REACTOR PHYSICS ASPECTS OF MODELLING IN-CORE SMALL LOSS OF COOLANT ACCIDENTS

E.V. Carruthers
M.Z. Farooqui
A.S. Adebiyi
G. Gaboury
Nuclear Design Engineers
H.J. Smith
Specialist Design Engineer
M.B. Gold
Supervising Design Engineer
Reactor Physics*, Ontario Hydro
11th Floor, 700 University Ave., Toronto, Ontario M5G 1X6

In-core small loss of coolant accidents(LOCAs) have been analyzed over a wide range of possible initial core states including that of a high moderator poison concentration. The in-core rupture of a pressure tube and its calandria tube represents a unique class of small breaks due to the possibility of damage to in-core reactivity devices such as the shutoff rod (SOR) and mechanical control absorber (MCA) guide tubes. In addition, the potential displacement of moderator poison by the unpoisoned coolant discharge would provide an additional positive reactivity source. The upper limit of moderator poison considered was that which would occur if the reactor is restarted after being shutdown, just after being overfueled by the maximum permissible amount, for a long enough period that all saturating fission products have decayed. In the case of a pre-equilibrium core analysis, the reactor is assumed to have shut down at or near the plutonium peak core state (with no overfueling prior to the shutdown).

ACCIDENT SCENARIO

The postulated break is a spontaneous rupture of a pressure tube and simultaneous failure of it calandria tube. This is unlikely[1].

Consequential Damage

The discharge of high pressure, high temperature coolant from the ruptured PT/CT into the moderator causes a complex hydrodynamic transient in the calandria, as well as other phenomena which could result in damage to the calandria or other in-core structures. Possible mechanisms for consequential damage are[2]:

1 Pressure Pulse. The initial, short duration, pressurization of a localized volume in the vicinity of the break could cause deflection and possible interference between guide tubes, fuel channels and other in-core structures.
2 Jet Force. The long-term, two phase jet force could exert a force on neighbouring channels or guide tubes.
3 Pipe whip. The thrust of the exhausting steam/water mixture could deflect the failed channel and cause it to impact upon other in-core structures.
4 Fuel Bundles. Ejected fuel bundles could potentially impact upon in-core structures or calandria walls.

The location of the PT/CT break is chosen to maximize the consequences of damage to in-core structures. In the assessment of reactor regulating system (RRS) and core response, the maximum damage to MCA guide tubes is assumed to challenge the ability of the RRS to maintain reactor power constant prior to a

---

* Thanks to N. Roy and A.L. Wight for early development of the methodology.

credited reactor trip, as shown in Figure 1. For determining the long term subcriticality margin, up until credited operator action, the maximum damage to SOR guide tubes is assumed (Figure 2).

Moderator Poison

A unique feature of this accident is that if soluble poison is present in the moderator for reactivity hold down, the displacement or dilution of the poison by unpoisoned heat transport system coolant will increase the reactivity of the system. A large amount of poison is present when the reactor has been fuelled ahead and has just been brought to power after a long shutdown The reactivity change depends also on the difference in isotopic purity between the coolant and the moderator. The situation which maximizes the increase in reactivity is when the maximum allowable amount of poison is present in the moderator and when the coolant isotopic purity is high. System response following a PT/CT rupture is analyzed for limiting conditions of maximum poison load in the moderator in combination with several representative values of differential isotopic purity between the moderator and the heat transport system coolant.

METHODOLOGY

The analysis of PT/CT breaks involve interaction with several nuclear disciplines. Thermal hydraulic and physics analysis must agree upon the core bulk power transient and void/fuel reactivity transients. This iterative process is minimal in PT/CT breaks as the void reactivity effect is small compared with the several neutronic and process reactivity transients due to changes in moderator conditions. Safety system initiation involves tracking reactor trip parameters to determine when the reactor is credited as tripped, also assessing parametric coverage of possible operating states**. Potential fuel or fuel channel failure and containment analysis complete the assessment. This is illustrated in Figure 3.

The physics analysis of in-core breaks consists of two distinct objectives. The first of these is to assess the reactor regulating system response (RRS) to a range of possible reactivity transients and determine the core bulk and spatial power transients. Through interaction with the thermal hydraulics and safety system groups, these power transients are further analyzed to assess channel dryout times and trip parameter effectiveness. The second objective is to calculate an overall reactivity balance to assess the subcriticality margin up until operator action is credited (fifteen minutes after unambiguous indication of the accident).

RRS and Core Response

For trip parameter effectiveness, the study considered a range of possible break sizes, a range of moderator to coolant isotopic purity differences, two characteristic initial zone controller levels (70 and 40 percent full), as well as various initial power levels. Reactor trip is credited only after both shutdown systems*** have registered two credible trips. As the transients are quite slow with respect to shutdown system response, activation

** cf. "Assessment of Shutdown System Trip Parameter Effectiveness of CANDU Reactors Following In-core Loss of Coolant Accidents" by A.F. Oliva and L.J. Watt elsewhere in these proceedings.

***One in Pickering NGS A

of the shutdown systems is not simulated. The location of the in-core damage zone is chosen to maximize the consequential damage to the reactor regulating system. For this range of parameters, the moderator poison load was assumed to be at its maximum level corresponding to a restart following a long shutdown. A reactivity equivalence between the moderator poison displacement and moderator purity degradation was established allowing assessment of lower moderator poison concentrations.

The SMOKIN[3] code was used to model the reactor regulating system response, the resulting power excursion, and the neutron overpower trip times. Spatial coolant voiding and fuel temperature transients were calculated based on thermal hydraulic data provided by the SOPHT[4] code. Moderator temperature changes were calculated by the SOMASS[5] code. The discharged coolant was assumed to mix rapidly (relative to the 900 second timeframe of the analysis) with the moderator, resulting in global reactivity transients due to moderator poison displacement, moderator purity degradation by the coolant, and changes in moderator temperature due to the higher temperature of the discharging coolant. These global reactivity contributions were calculated using a spreadsheet point reactivity model[6] based on reactivity variations calculated by the POWDERPUFS module of the OHRFSP[7] code.

The moderator poison displacement and moderator purity degradation reactivity transients are calculated based on the initial moderator poison load and relative moderator to coolant isotopic purities, the relative displacement of the moderator, and moderator poison and moderator purity reactivity coefficients. The moderator temperature reactivity equation consists of a third order polynomial in the primary variable (moderator temperature) with first order multivariable terms proportional to the product of the primary variable and the secondary variables moderator poison concentration and moderator isotopic purity. That is:

$$Rho(MT, MB, MP) = m1*MT+m2*MT^2+m3*MT^3+mb1*MT*MB+mp1*MT*MP$$

where:

Rho is the change in reactivity
MT is the change in moderator temperature
MB is the change in moderator poison concentration
MP is the change in moderator isotopic purity
m1, m2, m3, mb1, mp1 are reactivity coefficients as described below

The reactivity coefficients used in the spreadsheet reactivity model are calculated from a least-square fit of POWDERPUFS-generated data for a variety of moderator temperature, poison concentrations and purities. Figure 4 illustrates the interaction of these codes as used in the analysis of core and regulating system response for small LOCAs.

<u>Subcriticality Margin Assessment</u>

For assessing the subcriticality margin up until operator action is credited, the spreadsheet reactivity model is used. The location of the in-core damage zone is chosen to maximize the consequential damage to the shutoff rod guide tubes, reducing shutdown system one's reactivity depth. The reactivity worths of the shutoff rods and regulating system devices are calculated using the TIME-AVER or SIMULATE modules of OHRFSP for the equilibrium or pre-equilibrium core respectively. In addition to the three moderator reactivity equations described above (poison displacement, purity degradation, and temperature), the spreadsheet model includes polynomial equations for fuel temperature and coolant voiding as described in Reference 6. The coolant is assumed to be at its maximum value, with respect to the moderator, as opposed to the range of values considered in RRS and core response. Note that while the spreadsheet model includes contributions from the RRS prior to reactor trip, the implementation is quite simple compared with that of SMOKIN.

This model is used to calculate an overall reactivity balance as a function of time in order to assess the subcriticality margin after the reactor is shut down. The subcriticality margin is a function of the time of assumed reactor trip since the RRS is assumed to freeze when the reactor trips. The subcriticality margin is assessed both for an immediate trip following the break and at the latest possible trip time. This trip time is that which would occur if the RRS does maintain reactor power constant and NOP trips are not registered/credited. The subcriticality margin is also calculated both with and without boiler crash cooldown available. The emergency coolant injection system is assumed unavailable. The development and implementation of the spreadsheet reactivity model is illustrated in Figure 5.

RESULTS

RRS and Core Response

Although the results depend on the specific reactor being considered, generally the reactor regulating system could not completely compensate for reactivity transient for the limiting cases considered. Figure 6 illustrates the reactivity transients for a 225 kg/s rupture of a Bruce NGS B pressure tube and calandria tube with a moderator isotopic purity one percent higher than the coolant isotopic purity. Figure 7 illustrates the resulting core bulk power transient, a candidate hot channel power transient (selection of candidate hot channels is described in Reference 6), and the neutron overpower detector signals of the third safety channel of the shutdown system one and shutdown system two detectors. Figure 8 illustrates the wide range of possible core response depending on the differential isotopic purity between the moderator and the HTS coolant.

The spatial changes in the core power distribution resulting from such an asymmetrical MCA insertion are not easy to visualize. Figures 9 through 12 illustrate snapshots of a simulated core channel power distribution during the 225 kg/s transient described above. These figures are contour surface plots of channel power and capture only two of the three spatial dimensions involved. That is, they do not illustrate the axial power variations along the bundles in each channel. Figure 9 shows the initial (time-averaged) channel power distribution. Figure 10, just after NOP trips have been registered (but not credited) at 80 s, shows a little shift in the power shape towards the bottom left of the core. Figure 11 shows this combination top/bottom and side/side tilt more clearly as the two MCA are partially inserted. At 200 s in Figure 12, just before the second trip on SDS2, the tilt is predominantly side/side as the MCA are almost fully inserted. The reactor is shut down a few seconds later.

Subcriticality Margin Assessment

Table 1 illustrates the subcriticality margin calculation for the same break with the minimum operating limit of 0.2 weight percent isotopic difference. The case shown resulted in the smallest margin to criticality of 6.3 mk and assumed the latest possible trip time, loss of emergency coolant injection and no boiler crash cooldown following reactor trip.

CONCLUSION

The in-core rupture of a pressure tube and calandria tube has been analyzed over a wide range of initial core states. The positive reactivity transient associated with such a break occurring when the moderator is heavily poisoned has been shown to require special consideration during these short operating periods.

TABLE 1 REACTIVITY BALANCE IN-CORE BREAK - 103% power

Isotopic Difference = 1 wt %
Shutdown by : SDS1 Device Worths from: BB Poison= 44 mk
PT/CT Core Model: BB-Equil Transients File: BB_LOECI_225_TT

Time (seconds)	0	100	200	235	300	400	500	600	700	800	925
Parameter											
Mod. poison dillution	0.0	2.9	5.5	6.4	7.6	9.1	10.5	11.9	12.8	13.7	14.6
Coolant void	0.0	0.1	0.9	1.2	1.3	1.1	2.5	3.0	3.5	3.8	4.1
Fuel temperature	0.0	0.0	0.0	0.0	3.8	3.8	3.8	3.8	3.8	3.8	3.8
Moderator temp.	0.0	1.7	2.9	3.2	2.8	1.7	0.7	0.0	-0.3	-0.4	-0.5
Degrading moderator	0.0	-2.2	-4.2	-4.9	-5.8	-6.9	-8.0	-9.1	-9.8	-10.5	-11.2
Xenon reactivity	0.0	-0.1	-0.2	-0.2	-0.2	-0.2	-0.2	-0.2	-0.2	-0.2	-0.3
Excess reactivity	0.0	2.4	4.9	5.7	9.4	8.5	9.3	9.4	9.8	10.1	10.6
Regulating system	0.0	-2.4	-4.9	-5.7	-5.7	-5.7	-5.7	-5.7	-5.7	-5.7	-5.7
Total reactivity	0.0	0.0	0.0	0.0	3.7	2.8	3.5	3.7	4.1	4.4	4.9
Shut off rods	0.0	0.0	0.0	0.0	-23.0	-23.0	-23.0	-23.0	-23.0	-23.0	-23.0
Net reactivity	0.0	0.0	0.0	0.0	-19.3	-20.2	-19.4	-19.3	-18.9	-18.6	-18.1

Trip time	236	seconds	Void reactivity =	13.62	milli-k
Moderator Purity =	99.75	wt.%	Fuel temperature coefficient =	-0.006	milli-k/C
Coolant Purity =	98.75	wt.%	Void reactivity uncertainty =	1.4	milli-k
Poison Reactivity =	44	milli-k			
Max. RRS reactivity	9.9	milli-k			
SOR(T=0) Worth	25.2	milli-k			
SOR & RRS Worth	28.7	milli-k			
RRS react. at trip=	5.75	milli-k			

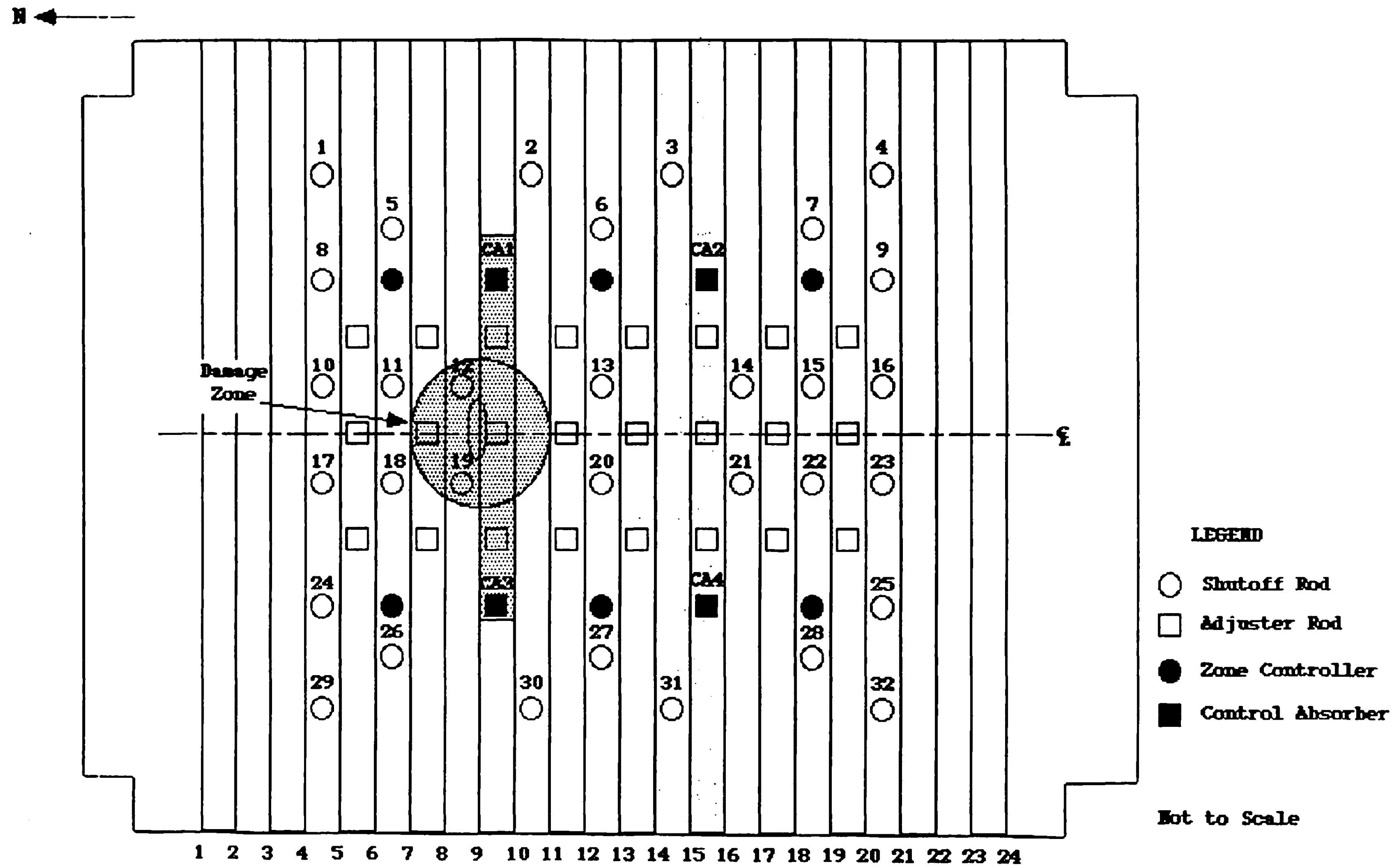


FIGURE 1
Maximum Control Absorber Disabling Following
a Pressure Tube /Calandria Tube Failure (Bruce B)

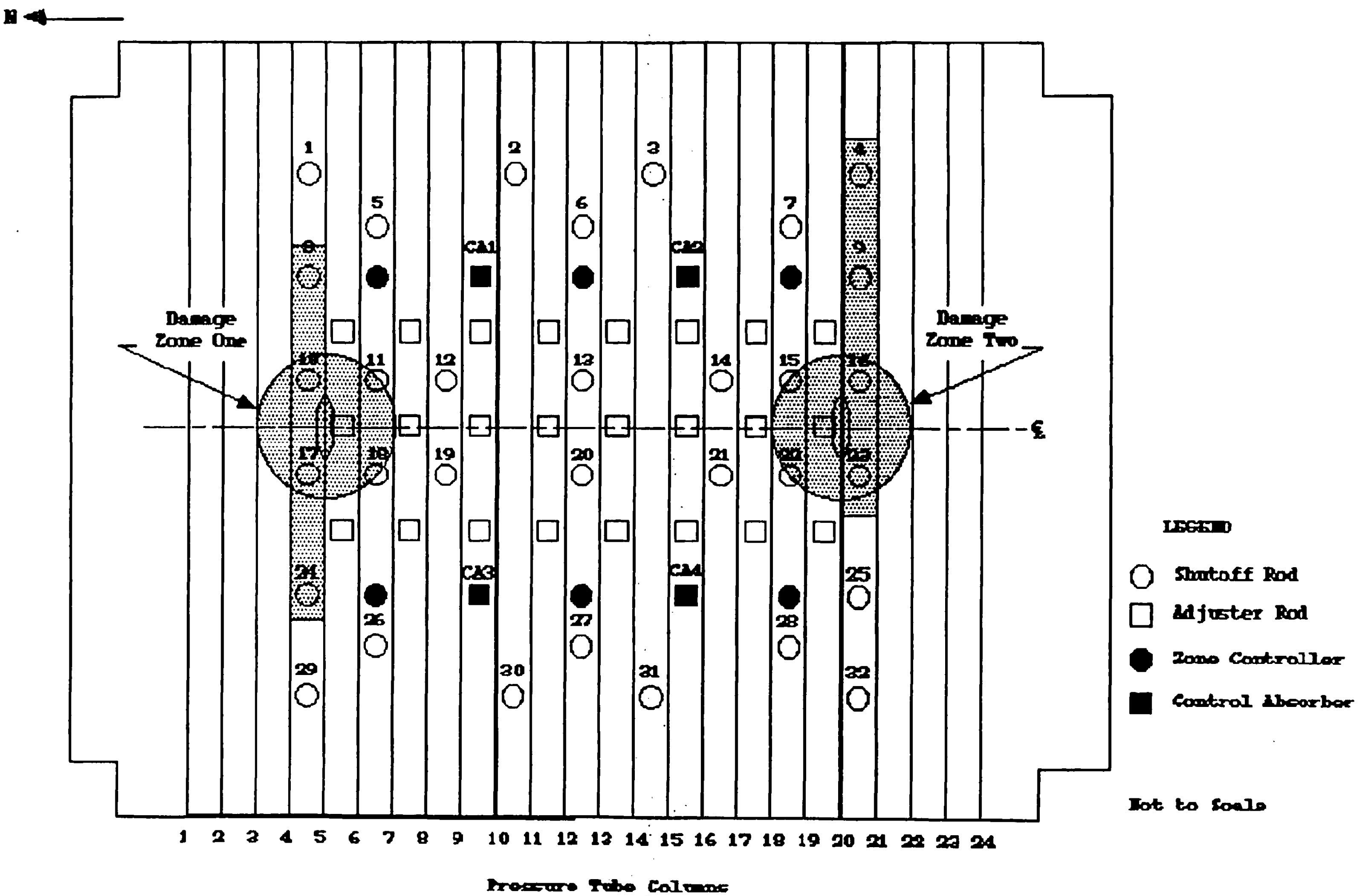


FIGURE 2
Typical Map of Shutoff Rod Disabling Following a Pressure Tube /Calandria Tube Failure (Bruce B)

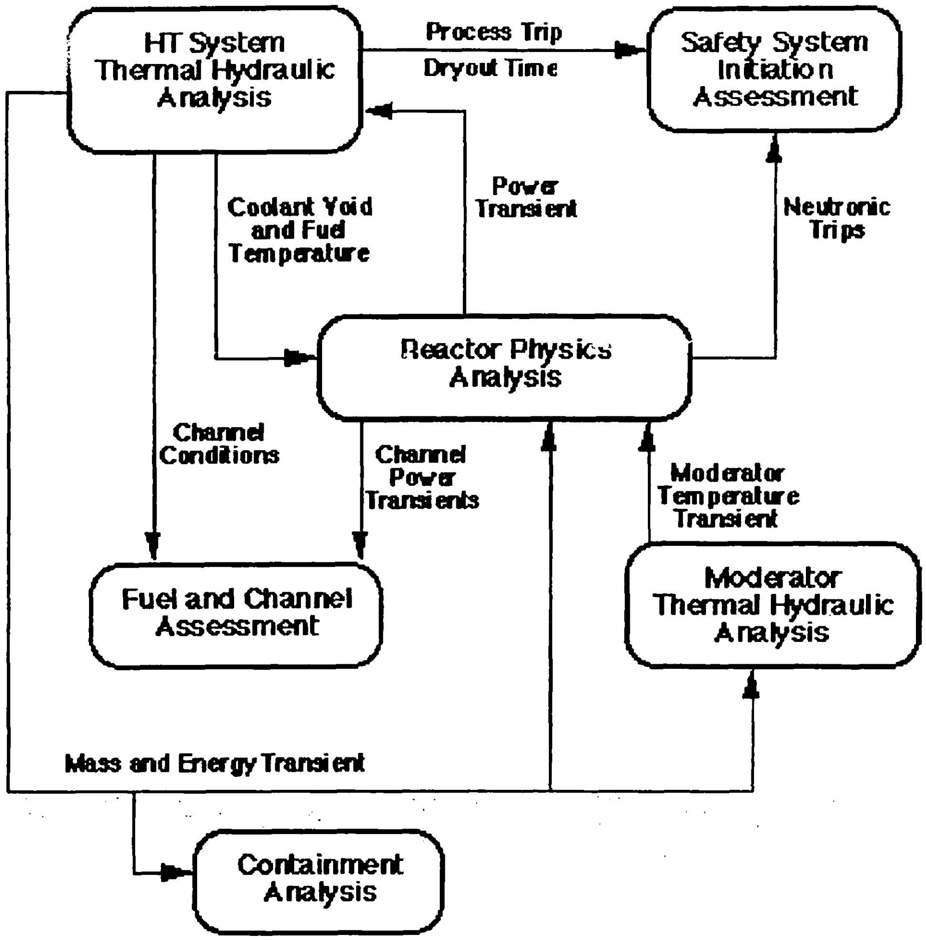


Figure 3
Basic Interactions in a Small LOCA Analysis

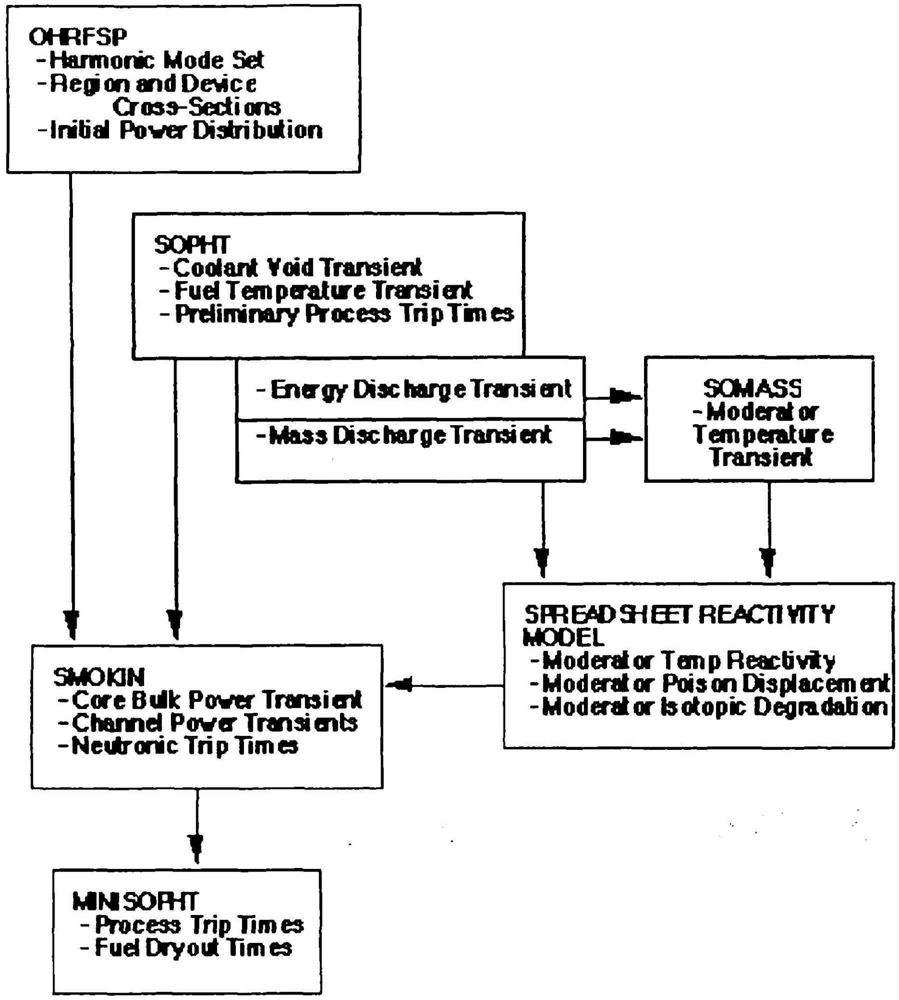


Figure 4
SMOKIN Analysis of an In-Core Small LOCA

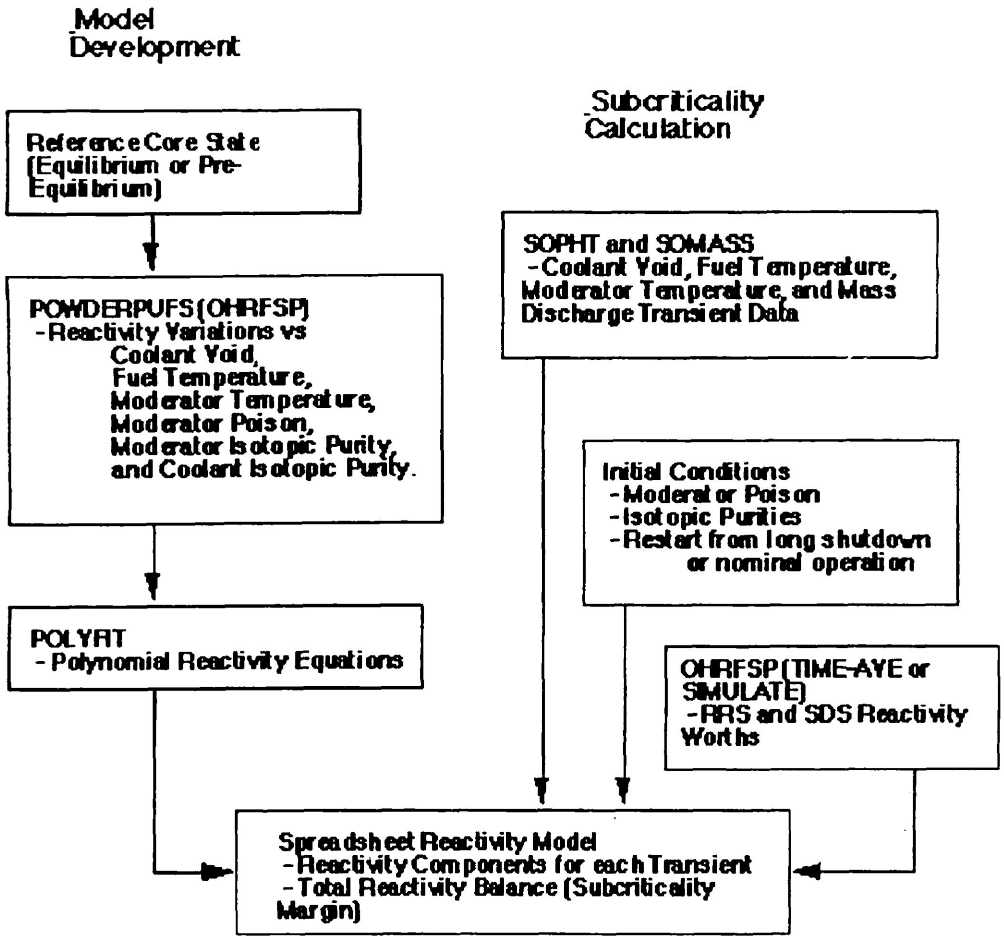
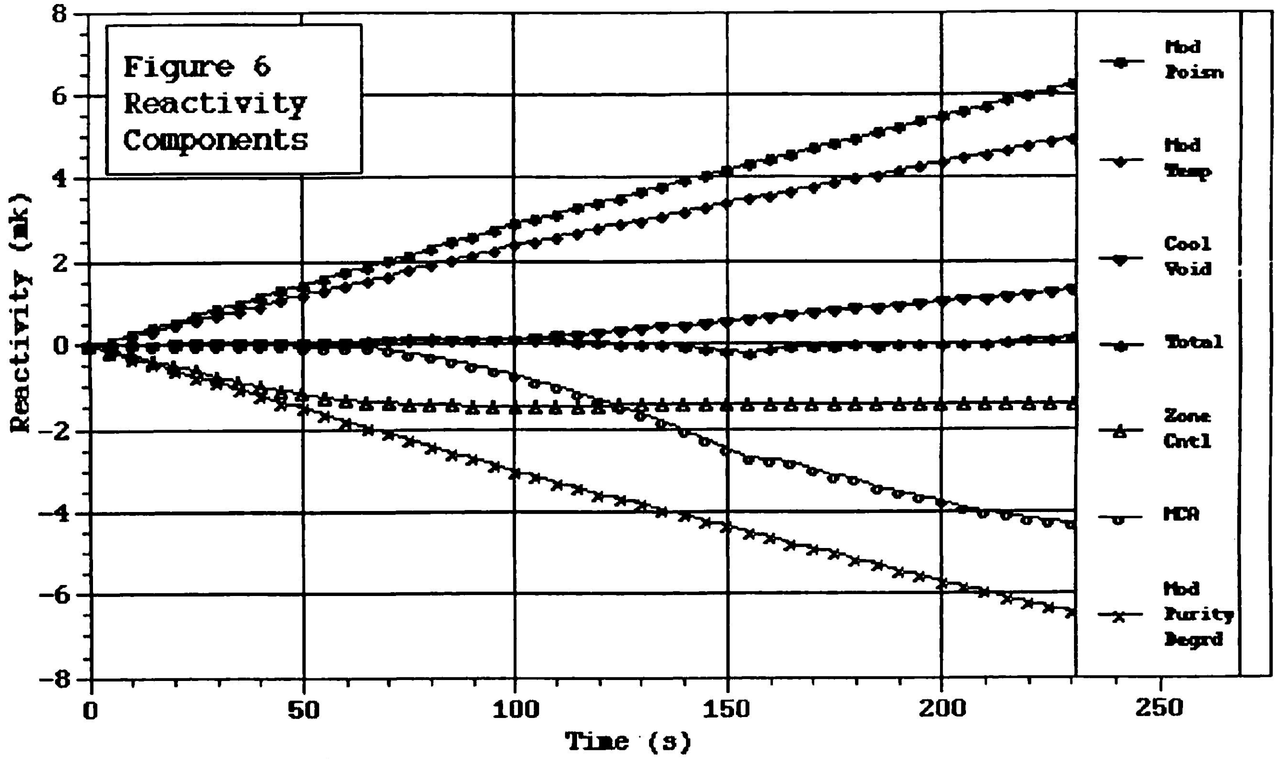


Figure 5
Spreadsheet Reactivity Model Development
and Subcriticality Margin Calculation

Bruce B 225 kg/s PT/CT Break, 1.0 % I.D.



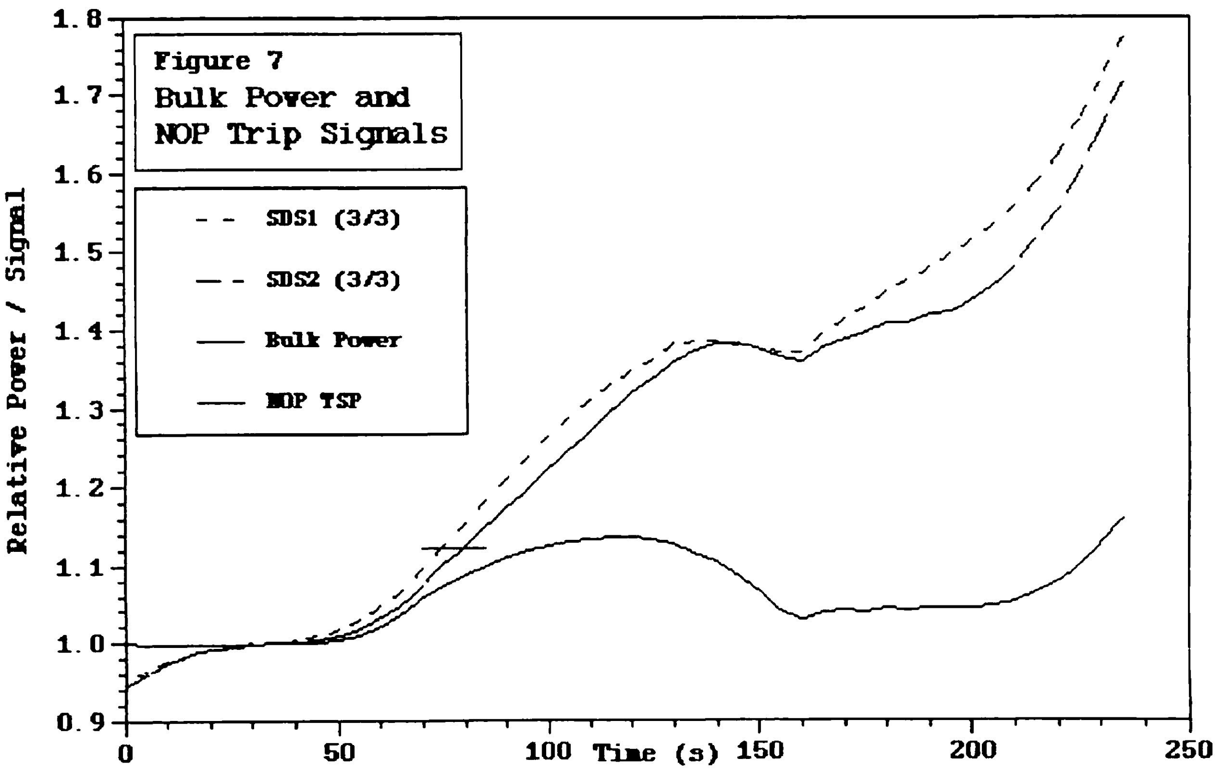


Figure 8
Bruce B 225 kg/s PT/CT Break
44 mk Moderator Poison, Two MCA Disabled
Bulk Power Transients
with Various Moderator to Coolant Isotopic Purity Differences

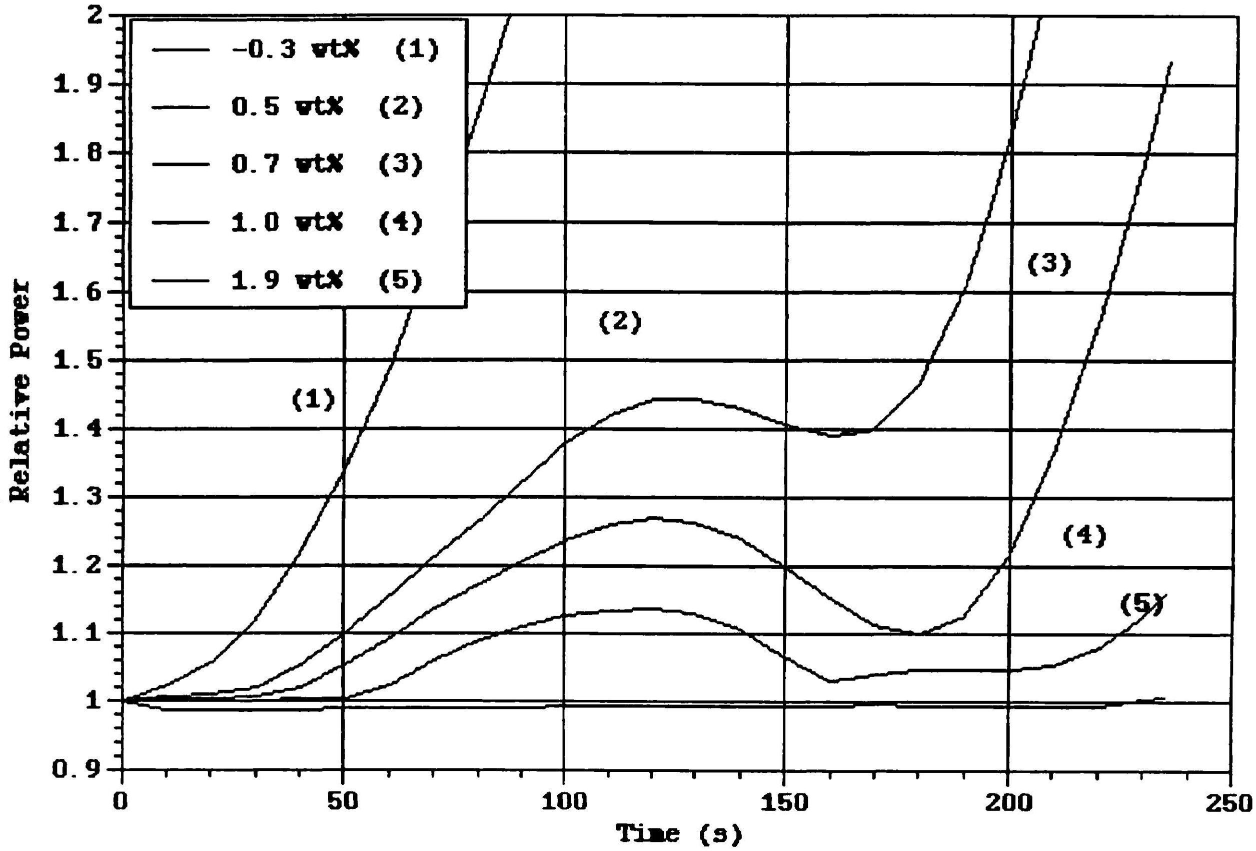
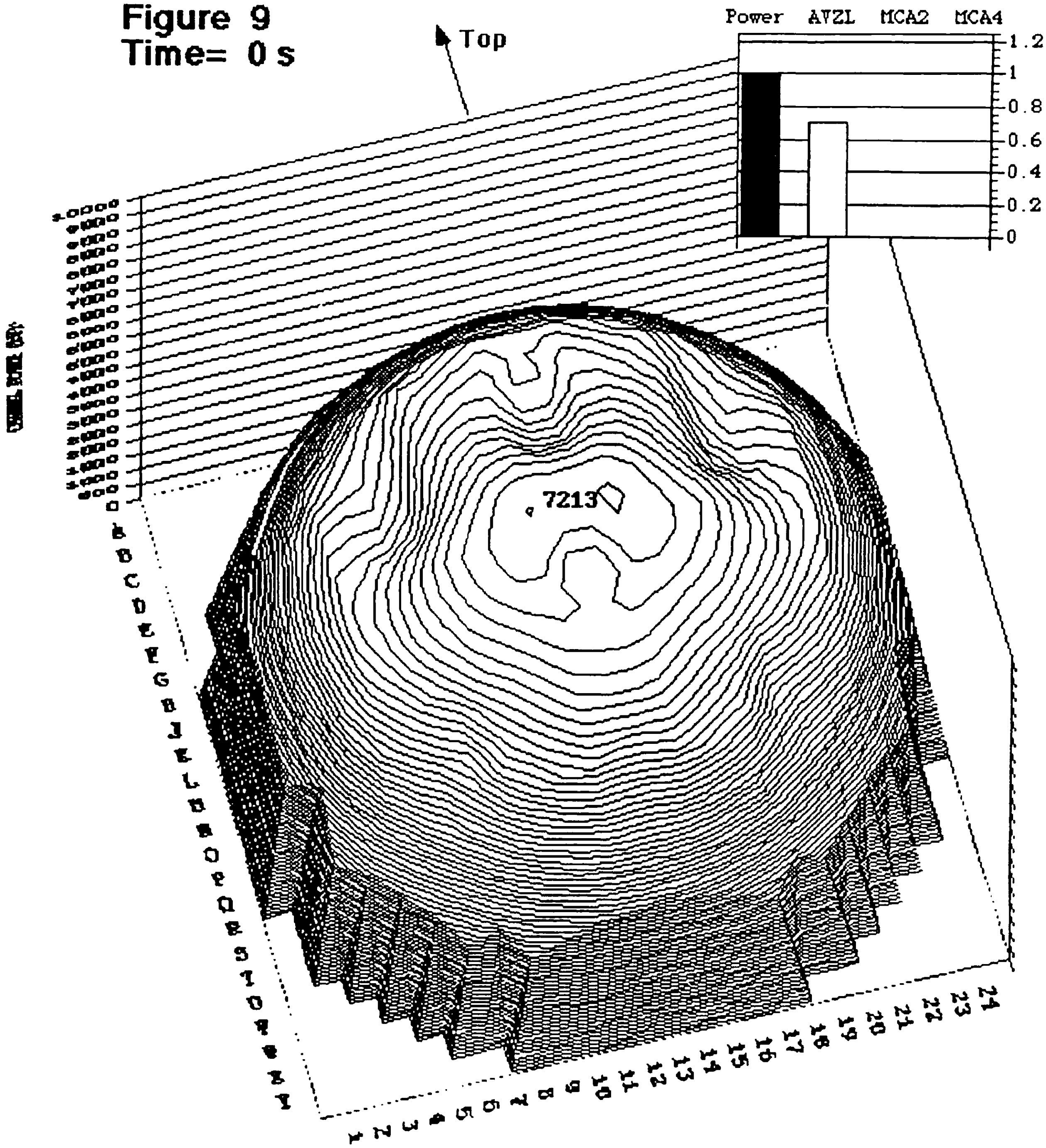
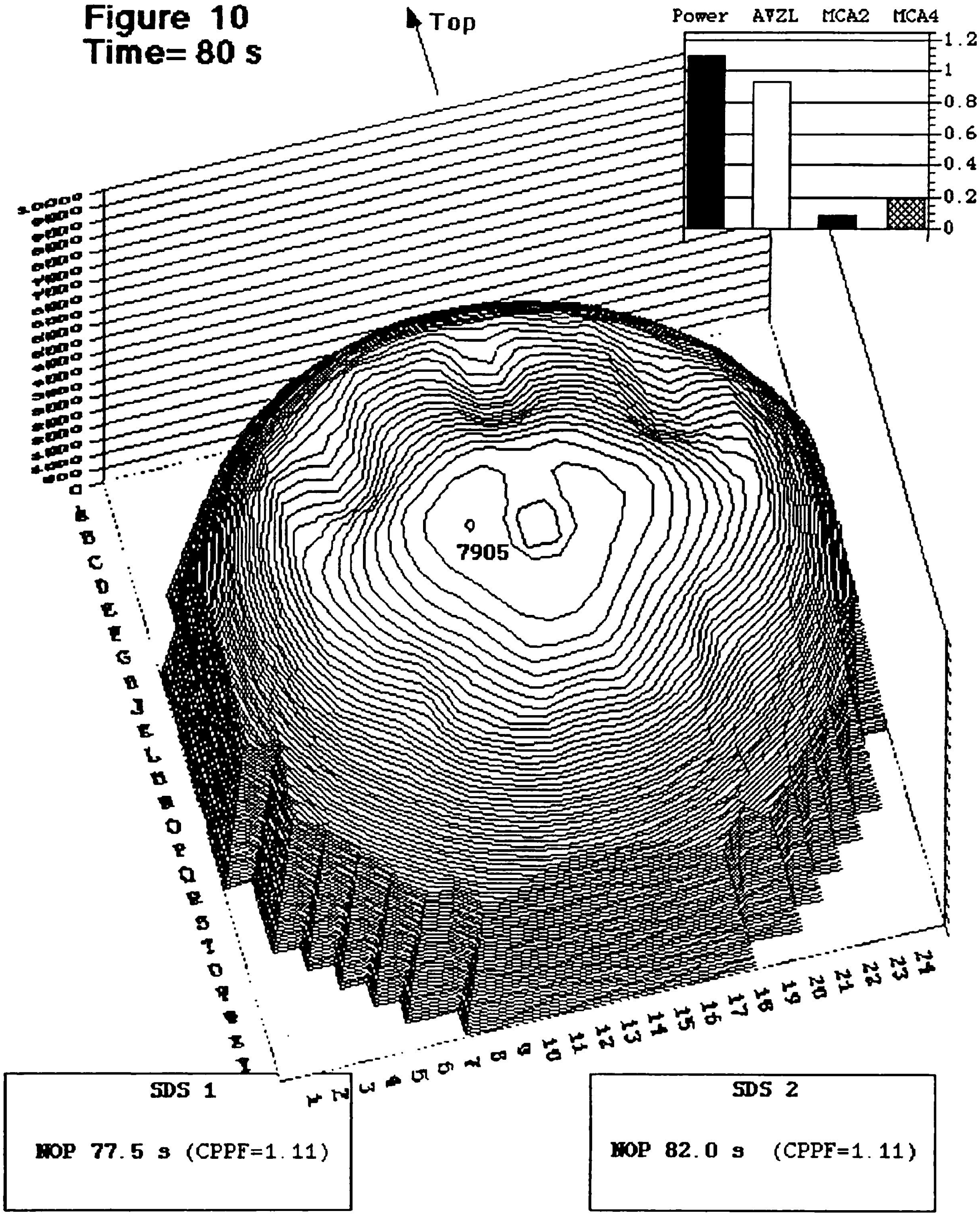


Figure 9
Time= 0 s

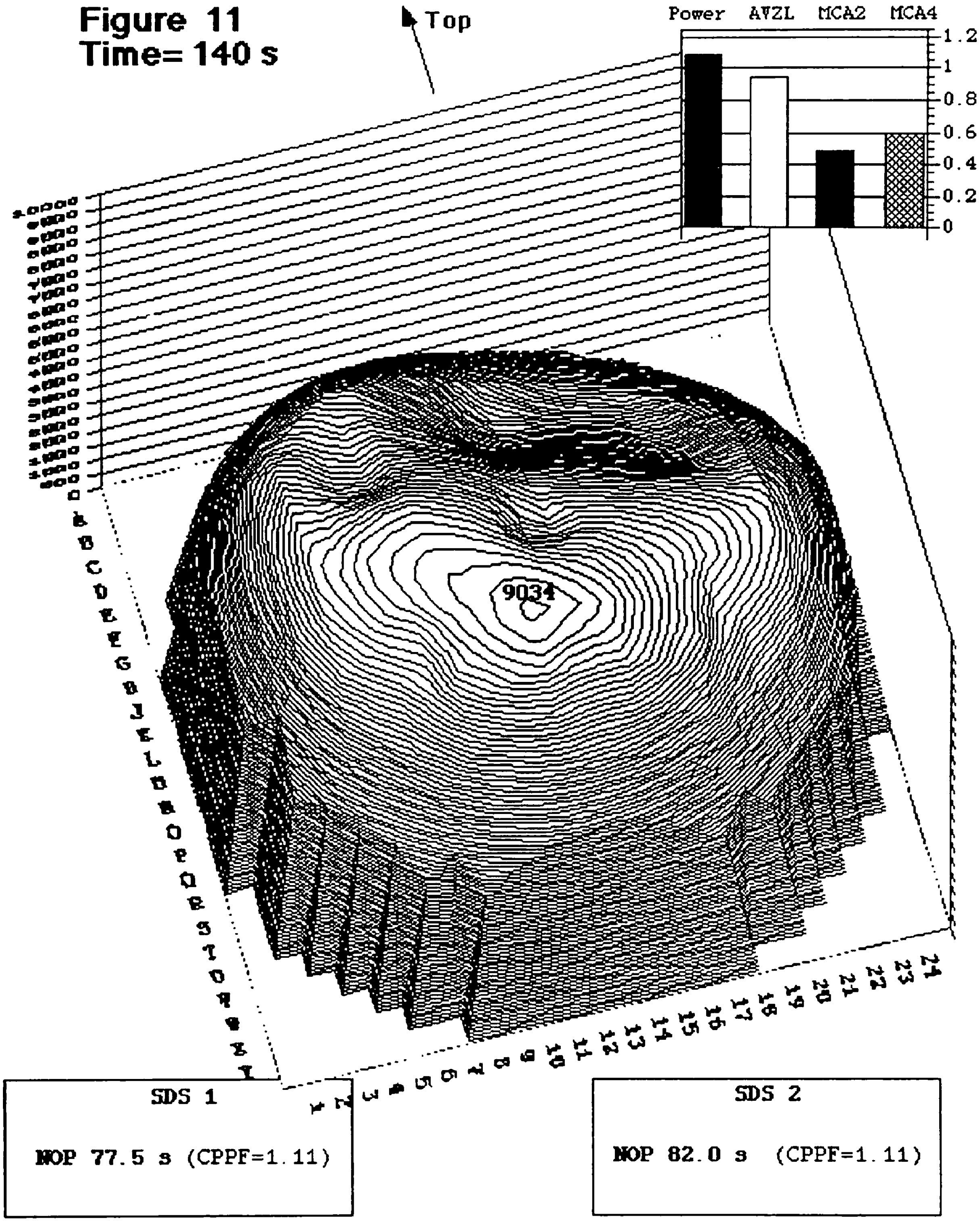


Bruce B 225 kg/s PT/CT, Long Shutdown, 1.0 wt% I.D.

Figure 10
Time= 80 s

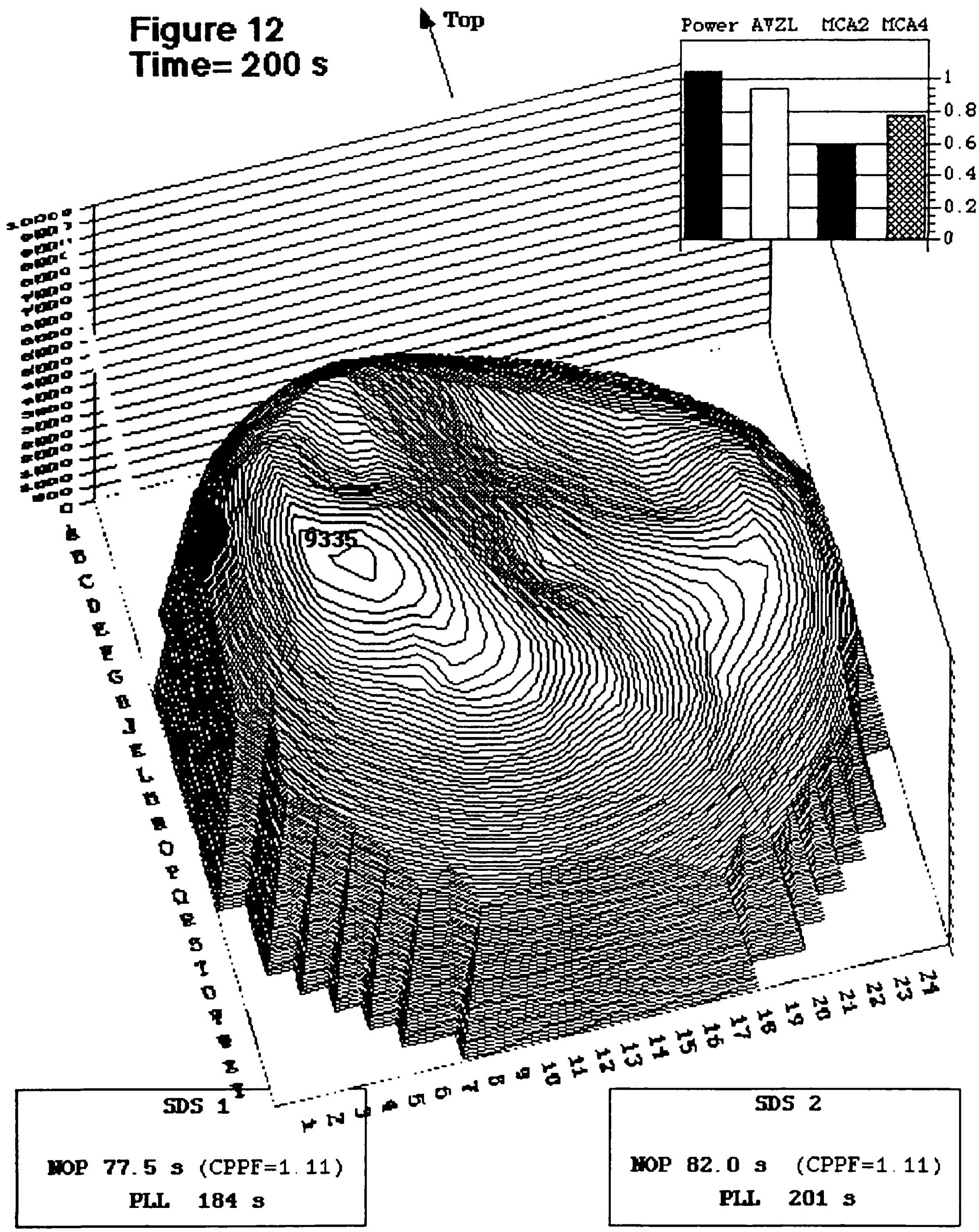


Bruce B 225 kg/s PT/CT, Long Shutdown, 1.0 wt% I.D.



Bruce B 225 kg/s PT/CT, Long Shutdown, 1.0 wt% I.D.

Figure 12
Time= 200 s



Bruce B 225 kg/s PT/CT, Long Shutdown, 1.0 wt% I.D.

REFERENCES

1.. STAFF OF THE NUCLEAR SAFETY DEPARTMENT, "Bruce NGS B Safety Report Section 3.6, Small Break Loss of Coolant Accident in the Heat Transport System, Chapter 5, Pressure Tube/Calandria Tube Failure", Ontario Hydro, June 1991. Various sections.

2.. Kundurpi, P.S. et al, "Consequences of Pressure/Calandria Tube Failure in a CANDU Reactor Core During Full-Power Operation", Nuclear Safety Department Report Number 84301, Oct 1984.

3.. Gold, M.B. and F.B. Tran, "SMOKIN - A Family of Codes for Reactor Space-Time Neutronics Calculations Based on Modal Kinetics - User's Manual", Nuclear Safety Department Report Number 87092, December 1987.

4.. Chang, Y.F., "A Thermal Hydraulic System Simulation Model for the Reactor Boiler and Heat Transport System (SOPHT)", Ontario Hydro Report Number CNS-37-2, Sept 1977.

5.. Riffert, K. and K.K. Fung, "SOMASS 2.0 Simulation of Moderator and Shield Systems Computer Code Description and Verification", Nuclear Safety Department Report Number 89401, Sept 1989.

6.. Carruthers, E.V. "Generic Methodology for Reactor Physics Analysis of Small Loss of Coolant Accidents", Nuclear Safety Department Report Number 90129 Rev 1, Oct 1991.

7.. Wight, A.L., "Ontario Hydro Reactor Fuelling Simulation Program - OHRFSP", Nuclear Safety Department Report Number 86000, January 1987.