POWER OPERATION, MEASUREMENT AND METHODS OF CALCULATION OF POWER DISTRIBUTION

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ABSTRACT

During the initial fuel loading of a BWR core, extensive checks and measurements of the fuel are performed. The measurements are designed to verify that the reactor can always be safely operated in compliance with the regulatory constraints.

The power distribution within the reactor core is evaluated by means of instrumentation and elaborate computer calculations. The power distribution forms the basis for the evaluation of thermal limits.

The behaviour of the reactor during the ordinary modes of operation as well as during transients shall be well understood and such that the integrity of the fuel and the reactor systems is always well preserved.

2.1 FUEL LOADING PROCEDURES

2.1.1 Neutron Flux Instrumentation

A. Introduction

The neutron flux is monitored by a number of detectors in the core. These detectors give the plant operator information about the total power level and the power distribution. The instrumentation also continuously checks that no rules and limitations set up on the flux level and flux change rates are violated.

Otherwise, the reactor may be scrammed. Thus, the detectors are an important part of the safety system.

Power is in the first approximation related to the thermal neutron flux density in the fuel, Φ, by the relation

$$p = e\ V\ \Sigma_f \cdot \Phi, \qquad (1)$$

where

Σ_f - macroscopic core average cross section

V - volume of the core

e - energy yield per fission

Equation (1) states that thermal power is proportional to the thermal flux, the proportionality constant being $c = e\ V\ \Sigma_f$.

As the power distribution changes or as the fuel is burned up the flux averaged fission cross section, and hence the constant c, changes. This is one of the reasons why the power range flux monitoring instruments have to be recalibrated occasionally. During power ascension from zero to full power the power profile varies considerably. As a consequence, the detector reading is not linearly proportional to the thermal output over the whole power range.

The fission product decay heat also contributes to the non-linearity of Eq. (1). The decay heat may be up to 8 % of rated power.

During transients, Eq. (1) should be interpreted with some care. Because of the fuels ability to store heat and the delayed fission product decay heat, the power transferred to the coolant, and hence the steam production, does not respond as rapidly as the neutron flux. In transients Eq. (1) yields fission power rather than thermal power.

To be able to cover the entire flux range from cold shut-down to full power the detector equipment has been divided into three systems, each covering a different flux range:

1. SRM - Source Range Monitoring
2. IRM - Intermediate Range Monitoring
3. PRM - Power Range Monitoring

The PRM system consists of two parts

3.a LPRM - Local Power Range Monitoring

3.b APRM - Average Power Range Monitoring

A fourth system, used for calibration of the LPRM detectors, is:

4. TIP - Traversing Incore Probe.

The SRM detectors are employed during the startup phase of nuclear operation (subcriticality and passage of criticality), the IRM detectors during the heating phase, and the PRM detectors during power operation, see figure 1. The monitoring systems must well overlap.

B. SRM - Source Range Monitoring

The SRM system usually consists of four identical channels, each with a detector, a detector drive and an amplifier.

There is one SRM detector in each core quadrant (fig. 2). The detector is a fission chamber designed to count pulses in the flux range from 10^3 neutrons/$cm^2 \cdot s$ to 10^9 n/$cm^2 \cdot s$, which corresponds to a relative power from 10^{-11} to 10^{-5}.

The SRM drive makes it possible to move the detectors in and out of the core. In the IRM and PRM ranges, the SRM detectors are moved out in order to prevent fast depletion of the fission chambers. Otherwise, the SRMs are always positioned inside the core, somewhere in the upper part where the flux peak occurs when the reactor is free of steam void and when control rods are first withdrawn from a shutdown condition.

Neutron sources are placed inside the fresh core to ensure a minimum measurable neutron activity.

A minimum counting rate of 3 pulses per second is required. Zero counting rate would not be acceptable since in this case the reactor operator would not know whether the counting rate was really zero and stable or the SRM instrumentation was out of order.

The SRM instruments display not only the flux level but also the flux doubling time.

The SRM system has a safety function when in use, i.e. before the IRM system is employed. Typical levels and ensuing actions are

		Counting Rate		Action
Low level	L1	3	p/s	WD Block x)
High level	H1	10^5	p/s	WD Block
High level	H2	$5 \cdot 10^5$	p/s	Scram
Short doubling time		25	s	WD Block

x) WD Block = Control Rod Withdrawals are prohibited.

High level H1 is an indication that the SRMs should leave the flux monitoring to the IRM system. High level H2 will be effective if the overlapping between the SRM and IRM ranges is inadequate.

Short doubling time leading to rod withdrawal block is, for instance, obtained when the operator is withdrawing control rods too fast.

One SRM channel indicating level H2 does not cause scram. Two or more channels are required to shut down the reactor. These logics are also used for the IRM and APRM safety systems.

Neutron sources of several types exist. The sources may be mounted either in separate thimbles in between fuel assemblies, as shown in figure 2, or mounted in a few selected fuel bundles. The neutron sources can make use of different active materials, e.g. Cf-252, AM-241/Be/Cm-242, Sb-124/Be, etc. The source strength should be in the range $1 \cdot 10^8$ to $5 \cdot 10^8$ neutrons per second.

Once a reactor core has attained an average burnup in excess of about 10 MWd/kg U, the inherent source will be sufficient to yield a neutron flux density so that the SRM system gives an acceptable count rate. Thus, the initially installed neutron sources may be removed from the core. The inherent source is due to spontaneous fission of several actinides (notably Pu-240, Cm-242, and Cm-244) and (α,n) reactions in the oxygen of UO_2.

C. IRM - Intermediate Range Monitoring

The IRM system consists of four or eight detectors, see figure 2. The IRM detector is a fission ionization chamber working in the flux range

10^8 n/cm^2·s to 10^{13} n/cm^2·s corresponding to the power range 10^{-6} to 0.3.

The IRMs have a retraction drive machinery similar to the SRM drives. During power operation the IRM detectors are withdrawn from the core.

The IRM range is divided into a number of overlapping linear subranges, two per flux decade, i.e. about 12 subranges. A manifold of subranges ensures a more detailed supervision than one big IRM range. When the flux monitoring is taken over by the IRM system from the SRMs the IRM range switch must be set to bottom range. The operator successively upgrades the range number as the flux increases.

The flux level is indicated on a linear scale for each subrange which is provided by low and high trip levels. The safety system automatically checks that the operator has a correct picture of the flux in the core by supervising the flux levels on the subrange in use and, eventually, trips the reactor if anything goes wrong. On the 0-125 linear scale the following may happen.

		Relative level	Action
Low level	L1	31	WD Block
High level	H1	110	WD Block
High level	H2	120	Scram

At H1 (L1) the operator should switch to a higher (lower) subrange.

At 5-10 % of full reactor power the IRMs leave the monitoring task to the PRM system.

D. LPRM - Local Power Range Monitoring

The LPRM system consists of a large number of detectors such that the entire core is well monitored. In our example (figure 2 and 3) there are 36 LPRM detector assemblies with 4 detectors in each assembly. The LPRM system is designed to monitor the power distri-

bution and its changes during rapid events as well as during normal operation.

The LPRM detectors are fission ionization chambers with highly enriched U-235 subject to depletion. The depletion is approximately exponential in neutron dose.

Newer designs of LPRM detectors have breeding material (U-234) addded to the U-235 in order to extend their lifetime by at least a factor of 2.

The interpretation of the LPRM signal is different for different reactor vendors. In certain BWRs the signals are calibrated to equal the average surface heat flux of the surrounding fuel assemblies at the axial level of the detectors. In this case the LPRMs have to be calibrated, by running the TIP system, whenever there is a major change of core status.

In a different approach, the LPRM value shown to the operator just equals the fission response of the detector (compare the different ways of evaluating power distributions and thermal limits described in Ch. 2.2.2) The LPRMs have to be calibrated only when the fission chamber depletion so warrants, i.e. about twice a month.

E. APRM - Average Power Range Monitoring

There are usually 4 or 8 APRM channels. To each APRM there are a number of LPRM detectors associated (20--30 detectors) that give a representative flux picture of the entire core. The APRM value is just the average of these representative LPRM readings.

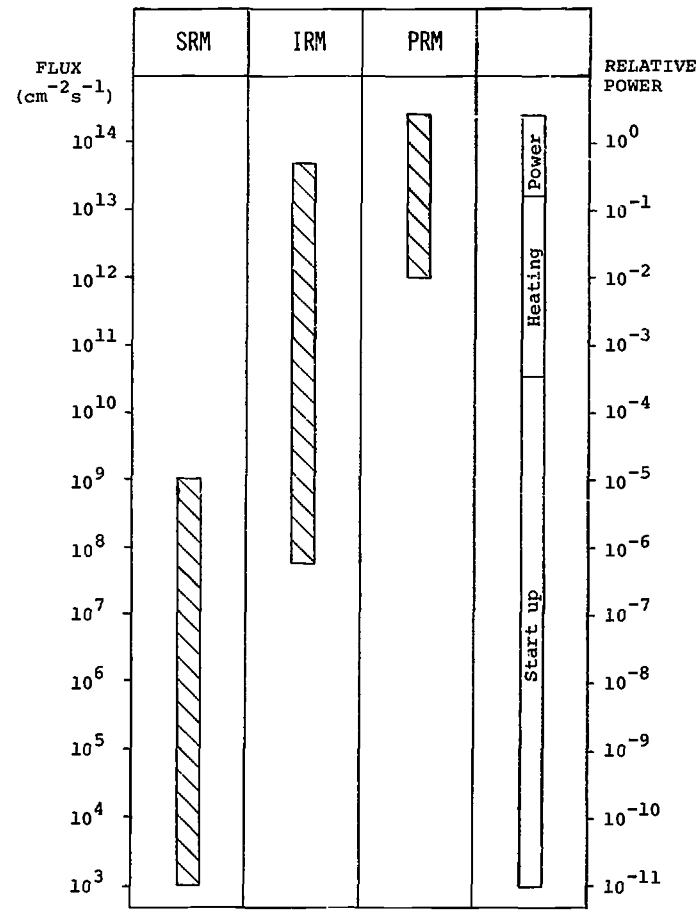
The APRM system covers a range from 1 % to 125 % of full power. The APRMs are calibrated via a gain adjustment which allows them to be set at the desired level, no matter what the average current coming from the LPRMs is. The APRMs readings are occasionally adjusted to equal the thermal reactor power obtained from boiler heat balance calculations (see Ch. 2.2.1).

The reactor is scrammed if the APRM reading exceeds 115-120 % or is too high at low values of the core coolant flow (figure 4). At 110 % there is a control rod withdrawal block and an automatic power decrease by means of recirculation pump speed reduction.

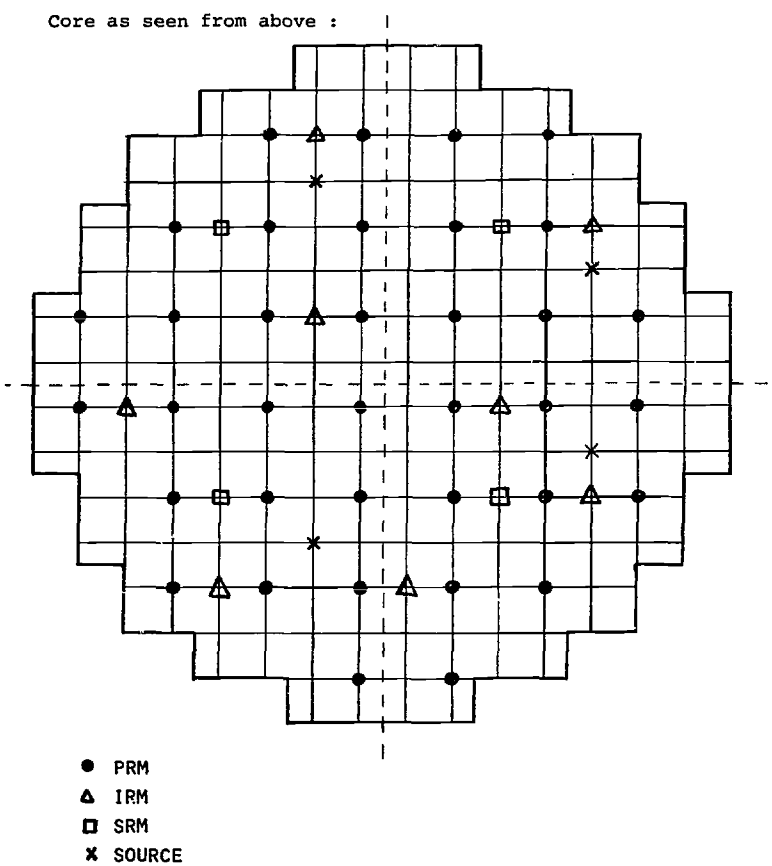
F. TIP - Traversing Incore Probe

The TIP system consists of a detector probe that can be moved inside each of the LPRM detector tubes. As a result a continuous picture of the axial flux is obtained, see figure 5.

Due to burnup the LPRM readings drift downwards, some more, some less, as time proceeds. To compensate for the drift the LPRMs are now and then intercalibrated. The LPRMs are compared to the relevant TIP curves and adjusted to equal the TIP reading.

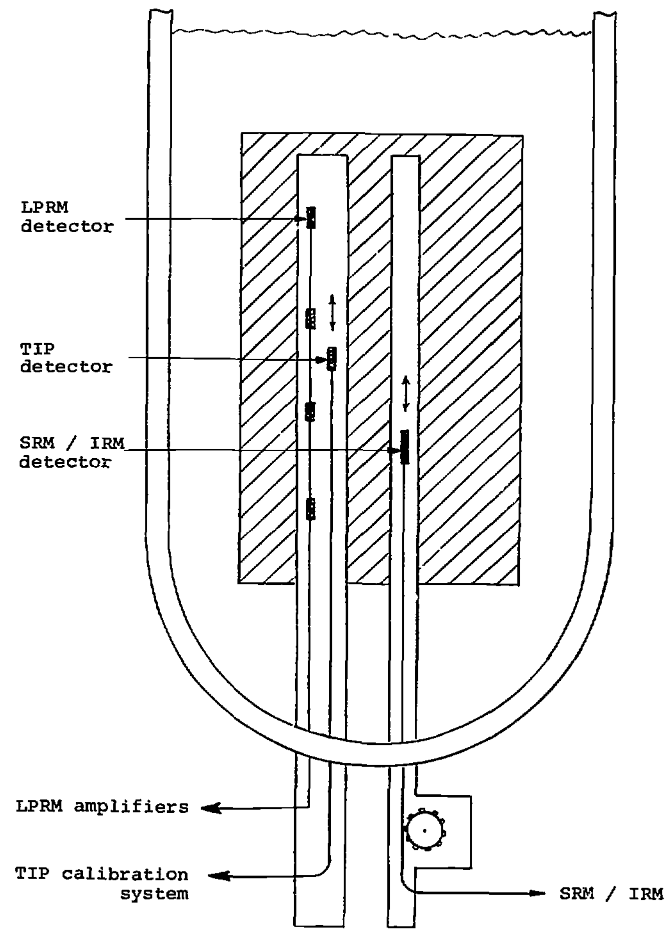


2.2.1
Figure 1 DETECTOR RANGES



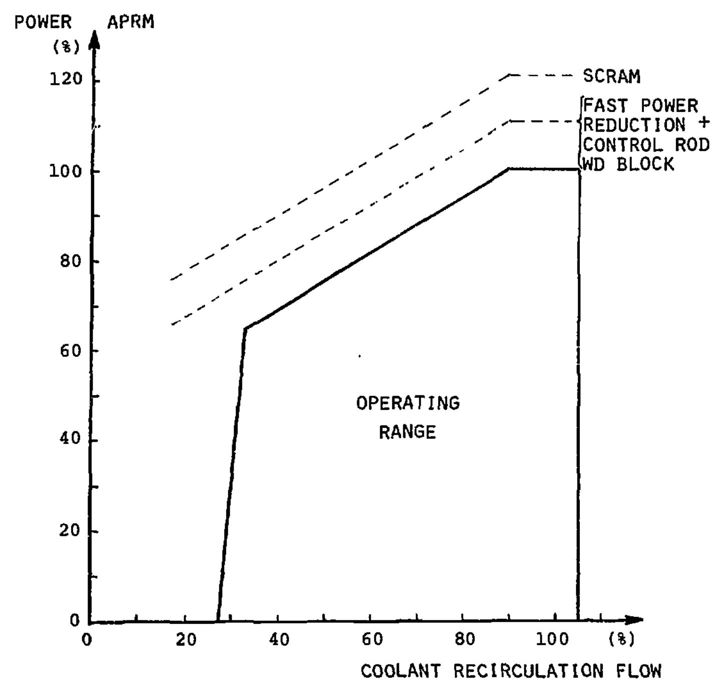
2.1.1
Figure 2

FLUX INSTRUMENTATION



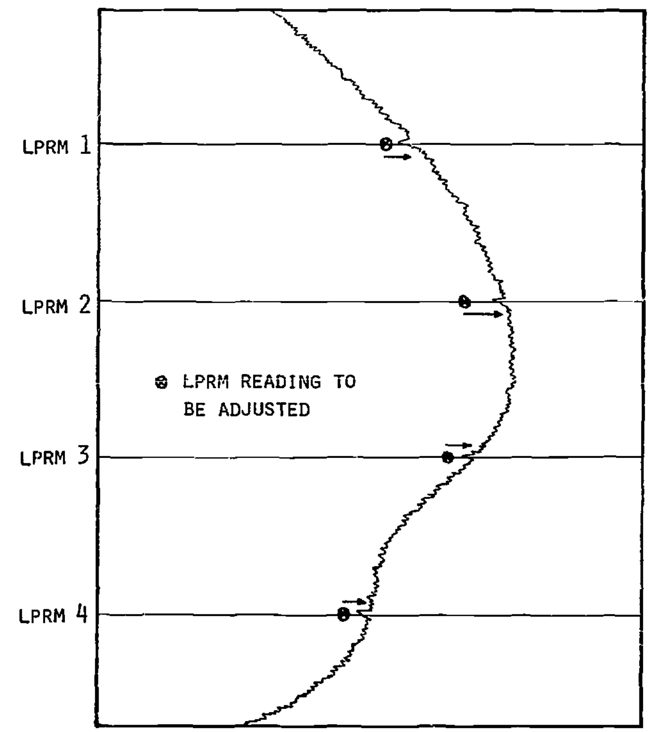
2.1.1
Figure 3

AXIAL DETECTOR POSITIONS



2.1.1
Figure 4

OPERATING RANGE
AND APRM SAFTY MONITORING



2.1.1
Figure 5

TIP CALIBRATION CURVE

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2.1.2 Control Rods, The Shutdown Criterion, Local and Global Criticalities

A. Control Rod Grouping and Maneuvering

It is convenient to dividie the control rods into two groups, one termed "black" and the other termed "white" (also denoted A and B rods), in a chequerboard pattern. A colour selector switch may be used which distinguishes between the two sets and during normal operation only one set of rods can be withdrawn at a time. This precludes adjacent rods to be withdrawn inadvertently through operator selection error.

In some reactor systems the control rods are withdrawn one at a time, at a fairly large withdrawal speed. In other systems, rod withdrawal speed will be slow but, instead, the rods are operated in maneuvering groups of various sizes, e.g. 2, 4, or 8 rods, with all rods within a group belonging to the same colour. This is sometimes called operation in a "ganged" mode. The rods in a maneuvering group will of course have to have adequate separation so as not to interact strongly. In general, they are arranged in symmetrical patterns (half core or quarter core symmetry).

For fast shutdown of the reactor the control rods are divided into scram groups (in case each rod does not operate singly and independently). A scram group typically consists of 8 to 10 rods as uniformly as possible distributed throughout the core. At scram, the rods are inserted by means of hydraulic systems, one for each scram group. Thus, if one hydraulic system fails only the rods belonging to that scram group are affected and, because of the shutdown criterion (see next subsection) the reactor can still be safely shutdown.

A successive set of control rod withdrawal patterns is called a control rod withdrawal sequence. An example of three successive patterns in a sequence is shown in figure 1.

A detailed and carefully preplanned withdrawal sequence should be the basis for all control rod movements. For every pattern in a sequence (or every second or third pattern) precalculations with a three-dimensional reactor core simulator should have been performed to show that operation with that control rod pattern will result in an acceptable power profile and in positive thermal margins (the concept of thermal margins is described in Ch. 2.2.2). Thus, if the operator follows the withdrawal sequence without any improvisation (which he always should), undesired power distributions are avoided.

At high power operation a pattern should be designed such that shallow and deep rods (shallow rod = rod that is mostly withdrawn) alternate in the central part of the core whereas all or most of the rods are fully withdrawn in the peripheral parts, see figure 1. The patterns are usually 90^{o} or 180^{o} rotational symmetric. Mirror symmetric pattern are also in wide use. However, from a power shaping point of view rotational symmetry is to be preferred over mirror symmetry.

Various hardware and software features can assist in avoiding errors in the withdrawal sequence ("operator errors") and in the control rod drive function. One such feature has already been mentioned - the division of rods into either white ones or black ones. Different vendors employ various other methods.

In modern power stations the plant process computer will assist both in selecting rods and in checking that the withdrawal sequence, which has been stored in the computer, is indeed followed. Alarm can also be given in case a control rod blade sticks in the core. Thus, for the reactor operator, control rod maneuvering will be a quite straightforward procedure.

B. Excess Reactivity - The Shutdown Criterion

Since light water reactors are generally in the cold state during fuelling there will be substantial excess reactivity which must be controlled by the reactivity control system. In the BWR this system consists of the movable cruciform control rods (usually containing boron carbide) and, in addition, any fixed neutron absorbers, often termed augmented reactivity control. Such fixed absorbers were earlier often in the form of borated stainless steel plates and were positioned in between the fuel assemblies but were used only for the very first operating cycle. Today, for both initial and reload cycles the fixed absorber consists of very high neutron cross-section nuclides, such as gadolinium, which is mixed into part of the ceramic uranium fuel. Since this additional neutron absorber is almost completely consumed or burned during an operating cycle, it is called a burnable absorber or burnable poison.

The cold excess reactivity may amount to about 20 % of reactivity and is required for taking up negative reactivity contributions from system temperature increase, fission product buildup (notably xenon and samarium), steam void, and fuel depletion, see the reactivity budget given in table I. It should be remembered that in some cases the depletion of burnable poison can give a net positive reactivity effect during the initial part of a cycle, thus increasing the excess reactivity which must be coped with by the control system at fuelling (Cf Fig. 2.2.4-2).

The main reactivity constraint for a reactor core in its most reactive condition (usually the cold state) is the shutdown reactivity criterion. This usually states that the reactor shall remain subcritical when the most reactive control rod scram group is withdrawn. (When the scram group - driven by a common actuating system - consists of more than a single rod, adequate rod spacing is required in the core to warrant a small reactivity interaction).

A common shutdown reactivity requirement is that the reactor shall be subcritical with a precalculated reactivity of at least 1 % or, as measured, by at least 0.25 to 0.5 % (depending on licensing authority). These figures would thus hold in case the core configuration at fuelling is the most reactive condition during the coming cycle. In other cases account has to be taken of reactivity contributions which arise because of deviations in the fuelling state from the most reactive state, e.g. due to temperature differences, burnable poison depletion, or possibly any remaining transient xenon poisoning from previous operation. A further contribution could in som systems arise due to a nonnegligible decrease in control rod worth as a result of boron burnup up to the point in time during the operating cycle when the most reactive cold condition occurs. To the extent that these various reactivity contributions need to be applied, it will generally suffice to use theoretical values from calculations with well qualified methods.

Shutdown reactivity measurements are typically done at initial fuel loading or after refuelling before startup when the system is still cold. The measurement is done as follows. The particular rod or rod scram group which is deemed most effective (according to calculations) is first fully withdrawn. So far the test will merely show that the reactor is still subcritical. In order to verify that the reactor is still subcritical by at least, say, 0.5 % of reactivity, a control rod adjacent to one of the already withdrawn rods is withdrawn a distance corresponding to a calculated reactivity increase of 0.5 %. If the reactor is then close to critical or clearly subcritical the shutdown requirement is fulfilled. The actual shutdown reactivity can be better determined by actually continuing rod withdrawal till local criticality is achieved. Again the reactivity value obtained will of course be a calculated one but the uncertainty will be quite acceptable.

C. Local Criticality

By "local criticality" is meant the core condition when criticality has been reached locally at some part of the core while the remainder is subcritical. Local criticality is obtained by withdrawing a few adjacent control rods while all other rods are fully inserted. The shutdown reactivity test is an example of a local criticality.

Local criticality measurements serve to check the uniformity, or reproducibility, of the neutron multiplication property of the core and to evaluate the reactivity shutdown margin. The measurements are compared to computer calculations and hence give a check of the accuracy of the theoretical methods.

Depending on circumstances, local criticality may involve several adjacent control rods. The procedure should always be to withdraw, in turn, successive rods to the fully withdrawn position until criticality is approached with the last of the rods partially withdrawn. The reason for this is to avoid rod banking, i.e. a condition with adjacent rods equally withdrawn, in which case very high differential rod reactivity worths might arise.

Still another reactivity limitation can in some systems and cases govern the order of rod withdrawals. For instance, if the rod by which criticality is reached is immediately adjacent to previously withdrawn rods and has a high differential reactivity in the vicinity of the critical position, difficulties may arise when approaching criticality - such as quickly reaching short positive reactor periods. In such cases it is recommended instead to use a rod which has less worth, for instance a diagonally adjacent rod which will have appreciably less differential reactivity. This kind of procedure is of special importance in reactor systems in which the minimum control rod withdrawal increments are fairly large, such as with notch mechanisms.

Examples of local criticalities are given in Fig. 2. Repetitive measurements at symmetrical positions of the core confirm the uniformity of the fuel.

Due to the fact that the top part of the core has high statistical weight when withdrawing rods in a local configuration, the differential rod reactivity worth of a rod will be highest for small withdrawals (in the range 15-35 % withdrawal) and quite small elsewhere. This effect must be understood and kept in mind by the control rod operator.

The integral reactivity worth of a control rod is rather large for local criticalities. However, as a general rule this worth decreases monotonously from the first rod in the order of withdrawal. As an example may be cited the following sequence for a local criticality involving 4 rods: 3.3, 3.1, 1.4, 1.2 %.

D. Global Criticality

The global criticality tests are performed to check that the fuel behaves as expected and that the computational methods can correctly predict the critical control rod positions. The measurements also serve to verify the adequacy of the rod withdrawal sequence employed for start-up.

Global criticality tests are done such that criticality is reached simultaneously in the entire core. Examples of global tests are shown in Fig. 3.

In making up a control rod sequence up to the point where global criticality is achieved, the following items should be considered:

a) Localized criticality is to be avoided

b) Near criticality, reactivity worths of individual rods shall not exceed a certain prescribed limit. (A typical value is 1 % of reactivity in order to limit energy release in a "rod drop" accident).

c) The rate of reactivity increase must be limited.

d) Power monitoring with the SRM system should result in a relatively smooth increase in SRM readings as control rods are withdrawn.

All these requirements are automatically satisfied when the control rods are withdrawn in a distributed manner such that the core at all times maintains a fairly uniform rod density. It follows that the withdrawal pattern will be symmetric. Another consequence is that only "white" rods, say, in a chequerboard pattern will be withdrawn first until they are all out, in the case "black" rods are desired for use at high reactor power. Conversely, "black" rods will be withdrawn first when "white" rods are used at high power.

The requirements under item b) should be verified by calculations. There is usually no difficulty in meeting the requirement.

There is no sharp limit on the rate of reactivity increase for item c). The value, e.g. in terms of pcm/s, that results with a sequence leading to uniform control rod density will be fully acceptable.

Item d) of course presupposes that the first rods to be withdrawn are chosen prudently with respect to the location of SRM detectors and neutron sources. It is an advantage to raise the detector signal level at an early stage in order to obtain better counting statistics. Rods close to neutron sources should be withdrawn first.

The number of control rods needed to be withdrawn before criticality is attained will vary greatly from time to time during an operating cycle. Thus the rod withdrawal at criticality may vary between about 25 and about 65 % of the total control rod inventory. The reason is that core reactivity depends on fuel and burnable posion depletion, on temperature, and on samarium and xenon transients. However, this veriability in core conditions poses no difficulty since the approach to criticality is readily monitored by the SRM system. Moreover, only two withdrawal sequences need to be used (one starting with "white" rods, the other with "black" rods) up to the point that all rods belonging to one colour are fully withdrawn, i.e. approximately 50 % of all the rods.

Consider now the procedure during rod withdrawal from a fully shutdown condition. Each successive group of rods will be fully withdrawn before the next is selected for withdrawal. In a ganged mode, the control rod grouping conveniently defines the set of rods. When the rods are operated singly, the rods defining a group will either be pulled out all the way one after another or they will be moved, in turn, only part of the way in two or more rounds until they are all out. The latter mode of operation brings out the importance of operating the control rods in (usually symmetric) groups. Behind this lies consideration of the very unlikely "rod drop" accident. Should one of the control rods happen to stick in the core and later to drop out, a severe reactivity transient is avoided since the other rods in the group have already contributed to a higher and controlled and therefore known reactivity level in the core.

E. Subcritical extrapolation

Subcritical extrapolation is a useful technique to estimate in advance at what value of a reactivity dependent parameter a reactor core will attain criticality. The inverse neutron detector signal is then plotted as a function of this parameter, and extrapolation to zero inverse count rate gives the critical value. The parameter will typically be control rod position or number of loaded fuel assemblies. The method is mostly applied when building up a new reactor core which has as yet experimentally unverified properties. It should be remembered, however, that the extrapolation curves which are obtained in actual practice may differ considerably from the theoretically ideal ones. The reason is, of course, that the spatial relations between the effective core volume, neutron source and neutron detector are not constant and that reactivity dependence on the parameter used may deviate strongly from a linear relationship. Nevertheless, extrapolation works well close to critical even for a control rod withdrawal and there is also a technique to monitor and demonstrate the subcritical state of a core during initial fuelling.

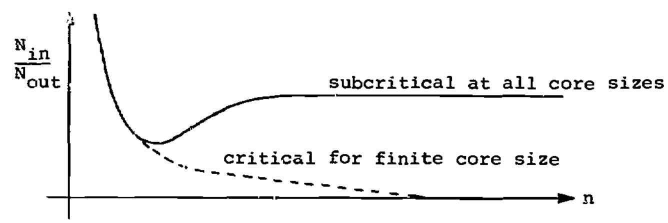
Extrapolation at approach to critical

One desires to plot the inverse neutron detector count rate as a function of control rod withdrawal position. However, since the rod differential reactivity varies strongly, especially in connection with local criticality, rod position should be converted to an equivalent reactivity scale. To obtain this conversion one can rely on a calculated relation between rod position and reactivity. Although the procedure is not very accurate initially, it is quite satisfactory with respect to the final extrapolation to zero inverse count rate, especially for reactivity in the range -0.2 % to critical.

Subcriticality monitoring

When a fresh core is being loaded for the first time the regular extrapolation procedure becomes very uncertain when plotting inverse count rate as a function of the number of loaded fuel assemblies. The uncertainties derive from a successively changing core - detector geometry because of increasing core size and changed detector positions.

A method which is rather insensitive to these disturbances is to plot the count rate ratio for all the rods inserted to the condition with the central control rod fully withdrawn as a function of the number of loaded fuel assemblies. According to elementary theory, the resulting curve will first exhibit a minimum and then asymptotically stabilize at a certain level as long as the core as a whole will remain subcritical for any size with the central rod withdrawn.



The method is valid as long as the core composition remains uniform and is applicable even for rather large subcritical reactivities. In practice, the minimum in the curve is not always clearly distinguished but most important is the trend towards a constant ratio for the large core.

2.1.3 Methods when loading initial core

A. Introduction

The initial core loading procedure shall warrant a safe and efficient way of building up the core. The core reactivity shall be well controlled at all times and any inadvertent reactivity increase shall not lead to any core damage or radiation exposure to operating personnel or the general public. At fuelling there is in effect no containment closure. On the other hand, with a fresh core there is no inventory of long-lived fission products. Therefore, procedures for taking the reactor critical at low power are acceptable. Maximum power is often set at 1 % of rated for the open reactor vessel but there is usually no need to exceed about 0 1 % of rated power, i.e. of the order of a few MW.

However, the procedures to be followed should be very stringent and all prerequisites with respect to plant status and all fuel transfers must be listed in detail in the fuel loading procedure document. The accompanying check list of operations should accordingly be signed by the responsible fuel engineer after each completed operation.

The considerations listed in the following paragraphs will aid in the formulation of detailed fuelling procedures. Only "wet" fuel loading, i.e. the reactor core is covered with water, will be treated here since it is the most common method.

B. Prerequisites for fuel loading

The requirements to be met at commencement of fuelling will differ in detail between different BWR designs and due to variations in overall commissioning procedures such as during preoperational testing. The following items should, however, be given consideration in most cases.

a) All fuel to be loaded has passed acceptance checkout.

b) Refuelling machinery fully checked out with the help of dummy fuel.

c) All control rod drives have undergone functional testing during preoperational tests.

d) All control rods installed in the core and fully inserted.

e) All incore instrumentation including extra neutron detectors shall be installed.

f) All startup neutron measuring instrumentation channels (SRM and extra channels) have been checked out using the startup neutron sources.

g) Neutron startup source(s) installed.

h) Reactor protection circuits relevant to the refuelling mode of operation checked out.

i) Results from precalculations of shutdown reactivity, local critical control rod patterns, and of the global (distributed) critical control rod pattern for the prescribed withdrawal sequence shall be available. The calculated differential reactivity worth of control rods to be used during shutdown reactivity testing and approach to local criticality should also be available.

j) Continuously open intercommunication between the control room and the refuelling platform shall be maintained during all fuel handling activities.

C. Control rod functional testing

Preoperational tests will have verified the reliable function of the control rod drives with respect to both slow maneuvering and fast insertion. However, such testing may not always be 100 % complete since the preoperational testing may have been limited to a cold system or to rod drive tests without the weight load of the absorber blade. For this reason and in order to preclude that fuel presence affects maneuverablility, some degree of control rod drive functional testing is generally done in connection with the loading procedure. In particular, before any one control rod is withdrawn in an approach to critical, it must have undergone a full functional testing in the cold condition.

The testing can be performed either as a part of the fuelling procedure or after fuelling (in case the rods are not withdrawn in any approach to critical during core buildup). In this connection withdrawal of a single rod (or single scram group containing well separated rods) is not taken to be an approach to critical. The reasoning behind this is that the reactor by design is still securely subcritical with the single rod withdrawn and that preoperational testing has verified the basic rod drive functions which are redundant. Moreover, power minitoring is done during all rod withdrawals in order to detect any inadvertent approach to critical. In addition, calculational results with respect to reasonable loading errors will probably have shown that subcriticality will subsist. (As an example of loading error may be mentioned the loading of fuel bundles that do not contain burnable poison in the central parts of the core. Such bundles are often used only at the core periphery in first cores.)

D. Power monitoring

An appropriate power or neutron flux monitoring involving satisfactory spatial relations between fuel, neutron source, and neutron detectors as well as adequate redundancy of measuring channels is essential to the loading procedure.

A startup neutron so. ; is required to provide a sufficiently large neutron flux so that a measurable neutron detector signal is obtained. The minimum count rate should exceed 3 cps with a signal-to-background ratio of at least 5:1.

The regular source range monitoring (SRM) system operates with fairly insensitive fission counters (typically four) in fixed radial core positions. Therefore, at least during the initial loading phase, supplementary high sensitivity neutron detectors are necessary. Such detectors, sometimes called "dunking chambers", will be inserted from above and are contained (possibly together with a preamplifier) in immersible containers or dunkers. The electric cabling will be contained in a water tight flexible tubing and drawn to an appropriate power and signal terminal in the reactor service hall. The dunkers are usually made to fit into a fuel assembly position in the core with the detector located in the top half of the core (at the level of neutron sources). The detector signals are transmitted to the SRM instrument cabinet and are fed to the ordinary SRM instrumentation so that all protection functions of the SRM system are operable with the extra detectors.

A fundamental power monitoring requirement during operations involving reactivity increments is that at least two redundant measuring channels must be operable. In general, further redundance is to be recommended in order to provide a margin for channel failures and to facilitate the loading procedure. Thus four dunking chambers are often used.

With a typical neutron source strength in the range $1 \cdot 10^8$ to $5 \cdot 10^8$ neutrons per second for the source unit used at start of fuel loading, a detector sensitivity of the order of 1 to 5 cps per flux unit is adequate. This is about 1000 times more sensitive than for the regular SRM detectors.

TABLE I

Reactivity Budget

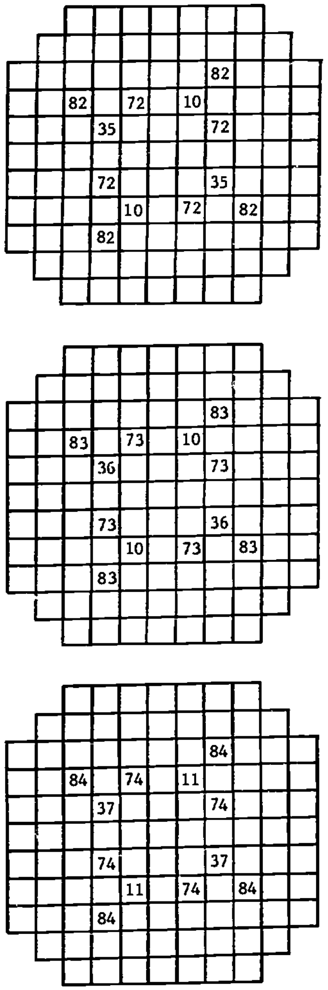
Fresh Cold Core: k_∞ = 1.15-1.20		
Excess reactivity needed for	Amount %	Reactivity Controlled by
Depletion	6-10	BA+CR
Xenon	2-3	CR+CRF
Temperature	3-4	CR
Void	3-4	CR
Leakage	3-4	-

BA = Burnable Absorber

CR = Control Rod

CRF = Coolant Recirculation Flow

2.1.2
Table I

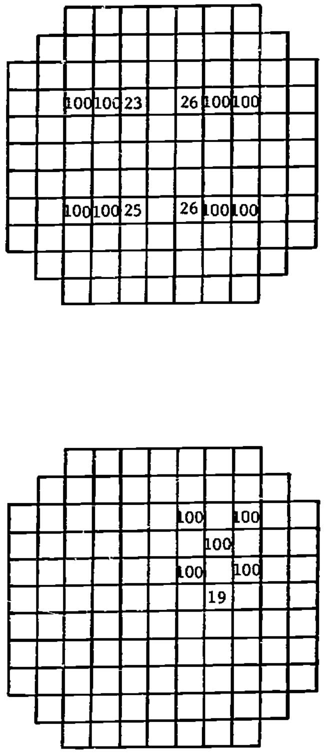


THE NUMBERS GIVEN INDICATE CONTROL ROD WITHDRAWAL PERCENTAGE

EMPTY SQUARES = 100% WITHDRAWAL

2.1.2
Figure 1

CONTROL ROD WITHDRAWAL SEQUENCE

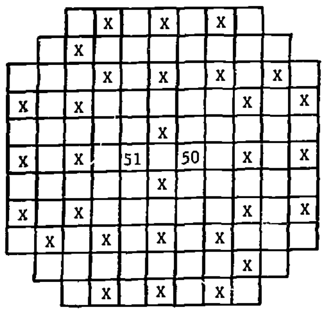


FOUR MEASUREMENTS SHOWN IN ONE PICTURE

DEMONSTRATION OF FUEL UNIFORMITY

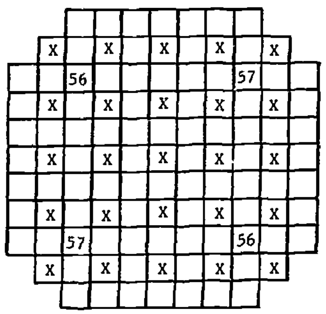
2.1.2
Figure 2

LOCAL CRITICAL MEASUREMENTS



ONLY WHITE RODS WITHDRAWN

X = 100 % WITHDRAWAL



ONLY BLACK RODS WITHDRAWN

COLD CORE CONDITIONS

 2.1.2 Figure 3 GLOBAL CRITICAL MEASUREMENTS

E. Loading procedure

A few basic principles will be detailed below. It should be born in mind, however, that various reactor vendors will have different procedures depending on the systems design in the specific cases.

The loading procedure should address the following items:

- Fuel loading sequence and instructions.
- Positions of neutron sources and dunking chambers.
- Control rod drive functional testing.
- Power monitoring instructions.
- Shutdown reactivity testing.
- Subcritical extrapolation procedures.
- Critical measurements (if any) for part size core.
- Results from calculations giving expected local critical configuration and differential reactivity of certain control rods.
- Critical measurements at end of loading (local and global).

The fuel loading sequence will be listed in detail giving assembly identifications, fetching and loading positions, as well as assembly angular orientation. Instructions are also given for when shutdown reactivity checking and subcritical multiplication measurements are to be done. It is convenient for the control room personnel to have a core display where the successive fuel additions are clearly marked. It is also advisable that the responsible fuel engineer carries out an independent check of fuel assembly positions in the core at various stages during the fuel loading.

The loading will naturally start at the core center where a neutron source will have been mounted. Only this neutron source should be used for most of the fuelling in order to give consistent neutron multiplication data. However, towards the end of the loading when all SRM detectors and regular neutron source positions are built into the core, all the neutron source units can be put in their final positions and power monitoring can be transferred to the ordinary SRM system. At this point the average reactivity properties of the fuel will have been verified so that any subcritical extrapolation procedures need not necessarily be continued.

The extra neutron detectors in the dunkers will initially be placed, one in each core quadrant (if 4 detectors are used), at such a distance from the central neutron source that a convenient count rate is achieved, e.g. in the range 10-100 cps. This will give acceptable counting statistics and allow a count rate increase according as core size increases. The dunkers will then be successively moved outwards in several steps. Each such repositioning should be done before fuel has to be loaded adjacent to the dunker since strong local effects on detector count rate will otherwise show up.

Fuel loading is typically done so that core size increases in a symmetrical fashion. It is usually convenient to load the four fuel assemblies in a control cell in an uninterrupted sequence. All control rods should always be fully inserted at fuelling. (No backup reactivity insertion is needed since the reactivity shutdown margin will be large, of the order of 5 % or more).

Any control rods that will be used for checking subcriticality must undergo complete functional testing beforehand. For instance, when two fuel assemblies have been loaded in diagonally opposite positions in a control cell, the rod can be withdrawn because of the lateral support from these assemblies and testing be done of slow and fast rod drive operation as required for the particular design. (Of course, lateral control blade support can be provided by using dummy fuel assemblies).

All increases in core reactivity must be accompanied by monitoring the neutron measuring channel readings in the control room. All fuelling operations must therefore be directed from the control room which calls for a continually open communication line with the fuelling platform. If the power monitoring shows strongly increasing count rates the control room operator will halt the fuel insertion and direct further actions. Detector count rate data are taken after fuel addition.

Verification of core subcriticality with the control rod of greatest worth withdrawn (i.e. usually the central control rod) should be done for several core sizes, e.g. after each new "ring" of loaded control

cells has been added. The ratio of the detector count rates for the central control rod inserted and singly withdrawn, when plotted as a function of loaded fuel assemblies, will check that the core will remain subcritical for any size (see procedure in ch. 2.1.2-E).

A further test-of shutdown reactivity is done by verifying that the reactor complies with the shutdown criterion. This can be done by first fully withdrawing the central rod (usually the most reactive rod) and then withdrawing a nearby control rod a distance that corresponds to a calculated reactivity increase as desired. Thus, this test, which should be done a few times during the complete loading, need not be a true or full approach to critical. (Often a third or even a fourth rod may have to be withdrawn to reach a local critical condition in an initial core).

The very first times that the central control rod is thus withdrawn, subcritical multiplication should be measured as function of rod position. The extrapolation procedure given in 2.1.2 may be used to predict the critical position of the rod.

Criticality tests are mostly done only after the completed loading and using the regular SRM-IRM system for power monitoring. However, if critical measurements are done for a part size core and the regular SRM detectors are not available, then the high sensitivity dunking chambers may have to be supplemented by additional neutron detector channels in order to expand the power monitoring range.

At end of loading final control rod drive functional testing will have to be carried out (if not completed earlier). This will only involve a single rod or scram rod group at a time and will give a detailed core-wide verification of adequate shutdown reactivity.

Local critical tests for the full core are also performed in addition to those done with part size core. The test sites in the core of local criticalities should be chosen in order to obtain convenient neutron flux monitoring with at least two measuring channels (SRM). Level trip of one or more of these channels should actuate the reactor protection system, i.e. scram.

During the first global criticality test the adequacy of the control rod withdrawal sequence with respect to power monitoring is checked. The critical rod positions will be compared with the precalculations.

In connection with the local and global criticality tests, rod differential reactivity worth may be measured for the partially inserted rods.

Note 1

The initial fuelling is in some respects done differently by various reactor vendors. Either the water level may be only a few meters above the core top (i.e. in the reactor vessel) or the complete reactor pool may be filled up entirely. In the first case, a fuel handling platform is in position on top of the vessel which in some ways facilitates the loading procedure and moving of detector dunkers. In the second case the regular fuelling machinery is used with the advantage of simultaneously testing the handling system and training the operating personnel.

Completely dry loading is also possible but will require different power monitoring procedures as compared to the methods described above.

Note 2

Some vendors will provide two dummy fuel assemblies for each control cell in the core in order to give the control blades the required lateral support when withdrawn for rod drive functional testing. This provision will facilitate the loading procedure with respect to the functional testing of control rod drives but at the same time it introduces more equipment handling.

2.1.4 Methods during reloading

A. Introduction

One may distinguish between two types of reloading:

a) Insertion of fresh reload fuel and any partly spent fuel ("reinsertion") in place of discharged fuel. This is the standard type of refuelling and will mostly involve some degree of shuffling, i.e. movement of fuel assemblies from one core position to another.

b) Reloading after complete core unloading e.g. after any repair or quality control work in the lower parts of the reactor vessel. The reload will include fuel from the previous operating cycle and any new fuel as in a).

The distinction is important with respect to criticality control. In the regular case continuous power monitoring (i.e. of subcritical multiplication) can be maintained with the SRM system throughout the operations for a contiguous core. In the second case the fuelling would be analogous to fuelling a first core and extra neutron measuring channels may be required and are known to have been used. However, in this case there is a way to avoid such extra instrumentation.

One important difference of the reload core compared to the initial fresh core is the distribution of source neutrons. The core which has attained an average burnup in excess of about 10 MWd/kgU has a strong inherent source, see ch. 2.1.1-B. It is now general practice in BWRs to remove the initially installed sources. The fact that the inherent neutron source is distributed throughout the core is of importance in criticality control.

Another important difference in the reload situation in comparison with the initial fuelling (as in Ch. 2.1.3) is the fact that the irradiated fuel - which even after refuelling constitutes most of the core - has wellknown and proven properties because of the previous power operation. Thus, for this fuel one need not take into account any deviations due to possible errors in the fuel delivery. Of course, any repositioning of irradiated fuel in the core will have to be done carefully.

A further point to be made is that since now the core contains large amounts of long-lived fission products, thorough consideration should be given to how shutdown reactivity tests will be made after refuelling but before the vessel is closed. Practice will vary in this respect. It is probably generally accepted to perform control rod functional testing with the open vessel in order to allow any required rod drive repair before the vessel head is mounted. At the same time a core-wide shutdown reactivity test is obtained with respect to single rod or single scram group withdrawal.

However, if local criticality tests are deemed necessary, they will in some countries be done only after vessel closure.

B. Standard refuelling procedure

All fuel handling shall be carried out with fully inserted control rods. Even with hypothetical gross errors in the fresh reload fuel (such as missing burnable poison), the generally scattered configuration of reload positions cannot then result in supercriticality.

A general procedure would be to

- discharge spent fuel to the pool
- reposition (shuffle) fuel assemblies within the core
- insert new fuel

Subcritical multiplication monitoring should be done during all shuffling and fuel insertion. Fuel handling instructions should be as with initial core fuelling.

Upon completion of refuelling, shutdown reactivity will be checked first when control rod drive functional testing is carried out but gives only proof of subcriticality for single rod withdrawals. The requirement that the reactor shall be subcritical by a defined amount (see ch. 2.1.2) will necessitate local approach to critical. The most reactive control rod positions will have been determined beforehand (by calculations) and typically four of these core regions should be checked to fulfil the shutdown reactivity requirement.

C. Refuelling after complete unloading

In case dunking chambers are available, refuelling can be done as with the fresh initial core. The different kind of neutron source distribution will, however, give somewhat different count rate variations as a function of the number of loaded fuel assemblies. In spite of this the method gives a reliable monitoring of subcritical multiplication. It should be further noted that the extra dunking detectors must have good gamma ray discrimination characteristics in order to give an acceptable neutron signal.

The use of dunkers is rather unwieldy, however, and the core may be refuelled without their use in the following manner. All previously used fuel is first loaded except that which was to be definitely discharged.

Much of this fuel will sit in its previous positions. The order of loading should be to first load fuel assemblies around the SRM detectors and then expand the loaded region inwards and to the space between the detectors. Subsequently the outer core regions are loaded. In this fashion a contiguous and continuously monitored core is built up.

Since fresh reload fuel has not yet been loaded the core will now resemble that which is attained in the standard refuelling procedure and the rest of the loading and shutdown reactivity testing will be done according to that procedure.

2.1.5 Determination of reactivity parameters

A. Introduction

During normal operation of a BWR, explicit use is seldom made of various reactivity quantities such as control rod worth or reactivity coefficients. One reason for this is that the complete range of reactivity coefficients for all core states contributes with ample margin to the excellent inherent stability and selfregulating property of the BWR reactor system. The sensitivity of power changes to control rod movements or core coolant flow control is measured rather in terms of rod displacements and pump speed changes. Moreover, the reactivity quantities vary considerably because of a large void range and rather large variation in control rod density.

However, implicitly quantitative assessments of reactivity quantities are of necessity for both static and dynamics evaluation of the BWR. Thus it is of interest to verify the calculational methods that involve reactivity changes either indirectly (eg reproducibility of critical control rod positions) or directly by means of actual reactivity measurements.

The reactivity quantities to be treated here are

- control rod integral and differential worth
- shutdown reactivity
- moderator temperature coefficient
- steam void and fuel temperature coefficient
- pressure coefficient
- power coefficient
- core excess reactivity

B. Control rod reactivity worth

In connection with startup and operation of a reactor, critical control rod patterns are obtained for all core states from the cold critical system to full power. Thus the ability of the calculational methods to reproduce the measured critical states will be a proof of how well control rod effectiveness can be determined. After such qualification, the methods can be relied upon to work out control rod withdrawal sequences and to check that these will not contain control rods that in case of a rod drop accident will increase core reactivity by some maximum value (usually 1 %).

Actual measurements of control rod integral worth are of little interest since they are valid only for the particular situations and hence are of little pratical use. However, for very low powers, two measurement techniques are readily available: integration of measured differential worths and the fast rod insertion method. At high powers, temperature and void reactivity feedback makes it difficult to separate the control rod worth component from power dependent reactivity quantities.

In the first method of measurement a series of differential rod worths with respect to successive small rod displacements are measured for a critical system. The rod displacements have to be accordingly compensated for by a corresponding reactivity change by using another control rod or by changing core temperature.

The second method of measurement is based on the prompt change in neutron flux level when a sudden reactivity change is introduced. At the outset the reactor is just critical and the particular rod (or rods) is then inserted with the scram system. Analysis of the neutron flux transient will then give the reactivity change. (Note that it may not always be possible to actuate scram of only part of the rods that are withdrawn).

It should be understood that neither of these methods is very accurate when applied in commercial reactors. With the first method the evaluated rod worth does not really correspond to a well-defined core condition. In the second method, analysis is difficult to perform due to the comparatively long rod insertion time and the nonoptimal core - detector geometry.

The measurement of the differential reactivity worth $\Delta\rho/\Delta Z$ of control rods for the various critical core states met with during very low power testing is of more direct interest. The results indicate the possible reactivity increase rates that are achievable and direct comparisons can be made with theoretical calculations. The measurement is conveniently done by withdrawing the particular rod or rods a predetermined increment ΔZ starting from an initial steady state criticality. The stable reactor period or doubling time is then measured from which the reactivity change $\Delta\rho$ is calculated from the inhour equation. An electronic reactivity meter may also be used to give a direct reactivity reading. An example of differential rod worths is given in Fig. 1.

C. Shutdown reactivity

This quantity is of immediate interest from a safety point of view since it concerns the capability to make the reactor subcritical even in the case of insertion failure of a control rod or control rod scram group of highest worth. The manner in which the shutdown criterion is measured is described in Ch. 2.1.2-B.

D. Moderator temperature coefficient of reactivity

With this quantity is understood the reactivity change due to a uniform change in core temperature (i.e. both moderator and fuel). At low temperatures the coefficient will be slightly negative (-5 to -10 pcm/degr C) for a freshly loaded core and slightly positive ($\lesssim$ 5 pcm/^{o}C) for a core at end of cycle.

It becomes more negative for increasing temperatures. Except for comparison with calculations the moderator temperature coefficient is of interest with respect to nuclear heat-up of the reactor system. The objective of a measurement is then to demonstrate that the coefficient is not unduly positive at beginning of the heat-up. Often this is done by merely showing that control rods must be withdrawn or only very little inserted to maintain criticality when the system temperature is increased from the cooled-down condition. Even in the case of a required rod insertion, the rods will have to be withdrawn again at somewhat elevated temperatures, eg before 100°C is reached.

Actual determinations of the temperature coefficient $\Delta\rho/\Delta T$ are done by period measurements at different temperatures. A common method is to start at a somewhat elevated temperature at which the reactor is taken critical at very low power (i.e. there is no reactivity feedback from power). Upon cooling the reactor is maintained critical by inserting control rods (for a negative coefficient). After a temperature decrease ΔT the control rods are withdrawn to the previous critical position and the positive reactor period is measured from which the reactivity change $\Delta\rho$ can be determined. The moderator temperature coefficient is plotted versus temperature in Fig. 2.

If a control rod is first calibrated, then the temperature coefficient can equally well be determined during heating up of the reactor system.

E. Steam void coefficient of reactivity

The void coefficient influences core stability and affects the transient response of the reactor system in connection with power control and system disturbances. It may be defined as the reactivity change per fractional change (eg volume %) in void at constant power. It is in principle a spatially dependent parameter but it is mostly adequate to evaluate the core averaged coefficient. A BWR is optimized to operate with a fairly strong negative coefficient which will vary over a certain range within an operating cycle.

The strong reactivity coupling between reactor power, fuel temperature, and void renders it very difficult to do accurate measurements of reactivity coefficients for void and fuel temperature. Instead, system perturbations can be introduced (eg in pressure, core

flow, or feed water temperature) which initiate power transients. Comparisons between the measured and calculated transients will then check whether correct dynamic parameters are obtained with the calculational method.

The interdependence between void and fuel temperature reactivity (see Fig. 3) manifests itself clearly in the power - core flow control line. An increase in recirculation coolant flow will decrease the steam void thus increasing reactivity. In turn, power will increase which raises the fuel temperature leading to a decrease in reactivity. Thus at every point along the control line there is a reactivity balance determined by steam void and fuel temperature. The ability of a theoretical method to reproduce the control line is again a good test of the method to calculate important reactivity parameters.

F. Pressure coefficient of reactivity

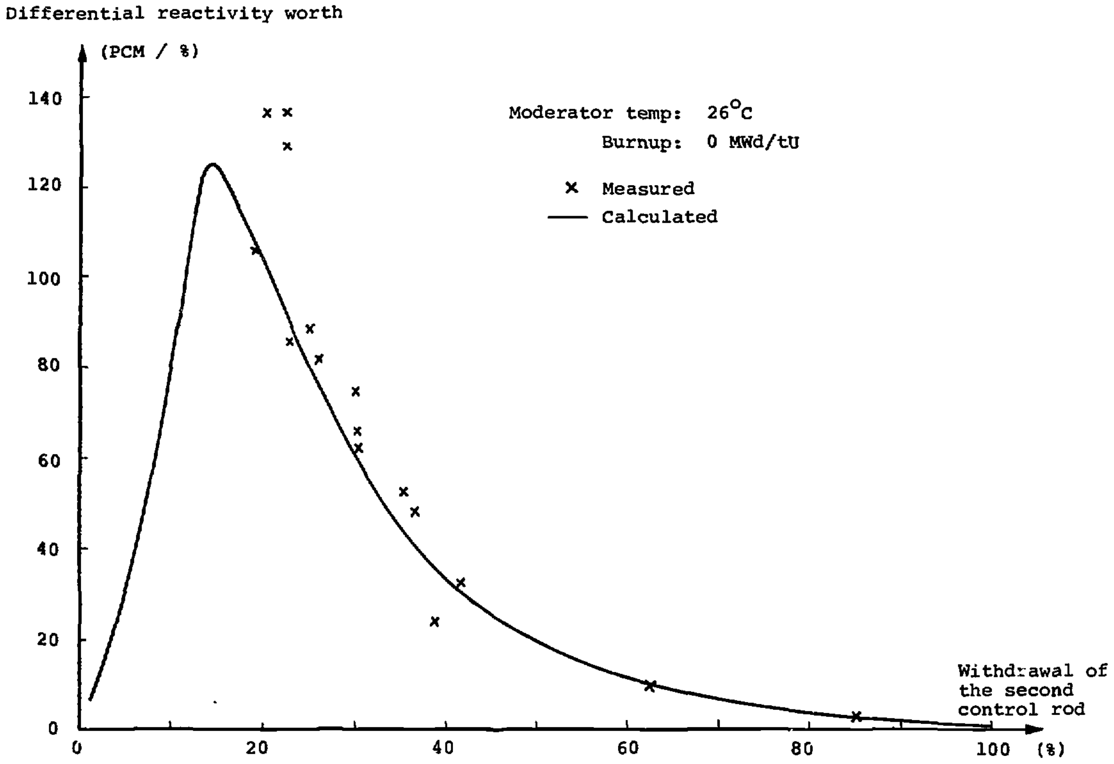
In a BWR the vessel dome pressure is held very nearly constant at about 70 bars (1000 psi) by menas of the pressure controller which operates on the turbine steam admission vavle. The pressure affects power through compression or decompression of the steam in the core i.e. through the void coefficient. The pressure coefficient which is positive is of interest whenever pressure transients arise. Again the pressure coefficient is not evaluated explicitly, but verified by means of power response tests when the pressure set point of the pressure controller is varied. A very convenient measurement is to impose a periodic pressure disturbance and register the power response. From this a transfer function (power to pressure) can be determined which is of significance in judging core dynamics and stability characteristics.

G. Power coefficient of reactivity

The power coefficient is of little practical use in its explicit form (e.g. as pcm per % power change) and is determined generally on a theoretical basis. It involves the void and fuel temperature coefficients and thus is difficult to measure. From an operating point of view it is instead convenient to use the core flow power control line (described in Ch. 2.2.3) as the basis for a measure of control capability, for instance, percent power change per percent change in core flow. (Note that control rods are seldom used to increase power at power levels above some 65 % of rated power).

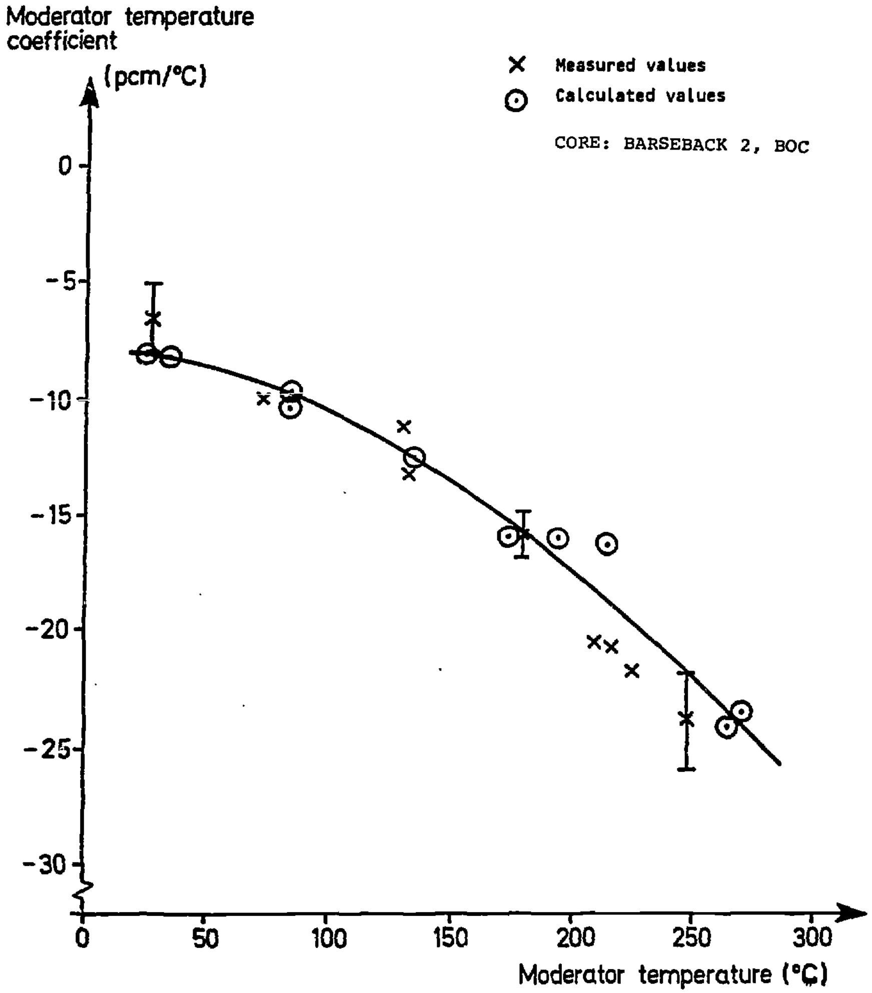
H. Core excess reactivity

Excess reactivity can of course never be directly measured in a BWR. As mentioned before, the theoretical methods are checked by reproducing the various critical core states that are obtained in going from a cold core to full power. A qualified method will then give adequate accuracy in calculating the various reactivity contributions. An indication of the size of the contributions is given in table 2.1.2-I.



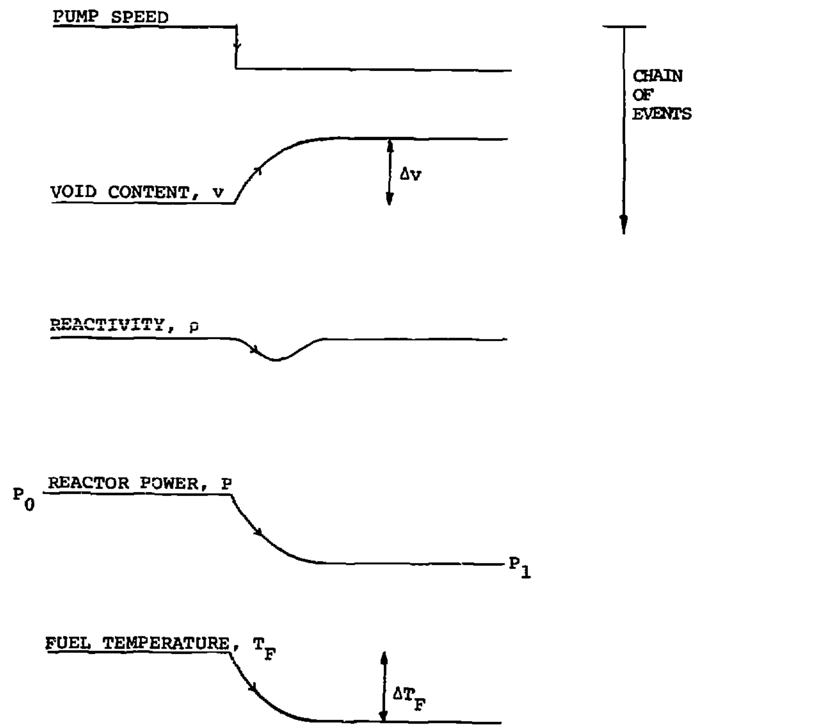
DIFFERENTIAL REACTIVITY WORTH OF THE SECOND CONTROL ROD

2.1.5
Figure 1



2.1.5
Figure 2

Moderator temperature coefficient



P is determined from

$-K_v \cdot \Delta v = K_{TF} \cdot \Delta T_F$ or $-K_v \cdot (v(P_1) - v(P_0)) = K_{TF} \cdot (T_F(P_1) - T_F(P_0))$

K_v = Void coefficient

K_{TF} = Doppler coefficient

2.1.5
Figure 3

REACTIVITY BALANCE

COOLANT RECIRCULATION FLOW VS FUEL TEMPERATURE

2.2.1 Total Thermal Power by Heat Balance Evaluation

The most accurate way to determine the total thermal reactor power, Q, is by making an overall heat energy balance over the reactor vessel taking all heat sources and losses into account.

The parameter Q appears in many calculations. Its value is essential in the evaluation of thermal margins, the thermal power is used to normalize the measured or calculated core power distribution, etc.

The value of Q has also economic implications. The generator output, G, is easily measured. The plant gross efficiency, G/Q, is a measure of how well the fuel is utilized.

The losses and sources of heat for the reactor vessel are sketched in figure 1. The energy balance of the system becomes

$$Q_{tot} = Q_{tur} + Q_{cl} + Q_{cr} + Q_{rad} - Q_p \qquad (1)$$

where

Q_{tot} - heat production of the reactor core

Q_{tur} - energy absorbed by turbine

Q_{cl} - losses in water cleanup demineralizer system

Q_{cr} - losses in control rod drive purge flow system

Q_{rad} - losses due to heat radiation

Q_p - energy input from recirculation pumps

It should be noted that equation (1) may vary slightly from one BWR type to another, but the basic principles are always the same.

The amount of heat to the turbine is obtained from measurements of reactor pressure (p), feedwater flow (W_{feed}) and temperature (T_{feed}), and the moisture (x) of the saturated steam going to the turbine. Then

$$Q_{tur} = W_{feed} \cdot \left[h_{steam}(p,x) - h_{feed}(p,T_{feed}) \right] \qquad (2)$$

where h is the quantity of enthalpy. It is here assumed that the steam flow equals the feedwater flow. The moisture content can not be continuously monitored, but has to be measured, for instance, during the start-up period.

The cleanup and control rod drive flow losses are known from temperature and flow measurements. For a 3000 MW unit they combined amount to approximately 10 to 15 MW.

The losses due to heat radiation is of the order 1 to 2 MW. The term Q_{rad} may be estimated during a pre-nuclear hot system test. In such a test the reactor water is heated to nominal values with electrical steam generators. Since all heat sources and heat sinks except Q_{rad} are well known, the value of Q_{rad} is easily computed. In case no pre-nuclear hot system test is performed, the heat radiation loss may be estimated from measurements of the temperature decrease rate during the early cooling phase of a reactor brought to zero power. A pre-condition for this procedure to be meaningful is that the decay heat is negligible, i.e. that the fuel is fresh.

The pump energy gain (a few megawatts) is a function of the coolant flow, the pump pressure head, the pump efficiency, and the water density.

$$Q_p = \frac{W_{cool} \cdot \Delta p}{\rho \cdot \eta} \qquad (3)$$

Using steam tables, Eq (1) may be approximated by

$$Q\,[\%] = \frac{W_{feed}}{W_{nom}} \cdot (100 + 0.3 \cdot \Delta T + 0.06 \cdot \Delta p + 0.8 \cdot \Delta X) + q_1 \qquad (4)$$

where

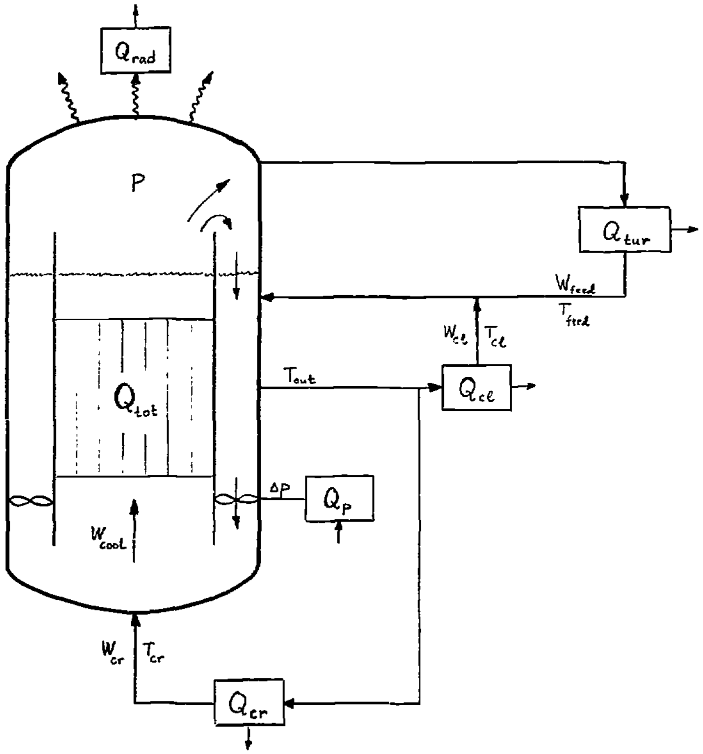
W_{nom} - "nominal" feed water flow

$\Delta T = 180 - T_{feed} \quad [^\circ C]$

$\Delta p = 70 - p \quad [bar]$

$\Delta X = 0.1 - X \quad [\%]$

$q_1 = (Q_{cl} + Q_{cr} + Q_{rad} - Q_p)/Q_{nom} \cdot 100$ = loss term



2.2.1
Figure 1 HEAT ENERGY SOURCES AND LOSSES

In practice, the loss term has an almost constant value, say 0.5 %, over the entire power range because of the nearly constant operating conditions of the cleanup system and the control rod drive purge flow.

Eq (4) is the starting point for our error analysis. From the values of its coefficients the relative importance of the various parameters is evident. The loss term, the feedwater temperature, the pressure, and the moisture are fairly easy to measure with good accuracy. Hence, their contribution to the overall uncertainly of the thermal power is small.

The dominating source of error is the feedwater flow. The flow can be measured with an accuracy of about one percent. Hence, this is also the uncertainty of the reactor power.

In addition to the above cited problem, the feedwater flow process signal is affected by noise caused by flow vacillation. The peak-to-peak noise value may be several times greater than the measurement inaccuracy. However, the importance of the noise on the thermal power determination may be reduced to any level by proper time averaging and noise filtering procedures.

2.2.2 Evaluation of Power Distribution and Thermal Limits

A. Introduction

The regulatory authorities have set up rules and limits for the reactor operation to maintain the integrity of the fuel and the safety of the plant. The reactor operator must show that the operation always complies with these rules.

As a measure of how far the core is from the limits the concept of "thermal margins" (or equivalently "thermal limits") is used. Roughly speaking, if the thermal margin is x % the reactor power may be increased x % before the regulatory constraints are reached. If the thermal limit is negative the operator should decrease the reactor power accordingly.

Another objective of the core supervision activities is to check that the actual core reactivity evaluation is satisfactoril; close to predicted behaviour and that the core is operated optimally from a fuel economical point of view. To meet the latter requirement the power shape should, during the entire operating cycle, be close to the so-called Haling distribution. An example of a typical power shape is given in Fig. 1.

The thermal limits are usually calculated, automatically or on request, by a process computer. For the event that the computer is down, a manual back up method must be available.

B. Limiting Parameters - CPR, LHGR, PCI

The main parameters of interrest for evaluation of the thermal loads of the fuel are the Critical Power Ratio (CPR) and the Linear Heat Generating Rate (LHGR).

The critical power is the point where film boiling breaks down and dryout occurrs.

During a dryout, the local heat transfer efficiency drops, causing in turn increased cladding temperatures, which may rise to 600-800°C. In consequence, the corrosion rate of the cladding accelerates, leading eventually to cladding failure and activity release to the coolant.

The operating conditions which determine the margin to dryout are, seen bundlewise, integral in character. The main parameters are bundle power and coolant flow. A dryout correlation, derived from laboratory experiments, determines the critical power.

The CPR is just the critical power divided by the actual power. A minimum CPR (MCPR) for the entire core is determined. The MCPR may not fall below some specified value, say 1.40, which will depend on the type of reactor system and dryout correlation used. The margin to dryout is, in this case, defined by

$$M_{CPR} = Q_{rel} \cdot \frac{MCPR - 1.40}{1.40} \qquad (1)$$

where Q_{rel} is the relative thermal power.

The other parameter, LHGR, pertains to the heat conduction through the fuel rod cladding. If the heat generation rate is too high the local temperature increase causes excessive release of gaseous fission products from the pellet structure to the gas plenum at the top of each rod as well as high cladding stresses due to pellet heat expansion. Eventually, the cladding may fail and release fission product gases to the coolant.

The maximum heat generation rate (MLHGR) of the fuel pins of the core is computed. It may not exceed, say, 46 kW/m. The margin to this LHGR limit is

$$M_{LHGR} = Q_{rel} \cdot \frac{46 - MLHGR}{MLHGR} \qquad (2)$$

The overall thermal margin of the core is
$M = \min (M_{CPR}, M_{LHGR})$.

Sometimes the Surface Heat Flux (SHF) is used instead of the LHGR. The SHF is just the LHGR divided by the circumference of the fuel pin and 46 kW/m corresponds to approximately 1.20 MW/m^2.

For LOCA (Loss of Coolant Accident) analysis the concept of Average Planar Linear Heat Generating Rate (APLHGR) is used. The APLHGR equals the LHGR divided by the local power peaking factor of the actual fuel bundle.

If the fuel is exposed to power shocks, i.e. steep increases or increase rates of the local heat load. Pellet-Cladding Interaction (PCI) induced fuel failures may occur. Those failures are statistical in nature and a limiting power increase rate has been determined below which the rod fuel failure rate should be acceptably small. This power increase rate limit is around 0.25 kW/m·h over a four-hour period or longer. Alternatively, a maximum power step increase of 1.0 kW/m at a time is allowed.

PCI failure only occurs for so-called unconditioned fuel, i.e. for fuel that is exposed to a heat load it has never experienced before. Thus, once the fuel has been exposed to a certain power level the PCI restrictions do not apply below that level, at least for a considerable burnup increase range.

To avoid PCI problems, specific operating procedures are implemented at high power governing control rod withdrawal and core coolant flow increases.

C. Calculational Methods

There are several methods employed for evaluation of the power distribution and the thermal limits. One method rests heavily on the LPRM and TIP readings. The LPRMs are calibrated to give a measure of the average surface heat flux of the four adjacent fuel bundles at the axial level of the LPRM detector (cf. the different LPRM calibration described in section 2.1.1). The local power distribution is obtained from an extrapolation of the TIP values to the four surrounding bundles. In this extrapolation a special void and burnup dependent factor is employed to translate the detector reading to power. Another factor modifies the equation for the event that control rods are present.

A difficulty with this method is to correctly estimate the void and the coolant flow distributions (the latter is needed for the dryout calculations). These are obtained by applying correlations which are based on precalculations. Other drawbacks are that a complete TIP run is needed whenever there is a more significant change of the core status and that core quarter symmetry has to be presumed.

In a different approach the detailed power distribution is evaluated online on the plant process computer by employing a three-dimensional core simulator that solves the neutronic and thermohydraulic equations of the core in a consistent way.

The calculational flow is shown in figure 2. A cell-program (see, for instance, references 1 and 2) is run off-site on a large computer to generate input data, like cross sections, to the core simulator. This computation is made once for every fuel batch. The core simulator (CS) determines a preliminary power profile based on actual process data. The CS is run on the plant computer after each major change of the core status, e.g. after control rod movements, once every n minutes if xenon is followed on-line, etc. The continuous core evaluation is provided by a fast Thermal Margins Program (TMP). It adjusts the preliminary power distribution to the actual process situation and computes the thermal limits of the core.

When selecting a core simulator some sacrifice in accuracy of the model must be made to acquire the necessary speed, say 10 minutes per calculation. The most popular models of today are based on socalled 1 1/2-group nodal theory (Ref 3-10). Each fuel bundle and control rod is treated individually. The fuel bundles are axially divided into a number of cubes, "nodes", of the approximate size 15x15x15 cm^3. A typical core consists of 10 000-20 000 nodes.

Cross section data for two groups are used, but the diffusion equation is reduced to one group by neglecting the leadage term in the thermal group diffusion equation. This approximation is based on the fact that the thermal diffusion coefficient is small compared to the node size and hence few neutrons cross the node boundary after they have been thermalized.

However, the use of two-group cross section data makes it possible to take doppler and xenon feedback effects into account.

Since the void content of the coolant strongly affects the cross sections, a thermohydraulic evaluation of the core is necessary. Power and void has to be found in an iterative fashion, see figure 3. After convergence, the expected values of the LPRM readings are calculated, i.e. the nodal powers around a detector are translated to equivalent fission chamber response.

New and more powerful generations of process computers will make it possible to relinquish some of the approximations that are made today, e.g. the elimination of the thermal group.

In addition to on-line evaluations of power profiles, the core simulator may be used for off-line purposes. The reactor engineer may use the program to update the burnup distribution regularily, to predict core behaviour, etc.

The Thermal Margins Program is based on actual LPRM readings, thermal power, coolant flow, CS power distribution and CS estimation of expected LPRM readings. The idea of the TMP is to adjust the CS power profile such that the CS LPRM values agree with the actual LPRM readings.

This method works well as long as the deviations between the core simulator power profile and the true power is large-scale (slowly varying) in nature.

Such situations exist after moderate changes of core coolant flow, thermal power, burnup profile, xenon distribution, etc. The model even corrects for model imperfections in the core simulator of large-scale nature.

Based on the modified power profile, the dryout margin and linear heat generation rate are computed and the overall thermal margin of the core is found. By comparing linear heat generation rates from different TMP runs the rate of change of the local heat loads is found. Hence, the program provides a PCI check.

D. Accuracy

The accuracy with which the local power, and hence the MCPR and the MLHGR, is finally determined depends of course on the size of the various error contributions. The sources of error may be grouped as follows.

1. Uncertainties in cross section data caused by model imperfections of the cell programs and inaccurate cross section libraries.

2. Uncertainties in process input data such as thermal power, core coolant flow and coolant temperature.

3. Imperfections in the core simulator equations caused by, for instance, the 1 1/2-group approximation and the spatial approximation of the diffusion equation. As already pointed out such errors are basically large-scale in nature and the errors are automatically corrected by the adjustment procedure of the Thermal Margins Program.

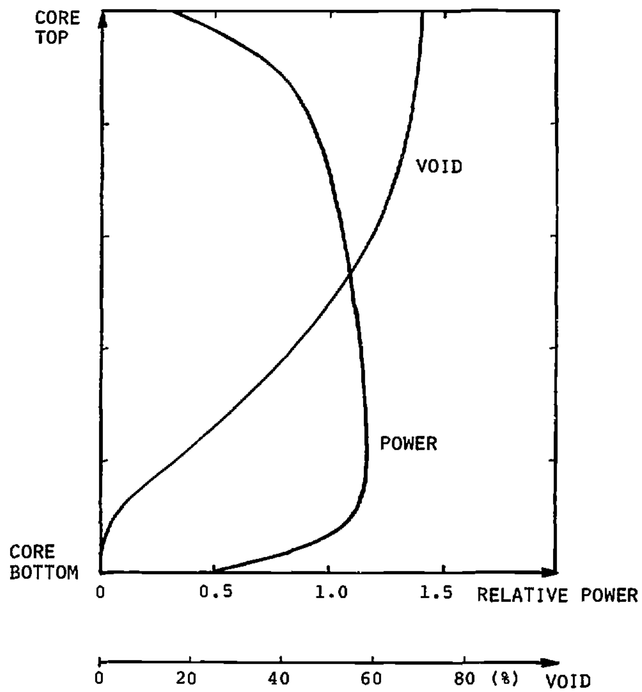
4. Uncertainties in the translation of the nodal power to detector response.

5. LPRM signal uncertainties caused by noise, signal calibration, drift, detector vibrations, etc.

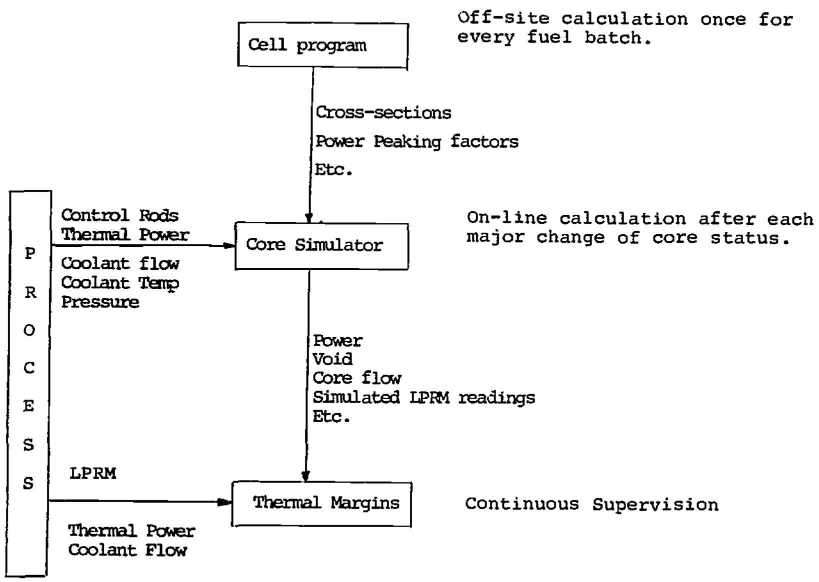
6. Interpolation errors in the power adjustment procedure of the Thermal Margins Program.

The overall nodal power error may be of the order 5-10 %. This uncertainty can be accounted for in the core surveillance by lowering the maximum linear heat generating rate allowed by some factor, say 2x5 % (from 46 kW/m to 42 kW/m). The uncertainty of the CPR

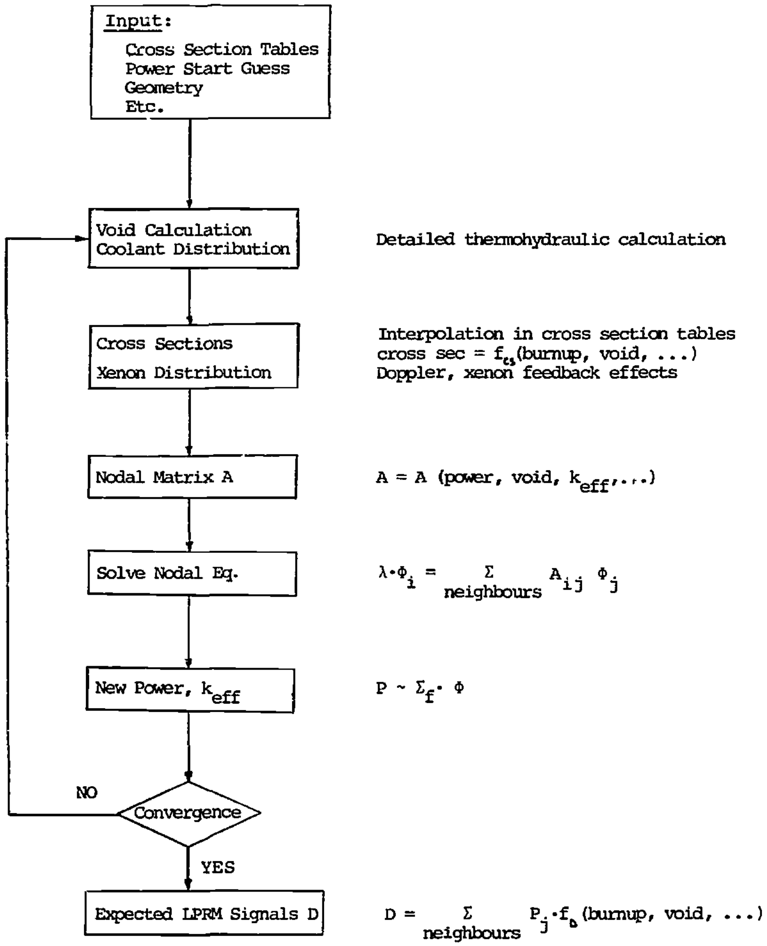
is usually lower than that of the LHGR since the margin to dryout is mainly dependent on the overall bundle power. When the bundle power is determined the axial errors are averaged out.



2.2.2 Figure 1 TYPICAL AXIAL POWER AND VOID DISTRIBUTIONS



2.2.2
Figure 2 CALCULATIONAL FLOW FOR EVALUATION OF THERMAL MARGINS



2.2.2
Figure 3 FLOW CHART FOR CORE SIMULATOR

2.2.3 BWR Control Systems - The Operating Range

A. Reactor Water Level and Pressure Control Systems

The main systems of the BWR steam cycle are shown in Fig. 1.

There are three important parameters - reactor water level, pressure, and power - that must always be maintained within close limits. For this purpose, special control systems are employed.

The water level controller adjusts the feedwater flow into the vessel, by means of the feedwater pumps, in such a way that the water level is kept constant, e.g. at 4.5 meters above the core. In case the level deviates too far from the nominal value (say $\pm$0.5 m) the reactor is scrammed. Should the feedwater pumps fail, or the main feedwater lines be closed, the water level is maintained by the auxiliary feedwater pumps or the core spray system.

During normal operation the reactor pressure is kept constant by the turbine admission valves. Should the pressure increase due to increased reactor power, more steam is admitted to the turbine until the pressure regains its nominal value (70 bar). This way the turbine plant acts as a slave to the reactor.

Under certain conditions the turbine valves are completely closed and the steam is dumped (bypassed) directly to the condenser. In that case, the pressure control is taken over by the dump control valves. If steam dumping is also prohibited, or when reactor isolation is initiated and the main steam valves are closed, the pressure is maintained by the relief control valves.

B. Power Control Systems

The two primary tools for controlling the reactor power are the coolant recirculation pumps and the control rods.

The power controller can be set in either of two modes: the plant power control mode (PPC) or the pump speed control mode (PSC). Using the PPC mode the power is kept at a given constant level by means of automatic core coolant flow adjustments with the recirculation pumps. In the PSC mode the pump speed,

and hence the core coolant flow, is kept at a given constant value while the power is allowed to drift or is adjusted with the control rods. Referring to the flow-power map of Fig. 2, the PPC (PSC) corresponds to operating the reactor along a horizontal (vertical) line.

If the reactor is operated in the plant power control mode under steady state conditions, reactivity will be lost continuously due to fuel depletion. The reactivity loss will be offset by a gain due to an automatic increase of coolant recirculation flow, i.e. the operating point moves from A to B in the example of Fig. 2. The operating point A may be reestablished by control rod withdrawals.

The physical reasons why the recirculation pumps can be used to gain reactivity are as follows. When the pump speed is increased the flow increases leading to decreasing void and the moderating capacity of the coolant is enhanced.

If the reactor is operated in the pump speed control mode, the depletion will force the operating point to move from A to C. The depletion reactivity loss is balanced by a gain due to a lower void content and lower fuel temperatures caused by the power decrease.

Another case when the operating point moves back and forth along the line A-B or A-C (depending on the control mode) is when the core is experiencing the reactivity effects of a xenon transient.

During high power operation, the plant power control mode is to be preferred over the PSC mode since under the PPC mode no power production loss takes place.

Let us assume that the reactor is operated in point D in the PPC mode. The plant power set point is switched to, say, 100 %. The power controller immediately responds by increasing the recirculation flow until the requested power is attained. As a result, the operating point moves continuously along the slightly curved line D-A. Such a line is called an "operating line" or, sometimes, a "control line".

The controller can change the power with a speed of up to about 20 % of full power per minute.

In the lower range of power (below approximately 65 %), low recirculation flow is normally preferred and the controller is set in the pump speed mode. Reactor power alterations are accommodated by adjustments of the control rod settings. In the flow-power map the operating point moves along the line E-D, see Fig. 2. The power increases are carried out at a rate of 1 to 2 % per minute.

This choice of power control is utilized at startup and shutdown and whenever power changes extend outside the recirculation flow control range capability.

In the high power range the power controller is normally in the PPC mode. During power increases the operating point moves along the line D-A. At steady power it moves along a horizontal line. Control rods are moved only to adjust the recirculation flow to a desired value (e.g. when flow changes move the operating point outside the permissible range).

Control rod movements should be avoided as much as possible at high power operation because of PCI (Pellet-Cladding Interaction) reasons. At the control rod tips the axial power gradiant is large, see Fig. 3, and even small control rod withdrawals may cause significant local power increases. If the minimum rod withdrawal increment is too large, control rod withdrawals are not acceptable at full power but have to be performed at reduced power.

C. The Operating Range

The design of the operating range is determined by the following factors.

1. Maximum permissible coolant recirculation flow is restricted either by the pump capacity or by the maximum permissible coolant flow through a fuel assembly. (Too high a coolant flow may cause vibrations or displacements within the assembly, etc).

2. Maximum power is the nominal power of the reactor on which the operating license is based.

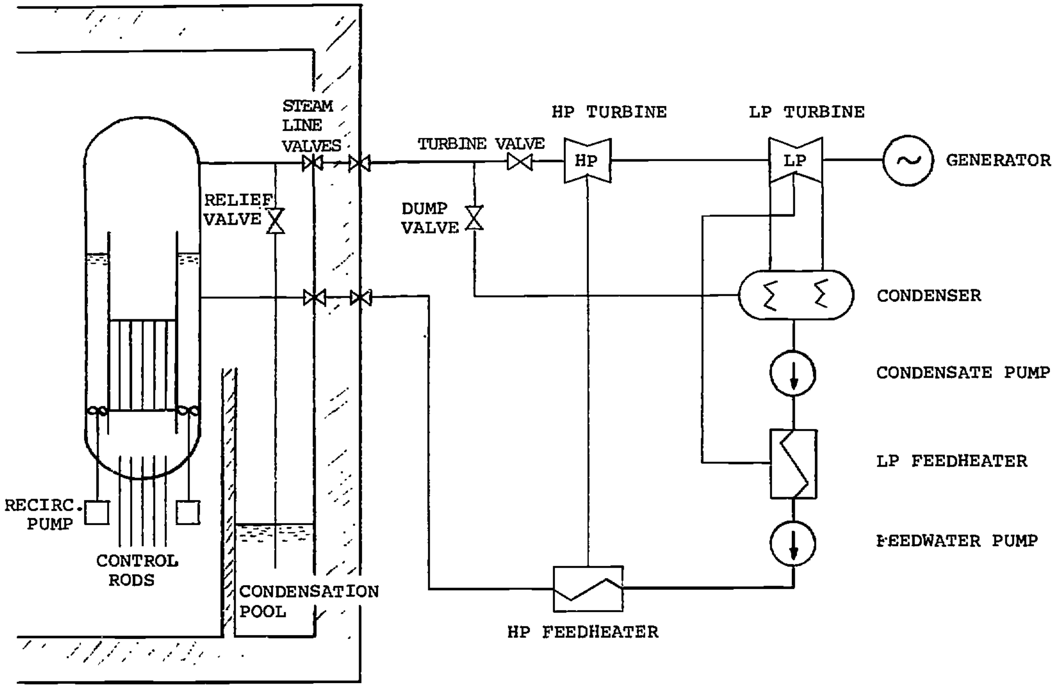
3. At high power the coolant flow must not be too low for two reasons. A low flow rate means poor cooling of the fuel pins and dryout may occur. A low flow rate also causes the reactor to become unstable, i.e. any power disturbance, such as results from a pressure perturbation, is not dampened out fast enough. A common stability condition for operation at rated power is that the ratio of two successive amplitudes in the core power oscillation caused by the disturbance shall be less than 0.25.

4. Below about 65 % of full power the minimum acceptable recirculation pump speed gives the minimum recirculation flow. For reactors with jet pumps the minimum flow of the operating range roughly equals the natural recirculation flow.

5. For jet pump reactors operation at low power and large flow is not permitted because of pump cavitation restrictions.

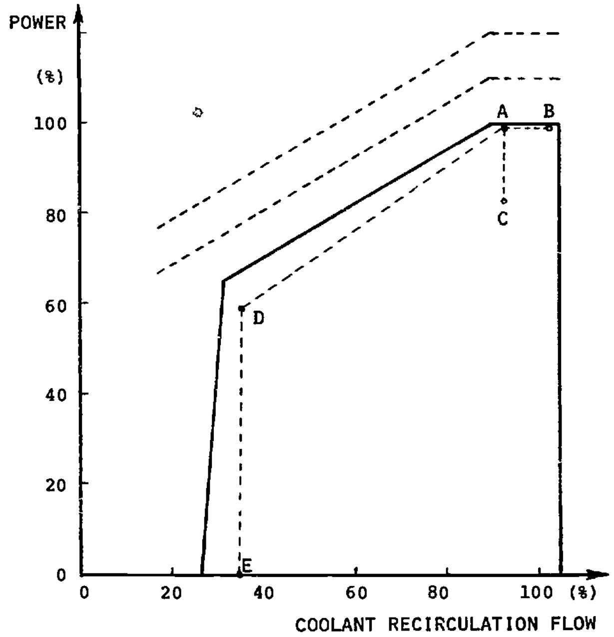
At 100 % of rated power the flow may, for a reactor with internal impeller pumps, vary between 90 % and about 105 % of nominal full core coolant flow. (The range 100-105 % of nominal flow is normally used only as reserve capacity.) For a reactor with jet pumps the flow range at 100 % of full power is rather limited.

If the power for some reason becomes too high the trip limits of the operating range may be reached, see Fig. 2. A fast decrease of the power by means of recirculation pump speed reduction (pump "runback") and a control rod withdrawal block may result or, eventually, the reactor is scrammed.



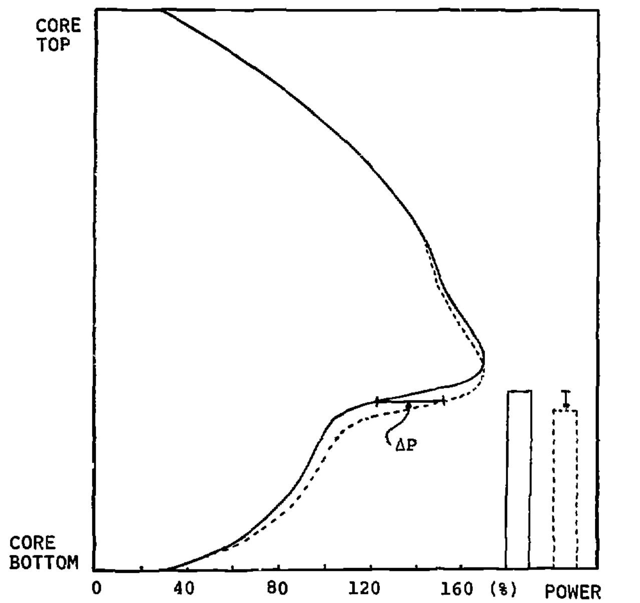
2.2.3
Figure 1

BWR STEAM CYCLE



2.2.3
Figure 2

POWER - FLOW OPERATING RANGE



2.2.3
Figure 3

EFFECT ON AXIAL POWER PROFILE OF CONTROL ROD WITHDRAWAL

2.2.4 Power Operation Modes

A. Reactivity control during operation cycles

In a power reactor, which normally is loaded with fresh fuel once a year, the core excess reactivity must be high enough to ensure that the reactor can be operated as intended to the next refuelling outage. In addition to compensation for temperature, void, and xenon, the excess reactivity must be sufficient to balance burnup during the operating cycle. The reactivity loss due to burnup is caused by the depletion of fissile material and the buildup of longlived fission products.

As a rule 20-25 % of the core is replaced each year.

The reactivity control system is designed to be able to absorb the core excess reactivity and to control the spatial power distribution in the core during any mode of operation. The system must be capable of keeping the reactor subcritical at any time according to the shutdown criterion.

Long-term reactivity control is provided by control rods and burnable absorbers, see Fig. 1. The short-term control is provided by recirculation pumps in addition to the control rods.

Fuel pellets containing a burnable absorber (BA) are present in a few rods in each reload fuel assembly and in most of the initial core fuel assemblies. The BA pellets consist of a solid solution of gadolinium oxide (Gd_2O_3) dispersed in sintered uranium oxide (UO_2). Depending on the various parameters of the operation of the reactor during the cycle, BA is typically present in two to seven of the 63 fuel rods in a fuel bundle with a content of 2-5 % BA by weight (8x8 fuel assemblies are assumed here). In advanced uses the BA content is axially varied in order to contribute to a favourable axial power distribution.

The introduction of burnable absorbers in the fuel reduces the maximum core excess reactivity during the operating season. The corresponding reduction of the reactivity control demand is enough to ensure that the cold shutdown condition is always satisfied. The primary objective of BA is thus to supplement the reactivity control capacity of the control rods.

The excess reactivity held by BA will decrease with irradiation in an almost linear fashion. Towards the end of the operating season practically all of the strong neutron absorbing isotopes Gd-155 and Gd-157 will have vanished and the residual reactivity effect of BA is quite negligible, see Fig. 2.

The control rods are designed to be able to bring the reactor into a shutdown condition at any time. During normal operation the objective of the control rods is to compensate for the burnup of the fuel and to shape the neutron flux distribution in the core in order to achieve an optimal power and burnup distribution. This shaping function is one of the reasons why the rods are inserted into the core from below.

The control of the reactor power during short-term ordinary operation is achieved either by control rod movements or by coolant recirculation flow changes. The control rods are used mainly during the start-up phase, up to about 65 % of rated power. Above this level, power is generally changed by varying the recirculation pump speed and thus the coolant flow.

Slow reactivity changes occurring at high power caused by xenon concentration variations or fuel burnup, are in the first place compensated for by the automatic coolant flow control. When the flow control approaches either of its range limits, control rods are moved at constant power in order to maintain coolant flow within these limits. This type of rod position adjustments will be made at intervals, which at certain times may be as short as a couple of days, at other times longer, depending on the temporary burnup - reactivity dependence.

The use of BA to reduce control rod inventory and to assist in axial power shaping and the employment of an appropriate refuelling strategy makes it possible to use a single control rod sequence during the whole operating cycle. This rod withdrawal strategy is called "Monosequence Operation". The idea is also known as the control cell core concept. An example of a monosequence is shown in Fig. 3.

However, during the first cycle it is generally desirable to change between black and white control rod sequences in order to obtain a radially uniform fuel burnup distribution. These changes of control rod sequence imply that those control rods which have been fully or partially inserted in the core are fully withdrawn, and certain others which have

previously been fully withdrawn, are fully or partially inserted into the core. A change of control rod sequence requires a temporary reduction of the reactor power (typically to 40 %).

During the first cycle 3 to 4 control rod sequences are used, depending on the length of the cycle and depending on vendor recommendations.

B. Reactor Startup from Cold Shutdown to Full Power

A varying amount of control rods (25-65 % of total inventory) must be withdrawn to make the reactor critical, depending on core reactivity condition (core life, xenon history, etc). From the beginning of control rod withdrawal, the power is monitored by the source range monitoring (SRM) system. Criticality is reached in the power range of 0.1-10 kW obtained from nuclear fission.

After criticality is attained, power is increased by further control rod withdrawal up to about 2 % of rated output to achieve a heat-up rate of typically 30-40°C per hour. During the heat-up the intermediate range monitoring (IRM) system is monitoring the neutron flux. While pressure and temperature are raised, steam may be increasingly admitted to the turbine system for establishing condenser vacuum and for heating the turbine. When the rated reactor vessel pressure is reached, pressure control is automatically initiated by the turbine control system. During this phase the feedwater system starts operating.

At full reactor vessel pressure, power is increased by further control rod withdrawals to about 15 %. The power controller is in the PSC mode. The coolant recirculation flow is still low and steam is bypassed to the condenser. In the range 8-15 % the power range monitoring (APRM) system takes over the neutron flux monitoring from the IRM system. At 15-20 % the turbine is brought into operation. The turbine speed is increased and the synchronization of the generator follows. During this operation reactor power is continuously increased. The pressure control is now transferred from the dump valves to the turbine governor valve.

Reactor and turbine generator output is increased up to about 65 % of rated power by withdrawal of control rods at low core coolant flow. At 65 % power the control rod pattern is close to that expected at full power.

Further power increases are carried out by increasing core coolant flow (i.e. recirculation pump speed) as illustrated in Fig. 2.2.3-2. An example of a reactor startup from a xenon free core is shown in Fig. 4 (in this case at least 3-4 days have elapsed after shutdown from full power operation). The xenon concentration approaches its steady-state value during the startup. This leads to an increasing reactivity loss which is compensated for by control rod withdrawals at certain high-power levels. At a constant power level the xenon buildup will at first result in increasing core coolant flow via the coolant recirculation flow control system. The coolant flow is then restored to recommended values by control rod withdrawals.

The thermal margin is evaluated intermittently. A permissible thermal margin is a prerequisite before a further increase in reactor power may be done.

Before full power can be reached the Xenon content in the core must not deviate to much from the full power steady-state value. After a short outage with a high Xenon content in the core, it is possible to reach almost full power in a couple of hours. The faster the return to the high-power region can be done, the less the Xenon content variation is.

For reactor shutdown, the procedure of approach to power is reversed. Thus the power is first decreased to about 60 % by reducing the recirculation pump speed to a minimum value. For further power reduction control rods are inserted at constant recirculation pump speed. The turbine valve closes concurrent with the power changes and keeps the reactor pressure constant. After disconnection from the outer grid and when the turbine control valve has closed, the task of maintaining the reactor pressure is automatically taken over by the dump valves and the residual power is handled by the turbine condenser.

C. Load follow

Nuclear power is taking an increasing part in the electric energy supply systems. The light water reactors will in a few years carry a major burden of electric energy generation in several countries. In this process the load follow capability of the reactors will become more and more important.

Load follow operation implies daily or weekend load swings as well as any other load adjustments that generally do not reduce power to less than about 40 % of rated output.

In some BWR systems two important factors that greatly facilitate load follow capability are the fine motion control rod drives (FMCRD) and the large permissible core coolant flow intervall at full power. The FMCRD permits control rod withdrawal at full power with acceptable small steps without violating the limitations on local power change rates (PCI restrictions).

With a large range within which the coolant flow may vary many Xenon reactivity transients can usually be absorbed by the automatic pump speed control system without power reductions.

Every change of reactor power is followed by relatively slow transients in local Xenon concentrations throughout the entire core. These transients affect both the core excess reactivity and the power distribution. A reduction in reactor power from an equilibrium condition implies at first a growing Xenon content. After 4-6 hours a maximum value is reached and then the amount of Xenon decreases to its new steady-state value. In the high-power region this reactivity variations is, as already mentioned, compensated for by a core coolant flow adjustment. In some cases control rod pattern changes are also required.

In the power range from about 60 % to 100 % of nominal power, daily variations can be accommodated with only insignificant restrictions. An example of a power reduction to 70 % during 8 hours is shown in Fig. 5. During the entire load follow the control rod pattern is held constant and the whole Xenon reactivity variation is absorbed by varying the core coolant flow.

For larger power ranges it is not sufficient to reduce power with the recirculation pumps alone. Below about 60 % power further reduction is performed with the help of control rods. This case is illustrated in Fig. 6, where the power is reduced to 40 % during 6 hours. The Xenon contant variation is in this case larger because of the larger power step.

After the core coolant flow has been reduced to about 55 % of the rated value at about 60 % reactor power, control rods are inserted into the core and the reactor power is decreased to 40 %. At this power level at constant low core coolant flow the growing Xenon reactivity loss is compensated by intermittent control rod withdrawals. The return to full power starts with control rod withdrawals until about 50 % of full reactor power is reached.

The power ascension above 50 % is done by increasing recirculation pump speed. Weekly variations, for example to meet reduced demands during weekends, are performed in a similar way. Control rod maneuvering is often required, in addition to core coolant flow changes, to absorb the Xenon reactivity variation. Weekly variations may require a more cautious return from 85 % to full power depending upon the past history of the movement of the control rods. The reason is that for longer periods at reduced power, the Xenon content in the fuel will reach a lower level and the return to full power will mean restoration of the Xenon content and thus a power redistribution.

D. Stretch-out operation

If all the excess reactivity of a reactor core has been consumed when approaching end-of-cycle, the chain reaction can no longer be maintained at full power. The reactor can, however, still operate for some time but only at a gradually reduced power output during a so-called coast down (or stretch-out) period. The reactor is then run in the constant pump speed mode. The depletion reactivity loss is compensated by the gain due to lower void content and lower fuel temperatures obtained when the power is decreased.

Coast down operation is employed for flexibility and economical reasons. The optimal coast down operating time is a trade off between the savings by augmented fuel burnup and the expenditure for replacement power. A typical cost optimum will often occur at about 10-15 % cycle extension.

More important than the economic gain is perhaps the increased flexibility in fuel management. Unpredictable parameters such as power demand and performance of other units connected to the grid will have a bearing on the decision to shut down the plant for refuelling. The reload batches can be repeatedly re-optimized to meet such demands. As the optimum coast-down period generally corresponds to a rather broad cost minimum, there is considerable flexibility in the choice of duration of the coast down period.

A typical power reduction during coast-down operation is shown in Fig. 7. For a BWR the rate of power reduction is about 0.35 %/day.

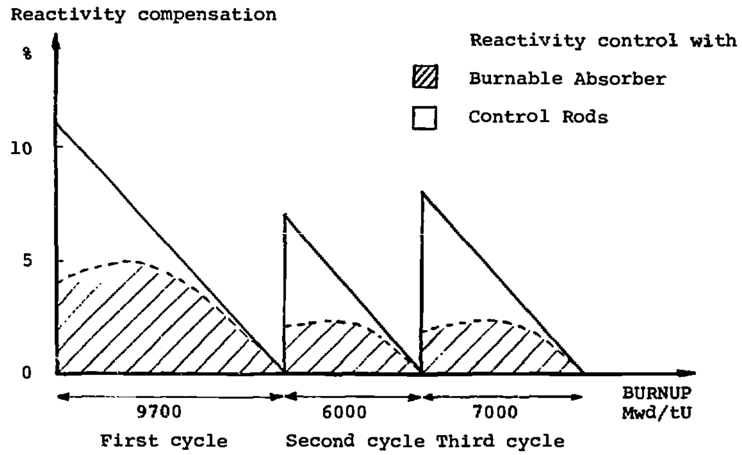
As an alternative to compensate the depletion reactivity loss in a stretch-out operation by reducing the power, the core inlet enthalpy may be decreased.

This move decreases the core void content and thus increases the reactivity. A lower enthalpy is obtained if one or more of the feedheaters are bypassed, thus lowering the feedwater temperature.

However, the efficiency of the turbine system is reduced to some extent by this procedure, so that this mode of operation may in some cases be unprofitable.

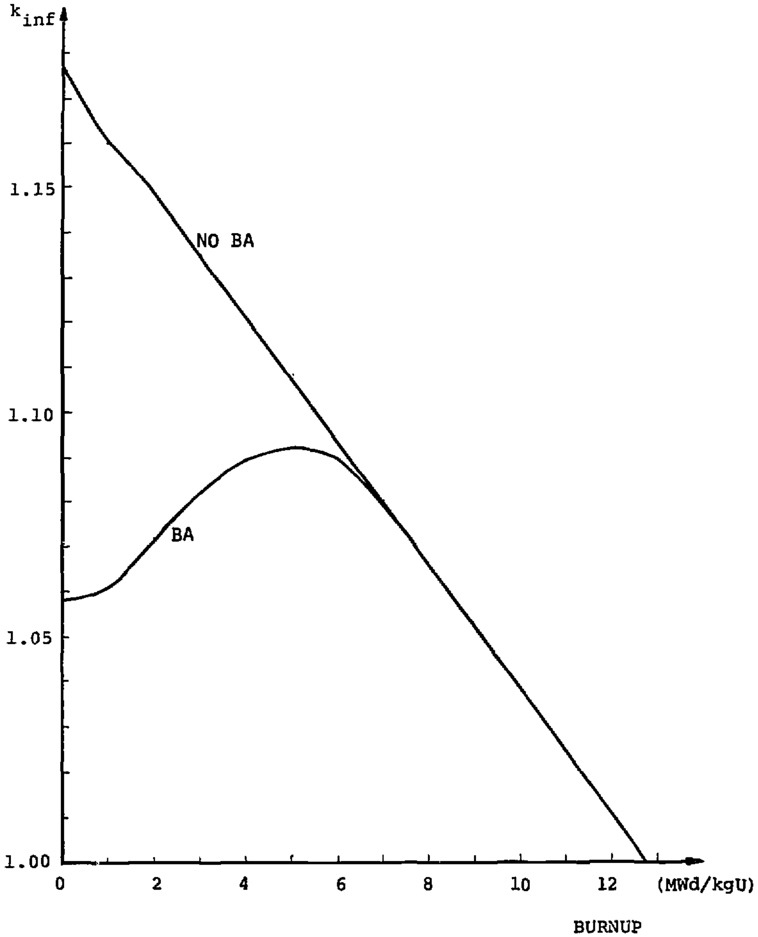
E. Spectrum Shift Operation

During steady-state operation it is desirable to keep the core coolant flow as low as possible within the operating range. This can be done during most of the cycle when excess reactivity is available. Operating at low core flow implies a high void content and a hard spectrum, which generates excess plutonium. This plutonium will provide an extended cycle life time when core flow is increased to its maximum value and thus the average void is decreased at end-of-cycle. This operating strategy is known as spectrum shift operation.

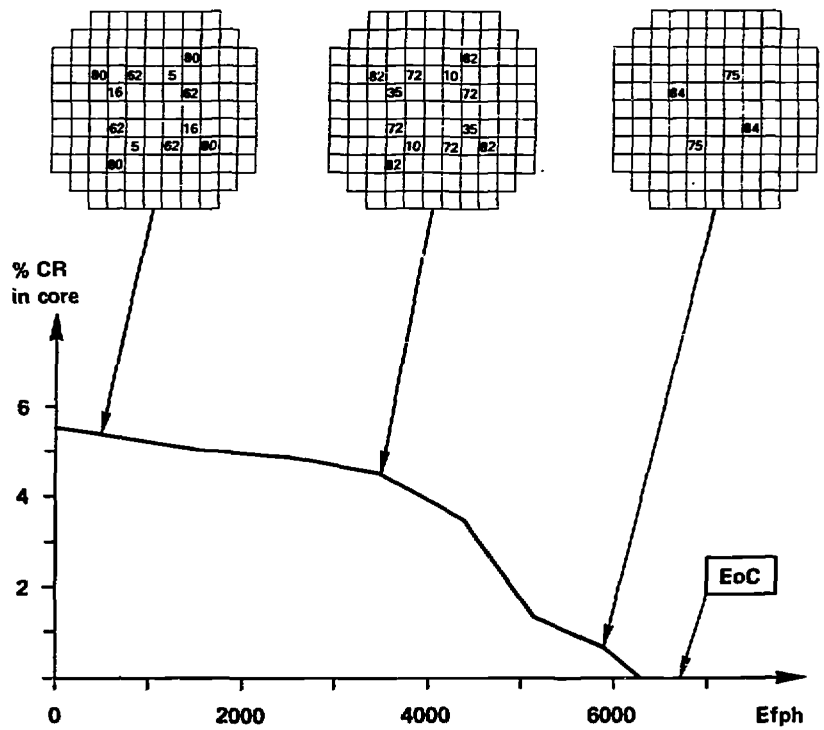


2.2.4
Figure 1

LONGTERM REACTIVITY CONTROL

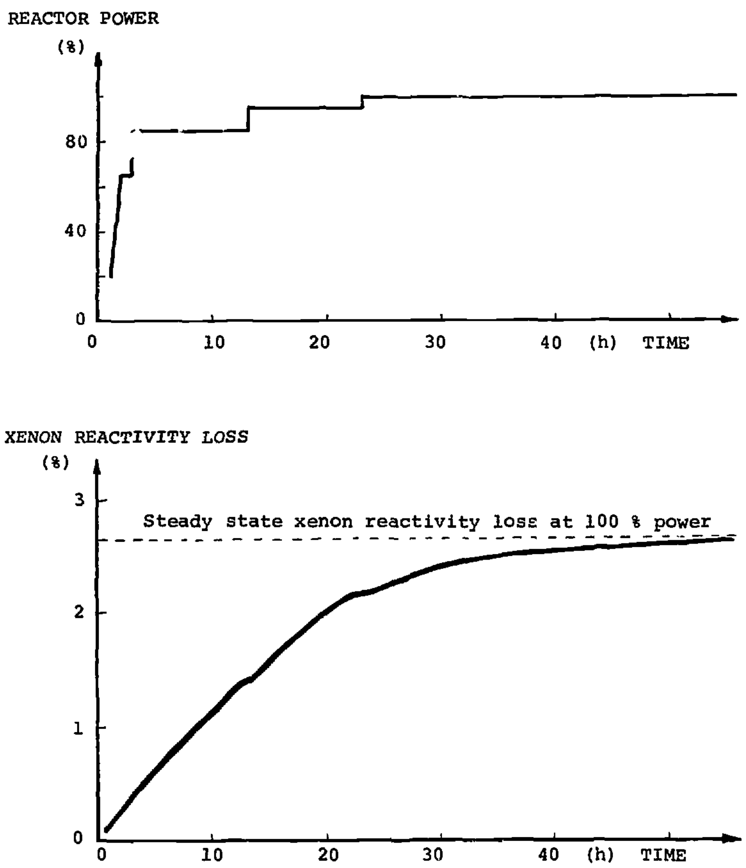


2.2.4
Figure 2 K_{INF} AS A FUNCTION OF BURNUP FOR FUEL WITH AND WITHOUT BA

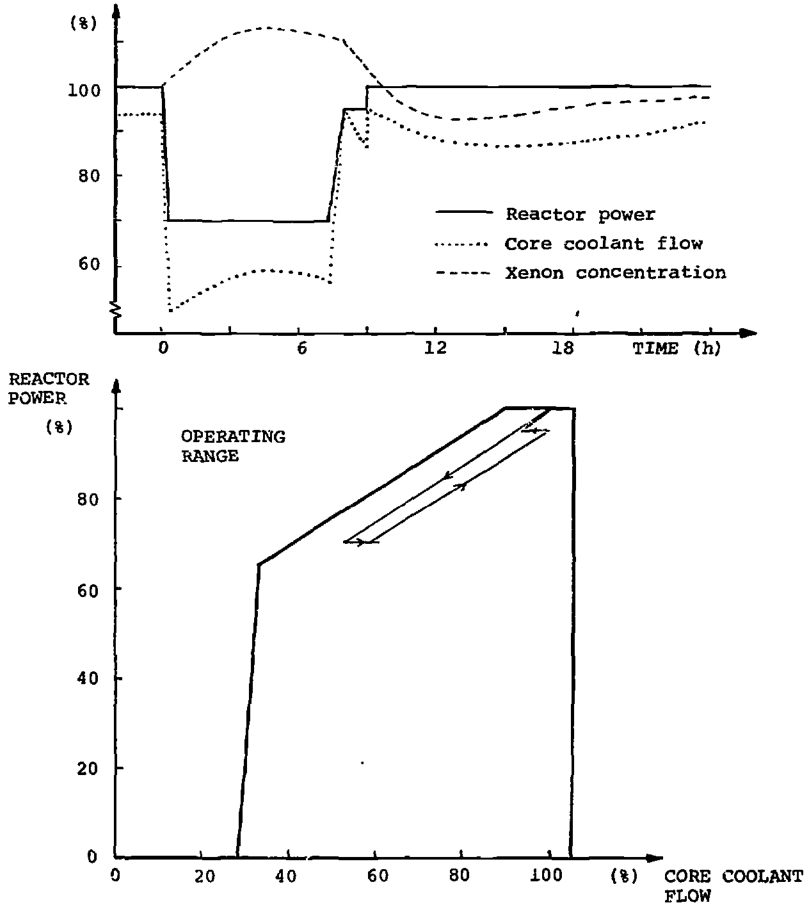


Note: Control rod withdrawal given in percent withdrawal.
Unmarked rods are completely withdrawn.

2.2.4
Figure 3 MONCSEQUENCE OPERATION

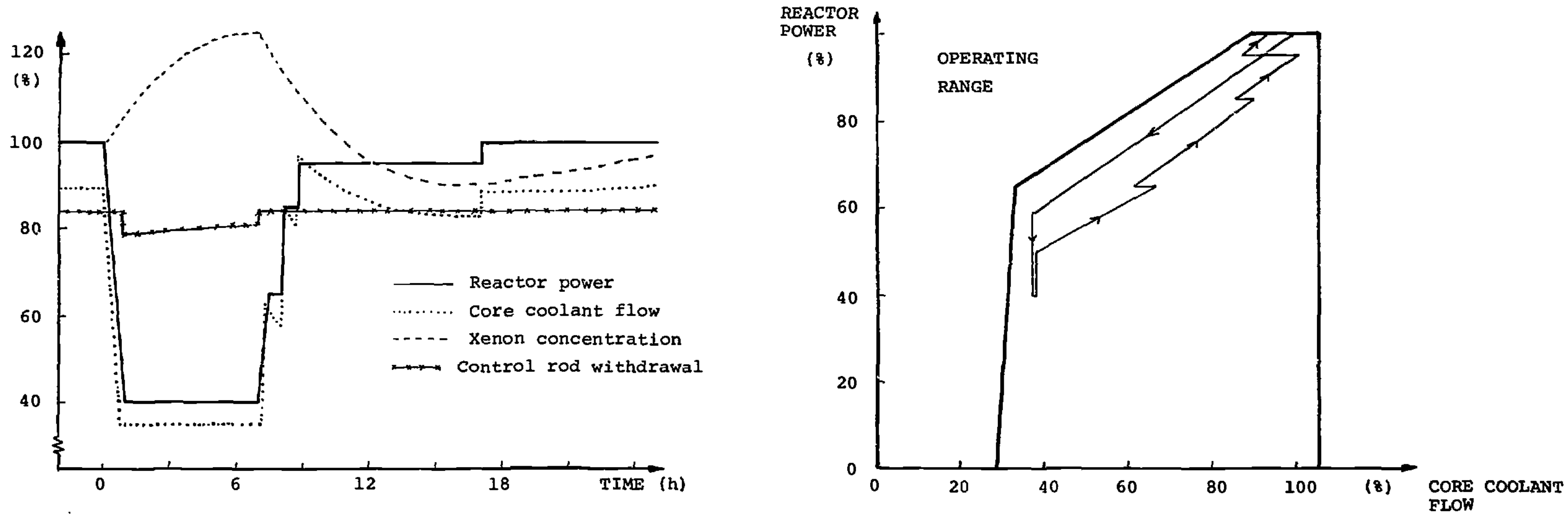


2.2.4
Figure 4 EXAMPLE OF REACTOR START-UP FROM XENON FREE CONDITION



BWR - DAILY LOAD FOLLOW

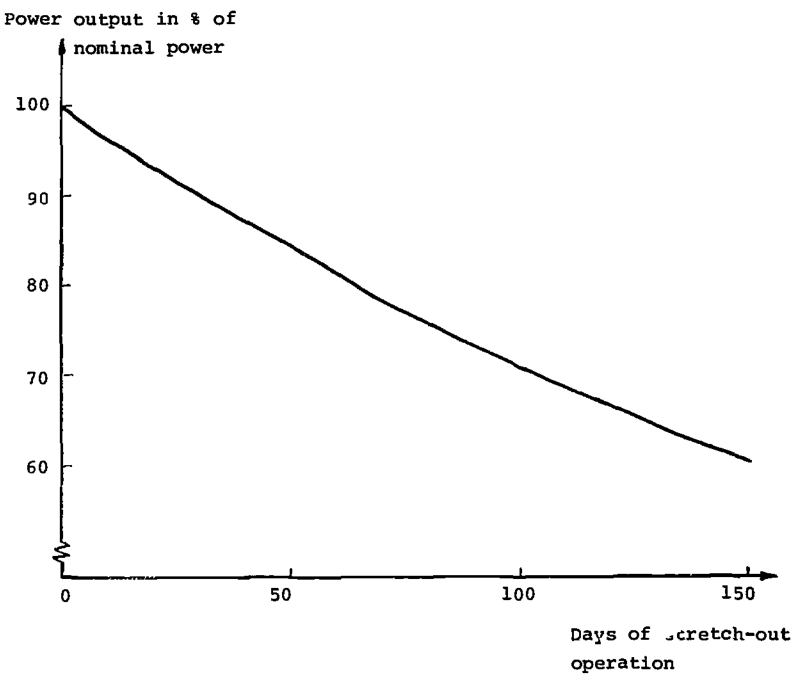
2.2.4
Figure 5 EXAMPLE OF POWER REDUCTION TO 70% WITH USE OF RE-CIRCULATION PUMPS AT CONSTANT CONTROL ROD PATTERN



2.2.4
Figure 6

BWR - DAILY LOAD FOLLOW

EXAMPLE OF POWER REDUCTION TO 40 % WITH USE OF RECIRCULATION PUMPS AND CONTROL RODS



2.2.4
Figure 7

STRETCH-OUT OPERATION

2.2.5 Reactor Transients

A. Introduction

During a transient the sequence of events that follow upon it depends to a certain degree on the specific type of BWR that is under consideration. For this chapter a BWR with internal impeller recirculation pumps will be presumed.

The main systems pertinent to a discussion of the most common transients are shown in Fig. 2.2.3-1.

The control systems necessary for the reactor to be able to cope with the transients have been discussed in Ch. 2.2.3.

The transient behaviour of the reactor is analyzed with regard to fuel cladding and reactor coolant system integrity. Parameters which have been investigated include

fission power
thermal power
reactor pressure
core steam void
feedwater flow
relief system flow
core coolant flow
reactor water level
dryout margin

For some of these parameters trip limits are defined. The reactor is scrammed, for instance, when the pressure exceeds a certain value (about 74 bar).

In addition to the scram limits the licensing authorities have defined design values which should never be exceeded. The vendor must show in the Final Safty Analysis Report (FSAR) that the reactor can always be operated within these limits. As an example, the maximum permissible pressure is about 94 bar. The critical power ratio may never fall below 1, or, taking calculational uncertainties into account, never below, say, 1.10.

B. Reactor Isolation

Complete steam flow interruption by closing of the main steam line isolation valves is automatically executed upon pipe break accidents inside the reactor containment or in the steam lines and upon high water level in the reactor pressure vessel.

The same signals which initiate steam line isolation also initiate reactor scram, fast reduction of recirculation flow and feedwater flow, and opening of the relief valves to blow off steam to the condensation pool. The steam line valves start to close a fraction of a second after the release of the isolation signal and they are completely closed after approximately one second. After four to five seconds the recirculation pump speed has reached its minimum value (20 % of full speed). At that time the control rods are fully inserted.

As a consequence of the scram and the reduction of the recirculation flow the neutron flux is rapidly reduced to decay power values. The thermal power decreases more slowly because of the fission product decay heat and the ceramic fuel's ability to store heat.

Initially, the pressure increases since the reactor is still producing steam. The relief system, however, reduces the pressure to its nominal value and keeps it constant by the relief control valves. Since the pressure reactivity coefficient is positive, the initial pressure increase will somewhat delay the power reduction.

The water level depends on three factors; average void fraction, relief flow and makeup water flow (= feedwater flow or auxiliary feedwater flow). As the thermal power decreases the steam bubbles collapse and the water level is reduced. The makeup water, however, slowly restores the level to its nominal value.

In all those transients where relatively cold feedwater and/or auxiliary feedwater is pumped into the reactor vessel, the pressure will decrease 10 to 30 bar below the normal value within a few minutes. When all of the water in the reactor is saturated, the pressure will increase slowly due to the decay power heating. Eventually, the pressure will be restored to its nominal value.

Condensation pool cooling is provided by a special cooling system.

An example of an isolation transient is shown in Fig. 1.

C. Turbine Trip

Turbine trip is initiated at certain abnormal conditions in different systems of the plant; turbine overspeed, large turbine vibrations, high condenser pressure, generator faults, reactor scram, etc.

Tripping causes all turbine control and stop valves to close. The reactor power is rapidly decreased by reducing the recirculation pump speed to a minimum within five seconds. Dumping of reactor steam to the condenser is initiated by opening of the dump control valves which control the reactor pressure. (It is here assumed that the condenser has a bypass capacity to receive the steam produced at about 60 % reactor power or more). Notice, that the reactor is not scrammed.

There is a small pressure increase as the turbine control valves close while the dump valves open. The fission power is rapidly reduced, apart from a small pressure induced increase during the first second. Concurrent with the power reduction, steam production goes down. The feedwater flow is reduced by the feedwater controller. The transient water level increase is negligible.

The reactor power temporarily stabilizes on an approximate level of 50 %. However, the turbine trip reduces feedwater temperature to about 30°C (from 180°C) after a few minutes as the feedwater reheating is stopped. The accompaning increase of core inlet subcooling causes the reactor power to rise slowly (cf Eq. 2.2.1-4 where the effect of a feedwater temperature drop is given explicitely in terms of total power increase). If the power reaches a prespecified value (60 %) while the feedwater temperature is still below 100°C a "partial scram" is actuated (only one scram group is inserted). On top of this rise there is a significant spatial redistribution of power.

This partial scram protects the fuel from any cladding failures.

The generator breaker will open at a generator output near zero. Auxiliary power is then provided from the external grid. The reactor response to a turbine trip is mild and the integrety of the nuclear process system and the fuel cladding is in no way threatend.

The turbine trip is studied in Fig. 2.

In certain situations, e.g. when a high condenser pressure is detected, steam dumping to the turbine condenser is prohibited. In such a case, the dump stop valves do not open, resulting in a complete interruption of steam flow. The turbine trip and dump blocking signal initiates reactor scram, fast reduction of recirculation flow, and relief valve opening.

D. Pump Trip

Tripping of one recirculation pump instantaneously yields a rapid reduction of its recirculation flow. The flow through the pump will reverse and within a few seconds a stationary back-flow will be reached.

As a consequence of the reduced core coolant flow, the reactor power starts to decrease. The power controller may be set in either the power control mode or the pump speed control mode. In the latter case the speed will be kept constant while the power is allowed to coast down and a sustained power reduction is experienced until the operator takes corrective actions. In the former case, which is the normal mode of operation at high power and which will be assumed from now on, the power controller counteracts the fission power reduction by increasing the speed of the remaining pumps. The pumps are normally operated at a speed corresponding to 90-100 % of full core coolant flow, but may for shorter periods of time be run at a speed equivalent to 105 % of full flow. This means that the power controller may normally, but not always, be able to restore power to 100 % of the pre-trip value.

In the example of a pump trip given in Fig. 3 the transient starts with a reactor power of 100 %. The fission power falls off to 50 % within a second. It returns to 90 % within another second and will finally reach its original steady state value of 100 %. The water level is constant within a few centimeters. The pressure varies between some ten kPa. If more than one pump trips it is not possible to restore full power.

E. Bypass of Feedwater

The feedwater heater system comprises a number of low pressure (LP) and high pressure (HP) heater stages. Each heater set is made up of two parallel heater strings with its own bypass equipment. Bypass is

actuated if a low water level is detected inside a heater.

Bypass of one or several LP heaters does not affect the short time reactor dynamics since the resulting cold water front will be smoothed and attenuated by the HP heaters before it reaches the reactor vessel. On the other hand, bypass of a HP heater will give rise to a realtively sharp cold water front, which will be transported to the reactor core. The shock, though, will be mitigated by the turbulent mixing in the feed line piping, the reactor downcomer, and the lower plenum. Loss of one HP heater string will now be assumed.

The average transit time through the downcomer and the lower plenum is about 10 s. The feedwater temperature decreases by 20^{o}C in 5 s. The disturbance of the inlet subcooling will thus be fully developed 15 seconds after the first cold water has reached the reactor vessel.

The emergence of cold water in the core is first noticed as a decrease of the water level by a few centimeters. The feedwater control system compensates by temporarily increasing the flow rate.

The negative void coefficient makes the neutron flux increase by some percent. The power controller responds by reducing the recirculation pump speed.

F. Control Rod Ramp

Erroneous control rod withdrawals can arise as a result of component faults or operator mistakes. The core is most sensitive to withdrawal errors during approach to criticality and full power operation. This subsection is going to analyse the full power case.

All normal control rod maneuvering should be done according to a preplanned and carefully evaluated rod withdrawal sequence. Various hardware features can also be employed to strongly limit the extent of errors, see Ch. 2.1.2. This should render rod selection and withdrawal errors to be infrequent.

In the analysis it shall be assumed that a withdrawal error that amounts to a distance of 5 % of core hight occurs at full power operation.

The influence on the global reactor power of an erroneous maneuvre depends very much on the initial control rod position and pattern type. The greatest reactivity disturbance is obtained if the rods are moved at a level of approximately 20 % of total withdrawal.

The rods may be maneuvered separately or in groups of at most 8 rods. A conservative figure for the maximum reactivity release is 14 pcm/% displacement per rod in the eight-rod group. Thus, the maximum reactivity release caused by a 5 % displacement error in an eight-group will be 560 pcm.

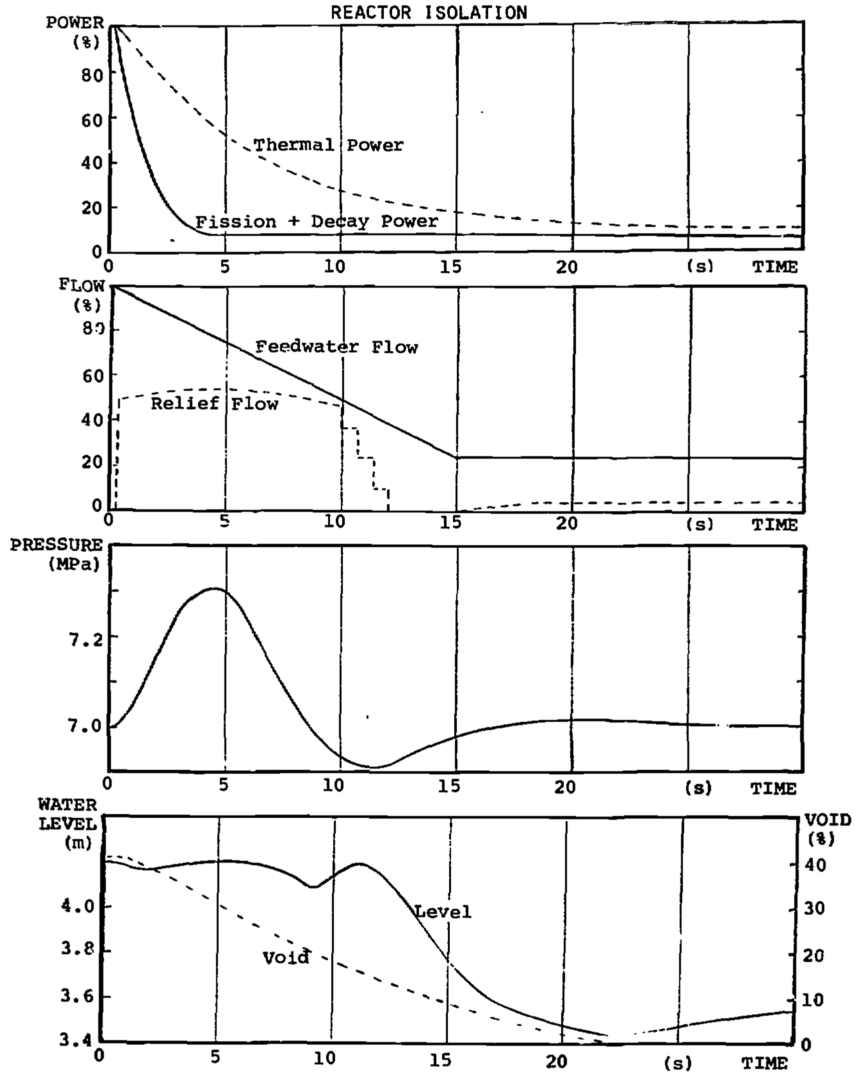
The effect of such a disturbance depends on the mode of control. When the automatic power control is employed the power increase is restricted to one to two percent. In the coolant recirculation flow mode the increase is higher, up to 10 % in extreme cases.

In the case of pump speed control mode, the reactor power increases monotonously. When the turbine maximum capacity (some percent above 100) is reached the dump valves open and the excess steam production is dumped to the condenser. If the neutron flux reaches the trip limit fast reduction of recirculation pump speed is initiated.

The point in time when the flux peak is reached is determined by the rod maneuvering speed. Assuming a speed of 0.4 % rod displacement per second, the maximum flux is obtained 12.5 seconds after initiation of the 5 % rod withdrawal.

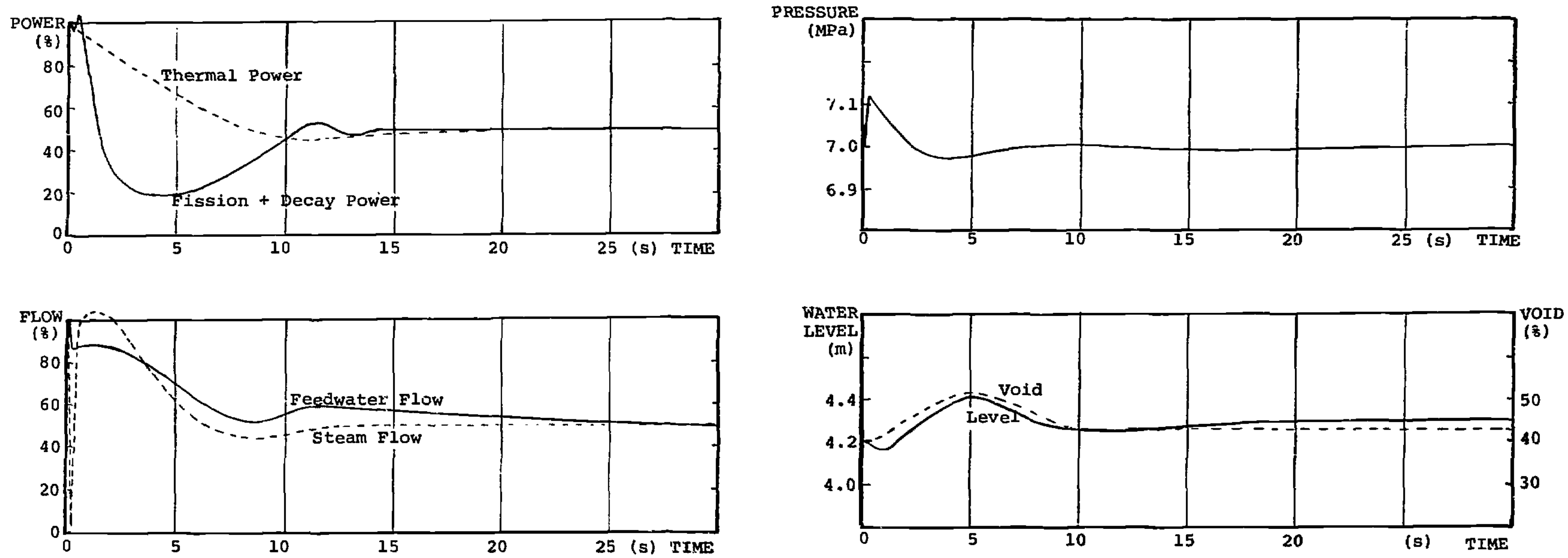
In the case of power control mode the reactor power increase is more modest since the power controller notices the rise in neutron flux and compensates by reducing the recirculation pump speed.

In addition to the global effects, erroneous rod withdrawal also distorts the core power distribution. This may lead to above-normal local heat loads in the fuel without exceeding rated power output. The local effect may be of the same order as the global disturbance.



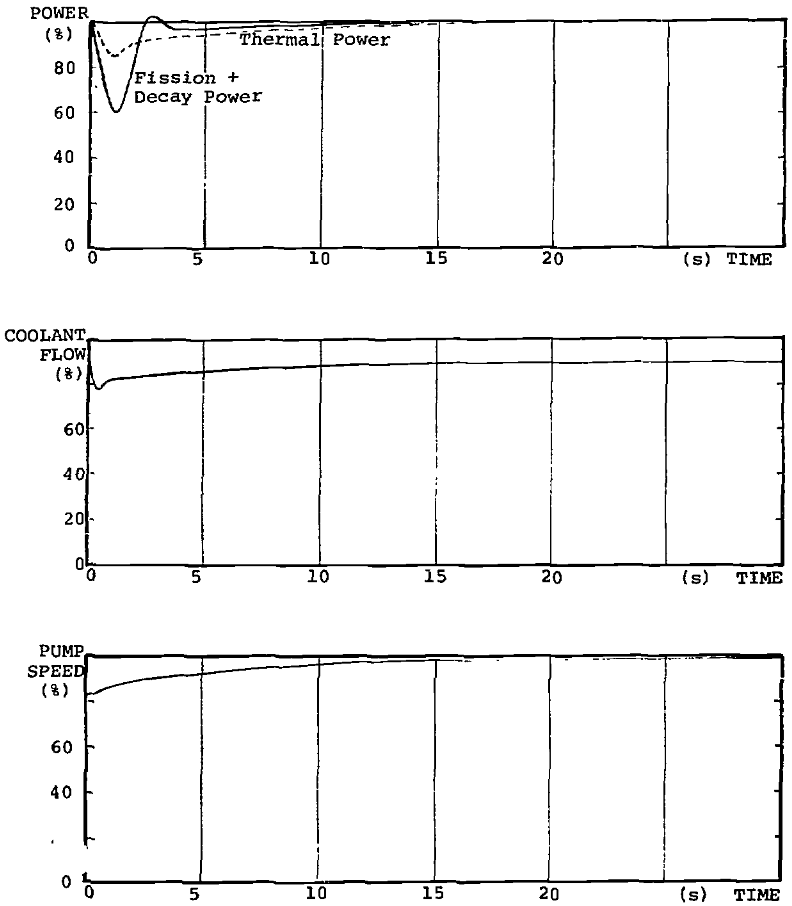
2.2.5
Figure 1

TURBINE TRIP



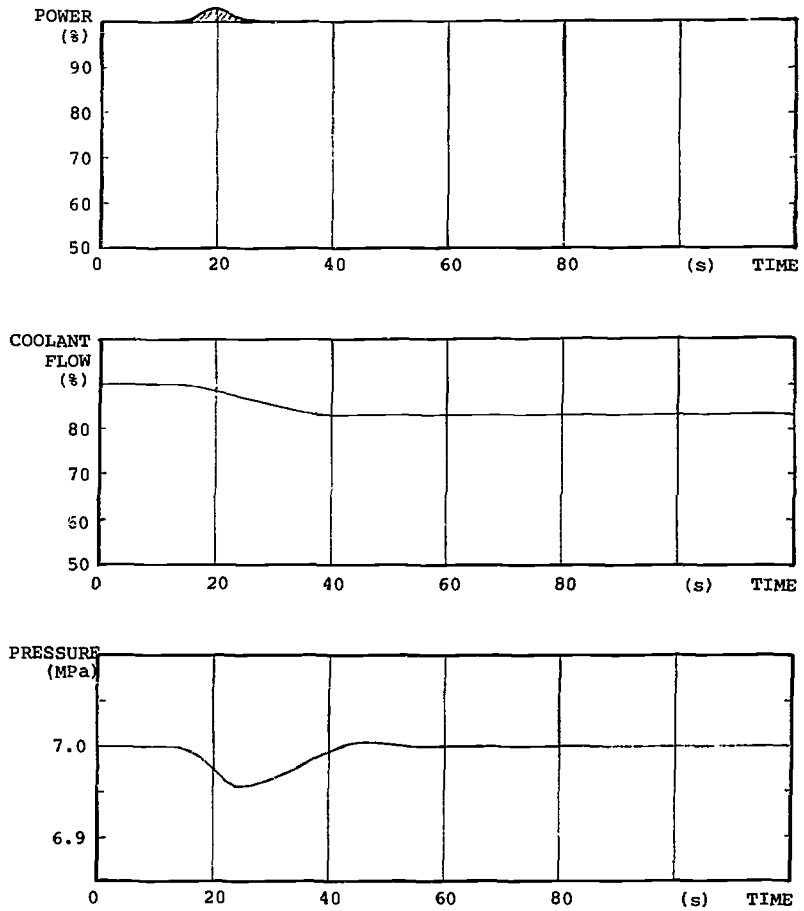
2.2.5
Figure 2

PUMP TRIP



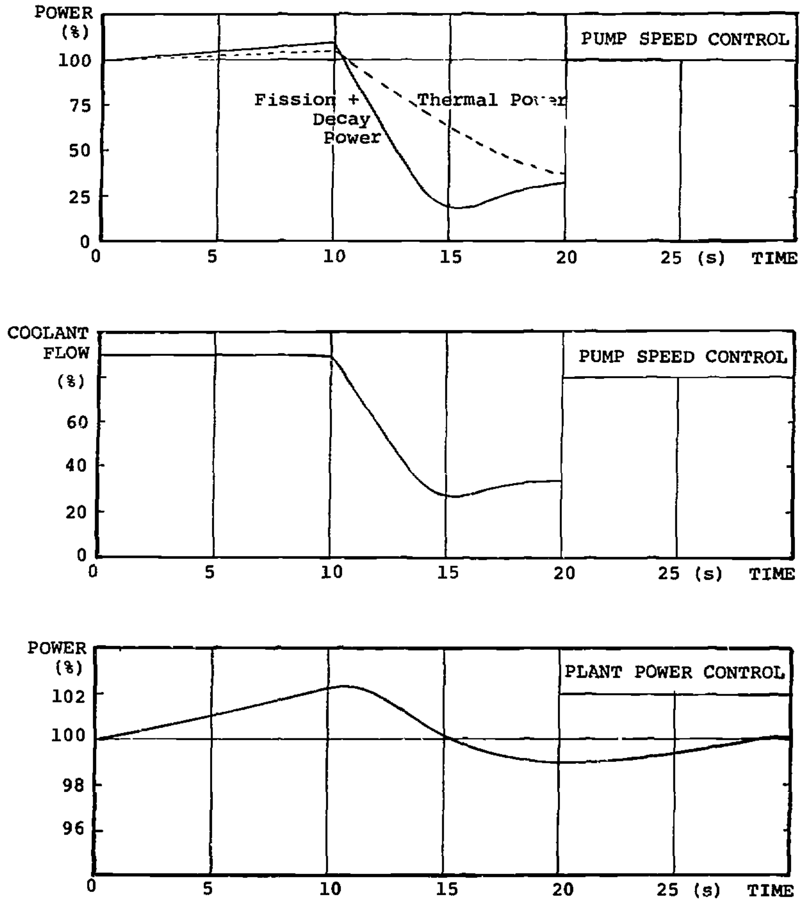
2.2.5
Figure 3

BYPASS OF FEEDHEATER



2.2.5
Figure 4

CONTROL ROD RAMP



2.2.5
Figure 5

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