive material to the environment. An example of a dual failure in a power reactor would be a loss-of-coolant accident coupled with unavailability of the emergency core cooling system.
has been published on the subject. It is difficult to develop absolute environmental protection criteria that are independent of human values and perceptions. Consequently, a central concept in development of these criteria is the assessment of the potential use of the environment by man, whether for habitation, recreation, economic use, or production of food. People are generally considered to be sensitive indicators of detrimental radiological effects. Thus one valid type of measure, which can be derived from radiation protection requirements, assumes that people would occupy and use the land in question. It can then be argued that the environment is adequately protected if people are protected.

Some of the concepts under study are outlined below; quantitative requirements are under development.

(1) Environmental impact during normal operation and anticipated operational occurrences shall be negligible. Releases to the ecosystem should be so low that no detrimental effects could be directly observed and no control or intervention action would be warranted.

(2) Environmental effects shall be minor for those accidents taken into account in the reactor design. There shall be little or no ground or aquifer contamination, no economic detriment or denial of land use beyond the facility exclusion zone, and no requirement for restrictions on consumption due to potential radionuclide uptake in the food chain.

(3) The likelihood of severe accidents with potential for more significant consequences to the environment shall be extremely small; the impact resulting from such accidents beyond the design basis shall be mitigated to the extent practicable. Although protection systems are provided to reduce the likelihood of core damage, a combination of other measures shall be available to limit the release of radioactive material and reduce their impact, including intervention measures to minimize transfer to the environment.

4. FUNDAMENTAL PRINCIPLES

The second level of the safety philosophy consists of fourteen fundamental principles of reactor safety which buttress the requirements of the safety objectives stated earlier. The objectives state what is to be achieved and the principles provide the basic safety concepts necessary to achieve these objectives. The principles are interrelated and must be taken as a whole. They are not meant to constitute a menu for selection.

Two of the fundamental principles, defence in depth and safety culture, are broadly based and encompass all the others. Thus the principles have been formulated in two groups as shown in Figure 2. All the other principles can be derived from these key principles and amplify their basic requirements. The two groups of principles are strongly interlocking. Application of
Defence in depth requires safety culture principles; in turn, the defence in depth concepts are integrated within the safety culture principles.

In this paper, the principles are stated and only a summary of their detailed evaluation criteria are presented.

4.1 Defence in Depth Principles

A defence in depth strategy shall be implemented at small reactor facilities to compensate for potential human and mechanical failure, and unexpected occurrences.

Defence in depth is one of the key principles essential for reactor safety to prevent, correct or compensate for deficiencies or failures. It should be in place for all safety activities, whether organizational, behavioural or equipment related.

Defence in depth strategy comprises four overlapping echelons of protection which are provided to prevent accidents or to ensure an appropriate level of safety in the event that prevention fails. These four echelons are prevention, control, protection and mitigation. In turn, each of these echelons employs two parallel and interlocking types of measures: measures related to equipment, including successive barriers, and measures related to human activities, particularly good operational practices.

Accident prevention, the first echelon of defence, is achieved by high quality in design and operation, proven engineering practice, conservative design, reliability, inherent safety characteristics, and effective surveillance.

The second echelon, reactor control, embodies good operational practice, reliable process systems, conservative operational limits and conditions, passive safety features, core stability, and fault tolerant design and operations.

The third echelon protects the reactor facility from accident sequences, whether of internal or external origin. Emergency equipment and procedures are provided to achieve stable and acceptable conditions following design basis events.

Accident mitigation, the final defence in depth echelon, protects the public from very rare but severe accidents which exceed the reactor design basis. This includes emergency measures to limit potential radiation exposures.

Six principles can be derived from defence in depth. These are shown in Table 4, together with a summary of typical evaluation criteria. There is a one-to-one correlation between four of the principles and the defence in depth echelons discussed above. The other two principles, "proven engineering practice" and "operational radiation safety", are applicable to all the echelons; these are important enough to be stated as separate principles.
Confirmation that all these principles are in place is provided by a strong safety culture, effective quality assurance, and safety assessment and verification activities.

4.2 Safety Culture Principles

A safety culture shall be established and govern the actions of all individuals and organizations engaged in activities related to small reactors.

Safety culture, the second key principle essential for reactor safety, can be described as the pursuit of excellence in all matters pertaining to reactor safety. It involves a pervasive safety thinking and personal dedication and accountability. It should govern the actions and interactions of all individuals and organizations engaged in any activity which has a bearing on safety. Safety culture can be characterized by four elements: commitment, direction, competence, and assessment.

The commitment element stresses the importance of an overall commitment to safety by organizations and individuals. Full attention is paid to safety; open, questioning attitudes are fostered; and established procedures are respected.

Direction requires that safety attitudes and requirements are instituted and encouraged by senior management. Clear lines of authority and responsibility are established, procedures developed, sufficient resources provided and quality assurance is implemented.

Competence stresses high standards of human performance. Staff selection and training emphasize inherent abilities, qualification, personal stability, integrity, and responsible attitudes.

The assessment element requires assessment and verification activities to be implemented. Reviews and audits are conducted for all activities important to safety and an ongoing safety assessment program is established. Lessons stemming from operational experience and safety research, both within the organization and internationally, are learned and acted on.

Table 5 states six principles which amplify safety culture and can be derived from it. A summary of typical evaluation criteria are also presented.

5. DISCUSSION

The safety philosophy presented in this paper is believed to be reasonably complete and appropriate for a wider peer review. A number of key issues have been identified in the course of this work - some have been resolved
<table>
<thead>
<tr>
<th>PRINCIPLE</th>
<th>TYPICAL CRITERIA</th>
</tr>
</thead>
<tbody>
<tr>
<td>ACCIDENT PREVENTION</td>
<td>- High quality design, construction and operation to prevent deviations from normal operation.</td>
</tr>
<tr>
<td></td>
<td>- Design conservatism to prevent failure.</td>
</tr>
<tr>
<td></td>
<td>- Selection of proper codes, standards, materials.</td>
</tr>
<tr>
<td></td>
<td>- Adequate inspection, monitoring, testing, and maintenance procedures to prevent failure.</td>
</tr>
<tr>
<td></td>
<td>- Operating procedures and training to minimize the likelihood of faults.</td>
</tr>
<tr>
<td>REACTOR CONTROL</td>
<td>- Prevention of deviations by conservative design, wide safety margins and high quality.</td>
</tr>
<tr>
<td></td>
<td>- Fault tolerant designs promoting inherent safety, stability, passive and fail-safe features.</td>
</tr>
<tr>
<td></td>
<td>- Operational limits and conditions, surveillance and good operating procedures.</td>
</tr>
<tr>
<td></td>
<td>- Monitoring, indication and human factors design.</td>
</tr>
<tr>
<td></td>
<td>- Reliable process systems.</td>
</tr>
<tr>
<td></td>
<td>- Process barriers and protection of barriers.</td>
</tr>
<tr>
<td></td>
<td>- Systems and practices to control radioactivity.</td>
</tr>
<tr>
<td>PROTECTION</td>
<td>- Passive shutdown and decay heat removal, or reliable, automatic safety systems.</td>
</tr>
<tr>
<td></td>
<td>- Barriers for protection against internal/external events and to confine/control radioactivity.</td>
</tr>
<tr>
<td></td>
<td>- Protection of systems and barriers; qualification.</td>
</tr>
<tr>
<td></td>
<td>- Emergency operating procedures established.</td>
</tr>
<tr>
<td></td>
<td>- High standards of quality assurance in design, construction, commissioning and operations.</td>
</tr>
<tr>
<td>ACCIDENT MITIGATION</td>
<td>- Siting and exclusion zones to reduce impacts.</td>
</tr>
<tr>
<td></td>
<td>- Precautions to prevent escape of radioactivity.</td>
</tr>
<tr>
<td></td>
<td>- Use of inherent features and/or structures and systems to limit releases.</td>
</tr>
<tr>
<td></td>
<td>- Adequate warning to the public affected in the event of significant radioactive releases.</td>
</tr>
<tr>
<td></td>
<td>- Ultimate emergency operating procedures.</td>
</tr>
<tr>
<td></td>
<td>- Emergency measures on/off site where required.</td>
</tr>
<tr>
<td>PROVEN ENGINEERING PRACTICE</td>
<td>- Design, construction and testing to quality standards commensurate with safety objectives.</td>
</tr>
<tr>
<td></td>
<td>- Use of recognized codes and standards; R and D programs, prototypes if information is lacking.</td>
</tr>
<tr>
<td></td>
<td>- Safety assessment for experiments where standards are not appropriate.</td>
</tr>
<tr>
<td></td>
<td>- Use of well established manufacturing/construction methods; prototypes for new/complex construction.</td>
</tr>
<tr>
<td></td>
<td>- Review of repairs or modifications to ensure systems are returned to appropriate state.</td>
</tr>
<tr>
<td>OPERATIONAL RADIATION SAFETY</td>
<td>- Doses to be ALARA and below prescribed limits.</td>
</tr>
<tr>
<td></td>
<td>- Establishment of radiation protection program.</td>
</tr>
<tr>
<td></td>
<td>- Control of sources, exposures; protect barriers.</td>
</tr>
<tr>
<td></td>
<td>- Monitoring program: personnel, area radiation, contamination, airborne, emissions, emergency.</td>
</tr>
<tr>
<td></td>
<td>- Supervision of radioactive material, wastes, emissions, shipments.</td>
</tr>
<tr>
<td></td>
<td>- Adequate staffing, training and procedures.</td>
</tr>
<tr>
<td>PRINCIPLE</td>
<td>TYPICAL CRITERIA</td>
</tr>
<tr>
<td>-----------</td>
<td>-----------------</td>
</tr>
</tbody>
</table>
| RESPONSIBILITY OF OPERATING ORGANIZATION | - Full responsibility for operational safety and financial capability.  
- Adequate number of trained, qualified personnel.  
- Availability of competent engineering and technical support throughout facility life.  
- Operation in accordance with approved procedures.  
- Operational limits and conditions established.  
- Inspection, testing, and maintenance programs.  
- Quality assurance program; reviews and audits.  
- Review and approval of experiments.  
- Adequate security and safeguards programs. |
| QUALITY ASSURANCE | - Quality assurance program instituted during the initial phases of any project.  
- Define required quality and means of achieving it.  
- Responsibilities defined; work planned/controlled.  
- Right information to right people at the right time; relevant experience sought and used.  
- Work verified; deficiencies corrected; changes controlled; records managed; reviews/audits done. |
| HUMAN FACTORS | - Organizational support: resources; operation, maintenance and inspection aids.  
- Staff selection: attitude, ability and stability.  
- Training to recognize safety significance, follow procedures and understand facility.  
- Design: human factors and ergonomic principles to reduce error/stress and facilitate correct action.  
- Control layout; diagnostic aids; clear indication.  
- Administrative controls verified and justified. |
| SAFETY ANALYSIS AND VERIFICATION | - Safety assessment process in parallel with all life-cycle activities: siting to decommissioning.  
- Evaluation, documentation, review, audit.  
- Safety report: justify safety case; updated when required by modifications or new information.  
- Safety analysis: reactor response to accident initiating events; methods validated.  
- Operational assessments: procedures; experiments. |
| FEEDBACK OF OPERATING EXPERIENCE | - Operator maintains effective system for collecting and interpreting operating experience.  
- Safety related abnormal events are identified, analyzed and measures taken to prevent recurrence.  
- Event data-bank maintained by Competent Authority.  
- Sharing of operational data coordinated nationally and internationally; feedback to designers.  
- Design and operations interaction; incorporation of results of relevant research. |
| REGULATORY CONTROL AND VERIFICATION | - Separation of regulator from promoters, vendors, and licensees to ensure independence.  
- Provide regulations, guides and safety criteria.  
- Review and assessment of safety documentation for each stage of licensing.  
- Issuing licenses and approvals authorizing action and placing conditions on licensee.  
- Surveillance: monitoring, inspection, enforcement.  
- Ensure compliance with requirements and stipulate corrective action to promote safety.  
- National and international liaison and dissemination of safety information. |
at the principles level, while others are still under discussion; comments on them are invited. They include:

- **Use of probabilistic risk criteria.** The numerical assessment criteria finally adopted are event frequency and consequence criteria as they are easier to apply than summed risk criteria. The confidence which can be placed on the results of the numerical evaluation is a function of how well the remaining fundamental principles are met.

- **Balance of front-end versus back-end protection.** Designs which place great emphasis on prevention of accidents need place less emphasis on consequence mitigation. Similarly, designs for which a severe accident is more likely should place more emphasis on mitigation measures. While a need for balance has been recognized by the group, we have not yet developed appropriate criteria.

- **Treatment of severe accidents.** These are less significant than for power reactors due to the smaller fission product inventory; we have not emphasized their investigation as being particularly productive, but have encouraged designers to briefly examine what lies "beyond the cliff" at frequencies a decade lower than $10^{-6}$ per year, and to provide mitigation measures where warranted.

- **Environmental goals.** We have not found much published work on defining environmental goals for small (or large) reactors. Our initial work presented here is fairly general at this stage; quantitative criteria are under development.

The work of the SRC is now focussed on producing criteria for specific applications, as described in Section 2 and illustrated in Figure 3. We are seeking broad peer comment on the safety philosophy presented here as
the foundation of our approach, and will be seeking review again once the applications sections are completed. Reviews will be requested from organizations involved in design, operation, and licensing of small reactors in Canada, and from the international community.

The hierarchical structure of the criteria has been chosen to provide a well-defined framework for subsequent development of third level guides by appropriate experts. This third stage is expected in Canada after the complete SRC document achieves some national consensus.

6. CONCLUSIONS

Recent small reactor proposals and licensing activities in Canada have indicated that safety criteria are needed. A Small Reactor Criteria group was formed to address this need and has made considerable progress. We have built on national and international experience for both power reactors and small reactors to develop comprehensive safety criteria in a tiered, hierarchical structure as an overall basis for subsequent detailed criteria. The work is meant to be compatible with current research reactor guide development projects by the IAEA, and with the Agency’s safety documentation practice and structure. We are seeking peer review of our work to date and are publishing to solicit comments.

This paper is a summary of a draft report in which complete and detailed criteria are presented. A comprehensive safety philosophy is proposed, consisting of basic safety objectives and fundamental principles. We believe that there is a need for such a philosophy and that it is essential to an understanding of the underlying concepts of reactor safety.

The development of criteria in Canada in response to a current need for application to real projects has proven to be beneficial. There has been interaction among designers, operators and regulators which has had a beneficial effect in advancing a small reactor safety culture.

7. REFERENCES


USE OF THE POWER BURST FACILITY 
FOR BORON NEUTRON CAPTURE THERAPY

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USE OF THE POWER BURST FACILITY FOR BORON NEUTRON CAPTURE THERAPY

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ABSTRACT

A program is under development at the Idaho National Engineering Laboratory (INEL) that involves using the Power Burst Facility (PBF) for research into Boron Neutron Capture Therapy (BNCT). BNCT utilizes the ionizing energy from boron-neutron capture to stop reproduction of or destroy cells in cancerous tissue in a two step process. The first step is to selectively concentrate a boron isotope within the tumor cell, that when activated by neutron capture emits highly ionizing, short range particles. The second step involves activation of the isotope only in the vicinity of the tumor with a narrow neutron beam. The \((^{10}\text{B}(n,^4\text{He})^7\text{Li})\) reaction with thermal neutrons produces fission products with track lengths approximately equal to a cell diameter. The INEL program includes the modification of the PBF by the addition of a neutron filter and treatment area. The filter will down scatter high energy neutrons into the epithermal range and remove thermal neutrons and excessively damaging gamma components. The intense source of epithermal neutrons from PBF is considered necessary to achieve optimum therapy for deep-seated tumors with minimum damage to surface tissue. The neutron filter conceptualized for PBF utilizes aluminum and heavy water to down scatter neutrons into the proper energy range. Bismuth will be used for gamma shielding and cadmium will remove the thermal neutron contaminant from the beam. The INEL program leads to human clinical trials at PBF which is intended to prove that brain tumors can be successfully treated through noninvasive techniques. Further research into BNCT at PBF for other cancer types is also anticipated.

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BACKGROUND AND SIGNIFICANCE OF BORON NEUTRON CAPTURE THERAPY:

Despite rapid progress in cancer research and reasonably effective efforts at prevention, cancer will soon become the foremost killer in the United States. While developing immunological and chemical therapies offer promise for controlling the systemic disease, relapse in the brain remains a significant problem. Over 80% of patients with metastatic melanoma, and many patients with metastatic tumors of other origin, have brain lesions. Brain tumors are not as accessible to treatment with drugs or biologicals because those agents are excluded by a physiological barrier.\(^{[1]}\) Approximately 4000 cases of the most aggressive brain tumor, Glioblastoma Multiforme, are diagnosed in the United States each year. Existing therapy offers little hope for survival with this disease. Median life expectancy is less than one year, and patients rarely survive therapy beyond two years.\(^{[2,3]}\)

Boron Neutron Capture Therapy (BNCT) was first proposed as a possible treatment for cancer by Locher in 1936.\(^{[4]}\) In concept, BNCT requires the uptake of an activatable nuclide within the tumor cell. The nuclide is then fissioned by neutrons resulting in a neutron-alpha reaction (i.e. \(^{10}\text{B} + ^1\text{n} \rightarrow ^7\text{Li} + ^{4}\text{He} + 2.3\text{ MeV}\) when a boron compound is used). This reaction produces, 94% of the time, \(^7\text{Li}\) and \(^4\text{He}\) (alpha) ions with 0.84 and 1.45 MeV of kinetic energy respectively, plus a 0.5 MeV gamma ray. Six percent (6%) of the time, no gamma ray is produced, with the \(^7\text{Li}\) and \(^4\text{He}\) ions sharing the 2.8 MeV kinetic energy. (Fig. 1.)

\[\begin{align*}
\text{\(\text{n}\)} & \quad \text{\(6\mu\text{m}\)} \\
\text{\(\text{\(7\mu\text{m}\)} +3\)} & \quad \text{\(0.85\text{ MeV}\)} \\
\text{\(\text{\(9\mu\text{m}\)} \gamma\)} & \quad \text{\(0.5\text{ MeV}\)} \\
\text{\(\text{\(\alpha\)} +2\)} & \quad \text{\(1.49\text{ MeV}\)} \\
\end{align*}\]

Fig. 1. Boron-10 Fission With A Thermal Neutron

The ionizing biologic effect of the fission fragments in the reaction has two particular advantages. First, the ionizing energy conveyed occurs within 10 micrometers with path length determined by the particular ion and its energy. Since the 10 micrometer path length is, generally speaking, equivalent to the diameter of the tumor cell, the ionizing damage is very cell selective. Second, the energy conveyed falls within the range of high linear energy transfer (LET) radiations, which have a greater biological effectiveness than more conventional radiation used in radiation therapy such as \(^{60}\text{Co}\) rays and high energy x-rays.
Boron Neutron Capture Therapy (BNCT) of malignant brain lesions was evaluated in the United States in the late 1950's and early 1960's. The early trials failed because of technological immaturity. The causes of failure were identified at the time of the trials as:

1. Excessive radiation damage to the vasculature system leading to hemorrhage
2. Treatment-related edema and intracranial pressure
3. Use of low energy (thermal) neutrons that caused surface (scalp) damage but failed to penetrate to the depth of the tumor.

**INEL PROGRAM**

A comprehensive program for BNCT using an epithermal neutron beam from the Power Burst Facility (PBF) has been prepared at the Idaho National Engineering Laboratory (INEL). The PBF must be modified to provide the epithermal neutron beam by the addition of a neutron filter which will downscatter fast neutrons into the epithermal range and remove the damaging thermal neutrons and gamma radiation. The reactor building must be modified with the addition of a patient treatment room, patient monitoring room and shielded shutter doors to reduce radiation levels during patient setup. (Fig. 2)

The use of epithermal neutrons should offer significant advantages over the early trials. It should be possible to deliver destructive doses to the internal tumor without surgery and without great damage to the healthy tissue. Optimum treatment conditions for brain tumors require boron activation by epithermal neutrons which have sufficient energy to penetrate the scalp and skull to the depth of the tumor but inadequate to damage tissue that does not contain boron. The epithermal neutrons then slow down (thermalize) forming a capture peak a few centimeters into the brain at the tumor site. Feasibility studies have indicated that capabilities and facilities presently available can be adapted or modified to generate all data required for scientific assessment of the BNCT process. Those same technological capabilities will also provide the ability to assure safety of human treatment and to limit human treatment to those individual cases for which there is a reasonable probability of efficacy.

United States research into BNCT, since the 1960 trials, has focussed on development of improved boron delivery agents that offer better tumor retention and blood clearance, thereby reducing vasculature damage. Borocaptate Sodium (Na$_2$B$_{12}$H$_{11}$SH) or BSH and the related dimer BSS (Na$_4$B$_{24}$H$_{22}$S$_2$) and BSSO the oxidized dimer (Na$_4$B$_{24}$H$_{22}$S$_2$O) appear to offer the necessary characteristics. Brookhaven National Laboratory (BNL) researchers have recently reported the first experimental proof of in-vivo efficacy using the BSS dimer and a malignant intracranial neoplasm implanted in a rat model. Dr. H. Hatanaka's clinical experience using BSH and a low-energy neutron source (summarized in Fig. 3), is very
Figure 2. BNCT Therapy In PBF-Artists Concept

11 Patients treated within the last 3 years

12 BNCT patients in 1968 to 1985 group having tumor within 6 cm of the brain surface

38 BNCT cases treated without regard to depth between 1968 and 1985

46 cases treated with photons

Fig. 3. Grade III/IV Astrocytoma Survival Data of Dr. Hatanaka
encouraging and has stimulated world wide effort to further develop the technology. (Grade III/IV Astrocytoma is a function classification of brain tumor patients. The referenced grades are advanced and in the symptomatic area typical of Glioblastoma Multiforme.)

Dr. Hatanaka's results have been criticized for lacking research vigor, but careful scrutiny by several physicians leaves little doubt that Dr. Hatanaka has survival rates unequalled by any other clinician.

The INEL program is a collaborative endeavor of leading U.S. technologists in each speciality field required for successful completion of overall program objectives. These objectives are: (1) to provide data required for a decision to proceed to human clinical trials of BNCT, using Na$_2$B$_{12}$H$_{12}$SH (BSH), as a nonsurgical treatment for Glioblastoma Multiforme and (2) to provide facilities, technology, data, and institutional approvals required for initiation of human clinical trials.

**Power Burst Facility**

The PBF is an open pool, light water cooled reactor capable of operating either in a steady state or transient mode. A closed loop coolant system provides forced cooling of the core. The reactor system is capable of continuous operation at power levels to 28 MW. The PBF core is contained within a 1.3 meter-diameter, 0.9 meter-high cylindrical envelope. The fuel rods are composed of ternary (20.6% urania, 61.8% zirconia, and 7.6% calcia) ceramic fuel pellets contained in a ceramic (zirconia, calcia) thermal insulator and a stainless steel cladding tube.

The core loading will be changed for BNCT by the addition of fuel to the core quadrant near the patient treatment location to enhance the neutron beam. The additional fuel will replace partially fueled canisters located at the edge of the core. The spacing of fuel rods in the core provides an intentionally under moderated core, which, combined with the relatively low $^{235}$U atom density, results in a high fast neutron flux. The PBF reactor is the most powerful U.S. source of an intense current ($10^{10}$ n/cm$^2$-s) of epithermal neutrons with contaminants (gamma photons and neutrons above 10 keV) reduced to the extent they are clinically insignificant. This intense, high-purity, epithermal neutron beam is required in order to evaluate the optimum therapeutic dose rate and the potential for BNCT to make the transition from the research laboratory to the medical community and become a practical, therapeutic tool available to the large number of patients needing the treatment.

**PBF Neutron Filter Design and Installation**

The design objective for the PBF reactor modification is to obtain the most desirable balance between the competing parameters of maximized epithermal neutron component and minimized fast neutron, thermal neutron, and gamma components.
Functional and operational requirements for the BNCT neutron beam and related hardware at the PBF reactor are:

1. The epithermal neutron flux, averaged across the beam, will be $\geq 7 \times 10^9 \text{n/cm}^2\text{-s}$ at the patient location for neutron energies in the range $1 \text{ eV} < E_n < 10 \text{ eV}$ for a 20-MW(t) reactor power level.

2. The fast neutron contaminate ($E_n > 10 \text{ keV}$) in the beam path shall result in a tissue dose $< 2.6 \times 10^{-11} \text{ cGy/(n/cm}^2)$ of the epithermal flux at the patient location.

3. The incident gamma radiation in the beam shall result in a tissue dose rate $\leq 2.0 \times 10^{-11} \text{ cGy/(n/cm}^2)$ of the epithermal flux at the patient location.

4. The reactor will operate at a constant power level not to exceed 20 MW during neutron therapy and $< 1 \text{ MW}$ during setup and patient positioning.

5. The center line of the treatment position will be approximately 3.4 meters from the vertical center line of the PBF reactor core.

6. The radiation shielding shutter doors will operate remotely and provide a reduction factor of $\geq 4.0 \times 10^4$ for incoming and induced radiation (neutron and gamma) in the neutron beam when the door is closed and the reactor is operating at $\leq 1 \text{ MW}$.

The neutron filter as designed is a 120-cm diameter, 100-cm long, 304 stainless steel tank, filled with closely spaced aluminum plates, (Fig. 4). Space between the aluminum plates is filled with heavy water (D.O). Spacing is adjusted to achieve a 90%/10% aluminum/D.O volume ratio for neutron spectra adjustment. The D.O will be circulated for cooling the 78, 0.99-cm aluminum plates. The aluminum plates and D.O coolant simulate a homogenous mixture in the filter region.

Design calculations for the PBF aluminum/D.O filter are performed with two independent computer models. Both models rely on core/reflector interface neutron currents calculated by the two-dimensional DOT core model. These currents are input to the RAFFLE Monte Carlo filter model and to the DOT filter model. The RAFFLE representation of geometry and physics data is more rigorous than the DOT results. The DOT model is required to provide the detailed angular fluxes at the beam exit for patient dose calculations. The calculational models used for the aluminum/D.O filter design for PBF were also used to design the aluminum oxide neutron filter for the Brookhaven Medical Research Reactor (BMRR). Measurements of the filtered neutron beam flux and spectrum were made at BMRR to verify the predictions made by the design codes.
Bismuth and cadmium for gamma shielding and thermal neutron removal is provided in a transition section on the patient end of the main neutron filter. The transition piece is connected to a 1.25-meter O.D. nozzle section, which extends through the reactor vessel wall. The inside diameters of the nozzle section and transition piece are lined with bismuth to shield the beam from gamma radiation produced by $^{16}$N decay in the reactor coolant water.

The neutron shutter, which will be opened and closed to commence and terminate treatment, will be placed outside the reactor vessel near the patient. The shutter is a two-piece, lead and polyethylene shield, 60-cm thick, with the two parts connected by a chain over sprocket. Hydraulic cylinders move the shielding to open or close the shutter. Conical openings in the shielding align when the shutter is opened to allow the neutron beam to reach the patient to commence treatment. Treatment is terminated by a timed sequencer, which will reverse the action of the hydraulic cylinders, closing the beam path with the shielded shutters. Reactor power will be lowered to < 1 MW after the shutter is closed.

The entire sequence will be monitored by a physician and a reactor operator from the patient monitoring room near the patient treatment room. The monitoring room is equipped with instrumentation to measure beam strength and quality before, during, and after treatment and is
shielded from the therapy room by a shielding labyrinth. Remote video observations of the patient and two-way voice communication will be provided between the patient and physician.

**BNCT Research and Analysis Laboratory**

The use of the PBF in BNCT represents a new mission at the INEL. Research into medical therapy, using an INEL reactor, has not been performed previously to the extent projected for BNCT. This research will require diagnostic tools not used in existing programs and not available at INEL. A laboratory facility near PBF has been proposed to provide an international center close to the source of the epithermal neutrons for research related to development of BNCT for treating a variety of human tumors. The laboratory will provide state-of-the-art medical and physics research facilities and diagnostic tools; laboratories and kenneling to support animal research; and offices, limited overnight human housing and various auxiliary facilities. The laboratory will host both domestic and international research and must provide the proper tools to support biological, medical, and physics research. Hospital facilities for human research will be provided by the Eastern Idaho Regional Medical Center (EIRMC) in Idaho Falls.

**CONCLUSION**

BNCT will first be applied to the aggressive brain tumor Glioblastoma Multiforme because of the insidious nature of the disease and because the orphan disease status facilitates United States Food and Drug Administration (FDA) approvals. Researchers expect, however, that BNCT will also be effective for other types of cancer in various parts of the body. Advanced research directed toward treatment of other cancer types is not part of the presently proposed three-year program plan, however, PBF is envisioned as becoming a baseload-funded international research tool, as well as a brain tumor treatment center.

It is anticipated that human clinical trials will prove that BNCT can be provided on an outpatient basis and provide an order of magnitude improvement over present treatment methods for Glioblastoma Multiforme, and subsequently for other tumor types for which present treatments are ineffective. Success in this endeavor will be a tremendous benefit to mankind and will be a powerful demonstration of the beneficial uses of nuclear reactors.
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"DISEÑO DE UN REACTOR DE INVESTIGACIÓN
DE PRUEBAS MÚLTIPLES"

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ABSTRACT

The availability of a research reactor is essential in any endeavour to improve the execution of a nuclear programme, since it is a very versatile tool which can make a decisive contribution to a country's scientific and technological development. Because of their design, however, many existing research reactors are poorly adapted to certain uses. In some nuclear research centres, especially in the advanced countries, changes have been made in the original designs or new research reactor prototypes have been designed for specific purposes. These modifications have proven very costly and therefore beyond the reach of developing countries. For this reason, what the research institutes in such countries need is a single sufficiently versatile nuclear plant capable of meeting the requirements of a nuclear research programme at a reasonable cost. This is precisely what a multipurpose reactor does.

The Mexican National Nuclear Research Institute (ININ) plans to design and build a multipurpose research reactor capable at the same time of being used for the development of reactor design skills and for testing nuclear materials and fuels, for radioisotope production, for nuclear power studies and basic scientific research, for specialized training, and so on.

For this design work on the ININ Multipurpose Research Reactor, collaborative relations have been established with various international organizations possessing experience in nuclear reactor design: Atomenergoeksport of the USSR; Atomic Energy of Canada Limited (AECL); General Atomics (GA) of the USA; and Japan Atomic Energy Research Institute (JAERI).
"DISEÑO DE UN REACTOR DE USOS MÚLTIPLES".

RESUMEN.

La disponibilidad de un reactor de investigación es una parte esencial para un mejor desempeño de un programa nuclear, dado que es una herramienta muy versátil que puede contribuir en forma determinante en el desarrollo científico y tecnológico de un país. Sin embargo, muchos reactores de investigación existentes son poco satisfactorios para algunas aplicaciones, debido a su diseño. En algunos centros de investigación nuclear, especialmente en países desarrollados, se han realizado cambios en los diseños originales o bien, se han diseñado nuevos prototipos de reactores de investigación para fines específicos. Estas opciones han resultado muy costosas, por lo que están fuera del alcance de países en desarrollo. Por lo tanto, lo que los Institutos de Investigación en estos países necesitan, es una sola instalación nuclear suficientemente flexible, que permita satisfacer las necesidades de un programa de investigación nuclear a un costo razonable. Este es el caso de un Reactor de Usos Múltiples.

En el Instituto Nacional de Investigaciones Nucleares (ININ) de México se ha pensado diseñar y construir un Reactor de Investigación de Usos Múltiples, que al mismo tiempo sirva para desarrollar la capacidad técnica y humana en el diseño de reactores y permita realizar pruebas de materiales y combustibles nucleares, producción de radioisótopos, investigaciones en energía nuclear y ciencias básicas, capacitación de personal especializado, etc.

Para la realización del diseño del Reactor de Usos Múltiples del ININ se han establecido relaciones de colaboración con diferentes organizaciones internacionales con experiencia en el diseño de reactores nucleares, como son: Atomenergoexport, de la Unión Soviética; Atomic Energy of Canada Limited (AECL), de Canadá; General Atomics (GA), de Estados Unidos; y Japan Atomic Energy Research Institute (JAERI), de Japón.
1. OBJETIVOS DEL PROYECTO.

Corresponde al Instituto Nacional de Investigaciones Nucleares (ININ), desarrollar en México la capacidad técnica para diseñar y construir un Reactor de Usos Multiples, que complemente los recursos disponibles y amplíe las bases para desarrollar las aplicaciones de la energía nuclear en el país.

Por esta razón, en el ININ existe un proyecto, cuyo objetivo es diseñar y construir un Reactor de Investigación de Usos Múltiples que permita al Instituto desarrollar la infraestructura técnica y humana necesaria para participar de manera más eficiente en los servicios y tecnología, requeridos por un futuro programa nucleoelectrónico nacional.

El Reactor de Usos Múltiples deberá ser una instalación nueva del Instituto Nacional de Investigaciones Nucleares, construida en el Centro Nuclear de México conforme a las normas y medidas de seguridad vigentes, para funcionar paralelamente a la operación del reactor nuclear TRIGA Mark III, en uso desde 1968 [2].

Con el trabajo de diseño de un reactor de este tipo, se pretende ampliar la experiencia existente en el ININ en las áreas de:

a) **Física de reactores**, mediante el cálculo neutrónico del núcleo.

b) **Termohidráulica**, al realizar el cálculo de los circuitos de enfriamiento tanto del núcleo, como de los circuitos de experimentación con que estará provisto el reactor.

c) **Seguridad y control del reactor**, mediante el conocimiento de las filosofías utilizadas en varios países para garantizar la operación segura del reactor, así como el tipo de instrumentación que se utiliza para la operación y control del mismo.

d) **Cálculos de blindajes**, al diseñar el blindaje requerido por el reactor.
2. ETAPAS DEL PROYECTO [33].

Para el desarrollo de este proyecto, se ha dividido el mismo en las siguientes etapas:

2.1 **Estudio Preliminar.** En el cual se determinaron las principales aplicaciones que tendría un Reactor de Usos Múltiples, así como algunos de los requerimientos del mismo.

2.2 **Diseño conceptual.** En el cual se identifica el tipo de reactor que mejor se adapte a las necesidades del ININ y se fijan las bases de diseño del reactor de usos múltiples seleccionado.

2.3 **Diseño básico.** Durante el cual, una vez definido el tipo de reactor y aprobado el concepto por el Órgano Regulador mexicano, se empezarán a generar los planos de construcción y los documentos de diseño.

2.4 **Ingeniería de detalle.** En esta etapa se especifican todas las dimensiones y materiales de todos los equipos y componentes de los sistemas.

2.5 **Desarrollo de la construcción.** Etapa durante la cual se lleva a cabo la ejecución de la obra civil y montaje de equipos y componentes.

3. ESTUDIO PRELIMINAR [32].

Esta primera etapa del proyecto se desarrolló con la ayuda de especialistas de la Unión Soviética mediante un acuerdo de cooperación con la compañía ATOMENERGOEXPORT que permitió que cinco especialistas de varios Institutos de investigación y una traductora vinieran a México para colaborar con personal del ININ para la realización del "Estudio Preliminar" del Reactor de Usos Múltiples.

Antes del arrivo de los especialistas soviéticos y durante su estancia en México se realizaron consultas con los diferentes grupos de investigación del ININ con el objeto de determinar las necesidades y los requerimientos que deberá tener un Reactor de Usos Múltiples para México.
Durante esta etapa, que duró seis semanas se determinaron las aplicaciones del reactor, los requerimientos de los canales experimentales, los parámetros neutrónicos y físicos, así como los parámetros termohidráulicos del reactor propuesto.

También se hizo una revisión de los diferentes sistemas del reactor que deberán ser considerados durante las siguientes etapas del diseño y se recopilaron datos sobre el posible sitio de emplazamiento y de los radioisótopos de mayor consumo en el país que pueden ser producidos en él mismo.

En las tablas I, II y III se muestran algunas de las características técnicas del Reactor de Usos Múltiples propuesto.

4. DISEÑO CONCEPTUAL

Para realizar el diseño conceptual, se obtuvo la colaboración de la compañía Atomic Energy of Canada Limited (AECL) de Canadá, mediante un convenio de colaboración conjunta por el cual, un grupo de especialistas mexicanos estuvo trabajando con especialistas canadienses de la Unidad de Tecnología de Reactores Pequeños de los Laboratorios Nucleares de White Shell en Pinawa, Manitoba, Canadá, para realizar los cálculos neutrónicos, termohidráulicos y blindaje del reactor, así como para revisar la filosofía de seguridad empleada en este tipo de reactores.

Este trabajo se desarrolló durante un período de nueve meses, culminándose con un reporte conjunto de los resultados obtenidos, algunos de los cuales se mencionan enseguida de manera somera.
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<td></td>
<td>- Con las barras de compensación de reactividad insertadas</td>
</tr>
<tr>
<td>9</td>
<td>Densidad máxima del flujo de neutrones x 10^{-14} n/cm²-s</td>
<td>(Con todas las barras de compensación de reactividad insertadas)</td>
</tr>
<tr>
<td></td>
<td></td>
<td></td>
</tr>
<tr>
<td></td>
<td></td>
<td>- En el CE de diámetro hasta 68 mm colocado en lugar del ensamble combustible.</td>
</tr>
<tr>
<td></td>
<td></td>
<td></td>
</tr>
<tr>
<td></td>
<td></td>
<td>- Térmicos en el reflector de Be.</td>
</tr>
</tbody>
</table>

* Las fig. 1, 2 y 3 verlas en el Anexo 1.
<table>
<thead>
<tr>
<th>No.</th>
<th>PARÁMETROS</th>
<th>CONFIGURACIONES DEL NÚCLEO.²</th>
</tr>
</thead>
<tbody>
<tr>
<td></td>
<td></td>
<td>1 Fig.1</td>
</tr>
<tr>
<td>1</td>
<td>Número de ensambles combustibles de 6 tubos en el núcleo</td>
<td>24</td>
</tr>
<tr>
<td>2</td>
<td>Superficie de transferencia de calor en el núcleo (m²)</td>
<td>32.9</td>
</tr>
<tr>
<td>3</td>
<td>Factor pico de potencia en el núcleo</td>
<td></td>
</tr>
<tr>
<td></td>
<td>- En la sección transversal (Ks)</td>
<td>1.69</td>
</tr>
<tr>
<td></td>
<td>- En la dirección axial (Kz)</td>
<td>1.3</td>
</tr>
<tr>
<td>4</td>
<td>Altura de la columna de agua sobre el núcleo (m)</td>
<td>7</td>
</tr>
<tr>
<td>5</td>
<td>Presión del agua en la entrada al núcleo (bars)</td>
<td>1.4</td>
</tr>
<tr>
<td>6</td>
<td>Gradiente de presión en el núcleo (bars)</td>
<td>0.5</td>
</tr>
<tr>
<td>7</td>
<td>Velocidad promedio del agua en los espacios de los ensambles combustibles (m/s)</td>
<td>3.73</td>
</tr>
<tr>
<td>8</td>
<td>Gasto total de agua a través de los ensambles combustibles (m³/h)</td>
<td>900</td>
</tr>
<tr>
<td>9</td>
<td>Gasto total de agua a través de los bloques de reflector (reemplazables y fijos)(m³/h)</td>
<td>1040</td>
</tr>
<tr>
<td>10</td>
<td>Gasto total de agua a través del reactor (m³/h)</td>
<td>2000</td>
</tr>
<tr>
<td>11</td>
<td>Temperatura del agua en la entrada a los ensambles combustibles (°C)</td>
<td>30/40</td>
</tr>
<tr>
<td>12</td>
<td>Temperatura del inicio de ebullición en subenfríamiento (°C)</td>
<td>124</td>
</tr>
<tr>
<td>13</td>
<td>Temperatura de la pared del elemento combustible (máxima) (°C)</td>
<td>98/99</td>
</tr>
<tr>
<td>14</td>
<td>Densidad máx. del flujo de calor (KW/m²)</td>
<td>1196/1075</td>
</tr>
<tr>
<td>15</td>
<td>Potencia máx. permitida del reactor (MW)</td>
<td>19.1/17.2</td>
</tr>
</tbody>
</table>

Las fig. 1, 2 y 3 veríasa en el Anexo 1.
<table>
<thead>
<tr>
<th>INVESTIGACION Y APLICACIONES</th>
<th>NUMERO DE CANALES: DIAMETROS (mm)</th>
<th>FLUJOS (n/cm²-s)</th>
<th>LONGITUD DEL CANAL (mm)</th>
</tr>
</thead>
<tbody>
<tr>
<td>1. Prueba de combustibles.</td>
<td>2</td>
<td>68</td>
<td>1.15x10^{13}</td>
</tr>
<tr>
<td>2. Irradiación de materiales.</td>
<td>14</td>
<td>40</td>
<td>1.5x10^{14}</td>
</tr>
<tr>
<td>3. Producción de radiisótopos.</td>
<td>41</td>
<td>25.4</td>
<td>1x10^{14}</td>
</tr>
<tr>
<td>4. Producción de molibdeno</td>
<td>1</td>
<td>1</td>
<td>1x10^{14}</td>
</tr>
<tr>
<td>5. Análisis por activación</td>
<td>1 con Cd</td>
<td>150</td>
<td>1x10^{13}</td>
</tr>
<tr>
<td>6. Circuitos Experimentales</td>
<td>3</td>
<td>68</td>
<td>150</td>
</tr>
<tr>
<td>7. Fuentes Frías</td>
<td>1</td>
<td>150</td>
<td></td>
</tr>
<tr>
<td>8. Espectroscopía de gammamas inmediatas</td>
<td>1</td>
<td>100</td>
<td></td>
</tr>
<tr>
<td>9. Difracción de neutrones</td>
<td>2</td>
<td>100</td>
<td></td>
</tr>
<tr>
<td>10. Neutrografía</td>
<td>1</td>
<td>100</td>
<td></td>
</tr>
<tr>
<td>11. Física Nuclear</td>
<td>2</td>
<td>100</td>
<td></td>
</tr>
<tr>
<td>12. Implantación de impurezas en silicio</td>
<td>1</td>
<td>150</td>
<td></td>
</tr>
</tbody>
</table>
4.1 Análisis neutrónico.

Para analizar el comportamiento neutrónico, se tomó como modelo el núcleo de un reactor MAPLE de AECL y se analizaron básicamente dos alternativas para satisfacer los requerimientos planteados por el ININ. La primera alternativa consiste de dos reactores de 14 MW cada uno y la segunda consiste de un reactor de 25 MW. Las Fig. 1, 2 y 3 muestran las configuraciones del núcleo del reactor para estas alternativas.

En la Fig. 1 se muestra una configuración del núcleo, en la cual están contemplados dos canales para prueba de materiales y uno para prueba de combustibles, designados como MT y FT respectivamente.

La Fig. 2 muestra otra configuración del núcleo, la cual contiene: una posición para producción del Molibdeno-99, una posición para producción de Fluor-18, además de otras dos posiciones para prueba de materiales y prueba de combustibles, MT y FT respectivamente.

La Fig. 3 muestra la segunda alternativa, que es un solo reactor de 25 MW, la cual posee todas las posibilidades de irradiación requeridas.

Para proceder al análisis neutrónico de estas dos configuraciones se utilizaron los programas de cómputo: WIMS-CRNL, MAPDDT y 3DDT, propiedad de AECL.

WIMS-CRNL.

Este programa obtiene los parámetros de celda utilizando la teoría de transporte de neutrones. Para esto, utiliza una biblioteca de secciones eficaces microscópicas para 89 grupos de energías de los neutrones, obtenidas de la base de datos ENDF/B-V.

Alimentando al programa los datos de los materiales y características de una supercelda (ver figura 4), se calculan las secciones eficaces macroscópicas promedio correspondientes a la celda de interés homogeneizada para cada uno de los 32 grupos de neutrones. El programa, además calcula el quemado del combustible por ensamble.

Posteriormente los grupos de energía se colapsan de 32 a 5.
Fig. 1

PRIMERA ALTERNATIVA CONSIDERADA.

Fig. 2
Fig. 3

SEGUNDA ALTERNATIVA CONSIDERADA.
Fig. 4

MODELO DE UNA SUPERCELDA PARA
EL WIMS-CRNA.
Fig. 5. Representación del Reactor de 14 MW con el código 3DDT

Fig. 6. Representación del Reactor de 25 MW con el código 3DDT
Este programa sirve para preparar los datos de entrada al programa 3DDT.

Este programa representa el núcleo del reactor en una serie de celdas rectangulares en el plano X-Y (ver Fig. 5 y 6) y resuelve la ecuación de difusión de neutrones en tres dimensiones proporcionando la $k_{eff}$ y la distribución del flujo de neutrones en cada celda en el plano.

La dirección axial queda representada por la superposición de estos planos.

En las tablas IV, V y VI se muestran estos resultados.

**TABLA IV.**

**VALORES DE LA $k_{eff}$ EN EL NUCLEO.**

<table>
<thead>
<tr>
<th></th>
<th>OPCIÓN 2 x 14 MW</th>
<th>OPCIÓN 25 MW</th>
</tr>
</thead>
<tbody>
<tr>
<td></td>
<td>React. 1 React. 2</td>
<td></td>
</tr>
<tr>
<td><strong>NUCLEO FRESCO</strong></td>
<td></td>
<td></td>
</tr>
<tr>
<td>Sin experimentos</td>
<td>1.07014 1.05189</td>
<td>1.09085</td>
</tr>
<tr>
<td>Con experimentos</td>
<td>1.06474 1.06122</td>
<td>1.09135</td>
</tr>
<tr>
<td><strong>NUCLEO CON 4</strong></td>
<td></td>
<td></td>
</tr>
<tr>
<td><strong>DIAS DE QUEMADO</strong></td>
<td></td>
<td></td>
</tr>
<tr>
<td>Con experimentos</td>
<td>1.01393 1.02273</td>
<td>1.03723</td>
</tr>
<tr>
<td>Position in Core</td>
<td>Group 1</td>
<td>Group 2</td>
</tr>
<tr>
<td>------------------</td>
<td>---------</td>
<td>---------</td>
</tr>
<tr>
<td>B3 (MT)*</td>
<td>5.33x10^13</td>
<td>5.16x10^13</td>
</tr>
<tr>
<td>C1 (FT)**</td>
<td>2.67x10^13</td>
<td>2.61x10^13</td>
</tr>
<tr>
<td>D3 (MT)</td>
<td>5.33x10^13</td>
<td>5.16x10^13</td>
</tr>
<tr>
<td>B3 (MT)*</td>
<td>5.33x10^13</td>
<td>5.16x10^13</td>
</tr>
<tr>
<td>C1 (FT)**</td>
<td>2.67x10^13</td>
<td>2.61x10^13</td>
</tr>
<tr>
<td>D3 (MT)</td>
<td>5.33x10^13</td>
<td>5.16x10^13</td>
</tr>
</tbody>
</table>

* MT - Materials Test Site
** FT - Fuel Test Loop Site

**FLUJO PROMEDIO POR CELDA PARA NUCLEO FRESCO DE LA ALTERNATIVA DEL REACTOR DE 14 MW.**
## TABLA VI

<table>
<thead>
<tr>
<th>Position in Core</th>
<th>Group 1</th>
<th>Group 2</th>
<th>Group 3</th>
<th>Group 4</th>
<th>Group 5</th>
</tr>
</thead>
<tbody>
<tr>
<td>A0 (FT)</td>
<td>3.28x10^{13}</td>
<td>3.17x10^{13}</td>
<td>3.47x10^{13}</td>
<td>9.52x10^{12}</td>
<td>2.02x10^{14}</td>
</tr>
<tr>
<td>B3 (MT)</td>
<td>7.76x10^{13}</td>
<td>7.57x10^{13}</td>
<td>7.33x10^{13}</td>
<td>1.85x10^{13}</td>
<td>3.07x10^{14}</td>
</tr>
<tr>
<td>C2 (MT)</td>
<td>8.04x10^{13}</td>
<td>7.92x10^{13}</td>
<td>7.62x10^{13}</td>
<td>1.91x10^{13}</td>
<td>3.07x10^{14}</td>
</tr>
<tr>
<td>C5 (NO)</td>
<td>5.83x10^{13}</td>
<td>5.65x10^{13}</td>
<td>5.50x10^{13}</td>
<td>1.40x10^{13}</td>
<td>2.42x10^{14}</td>
</tr>
<tr>
<td>D3 (MT)</td>
<td>7.12x10^{13}</td>
<td>6.95x10^{13}</td>
<td>6.71x10^{13}</td>
<td>1.69x10^{13}</td>
<td>2.81x10^{14}</td>
</tr>
<tr>
<td>E0 (F18)</td>
<td>3.15x10^{13}</td>
<td>3.04x10^{13}</td>
<td>3.33x10^{13}</td>
<td>8.86x10^{12}</td>
<td>1.92x10^{14}</td>
</tr>
<tr>
<td>E4 (FT)</td>
<td>2.67x10^{13}</td>
<td>2.57x10^{13}</td>
<td>2.80x10^{13}</td>
<td>7.45x10^{12}</td>
<td>1.65x10^{14}</td>
</tr>
</tbody>
</table>

<table>
<thead>
<tr>
<th>Position in Reflector</th>
<th>( \phi ) (n/cm²·s⁻¹)</th>
</tr>
</thead>
<tbody>
<tr>
<td>Site Name</td>
<td>Group 1</td>
</tr>
<tr>
<td>SU31</td>
<td>4.48x10^{11}</td>
</tr>
<tr>
<td>SU21</td>
<td>5.53x10^{12}</td>
</tr>
<tr>
<td>SU22</td>
<td>6.58x10^{12}</td>
</tr>
<tr>
<td>SU11</td>
<td>3.65x10^{12}</td>
</tr>
<tr>
<td>SU12</td>
<td>7.52x10^{12}</td>
</tr>
<tr>
<td>L11</td>
<td>4.18x10^{12}</td>
</tr>
<tr>
<td>L12</td>
<td>4.11x10^{12}</td>
</tr>
<tr>
<td>L22</td>
<td>2.72x10^{11}</td>
</tr>
<tr>
<td>L21</td>
<td>2.75x10^{11}</td>
</tr>
<tr>
<td>R22</td>
<td>2.39x10^{11}</td>
</tr>
<tr>
<td>R21</td>
<td>2.93x10^{11}</td>
</tr>
<tr>
<td>R11</td>
<td>5.22x10^{12}</td>
</tr>
<tr>
<td>R12</td>
<td>3.48x10^{12}</td>
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<tr>
<td>IN31</td>
<td>3.60x10^{11}</td>
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<tr>
<td>IN22</td>
<td>4.59x10^{12}</td>
</tr>
<tr>
<td>IN21</td>
<td>4.95x10^{12}</td>
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<td>IN11</td>
<td>3.45x10^{12}</td>
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<tr>
<td>IN12</td>
<td>2.99x10^{12}</td>
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<tr>
<td>BTR1</td>
<td>2.22x10^{12}</td>
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<tr>
<td>BTR2</td>
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<td>BTT1</td>
<td>4.20x10^{11}</td>
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<tr>
<td>BTT2</td>
<td>4.20x10^{11}</td>
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<tr>
<td>BTT3</td>
<td>3.50x10^{11}</td>
</tr>
<tr>
<td>BTT4</td>
<td>3.50x10^{11}</td>
</tr>
</tbody>
</table>

**FLUJO PROMEDIO POR CELDA PARA**

**NUCLEO FRESCO DE LA ALTERNATIVA**

**DEL REACTOR DE 25 MW.**
4.2 Análisis termohidráulico.

El análisis termohidráulico de un Reactor de Usos Múltiples basado en el concepto MAPLE, se efectuó utilizando el programa de computadora SPORTS-M, propiedad de AECL. Este programa efectúa cálculos de redes de tuberías tanto en estado estacionario como transitorio. En el modelo empleado por el programa están incluidas las ecuaciones de balance de masa, momento y energía, además de una ecuación de estado para flujo en dos fases, homogéneo en una sola dimensión. También se incluyen, acoplados a los módulos hidráulicos, un paquete de transferencia de calor y un modelo de conducción radial de calor.

Este programa puede ser usado para modelar circuitos hidráulicos en circuitos cerrado o abierto.

En la Fig. 7 se muestra el diagrama esquemático de flujo utilizado para el análisis termohidráulico del Reactor de Usos Múltiples tipo MAPLE para México.

La Fig. 8 muestra el esquema de la nodalización utilizada en el programa SPORTS-M.

Para obtener una estimación del comportamiento termohidráulico del núcleo, se efectuaron algunos estudios de sensibilidad con el programa para asegurar una refrigeración adecuada del núcleo y los resultados se muestran en la tabla VII.

4.3 Cálculo del blindaje.

Los cálculos del blindaje para las alternativas de 14 MW y 25 MW, se efectuaron utilizando el programa de cómputo ONEDANT, utilizando geometría esférica. Las razones de dosis obtenidas para los espesores de blindaje propuestos resultaron por debajo de los límites máximos permisibles.

Se realizaron además, los cálculos de las razones de deposición de calor en el blindaje, encontrándose que estos valores se encuentran dentro de un rango aceptable.
Fig. 7

DIAGRAMA DE FLUJO USADO PARA EL ANALISIS TÉRMICO HIDRAULICO
ESQUEMA DE LA NODALIZACIÓN USADA PARA EL ANÁLISIS TERMOHIDRAULICO
### TABLA VII
**ESPECIFICACIONES DEL SISTEMA DE ENFRIAMIENTO PRIMARIO DE REACTOR.**

<table>
<thead>
<tr>
<th>OPCIÓN</th>
<th>25 MW</th>
<th>14 MW</th>
</tr>
</thead>
<tbody>
<tr>
<td><strong>NUCLEO</strong></td>
<td></td>
<td></td>
</tr>
<tr>
<td>Gasto másico</td>
<td>446.5 Kg/s</td>
<td>307 Kg/s</td>
</tr>
<tr>
<td>Temperatura de entrada</td>
<td>35°C</td>
<td>35°C</td>
</tr>
<tr>
<td>Temperatura de salida</td>
<td>48.4°C</td>
<td>46.4°C</td>
</tr>
<tr>
<td>Caída de presión</td>
<td>148.5 KPa</td>
<td>260 KPa</td>
</tr>
<tr>
<td>Razón máxima ONB</td>
<td>0.87</td>
<td>0.6</td>
</tr>
<tr>
<td><strong>INTERCAMBIADOR DE CALOR</strong></td>
<td></td>
<td></td>
</tr>
<tr>
<td>Razón de remoción de calor</td>
<td>25 MW</td>
<td>14 MW</td>
</tr>
<tr>
<td>Gasto másico</td>
<td>478 Kg/s</td>
<td>343 Kg/s</td>
</tr>
<tr>
<td>Caída de presión</td>
<td>1733 KPa</td>
<td>747 KPa</td>
</tr>
<tr>
<td><strong>BOMBAS</strong></td>
<td></td>
<td></td>
</tr>
<tr>
<td>Número de bombas</td>
<td>1</td>
<td>1</td>
</tr>
<tr>
<td>Gasto másico</td>
<td>478 Kg/s</td>
<td>343 Kg/s</td>
</tr>
<tr>
<td>Incremento de presión</td>
<td>2038 KPa</td>
<td>1069 KPa</td>
</tr>
<tr>
<td>Carga de la bomba</td>
<td>210 m</td>
<td>110 m</td>
</tr>
</tbody>
</table>
5. CONCLUSIONES.

La experiencia ganada al realizar los cálculos neutrónicos, termohidráulicos y del blindaje para las alternativas propuestas, ha sido muy importante, así como la asesoría brindada por AECL y Atomenergexport, ya que ha servido de base para establecer una metodología para el análisis del diseño de otros prototipos de reactores de usos múltiples. Ha servido también para consolidar los grupos de trabajo del ININ en México.

En base a la experiencia obtenida en estas etapas del proyecto, un grupo de trabajo del ININ se ha dedicado a desarrollar un programa de computadora para efectuar estudios en estado estacionario, sobre las caídas de presión y los gastos móviles en cada canal del núcleo, así como la estimación de los márgenes térmicos y otros grupos de trabajo se están dedicando a la tarea de adaptación y validación de programas de reciente adquisición o ya existentes en el Instituto.

6. REFERENCIAS.


THE PRESENT STATUS AND THE PROSPECT OF CHINA RESEARCH REACTORS

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CHINA ZHONGYUAN ENGINEERING CORPORATION
CAO YIZHENG
INSTITUTE OF ATOMIC ENERGY

BEIJING
CHINA
ABSTRACT

A total of 100 reactor operation years' experience of research reactors has now been obtained in China. The type and principal parameters of China research reactors and their operating status are briefly introduced in this paper. China research reactors have been playing an important role on nuclear power and nuclear weapon development, industrial and agricultural production, medicine, basic and applied science research and environmental protection, etc. The utilization scale, benefit and achievement will be given.

There is a good safety record in these reactors operation. A general safety review is discussed. The important incidents and accidents happened during a hundred reactor operating years are described and analysed. China has got the capability to develop any type of research reactor. The prospective projects are briefly introduced too.

THE PRESENT STATUS AND THE PROSPECT OF CHINA RESEARCH REACTORS

1. China research reactor's type, main parameters and operation situation

China research reactor has an operation experience of over 100 reactor years. Its type and main parameters are listed in Table I and its operation situation is introduced as follows.
1.1. HWRR (1) (2)

HWRR was China's earliest research reactor imported from USSR. It took 2% U-235 metal uranium as fuel, heavy water as moderator and coolant, graphite as reflector. Its nominal power was 7 MW, strengthened power was 10 MW, and the maximum thermal flux was $1.2 \times 10^{14} \text{n/cm}^2\text{s}$.

HWRR's historical development might be divided roughly into three periods. The first period was from 1958 to 1978 with assimilating, digesting, improving and safe operating as its main task. In this period, the Chinese technicians made improvement in every system of the reactor and in reactor character. The reactor's utilization was enlarged and meanwhile a great number of reactor engineering scientists and technicians were trained.

The second period was from 1979 to 1982 with the aim of reactor's reconstruction. In the first phase of the reconstruction project, stress was laid on core replacement and heavy water system reform, and these works was completed on June of 1980. The second phase was implemented from 1980 to 1982, mainly to renew the measuring instruments and introduce the process control computer. The data measured after reconstruction show that the reactor's technical index have totally reached the reconstruction design requirement, the major characteristic parameters are showed in Table 1.

After 1982 began the third period of the reactor's further application. During this period, HWRR completed fuel assembly test for Qinshan Nuclear Plant, let NTD silicon be produced commercially and cold neutron source be installed and put into operation. Now thirty years have gone, but HWRR
still plays its important role in the fields of China’s nuclear technology. At present HWRR works 10 to 20 days each month in the light of needs.

1.2. SPR

There are three swimming pool reactors in China now. SPR-IAE (3) is a reactor put into operation on Oct. of 1964, with the construction aim of testing fuel and materials. The reactor has been improved greatly during the past 23 years. Its rated power is gradually increased to 3.5MW and it gets various accessory facilities. The reactor then becomes a multipurpose reactor that works on reactor physics experiment, neutron radiograph, radioisotopes production and NTD silicon irradiation apart from its testing work. Now the reactor works about 10 days each month.

SPR-QHU Core No.1’s original design power is 2 MW. At present natural convection cooling is applied and its working power is only 50KW. It is mainly for making shielding material experiment and other irradiation test. In 1975, a 2.8MW core No.2 was installed opposite the core NO.1 for comprehensive utilization. The two cores share one water pool. In 1983 a low temperature heating experiment was made on core No.2 and this provides data and experiences for developing the low temperature heating reactor.

SPR(4)-IPC is China’s third swimming pool reactor. Apart from completing special tasks assigned by the state, it also works on monocrystal silicon doping, neutron activation analysis, material irradiation test, neutron radiograph, etc. And remarkable success is specially achieved in neutron radiograph’s practical application.

1.3 HFETR (5)(6)
HFETR is mainly used in the irradiation character research of power reactor's fuel and material and the production of radioisotopes. High power operation began in 1980, and a total of 16 loadings had been operated by the end of 1987, 309 fuel assemblies and 40 follower assemblies were loaded. The accumulative power is 20560MWD. From first to sixteenth loading the fuel assemblies for each loading were increased from 25 assemblies to 57 assemblies, power from 40MW to 90MW. Since the refueling pattern of zoned replacement is applied, the unloading element' average burn-up is 42%.

In HFETR's main building, there are places for installing various experiment facilities. It is possible for constructing 9 test loops such as water cooling, gas cooling and sodium cooling test loop. But due to some reasons there is only a high temperature and pressure water loop under construction by now. The reactor is not fully utilized and operated only 8 to 10 days each month for material irradiation research and irradiation products.

1.4. MNSR (7)(8)

MNSR is a safe, compact, economic and efficient tool which is applicable for neutron activation analysis, radioisotope production, education and other purposed. It is specially suitable for use of hospital, university and research institutes. In 1984, a prototype reactor was constructed and several years operation shows the reactor has reached its design parameters with a good properties and safety. In recent years, two commercial reactors were constructed in Shenzhen and Shandong in succession.

2. China research reactor's application

China research reactor has played an important role in
developing nuclear power and weapon, industrial and agricultural production, medicine science, basic and applied science and research.

2.1. Plutonium reactor and power reactor's fuel element test.

The fuel element test are mainly made on HWRR and SPR-IAE. Since 1965, SPR-IAE has built 4 sets of high temperature and pressure, middle temperature and pressure test loops in core, and a total of ten types' home-made plutonium reactor and power reactor fuel elements have been tested. These tests have made important contribution to the increase of prototype reactor power and fuel burn-up index, the element lifetime prolongation, the optimum manufacture process selection, and the increase of the element qualification rate. At the same time, research has also been made on UO2 pellet's temperature and thermal conductivity measurement. All elements tested are short rod because of size limitation of SPR core.

During the year of 1966 to 1970, a high temperature and pressure test loop was installed in HWRR's central thimble to test China's first nuclear submarine element. It had two loading and gained the expected results. It provided reliable basis for the elements production, and we have tested several kinds of metallic uranium elements in the reactor.

After the reconstruction, a new and high temperature and pressure test loop(9) was set up in HWRR's central thimble (φ120mm) to test the fuel element of Qinshan Nuclear Plant. The test loop parameters are 15.5Mpa, 320℃. The assembly U-235's enrichment is 10%. The assemblies
structure is 3X3-2. Its maximum power is 290KW and the maximum heat flux $1.45 \times 10^6$ Kcal/m$^2$h. The unloading assemblies' mean burn-up is 25,000MWD/T. The test work has made contributions to the elements production of the Qinshan Nuclear Plant.

HFETR also fulfilled some fuel single rod's test inserted directly in the core, such as the UO2 fuel rod containing burnable poison and the ThO2 fuel samples. A new test loop is under the construction.

2.2. Research on irradiated character of the reactor's structure material.

Research works are mainly done in reactor HFETR and SPR, for their neutron energy spectra is harder and their fast neutron flux is higher.

The pressure vessel steel, such as 645 steel, 728 steel, LT21, LT24 aluminium alloy, Al-Li alloy, Zr-Al alloy, cables and paintings etc, were irradiated in reactor SPR-IAE. The alloy material's moving water corrosion test was developed on strong radiation fields.

Material in core irradiation and temperature control irradiation are carried on reactor HFETR. The material being researched are A 508-3 steel, S271 steel, fast reactor steel, B-10 steel, permeated boron stainless steel, Zr-4 etc, and the detective devices and elements such as transducer, new type thermo-couple and Pt detector.

2.3. The production of radioisotops

The man-made radioisotopes' research started in 1958. Along with the development of the research reactor, China's radioisotopes research and production has a considerable scale. There are more than 800 kinds of products in the country, and 150,000 pieces of products are delivered each
year to over 1500 consumers. Most of the radioisotope products are produced by HWRR. The main products are I-131, I-125, Au-198, P-32, Ba-131, C-14, I-192 and Po-210 etc. The radioisotope products produced by HWRR meets the 70% of domestic demand.

SPR-IAE’s main products are Mo-99-Tc-99m, Cr-51, and non carrier P-32 using the characteristic of its low temperature of moderator.

HFETR mainly takes its high excess reactivity as advantage to produce medical, industrial and agricultural intensive cobalt source, Ir-192—the source of defect detector, high specific intensity C-14, Sn-113-In-113M, Mo-99-Tc-99m isotope generator and transplutonium.

2.4. Neutron transmutation doping in monocrystal silicon.

China started to develop NTD technology since 1980. Now five research reactors are taking monocrystal silicon irradiation. The irradiation ability per year is over 40 ppb-tons, its actual irradiation quantity is 10 tons. The doping scope is from $10^{10} \Omega \cdot \text{cm}$ to $10^2 \Omega \cdot \text{cm}$, the doping uniformity, accuracy and annealing technology have reached international level. NTD technology have been basically adopted for The China-made monocrystal silicon used as producing power device, hence the apparent improvement of the device quality have been realized.

2.5. Neutron activation analysis

All the research reactors do neutron activation analysis work, HWRR, MNSR-IAE, SPR-QHU, these three reactors undertake a bigger work quantity. The analysis sample quantity each year are 10,000 to 30,000 most of them are environmental samples, superpure material, mineral samples,
and biological samples. They can analysis over 70 kinds of chemical elements with a sensibility from several hundred ppm to grade pph. meanwhile, the neutron activation analysis has become one of the main analysis methods for quantification of standard reference materials.

12.6. Neutron scattering and neutron diffracting experiments (11)

Neutron scattering research is carried on HWRR's horizontal experimental channels. In recent years, cold source with 0.4 litres, cold guide tube with 30 meter long and a small angle scattering spectrometer are installed in HWRR NO. 4 horizontal experimental channel by China-France cooperation. At cold guide tube exit, the cold neutron beam intensity can reach to $10^6 \text{n/cm}^2\text{sec}$ with a better cold neutron gain. This set of equipment can take into use at the end of this year.

4-circle diffractometers, triple-axis spectrometer and flying time spectrometer are installed on the other horizontal experimental channels mainly for the research work of crystal and magnetic structure, phonon dispersion study, phonons in metal hydrides and superconductors diffusion process, etc.

Besides the above application, we adopt the technology of neutron radiograph, irradiation seed breeding and nuclear pore filter film etc. A gamma irradiation device with a diameter of 180mm using the unloaded element has been installed in the storage pool of the HFETR, and gamma field intensity can reach to $2 \times 10^7 \text{R/h}$. A successful irradiation-crosslinking treatment to 30 tons of polythene has been done in the device.

3. Commentary on the safe operation of China research reactors (12)
3.1. The brief information of safe operation of China research reactor.

China research reactor has an operation experience of over 100 reactor years with a good record for safety operation. The collective dose equivalent per MWY of the reactor's operating staff can be seen in Table II. After HWRR reconstruction, since we substitute metal uranium with UO2 for fuel and replace the bead welding cobalt alloy to the sputter coating with non-cobalt wear-resistant alloy on shaft sleeve and thrust bearing disk of the heavy water pump, since we have improved the reactor operation process, collected sufficient experience in reactor's maintenance work and strengthened the management work of the reactor hall and chamber, the value of collective dose equivalent of releasing unit energy is apparently lower than that before the reconstruction. Compared with such value of reactors outside China, we do believe that China research reactor is in line with them.

The intensity of radioactivity from HWRR to the environment can be seen in Table III. After the reconstruction, since we have added HEPA filter and deiodine filter in the ventilation system of the reactor chamber, the discharging amount of radioactive iodine and aerosol to the environment have been dropped apparently. Since the discharging amount of radioactivity from the other China research reactors are far less than that of HWRR, we will not give the exact value in this paper. The amount of radioactivity to the environment from the China research reactors are less than the permissible limitation, it never cause any sensible influence to inhabitants around and we have not found any case of radioactive sufferer from the
reactor operators whom we give a thorough health checking regularly.

3.2. The incidents and accidents of China research reactors happened in the period of 1958-1988 and analysis.

From Table IV we can see the main incidents and accidents of China research reactors happened within the past three decades. For each incident or accident, we give a brief status, causes and radiological consequences. There are two common characters for all these incidents and accidents listed in Table IV.

a. The main causes for the incidents (accidents) or extending radiological consequence are due to the misjudgement or maloperation. Over 70% of incidents or accidents listed in Table IV are man induced events.

b. Most of the incidents or accidents which bring about radiological consequence are due to the failure of the test loop, isotopic target, experimental source and sample etc. Incident or accident caused by the defect or failure of the equipment of the reactor is rare.

We believe that the most effective improvement to the safety operation of research reactor is to train a qualified operating staff, improve their ability of coping with emergency and strengthen the management to the experimental items of the research reactor including a strict safety review to every device and sample which is going to be sent into the reactor.

3.3. The publication and execution of The Safety Regulation to China Research Reactor.

In Oct. 1986, the State Council of the People's Republic of China issued "The PRC Civil Nuclear Facilities Regulatory Supervision Rule" and the Nuclear Safety Bureau
gave a detailed and definite interpretation to the rule for its execution. China research reactors are under the direct jurisdiction of the above two documentations.

In Aug. 1988, the Nuclear Safety Bureau of the People's Republic of China issued "Code for Safe Operation of Research Reactors and Critical Assemblies". The other safety guides for the China research reactor are being prepared and discussed and three of them are going to issue in the near future.

Now all the research reactors and the critical devices in China are under the supervision of the Nuclear Safety Bureau of the People's Republic of China. Safety analysis report for HWRR, HFETR and SPR were submitted, and the adding safety facilities in order to improve the safety have almost completed according to the schedule. The Nuclear Safety Bureau will issue operation licences for them. Concerning MNSR, pulse reactor and low temperature heating reactor which are under the construction, they will accept the safety review from the Nuclear Safety Bureau and are waiting for approvement of fuel loading and operation licences.

The publication and execution of The Safety Regulation to China Research Reactors will further raise the safety operation level of the research reactor.

4. The developing prospect of China Research Reactors

4.1. Full utilization and reformation to the completed reactors in China

From our statement in Chapter 1 we can see that the utilization efficiency of China research reactor are not so ideal, the operating time for each reactor is only 10 days per month in average reflects a rather low operating
efficiency and the utilization efficiency of experimental thimbles is expecting for further raising, on one hand, we should further exploit application items of the research reactors and attract more users, on the other hand, we must strengthen the reformation of our research reactors, add the auxiliary facilities so as to enlarge the capability of comprehensive utilization and decrease the using cost.

We are planning to build MNSR in the coastal area for activation analysis, education and training.

In Southwest China Reactor Research Center, we will build a low power reactor using the unloaded element from HFETR. The reactor will load 32 assemblies of HFETR unloading element of forty percent of burn-up. The reactor power is 5MW. The generating energy for each loading is 340 MWD and element burn-up of final stage is 45%.

This low power reactor can be used for the irradiation of monocrystal silicon and production of radioisotopes like fission Mo-Tc. The reactor will be operating in 1990 according to the schedule and it will take a full use of HFETR element and realize the aim of economization, tapping the potentialities and low irradiation cost.

4.2. The development of new type reactors
   a. Pulse reactor (13)

To develop the reactor with possessed inherent safety, A prototype pulse reactor designed by IRERD is under the construction, this pulse reactor will be similar to TRIGA-II and the main parameters will be as follows.

1. Reactor core
   Diameter: 540mm Height: 390mm
   Fuel loading: 3.45kg U-235
   Core total life time: 150MWD
2. Fuel element
- Diameter: 37.2mm
- Ingredient: UZrH1.6
- Concentration of U-235: 20%
- Uranium content: 8.5Wt%

3. Stationary condition
- Rated power: 1000KW $\ \bar{\Phi}_{th}; \ 1.4 \times 10^{13} \text{n/cm}^2 \text{s} \ \bar{\Phi}; \ 2.4 \times 10^{13} \text{n/cm}^2 \text{s}$
- Mean power density: 15.8W/cm$^3$
- Max. power non-uniform coefficient: 2.31

4. Pulse condition
- Max. pulse reactivity add: $3\$$
- Pulse peak power: 4900MW
- Pulse peak flux: $6 \times 10^6 \text{n/cm}^2 \text{s}$
- Pulse half-width: 7ms
- Generate energy per pulse: 38.8MWS

We are expecting to complete the prototype reactor in the coming year and carry out the verifying experiments of physics, thermodynamics, control and nuclear measurement, under the stationary and pulse conditions. After the completion of safety analysis and optimizing design, we are going to build a UHZrPR-A pulse demonstrative reactor and realize the commercial operation. Further exploitation to the low temperature heating reactor UHZrPR-B will be done by adopting the technology and experience from UHZrPR-A reactor.

b. Low temperature heating reactor (14)

We might see the developing future of low temperature heating reactor in the areas in China where are lack of energy sources. Engineers and technical staff in the Nuclear Energy Institute of Qinhua University are installing and commissioning a low temperature heating
prototype reactor. Prof. Lui Yingzhong and his Colleague have obtained the inventive certificate from the patent Office of the People's Republic of China for the deep water type heating reactor.

With its own qualified design units and construction teams, China has collected a considerable experience in developing, making and operating the research reactor, and is able to design and build any type of research reactor and its fuel element. We are willing to offer the Chinese technology and construction experience in the nuclear fields and promote the international cooperation.
Table 1 Type and main parameters of China research reactors

<table>
<thead>
<tr>
<th>Reactor Name</th>
<th>Critical Date</th>
<th>Reactor Address</th>
<th>Reactor type, fuel type, ( \text{U}^{235} ), Concentration, moderator, coolant, reflector.</th>
<th>Power</th>
<th>The ( \phi_{\text{th}} ) max n/cm(^2) sec.</th>
</tr>
</thead>
<tbody>
<tr>
<td>HWRR</td>
<td>1958</td>
<td>IAE Beijing</td>
<td>Heavy Water ( \text{U}\text{O}_2, 3% \text{U}^{235}, \text{D}_2\text{O}, \text{D}_2\text{O}, \text{graphite.} )</td>
<td>15</td>
<td>( 2.4 \times 10^{14} )</td>
</tr>
<tr>
<td>SPR --IAE</td>
<td>1964</td>
<td>IAE Beijing</td>
<td>Swimming Pool ( \text{U}\text{O}_2, 87% \text{wt} \text{U}\text{O}_2 + 13% \text{wt} \text{Mg}, 10% \text{U}^{235}, \text{H}_2\text{O}, \text{H}_2\text{O}, \text{Be} + \text{C} )</td>
<td>3.5</td>
<td>( 4.0 \times 10^{13} )</td>
</tr>
<tr>
<td>SPR --QHU</td>
<td>1964</td>
<td>INET Qinghua University</td>
<td>Swimming Pool ( \text{U}\text{O}_2, 87% \text{wt} \text{U}\text{O}_2 + 13% \text{wt} \text{Mg}, 10% \text{U}^{235}, \text{H}_2\text{O}, \text{H}_2\text{O}, \text{C} )</td>
<td>2.8</td>
<td>( 3.5 \times 10^{13} )</td>
</tr>
<tr>
<td>SPR --IPC</td>
<td>1979</td>
<td>Southwest IPC</td>
<td>Swimming Pool ( \text{U}\text{O}_2, 87% \text{wt} \text{U}\text{O}_2 + 13% \text{wt} \text{Mg}, 10% \text{U}^{235}, \text{H}_2\text{O}, \text{H}_2\text{O}, \text{C} )</td>
<td>2.0</td>
<td>( 3.0 \times 10^{13} )</td>
</tr>
<tr>
<td>HFETR</td>
<td>1979</td>
<td>Southwest IRERD</td>
<td>Tank ( \text{UAl}_4, 25.4% \text{wt} \text{U}, 90% \text{U}^{235}, \text{H}_2\text{O}, \text{H}_2\text{O}, \text{Be} + \text{Al} )</td>
<td>125</td>
<td>( 6.2 \times 10^{14} )</td>
</tr>
<tr>
<td>MNSR --IAE</td>
<td>1984</td>
<td>IEA Beijing</td>
<td>Tank in Pool ( \text{UAl}_4, 26.7% \text{wt} \text{U}, 90% \text{U}^{235}, \text{H}_2\text{O}, \text{H}_2\text{O}, \text{Be} )</td>
<td>27KW</td>
<td>( 1 \times 10^{12} )</td>
</tr>
<tr>
<td>MNSR --SZ</td>
<td>1988</td>
<td>Shenzhen University</td>
<td>Tank in Pool ( \text{UAl}_4, 26.7% \text{wt} \text{U}, 90% \text{U}^{235}, \text{H}_2\text{O}, \text{H}_2\text{O}, \text{Be} )</td>
<td>27KW</td>
<td>( 1 \times 10^{12} )</td>
</tr>
<tr>
<td>MNSR --SD</td>
<td>1989</td>
<td>Shandong Geology Bureau</td>
<td>Tank in Pool ( \text{UAl}_4, 26.7% \text{wt} \text{U}, 90% \text{U}^{235}, \text{H}_2\text{O}, \text{H}_2\text{O}, \text{Be} )</td>
<td>27KW</td>
<td>( 1 \times 10^{12} )</td>
</tr>
</tbody>
</table>
Table 2.  Collective dose equivalent of personnel per MWY (Person - Sv)

<table>
<thead>
<tr>
<th>HWRR</th>
<th>Before reconstruction</th>
<th>HWRR</th>
<th>After reconstruction</th>
<th>HFETR</th>
<th>SPR - IAE, QHU, IPC</th>
</tr>
</thead>
<tbody>
<tr>
<td></td>
<td>3.46 x 10^-2</td>
<td>7.9  x 10^-2</td>
<td>4.6 x 10^-2</td>
<td>(1 -- 4) x 10^-2</td>
<td></td>
</tr>
</tbody>
</table>

Table 3.  HWRR radioactivity released to environment (Ci/yr)

<table>
<thead>
<tr>
<th></th>
<th>Through exhaust stack</th>
<th>Through drain of effluent</th>
</tr>
</thead>
<tbody>
<tr>
<td><strong>Before</strong></td>
<td>41Ar 2700</td>
<td>3H 40</td>
</tr>
<tr>
<td><strong>After</strong></td>
<td>3980</td>
<td>54</td>
</tr>
</tbody>
</table>

* Integral released quantity of IAE.
Table 4. The main incidents /accidents occurred in 1958 - 1988

<table>
<thead>
<tr>
<th>Reactor</th>
<th>Date</th>
<th>Status</th>
<th>Cause</th>
<th>Radiological consequences</th>
</tr>
</thead>
<tbody>
<tr>
<td>HWRR -1</td>
<td>1958 -1986</td>
<td>Heavy water leakage from reactor system to processing rooms, 9 times.</td>
<td>Cracking of isolation shell of Heavy water pump and instrument valves; welding seam crack of instrument connecting pipes.</td>
<td>Release $^3\text{H}$ of 0.05 - 40 Ci.</td>
</tr>
<tr>
<td>HWRR -2</td>
<td>1959 -1987</td>
<td>Period over short, 5 times.</td>
<td>Insufficient valuation for experiment reactivity and maloperation.</td>
<td>/</td>
</tr>
<tr>
<td>HWRR -3</td>
<td>1961 -1987</td>
<td>Breakdown of process tube, 10 times.</td>
<td>Mechanical damage from flow vibration.</td>
<td>/</td>
</tr>
<tr>
<td>HWRR -4</td>
<td>1970.9</td>
<td>Fall of fuel element on the bottom of the reactor vessel.</td>
<td>Break down at the bottom of the process tube during handling.</td>
<td>Integral dose equivalent: -5 person. rem</td>
</tr>
<tr>
<td>HWRR -5</td>
<td>1969 -1977</td>
<td>Failure of fuel elements, 9 times. 1 in which meltdown.</td>
<td>Malfunction of non-destructive test in production, maloperation.</td>
<td>2 months for dispos. integral dose equivalent: -12 person. rem</td>
</tr>
<tr>
<td>HWRR -6</td>
<td>1962</td>
<td>Blockage of one safety rod.</td>
<td>Rust of iron cable.</td>
<td>/</td>
</tr>
<tr>
<td>HWRR -7</td>
<td>1981</td>
<td>Fall of regulation rod.</td>
<td>Break down of cable from mechanical wear.</td>
<td>/</td>
</tr>
<tr>
<td>HWRR -8</td>
<td>1974.7</td>
<td>Heavy water leakage to heat exchanger shell cavity.</td>
<td>Cracking at welding seam of exchanger vessel.</td>
<td>/</td>
</tr>
<tr>
<td>Reactor</td>
<td>Date</td>
<td>Status</td>
<td>Cause</td>
<td>Radiological consequences</td>
</tr>
<tr>
<td>---------</td>
<td>----------</td>
<td>---------------------------------------------</td>
<td>--------------------------------------------</td>
<td>------------------------------------------</td>
</tr>
<tr>
<td>HWRR 9</td>
<td>1985.12</td>
<td>Heavy water leakage from heat exchanger to secondary water.</td>
<td>Welding cracking of tube end at tube plate.</td>
<td>7 days for disposal</td>
</tr>
<tr>
<td>HWRR 10</td>
<td>1964.8</td>
<td>Spurt of ThO₂ powder sample.</td>
<td>Burst of glass capsule from radiation.</td>
<td>Release ThO₂ of 1 mci to Reactor cabinet.</td>
</tr>
<tr>
<td>HWRR 11</td>
<td>1971.12</td>
<td>Leakage of ammonium dichromate.</td>
<td>Burst of irradiation capsule from radiation decomposition.</td>
<td>5 persons are contaminated.</td>
</tr>
<tr>
<td>HWRR 13</td>
<td>1981-1986</td>
<td>Leakage of sample I.</td>
<td>Failure of irradiation capsule.</td>
<td>Release ¹³¹I of 0.05 - 30 mci.</td>
</tr>
<tr>
<td>HWRR 14</td>
<td>1975.1</td>
<td>Leakage of sample Hg to the reactor vessel and heavy water loop.</td>
<td>Burst of quartz capsule, damage of irradiation facility and experimental channel from corrosion mercuride.</td>
<td>Release ²⁰⁵Hg of 53mci to atmosphere.</td>
</tr>
<tr>
<td>Reactor</td>
<td>Date</td>
<td>Status</td>
<td>Cause</td>
<td>Radiological consequences</td>
</tr>
<tr>
<td>-----------</td>
<td>--------</td>
<td>---------------------------------------------</td>
<td>-------------------------------------------------------------------------------------------</td>
<td>-------------------------------------------------</td>
</tr>
<tr>
<td>HWRR -16</td>
<td>1977.5</td>
<td>Meltdown of test fuel element.</td>
<td>Fabrication defects of test loop facilities and transgressing the operating instruction.</td>
<td>Release $^{133}$Xe of 1650 Ci and $^{131}$I of 20 mCi to atmosphere.</td>
</tr>
<tr>
<td>HWRR -17</td>
<td>1964.3</td>
<td>PuO$_2$ micro-particles spillage from sample.</td>
<td>Maloperation of experimental personnel.</td>
<td>Contaminated area: 1500 m$^2$. Radioactivity: 1.1 mCi.</td>
</tr>
<tr>
<td>HWRR -18</td>
<td>1964.4</td>
<td>Radium microparticles spillage from sample.</td>
<td>Transgressing the operating instruction.</td>
<td>Contaminated area: 100 m$^2$. Radioactivity intensity: 1 mCi.</td>
</tr>
<tr>
<td>HWRR -20</td>
<td>1980.7</td>
<td>Spread of Heavy water with $^3$H.</td>
<td>Breakdown of rubber filling tube during critical experiment.</td>
<td>Release $^3$H of 1 Ci to atmosphere.</td>
</tr>
<tr>
<td>SPR-IAE -21</td>
<td>1965.4</td>
<td>Shaft break down of test loop pump.</td>
<td>Defect of material</td>
<td>/</td>
</tr>
<tr>
<td>SPR-IAE -22</td>
<td>1967.7</td>
<td>Burst of test loop heating section.</td>
<td>Maloperation.</td>
<td>/</td>
</tr>
<tr>
<td>SPR-IAE -23</td>
<td>1969.6</td>
<td>Burn down of pump packing in secondary loop.</td>
<td>Maloperation.</td>
<td>/</td>
</tr>
<tr>
<td>SPR-IAE -24</td>
<td>1969.10</td>
<td>Meltdown of test fuel element.</td>
<td>Test loop lose flow</td>
<td>Release inert gas of hundreds Ci, Aerosol of 1.5 mCi to atmosphere.</td>
</tr>
<tr>
<td>Reactor</td>
<td>Date</td>
<td>Status</td>
<td>Cause</td>
<td>Radiological consequences</td>
</tr>
<tr>
<td>----------</td>
<td>-----------</td>
<td>--------------------------------------------------</td>
<td>-------------------------------------------------</td>
<td>----------------------------</td>
</tr>
<tr>
<td>SPR-IAE -25</td>
<td>1970.10</td>
<td>Abnormal oscillation of reactor power.</td>
<td>Boiling of pool water penetrated into the central experimental channel.</td>
<td>/</td>
</tr>
<tr>
<td>HFETR -29</td>
<td>1981.</td>
<td>Interruption of two independent outside electrical power supply.</td>
<td>Sounder stroke on large area.</td>
<td>/</td>
</tr>
<tr>
<td>HFETR -30</td>
<td>1983.</td>
<td>Overpower operation.</td>
<td>Maladjusted power setting device.</td>
<td>/</td>
</tr>
</tbody>
</table>
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RATIONALIZATION AND FUTURE PLANNING FOR AECL'S RESEARCH REACTOR CAPABILITY

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RATIONALIZATION AND FUTURE PLANNING FOR AECL'S RESEARCH REACTOR CAPABILITY.

ABSTRACT

AECL's research reactor capability has played a crucial role in the development of Canada's nuclear program. All essential concepts for the CANDU reactors were developed and tested in the NRX and NRU reactors, and in parallel, important contributions to basic physics were made. The technical feasibility of advanced fuel cycles and of the organic-cooled option for CANDU reactors were also demonstrated in the two reactors and the WR-1 reactor. In addition, an important and growing radio-isotope production industry was established and marketed on a world-wide basis.

In 1984, however, it was recognized that a review and rationalization of the research reactor capability was required. The commercial success of the CANDU reactor system had reduced the scope and size of the required development program. Limited research and development funding and competition from other research facilities and programs, required that the scope be reduced to a support basis essential to maintain strategic capability.

Currently, AECL is part-way through this rationalization program and completion should be attained during 1992/93 when the MAPLE reactor is operational and decisions on NRX decommissioning will be made. A companion paper describes some of the unique operational and maintenance problems which have resulted from this program and the solutions which have been developed.

Future planning must recognize the age of the NRU reactor (currently 32 years) and the need to plan for eventual replacement. Strategy is being developed and supporting studies include a full technical assessment of the NRU reactor and the required age-related upgrading program, evaluation of the performance characteristics and costs of potential future replacement reactors, particularly the advanced MAPLE concept, and opportunities for international co-operation in developing mutually supportive research programs.

The growth and maturing of the nuclear industry has not reduced the need for a strong and flexible research reactor capability. The Canadian program has been fortunate in developing the NRX/NRU reactor combination, and careful planning will be required to maintain this strength for future application.
1. INTRODUCTION

This paper reviews the current planning and studies being undertaken to secure a future research reactor capability for Atomic Energy of Canada Limited (AECL). This capability also provides an important resource for the Canadian nuclear industry and the University research community.

AECL is currently mid-way through a Research Reactor Rationalization Program which was implemented early in 1985. This program will be completed during the 1992/93 fiscal year and will define AECL's research reactor capability during the 1990's. Current planning and studies are part of a program which will provide the future capability for the early decades of the next century.

2. RESEARCH REACTOR RATIONALIZATION PROGRAM

In the early 1980's AECL owned and operated two zero-power critical facilities (PTR, ZED-2), three prototype power reactors (NPD, Douglas Point, Gentilly-1) and three research reactors (NRX, NRU, WR-1). Table I below gives the characteristics of the three research reactors.

<table>
<thead>
<tr>
<th>Reactor</th>
<th>Power (MWt)</th>
<th>Coolant</th>
<th>Moderator</th>
<th>Fuel</th>
<th>First Power</th>
</tr>
</thead>
<tbody>
<tr>
<td>NRX</td>
<td>42</td>
<td>LW</td>
<td>HW</td>
<td>HEU-A1</td>
<td>1947/07</td>
</tr>
<tr>
<td>NRU</td>
<td>135</td>
<td>HW</td>
<td>HW</td>
<td>HEU-A1</td>
<td>1957/11</td>
</tr>
<tr>
<td>WR-1</td>
<td>60</td>
<td>ORG</td>
<td>HW</td>
<td>UC, 1.8%</td>
<td>1965/11</td>
</tr>
</tbody>
</table>

LW-Light Water, HW-Heavy Water, ORG-Organic Coolant

Since that time, all three prototype power reactors have been shut down and decommissioned. The two critical facilities are still in operation, but the research reactor capability has been reduced as part of a rationalization program.
2.1 Rationalization Program

Development of the program started in 1984, when it was recognized that a review and rationalization of the research reactor capability was required. There were a number of factors important in shaping the program.

1. The successful operating and maintenance record of the CANDU reactor system had reduced the scope and size of the development program required to support the current design of power reactor. In parallel, the slowdown in the world-wide nuclear construction programs had reduced the need for and pace of development of advanced types of reactor and fuel cycles. Consequently, the facilities available in the three reactors were not being utilized to full capability.

2. Funding for fission reactor research and development had peaked and was beginning to decline. It could be foreseen that the operating and maintenance costs for the reactors and other research facilities under construction would consume an increasing fraction of the available funds, and impact adversely on the amount available for direct research and development activities. Consequently, while maintaining an adequate research reactor capability, the associated costs must be reduced.

3. An important and growing radio-isotope production industry was being established to serve a world-wide market. This activity required that at least two reactors were available to provide short-lived isotopes (one as primary producer, and the other reactor as backup to cover primary reactor outages).

4. A new type of multipurpose research reactor was under development. This MAPLE reactor (Multipurpose Applied Physics Lattice Experimental) combined the advantages of a pool-type compact core associated with light-water cooling, with a large volume of irradiation space resulting from the use of a heavy water radial reflector. (This reactor is described in a companion paper [1]). Construction of an initial reactor could meet some of AECL's future needs, as well as providing a demonstration facility which may interest other organizations who were also replacing or expanding their research capability.
The result of the studies was a rationalization program with four components;

- mothball the WR-1 reactor (with the potential to restart or eventually decommission),
- maintain the NRU reactor as the prime research reactor capability,
- construct a MAPLE reactor to demonstration the technology and provide the prime capability for production of short-lived isotopes,
- maintain the NRX reactor in a hot standby mode to provide backup isotope production capability while the MAPLE reactor was being constructed and then decommission.

2.2 Current Status

Currently AECL is part-way through this program and completion should be attained during the fiscal year 1992/93.

- The WR-1 reactor was shut down during 1985, and a recent decision has been made to decommission the facility.

- NRU is operating at high operating efficiency, providing research and development services as well as being the prime isotope producer.

- Design of the MAPLE reactor is more than 75% complete and procurement of the major components is underway. Major construction will start in spring of next year. The project has been the subject of two major reviews (1987 and recently in 1989) and a recent decision of the Board of Directors has confirmed the decision to complete the program.

- The NRX reactor is being maintained in a hot, standby condition to provide backup isotope production capability. A companion paper [2] describes the operational methods which have been developed to deal with the problems of an aging reactor and maintain low operating costs.

Overall, the program is meeting the strategy and objectives developed in 1984. This program and other steps have resulted in a reduction in operating costs of approximately $16 million [2] and they are now 58% (in constant dollars) of the 1985 level.
3. FUTURE REQUIREMENTS

The impact of the rationalization program has been to reduce the AECL research reactor capability from three facilities to one. The capabilities of the NRU reactor will satisfy most of the needs of the pure and applied research community in the 1990's together with support for other associated commercial activities.

The reactor, however, is aging (over 32 years old) and will require future repair and refurbishment to maintain safe and efficient operation. Future upgrading will also be required to meet the changing and more demanding needs of the researchers, as well as keep abreast of current licensing requirements. While NRU can meet AECL's requirement during the 1990's, a major issue is whether the reactor is a suitable vehicle on which to plan AECL's capability for the early decades of the next century. Part of the answer to this question lies in the expected future role of research reactors in the next century.

3.1 The Environment in 2010

Long-range planning requires that the expected future environment be analyzed to provide direction for both strategy and investment. Currently a 20-year perspective is being developed, and by 2010 it is expected that there will be a strong resurgence in the pace of nuclear power installation and use. This will be fuelled, not only by economic considerations, but also by environmental concerns about alternative energy sources, and the perception that fission power is a long-term sustainable technology.

Arising from this environment, it is expected that AECL's activities will be influenced by three dominant features,

- a continuing strong nuclear technology focus,
- a strong associated pure research program, and
- commercial exploitation of the technologies developed.

In turn, these features will require that the researchers have access to major facilities with world-class characteristics and designed to be beneficial to a broad range of R&D programs ranging from pure scientific research to applied R&D.
3.2 Research Reactor Characteristics

With a continuing strong focus on fission power, it is expected that future facilities will require capabilities similar to those available today. The needs can be grouped into seven major areas:

- the ability to provide tailored beams of neutrons for pure research work in condensed matter physics and also commercial applications (e.g. neutron radiography),

- facilities to investigate the behaviour of new materials in the Advanced CANDU nuclear environment (e.g. to study corrosion, deuterium uptake, water chemistry, material fracture properties and deformation),

- facilities in which to test new advanced fuels and their behaviour under power manoeuvering conditions,

- facilities in which to test new components (e.g. pressure tubes),

- facilities in which to test new fusion breeder blanket materials,

- facilities to test fuel behaviour under abnormal conditions (e.g. Blowdown Test Facility [4]),

- facilities to produce radio-isotopes.

Many of these needs can be satisfied by similar research reactor capabilities (e.g. high fast and thermal neutron fluxes) but can also place differing emphasis on reactor operating characteristics and schedules.

4. CURRENT PLANNING AND STUDIES

To develop a long-range plan to meet future needs, a number of activities are currently underway.
4.1 Technical and Safety Assessment of NRU

Planning for a major technical and safety assessment of NRU is currently underway. The objectives are to:

- determine what safety improvements are required and to provide a safety justification that will enable the reactor to remain licensable at least to the year 2010,

- establish what refurbishment and upgrading needs to be done, and the associated schedule, to enable continued operation of NRU until at least 2010,

- estimate the associated costs (capital, operating and maintenance).

It is expected that the full assessment will take three years and requires a dedicated effort of approximately 60 person-years. The program, however, is divided into three phases and after phase 1 has been completed (towards the end of 1990) it is expected that sufficient information will have been collected to start comparison with other options (described below) and make decisions whether to proceed with the later phases.

4.2 Evaluation of Potential Replacement Reactors

Alternatives to continuing with the NRU reactor are also being evaluated. In particular, attention is being focused on a 50 MWt advanced MAPLE concept, [3] which builds on the technology being developed by AECL for the current MAPLE-X10 project.

Studies include investigating the ability of the reactor to supply the multipurpose functions described earlier in Section 3. Both capital and operating costs will also be estimated with a target for completion by late 1990. These data, and information on other alternatives will then be compared with similar information being developed by the NRU technical assessment, described above.
4.3 **International Cooperation**

AECL is already utilizing the capabilities of other reactors to meet those needs which cannot be supplied by NRU, particularly ultra-high fast neutron irradiation of materials. Other areas being investigated include,

- joint studies with other countries, who also need replacement reactors, to explore the possibilities of a common development program,

- participation in international groups studying the development of advanced neutron sources,

- studies on enhanced utilization of off-shore reactors or development of mutually supportive reactor programs.

Results of these studies cannot currently be scheduled, but the status will also be factored into the decision-making process.

4.4 **Factors**

The future program will be shaped by a number of factors including the perceived needs, the overall costs and potential for attracting a broad range of funding support, and the opportunities available for international cooperation. Two other factors will also influence the nature of the program.

National review - currently a national review is being conducted by both federal and provincial agencies with a view to rationalization of the Canadian nuclear industry. This is similar to reviews being conducted in other countries. It is expected that the review will confirm the commitment to retention of the CANDU nuclear option and, hence, the need to maintain a strong major facility capability. It may also, however, broaden the range of participants in the decision-making process.

Multi-versus single-purpose facilities - past Canadian perspectives on research reactors have been shaped by the design use of heavy water as a moderator in the core design. Prior reactors have been relatively large both in terms of power output, core length and overall core volumes. This has facilitated the development of multi-purpose use. However, future designs based on the MAPLE concept will have more compact, shorter cores and there is an economic incentive to keep core power as low as possible to minimize fuelling costs. For the future, more emphasis will be placed on evaluating the relative merits of a large multipurpose facility relative to a number of smaller, limited-purpose ones.
5. SUMMARY

AECL is mid-way through a research reactor rationalization program initiated in 1985. The net result has been to effectively reduce the number of operating reactors in support of R&D from 3 to 1. A new reactor, based on the MAPLE concept, is currently being constructed, which could provide the basis for future replacement reactors. Completion of the rationalization program will occur in the early 1990's, and AECL will rely on the NRU reactor to support the R&D program for the rest of this century.

Planning and investigations are currently underway to define the long-range program for AECL's research reactor capability in the early decades of the next century. Options include continuing with the NRU reactor; constructing a replacement reactor based on the Advanced MAPLE concept, and the opportunities provided by international cooperation. Results from these studies should start to become available in late 1990 and the major features of the long-range program should then begin to emerge.

6. REFERENCES


Highlights in KMRR Nuclear Design

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ABSTRACT

The Korea Advanced Energy Research Institute is constructing the KMRR (Korea Multi-purpose Research Reactor) to meet the requirements of the nationwide nuclear programs. The KMRR is an open-tank-in-pool type reactor which uses LEU U_3Si-Al fuel and features an LWR/HWR hybrid configuration that has advantage of easily tailoring experimental facility layout to ensure supply of effective neutron quality. The KAERI established the physics design system including the shielding design specifically useful for the KMRR. Based on the design analyses, the reactor can be operated at a thermal power output of 30 MW to suit applications ranging from material/fuel test to beam utilization, without violating the design limits concerning the fuel integrity, and the radioactivity level in the access area is confined below the design criteria of 1.25 mrem/hr.
1. Introduction

The 30 MWth KMRR, an open-tank-in-pool type multi-purpose research reactor, is being constructed at Daeduk, Korea by KAERI and is expected to achieve its first criticality on July, 1992. In order not only to support the active nuclear power development program, but also to supply intensive neutron source for nationwide R & D program, KAERI conducted a feasibility study, cooperated with Atomic Energy of Canada Limited, to select the reactor type to meet the requirements needed for:

1. Testing of fuels/reactor materials to support the localization of fuel and reactor components for PWR's and CANDU's.
2. Production of radioisotopes including Tc$^{99}$, I$^{131}$, Ir$^{192}$, and Co$^{60}$ for the medical/industrial utilization, and silicon doped by neutron transmutation.
3. Neutron activation analysis of nuclear-grade materials and basic/applied research employing neutron beam.
4. Neutron radiography for scanning experimental power reactor fuel assemblies and for performing non-destructive examinations of materials and components used in nuclear/non-nuclear applications.

Through reviewing the existing and forthcoming research reactor concepts, the preferable KMRR was conceptually determined as described in Sec. 2.

2. The KMRR Core Description

The outstanding design features of the reactor core as shown in Fig. 1 are:

1. LEU Fuel - The fuel comprises LEU(20 w/o U$^{235}$) U$_3$Si dispersed in aluminum matrix; by the coextrusion method the fuel meat is sheathed with aluminum to form 8 finned rod. The active fuel
length is 70 cm. This fuel is assembled into two types of fuel bundles; the driver bundle which contains 36 elements in a closely packed hexagonal array and the shim bundle which consists of 18 elements in a circular arrangement. Each bundle is assembled with a central tie rod for safe locking inside flow tube.

(2) Compact, $H_2O$ cooled and moderated Inner Core - The KMRR grid plate accommodates 23 hexagonal and 8 circular flow tubes. Inside hexagonal flow tubes except 3 sites will be loaded the driver assemblies and the remaining three flow tubes are to be used as irradiation holes (CT, IR1 and IR2). The circular flow tube contains shim assemblies inside and natural hafnium absorber shrouds outside. Hafnium shrouds are inserted into the water annuli outside the circular flow tubes to provide reactivity hold down or to ensure reactor safe shutdown. Among those eight absorbers, four (C/A) are assigned to power control in conjunction with the reactor regulating system, and the other four (S/O) are for immediate reactor shutdown signaled from the reactor protection system. In the center of the inner core, the central flux trap (CT) is furnished so that this site may be powerful for test rig or loop installations. This inner core lattice is an LWR lattice.

(3) $H_2O$ cooled and $D_2O$ moderated Outer Core - Eight circular flow tubes outside the inner core are equipped either with the shim assemblies for enhancing core excess reactivity or with capsules and rigs for irradiation especially requiring epithermal neutrons. This site has characteristics of HWR lattices.

(4) $D_2O$ Reflector - The KMRR reflector tank is a zircaloy vessel (2.0 m diameter by 1.2 m high) containing $D_2O$. It is penetrated by a user specified array of 25 vertical irradiation tubes and 7 horizontal beam tubes. The use of $D_2O$ as the primary reflector provides near-optimal transmission of neutrons from the compact core to the beam ports and the vertical holes for fuel testing, silicon doping, neutron activation analysis, radioisotope production, and cold neutron source housing.
3. Reactor Physics Design

3.1 Physics Design System

Generally, the lattice design has objectives to get fine structure of nuclear properties of a lattice as well as to prepare cell-averaged parameters for core design as a function of temperature, density and fuel burnup. Since the WIMS/D4[1], a multigroup one-dimensional neutron transport code, can treat a cluster type fuel assembly with fine energy groups and its library contains nuclides encountered in fission process and actinide burnup chains, the WIMS/D4 was selected as a basic tool for the KMRR lattice design. The WIMS-KAERI[2], the KAERI version of WIMS/D4, was born through improvement of the WIMS/D4 to be suitable for the KMRR design analysis. The WIMS-KAERI uses a 69-group structure library which contains the conventional microscopic cross sections for nuclides and subsidiary data for burnup calculations. Parts of the cross section in the original WIMS library were updated to satisfy the KMRR physics design considerations. The contents of the updated or augmented data are:

- $^{235}\text{U}$: thermal absorption and fission cross sections ($< 0.3$ eV)
  
epithermal capture-to-fission ratio ($4 \text{ eV} < E < 1.363 \text{ eV}$)
- $^{238}\text{U}$: fast and resonance transport and capture cross sections
  ($> 4.0$ eV)
- $^{239}\text{Pu}$: a scaling factor on thermal fission cross section of 0.99
- addition of data for $^{237}\text{Np}$, $^{239}\text{Np}$, $^{238}\text{Pu}$ and $^{243}\text{Am}$
- addition of data for Hf isotope, i.e., $^{174}\text{Hf}$, $^{176}\text{Hf}$, $^{177}\text{Hf}$, $^{178}\text{Hf}$, $^{179}\text{Hf}$, and $^{180}\text{Hf}$

Reference 3 gives the bases of updating the first three items. $^{237}\text{Np}$, $^{238}\text{Pu}$, and $^{243}\text{Am}$ data were stemmed from ENDF/B-IV. The remaining came from JENDL-2. Since the KMRR fuel is expected to reach about 120 GWD/MTU in the average discharge burnup, which is extremely high compared with that of the existing power reactors, the actinide burnup chain was extended. For validation prior to being
used, WIMS-KAERI and its library data were benchmarked against the internationally recognized criticality experiments such as TRX, ZEEP, MIT and BAPL.[4].

In order to select proper combination among calculation options provided in WIMS, investigated were the effects on the lattice characteristics by different calculation methods such as the transport solution method, the leakage treatment method, the diffusion coefficient option, the energy group, and the mesh[5]. DSN of transport solution method is selected for all lattices except for the control/shutoff lattices which is solved using the collision probability method. The leakage treatment is based on the diagonal corrected diffusion theory with the simple transport diffusion coefficient. The energy group condensation was optimized to 18 group structure. The mesh grid was constructed on the basis of the neutron mean free path.

Since the KMRR core is strongly heterogeneous, particular attention is given to geometry modeling of the lattices. In particular, control rods and beam tubes were emphasized in modeling. In the control rod modeling investigated were the effects by the fuel burnup, the absorber thickness, surrounding materials, and self-degradation as well[6]. Using the two-dimensional neutron transport code TWOTRAN[7], the homogenization method for the control absorber lattice was investigated in depth[8]. The reactivity worth of the control shrouds with a shim assembly inside was also studied with the real geometry using the KENO-IV[9]. For the beam tube modeling, the reactivity effect of the beam tube was checked with the KENO-IV explicit model. The other lattices representing parts of the reactor core were handled with supercell models taking into account the actual spectrum influenced by environment.

Using WIMS-KAERI output, REGAV-K[10] was used for homogenization of group constants over the region-of-interest and WIMPAX for generation of the few-group cross section table in VENTURE[11] input form. The multigroup 3-dimensional neutron
diffusion code VENTURE simulates the horizontal layout of the reactor as a set of the rectangular cells in the X-Y plane and the vertical layout by stacking the planes; it computes the multiplication factor, the detail neutron flux and power distributions, and the reaction rates in each node. Evaluation by sensitivity studies on mesh size and energy group numbers indicates that the current 5 group and 57 x 47 x 20 mesh model is expected to give reasonable accuracy for detail core analysis. In this geometry, one fuel bundle is represented by 4 x 4 meshes in the horizontal section. Shown in Fig.2 is a schematic flow diagram of detailed physics design activities used for the KMRR. Reference[12] gives details of the reactor physics modeling methods.

3.2 Neutronic Performance Characteristics

The primary concern of a research reactor performance is to provide the high quality neutron fluxes in the experimental facility furnished inside, without violating the safety considerations. Table I shows the maximum and average flux levels at various irradiation holes under the design power of 30 MWth. The average values in the Table were estimated over the active fuel length. Generally the flux levels and quality are sufficient to satisfy the design targets. The axial flux distributions are reasonably flat to allow high quality experiments.

In order to confirm that the reactor is inherently safe at the upset and accident conditions, the reactivity coefficients are usually referred. The reference temperatures of the KMRR for the fuel, coolant and D₂O are 150°C, 40°C, and 40°C, respectively. The reference purity level of D₂O is 99.75 w/o. Due to variations of operating conditions, the system temperatures are changed. For the fuel temperature coefficient, the KMRR has negative value due to Doppler effect. For the H₂O coolant and D₂O temperature coefficients, the neutron spectrum shift and the density change are the principal factors in determining the values. The KMRR has negative values for these coefficients. If D₂O is degraded, the
impurity gives less reflection and more absorption effect to the core. The $D_2O$ impurity coefficient is negative in the KMRR. The KMRR void coefficient turns out to be negative which results from the density decrease of the coolant when the coolant boiled off.

Generally a research reactor should have sufficient excess reactivity for experiments and shutdown margin. A sufficient excess reactivity of 69.54 mk is available for control, experiments and burnup compensations. In the KMRR design basis, the shutdown margin is defined as the negative reactivity present when all the control and shutoff rods except the most reactive one are fully inserted to achieve minimum core multiplication. Four control rods have 137.06 mk and the other four shutoff rods 141.81 mk. The reactivity worth of the most reactive rod is 63.66 mk. Therefore the shutdown margin becomes 215.21 mk and is sufficient to shut down the reactor in the upset and accident condition.

To maintain the fuel integrity during operation, the power level shall be controlled not to exceed a limit value, which was established through fuel management and demonstration test in the NRU reactor. So as not to exceed the limits imposed on the fuel centerline temperature, the sheat temperature, and the ONB/DNB margin under normal operation, the linear element ratings should be lower than 95 KW/m, and the permissible maximum channel powers are 1650 KW in driver and 850 KW in shim assemblies. The peak linear element rating is 61.36 KW/m in driver and 79.04 KW/m in shim assemblies, respectively. The maximum channel powers are 1313 KW in driver and 817 KW in shim assemblies, far below design limits.

4. Shielding Design

4.1 Shielding Design System

The main purposes of the radiation shield structures in a nuclear reactor are to protect operating personnel from nuclear
radiation hazard and in some cases to reduce the radiation damage of structure and components. The KMRR shielding design is risk-free access to the facilities for charging and discharging fuels, for conducting various experiments, and for performing process system maintenance. Extensive efforts have been given to develop a reliable design method for beam tube because of their complex geometries and radiation streaming phenomena.

Scope of KMRR shielding design includes (1) identification of radiation source to examine the characteristics of radiation on the various shields and (2) determination of shielding adequacy to meet the design requirements by suitable analysis methods. Fig. 3 shows the outline of design methods adopted for the KMRR shielding design.

The principal nuclear data libraries and computer codes used in the shielding calculations are as follows.

1. DLC-41C/VITAMIN-C[13]: for neutron and gamma-ray fine group cross section,
2. MACKLIB-IV-82[14]: for nuclear response function,
3. ANSI recommended data: for flux-to-dose rate conversion factor,
4. ANISN[15]: for generation of the zone-dependent few-group data from the fine-group data library and/or calculation of the reactor bulk shielding,
5. DOT4.2[16]: for calculation of the reactor bulk shielding, and/or the spatial source term for three-dimensional problems,
6. MORSE-CG[17]: for assessment of the radiation streaming effect,
7. ORIGEN-II[18]: for generation of the radiation source term of spent fuel,
8. QAD-CG[19]: for calculation of the gamma-ray shieldings of auxiliary facilities.
General approach taken in evaluations was simple method using one-dimensional transport code, ANISN, and point kernel method, QAD-CG, with conservatism. However, verifications of these simple calculation relied on multi-dimensional transport and Monte Carlo methods. For the beam tube systems, the rigorous method combining the discrete ordinates method (DOT4.2) and the Monte Carlo method (MORSE-CG) was developed to be able to describe the phenomena of streaming through duct. The brief description of this coupling procedure are as follows.

- step 1: calculation of the spatial source term around the core using DOT4.2,
- step 2: calculation of the duct streaming and angular source term inside the beam tube using MORSE-CG,
- step 3: calculation of the shielding data for shield design in the beam tube using DOT4.2.

4.2 Shielding Adequacy

The activities at the outside of shields are as follows. The dose rates at the outside of biological shield are 0.32 mrem/hr in radial direction of magnetite concrete (3.45 g/cm^3) and 0.0053 mrem/hr in axial direction of water. For the canal, the activities are $10^{-4}$ mrem/hr at the outside of concrete shield in radial direction and 0.03 mrem/hr at the water surface in axial direction. The shields for spent fuel storage pool attenuates doses in radial direction of concrete to 0.6 mrem/hr and in axial direction of water to $10^{-4}$ mrem/hr. In beam tube shielding design, the activity at the outside of shield is estimated to 0.135 mrem/hr.

The evaluated dose rates for all shield systems satisfies the design criteria 1.25 mrem/hr.
5. Conclusion

The KAERI established methods and tools for designing the KMRR. Through benchmarking test and sensitivity study, all of computer code systems and modeling techniques were verified prior to being applied to the actual design. Without exceeding the design limits or design criteria, the design results, from neutronics viewpoint, confirm that the KMRR has the following characteristics:

- providing high quality neutron fluxes to meet wide range of applications using holes furnished inside the reactor.
- having negative reactivity coefficients enough to be inherently safe.
- having sufficient excess reactivity for experiments.
- possessing the margin sufficient to control and shut down the reactor.
- not exceeding the constraints concerning the channel powers and the linear element ratings

The shielding design justified that the activity level in the access area is confined within the design criteria.

References


Table I  Maximum and Average Neutron Fluxes (unit: n/cm²·sec) in the Irradiation Holes of the KMRR Reference Core

<table>
<thead>
<tr>
<th>Holes</th>
<th>Group 1</th>
<th>Group 2</th>
<th>Group 3</th>
<th>Group 4</th>
<th>Group 5</th>
<th>Total</th>
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<td>In Inner Core</td>
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<td></td>
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Note:
Group 1 0.821 MeV < E < 10.0 MeV
Group 2 9.118 KeV < E < 0.821 MeV
Group 3 4.0 eV < E < 9.118 KeV
Group 4 0.625 eV < E < 4.0 eV
Group 5 0.0 eV < E < 0.625 eV

Maximum
Average
Fig. 1 Plan View of the KMRR Configuration
UKNDL, ENDF/B-IV, JENDL-2

69-group library

WIMS-KAERI

gain detail information inside lattice

REGAV-K

prepare lattice parameters

TWOTRAN

evaluate control absorber lattice in detail

WIMPAK

prepare X-section tables

KENO-IV

venteure

get flux/power distributions

investigate control absorber and beam tube worth

Fig. 2 Schematic Diagram of the KMRR Physics Design
Fig. 3.1 for coupled neutron and gamma-ray

Fig. 3.2 for only gamma-ray

Fig. 3 Calculational Flow for KMRR Shielding Design
Dynamic Modelling of KMRR and Its Application to Setback Simulation

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Dynamic Modelling of KMRR and Its Application to Setback Simulation

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Abstract

The dynamic model of the reactor and its cooling systems has been derived to study the system performance at transients and steady states and to support the design of an automatic control system for the Korea Multi-purpose Research Reactor. The dynamic model including neutron kinetics, heat transfer of the cooling system, hydraulics of the primary cooling system, and the automatic controller has been extensively described in the present paper. As an application of the dynamic model, we present the simulation results of the plant start-up from source level and the flow induced automatic setback through an automatic controller. The verification of the dynamic model will be done by the planned experiments during and after commissioning.

1. Introduction

Korea Advanced Energy Research Institute (KAERI) is developing the 30 MWth research reactor, Korea Multi-purpose Research Reactor (KMRR), in order to provide an intensive neutron source for the nation wide research programs by 1992. A large number of experimental facilities are provided inside the reactor core and D2O tank to meet the utilization programs such as fuel and material testing for PWR's and CANDU's, production of key radioisotopes and neutron transmutation doped silicon, neutron activation analysis of nuclear grade materials, and neutron radiography.
The KMRR has been designed to satisfy the above mentioned utilization programs in mind and to have the inherent safety features. The reactor consists of the inner and outer core to form an LWR-HWR hybrid. The inner core, 50 cm of effective diameter and 1.2 m high has 23 hexagonal and circular flow channels. Each hexagonal flow channel is loaded with 36-element driver fuel assembly while the circular flow channel is loaded with 18-element fuel assembly. The fuel element is a 6.35 mm diameter and 700 mm long rod type made of 20% enriched U3Si-Al in an aluminum clad. 3 out of 23 hexagonal flow channels are vacant to be able to accommodate fuel test loop. The 18-element fuel assembly is enclosed by the hafnium shroud tube which can be moved up and down. Four of eight hafnium shroud tubes are used for reactor trip and the rest are used for reactor power control. The outer core consists of the eight circular flow channels loaded with 18-element fuel assembly and embedded in D2O tank (reflector), with an effective diameter of 2 m and height of 1.2 m. The large amount of heavy water in D2O tank works as a reflector for the inner core, thus preventing the escape of fast and epithermal neutrons and providing a sufficient site for the installation of irradiation and experimental facilities. The structural material used in the reactor core is made of zircaloy 4 to minimize the neutron absorption. The light water is used as coolant and moderator for the inner core and the heavy water is used as a moderator for the outer core.

In order to investigate the performances of KMRR at normal and transient conditions, we have developed the numerical model of the reactor and its cooling system, KMRRSIM. The KMRRSIM has been extensively used during the detail design phase of KMRR for the control system design, effect analysis of design proposals on reactor control, and prediction of reactor performance under various operating conditions. Before describing the dynamic model of KMRR in detail, we briefly describe the function of the systems used in the model.

Basically there are safety and process systems in KMRR. The safety system which will be activated as soon as the accident occurs includes reactor protection system (RPS) and shutoff rods. Other systems regarded as improved safety factor are emergency water supply system (EWSS), and emergency ventilation system (EVS). The process system which will be continuously operating during normal operation includes reactor regulating system (RRS), primary cooling system (PCS), secondary cooling system (SCS), reflector cooling system (RCS), and auxiliary systems to make the process system work properly.

- Reactor protection system (RPS)

In order to prevent the reactor from operating in a condition which could lead to fuel failure, significant release of radioactivity resulting undue risk to public and operating staff, the RPS must monitor the various system parameters and initiate a reactor trip when predetermined trip setpoints have been exceeded. Eleven trip parameters are selected in KMRR to cover the anticipated postulated events such as loss of primary cooling, loss of heat sink,
and loss of regulation. The RPS consists of trip parameter measuring devices, two out of three general coincidence trip relay logic, trip actuators such as solenoid valves. To ensure its function when required and minimize any interference from the other systems, the RPS has been designed to be physically and functionally separated from any process system. In response to any indication of reactor over-power, loss of regulation, or mismatch of fuel cooling, it opens the solenoid valves and 4 shut off rods which are normally held poised above the core by hydraulic pumps drop into the reactor core. The fail-safe concept has been applied to the design of RPS.

- Emergency Water Supply System (EWSS)

  The main function of EWSS is to enhance that there is an adequate heat sink available for decay heat removal following a loss of the reactor pool inventory caused by a beam tube rupture. The EWSS consists of two modes - an injection mode and a recirculation mode, and it is essentially made up of a storage tank, various valves, two submersible sump pumps, and pool water level detectors. The storage tank can supply water up to 5.5 hr if the water level drops below 60 cm from the chimney top (injection mode). The sump pumps start to operate if the water level in the sump becomes high (recirculation mode).

- Emergency Ventilation System (EVS)

  The EVS filters the radioactive materials at the accident and maintains negative pressure in the reactor concrete island equipment room, reactor pool surface, and fuel test loop room. The EVS consists of the radiation monitoring instruments with associated control logic, fans, filter, and damper. The filtered air is discharged through the stack.

  The safety systems described above are not included in the KMRRRSIM program because they work only at the postulated accident case.

- Primary Cooling System (PCS)

  Fig.1 shows the PCS of KMRR. The core cooling is accomplished by the upward forced convective flow of light water through the common return line and inlet plenum. About 10% of PCS total flow dumped into the bottom of the reactor pool via the bypass line is sucked into the chimney. The mixture of the upward core flow and the downward bypass flow is discharged into a pair of outlet suction nozzles in chimney wall. This kind of combinational flow concept is one of the effective method to prevent the migration of $^{16}\text{O}$ and other short-lived activation products from the chimney to the reactor pool surface. Two PCS pumps and plate type heat exchangers with 50% capacity are installed in parallel to circulate the coolant and to remove heat generated in the reactor core. There are three operating modes in PCS of KMRR depending on the power level of reactor; two pump two heat exchanger mode, one pump one heat exchanger mode, and natural circulation mode. The detailed description about the operating modes can be found on the companion paper of the symposium.
Secondary Cooling System (SCS)
The heat in PCS is transferred to the SCS through two heat exchangers and subsequently discharged into the atmosphere via cooling tower. Two pumps in SCS are normally operating and the third one is in a hot stand-by state in order to circulate the coolant in SCS. In order to keep the inlet temperature of the secondary side of heat exchanger within an operation range the cooling fan and bypass valves are controlled by its own controller.

Reflector Cooling System (RCS)
The heat generated inside the reflector is removed by circulating the heavy water through two pumps (each of them has 100% heat removal capacity) and one heat exchanger. To prevent the disruption of D2O tank due to high temperature, the reactor regulating system constantly monitors the flow and temperature of RCS and initiates reactor shutdown if either a flow reduction or temperature rise occurs.

Reactor Regulating System (RRS)
The reactor power is controlled by the RRS which is composed of a direct digital controller, stepping motor drive mechanism, and four hafnium absorber rods. The direct digital controller receives the nuclear and process signals from the field instruments and converts them into the rod movement signal as specified in the control algorithm which is developed by KAERI design group with a large number of computer simulations. There are two modes of operation in KMRR, automatic and manual. In the automatic control mode, the operator need only choose a demand power and the controller will raise or lower 4 control absorber rods to attain and maintain that power. The automatic controller is necessary to achieve full power from subcritical state in time to avoid xenon poison-out. To increase the flexibility a manual control mode is available where a specific control absorber rod can be moved by the operator. The manual control mode is intended for uses in start-up, low power physics testing, and specific experiments. In order to simplify design, maintain safety, and reduce costs, the manual control has also been realized using the direct digital controller. The detail design of RRS and RPS has been done with a collaboration of AECL and KAERI and that of the other systems have been done by KAERI and local engineering company (KOPEC).

2. Dynamics of Reactor and Its Cooling Systems

The numerical model of the reactor and its cooling systems has been developed to study the system performances at transients and steady states and to support the design of RRS of KMRR. The main features of the KMRRSIM code are:

- Two point reactor kinetics with delayed neutron plus photo-neutron
- Temperature and poison reactivity feedback
- Decay heat model
- Fuel and coolant model of reactor core
- PCS and RCS components (heat exchangers, pumps etc)
- Reactor pool
- Automatic controller including nuclear and process detector models

The two important modifications compared to the previous model\(^{[3]}\) are the inclusion of hydraulic model with a separate treatment of PCS loops for the study of power setback operation and of nuclear and process instrument models used in an automatic controller for the study of system performance in response to the superposition of noise. The reactor and its cooling systems are nodalized to represent the components as correctly as possible and to attain the simplified dynamic equations which can be easily tackled without losing the node characteristics. The nodes used in the KMRRSIM are shown in Fig.2.

2.1 Reactor Model

- Neutron Kinetics

Two point kinetics equations, one point for the core and the other for the reflector, have been derived to correctly represent the effect of the photoneutron generation in a large D\(_2\)O tank and to investigate the controllability of the reactor power with out-core neutron detectors. For the detailed description of the two point kinetics model, refer to the reference given by Oh and Noh\(^{[4]}\). Using the variables defined in the list of symbols of the appendix, the normalized power at the core is

\[
\frac{dN_c}{dt} = \frac{\rho_{loc} - \gamma_c}{\Lambda} N_c + \sum_{i=1}^{6} \gamma_{ci} C_i - \omega \left( 1 - \tau_{cr} \right) (N_c - N_r) + S
\]

and delayed neutron precursor for the i-th group is

\[
\frac{dC_i}{dt} = \lambda_{ci} \left\{ N_c - C_i \right\} = 1, 2, \cdots, 6
\]

The normalized neutron power at the reflector is

\[
\frac{dN_r}{dt} = \frac{1}{\omega} \left[ \sum_{j=1}^{9} \gamma_{dj} D_j + \frac{\alpha_{rc}}{\Lambda} \left( 1 - \tau_{rf} \right) \left\{ N_c - N_r \right\} - \frac{\gamma_d}{\Lambda} N_r \right]
\]

and the photoneutron precursor for the j-th group is

\[
\frac{dD_j}{dt} = \lambda_{dj} \left\{ N_c - D_j \right\} \quad j = 1, 2, \cdots, 9
\]
Xenon Dynamics

The net rate of formation of iodine-135(I) and xenon-135(X) concentrations become

\[
\frac{d I}{dt} = \lambda I \left\{ Nc - I \right\}
\]

(5)

\[
\frac{d X}{dt} = \frac{\lambda X + \lambda e}{\gamma I + \gamma x} \left\{ \gamma X Nc + \gamma I I \right\} - \left\{ \lambda X + \lambda e Nc \right\} X
\]

(6)

Decay Heat

By choosing the appropriate decay constants ($\lambda wk$) for the fission product concentrations, the decay heat can be included as follows

\[
\frac{d Wk}{dt} = \lambda wk \left\{ Nc - Wk \right\} \quad k = 1, 2, 3
\]

(7)

And the fuel power including decay heat becomes

\[
Nt = Nc - \sum_{k=1}^{2} \gamma wk \left\{ Nc - Wk \right\}
\]

(8)

2.2 Heat Transfer Model of PCS

In modelling the heat transfer of PCS and RCS the spatial variation of temperature in a node is ignored using the average value, and transport delay between the reactor and heat exchanger is treated by the mixing volume model.

Reactor Thermal Power

As the reactor thermal power is originated by the heat produced both in the core and in the reflector, the core power ($Qc$) and the reflector power ($Qr$) can be represented by

\[
Qc = Nt \cdot Qt \cdot \zeta c
\]

\[
Qr = Nt \cdot Qt \cdot (1 - \zeta c)
\]

(9)

where $Qt$ is the nominal thermal power at 100% full power (30 MWth). The total thermal power is the sum of $Qc$ and $Qr$.

Reactor Core

An energy balance for the reactor core allowing for the energy transported by the coolant passing through the fuel assembly is given by

\[
M_{e Cre} \frac{d Tc}{dt} = \zeta c Qc - U_{r} A_{r} \left( Tc - T_{e} \right)
\]

(10)

with $M_{e Cre} = M_{C r e} + M_{e} \cdot C_{e1}$ and $U_{r} = 115.05 \text{ We}^0.4$.
\[ \frac{d \dot{M}_\text{Ch}}{dt} = (1-\zeta)Q_p + U_{\text{Ar}}(T_\text{Ch} - T_i) - 2\dot{M}_\text{Ch}(T_\text{Ch} - T_i) \]  

**Chimney**

As the core flow rate \((\dot{W}_\text{c})\) and the bypass flow rate \((\dot{W}_\text{b})\) are mixed in the chimney and divided into primary flow rates \(\dot{W}_{\text{pi}}\) and \(\dot{W}_{\text{bpi}}\), the chimney temperature, \(T_{\text{m}}\), can be calculated from

\[ \frac{dT_{\text{m}}}{dt} = \frac{\dot{M}_\text{Ch}}{C_{\text{m}}}(2T_\text{Ch} - T_i) + \dot{W}_{\text{bpi}}(2T_\text{Ch} - T_i) - (\dot{W}_{\text{pi}} + \dot{W}_{\text{bpi}})C_{\text{m}}T_{\text{m}} \]

**HOT1 and HOT2**

In the nodes HOT1 and HOT2 of Fig.2, the temperature increases a little bit due to the heat produced by the reactor coolant pump \((Q_{\text{pi}} = Q_{\text{bpi}} = 0.2\ \text{MW})\)

\[ \frac{dT_k}{dt} = \frac{\dot{M}_k}{C_{\text{m}}}(T_k - T_{\text{m}}) \]

where \(k = 1\) or \(2\) for the HOT1 and HOT2, respectively.

The transport time delay occurs in the intermediate pumping from the reactor to the heat exchanger and vice versa. Assuming no heat loss in the pipe transportation system, the heat transfer equations for the nodes such as POT1, POT2, COLD, LCOM, HCOM, INLET, BYPASS, LPOOL, and HPOOL are given by:

\[ \frac{dT_k}{dt} = W_{\text{pi}}C_{\text{m}}(T_k - T_{\text{m}}) \]

\[ \frac{dT_1}{dt} = \frac{\dot{M}_1}{C_{\text{m}}}(2T_1 - T_{\text{m}}) + \dot{W}_{\text{pi}}C_{\text{m}}(2T_1 - T_{\text{m}}) - (\dot{W}_{\text{pi}} + \dot{W}_{\text{bpi}})C_{\text{m}}T_1 \]

\[ \frac{dT_2}{dt} = \frac{\dot{M}_2}{C_{\text{m}}}(2T_2 - T_{\text{m}}) + \dot{W}_{\text{bpi}}C_{\text{m}}(2T_2 - T_{\text{m}}) - (\dot{W}_{\text{pi}} + \dot{W}_{\text{bpi}})C_{\text{m}}T_2 \]

\[ \frac{dT_3}{dt} = \frac{\dot{M}_3}{C_{\text{m}}}(T_3 - T_{\text{m}}) \]

\[ \frac{dT_4}{dt} = \frac{\dot{M}_4}{C_{\text{m}}}(T_4 - T_{\text{m}}) \]

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\[ \frac{dT_10}{dt} = \frac{\dot{M}_10}{C_{\text{m}}}(T_10 - T_{\text{m}}) \]

\[ \frac{dT_11}{dt} = \frac{\dot{M}_11}{C_{\text{m}}}(T_11 - T_{\text{m}}) \]

\[ \frac{dT_12}{dt} = \frac{\dot{M}_12}{C_{\text{m}}}(T_12 - T_{\text{m}}) \]
\[
\frac{d T_w}{dt} = \frac{W_b C_{Cc}(T_r - T_w)}{} \tag{21}
\]

- Heat Exchanger

The plate type heat exchanger used in PCS has been modelled into 3 parts such as primary coolant (p), plate metal (t), and secondary coolant (s). By application of energy balance for each node we get;

\[
\frac{d T_{ppk}}{dt} = \frac{M_{ppk} C_{Cc} (T_{hk} - T_{ppk}) - U_{ppk} A_{ppk} (T_{ppk} - T_{ptk})}{dt} \tag{22}
\]

\[
\frac{d T_{ptk}}{dt} = \frac{M_{ptk} C_{t} = U_{ppk} A_{ppk} (T_{ppk} - T_{ptk}) - U_{psk} A_{psk} (T_{ptk} - T_{psk})}{dt} \tag{23}
\]

\[
\frac{d T_{psk}}{dt} = \frac{M_{psk} C_{Cc} = U_{pek} A_{psk} (T_{ptk} - T_{pel}) - 2W_{spk} C_{Cc} (T_{psk} - T_{si})}{dt} \tag{24}
\]

where \(k = 1\) or \(2\) for loop 1 or loop 2, respectively

\(U_{ppk} = 135.57 \hspace{1mm} W_{pk}^{0.721}\)

\(U_{psk} = 135.57 \hspace{1mm} W_{ps}^{0.721}\)

2.3. Heat Transfer Model of RCS

There is a large amount of heavy water reflector surrounding the core and heat produced in the reflector tank \(Q_r\) contributes to increasing the secondary outlet temperature, \(T_{so}\), through the reflector heat exchanger.

The temperature at the reflector tank \(T_{rc}\) is

\[
\frac{d T_r}{dt} = \frac{M_{rc} C_{r} = Q_r - 2W_{Cr} (T_{rc} - T_{ri})}{dt} \tag{25}
\]

and the intermediate piping nodes ROUT \(T_{ro}\) and RIN \(T_{ri}\) are

\[
\frac{d T_{ro}}{dt} = \frac{M_{ro} C_{r} = W_{Cr} (2T_{rc} - T_{ri} - T_{ro})}{dt} \tag{26}
\]

\[
\frac{d T_{ri}}{dt} = \frac{M_{ri} C_{r} = W_{Cr} (2T_{rp} - T_{ro} - T_{ri})}{dt} \tag{27}
\]

By applying the same principle used for PCS heat exchanger, the energy balance equations for reflector heat exchanger are

\[
\frac{d T_{rp}}{dt} = \frac{M_{rp} C_{r} = 2W_{Cr} (T_{ro} - T_{rp}) - U_{rpArp} (T_{rp} - T_{rt})}{dt} \tag{28}
\]
\[
\begin{align*}
\frac{d}{dt} M_{\text{rtC}} &= U_{\text{rp}} \Delta T_{\text{rp}} - U_{\text{rs}} \Delta T_{\text{rs}} \\
\frac{d}{dt} M_{\text{rsC}} &= U_{\text{rs}} \Delta T_{\text{rs}} - 2 W_{\text{rc}} \Delta T_{\text{rs}} - T_{\text{si}}
\end{align*}
\]

with \( U_{\text{rp}} = 435.28 \, \text{Wr}^{0.721} \) and \( U_{\text{rs}} = 435.28 \, \text{Wr}^{0.721} \)

2.4 Secondary Cooling System

As the numerical model for the SCS of KMRR has not been developed until now, we have assumed that the heat transferred through the heat exchangers of PCS and RCS has been completely removed by the SCS while keeping the secondary inlet temperature \((T_{\text{si}})\) to a maximum design temperature of 32°C. Therefore, the secondary outlet temperature is calculated from the following equation.

\[
T_{\text{so}} = \frac{W_{\text{sp1}} T_{\text{psol}} + W_{\text{sp2}} T_{\text{psol}} + W_{\text{rs}} T_{\text{rs}}}{W_{\text{s}}}
\]

2.5 Hydraulic Model of PCS

To simulate the setback operation in KMRR, we need to develop a hydraulic model of PCS. The PCS of KMRR is simplified to 4 path loops from junction "a" to "b" shown in Fig.2 using inertia, acceleration and viscous, hydrostatic, and pump head terms for each loop. We utilize the continuity and mechanical energy balance equation for each loop with the assumption that PCS flow is single-phase turbulent and incompressible. Then we have 5 equations for 5 unknowns (pressure difference between junction "a" and "b", mass flow rates of 4 loops). The acceleration and viscous pressure drop along PCS loop within turbulent flow range is approximated to be proportional to 1.8th power of flow rate and the pump head was obtained by solving the pump dynamic equations using the manufacturer's correlation. The detailed equations can be found on the reference [2].

2.6 Automatic Controller

The automatic controller receives flux informations such as log power, linear power, and log rate from the fission chamber and process informations such as flow rates and temperature from the field instruments of the related systems. There exists a finite time delay in the response of the measuring system. To reflect the finite time delay, we used the first order differential equation for the log, linear amplifier, and temperature measuring instruments involved in RRS. They are represented by the Laplace transformation form with an appropriate time constants (\( \tau' \)).
- For log and linear signal \( N \)

\[
N(s) = \frac{1}{1 + s \tau_n} \text{Nr}(s) \quad (32)
\]

where \( \text{Nr} \) is the log or linear power measured by the fission chamber.

- For lograte signal

\[
\text{Lograte}(s) = \frac{s \tau_2}{(1 + s \tau_1)^2} N(s) \quad (33)
\]

- For temperature signals

\[
T_m(s) = \frac{1}{1 + s \tau_{\text{temp}}} T_{\text{real}}(s) \quad (34)
\]

where \( T_m \) is the time delayed temperature which the automatic controller recognizes and \( T_{\text{real}} \) is the measured value by the field instruments.

The required rod movement signal, \( Y_1 \), is derived from the amplified error signal subtracting the appropriate lograte signal amplified by \( G_2 \), which limits the power increase rate below 5% of present power (PP) per sec.

\[
Y_1 = \left[ G_1 \log\left( \frac{P_{\text{m}}}{n \pm 1} \right) - G_2 \text{lograte} \right] \pm 1 \quad (35)
\]

The \( Y_1 \) is further filtered and multiplied by the maximum rod withdrawal speed, then the control rod as specified by the rod selection algorithm moves accordingly. The determination of the control parameters such as \( G_1 , G_2 \) of eq.(25) are described in the reference [3].

For the on-line calibration of the fission chamber signals, (thermal mode), the automatic controller computes the thermal output by measuring the differential temperature of the heat exchanger secondary side of SCS loop. The thermal mode can be applicable only if the reactor power becomes higher than 20% of full power because of the measurement error of the differential temperature [5]. The total reactivity \( (\dot{\rho}_{\text{tot}}) \) in eq.(1) can be derived from the control rod reactivity worth curve \( (\rho_{\text{CAR}}) \), the feedback reactivity from fuel and coolant temperatures variations, and xenon load.

\[
\dot{\rho}_{\text{tot}} = \rho_{\text{CAR}} + \alpha_f (T_f - T_{\text{ref}}) + \alpha_c (T_c - T_{\text{co}}) + \alpha x
\quad (36)
\]

where \( T_f \) and \( T_c \) are fuel and coolant temperatures at the initial steady state, respectively.
3. Application of KMRRSIM

The simultaneous 1st order differential equations described in section 2 are solved by the Runge Kutta-Fehlberg 5th and 6th order algorithm. To check the validity of the two point kinetics model, the comparison study was extensively done with the space-dependent HEXKIN[8] program for the various reactivity insertions.

Also many simulation studies[7] have been done to test the various functions of the automatic controller under four modes of reactor operation: automatic, manual, shutdown, and setback. As an example of the application of KMRRSIM, we will present two simulation results in this paper. One is the reactor startup simulation from source power level to 100% FP via automatic controller which covers almost 10 decades of power range. The other is the power setback operation introduced to KMRR for the purpose of increasing the reliability of PCS and plant capacity factor.

3.1 Reactor Start-up Simulation

To demonstrate the controllability of the automatic controller, we have simulated a reactor start-up from source level to 100% FP. The 4 shutdown rods are poised at the top of the core and 4 control absorber rods are fully inserted at the bottom of the core. As the transition from the natural circulation to one pump one heat exchanger mode in PCS is so short, we have assumed that the flow in PCS is at the state of nominal full flow rate condition. In order to have an adequate reading on the neutron flux, the neutron source with an intensity of 10^7 neutrons/sec (normalized) has been inserted into the core. On this subcritical equilibrium state, the shutdown power level can be determined from source level and subcritical reactivity present in the reactor. Fig.3 shows the transient responses of neutron power and xenon concentration as a function of time. It takes about 20 min. to raise a reactor power from 2 x 10^-1 % FP to 100% FP because of the limitations both on the lograte (≤ 5 % PP/sec) and on the reactivity insertion rate (0.33 mk/sec). The 4 control absorber rods are withdrawn at the maximum speed up to 700 seconds. As soon as the lograte signal approaches to 5% PP/sec, the reactor power increases exponentially with a stable period of 20 sec as shown in Fig.3. (800 - 1200 sec)

When the reactor power approaches to the demand power of 100% FP, the automatic controller adjusts control rods to maintain that power. Fig.4 shows the transient responses of the important process parameters such as fuel, reactor outlet, reflector outlet, and secondary outlet temperatures. As the reactor power is very low up to 1100 sec, those temperatures stay at 32°C which is the inlet temperature to the heat exchanger secondary side. Then they increase quite fast and approach to their respective 100% FP steady state values.
As the process system achieves its equilibrium state after 1500 sec, the automatic controller needs only to adjust the control rods to compensate the build up of xenon which takes quite a long time to approach its equilibrium (approximately 3 days).

3.2 Power Setback Simulation

A power setback is initiated by RRS whenever a process system fails such as the case that the reactor safety is not impaired but the full heat load can not be removed. Thus the events such as a single component failure in the PCS and SCS which can be repaired during plant operation do not need a reactor shutdown. By introducing the setback concept into KMRR, we can avoid the long shutdown time due to xenon load.

The setback in KMRR means that if the RRS detects any malfunction of the components in PCS and SCS and the reactor is operating under two pump two heat exchanger mode with a reactor power higher than 50 % FP, the RRS initiates power setback and subsequently lowers the reactor power to 50 % FP before the reactor trip parameters such as flow rate, temperatures in PCS and SCS reach their respective setpoints.

There are two ways to initiate setback in KMRR, automatic and manual. The automatic setback occurs whenever RRS control algorithm detects the following events with a condition of reactor power higher than 50 % FP under two pump two heat exchanger operation of PCS.

- The PCS flow drops less than 90 % of nominal flow
- The SCS flow drops less than 90 % of nominal flow
- The heat exchanger inlet temperature on the secondary side exceeds 33°C

The manual setback can also be initiated if the operator detects any problem on the PCS pumps, heat exchangers, SCS pump or cooling tower.

Fig.5 shows the transient response of neutron power and the flow rate of each loop under setback induced by the low flow rate in PCS such as one pump off. The flow rate through a failed pump decreases quite fast while that through an intact pump increases slightly because of the reduction of the flow resistance through the normally operating loop. As the core flow rate reaches the setback initiating setpoint (90 % of nominal flow rate), the neutron power drops very rapidly to 50 % FP before the flow rate through reactor core reaches the low flow trip setpoint (70 % of nominal flow rate).

4. Conclusion

The dynamic model of KMRR has been developed to investigate the system performance and to support the design of RRS. The model covers the neutron kinetics, reactivity feedback from fuel and
coolant temperature variations, decay heat, thermal hydraulics of cooling system, and the automatic controller.

The KMRRSIM code has been extensively used to verify the proposed design concept and to determine the control parameters of RRS. From the simulation results for a reactor start-up shown in Fig.3, we have demonstrated the controllability of the automatic controller for all power ranges from source power level to 100% FP. The power setback concept which has been introduced for the normal operation mode of RRS has been demonstrated as a possible means of a reasearch reactor control without tripping a reactor in the case of a minor failure event on a PCS or SCS.

The control algorithm will be tested by mocking up the plant dynamic equations on the real-time digital computer at Chalk River national laboratory before commissioning.

5. References


Appendix

List of Symbols

\( N_c = \) normalized neutron power at core region
\( N_r = \) normalized neutron power at reflector region
\( \Lambda = \) neutron generation time
\( \gamma_{ci} = \) the i-th delayed neutron yield fraction (\( \gamma_c = \sum_{i=1}^n \gamma_{ci} \))
\( \alpha_{cr}(1-\tau_{cr}) = \) coupling coefficient from core to reflector
\( \alpha_{rc}(1-\tau_{rc}) = \) coupling coefficient from reflector to core
\( \lambda_{ci} = \) the i-th delayed neutron decay constant
\( \omega = \) fraction of steady state neutron power of core to that of reflector
\( C_i = \) delayed neutron precursor of the i-th group
\( D_j = \) photoneutron precursor of the j-th group
\( \gamma_{dj} = \) the j-th photoneutron yield fraction (\( \gamma_d = \sum_{j=1}^m \gamma_{dj} \))
\( \lambda_{dj} = \) the j-th photoneutron decay constant
\( \lambda_I = \) decay constant of \(^{135}\text{I} \)
\( \lambda_X = \) decay constant of \(^{135}\text{Xe} \)
\( \gamma_I = \) yield fraction of \(^{135}\text{I} \)
\( \gamma_X = \) yield fraction of \(^{135}\text{Xe} \)
\( \lambda_e = \) effective decay constant of \(^{135}\text{Xe} \)
\( W_k = \) fission product of the k-th group
\( \lambda_{wk} = \) the k-th fission product decay constant
\( N_t = \) thermal power including decay heat
\( T_{sub} = \) temperature of node "sub"
\( M_{sub} = \) mass of node "sub"
\( W_{loop} = \) mass flow rate of a loop "loop"
\( U_{sub} = \) heat transfer coefficient at the heat exchanger "sub"
\( A_{sub} = \) heat transfer area at the heat exchanger "sub"
\( C_{sub} = \) specific heat of node "sub"
\( \zeta_f = \) fraction of fission energy absorbed in fuel element
\( \zeta_c = \) fraction of power produced in core
S = normalized source neutrons

Subscript (sub)

f = fuel
c = coolant
c1 = clad
m = chimney
uk = pout for k = 1 or 2
hk = hot for k = 1 or 2
d = cold
b = lcom
e = hcom
l = inlet
v = bypass
r = lpool
\[ w = \text{hpool} \]
\[ t = \text{plate of heat exchanger} \]

**Loop**

\[ p = \text{primary} \]
\[ s = \text{secondary} \]
\[ r = \text{reflector} \]
\[ b = \text{bypass} \]

**Example**

\( W_{p1} \): stands for the mass flow rate at the secondary loop through the primary heat exchanger number 1.

\( M_{ppk} \): stands for coolant mass in a heat exchanger \( k \) of a primary loop.

---

*Fig. 1. Schematic Diagram of PCS*
Fig. 2. Simplified Block Diagram of KMRR Simulation Model

Fig. 3. Neutron power and Xe concentration for initial start-up
Fig. 4. Temperatures for initial start-up

Fig. 5. Neutron power and flow rates for setback
RESEARCH REACTOR DHRUVA

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RESEARCH REACTOR DHARUV

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ABSTRACT

Dhruva, a 100 MWt research reactor located at the Bhabha Atomic Research Centre, Bombay, attained first criticality during August, 1985. The reactor is fuelled with natural uranium and is cooled, moderated and reflected by heavy water. Maximum thermal neutron flux obtained in the reactor is $1.8 \times 10^{14}$ n/cm$^2$/sec. Some of the salient design features of the reactor are discussed in this paper. Some important features of reactor coolant system, regulation and protection systems and experimental facilities have been presented. A short account of the engineered safety features is provided. Some of the problems that were faced during commissioning and initial phase of power operation have also been dealt upon.

1. INTRODUCTION

India's fifth research reactor DHARUV which became critical on August 8, 1985, is a natural uranium fuelled, heavy water moderated and cooled thermal neutron research reactor with an operating power level of 100 MWt and a maximum thermal neutron flux of $1.8 \times 10^{14}$ n/cm$^2$/sec. The indigenously designed and built reactor is located at the Bhabha Atomic Research Centre (BARC), Trombay, close to the city of Bombay. A number of Indian industrial firms and BARC, have between them, shouldered the responsibilities of fabricating all the major components of the reactor such as reactor vessel, fuelling machine and heat
exchangers. The reactor employs natural metallic uranium seven rod cluster fuel assemblies installed in zircaloy guide tubes in a stainless steel reactor vessel. Heavy water is used as the moderator, reflector and primary coolant and helium is used as cover gas. Reactor power regulation is achieved by moderator level control. Fast shutdown of the reactor is effected by nine cadmium shutoff rods with simultaneous dumping of the heavy water moderator. Heat from the primary coolant is transferred to a closed loop recirculating secondary light water system in a set of heat exchangers. The secondary coolant in turn is cooled by sea water in another set of heat exchangers. The sea water coolant is drawn from the Bombay harbour bay and flows through the heat exchangers on a once-through basis. Salient design data are given in Table I.

2. PLANT LAYOUT

Figure 1 shows the general plant layout. The reactor containment building houses the pile and storage blocks, the fuelling machine and the heavy water primary coolant pumps and heat exchangers. The service building houses the process water (secondary coolant) pumps and heat exchangers, air compressors and normal and emergency power supply equipment and air handling units. An annexe building houses the main control and instrumentation rooms. The spent fuel storage building communicates with the reactor building and houses water-filled trenches and bays for handling and storage of spent fuel. A once-through ventilation system is employed for the reactor building with air-flow maintained from lower to higher radioactive zones, the air being finally exhausted through high efficiency particulate air filters through a high stack.

3. PILE BLOCK

Figure 2 shows a schematic sectional view of the pile block. The reactor has a vertical core housed in a stainless steel vessel (or calandria) which is located inside a concrete vault filled with light water for simplifying the bulk shield design. The reactor vessel is supported
at the bottom on a support structure. The vault is lined with stainless steel for providing water leak tightness. Shielding at the top is provided by an annular shield and an end shield assembly. Extension tubes are rolled into the top tube sheet of the reactor vessel and these extend the reactor vessel boundary to the top deck plate. Inside the extension tubes, zircaloy-stainless steel integral guide tubes are placed forming the reactor fuel channels extending from the inlet plenum to the tail pipes in the service space and further on to the top of the deck plate for facilitating fuel installation.

4. FUEL ASSEMBLY

4.1 Figure 3 shows a fuel assembly which consists of three sub-assemblies viz. fuel cluster sub-assembly, shield sub-assembly and seal-and-shield plug sub-assembly. The fuel cluster sub-assembly is about 3 metres in length and forms the bottom portion of the 9.3 metre long fuel assembly. It consists of 7 nos. aluminium cladded uranium fuel rods assembled inside an aluminium flow tube and is located in the zircaloy portion of the guide-tube in the core region of the fuel channel. Aluminium spacers are fixed to the central rod of the fuel cluster and are distributed evenly over the length.

4.2 Towards ensuring that the assembly sits snugly inside the guide tube, a split-bulge is provided at the bottom of the flow-tube. The free diameter of this bulge is slightly more than the inside diameter of the fuel channel at the bottom. However, due to the split type design, the bulge collapses slightly inwards like a leaf-spring during installation and sits without any clearance inside the channel. Another split collar of a slightly different design, is provided at the top of the flow-tube for the same purpose. A solid top bulge, just below the top split-collar is provided for restricting the coolant flow bypassing the fuel rods. The shield sub-assembly and seal-and-shield plug sub-assembly form the extension of the fuel cluster sub-assembly upto the top of the reactor and provide shielding and means for locking the fuel assembly to its channel.
4.3 Refuelling is carried out by a fuelling machine which can move over the top of the pile and storage blocks. The machine has provisions to load and unload fresh and irradiated fuel into and from the core. Heavy water cooling provisions have been made in the machine to provide transit cooling to irradiated fuel assemblies, before they are discharged into the storage block or spent fuel storage bays.

5. NORMAL REACTOR COOLING

The reactor coolant circuit is shown schematically in Fig.4. The heavy water coolant enters through the inlet plenum at the bottom of the reactor vessel and then flows up through the fuel assemblies and exits through the tail pipes at the top of each fuel channel and joins into a common outlet header. From the outlet header the coolant flows through three down comers connected to the suction of three main coolant pumps. Thereon it passes through heat exchangers back to the inlet plenum, thus flowing in a closed loop recirculating circuit. A part of the coolant is diverted for cooling the upper tube sheet of the reactor vessel and other structural components, thence discharged into the moderator and finally joined at the suction of the main coolant pumps via the moderator return flow path. This feature inter-connects the moderator and coolant systems and provides cooling to the moderator also without any need for separate heat exchangers for moderator cooling. Heat transferred to the secondary light water coolant circuit is removed by circulating the secondary water through light water-sea water heat exchangers. Sea water is drawn from the Bombay harbour bay with the help of pumps located in a pump house at the end of a jetty and flows on a once-through basis. Thus the heat generated in the reactor is finally rejected to the sea. The design intent for adopting the intermediate light water circuit was to preclude any possibility of radioactive heavy water release into the sea as well as to ensure a long and reliable service life for the heavy water heat exchangers by preventing direct sea water cooling and consequent corrosion effects.
6. SHUTDOWN COOLING

Adequate provisions have been made in the design to assure uninterrupted fuel cooling to preclude any possibility of damage to the fuel. For the situation of a Class IV mains power supply failure, the reactor is automatically tripped upon loss of power to the main coolant pumps. Large flywheels installed on the main coolant pump shafts ensure a slow flow-coastdown while small auxiliary coolant pumps (operated on Class II power supply and piped in parallel with the main coolant pumps) take over and supply the required coolant flow to the core (Fig. 5). A safety feature of these pumps is the provision of a water-turbine prime-mover on the same shaft (in addition to the electrical motor prime-mover) which ensures uninterrupted operation of the pump even under a power supply black-out situation. The overhead light water storage tank ensures gravity feed of power-water to the water-turbines for several hours even without operator intervention.

7. EMERGENCY CORE COOLING

Because of the inter-connection of the heavy water coolant and moderator systems, the reactor is intrinsically safe against a small break loss of coolant accident (LOCA) situation. For example, for a heavy water coolant loss rate as high as 1000 litres per minute, the moderator system can supply the required coolant inventory towards uninterrupted fuel cooling for at least 30 minutes without operator action. Notwithstanding this intrinsic feature, the reactor has been provided with an emergency core cooling system (ECCS) with provisions for detecting the LOCA situation, collecting the leaking heavy water in separate tanks and recirculating the same with ECCS pumps and heat exchangers through the core. Provision also exists for injecting light water into the core through a system of valves and rupture discs, should this highly improbable situation arise.

8. REACTOR POWER REGULATION

Reactor power is regulated by control of moderator heavy water level in the reactor vessel. Level control is achieved by an inde-
pendent level control circuit consisting of three level control pumps which pump heavy water into the moderator space of the reactor vessel from a dump tank below (Fig. 4). Three control valves control the return flow of the moderator from the reactor vessel into the heavy water dump tank. Automatic power regulation is achieved by the power regulation system neutronic signals which adjust the valve positions for raising, maintaining or lowering the moderator level depending on the desired power level. Twelve neutron detectors (fission and boron coated ion chambers) installed in three instrument tubes in the concrete biological shield of the reactor provide log rate, linear rate, log power, linear power and over-power signals for the power regulation and safety systems.

9. REACTOR PROTECTION SYSTEM

9.1 Nine cadmium shutoff rods are provided as primary fast-acting shutdown devices. These are parked above the core-region during power operation and are rapidly inserted into the core on a scram signal. Insertion is primarily by gravity assisted initially by an accelerating spring. Moderator dump has been provided as a slow-acting back-up shutdown system which also ensures long term shutdown safety. In addition, a fast-acting emergency shutdown (ESD) system comprising injection of a liquid poison into a set of 20 zircaloy tubes located in the central region of the core has also been provided. The ESD system is actuated in the event of failure of the shutoff rod/moderator dump system as also in the event of certain other abnormal conditions such as reactor power exceeding a pre-set limit.

9.2 Computations show no fuel damage for anticipated operational occurrences like main coolant pump trip, loss of class IV power supply and loss of reactor power regulation for a complete shutoff rod system failure due to provision of the back-up automatic heavy water dump and liquid poison reactivity shutdown mechanisms and favourable primary coolant flow coastdown characteristics.
10. EXPERIMENTAL FACILITIES

These are given below:

<table>
<thead>
<tr>
<th>Type</th>
<th>No.</th>
<th>Maximum thermal neutron flux $\times 10^{-13}$ $(n/cm^2/sec.)$</th>
</tr>
</thead>
<tbody>
<tr>
<td>Engineering loop (150 mm)</td>
<td>1</td>
<td>14.6</td>
</tr>
<tr>
<td>Engineering loop (100 mm)</td>
<td>1</td>
<td>16.4</td>
</tr>
<tr>
<td>Radial beam holes (100 mm)</td>
<td>4</td>
<td>3.8</td>
</tr>
<tr>
<td>Tangential beam holes (100 mm)</td>
<td>4</td>
<td>8.1</td>
</tr>
<tr>
<td>Radial beam holes (300 mm)</td>
<td>2</td>
<td>7.0</td>
</tr>
<tr>
<td>Cold neutron source (300 mm/300 mm)</td>
<td>1/1</td>
<td>13.4</td>
</tr>
<tr>
<td>Hot neutron source (300 mm/100mm)</td>
<td>1/2</td>
<td>14.3</td>
</tr>
<tr>
<td>Upper through tube (100 mm) for isotope production</td>
<td>1</td>
<td>8.1</td>
</tr>
<tr>
<td>Lower through tube (100 mm) with scatterer</td>
<td>1</td>
<td>16.3</td>
</tr>
<tr>
<td>Pneumatic carrier facility</td>
<td>1</td>
<td>18.4</td>
</tr>
<tr>
<td>Isotope tray rods (including cobalt slug rods)</td>
<td>4</td>
<td>17.7</td>
</tr>
<tr>
<td>Creep and corrosion facilities</td>
<td>3</td>
<td>11.3</td>
</tr>
</tbody>
</table>

11. COMMISSIONING EXPERIENCE

11.1 During commissioning and initial operating phases of the reactor, a number of problems were encountered which delayed full power regular operation till January 1988.

11.2 Considerable time and effort had to be spent during lightwater testing of the heavy water system for flushing out construction and fabrication debris like stainless steel metal turnings, stainless steel dust etc arising predominantly from inadequate cleaning of calandria after fabrication prior to site installation.
11.3 Initial design of the fuel assembly employed solid bulges at the top and bottom of the flow tube with a small radial clearance between the bulges and the inner surface of the housing guide tube. This caused large amplitude flow-induced vibrations leading to excessive wear of fuel rod aluminium cladding, especially at spacer locations, exposing the uranium to the coolant and making sustained power operation difficult due to high radioactivity in the coolant. The problem was solved by ensuring a snug fit of the fuel assembly in the guide tube by providing split bulges in place of solid bulges to act like springs bearing on the guide tube inner surface with zero radial clearance.

11.4 Aluminium turbidity appeared in the system heavy water due to the excessive fuel vibration problem mentioned earlier. The turbidity was removed by purifying the heavy water through special acrylic type weak acidic magnesium-magnesium oxide loaded resins as also by employing a centrifuge-separator in the system.

11.5 The fuel assembly lock/unlock (to the channel) mechanism was not hundred percent reliable raising the concern of inadvertent ejection of the assembly during full flow operation. Back-up mechanical locks were provided for all in-core assemblies to solve this problem.

11.6 The original design of the radial bearing lubrication circuit of the heavy water primary coolant pumps had to be modified for correcting excessive oil leakage to prevent fire hazards and to enable sustained pump operation.
## Table 1

**SALIENT DESIGN DATA**

<table>
<thead>
<tr>
<th>Description</th>
<th>Specification</th>
</tr>
</thead>
<tbody>
<tr>
<td>Reactor Power (thermal)</td>
<td>100 MW</td>
</tr>
<tr>
<td>Overhead (emergency cooling) water storage tank capacity</td>
<td>1.8 x 10⁶ Litres</td>
</tr>
<tr>
<td>Reactor (pile) block</td>
<td>11.05 m dia x 11.94 m high</td>
</tr>
<tr>
<td>Reactor vessel (stainless steel calandria)</td>
<td>3.72 m dia x 2.875 m high</td>
</tr>
<tr>
<td>Annular bulk water shield thickness around calandria</td>
<td>1.22 m</td>
</tr>
<tr>
<td>Annular bulk concrete shield thickness</td>
<td>2.44 m</td>
</tr>
<tr>
<td>Number of lattice positions</td>
<td>146</td>
</tr>
<tr>
<td>Lattice pitch and geometry</td>
<td>18 cm square</td>
</tr>
<tr>
<td>Typical operational loading</td>
<td></td>
</tr>
<tr>
<td>Fuel assemblies</td>
<td>127</td>
</tr>
<tr>
<td>Shut-off rods</td>
<td>9</td>
</tr>
<tr>
<td>Engineering loops</td>
<td>2</td>
</tr>
<tr>
<td>Creep and corrosion facilities</td>
<td>3</td>
</tr>
<tr>
<td>Pneumatic carrier</td>
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</tr>
<tr>
<td>Isotope tray and slug rods</td>
<td>4</td>
</tr>
<tr>
<td>Fuel Assembly</td>
<td></td>
</tr>
<tr>
<td>Natural uranium metal clad with 1 mm aluminium</td>
<td>1.27 cm dia x 303 cm long</td>
</tr>
<tr>
<td>Flow tube (aluminium)</td>
<td>5.23 cm ID and 1 mm thick</td>
</tr>
<tr>
<td>D₂O coolant flow rate</td>
<td>490 LPM</td>
</tr>
<tr>
<td>Maximum assembly power (thermal)</td>
<td>1125 kW</td>
</tr>
<tr>
<td>D₂O gross coolant flow rate</td>
<td>69000 LPM</td>
</tr>
<tr>
<td>D₂O gross coolant temp. rise</td>
<td>19°C</td>
</tr>
<tr>
<td>D₂O coolant exit temp.</td>
<td>70°C max.</td>
</tr>
</tbody>
</table>
Fig 1  PLANT LAYOUT
Fig. 2 PILE BLOCK SCHEMATIC
Fig. 3  FUEL ASSEMBLY
Fig. 4  NORMAL COOLING SCHEMATIC
OVERHEAD $\text{H}_2\text{O}$ STORAGE TANK

HEADER

FUEL CHANNEL

NORMAL COOLING

MAKE-UP PUMP

$\text{D}_2\text{O}$ MAIN PUMP

FLYWHEEL

$\text{D}_2\text{O}$ AUX PUMP

TURBINE

MOTOR (CL. II SUPPLY)

$\text{H}_2\text{O}$ DUMP TANK

Fig. 5 SHUTDOWN COOLING SCHEMATIC
современное состояние и перспективы
исследовательских ядерных реакторов в СССР

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PRESENT STATUS AND FUTURE PROSPECTS OF RESEARCH REACTORS IN THE SOVIET UNION

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ABSTRACT

The research reactors which are currently in use in the USSR are employed in a wide range of research in various scientific fields, as well as for certain applied tasks. Most of these reactors are pool-type reactors. Since it is significantly cheaper to upgrade research reactors rather than to build new ones, the vast majority of them have been upgraded and their experimental capabilities significantly expanded. In the USSR the future of research reactors lies in the continued modernization of currently operating research reactors and the building of new powerful research reactors for which designs are being developed. Some are already under construction (for example, the PIK reactor). These designs are developing Soviet research reactor concepts which centre around pressure-vessel-type reactors and channel-type reactors in tanks. Other technical ideas are also being used. Research reactor safety meets current requirements on the whole; however, their long operating life, their proximity to heavily populated areas, and several other features of research reactors makes safety a higher priority. A series of organizational and technical measures are being undertaken to improve research reactor safety.
СОВРЕМЕННОЕ СОСТОЯНИЕ И ПЕРСПЕКТИВЫ
ИССЛЕДОВАТЕЛЬСКИХ ЯДЕРНЫХ РЕАКТОРОВ В СССР

АННОТАЦИЯ
В настоящее время эксплуатирующиеся в СССР исследовательские реакторы (ИР), широко используются для исследований в различных областях науки, а также для прикладных задач. В основном это реакторы бассейнового типа. Поскольку реконструкция ИР существенно дешевле создания новых, подавляющее большинство их реконструировано со значительным расширением экспериментальных возможностей. Перспективы ИР в СССР связаны с продолжением модернизацией действующих ИР и с созданием новых мощных ИР, проекты которых разрабатываются. Некоторые из них уже сооружаются (например, реактор ПИК). Эти проекты развивают концепции советских ИР, основанные на конструкциях корпусного и канального в бассейне типов реакторов. Имеются и другие технические идеи. Безопасность ИР, в целом, удовлетворяет современным требованиям, однако длительное время эксплуатации, близость жилых массивов и ряд других особенностей ИР предъявляют повышенные требования к их безопасности. Предпринимаются ряд мер организационного и технического характера для повышения безопасности ИР.

I. ВВЕДЕНИЕ
Советские ИР имеют более, чем сорокалетнюю историю развития. Первый советский ИР – Ф-1 – достиг критичности 25 декабря 1946 года и с тех пор успешно используется для проведения различных исследований. Он имеет самую большую продолжительность эксплуатации из всех когда-либо созданных ядерных реакторов во всем мире.

ИР – сложные, дорогостоящие установки, обладающие повышенной опасностью. В то же время они являются, по-видимому, самым экономичным источником нейтронов высокой интенсивности, что делает их использование крайне привлекательным для исследований по нейтронной и ядерной физике, физике твердого тела, радиационной биологии и медицине, нейтронно-
активационному анализу. Ясно также, что развитие атомной энергетики немыслимо без проведения широких исследований по радиационному материаловедению, в т.ч. и в аварийных и переходных режимах. Все эти задачи решались и продолжают решаться в СССР на ИР. Некоторые из них приведены в таблице I. Кроме этих реакторов в СССР действуют и другие ИР, а также некоторое количество реакторов специального назначения и опытных энергетических реакторов.

Основным типом советских ИР являются, как и во всем мире, реакторы бассейнового типа. Другим типом реакторов, получившим развитие лишь в СССР, являются реакторы канального типа в бассейне — ИР и МИР (реактор этого типа — "Мария" — построен также в ПНР). Использование этих реакторов оказалось исключительно удобным для проведения петлевых исследований. Все советские реакторы построены в основном в 60-е годы, а в дальнейшем их экспериментальные возможности расширялись за счет реконструкций [I].

2. ПОВЫШЕНИЕ ПАРАМЕТРОВ СОВЕТСКИХ ИР ЗА ПОСЛЕДНИЕ ГОДЫ

Как уже отмечалось, основной тип ИР в СССР — реакторы бассейнового типа. Эти реакторы обеспечивают достижение уровня мощности до ~30 МВт при физически оптимальных размерах активной зоны. Они до сих пор продолжают сохранять немалую привлекательность для экспериментаторов, поскольку им присущи следующие основные преимущества:

- простота конструкции;
- безопасность эксплуатации, подтвержденная многочисленными расчетными и экспериментальными исследованиями;
- возможность многоцелевого использования;
- высокая приспособляемость к новым экспериментам;
- относительно небольшие расходы на создание реактора и его эксплуатацию.

Реакторы бассейнового типа, обладающие высокой доступностью активной зоны, могут быть реконструированы легче и с более значительным расширением экспериментальных возможнос-
Таблица I
Исследовательские реакторы СССР

<table>
<thead>
<tr>
<th>Название</th>
<th>Дата пуска (реконструкции)</th>
<th>Мощность, MBт</th>
<th>Тип</th>
</tr>
</thead>
<tbody>
<tr>
<td>Режектоны со стационарной плотностью потока нейтронов</td>
<td></td>
<td></td>
<td></td>
</tr>
<tr>
<td>Ф-I</td>
<td>26.12.1946</td>
<td>0,024</td>
<td>графитовый</td>
</tr>
<tr>
<td>ВВР-СМ</td>
<td>1959 (1971)</td>
<td>10</td>
<td>баковый</td>
</tr>
<tr>
<td>ВВР-M (Гатчина)</td>
<td>1959</td>
<td>18</td>
<td>бассейновый</td>
</tr>
<tr>
<td>ИРТ (Тбилиси)</td>
<td>1959 (1973)</td>
<td>5</td>
<td>бассейновый</td>
</tr>
<tr>
<td>БР-I0</td>
<td>1959 (1982)</td>
<td>8</td>
<td>жидкокометаллический</td>
</tr>
<tr>
<td>ВВР-M (Кiev)</td>
<td>1960</td>
<td>10</td>
<td>баковый</td>
</tr>
<tr>
<td>ИРТ (Рига)</td>
<td>1961 (1975)</td>
<td>5</td>
<td>бассейновый</td>
</tr>
<tr>
<td>СМ-2</td>
<td>1961 (1974)</td>
<td>100</td>
<td>корпусной</td>
</tr>
<tr>
<td>ИРТ (Минск)</td>
<td>1962 (1977)</td>
<td>5</td>
<td>бассейновый</td>
</tr>
<tr>
<td>МР</td>
<td>1963 (1967)</td>
<td>40</td>
<td>канальный в бассейне</td>
</tr>
<tr>
<td>БВР-Ц</td>
<td>1964</td>
<td>13</td>
<td>баковый</td>
</tr>
<tr>
<td>МИР</td>
<td>1966</td>
<td>100</td>
<td>канальный в бассейне</td>
</tr>
<tr>
<td>ИВВ-2М</td>
<td>1966 (1983)</td>
<td>15</td>
<td>бассейновый</td>
</tr>
<tr>
<td>ВВР-K</td>
<td>1967</td>
<td>10</td>
<td>баковый</td>
</tr>
<tr>
<td>ИРТ (Москва, МИФИ)</td>
<td>1967 (1975)</td>
<td>2,5</td>
<td>бассейновый</td>
</tr>
<tr>
<td>РГ-1М</td>
<td>1970 (1975)</td>
<td>0,1</td>
<td>бассейновый</td>
</tr>
<tr>
<td>РБТ-6</td>
<td>1975</td>
<td>6</td>
<td>бассейновый</td>
</tr>
<tr>
<td>ИР-8</td>
<td>1981</td>
<td>8</td>
<td>бассейновый</td>
</tr>
<tr>
<td>Аргус</td>
<td>1981</td>
<td>0,05</td>
<td>гомогенный</td>
</tr>
<tr>
<td>РБТ-I0-1</td>
<td>1983</td>
<td>10</td>
<td>бассейновый</td>
</tr>
<tr>
<td>РБТ-I0-2</td>
<td>1984</td>
<td>10</td>
<td>бассейновый</td>
</tr>
</tbody>
</table>

Импульсные и пульсирующие реакторы

<table>
<thead>
<tr>
<th>Название</th>
<th>Дата пуска</th>
<th>Мощность, MBт</th>
</tr>
</thead>
<tbody>
<tr>
<td>ИГР</td>
<td>1960</td>
<td>10^{-22}</td>
</tr>
<tr>
<td>Гидра</td>
<td>1972</td>
<td>5.10^{21}</td>
</tr>
<tr>
<td>ИБР-30</td>
<td>1969</td>
<td>10^{18}</td>
</tr>
<tr>
<td>ИБР-2</td>
<td>1977</td>
<td>4.10^{19}</td>
</tr>
</tbody>
</table>

Вместо мощности приведена максимальная плотность потока нейтронов (м^{-2}.с^{-1}).
тей, чем реакторы других типов, поэтому большинство советских ИР этого типа было реконструировано. При этом решались следующие задачи: увеличение плотностей потоков нейтронов в экспериментальных устройствах; создание новых экспериментальных установок; ремонт и замена устаревшего оборудования.

Эти задачи решались индивидуально в каждом конкретном случае, однако можно выделить следующие общие для многих ИР пути повышения параметров:

- разработка новых тепловыделяющих сборок (ТВС) с более развитой поверхностью теплоотдачи, что привело к существенному увеличению удельных мощностей. Созданные в СССР ТВС типа ВВР-5М с трубчатыми тепловыделяющими элементами (твэлами) [2] имеют удельную поверхность теплоотдачи, равную 6,6 см$^{-1}$, что является рекордной величиной. Созданы и другие ТВС с приблизительно таким же развитием поверхности теплоотдачи [3] (таблица П). При использовании ТВС с созданными по оригинальной советской технологии твэлами (в отличие от ТВС типа MTR) в активную зону вводится только такое количество конструкционных материалов, которое необходимо для изготовления самих твэлов, что и явилось основной причиной обеспечения рекордных величин удельной поверхности теплоотдачи. Использование этих твэлов диспергированного типа почти во всех ИР с 1952 г. показало их высокую надежность;

- увеличение концентрации урана-235 в активной зоне, что позволяет уменьшать критический объем, а также повышать величину качества - отношения плотности потока нейтронов к мощности реактора [4]. В новых ТВС, предназначенных для применения в бассейновых реакторах, концентрация урана-235 в активной зоне доведена до 130 г/л при обогащении урана, равном 90%. Эта же мера приводит к увеличению объема экспериментальных работ при сохранении неизменным выгорания топлива в выгружаемых из реактора ТВС;

- широкое использование бериллиевого отражателя, что также приводит к существенному повышению качества реактора. Кроме того, бериллий позволяет значительно увеличить отно-
Таблица II

Современные ТВС советских ИР

<table>
<thead>
<tr>
<th>Параметр</th>
<th>Тип ТВС</th>
<th>Кассеты с твэлами МИ МК-106</th>
<th>ВВР-М2</th>
<th>ИРТ-2М</th>
<th>ВВР-М5</th>
<th>ИРТ-3М</th>
<th>ИВВ-2М</th>
<th>МР (МИР)</th>
</tr>
</thead>
<tbody>
<tr>
<td>Обогащение по урану-235, %</td>
<td>I0</td>
<td>36</td>
<td>36</td>
<td>90</td>
<td>36</td>
<td>90</td>
<td>36</td>
<td>90</td>
</tr>
<tr>
<td>Длина активного слоя, см</td>
<td>50</td>
<td>60</td>
<td>60</td>
<td>50</td>
<td>60</td>
<td>58</td>
<td>60</td>
<td>50</td>
</tr>
<tr>
<td>Удельная загрузка топлива a), г/л</td>
<td>50</td>
<td>70,6</td>
<td>77,6</td>
<td>124</td>
<td>123</td>
<td>101</td>
<td>119</td>
<td>130</td>
</tr>
<tr>
<td>Толщина твэла, мм</td>
<td>10 Б</td>
<td>2,5</td>
<td>2</td>
<td>1,25</td>
<td>1,4</td>
<td>1,35</td>
<td>2,65</td>
<td>2,0</td>
</tr>
<tr>
<td>Объемная доля воды</td>
<td>0,7</td>
<td>0,542</td>
<td>0,726</td>
<td>0,579</td>
<td>0,624</td>
<td>0,620</td>
<td>0,61</td>
<td>3,67</td>
</tr>
<tr>
<td>Удельная поверхность теплоотдачи, см⁻¹</td>
<td>0,98</td>
<td>3,67</td>
<td>2,65</td>
<td>6,6</td>
<td>5,25</td>
<td>4,50</td>
<td>3,86</td>
<td>3,67</td>
</tr>
</tbody>
</table>

a) в качестве топливной композиции в твэлах всех ТВС используется двуокись урана в алюминиевой матрице;

b) кассеты практически не используются в современных советских ИР и данные по ним приведены в таблице для сравнения;

в) диаметр стержневого твэла.
шению плотности потока тепловых нейтронов к плотности потока быстрых в отражателе реактора, что крайне важно для многих экспериментов;
- повышение мощности реактора за счет расширения возможностей системы охлаждения.

Из 12 бассейновых ИР многоцелевого назначения, сооруженных в СССР в конце 50-х - начале 60-х годов, реконструировано 8, и 4 планируется реконструировать в ближайшее время. В качестве примера реконструкции серийного реактора можно привести реконструкцию реактора ВВР-С в Ташкенте с повышением мощности с 2 до 10 МВт [5] . В основе ее лежал переход к использованию ТВС типа ИРТ-2М, которые имеют значительные преимущества по сравнению с ранее применявшимися кассетами с твэлами ЭК-10.

Другим примером последовательного расширения экспериментальных возможностей ИР является эволюция реактора ИРТ в ИАЭ имени И.В.Курчатова. Он был создан в 1957 году и имел проектную мощность 1 МВт. В 1965 году была проведена реконструкция реактора, основанная на применении ТВС типа ИРТ-М вместо кассет с твэлами ЭК-10 и переходе к новой системе охлаждения активной зоны, что позволило поднять мощность реактора до 5 МВт, а плотность потока тепловых нейтронов в ловушке до 1,8.10^{18} м^{-2}.с^{-1} [6] . В 1971 году реконструкция была продолжена: осуществлен переход на новые ТВС ИРТ-2М и расширена система отвода тепла. Это позволило увеличить мощность реактора до 8 МВт. В конце 1978 года на реакторе была обнаружена течь алюминиевого бака и деформация его стенок. Было принято решение об остановке реактора и проведении новой реконструкции, суть которой состояла в следующем [7]:
- алюминиевый бак реактора заменяется на бак из коррозионностойкой стали. Новый бак имеет большую высоту;
- внутри нового бака монтируется активная зона со стационарным бериллиевым отражателем, а также эжектор и другие компоненты системы отвода тепла (рис. 1). Активная зона реактора собирается из ТВС типа ИРТ-3М с высоким коэффициентом размножения и малой длиной миграции нейтронов, что позволяет
Рис.1. Продольный разрез реактора ИР-8: 1-каналы блоков детектирования СУЗ, 2-вертикальные экспериментальные каналы (ЭК), 3-каналы со стержнями СУЗ, 4-бак реактора, 5-корпус реактора, 6-канал горизонтальный экспериментальный, 7-сталевые экраны, 8-тепловыделяющая сборка, 9-бериллиевый отражатель, 10-промежуточное дно, II-канал с источником ультрахолодных нейтронов, 12-эжектор, 13-трубопровод напорный, 14-емкость задерживающая, 15-вертикальная перегородка, 16-хранилище отработанных ТВС, 17-трубопровод всасывающий, 18-контейнер перегрузочный.
получить геометрически малые размеры активной зоны и боль-
шую утечку нейтронов в бериллиевый отражатель, состоящий из
стационарной части, которая вместе со сменными блоками имеет
толщину 30 см.

Фактически это означает не реконструкцию, а создание
нового ИР. Этот реактор получил название ИР-8.

Во время энергопуска reactora ИР-8 в 1981 году были
выполнены измерения абсолютных значений плотностей потоков.
При номинальной мощности реактора 8 МВт максимальная плот-
ность потока тепловых нейтронов в отражателе равна
2,3.10\(^{18}\) м\(^{-2}\).с\(^{-1}\). Плотность потоков тепловых нейтронов на
выходе из ГЭКов достигла рекордного значения по сравнению
со всеми другими реакторами СССР, работающими даже при боль-
шей мощности, включая и реактор СМ-2. Плотность потока на
выходе из ГЭКов достигает 1,8.10\(^{14}\) м\(^{-2}\).с\(^{-1}\), что в 4 раза
выше, чем на старом реакторе ИРТ-М. Такая плотность потока
блика к величине, достигнутой на одном из лучших в мире
специализированном реакторе HFR мощностью 57 МВт в Гренобле
для исследований в области физики твердого тела и ядерной
физике, и превосходит плотность потока на Окриджском реак-
торе HFIR мощностью 100 МВт. Это связано с высоким качест-
вом реактора ИР-8, а также относительно малой толщиной эф-
fективной биологической защиты. Таким образом, от первона-
чального уровня 1 МВт мощность реактора повысилась до 8 МВт,
a по своим экспериментальным возможностям он почти достиг
уровня высокопоточного реактора.

Примером современного многопетлевого материаловедчес-
кого реактора является реактор МР в ИАЭ [8]. Он является
реактором оригинальной отечественной конструкции — это ре-
актор канального типа, погруженный в бассейн с водой
(рис. 2). Реактор МР, достигший критичности в декабре 1963
года, начал работать в 1964 году на проектной мощности
20 МВт. Благодаря конструктивным особенностям реактора МР
(полной разборяемости кладки активной зоны и отражателя,
отсутствию корпуса), в дальнейшем оказалось возможным при не-
значительных затратах существенно расширить эксперименталь-
Рис. 2. Продольный разрез реактора МР: 1-корпус кладки реактора, 2-берилиевые и графитовые блоки, 3-рабочий канал с неподвижной ТВС, 4-рабочий канал с подвижной ТВС, 5-опорная плита стаканов рабочих и петлевых каналов, 6-коллекторы контура охлаждения ТВС, 7-прямоточный U-образный петлевой канал, 8-каналы блоков детектирования СУЗ, 9-тележка с приводами СУЗ и подвижных ТВС, 10-трубопроводы контура охлаждения кладки, 11-канал со стержнем СУЗ.
ные возможности реактора. Количество петлевых каналов для испытаний опытных ТВС, которые могут быть установлены в реактор в настоящее время, доведено до 26 вместо 13 в 1964 году. Мощность реактора (с петлевыми каналами) может достигать 50 МВт. Расширение экспериментальных возможностей действующих петлевых реакторов, в частности увеличение в них количества петлевых и материаловедческих каналов, имеет важное значение, поскольку затраты на сооружение каждого нового петлевого реактора весьма значительны. Таким образом, модернизация реактора МР, проведенная в несколько этапов, фактически равнозначна созданию второго петлевого реактора.

Конструктивные особенности реактора МР обеспечили возможность осуществления на нем самых разнообразных экспериментов, связанных с созданием ядерных реакторов, в том числе и тех, необходимость в которых возникла после пуска реактора. Последнее показывает его высокую мобильность.

3. ПЕРСПЕКТИВЫ РАЗВИТИЯ ИР В СССР

При обсуждении вопроса о создании современного высокооточного реактора следует иметь в виду следующие обстоятельства:

- сооружение такого реактора должно способствовать развитию качественно новых фундаментальных и прикладных исследований;
- достаточным основанием для принятия решения о его строительстве может быть только компетентная, перспективная программа работ;
- использование реактора будет эффективным только в том случае, если на нем будет установлено уникальное экспериментальное оборудование, обеспечивающее полное использование всех его возможностей.

Немало задач, хотя и на более низком уровне, могут быть решены на ИР с плотностью потока порядка $10^{18}$ м$^{-2}$·с$^{-1}$. Необходимо максимально использовать возможности модернизированных реакторов.

Среди возможных направлений использования ИР весьма
важным является радиационное материаловедение.

Удобство проведения петлевых испытаний в реакторах типа МР и МИР привело к естественному желанию — при сохранении основной особенности конструктивной схемы — "каналы в бассейне" существенно повысить плотности потоков нейтронов в реакторе. За счет тесного расположения рабочих каналов и исключения бериллиевого замедлителя удается обеспечить более жесткий спектр нейтронов, что очень важно для проведения материаловедческих исследований, а за счет повышения удельных тепловых нагрузок поднять абсолютную величину плотности потоков быстрых нейтронов в активной зоне и тепловых в отражателе до $\sim 10^{19}$ м$^{-2}$·с$^{-1}$. Описанию конструкции реактора, реализующего эти идеи и разрабатываемого в настоящее время в СССР, посвящен специальный доклад.

В СССР начаты проработки высокопоточного реактора ВРТТ мощностью 100 + 250 МВт. В этом реакторе топливо в виде твэлов стержневого типа циркулирует по замкнутому пути и на ограниченном участке создаются условия для протекания цепной реакции за счет применения хорошего отражателя. В основе этого реактора лежит идея циклокотла, предложенная С.М.Фейнбергом еще в шестидесятые годы [9]. Сложность эксплуатации реактора ВРТТ-250 может быть связана с резким уменьшением доли запаздывающих нейтронов (до $\sim 0,03\%$) и возможными вследствие этого значительными флуктуациями мощности. При мощности 250 МВт и объеме активной зоны 50 л плотность потока быстрых нейтронов в экспериментальных каналах реактора достигает $7,5 \cdot 10^{19}$ м$^{-2}$·с$^{-1}$, что достаточно для проведения материаловедческих исследований по всем реакторным направлениям, включая реакторы на быстрых нейтронах.

Сооружаемый в настоящее время в Гатчине реактор ПИК мощностью 100 МВт предназначен в основном для экспериментов на выведенных пучках нейтронов, причем для увеличения плотности потока нейтронов на выходе из канала толщина защиты уменьшена, что даст возможность достигнуть уровня $3 \cdot 10^{15}$ м$^{-2}$·с$^{-1}$, который примерно на порядок выше, чем в лучших современных реакторах.
Описанию некоторых аспектов его безопасности посвящен специальный доклад.

Для проведения испытаний твэлов и фрагментов ТВС действующих и перспективных реакторов в аварийных условиях проектируется специализированный реактор ПРИМА, который предполагается создать в НИИАР. Описанию конструкции и экспериментальных возможностей реактора также посвящен специальный доклад. Мощность реактора в стационарном режиме 100 МВт, причем имеется возможность работы реактора и в импульсном режиме. Особенностью реактора является возможность проведения исследования режимов, которые имитируют аварии с тяжелыми последствиями, вплоть до полного оплавления экспериментальных ТВС.

4. БЕЗОПАСНОСТЬ ИР

В последние годы особую актуальность приобрели вопросы, связанные с безопасностью ИР. Это связано со следующими основными аспектами. Часть исследовательских реакторов расположена либо в черте крупных городов, либо вблизи их границ, что существенно повышает требования к обеспечению безаварийной работы с тем, чтобы уменьшить радиационный риск для населения. Большинство исследовательских реакторов разрабатывались и создавались на рубеже 50-х - начала 60-х годов (таблица I). В то время не было жестких требований к определению сроков службы основного оборудования реакторов, анализу надежности аппаратуры СУЗ и ответов СУЗ на возможные неисправности. Мощности большинства реакторов повышались, в настоящее время они достигли значений, в несколько раз превышающих проектные. В отличие от действующих энергетических реакторов, число типов которых невелико, для исследовательских реакторов характерно несравненно большее разнообразие конструкций, связанное со спецификой назначений и выбором оптимальной мощности. Анализ безопасности для каждого типа реактора в связи с этим имеет свои особенности.

Особенности ИР с точки зрения безопасности, проанализированные в работе [10], показывают, что, несмотря на су-
яственно меньшие мощности и накопление осколков деления, чем в реакторах АЭС, ИР имеют ряд особенностей, делающих их достаточно опасными (высокое обогащение урана, большой запас реактивности и т.д.).

Безопасность реакторных установок обеспечивается техническими мерами и организационными мероприятиями. При безусловном выполнении требований регламентирующих документов и контролирующих органов безопасность в сильной степени зависит от надежности основного оборудования, систем электрообеспечения, локализующих систем, обеспечивающих минимальное воздействие на окружающую среду и население прилегающего района, и организации надлежащего хранения отработавшего топлива. Разумное резервирование оборудования и систем безопасности, точное соблюдение регламента их проверок и поверок средств измерения позволяют поддерживать надежность на высоком уровне.

Важным моментом обеспечения безопасной эксплуатации исследовательских реакторов является квалификация, подготовка и переподготовка персонала реактора, регулярные тренировки его действиям при возникновении возможных аварийных ситуаций и аварий. Для этого необходимо форсировать разработку и создание всережимных математических моделей реакторных установок и соответствующих тренажеров. Новые специальные технические средства нужны также для проведения исследований в области динамики реакторных установок и управления, они должны обеспечить в реальном масштабе времени моделирование и исследование аварийных ситуаций, изучение возможных причин (исходных событий) возникновения, механизмов развития и последствий проектных и любых запроектных (гипотетических) аварий, а также способов их предупреждения и локализации (минимизации отрицательных последствий), исследование и отработку методов режимной диагностики и взаимодействия человек-машина. Все это вместе с имеющимся опытом позволит решить важнейшую проблему повышения надежности исследовательских реакторов и создание предельно безопасного реактора для эксплуатационного персонала, населе-
ния и окружающей среды, реактора с максимальным использованием внутренних пассивных эффективных и надежных средств предотвращения аварийных событий.

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Operating and Safety Features of the MAPLE Reactor

by

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ABSTRACT

Atomic Energy of Canada Limited is developing the MAPLE reactor as a multipurpose neutron source for small-scale materials analysis and testing, for radioisotope production, and for scientific, medical, and commercial applications of neutron beams. The basic concept incorporates a compact H₂O-cooled and -moderated core fueled by low-enrichment-uranium fuel rods within a D₀ primary reflector. This results in strong fast and intermediate neutron fluxes per unit power within the core and very strong thermal neutron fluxes at irradiation facilities in the core and reflector. Notwithstanding MAPLE's novel configuration, its safety characteristics are similar to those of typical pool-type reactors fueled with low-enrichment uranium. MAPLE facilities may be oriented to various applications of contemporary interest. A 1-MW MAPLE facility can generate key medical radioisotopes, e.g. ⁹⁹Mo, for a regional health-care system and also perform sensitive neutron-activation analysis. Via the production of peak perturbed thermal neutron fluxes above 1 x 10¹⁸ n·m⁻²·s⁻¹, a MAPLE of 6 MW or more can facilitate advanced neutron-beam research. At higher operating powers (10-20 MW), MAPLE can serve as the basis for a fully independent national nuclear program.

1. INTRODUCTION

For several years, Atomic Energy of Canada Limited (AECL) has been developing the MAPLE (Multipurpose Applied Physics Lattice Experimental) reactor concept [1-4] as an efficient multipurpose neutron source for radioisotope production, for small-scale materials testing and analysis, and for scientific, medical and commercial applications of neutron beams. The basic MAPLE concept incorporates low-enrichment-uranium (LEU) fuel rods within a compact H₂O-cooled and -moderated core that is surrounded radially by a D₀ primary reflector. The combination of a small core (about 63 L volume) within a spacious D₀ reflector (1300 to 1700 L) results in the generation of very strong fast and intermediate neutron fluxes per unit power within the core and the availability of unusually strong thermal neutron fluxes at irradiation facilities in the core and reflector.

AECL is currently building the MAPLE-X10 prototype facility at the Chalk River Nuclear Laboratories to commercially produce key short-lived radioisotopes (such as ⁹⁹Mo, ¹²⁵I and ¹⁹²Ir) as well as to demonstrate MAPLE technology. In support of MAPLE-X10 construction and licensing, an AECL development program is underway to verify the performance of reactor components and characteristics unique to the MAPLE concept during both normal operating and upset conditions.
2. DESIGN FEATURES

MAPLE reactors derived from the MAPLE-X10 prototype are distinguished by the open-tank reactor assembly (Fig. 1 and 2) which may be installed in a new or existing pool. Principal MAPLE design features are:

1) **Compact, Light-Water-Cooled and -Moderated Core** - The grid plate of the reactor assembly accommodates a hexagonal arrangement of 19 core modules. Core modules comprise either a fuel assembly installed in a flow tube or an irradiation rig. Standard fuel assemblies contain 36 rods in a close-packed hexagonal array; each assembly is mounted on a hanger rod that can be locked into a hexagonal-shaped zirconium-alloy flow tube. The six reactivity-control assemblies, which are located in the middle-outer lattice positions, are similar to standard assemblies except they contain either 18 or 36 rods within circular flow tubes. Reactivity compensation and reactor shutdown are provided by hafnium absorber cylinders that insert into the water annuli outside the circular flow tubes.

2) **LEU-Silicide Fuel Particles Dispersed in Aluminum Rods** - MAPLE fuel comprises low-enrichment (about 19.7 weight percent $^{235}$U in total uranium) U$_3$Si particles dispersed in an aluminum matrix; the fuel meat
is extruded to form rods 5.48 mm or 6.35 mm in diameter and then coextrusion clad with aluminum of thickness 0.76 mm. Standard fuel assemblies and 18-rod control assemblies use the larger rods; 36-rod control assemblies based on the smaller rods are employed to extend the operation of the MAPLE-X10-type core from 12 MW to 20 MW.

3) **Heavy-Water Primary Reflector** - Radially surrounding the core region is an annular cylindrical zirconium-alloy vessel (1.4 to 1.6 m dia. by 0.9 m high) containing heavy water. It is penetrated by up to eight horizontal beam-tubes and a custom-built array of fixed vertical irradiation tubes for fuel-test loops, a cold-neutron source, silicon doping, neutron activation analysis, radioisotope production, etc.

4) **Core Access via an Open Chimney** - An aluminum chimney mounted on the reflector tank receives primary cooling water from the reactor core plus a smaller flow from the pool and directs them to the primary-coolant piping. It also houses the reactivity-control devices and provides access to the core for fuel and irradiation-target handling.

5) **Primary Cooling** - Primary cooling water enters the reactor assembly via a stainless-steel inlet plenum which supports the other reactor-assembly components. It is pumped upward through the flow tubes past the fuel assemblies and into the chimney. The pump draws the heated water from two nozzles in the chimney through the outlet piping to a plate-type heat exchanger and returns cooled water to the reactor.

6) **Control System** - A dual/redundant commercial digital-computer system performs power regulation, process control, and data collection. It senses the state of the reactor and its process systems by signals received from field instruments and establishes control by vertically positioning a set of three hafnium reactivity-control mechanisms and by manipulating other process control devices. The operator employs a dual keyboard/display console to interact with the control system in either automatic or manual control mode. The control system precludes operator action that might create an unsafe state.

7) **Shutdown Systems** - One of two independent reactor shutdown systems is physically separate from the reactor regulation system; in response to specified abnormal conditions, it inserts a set of three hafnium absorbers that are normally held poised above the core by hydraulic cylinders. The second system releases a magnetic latch to override control-system action and rapidly insert the other three absorbers.

3. **MAPLE PERFORMANCE**

For a MAPLE reactor operating at 10 MW, the calculated [5] peak thermal neutron flux is $4 \times 10^{18} \text{ n}\cdot\text{m}^{-2}\cdot\text{s}^{-1}$ in a central flux trap and $2 \times 10^{18} \text{ n}\cdot\text{m}^{-2}\cdot\text{s}^{-1}$ in the reflector. These very strong thermal neutron flux levels arise from the combination of the compact core and the spacious D$_2$
reflector. While traditional pool-type reactors produce similar thermal fluxes in a central flux trap, the peak thermal flux levels available in their reflectors is typically one half that of MAPLE; moreover, the near-peak thermal fluxes of light-water, graphite, and beryllium reflectors persist over a much smaller volume, relative to heavy water.

Perturbed neutron fluxes have also been computed [5] for a 10-MW MAPLE reactor with 70-mm wide by 140-mm high beam tubes centred on the axial neutron flux peak (50 mm below core mid-plane) and arranged as shown in Fig. 1. At the entrance to a tangential beam tube 100 mm from the core wall, the thermal neutron flux is $1.6 \times 10^{18} \text{n} \cdot \text{m}^{-2} \cdot \text{s}^{-1}$ while the intermediate ($0.625 \text{eV} \leq E < 5.9 \text{keV}$) flux is $0.51 \times 10^{18} \text{n} \cdot \text{m}^{-2} \cdot \text{s}^{-1}$ and the fast ($5.9 \text{keV} \leq E < 10 \text{MeV}$) flux is $0.35 \times 10^{18} \text{n} \cdot \text{m}^{-2} \cdot \text{s}^{-1}$. The improved purity further into the heavy-water tank is indicated by the neutron fluxes for a tangential tube at 200 mm, $1.4 \times 10^{18} \text{n} \cdot \text{m}^{-2} \cdot \text{s}^{-1}$, $0.24 \times 10^{18} \text{n} \cdot \text{m}^{-2} \cdot \text{s}^{-1}$, and $0.06 \times 10^{18} \text{n} \cdot \text{m}^{-2} \cdot \text{s}^{-1}$ for thermal, intermediate, and fast, respectively. For either beam tube, the corresponding thermal neutron flux calculated [5] at the working face of the beam hall (allowing for 350 mm of light water between the reflector tank and the pool wall and a heavy-concrete shield thickness of 2000 mm) is $2.0 \times 10^{12} \text{n} \cdot \text{m}^{-2} \cdot \text{s}^{-1}$.

4. MAPLE SAFETY CHARACTERISTICS

A review [6] of the safety analysis of the MAPLE-X10 prototype reactor shows that its operation poses an acceptable risk to employees and to the public. MAPLE-X10 licensing is based on Canadian reactor safety philosophy and practice, which is consistent with international requirements [7]. The licensee is responsible for following adequate safety procedures. This especially requires good design practice, incorporating "defence in depth" (e.g. via multiple barriers to fission-product release) and using redundancy, diversity, separation, and fail-safe components. Important inherent and engineered MAPLE safety characteristics are:

- The large reactor pool provides shielding plus extensive decay heat removal during upset conditions.

- The undermoderated core design provides stability by appropriately negative reactivity coefficients of void, coolant temperature, and fuel temperature.

- The fuel has excellent heat-transfer properties at high rod ratings (up to 100 kW/m) up to high burnup (93% initial $^{235}\text{U}$).

- Upward-forced core cooling allows higher flows and pressures than downward-forced and avoids flow reversal during the transition to natural-circulation cooling.

- A large margin exists between normal operating conditions and the critical heat flux because the core operates at low temperature and pressure with highly subcooled water.
In the event of loss of flow, the primary cooling system is designed to remove decay heat by natural water circulation, either through the primary heat exchanger or through the pool.

A useful perspective on MAPLE safety behaviour is provided by the comparison of its principal safety-related physics parameters with those of the IAEA Generic 10-MW reactor [8]. The generic reactor closely represents sixteen facilities currently operating at powers of 5-15 MW and generally represents roughly one hundred operational pool-type and tank-in-pool reactors of power levels up to 45 MW. The summary presented in Table I shows similar safety-system reactivity worths and shutdown margins, despite major differences in fuel-element design and core configuration. Thus, the MAPLE response to typical upset conditions is expected to be similar to that of current pool-type reactors after their conversion to LEU fuel.

The response of MAPLE reactors to postulated reactivity accidents is being investigated using TANK [9], a 2D, 2G space-time reactor kinetics computer code. TANK dynamically adjusts the void and temperature status of individual fuel channels using the heat-transfer package of SPORTS-M [10], a thermalhydraulics code developed for small reactors operating at high power density with low temperature and pressure coolant. TANK's dynamic spatial representation of the effect of void and temperature changes reduces the uncertainties inherent in large-scale averaging of neutronic properties for point-kinetics models. TANK predictions have been verified by comparing them with SPERT-1B experiments [11]; TANK predictions compare favourably with experimental measurements.

MAPLE reactor behaviour as predicted by TANK is demonstrated by the results of a loss-of-regulation accident scenario with failure to shut down, starting from low power [12]. The reactor power peaks at 21 MW at about 100 s from accident initiation, then subsequently reduces to about 14 MW. As all peak temperatures are well below any failure thresholds, the reactor is expected to survive without damage. In this particular case, coolant-density decrease and fuel-temperature increase stabilize the power and substantial void generation is not a factor. The analysis shows the reactor's ability to cope with postulated severe reactivity accidents.

5.0 MAPLE APPLICATIONS

Depending on the program requirements of the host institute, MAPLE multipurpose facilities can be oriented to radioisotope production, beam-tube applications, or materials testing [4].
TABLE I
COMPARISON OF MAPLE WITH GENERIC POOL-TYPE REACTOR

<table>
<thead>
<tr>
<th></th>
<th>MAPLE</th>
<th>IAEA Generic Reactor</th>
</tr>
</thead>
<tbody>
<tr>
<td><strong>Fuel</strong></td>
<td></td>
<td></td>
</tr>
<tr>
<td>Material Enrichment (% U-235)</td>
<td>U Si-Al 19.75</td>
<td>U Si-Al 19.75 Plates</td>
</tr>
<tr>
<td>Geometry</td>
<td>Rods</td>
<td></td>
</tr>
<tr>
<td><strong>Core</strong></td>
<td></td>
<td></td>
</tr>
<tr>
<td>Moderator/Coolant</td>
<td></td>
<td></td>
</tr>
<tr>
<td>Geometry</td>
<td>H2O</td>
<td>H2O</td>
</tr>
<tr>
<td>Volume (L)</td>
<td>63</td>
<td>Rectangular 112</td>
</tr>
<tr>
<td>Max. Fissile Load (kg)</td>
<td>6.3</td>
<td>10.4</td>
</tr>
<tr>
<td><strong>Reflector</strong></td>
<td></td>
<td></td>
</tr>
<tr>
<td>Material</td>
<td>D2O</td>
<td>H2O, Graphite</td>
</tr>
<tr>
<td>Peak Thermal Flux at 10 MW\textsubscript{th} (n\textsuperscript{-}\textsuperscript{m}^2\textsuperscript{-}\textsuperscript{s}^{-1})</td>
<td>2.0 \times 10^{10}</td>
<td>1.1 \times 10^{10}</td>
</tr>
<tr>
<td><strong>Reactivity Control Systems</strong></td>
<td>Hf Cylinders 3 Absorbers</td>
<td>Ag-in-Cd Forks 5 Absorbers Override Reg. Abs.</td>
</tr>
<tr>
<td>Shutdown #1</td>
<td></td>
<td></td>
</tr>
<tr>
<td>Shutdown #2</td>
<td></td>
<td></td>
</tr>
<tr>
<td>Maximum Absorber</td>
<td>81</td>
<td>68</td>
</tr>
<tr>
<td>Average Absorber</td>
<td>57</td>
<td>34</td>
</tr>
<tr>
<td>Typical Excess Reactivity after Refueling</td>
<td>71</td>
<td>71</td>
</tr>
<tr>
<td>Shutdown System</td>
<td>170 (either)</td>
<td>170</td>
</tr>
<tr>
<td><strong>Reactivity Worth (MK)</strong></td>
<td></td>
<td></td>
</tr>
<tr>
<td>SS#1 (or SS#2) Fully Effective</td>
<td>99</td>
<td>98</td>
</tr>
<tr>
<td>SS#1 (or SS#2) with Maximum Absorber Ineffective</td>
<td>18</td>
<td>29</td>
</tr>
<tr>
<td>Minimum Acceptable (with single failure)</td>
<td>10</td>
<td></td>
</tr>
</tbody>
</table>

One of the principal contemporary applications of research reactors is local production of radioisotopes for nuclear medicine. Canadian-developed technology offers the basis for a practical system for \(^{99}\)Mo generation: irradiating molybdenum in an indigenous reactor, transporting it to major cities, extracting \(^{99m}\)Tc in centralized radiopharmacies, and distributing
$^{99m}$Tc compounds to regional hospitals and clinics. A 1-MW MAPLE can readily generate 2000 GBq $^{99}$Mo weekly as low-specific-activity $^{99}$Mo for subsequent processing via proven self-shielded $^{99m}$Tc-extraction units. It can also produce 100 GBq $^{131}$I and 0.3 GBq $^{125}$I weekly plus significant quantities of $^{18}$F thereby yielding a range of medically important radionuclides. Additionally, a 1-MW MAPLE furnishes thermal neutron fluxes of $2 \times 10^{17} \text{n}\cdot\text{m}^{-2}\cdot\text{s}^{-1}$ for sensitive neutron-activation analysis and offers beam currents of $1-2 \times 10^{11} \text{n}\cdot\text{m}^{-2}\cdot\text{s}^{-1}$ at the instrument working face, which makes it comparable to nuclear facilities that have made important contributions to many areas of basic research.

A major use of the neutron sources such as the MAPLE reactor involves extracted neutron beams for neutron radiography, neutron scattering, prompt gamma-ray neutron activation analysis, and boron neutron capture therapy. Each application has different requirements for neutron and gamma radiation fields. A MAPLE facility of 6-MW or more produces peak perturbed thermal neutron flux levels above $1 \times 10^{14} \text{n}\cdot\text{m}^{-2}\cdot\text{s}^{-1}$ at the beam-tube entrances and neutron currents above $1.5 \times 10^{12} \text{n}\cdot\text{m}^{-2}\cdot\text{s}^{-1}$ in the beam hall. This enables such a MAPLE facility to support a vigorous program of basic and applied research using neutron beams.

At higher operating powers, a MAPLE facility is capable of supporting a fully independent national nuclear program. At 10-20 MW, the MAPLE reactor will generate fast (> 0.8 MeV)-neutron fluxes of $1-2 \times 10^{15} \text{n}\cdot\text{m}^{-2}\cdot\text{s}^{-1}$ for studying materials-damage or proof-testing light-water-reactor fuel and peak (unperturbed) thermal-neutron flux levels of $2-4 \times 10^{15} \text{n}\cdot\text{m}^{-2}\cdot\text{s}^{-1}$ for proof-testing CANDU fuel. Moreover, all of the applications possible at lower power levels are significantly enhanced; with thermal neutron fluxes of up to $3 \times 10^{18} \text{n}\cdot\text{m}^{-2}\cdot\text{s}^{-1}$ available at the beam-tube entrances, very sensitive neutron-scattering experiments are feasible. Radioisotope production also becomes more efficient, with either reduced target mass or greater product quantities possible.

REFERENCES


THE ADVANCED NEUTRON SOURCE--
DESIGNING TO MEET THE NEEDS
OF THE USER COMMUNITY

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THE ADVANCED NEUTRON SOURCE--DESIGNING TO MEET
THE NEEDS OF THE USER COMMUNITY

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Abstract

The Advanced Neutron Source (ANS) is to be a multipurpose neutron research center, constructed around a high-flux reactor now being designed at the Oak Ridge National Laboratory (ORNL). Its primary purpose is to return the United States to the forefront of neutron scattering in the twenty-first century. Other research programs include nuclear and fundamental physics, isotope production, materials irradiation, and analytical chemistry. The ANS will be a unique and invaluable research tool because of the unprecedented neutron flux available from the high-intensity research reactor. But this reactor would be ineffective without world-class research facilities that allow the fullest utilization of the available neutrons. And, in turn, those research facilities will not produce new and exciting science without a broad population of users from all parts of the nation and the world, placed in a stimulating environment in which experiments can be effectively conducted and in which scientific exchange is encouraged. This paper discusses the measures being taken to ensure that the design of the ANS focuses not only on the reactor, but on providing the experiment and user support facilities needed to allow its effective use.

1. INTRODUCTION

The Advanced Neutron Source (ANS) is to be a multipurpose neutron research center, constructed around a high-flux reactor now being designed at the Oak Ridge National Laboratory (ORNL). Its primary purpose is to return the United States to the forefront of neutron scattering in the twenty-first century. It will provide state-of-the-art facilities for nuclear and fundamental physics research using neutron beams ranging from thermal to "ultra-cold" energies and specialized cold neutron beam facilities for analytical chemistry applications. Exceptional thermal flux irradiation facilities will be available for activation analysis and isotope production, with low fast neutron and gamma contamination levels. Although spatially limited, extremely high flux facilities will be available for epithermal and fast neutron irradiations, supporting structural materials irradiation programs and the production of californium, einsteinium, and other transuranic isotopes. The ANS is to be a national user's facility, catering to researchers from throughout the United States and the world.
Often, in designing such a facility, so much attention is given to the reactor itself that the research facilities do not allow the fullest use of the facility. Therefore, this paper addresses those early planning and design efforts that address the needs of the users, both in terms of experiment facilities and of the support facilities and programs for the users themselves. But first, an overall description of the reactor and facilities must be given.

The ANS reactor will be a compact, 300-MW, heavy-water-cooled, moderated and reflected reactor fueled with enriched uranium. Innovative arrangements of coolant and fuel elements will allow the production of a thermal neutron flux approaching $10^{20} \text{ m}^{-2} \cdot \text{s}^{-1}$ in the large heavy water reflector. Two annular fuel elements are provided, each with hundreds of involute-shaped fuel plates. The lower element has an outer radius smaller than the inner radius of the upper element (Fig. 1). Separate, parallel coolant paths lead into each element, reducing the effective heated length of the core. This allows a relatively thin fuel zone that results in a high efficiency for neutron leakage into the reflector while increasing the amount of heat that can be removed from the core. Control rods are located in the central hole of the core, minimizing their impact on the flux in the reflector. Pressurized-heavy-water coolant flows up through a core pressure boundary tube located just outside the fuel; no flow reversal is necessary in the transition to natural circulation for long-term decay heat removal. Two liquid deuterium cold sources are located in the reflector, each feeding seven horizontal cold beam guides and a single-slant, very-cold neutron (VCN) guide. *A graphite hot source located at the outer perimeter of the reflector feeds two beam tubes. Thermal beams include six tangential beam tubes, one through tube, and a radial supermirror guide.

Positions for fast neutron materials irradiation samples, including five instrumented capsules, are provided just inside the upper, wider fuel element. Positions for production of californium and einsteinium are located just outside the lower fuel element. Both are cooled by the pressurized-heavy-water coolant. Thermal irradiation facilities include static irradiation tubes and both hydraulic and pneumatic rabbit facilities. Regions near the outside of the reflector tank provide thermal fluxes in excess of $5 \cdot 10^{20} \text{ m}^{-2} \cdot \text{s}^{-1}$, with very low fast flux contamination and relatively low gamma heating rates. A single hydraulic rabbit tube located near the core pressure boundary tube provides a high epithermal component, and slant tubes allow irradiation of materials capsules in a similar flux region.

The ANS is designed around four main, interconnected structures (Fig. 2). The central reactor building is a cylindrical structure with a diameter of 60 m. This provides ample space for beams around the reactor shield structure located in the center. An attached guide hall, which provides experiment space along the horizontal cold guides, is a wedge-shaped structure with radial-segment bridge cranes moving across the guides. Reactor equipment, including the primary pumps and heat exchangers, is located in the reactor support building. An office building provides space and laboratories for users and for certain reactor staff members.

The elevations around the reactor pool are seen in Fig. 3. The ground floor is dedicated to beam use. The second floor (Fig. 4) is used by a variety of experimenters and is partially interrupted by the spent fuel pool and pipe chase. Preliminary concepts for beam facilities include the two VCN guides and a single slant cold guide for depth profiling work. One of the two VCN guides feeds a "neutron turbine" to provide ultra-cold neutrons, which are transmitted to a number of stations surrounding the turbine. The other may be used...
Fig. 1. ANS core and central irradiation facilities.
Fig. 3. Section of ANS reactor pool.
Fig. 4. Plan view of second floor experiment areas.
directly in the very-cold range, for example in neutron optics work. Other portions of the second floor are occupied by the lower structures of the refueling machinery, including cells for removing, segmenting, and packaging of irradiation capsules. Rabbit tube stations and gas and instrument facilities for irradiation facilities are also located on the second floor. The high bay floor is used for refueling and maintenance of the reactor.

2. DEFINING THE USERS

The ANS will serve as a focal point for neutron beam research both in the United States and internationally. More than 1000 users are expected annually, with the typical user being a university, government, or industrial researcher who comes to the ANS for about two weeks to run an experiment, then returns to his own laboratory to analyze data and publish results.

The first step in designing such a facility to meet the needs of the users is, obviously, to establish who the users are and what needs they have. Early in the ANS project, a survey was made of plans for new or upgraded facilities throughout the world. This survey was judged to be an excellent approach to examine other opinions as to where priorities of users at other sites were being placed. It also offered a means of placing the ANS into a more global perspective and assessing what objectives are appropriate for the ANS and what needs are already being met at other locations. Workshops were held with various user communities, such as the neutron scattering community and the users of californium isotopes. Representatives of many of the user communities have been working at ORNL for many years, and representatives from these researchers were recruited for participation on the ANS design teams. Not only were these representatives solicited for input into routine design decisions, but they were, in turn, urged to contact coworkers at other sites. In several cases, questionnaires were given wide distribution in an attempt to broaden the input into the facility design. As the project continues to mature, regular newsletters are being produced, again soliciting input on project priorities and other design matters.

A more formal procedure for polling the user communities and obtaining the appropriate balance of emphasis for the ANS is the formation of the National Steering Committee for an Advanced Neutron Source (NSCANS). Participation on this steering committee was carefully apportioned to ensure balanced membership from the primary disciplines involved in beam research and the various communities interested in the use of irradiation facilities. It was also balanced between university, government, and industrial researchers. Regular meetings of the executive and full committee provide review and overall guidance on the ANS design from a user's perspective; subcommittee meetings of the individual disciplines offer the potential for more focused discussions. Membership on NSCANS is now being rotated to ensure that guidance is not limited to the opinions of a small group.

3. DESIGNING FOR THE USER COMMUNITIES

The general requirements for a research reactor such as the ANS call for two distinct populations on the site, each with its own procedures and obligations. The first is the reactor operations staff, which is trusted with the responsibility of ensuring that the reactor is operated safely and that no radioactive releases occur during either routine long-term...
operation or any upset, and which is responsible for both the well being of others at ORNL and of the general public. The second is the outside user community, which requires open access to research instruments and supporting laboratory, office, and computing facilities with on-site safety training programs commensurate with visits as short as a few days.

The design of the ANS addresses the basic needs of these two populations by zoning the facility into research and operations areas. The guide hall and office/laboratory buildings are accessible to users with minimal restrictions. The ground floor of the reactor building is also readily accessible, with personnel training and monitoring levels as appropriate for entering reactor containment. The high bay of the reactor building and the entire reactor support building (including the reactor control room) are designated as operating areas, and access is restricted to the operations staff. Here, greater levels of training, security, and monitoring are applied to limit the possibility of a reactor incident or a release of contamination. All reactor equipment likely to become significantly activated or contaminated (with the exception of beam tube assemblies and the small activity present in research samples) is located in the operations areas. The second floor of the reactor building, including facilities associated with irradiation experiments, is split between the two groups. Part of the second floor is used for nuclear and fundamental physics and analytical chemistry applications, using bent guides or slant beams; the other part is associated with the operations and handling of irradiation capsules and rabbits. In the latter case, the actual loading and handling of capsules and rabbits is conducted by the operations staff. Control of irradiation capsules and rabbits is closely supervised by the operations staff, in accordance with preapproved procedures. At the juncture of the office, guide hall, and reactor buildings is a focal point area where incoming guests can obtain badges and directions to the appropriate facilities. It will also be used as an information center, so that users will be provided with ample data on reactor operation and will have no need for access to the control room.

The design activities for experiments are again split along the lines of beam instruments and irradiation facilities. The primary purpose of the ANS is neutron scattering, and thus the facility is optimized for beam work. The ground floor beam room totally encircles the reactor, with the exception of the pool needed to pass the cold guides into the guide hall. Every attempt is also being made to provide space on the second floor to meet all needs, spatial and otherwise, for beam research applications without precluding irradiation facilities. The guide hall is a wedge-shaped structure, rather than a rectangle, to allow an open layout of instruments without limiting the useful length of any guide as a result of that guide meeting a side wall. The overall layout of the main complex limits the arc of the reactor building taken up by the support and office buildings, allowing for the addition of further guide hall space in a direction counterclockwise from the initial hall; this may be of particular interest should the development of supermirror technology allow the efficient transmission of thermal neutron beams, eliminating much of the crowding around the shield wall.

Designs for beam transport systems must balance flexibility over the life of the facility against unusual needs of specific experiments. To the greatest extent practical, standard beam-instrument interfaces will be used. Even when special beam interfaces are required, these will be designed to be replaceable with a standard plug. Straight neutron guides will generally be used rather than curved guides, to avoid incorporating a wavelength filter that
could not easily be altered. Provisions for filters will be accommodated in the beam-instrument interface. The ANS will be used to extend neutron-scattering techniques into new subject areas, including problems where the effects being studied are very small or where only very small samples can be prepared. These needs will be met by providing a mix of the very best classical instruments, having the highest possible intensity, together with instruments that push the state-of-the-art into regions of uniquely high resolution. Specialty instruments, such as nuclear physics stations, will be designed for complete removal and replacement. Handling facilities will be designed to allow replacement of any instrument while adjacent stations are in operation.

Likewise, irradiation facilities will be designed so that positions of fast, high-epithermal, and thermal spectrum can be made available to all users. Irradiation capsules will be designed to allow instrumentation and gas lines to the extent that the integrity of the primary coolant boundary is not compromised. Facilities not only include the irradiation positions themselves, but also target handling cells (including cask loading facilities), pneumatic and hydraulic rabbit tube unloading stations, and instrumented capsule control stations. Supporting laboratory space will also be provided.

Finally, the facility design must accommodate the needs of the users themselves. Estimates are being made of desirable office and laboratory space, based on experience at successful European and American centers. It has been estimated that space and facilities should be provided for six persons, on the site, for each beam instrument. A central data collection and communication center is also planned, aiding the user in transporting data to his home location and allowing communication and collaboration with colleagues during an experiment. An organizational structure is being developed in which permanent staff will be assigned to groups of instruments, ensuring support for outside users in setting up and operating experiments. The office building is designed with a central open area, encouraging free discussions between the occupants. Researchers will also be able to take advantage of research teams and other facilities located at ORNL, including libraries, supercomputer facilities, and the nearby Holifield Heavy Ion Research Facilities (HHIRF). A number of general personnel support facilities will be located between the HHIRF and the ANS, creating a large users complex on the east end of ORNL.

4. CONCLUSION

The ANS will be a unique and invaluable research tool because of the unprecedented neutron flux available from the high-intensity research reactor. But this reactor would be ineffective without world-class research facilities that allow the fullest utilization of the available neutrons. And, in turn, those research facilities will not produce new and exciting science without a broad population of users from all parts of the world, placed in a stimulating environment in which experiments can be effectively conducted and in which scientific exchange is encouraged. The design of the ANS focuses not only on the reactor, but on providing the experiment and user support facilities needed to ensure its effective use.
5. REFERENCES

5. HAYTER, J. B., personal communication (February 1989).