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THERMAL-HYDRAULIC DESIGN OF THE 200 MW NHR

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Abstract

The thermal hydraulic design of the 200-MW Nuclear Heating Reactor (NHR), design criteria, design methods, important characteristics and some development results are presented in this paper.

1. DESIGN TASK AND IMPORTANT CHARACTERISTICS

The 200-MW NHR is a demonstration nuclear heating reactor. It operates at low temperature and low pressure conditions in an integrated primary circuit with natural circulation and a self-pressurizer filled with nitrogen gas. Its longitudinal section is shown in Fig. 1.

The basic task of the thermal hydraulic design of the 200-MW NHR is to provide an adequate heat transfer ability that is matched with the heat-generating ability in the core, to provide a set of reasonable parameters for the intermediate loop, to enable the 200-MW NHR to have good economic benefits under the condition that the integrity of the three levels of radioactivity barriers for preventing the release of radioactive products is ensured, and that the requirements of various operation safety are satisfied.

The 200 MW NHR will be used for domestic heating for cities, so adequate safety is highly required. Based on the design and operation experience with the 5-MW Experimental Heating Reactor (NHR-5), the design of the 200-MW NHR is underway. The thermal hydraulic design has different characteristics in comparison with conventional PWR nuclear power plants as follows:

- integrated arrangement of the primary circuit, natural circulation operating pattern,
- a self-pressurizer in the upper dome of the reactor vessel with nitrogen gas,
- fuel bundle with box, flow distribution by throttling at the core inlet,
- lower temperature, lower pressure, lower reactor thermal parameters (volume power density, linear power density).

2. DESIGN CRITERIA

There are no official design standards for NHRs in China up to now. Under the agreement of the National Nuclear Safety Administration of China, the following criteria are used in the design of 200-MW NHR [1].

General design criteria:

- To assure fuel elements not to be damaged in normal operation, anticipated operating occurrences and infrequent faults.
- Fuel elements may be damaged in limited accidents, but the effective radioactive

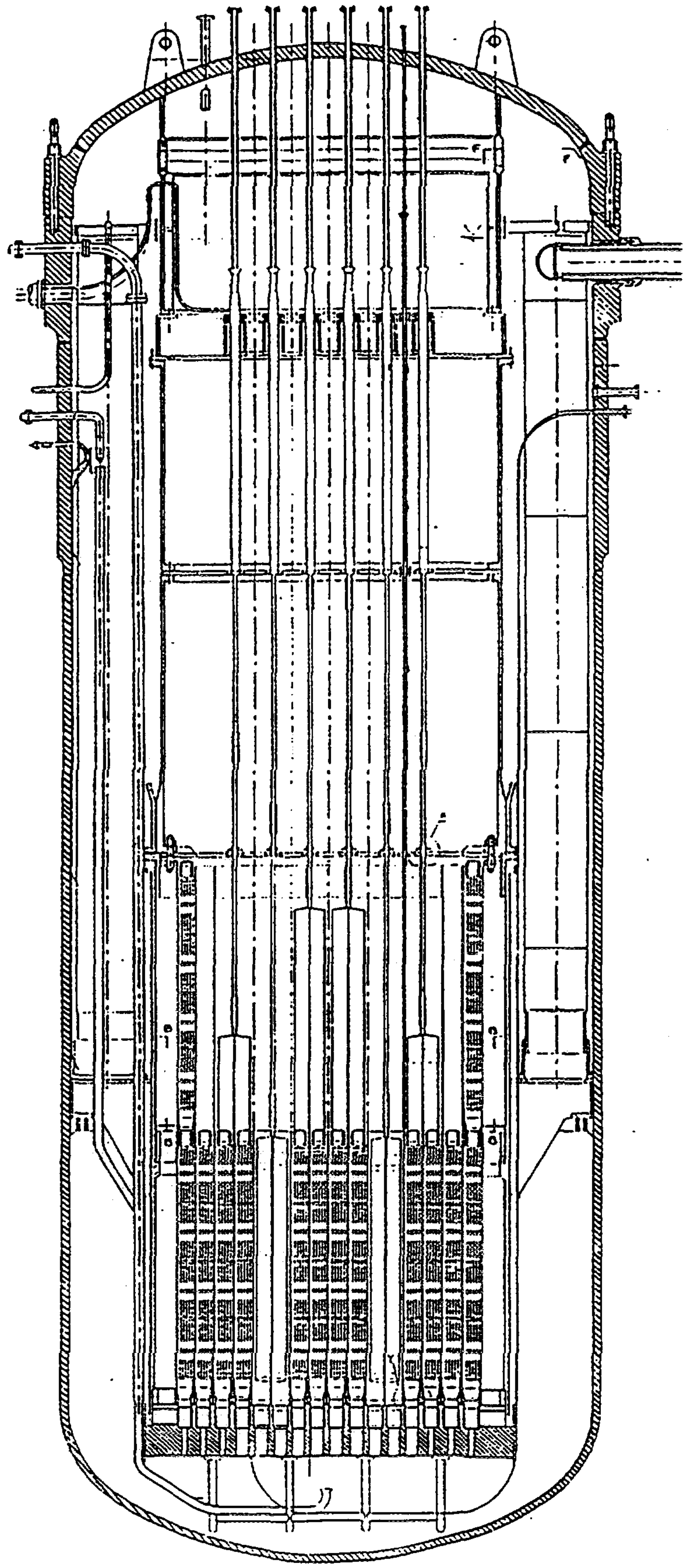

Fig.1 Longitudinal section of NHR-200

dose-equivalent accepted by any individual should be lower than 5 mSv, and the thyroid gland dose-equivalent should be lower than 50 mSv.

- The reactor core is not to melt in limiting beyond design accidents; the effective radiation dose-equivalent received by any individual should be lower than 5 mSv and the thyroid gland dose-equivalent should be lower than 50 mSv.

Design criteria used in the design:

- Departure from nucleate boiling ratio (DNBR)

  The advanced Barnett formula is used to analyze DNBR. The minimum DNBR (MDNBR) should be larger than 1.35 in normal operation, anticipated operating occurrences and infrequent faults.

  According to the analysis of DNB experiments of twenty-rod bundles and simulation rod bundle experiments for the NHR-5, satisfying a 95% confidence and a 95% probability level, the limit value of MDNBR for the 200 MW NHR should be 1.15. Considering fuel rod bending and a suitable DNB margin, the final DNBR limit value of 1.35 is suitable.

- Fuel temperature:

  In operation (see paper "Safety Objectives and Design Criteria for the NHR-200") Categories I, II and III, the maximum temperature of the fuel pellets should not be larger than 2590°C.

- Hydraulic instability:

  In operating Categories I, II and III, there should be no hydraulic instability in the 200-MW NHR primary circuit.

- Core covered with water:

  The 200-MW NHR is operated under lower pressure; the surface temperature of the fuel rods is not a limiting factor as long as the core is covered with water in any condition.

- Other limits:

  There are still other limits associated with other systems or components that should be satisfied.

3. THERMAL DESIGN EXPLANATION

3.1. Computer codes

The RETRAN-02, COBRAIIIC/MIT-2 and the STEADY-LTHR codes are used in the thermal hydraulic design of the 200-MW NHR.

The RETRAN-02 code [2] is used to analyze steady and dynamic processes of 200 MW NHR systems. It can provide the distribution of mass flow, pressures and temperatures, etc., in the primary circuit.

The COBRAIIIC/MIT-2 code [4] is used for the detailed analysis of subchannels and rod bundles in the core. It can provide detailed distributions of mass flow, temperatures, pressures, and DNBR in subchannels and rods, respectively.

The STEADY-LTHR code [5] is used for the analysis of the steady parameters in the primary and the intermediate loop, and in the thermal network. It can also be used to analyze the mass flow distribution in side-by-side channels.

3.2. DNBR analysis

The AD-Barnett formula is used for the DNBR analysis in the thermal-hydraulic design of the 200-MW NHR. The DNB formula is concluded on the basis of DNB experiments of rod bundles [3]. Its suitable range and the design parameters of the 200-MW NHR are shown in Table 1.

TABLE 1: EXPERIMENT DATA SCOPE AND 200 MW NHR PARAMETER

Parameter	Unit	Data scope	200-MW parameter
Rod diameter	mm	10-14.3	10
Heating section length	mm	836-4440	1900
Pressure	MPa	1.03-5.0	2.5
Mass flow flux	kg/s-m^2	34-2288	494
Inlet subcooling	kJ/kg	13.9-798	350

The calculated results of the AD-Barnett formula are compared with 743 experimental data in Fig. 2. 95% of the data are inside a band of 15% difference, the spread of all data differs not more than 20%. The Root-Mean-Square (RMS) is 6.01%. The simulating experiments concerning DNB experiment for NHR-5 were conducted at the Nuclear Power Institute of China.

Experiment parameters:

Heating rod diameter:	10 mm
Rod lattice pitch:	13.3 mm
Rod cluster:	square 3x3 or 4x4
Heating section length:	800 mm
System pressure:	1.2-1.8 MPa
Mass flux:	500-1400 kg/s-m2
Critical quality:	-0.05 $\pm$ 0.05

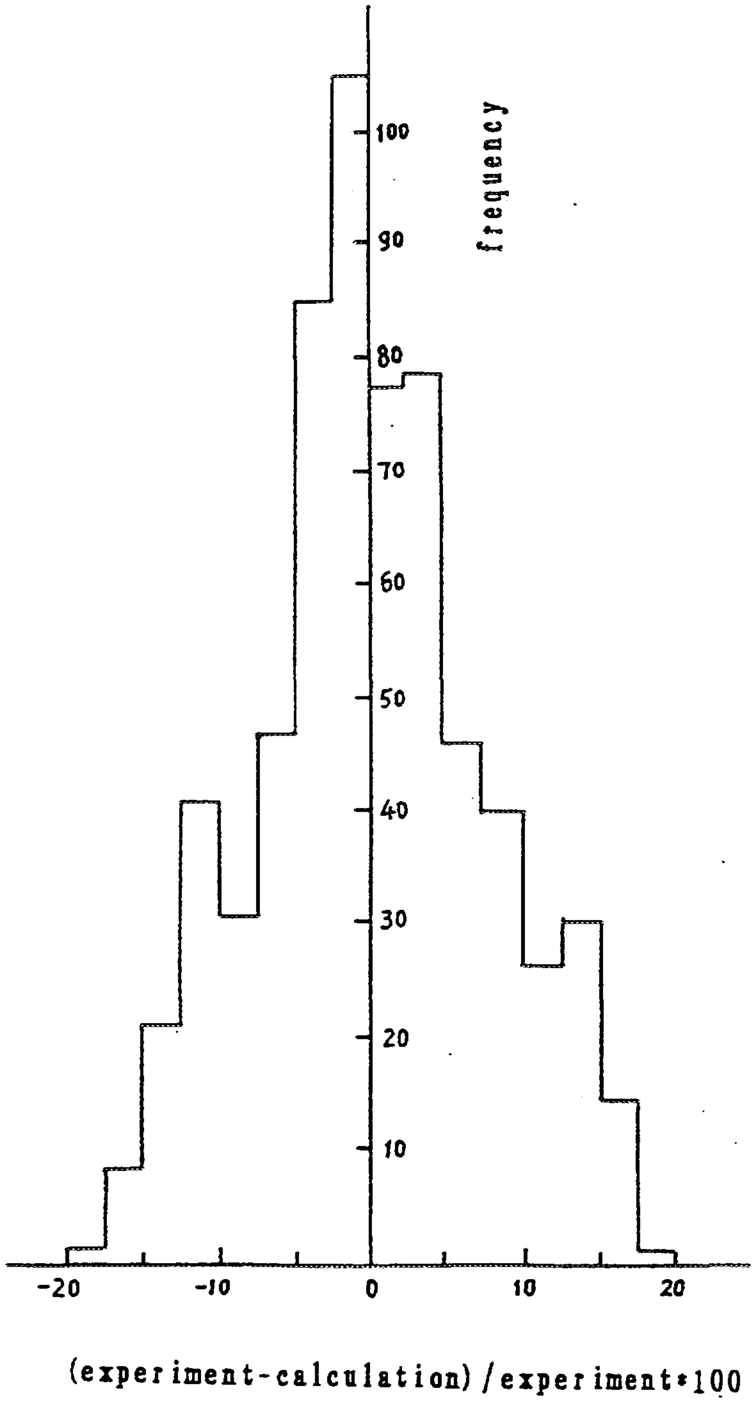

Fig.2 comparison between AD-Barnett and 743 experiments

The results calculated with the AD-Barnett formula are compared with 98 experimental data. The comparison of analysis and experiments shows for:

55.1% of the data	a relative difference < 5%
82.7% of the data	a relative difference < 10%
95.9% of the data	a relative difference < 15%
100% of the data	relative difference < 20%

The result is the same as for the 743 experimental data [3].

3.3. Nuclear heat flux factor and engineering hot spot factor

The axial nuclear heat flux distribution at the beginning of life (BOL) of the 200-MW NHR is shown in Fig. 3. The radial power distribution of the rod clusters in the core is shown in Fig. 4. The maximum local nuclear heat flux factor for the fuel rods in rod bundles is 1.32. At BOL the overall hot spot factor is 4.39. The overall hot spot factor at mid-life and end-life will be lower than that at BOL.

The effects of the difference between real parameters and the rated values such as the fuel pellet diameter, pellet density, fuel enrichment, and the heat flux will be included by the Engineering Hot Spot Factor (EHSF). The EHSF is 1.04 in the 200-MW NHR. Considering other effects, a total EHSF of 1.10 is used in the 200-MW NHR design.

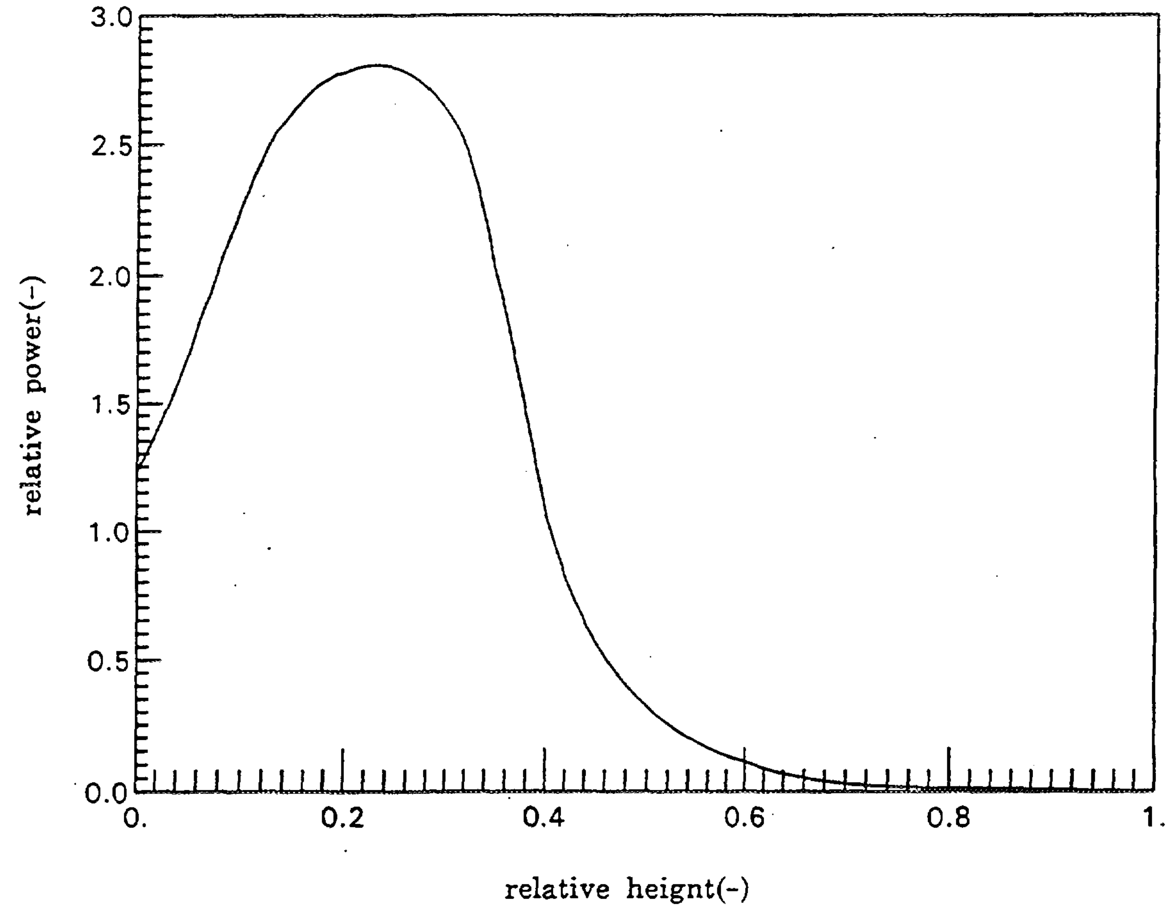

Fig.3 Axial power distribution at BOL

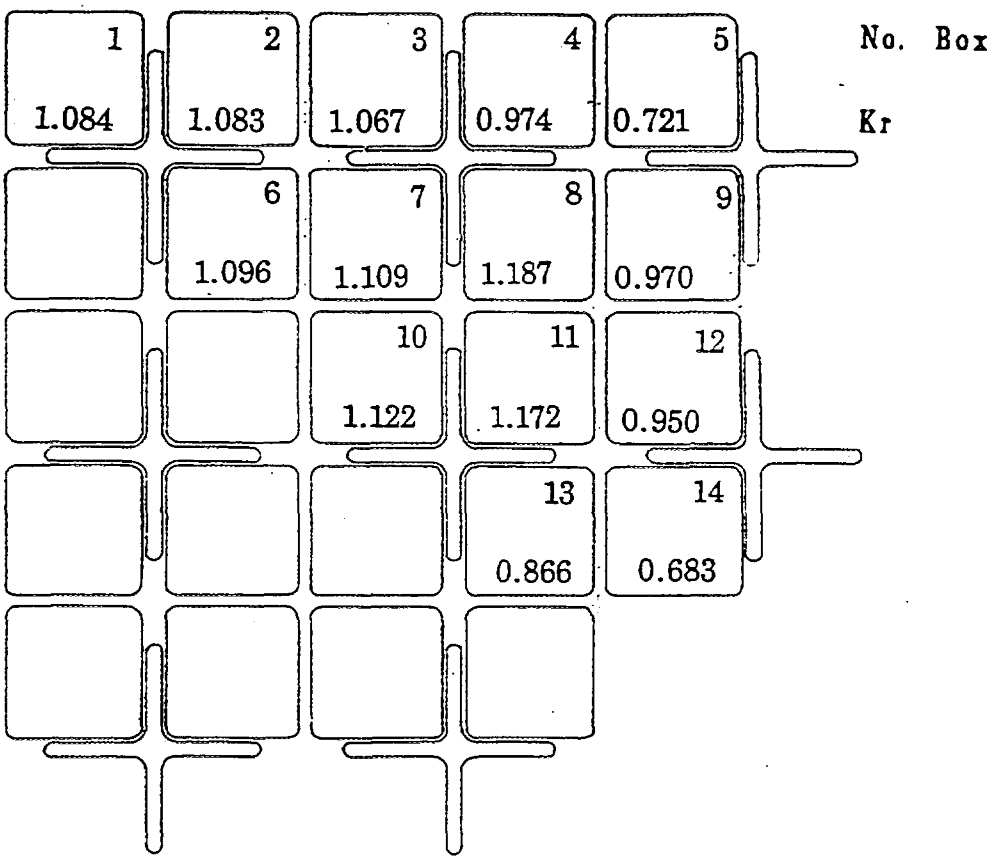

Fig.4 Radial power distribution of rod bundles at BOL

3.4. Temperature analysis in fuel rod

The following UO_2 thermal conductivity formula is used in calculating the temperature of fuel rods.

$$K=38.24/(t+402.55)+4.788e^{-13}(t+273.15)^3$$

with:

t: fuel temperature, °C,
K: thermal conductivity, W/cm-°C

It is assumed that the equivalent heat transfer coefficient between fuel pellet and cladding is 5678 W/m²-°C.

3.5. Thermal parameters of the 200-MW NHR

Some thermal parameters of the 200-MW NHR at BOL and at rated operating condition are shown in Table 2.

TABLE 2: SOME THERMAL PARAMETERS

Parameter	Unit	NHR-5	200-MW NHR
Reactor power	MW	5	200
Primary system pressure	MPa	1.47	2.5
N_2-pressure	MPa	0.32	0.59
Total mass flow	t/h	106.6	2527
Effective core flow	t/h	100	2376
Bypass/total flow	--	0.06	0.06
Average velocity in core	m/s	0.24	0.57
Average mass flux in core	kg/s-m^2	202	494
Core inlet temperature	°C	147	145
Riser outlet temperature	°C	186	210
Average thermal flux	w/cm^2	17.8	24.6
Maximum thermal flux	w/cm^2	78.1	107.9
Average liner power density	w/cm	56.1	77.2
Maximum liner power density	w/cm	246	338.7
Core power density	kW/l	24.0	36.2
MDNBR	--	2.39*	2.65
Maximum fuel temperature	°C	1125	1366
DNB formula	--	AD-Barnett	AD-Barnett

* Including a safety factor of 1.3.

3.6. Comparison with NHR-5

The NHR-5 has been successfully operated for 5 heating seasons since 1989. A set of experiments associated with operation and safety were conducted. The results of the experiments show that the NHR-5 has a good inherent safety. The NHR-5 design was very successful.

The design and operational experience of the NHR-5 are fully taken into account in the design of the 200-MW NHR. A comparison of some important parameters between the NHR-5 and the 200-MW NHR is shown in Table 2.

After a full consideration of safety issues, higher primary loop pressure, higher core heat flux, higher core power density than NHR-5 were selected for the design of the 200-MW NHR. This is helpful to advance the economics of the 200-MW NHR.

3.7. Hydraulic instability

In some conditions a hydraulic instability may take place in the steam-water two-phase flow under low pressure in a natural circulation loop. This type of instability normally is a density wave oscillation or dynamic instability. To avoid this type of instability, nitrogen gas is filled in the upper dome of the reactor vessel. Under rated operating conditions, the total system pressure in the primary loop is 2.5 MPa. The nitrogen partial pressure is 0.59 MPa and the steam partial pressure is 1.91 MPa. The riser outlet temperature is 210°C. The saturation temperature is 224°C at 2.5 MPa pressure. The subcooling temperature of 14°C between riser outlet temperature and saturation temperature is maintained during operation by regulating the intermediate loop parameters. The subcooling temperature of 14°C at riser outlet will make the following possible:

- there is no boiling in the core at rated operating conditions, and at anticipated operating deviations,
- the 200-MW NHR will not go into an instability region of steam water two phase flow in accident conditions.

3.8. Flow distribution in the core

Rod clusters within a fuel box are used in the 200-MW NHR. Through flow distributors in the fuel clusters, a limited natural circulation flow in the primary loop is effectively used. The total mass flow is 702 kg/s in the primary loop at rated operating conditions. The mass flow through the gap between cluster boxes and other bypass gaps is considered ineffective. It is about 6% of the total flow. Total effective flow through the fuel boxes is 660 kg/s.

According to the radial power distribution an appropriate flow distribution is achieved by installing throttles at the inlet of the cluster boxes. By distributing flow, the clusters with larger heating power have larger mass flow. The difference of radial power distribution at BOL, MOL and EOL are considered in flow adjustment to get the best use of flow in any period of the life.

The mass flow distribution at BOL is shown in Fig. 5. The outlet water temperature distribution at BOL, MOL and EOL are shown in Fig. 6.

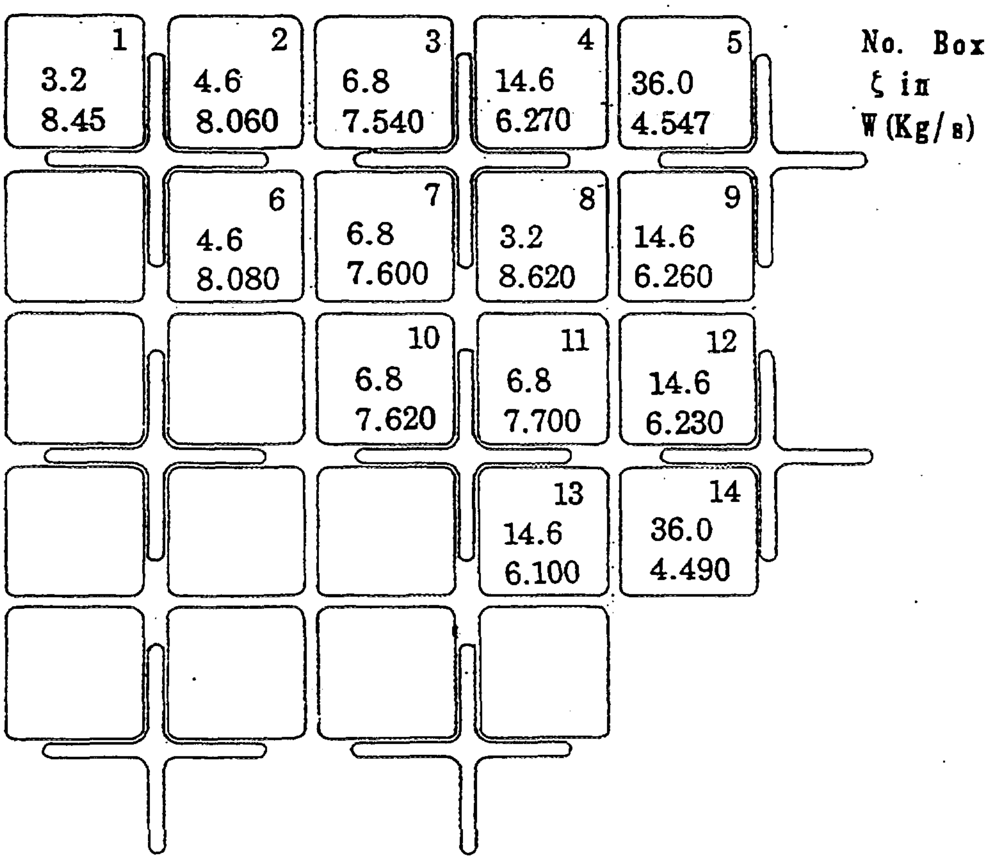

Fig.5 Core flow distribution at BOL

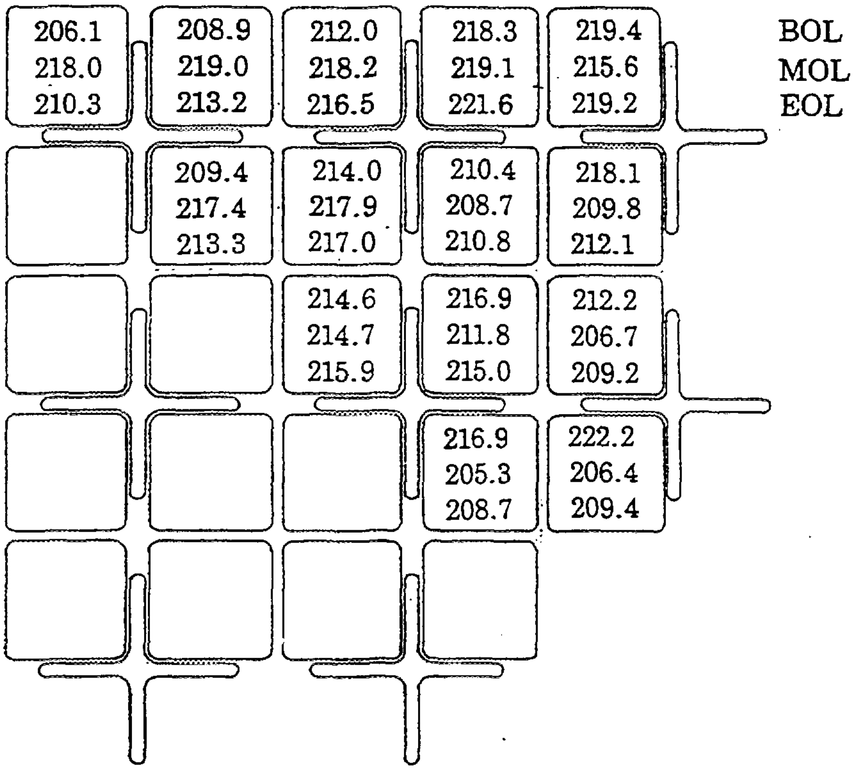

Fig.6 Outlet temperature distribution at BOL, MOL and EOL

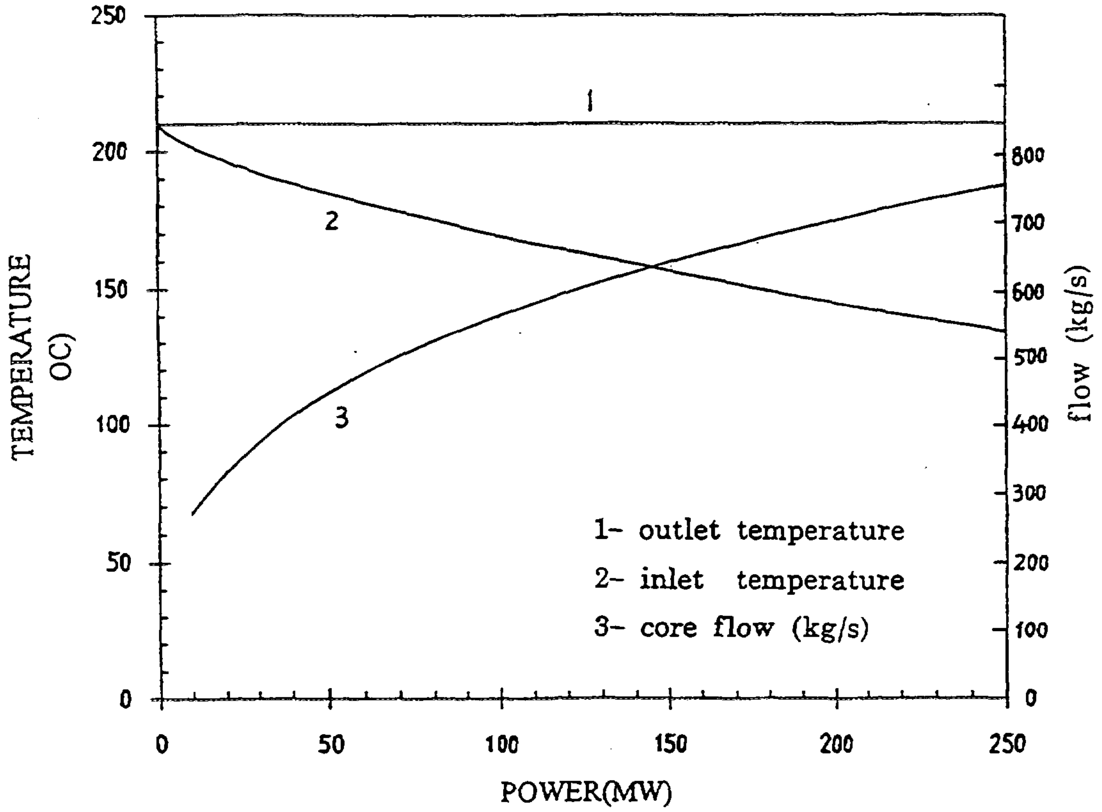

Fig.7 Natural circulation character:
flow and inlet temperature chage with power

3.9. Natural circulation character of primary loop

Natural circulation operation is used in the integrated primary loop of the 200-MW NHR. The flow circulation is somewhat different from PWRs in which the flow circulation is driven by pumps in the primary loop. The natural circulation flow in the primary loop, and the core inlet temperature in the 200-MW NHR will change with different operating power. By maintaining constant outlet temperature of the riser and constant pressure in the upper dome of the reactor vessel, the natural circulation flow increases and core inlet temperature decreases with reactor power rising. That is shown in Fig. 7.

3.10. The thermal response in accidents

The most dangerous accident for the 200-MW NHR regarding MDNBR and fuel maximum temperature is an unexpected control rod drop. When the induced negative reactivity by dropping one control rod is unable to scram the 200-MW NHR, assuming that the protection of the fast decrease of the neutron flux is not available, and the intermediate circuit and the thermal network circuit continue in normal operation, the accident will cause the moderator temperature in the core and the fuel temperature to decrease. The induced positive reactivity feedback by the decrease of moderator and fuel temperature will be compensated the induced negative reactivity by dropping the control rods. Finally the reactor power will increase and return to about the initial power level. The change of the reactor power by dropping a 0.5$ control rod under the described severe assumptions is shown in Fig. 8.

Because of the drop of a control rod into the core, a radial power distortion is induced. The neutron flux in 1/8 quadrant that is symmetric to the quadrant of the dropped control rod will rise. The MDNBR will decrease and the maximum fuel temperature will increase.

The radial power worsening coefficient for fuel elements is defined as the ratio of the radial power factor of a cluster after dropping a control rod to that before dropping:

$$K_{rw}=K_r'(\text{after dropping}) \,/\, K_r\,(\text{before dropping})$$

Kr' is the radial power factor of a cluster after dropping. Kr is the radial power factor of a cluster before dropping.

The results of a two-dimension neutron physics analysis with the SIMULATE code show that the maximum radial power worsening coefficient is 1.34 when dropping one control rod into core. The location of the maximum radial power worsening coefficient is the location of the maximum radial power factor (1.187) before dropping the rod. The subchannel analysis results show that the MDNBR decreases from 2.65 at normal operating condition to about 2.0 after dropping a control rod. The maximum fuel temperature increases from about 1360°C increasing to about 2000°C.

4. CONCLUSIONS

The analysis presented in above sections show that the design criteria for the 200-MW NHR are satisfied. Although the power density is increased in the 200-MW NHR over that in the NHR-5, safety is still maintained under normal operating and under accidents conditions.

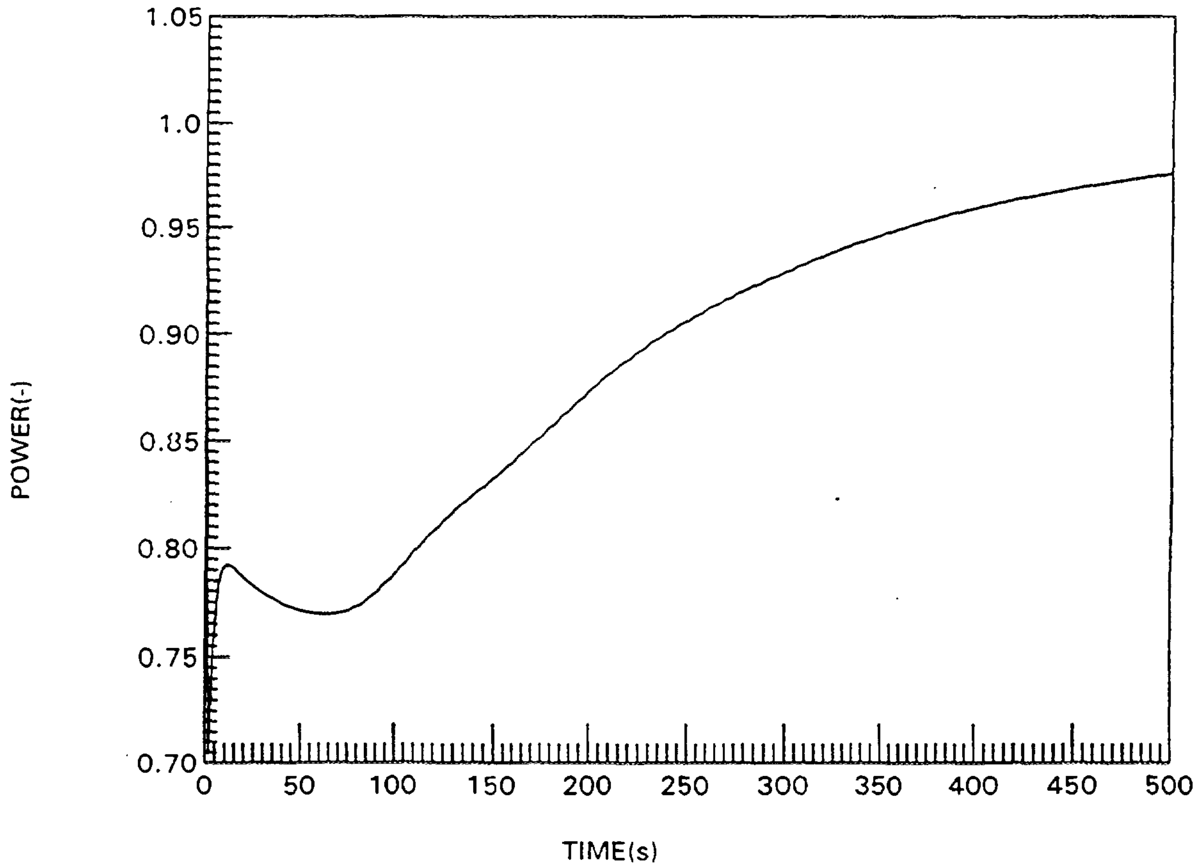

Fig.8 0.5$ control rod drop accident:
change of reactor power against time.

REFERENCES

[1] The design criteria for 200 MW nuclear heating station, Institute of Nuclear Energy Technology of Tsinghua University, 1994.

[2] RETRAN-02: A Program For Transient Thermal Hydraulic Analysis of Complex Fluid, Flow Systems, NP-1859-CCM, vol. 1 equations and numerics, 1981.

[3] D.Hughes, A correlation of rod bundle critical heat flux for water in the pressure range 150-725 Pisa, IN-1412, July 1970, IDAHO Nuclear Corporation.

[4] J.W.Jackson,N.E.Todreas, COBRAIIIC/MIT-2: A Digital Computer Program For Stead And transient Thermal-Hydraulic Analysis Of Rod Bundle Nuclear Fuel Elements, PB 82 180233.

[5] Gao Zuying etc., STEADY-LTHR: A Stead Analysis Program For Low Temperature Heating Reactor, Third Topical Meeting On Reactor Thermal Hydraulics, Shanghai, Sept. 1988.