METHODOLOGY USED TO CALCULATE MODERATOR-SYSTEM HEAT LOAD AT FULL POWER AND DURING REACTOR TRANSIENTS IN CANDU REACTORS

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ABSTRACT

Nine components determine the moderator-system heat load during full-power operation and during a reactor power transient in a CANDU® reactor. The components that contribute to the total moderator-system heat load at any time consist of the heat generated in the calandria tubes, guide tubes and reactivity mechanisms, moderator and reflector; the heat transferred from calandria shell, the inner tubesheets and the fuel channels; and the heat gained from moderator pumps and heat lost from piping.

The contributions from each of these components will vary with time during a reactor transient. The sources of heat that arise from the deposition of nuclear energy can be divided into two categories, viz., a) the neutronic component (which is directly proportional to neutronic power), which includes neutron energy absorption, prompt-fission gamma absorption and capture gamma absorption; and b) the fission-product decay-gamma component, which also varies with time after initiation of the transient.

An equation was derived to calculate transient heat loads to the moderator. The equation includes two independent variables that are the neutronic power and fission-product decay-gamma power fractions during the transient and a constant term that represents the heat gained from moderator pumps and heat lost from piping.

The calculated heat load in the moderator during steady-state full-power operation for a CANDU 6 reactor was compared with available measurements from the Point Lepreau, Wolsong 1 and Gentilly-2 nuclear generating stations. The calculated and measured values were in reasonably good agreement.

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1. INTRODUCTION

Nuclear radiation absorbed in the reactor components generates heat, which is removed by the heat-transport system (HTS), and the moderator and shield-cooling systems. These components include the following:

- the lattice-cell components (fuel, fuel sheath, coolant, pressure tube, calandria tube and moderator);
- other in-core components (reactivity mechanisms, mainly adjuster rods and their guide tubes); and
- materials and structures outside the reactor core (reflector, calandria shell, vault water, primary end shields, primary side shields, etc.).

The different components in the reactor system are shown in Figure 1. In this paper, a lattice-cell calculation for a 37-element fuel bundle and bulk shield calculations for the reactor structures were combined to evaluate the moderator-system heat load for a CANDU 6 station during reactor operation and during various reactor transients.

Five transient conditions are considered in this paper (without any reference to the associated service conditions for which a certain reactor power transient scenario may be applicable):

a) a SDS1 trip transient associated with a short shutdown,

b) a startup following a short trip,

c) a startup following a shutdown,

d) a setback transient, and

e) a stepback transient

2. MODERATOR-SYSTEM HEAT LOAD MECHANISM

Nine components determine the moderator-system heat load during full-power operation and during a reactor power transient in a CANDU reactor. The components that contribute to the total moderator-system heat load at any time consist of the following:

i) heat generated in the calandria tubes, q_1

ii) heat generated in the guide tubes and reactivity mechanisms, q_2

iii) heat generated in the moderator, q_3

iv) heat generated in the reflector, q_4

v) heat transferred from the calandria shell to the moderator, q_5

vi) heat transferred from the inner tubesheets to the moderator, q_6

vii) heat transferred across the gas annulus from fuel channels, q_7

viii) heat loss from piping, q_8

ix) heat gained from moderator pump, q_9

The schematic diagram of the heat-load components for the moderator system is shown in Figure 2.

3. SOURCES OF RADIATION IN THE NUCLEAR HEAT DEPOSITION PROCESS

The nuclear heat deposition rates in items (i) to (vi) have essentially two components. The first component (neutronic heating) is directly proportional to neutronic power, which arises from neutron energy absorption, prompt-fission and capture-gamma energy absorption. The second component is due to fission-product decay-gamma energy absorption.

Note that the energy deposited in the moderator system as a result of activation gammas in the calandria shell and in the calandria side tubesheets (^{56}Mn, ^{60}Co decay gammas) was evaluated based on 1.5 wt % manganese and 1000 ppm cobalt impurity levels in the steel material, which is irradiated to saturation. The contribution to the total heating was taken into consideration, together with the fission-product decay-gamma heating.

In general, the total heating in these components during full-power operation includes both the neutronic and fission-product decay gamma (plus activation gamma) heating. During reactor transients, the contribution from each of these components varies with time. The neutronic component, which is directly proportional to the neutronic power, follows the power rundown transient curve; the fission-product decay-gamma component varies with time following the initiation of a transient from full-power operating conditions.

To calculate the total heating as a function of time, each component needs to be isolated so that it can be followed as a function of time using its rundown and decay characteristics. The following sections describe the methodology used to calculate the neutronic and the fission-product decay-gamma related heating values.

3.1 Lattice Cell Calculation

The one-dimensional discrete ordinates transport code ANISN[1] was used to perform the lattice-cell calculation. The fuel bundle was assumed to have an average exit burnup of 180 MW.h/kg(U), for which $\nu = 2.62$ neutrons/fission. These calculations show that the fractions of gamma and neutron energy deposited in the calandria tube are 0.018 and 0.004, respectively[2]. In a typical CANDU 6 reactor, the total gamma energy emitted per fission is 22.45 MeV, and the neutron energy emitted per fission is 5.28 MeV. The former consists of gamma energy releases from prompt gammas (7.26 MeV/fission), fission-product decay gammas (5.99 MeV/fission) and capture gammas (~9.2 MeV/fission), a total of 22.45 MeV/fission.

3.1.1 Calandria Tubes

The lattice-cell calculation shows that the calandria-tube full-power heating becomes 4.34 MW, viz., [0.018 x (N_k x 22.45 - 6.16) from gammas] + [0.004 x (N_k x 5.28 - 0.70) from neutrons]. The gamma heating is 4.12 MW, and the neutron heating is 0.22 MW. The N_k is equal to 10.4657 MW.fission/MeV and was based on the nuclear energy deposition of 2061.4 MW(th) in the fuel channels in a CANDU 6 reactor. The 6.16 MW is the sum of the total gamma energy escaping from the fuelled core and the gamma energy deposited in other in-core components, including adjuster rods; the 0.7 MW is the corresponding value for neutrons.

The next step is to evaluate the fission-product decay-gamma contribution to the total heating. The fractions of fission-product decay-gamma energy deposited in the fuel-channel components were calculated in Reference 3 using ANISN. The energy grouping was based on the isotope generation and depletion code ORIGEN[4]. The gamma source was divided into 12 energy groups, which correspond to the 12 energy groups used in ORIGEN for the fission products; the lowest energy group is divided into seven subgroups, corresponding to the ORIGEN grouping for the actinides.

The fractions of the total gamma energy absorbed in the calandria tubes in each energy group were multiplied by the gamma power at time zero from an ORIGEN calculation of an average power bundle (473 kW irradiated to an exit burnup of 180 MW.h/kg(U)), and summed over all energy groups to obtain decay power in watts in a calandria tube associated with one fuel bundle. The sum is about 225 W (for one fuel bundle), which corresponds to 0.096 MeV/fission. This value indicates that 22.6% (i.e., 0.096/0.425) of the total heat load in the calandria tube is associated with fission-product decay gammas and 77.4% is associated with neutronic power; 0.425 MeV/fission is the full-power fraction in the calandria tube, viz., 0.018 x 22.45 + 0.004 x 5.28.

3.1.2 Reactivity Mechanisms and Guide Tubes

This is the heat deposited in the guide tubes of the reactivity mechanisms, including the stainless-steel adjuster rods. The total zirconium mass in the guide tubes is about 1.4 Mg. The nuclear energy deposition in the guide tubes was calculated using the assumption, that the nuclear energy deposition per unit mass of zirconium is the same as that per unit mass of the calandria tubes. The total mass of the calandria tubes in a CANDU 6 reactor is about 8.3 Mg, and the energy deposited in the calandria tubes is 4.34 MW. Thus the corresponding heat deposition in the guide tubes is about 0.7 MW (i.e., 4.4 x 1.4/8.3).

An allowance of 2.0 MW was made for the nuclear energy deposited within the adjusters and in the moderator because of energy released from the adjusters. The total heat deposition was therefore 2.7 MW. The neutronic and fission-product decay gamma contributions were assumed to be the same as those for the calandria tubes, i.e., 77.4% and 22.6% respectively.

3.1.3 Moderator

Following the same methodology given above and for fractions of the gamma and neutron energy deposited in the moderator, 0.185 and 0.666 respectively, the moderator full-power heating is equal to 78.6 MW, viz., [0.185 x (N_k x 22.45 - 6.16) from gammas] + [0.666 x (N_k x 5.28 - 0.70) from neutrons]. The gamma heating is 42.3 MW, and the neutron heating is 36.3 MW.

Based on an average bundle power of 473 kW at an exit burnup of 180 MW.h/kg(U), the product of the fractional fission-product decay gamma heat distribution in the moderator and the gamma powers at time zero in each gamma energy group calculated by ORIGEN, a total of 1970 W of energy is absorbed in the moderator. This corresponds to 0.84 MeV/fission. Because the full-power fraction is 7.5 MeV/fission, i.e., 0.185 x 22.45 + 0.633 x 5.28, the fission-product decay-gamma contribution is about 11% (0.84/7.5); the neutronic contribution is therefore 89%.

3.2 Bulk Shield Calculations

3.2.1 Reflector

In a CANDU 6 reactor, the D_2O moderator contained outside the average fuelled-core radius (which is 3.14 m) and inside the calandria main shell and subshells is called the reflector. The total amount of nuclear energy deposited in the reflector region was calculated by a two-dimensional discrete ordinates transport code DOT 4.2[5]. The model used in that analysis is shown in Figure 3. The neutronic and fission-product decay-gamma contribution to the reflector heat load was assumed to be the same as that of the moderator associated with the fuel channels. The total amount of heat deposited in the reflector was calculated to be 5.9 MW.

3.2.2 Calandria Shell

The calandria shell consists of the main shell, two annular plates and two subshells. The total amount of heat deposited in these components during full-power operation was calculated by the DOT 4.2 code. The total heat load was calculated to be about 2 MW. This heat is removed by the reflector on the inside and by the vault water on the outside. The former ends up in the moderator system and the latter in the shield cooling system.

From an ANISN calculation, the fission-product decay-gamma component of the heat deposited in the calandria shell was estimated to be less than 1% during reactor operation at full-power conditions. The activation-gamma heating (^{56}Mn and ^{60}Co decay gammas) was calculated from a DOT 3.5 code (predecessor of the DOT 4.2 code). It amounts to a total of 2% of the 2 MW heat load calculated above; the moderator portion is only 0.5%, i.e., 10 kW. Consequently, it was not included in the 2 MW heat load. However, this was taken into account during the transients as a component within the fission-product decay-gamma heating. This contribution was taken conservatively to be 5% of the total heating in the calandria. Note that two thirds of the activation gamma heating is due to ^{56}Mn radiation, which decays rapidly with a half-life of 2.5 h. The remaining one third of the activation heating is due to ^{60}Co radiation.

The calandria shell is cooled by the vault water on the outside and the moderator on the inside. The amount of heat transferred to the moderator was estimated using Reference 6 for heat transmission in slabs, with exponential heat source and known surface temperature on both sides. The peak temperature occurs at χ = 4 mm into the calandria shell. The amount of heat removed on the moderator side of the calandria shell can be obtained by integration, which is 0.42 W/cm^2.

The total amount of heat removed by the moderator was estimated to be 0.38 MW, i.e., [0.42 W/cm^2 x 2πRH x FF x 10^{-6} W/MW], where 2πRH is the calandria shell surface area, R = 3.79 m, H = 5.94 m, and FF is the axial form factor at the core edge, 0.635. The factor FF was calculated using bundle power distribution for peripheral channels calculated by RFSP[7].

3.2.3 Calandria Side Tubesheets (Inner Tubesheets)

The nuclear heat deposited in the calandria side tubesheets was calculated by ANISN and was about 1.9 MW for both tubesheets. Following the methodology given earlier for the calandria shell, the amount of heat removed by the moderator was calculated to be about 1.4 MW.

From an ANISN calculation, the fission-product decay gamma component of the total heating was estimated to be ~8%. The activation gamma component was ~2% at time zero. Adding the latter component to the former, the fission-product decay gamma heating was taken to be 10%.

The calandria-side tubesheet is cooled by the end-shield cooling system on the outside and by the moderator on the inside. The distance χ at which the peak temperatures occurs in the slab was calculated to be at 28 mm into the calandria side tubesheet.

The heat removed on the moderator side of the calandria-side tubesheet was found to be 3.22 W/cm^2. This heat current was multiplied by the area of the end shields (2 x 28.575 x 380) and by the radial form factor at the core end, 0.70, and the total heat removed by the moderator was calculated to be 1.40 MW (2 tubesheets).

3.3 Estimated Heating from Fuel Channels, Piping Loses and Pump Energy Gains

At full power, the amount of heat transferred from the fuel channels across the gas gap between the calandria tube and pressure tube was taken to be about 3 MW. It was assumed to be constant during short transients following steady-state operation at full power. For transients associated with longer shutdown and following startups, it was assumed to be proportional to the calandria-tube heating, i.e., 3 MW at 100% full power with 77.4% contribution from neutronic power, 22.6% contribution from fission-product decay gammas.

The piping losses, 0.3 MW, and the pump energy appearing in the moderator, 0.7 MW, result in a net energy transfer of 0.4 MW to the moderator during full-power reactor operation.

4. TRANSIENT HEATING CALCULATIONS

Table 1 shows a typical moderator system heat load for a CANDU 6 station during steady-state operation at 100% full power. Using the data given in the previous section and Table 1, a transient heating expression can be formulated as follows.

If there is a power change, then the contribution from each of these components will vary with time and can be calculated by modelling the source of heat and its behaviour during a transient. The sources of heat that arise from the deposition of nuclear energy can be divided into two categories, a) the neutronic component (which is directly proportional to the reactor power), which includes neutron energy absorption, prompt-fission gamma absorption and capture gamma absorption; and b) the fission-product decay gamma component, which varies with time after the initiation of the transient.

Using a lattice cell and bulk shield calculations performed by ANISN[1], DOT-4.2[5], and the ORIGEN[4] codes, the neutronic and fission-product decay gamma components of the total heat deposition rates in items (i) to (vi) listed in Section 2 were obtained during reactor operation. The total heating in these components at time "t" after a reactor transient can be expressed as follows:

$$q_1(t) = Q_1[f_1P(t) + p_1FP(t)]$$

$$q_2(t) = Q_2[f_2P(t) + p_2FP(t)]$$

$$q_3(t) = Q_3[f_3P(t) + p_3FP(t)]$$

$$q_4(t) = Q_4[f_4P(t) + p_4FP(t)]$$

$$q_5(t) = Q_5[f_5P(t) + p_5FP(t)]$$

$$q_6(t) = Q_6[f_6P(t) + p_6FP(t)]$$

$$q_7(t) = Q_7$$

$$q_8(t) = Q_8$$

$q_9(t)$ = Q_9

where Q_i is the heating in a given component "i = 1 to 9" at 100% full power (FP).

P(t) is the neutronic power fraction at time "t".

FP(t) is the ratio of the fission-product decay power at time "t" to that of the power at time zero at 100% FP operation. [This value was obtained from ORIGEN and ANISN calculations, as documented in Reference 4.]

f_i,p_i are the neutronic and fission-product decay gamma power fractions of heat deposition rates in that component.

Summing up those heating components q_1, q_2,......,q_9 and normalizing it to Q(0) gives the moderator heat load q(t) at time "t" as shown below.

$$q(t) = Q(0)\left\{\sum_{i=1}^{i=6}[f_iP(t) + p_iFP(t)]\ Q_i / Q(0) + [Q_7 + Q_8 + Q_9] / Q(0)\right\}$$

where Q(0) is the total moderator heat load at 100% FP.

The above expression can be reduced to the following after substituting the terms inside the brackets.

q(t) = Q(0) [fP(t) + pFP(t) + constant]

where f, p are the overall neutronic and fission-product decay gamma power fractions of the moderator heat load and the "constant" is the ratio $(Q_7+Q_8+Q_9)/Q(0)$.

Note that the expression for q(t) assumes that the heat transferred from the fuel channels (q_7), the heat loss from moderator piping and heat gained from moderator pumps (q_8 and q_9) remain constant during a transient. These heat loads may need modification for some transients, e.g., for startups following a long shutdown.

Parameters given in these expressions were evaluated in Section 1; thus $q_i(t)$ can be expressed explicitly as

q_1 = 4.3 [0.773 P(t) + 0.227 FP(t)]

q_2 = 2.7 [0.773 P(t) + 0.227 FP(t)]

q_3 = 78.6 [0.89 P(t) + 0.11 FP(t)]

q_4 = 5.9 [0.89 P(t) + 0.11 FP(t)]

q_5 = 0.4 [0.95 P(t) + 0.05 FP(t)]

q_6 = 1.4 [0.9 P(t) + 0.10 FP(t)]

q_7 = Q_7 = 3.0 MW(th)*

q_8 = Q_8 = -0.3 MW(th)

q_9 = Q_9 = 0.7 MW(th)

Summing up the heating components, the following expression is obtained:

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* Use $q_7(t)$ = 3.0 [0.773 P(t) + 0.227 FP(t)] as a first approximation for transients involving long shutdowns, i.e., days, weeks and for subsequent startups.

$$q(t) = 82.24\,P(t) + 11.04\,FP(t) + 3.40$$

Dividing both sides by the total heat load Q(0) = 96.7 MW(th) in the moderator system, an expression for the transient heating can be evaluated as

$$q(t) = Q(0)\,[0.851\,P(t) + 0.114\,FP(t) + 0.035] \qquad [1]$$

5. APPLICATION OF THE METHODOLOGY TO VARIOUS REACTOR TRANSIENTS

Reactor transients associated with service conditions in terms of temperatures and pressures in the calandria and in the moderator are generally defined in the reactor structure assembly documents. Five different transients were considered in the paper.

During each transient, the fission-product decay gamma heating at time "t" that is due to previous operation at power higher than the power at time "t" is excluded in the above expression. Generally, this component is about zero, particularly if there is an abrupt reduction in power level after extended operation at steady-state full power, e.g., SDS-1 trip or a stepback transient. In other words, the neutronic power drops to zero instantaneously at time zero. However, for a setback or any other transients, during which the power level is reduced over a period of time (minutes), the additional heating resulting from the generation of fission products during the power reduction should be taken into consideration.

5.1 Transient Conditions

The transient conditions for the five different reactor transients are described below:

i) SDS1 Trip Transient

A power rundown transient associated with a short shutdown with 2 out of 28 shutoff rods (SORs) missing was considered. The dynamic reactivity worth of -75.5 mk was used and the SORs were assumed to drop in 1.54 s. Equilibrium fuel conditions were assumed at the start of the transient and the zone controllers were 50% full on average. The neutronic power transient (up to 1.54 s) was taken from a CERBERUS[7] analysis.

After all the shut-off rods were dropped, residual neutronic power of the core follows the decay of the neutron source in the core. The decay in neutron source will follow the decay of the delayed neutron precursor fission products and the fission-product decay gammas that generate photoneutrons. The shape of the neutronic power curve give in Figure 4 was derived from a point kinetics code. The reactor power transient was simulated using a point-reactor model. The point-kinetics equations were solved by giving the pre-defined system transient reactivity and delayed-neutron data (6 delayed neutrons and 9 photoneutron groups were used). The transient was extended to 8.64×10^3 s (1 day).

ii) Startup Following a Short Trip

The startup was taken to be initiated following a short trip. The neutronic power at the start of the transient (assumed to be about 1500 s following the trip) was taken to be 2.0×10^{-4} full power (FP).

The demanded startup rate was considered to be 4% present-power per second (PP/s) up to 25% FP, 1% FP/s up to about 70% FP above which the adjusters would be inserted and the power would be increased in steps. It would take several hours to reach full-power conditions. The neutronic power transient is given in Figure 5.

iii) Startup After a Long Shutdown

The startup was assumed to be initiated about 33 h after initial shutdown at which time the neutronic power has dropped to about 2 x 10^{-7} FP. The startup time has two components: the time required to raise the temperature of the HTS from cold conditions to hot conditions (heat-up time) and the time required to raise the reactor power from 1% to 100% FP (power run-up time). The heat-up time was taken to be 1.3 h, during which time power was raised to 0.1% FP. From this point on, the demanded power increase in the section above was taken into account to raise the reactor power from 0.1% to 100% FP conditions. The neutronic power transient is given in Figure 6.

iv) Setback Transient

The demanded setback rate was considered to be 0.5% FP/s from 100% to 25% FP, followed by 2% PP/s to 2% FP. The reactor was assumed to be kept critical until it poisoned out. The equilibrium fuel conditions were assumed at the start of the transient and the zone controllers were 50% full on average. Note that the transient curve was based on the calculations of the Darlington station in which one of the two banks of mechanical control absorbers (MCAs) was inserted about 50% at about 275 s. To keep the reactor critical at 2% FP, the MCAs and adjusters had to be withdrawn to compensate for xenon buildup. The MCAs were out at 315 s, whereas the adjusters were all out at 2590 s. The transient was extended to 5000 s by extrapolation with the neutronic power fractions shown in Figure 7.

v) Stepback Transient

This transient is a shutdown by insertion of 4 mechanical control absorbers. The reactivity used for the control absorbers (MCAs) was -10.8 mk. All MCAs were taken to be fully inserted at about 2 s. The neutronic power rundown curve extended to 5000 s is shown in Figure 8.

5.2 Normalized Neutronic and Decay Powers During the Transients

The normalized values of the neutronic and fission-product decay gamma values are required in the transient heating expression for the moderator. Therefore, the normalized values for these two heating components are given in Figures 4 to 8.

The normalized values for the fission-product decay gamma heating in the moderator (i.e., decay power at time "t" to that of the power at time zero during 100% FP operations) for an SDS1 trip setback or stepback transient, were calculated using CANDU channel decay power curves [3]. These fractions (see Figures 4, 7 and 8) were based on the assumption that the neutronic power drops to zero at time zero. That is, the assumption neglects the contributions to the decay power from reactor operations during a transient. This amount is negligible for SDS1 trip transient, and small for setback or stepback transients.

For reactor startups, the fission-product decay gamma fraction of reactor power FP(t) was given at the time when a startup was initiated, i.e., 1500 s after an SDS1 trip (Figure 5) and about 34 h after a shutdown (Figure 6).

Note that at relatively low powers (up to a few percent) during startup, the buildup of fission products is not significant. Consequently, the decay of fission products after initial shutdown was considered while giving FP(t) values. At neutronic powers 10% and above, the fission-product gamma fraction was assumed to be proportional with the neutronic power rise, viz., equal to P(t).

5.3 Normalized Transient Heating in the Moderator System

Using Equation [1] and data on neutronic and fission-product decay gamma fractions from Figures 4 to 8, the calculated moderator-system heat loads are shown in Figures 9 to 13 during an SDS1 trip, a startup after a short trip, a startup after a long shutdown, a setback and a stepback transient respectively.

6. COMPARISON OF MEASURED AND CALCULATED MODERATOR SYSTEM HEAT LOADS AT STEADY-STATE FULL POWER

The methodology used to calculate the moderator-system heat load during steady-state operations at 100% FP was applied to not only CANDU 6 reactors, but also to Bruce A and B, Pickering B and Darlington A reactors.

The moderator-system heat load was calculated to be about 96.7 MW(th) for a typical CANDU 6 reactor during full-power operation. Measurements made at the Point Lepreau reactor ranged between 89.0 and 90.0 MW(th). In Gentilly-2, the measurement at the heat exchangers in November 1984 at 100% FP was reported to be 87.7 MW(th). In addition, measurement data from Wolsong 1 in December 1987 indicated a moderator-system heat load of 88.5 MW(th), which was derived from the temperature rise measured with local dial-type temperature indicators. The calculated heat load value is an overestimation of the measured values, though no uncertainty values were specified with those measurements. An uncertainty of about ±7% was estimated for the calculated heat load.

From a measurement made at Bruce A heat exchangers, the moderator-system heat load was extrapolated to a value of 129 ± 14 MW for which the uncertainty in the measured value is about 11%. The calculated value of the moderator-system heat load for Bruce A during steady-state operation at full power was calculated to 125 MW ± 9 MW. The estimated uncertainty in the calculation is about ±7%. Thus, with the uncertainty included, the calculated values appear to be the best estimates for CANDU 6 and Bruce A station moderator systems. The methodology used to calculate the heat loads is the same for all CANDU reactors.

7. CONCLUSIONS

This paper has documented the nuclear heat deposition calculations for the moderator system during full-power operation and during various reactor transients for CANDU 6 reactors. The lattice cell and bulk shield calculations were combined to obtain the total heating values using a derived transient heat load expression. The methodology and the analysis can be applied to other CANDU reactors, using an appropriate neutronic power curve during desired reactor transients demanded by certain service conditions. Reasonably good agreement between the measured and calculated full-power heat loads in the moderator system for CANDU reactors demonstrates that the methodology is sound and that the analysis results can be applied to full power or transient conditions for CANDU reactors. However, the overestimation of the CANDU 6 heat load indicates that the modelling assumptions need to be reviewed and the methodology needs some refinement.

8. REFERENCES

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Table 1
Moderator System Heat Load During Steady-State Operation at 100% Full Power for a Typical CANDU 6 Reactor

Component	Heating MW(th)
Heat Deposited in Calandria Tubes	4.3
Heat Deposited in Reactivity Mechanisms & Guide Tubes	2.7
Heat Deposited in Moderator	78.6
Heat Deposited in Reflector	5.9
Heat from Calandria Shell and Calandria Side Tubesheets	1.8
Heat from Fuel Channels	3.0
Total Fission Heat to Moderator	96.3
Heat Loss to Moderator Piping	-0.3
Net Fission Heat to Moderator	96.0
Pump Energy Appearing in Moderator	0.7
Net Fission Heat to Heat Exchangers	96.7 MW(th)

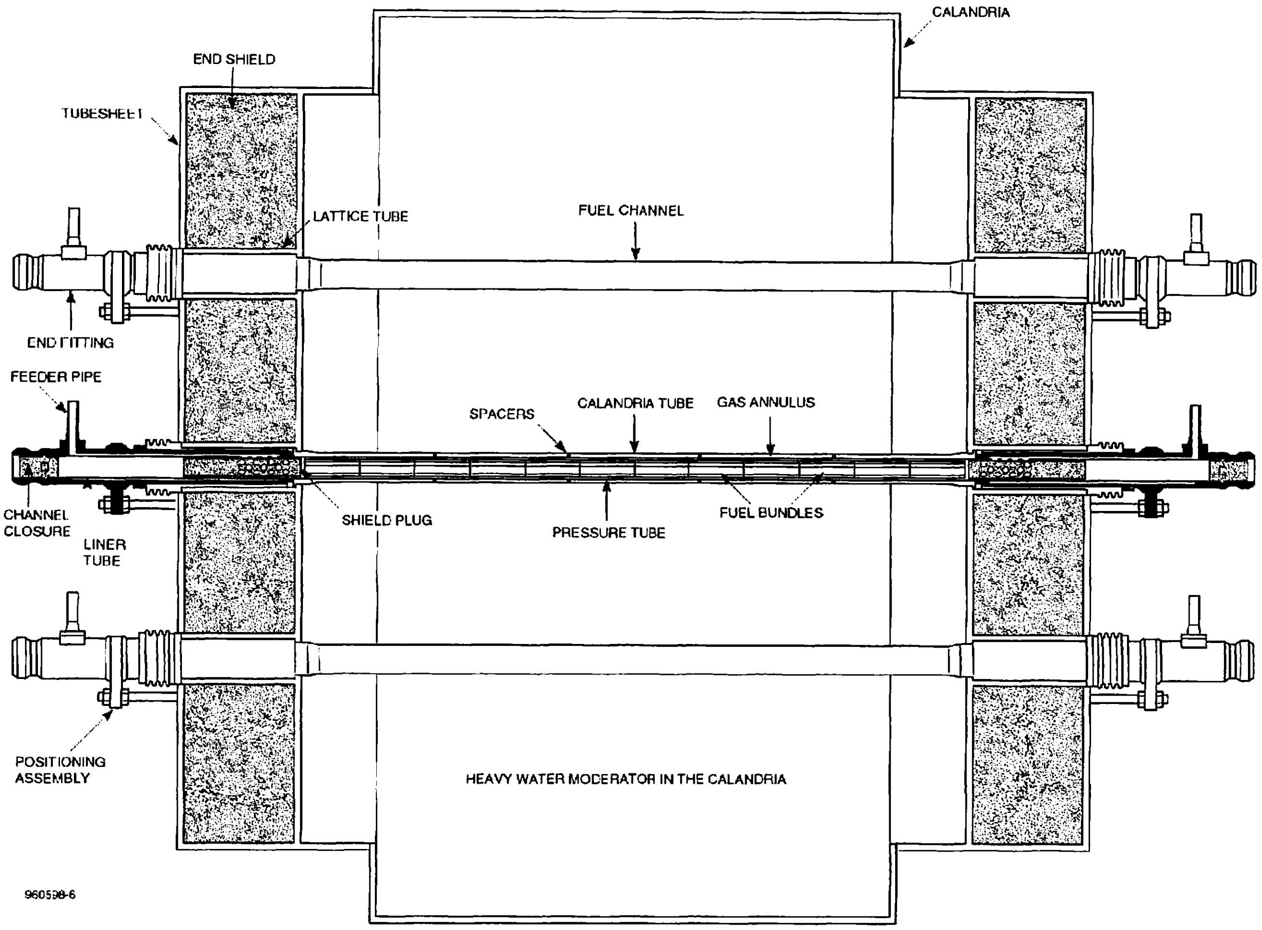

Figure 1 Calandria Assembly Scheme

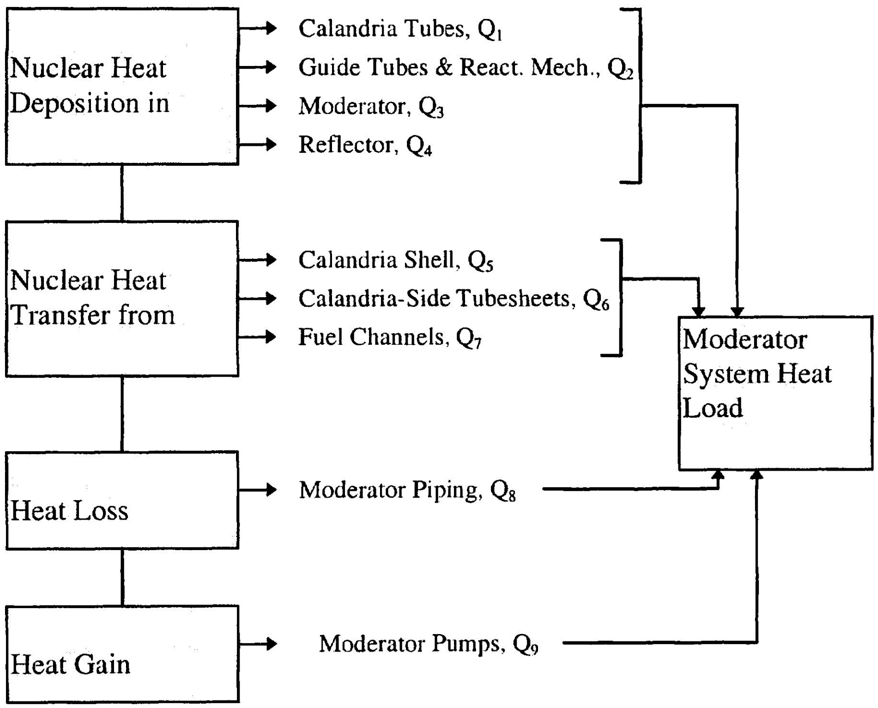

Figure 2 Moderator System Heat Load Components During Reactor Operation and During Reactor Transients

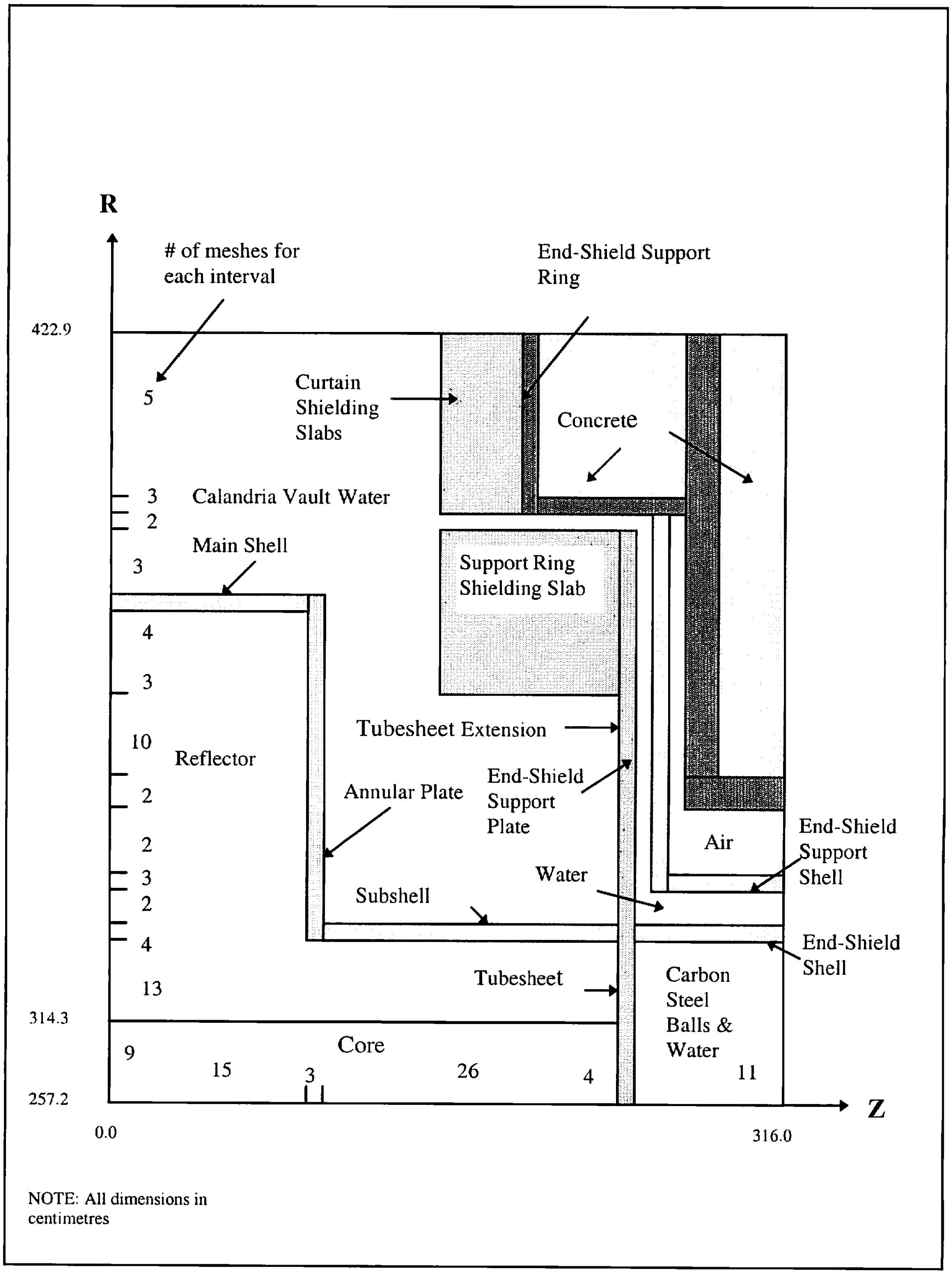

Figure 3 Showing R-Z Geometry DOT IV Model Used in Calandria Shell and End-Shield Ring Analysis

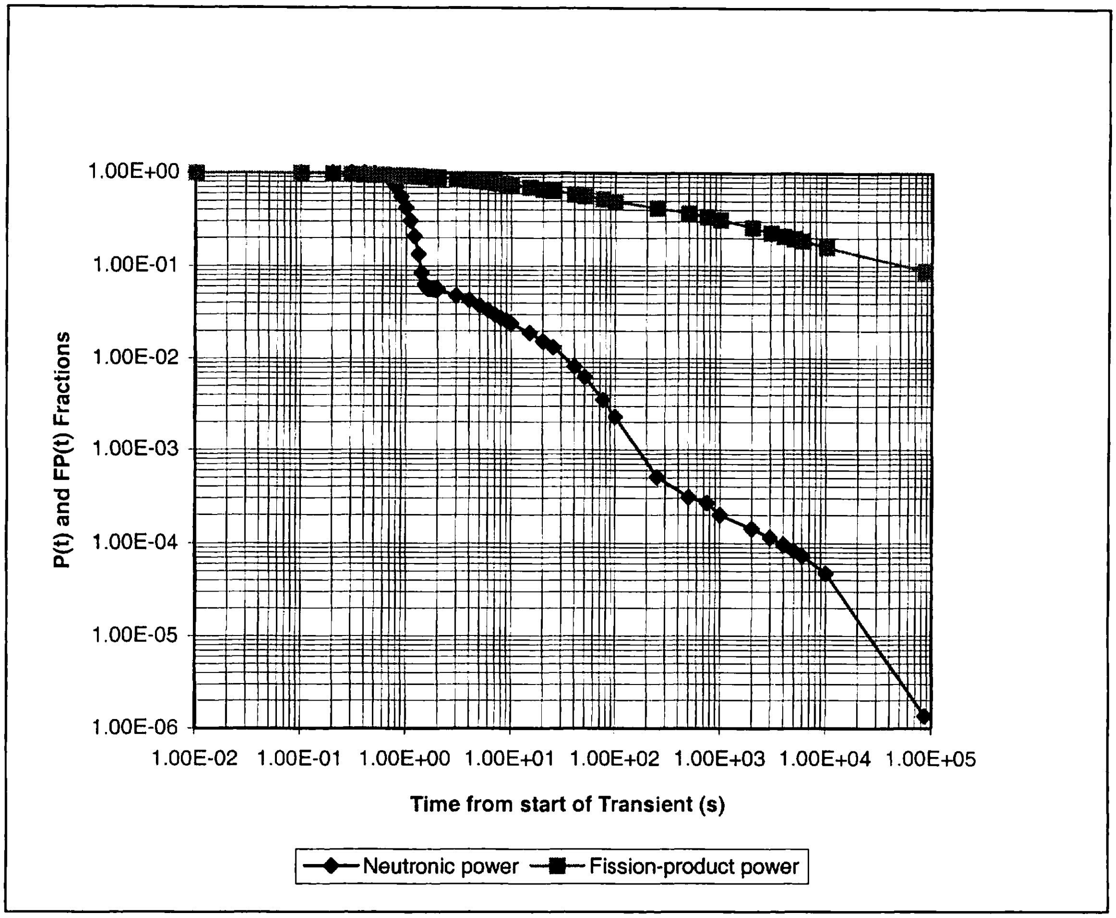

Figure 4 Power Fractions During SDS1 Trip Transient

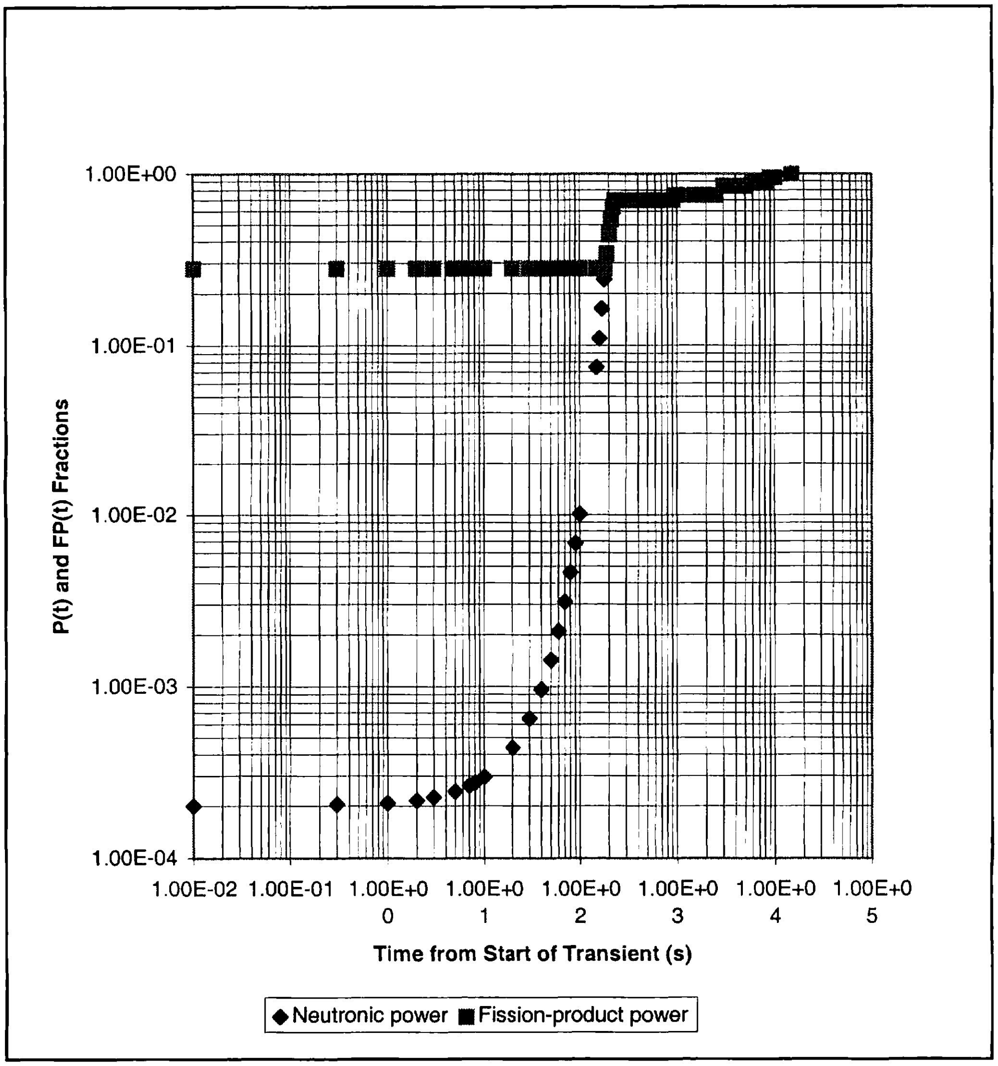

Figure 5 Power Fractions during Startup after a Short Trip

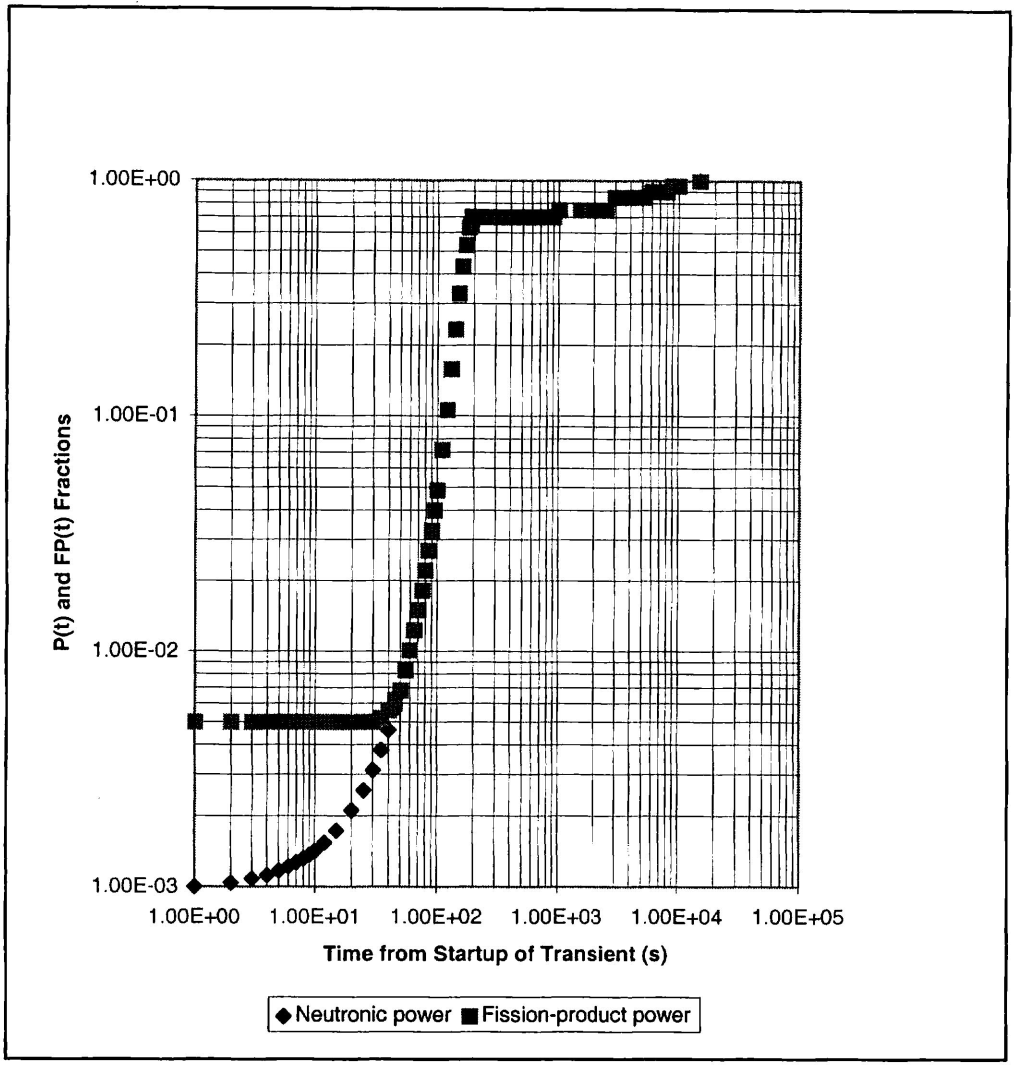

Figure 6 Power Fractions during Startup after a Long Shutdown

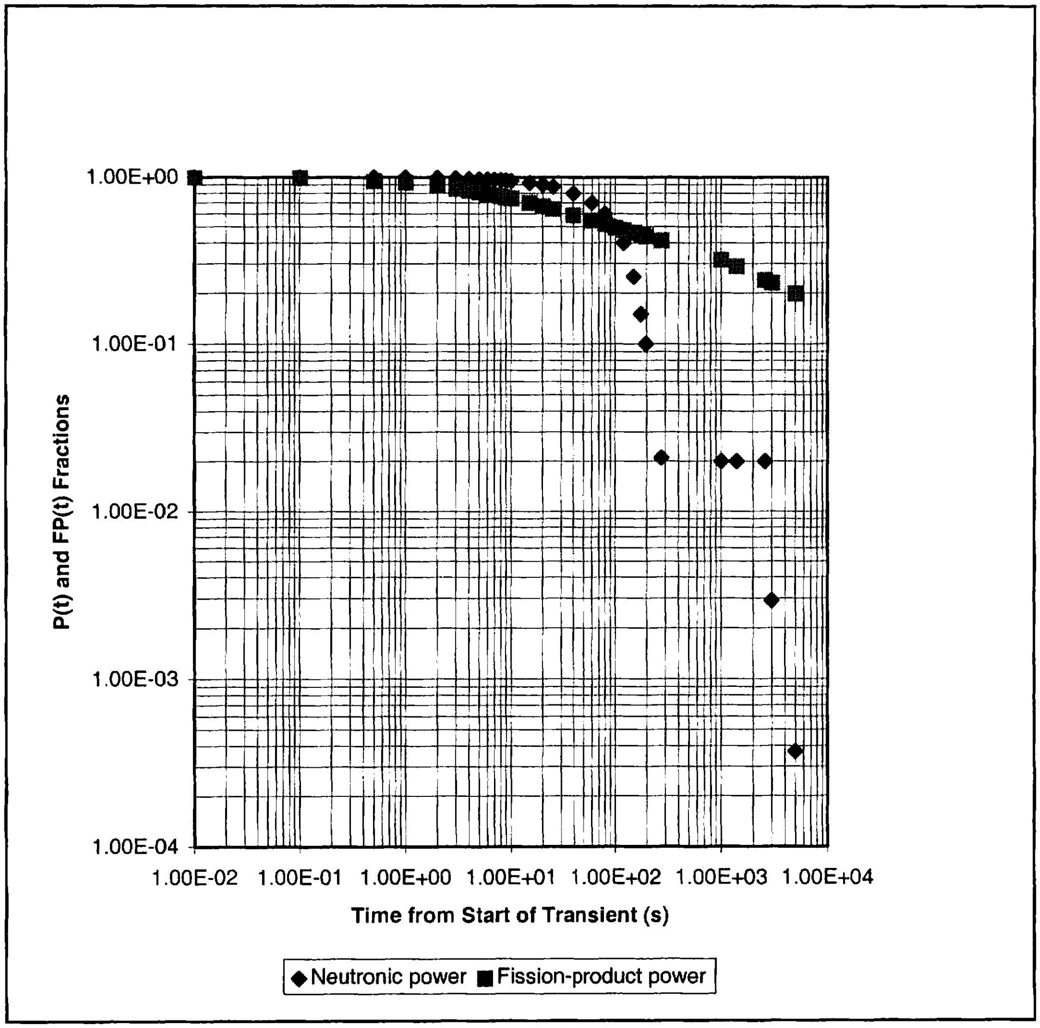

Figure 7 Power Fractions during a Setback Transient

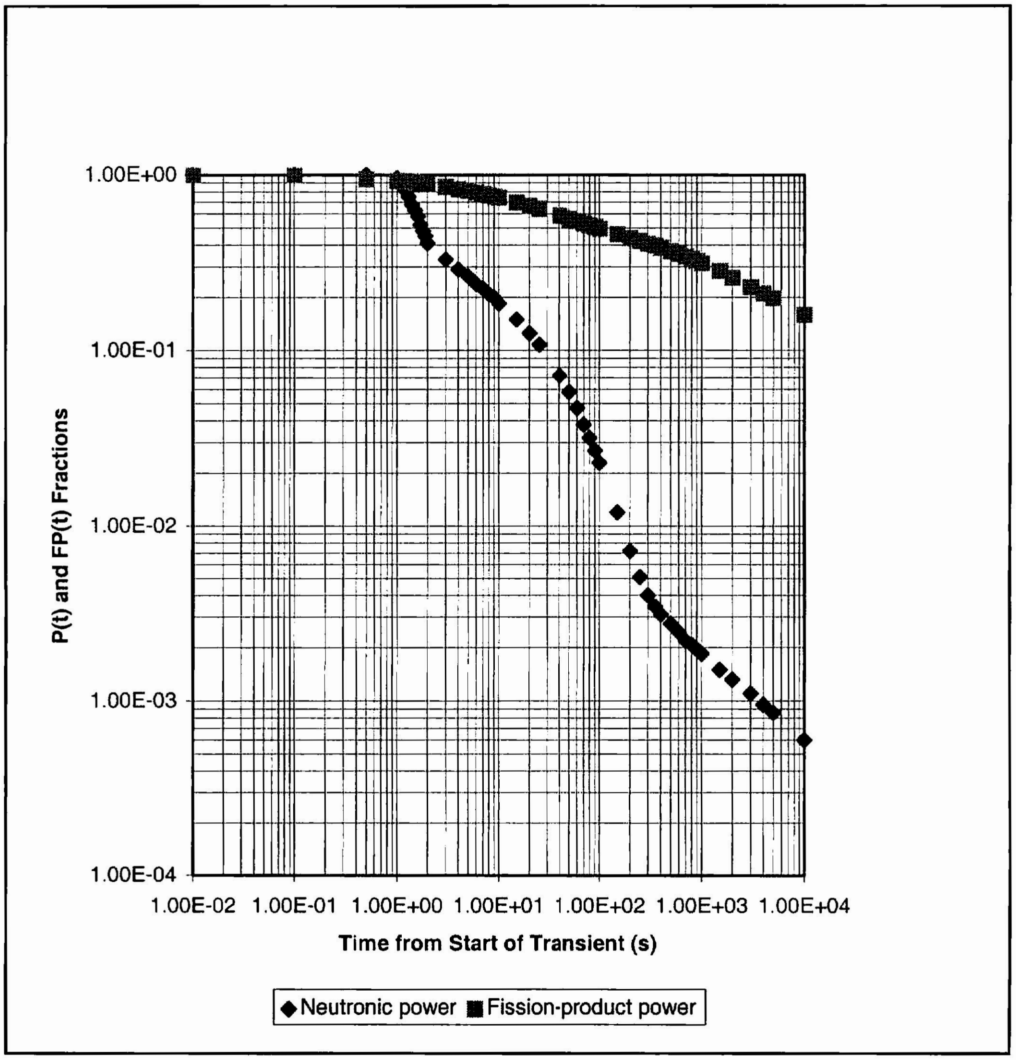

Figure 8 Power Fractions during a Stepback Transient

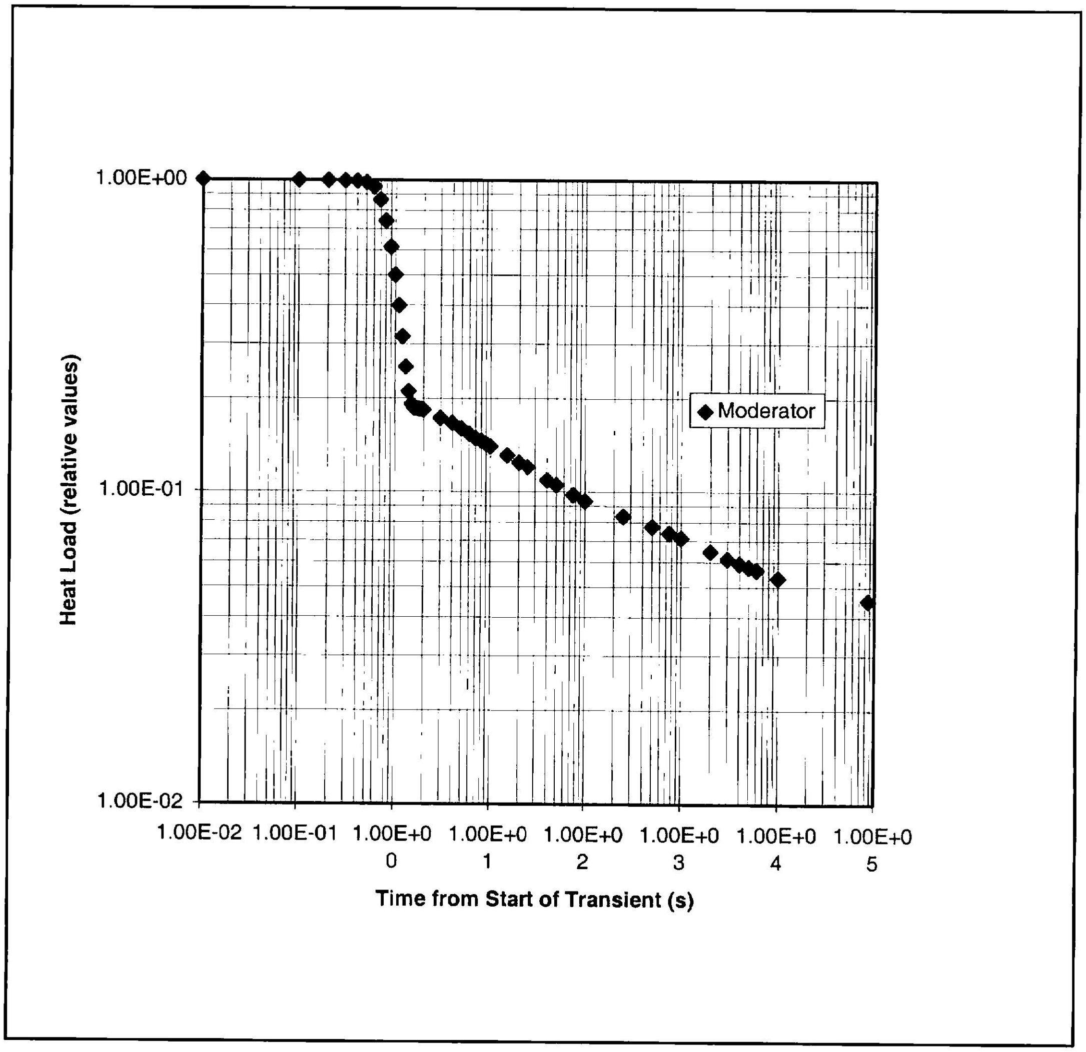

Figure 9 Moderator System Heat Load during SDS1 Trip Transient

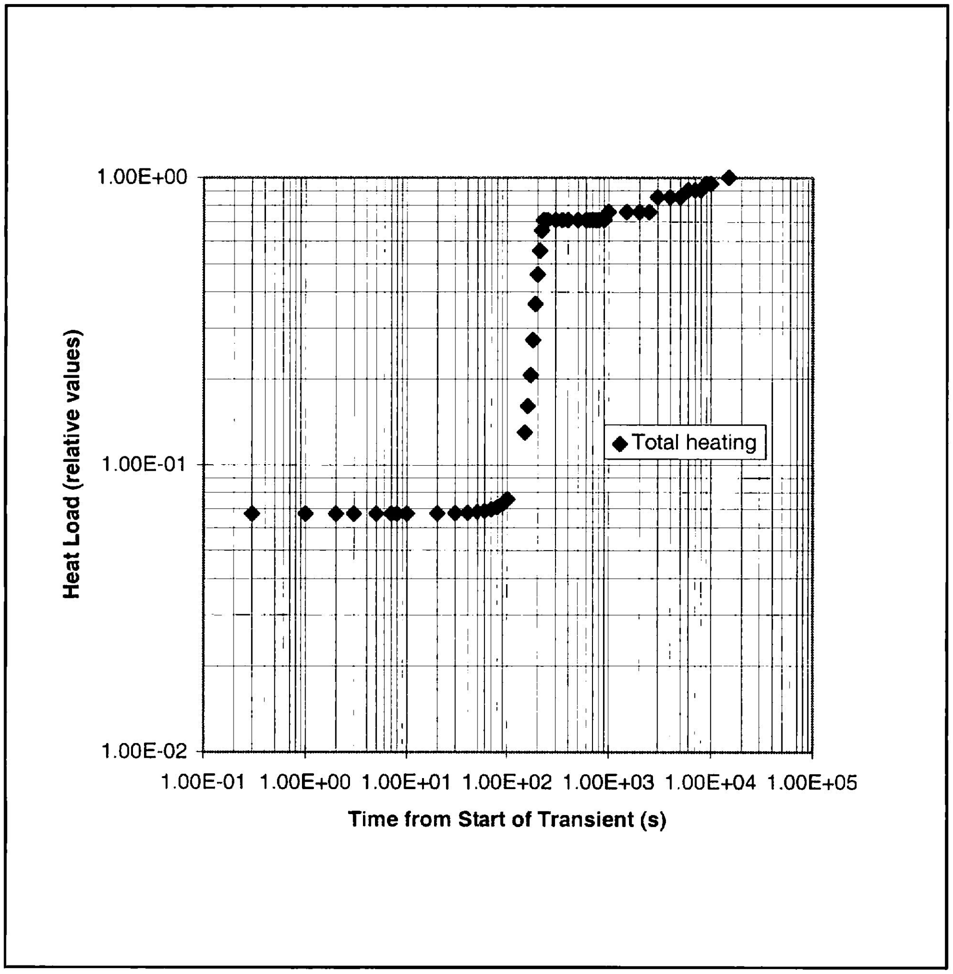

Figure 10 Moderator System Heat Load during Startup after a Short Trip

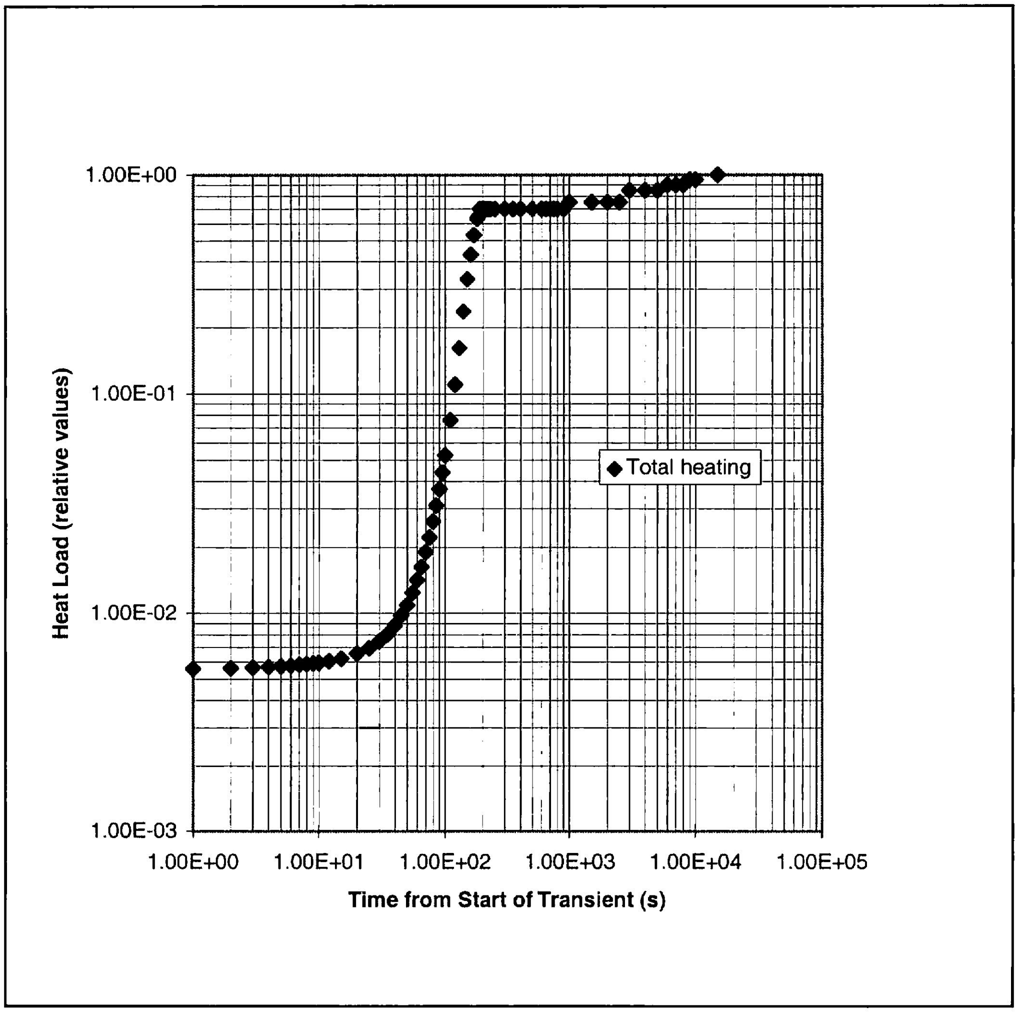

Figure 11 Moderator System Heat Load during Startup after a Long Shutdown

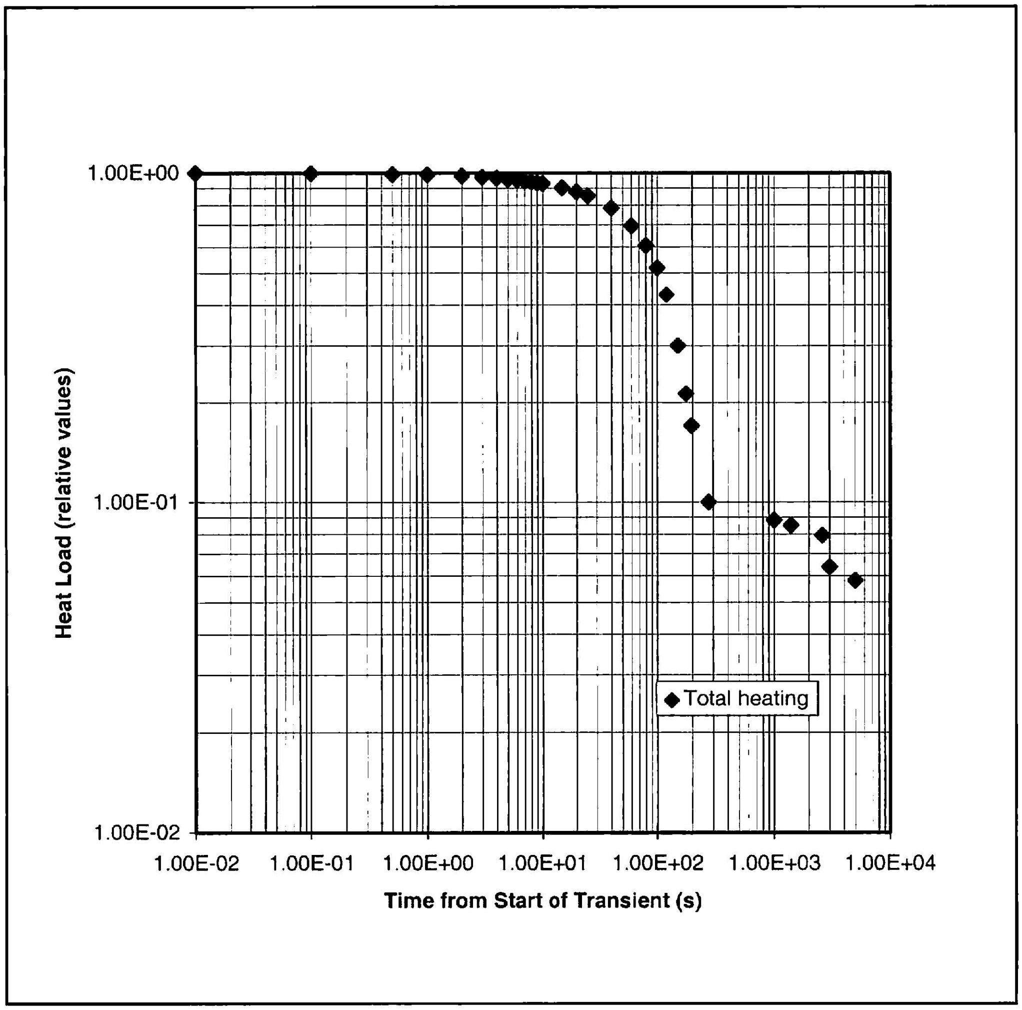

Figure 12 Moderator System Heat Load during a Setback Transient

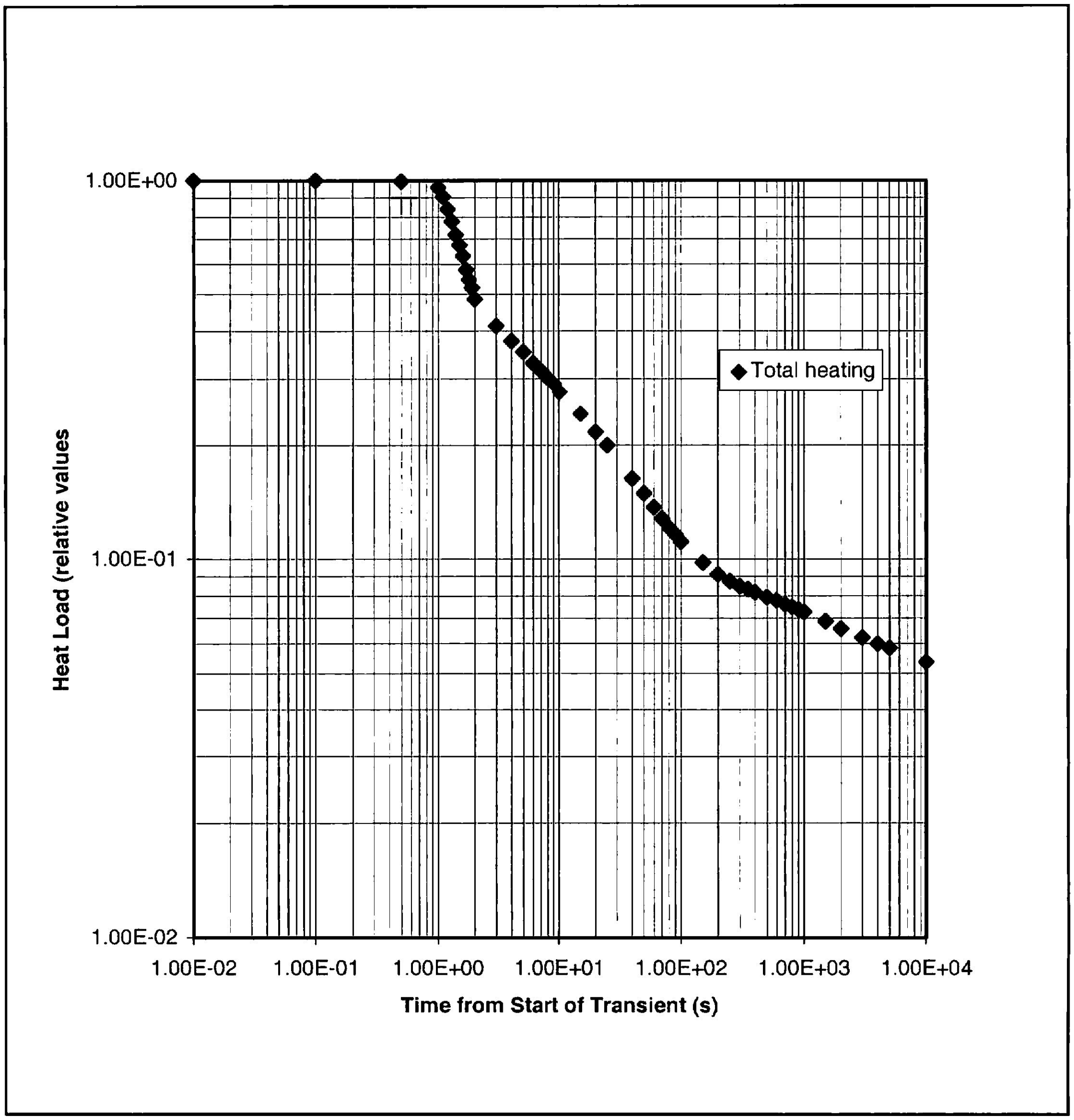

Figure 13 Moderator System Heat Load during a Stepback Transient