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Advanced Neutron Source Reactor (ANSR) Phenomena Identification and Ranking (PIR) for Large Break Loss of Coolant Accidents (LBLOCA)

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ADVANCED NEUTRON SOURCE REACTOR (ANSR) PHENOMENA IDENTIFICATION AND RANKING (PIR) FOR LARGE BREAK LOSS OF COOLANT ACCIDENTS (LBLOCA)

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ABSTRACT

A team of experts in reactor analysis conducted a phenomena identification and ranking (PIR) exercise for a large break loss-of-coolant accident (LBLOCA) in the Advanced Neutron Source Reactor (ANSR). The LBLOCA transient is broken into two separate parts for the PIR exercise. The first part considers the initial depressurization of the system that follows the opening of the break. The second part of the transient includes long-term decay heat removal after the reactor is shut down and the system is depressurized. A PIR is developed for each part of the LBLOCA. The ranking results are reviewed to establish if models in the RELAP5-MOD3 thermalhydraulic code are adequate for use in ANSR LBLOCA simulations. Deficiencies in the RELAP5-MOD3 code are identified and existing data or models are recommended to improve the code for this application. Experiments were also suggested to establish models for situations judged to be beyond current knowledge. The applicability of the ANSR PIR results is reviewed for the entire set of transients important to the ANSR safety analysis.

ACRONYMS AND ABBREVIATIONS

1D	One-dimensional
2D	Two-dimensional
A	Accumulator
APT	Accumulator and pump test
AP	Applicable
AHP	Analytical heirachy process
ACSL	Advanced continuous simulation package
ANSR	Advanced neutron source reactor
BASIN	Cooling tower basin
BR	Break
B	Bypass
CL	Cold leg
CPBT	Core pressure boundary tube
CR	Control rods
CSAU	Code scaling, applicability, and uncertainty
D2O	Heavy water
DNB	Departure from nucleate boiling
EHX	Emergency heat exchanger
EMER	Emergency
FBE	Flow blockage experiment
FA	Fuel assembly
FF	Friction factor
G	Mass flow
HL	Hot leg
HT	Heat transfer
HX	Heat exchanger
LBLOCA	Large break loss-of-coolant accident
NCT	Natural circulation test
NA	Not applicable
ONVG	Onset of net vapor generation
P	Pump
PA	Partially applicable
PB	Pipe break
PHX	Primary heat exchanger
PIR	Phenomena identification and ranking
POOL	Reactor pool
PR	Pressurizer
R	Reflector tank
S	Secondary side
T	Target
THTL	Thermal hydraulic test loop
t	Time

ADVANCED NEUTRON SOURCE REACTOR (ANSR) PHENOMENA IDENTIFICATION AND RANKING (PIR) FOR LARGE BREAK LOSS OF COOLANT ACCIDENTS (LBLOCA)

1. INTRODUCTION

The Advanced Neutron Source Reactor (ANSR) is a research facility planned by Oak Ridge National Laboratory (ORNL) to meet the need for an intense steady state source of neutrons (C. D. West, 1988). The ANSR is being designed for condensed matter physics, isotope production and fundamental physics research. The design effort is in the conceptual phase. Analysis tools have been developed to support the design process and to produce the simulations required for the Safety Analysis Report (SAR). These tools will require verification and validation.

The RELAP5-MOD3 code (Carlson et. al., 1990) and an Advanced Continuous Simulation Package (ACSL) based lumped parameter code (Chen et. al., 1993) are presently used for most of the safety and design evaluations. Many of the models incorporated in these codes are known to be deficient when applied to the thermalhydraulic conditions of the ANSR. It is expected that several experiments will be needed to produce data to support development of credible models. However, the selection and prioritization of the experiments requires that some measure of benefit be established to weigh against the cost. A Phenomena Identification and Ranking (PIR) exercise was conducted for a LBLOCA transient in the ANSR in order to identify the phenomena for which credible models are most needed.

A Team of individuals knowledgeable in reactor thermalhydraulics was assembled by the Advanced Neutron Source Reactor program for a series of four two day meetings. The team members were selected based on their familiarity with research and production reactor systems and the PIR process. The team members and a brief description of their experience relative to the PIR task for the ANSR follows:

Lap Y. Cheng of the Brookhaven National Laboratory. Lap is involved with the safety analysis of the High Flux Beam Reactor (HFBR) including planning of experiments to support these analyses.

Richard Dimenna of the Savannah River Technology Center. Richard was involved in the development of the RELAP5-MOD2 Code and has performed safety analyses for production reactors (e.g., K-Reactor restart) at Savannah River. Richard Dimenna is one of the developers of the Code Scaling, Applicability, and Uncertainty (CSAU) process from which the PIR process was derived, Technical Program Group, 1989.

Peter Griffith of MIT is internationally recognized as a researcher in two-phase flow and thermalhydraulics. He is also familiar with safety issues in reactor systems and is a very experienced experimentalist in thermalhydraulics.

Art Ruggles of the Oak Ridge National Laboratory is familiar with the safety analysis of the ANSR and is responsible for verification and validation of the thermalhydraulic analysis tools to be used in these analyses. Art is also familiar with the RELAP5-MOD3 thermalhydraulic simulation code.

Gary Wilson of the Idaho National Engineering Laboratory is also one of the developers of the CSAU process from which the PIR process was derived. Gary is familiar with safety issues in conventional reactor systems and with the history of Nuclear Reactor Safety Regulations developed by the United States Nuclear Regulatory Commission (USNRC).

The ANSR PIR team accomplished the following tasks:

(a) Ranked thermalhydraulic phenomena according to their importance during safety related transients. The LBLOCA was the transient considered explicitly in this PIR process. However, other transients and segments of transients were identified as subsets to the LBLOCA response, and are covered by the results of this evaluation.

(b) Established if the important phenomena are (or can be) adequately modeled in RELAP5-MOD3.

(c) Established which of the important phenomena require experimental investigation before reliable models can be formulated.

The configuration of the ANSR was fixed for the purposes of the ANSR PIR Team as the preconceptual design configuration at the time of the first ANSR PIR Team meeting on September 17 and 18, 1991. Appendix A contains the description of the ANSR configuration considered by the PIR Team. Additional material was provided to the ANSR PIR team prior to the first team meeting to improve their familiarity with the ANSR. This material included the Advanced Neutron Source Project Report from February 1991 (ORNL-6656), Analysis of Loss of Coolant Accidents in the Advanced Neutron Source (EGG-EAST-8700, Fletcher and Ghan), and (Reactor Design of the Advanced Neutron Source, Nuclear Technology, Vol. 93, March 1991, Ryscamp, Selby, and Primm). General suggestions and comments from the PIR Team during presentation of the ANSR design are given in Appendix B.

The ANSR PIR Team examined a LBLOCA for the ANSR. The ANSR PIR Team decided that the LBLOCA transient was best considered in two parts. The first part of the LBLOCA involves the initial system depressurization and reactor shutdown. Simulations of the performance of the ANSR during this part of the LBLOCA were run on the RELAP5-MOD2.5 code. The second part of the LBLOCA transient includes the long term decay heat removal occurring subsequent to the reactor shutdown. Simulation of this part of the transient was performed using the Advanced Computer Simulation Language (ACSL) based lumped parameter model developed to support the design effort of the ANSR. These transient simulations were presented to the ANSR PIR team prior to performing the PIR.

2. THE PIR PROCESS

The phenomena identification and ranking (PIR) process was adapted from the PIR process incorporated with the Code Scaling, Applicability, and Uncertainty (CSAU) evaluation methodology developed by the USNRC. A brief description of the PIR process employed by the ANSR PIR Team is included here since the objectives of this PIR are slightly different from those that exist during a CSAU evaluation.

The PIR process is focused on a specific transient occurring in a specific piece of hardware. The PIR results presented herein were developed with a specific transient, the LBLOCA, in mind. The PIR process results in a ranking of components and a ranking of phenomena. The ranking of the components and phenomena represents their importance to the outcome of the transient.

The PIR process begins with the specification of the specific system and the specific transient to be considered. In this PIR the transient was a LBLOCA occurring near the fuel assembly inlet in the core pressure boundary tube (CPBT). The system configuration is given in Appendix A. The transient to be considered is normally simulated in some manner to facilitate discussion of the important components and phenomena. The simulations used as the basis for these discussions were developed using RELAP5-MOD2.5 and an ACSL based lumped parameter (i.e., lumped components) model. The simulation results are used as a guide to the ranking of components and phenomena. Care is taken to base the results of the PIR on real physics, not artifacts associated with the particular simulation tool used for the "strawman" simulation.

A clear definition of what must be simulated (i.e., the so-called critical indicator) is required before the actual PIR process can begin. It is desired to predict the incipience of fuel damage during the early part of the LBLOCA. The critical indicator for the early part of the LBLOCA was established as the fuel centerline temperature since this was judged to be the single most important indicator of fuel integrity. The onset of net vapor generation (ONVG) was the critical indicator used for the PIR of decay heat removal.

The components important to the transient simulation are listed to begin the PIR. These components are then ranked according to their importance to the transient behavior. The ranking is accomplished by forming a matrix of component versus component as shown in Fig. 2.1. Each component is then pairwise ranked against all the other components in the system according to a ranking scheme indicated in Table 2.1. Intermediate integer numbers were used in the ranking scheme when the team was not able to agree on a specific level. The ranking procedure compares matrix row components to matrix column components such that if the row component is much more important than the column, a five is entered for that matrix position.

		1.	2.	3.	4.	5.	6.	7.	8.	9.	10.	11.	12.	13.	14.
Fuel assy	1.	1	3	5	5	5	5	5	5	5	3	5	5	1	5
Control rods	2.		1	3	5	5	5	3	5	5	1	5	5	-3	5
Targets	3.			1	2	1	3	1	5	3	-5	3	5	-5	5
CPBT	4.				1	1	3	1	5	3	-5	3	5	-5	5
Bypass	5.					1	1	1	5	3	-5	1	3	-5	5
Cold leg	6.						1	-3	5	2	-5	1	3	-5	5
Pumps	7.							1	5	3	-3	3	5	-5	5
Emer. HX	8.								1	-5	-5	-3	1	-5	1
Primary HX	9.									1	-5	-3	3	-5	5
Accumulator	10.										1	5	5	-3	5
Hot leg	11.											1	3	-5	5
Pressurizer	12.												1	-5	5
Break	13.													1	5
Secondary	14.														1

Figure 2.1: Component Ranking Matrix for LBLOCA

Table 2.1. Pairwise ranking scheme

5	Much more important
3	More important
1	Equally important
-3	Less important
-5	Much less important

The ranking process is easily organized in the matrix format shown in Fig. 2.1. Specific definitions for some components are given in Appendix C to avoid ambiguity in interpreting the ranking. The pairwise ranking matrix is used to produce a composite ranking via an Analytical Hierarchy Process (AHP), (Saaty, 1977, 1988). Note that the various pairwise rankings that combine to produce the composite rank may not be perfectly consistent. A measure of the consistency of the pairwise rankings with the composite ranking is provided by the AHP.

The process continues with the identification of the phenomena important to the simulation. These phenomena are identified by component and are pairwise ranked for each component. Figure 2.2 shows the phenomena ranking matrix for the fuel assembly during LBLOCA. Definitions for some phenomena are given in Appendix C. The phenomena are composite ranked by AHP with consideration of the rank of the phenomena within the component along with the composite rank of the component.

The composite rankings of components and phenomena are reviewed by the PIR team for sensibility and consistency with expectations. The composite rankings are then accepted by the team. No significant surprises or inconsistencies with expectations were encountered in this PIR development. However, the high ranking of the break performance did motivate some discussion regarding the credibility of large breaks in the primary loop.

The PIR methodology for establishing the composite rankings is structured such that the team members invest in the decision making process in a continuous and manageable way (i.e., via numerous pairwise rankings). The process is very effective in establishing a consensus from a group of technical experts.

		1.	2.	3.	4.	5.	6.	7.	8.	9.	10.	11.	12.	13.	14.	15.	16.	17.
1 phase HT	1.	1	-3	-3	-5	1	-3	2	5	-5	-3	5	1	1	1	1	-3	3
1 phase FF	2.		1	4	-3	3	-3	3	5	-5	1	-5	1	1	1	1	-3	3
Form factor	3.			1	-3	3	-2	3	5	-5	1	-5	1	-2	3	1	-3	3
Power dist	4.				1	5	3	5	5	1	2	1	2	3	5	4	3	3
Oxide growth	5.					1	-3	1	3	-5	-3	-5	-3	1	2	-3	-3	2
Plate spacing	6.						1	3	5	-3	3	-3	2	3	3	-3	1	3
2D conduction	7.							1	3	-3	-3	-5	-3	-2	1	-5	-2	1
D2O prop	8.								1	-5	-5	-5	-5	-5	-3	3	-5	-3
Vapor gen	9.									1	2	1	3	5	5	1	3	5
2 phase resist	10.										1	-3	1	3	3	3	1	3
DNB	11.											1	3	5	5	-2	3	5
2D flow	12.												1	2	2	-3	1	3
Crit. flow	13.													1	-2	2	-3	2
Thermal strain	14.														1	1	1	2
Hydraulic loads	15.																-3	3
Inlet cond.	16.																1	3
2 phase HT	17.																	1

Figure 2.2: Phenomena Ranking Matrix for the Fuel Assembly During LBLOCA

3. CRITICAL INDICATORS DURING THE EARLY PART OF THE LBLOCA

The incipience of fuel damage is the event to be predicted during the early portion of the LBLOCA. Presentations were given to the ANSR PIR Team evaluating fuel damage mechanisms additional to melting. These presentations included a discussion of outgassing of the fuel at high temperatures and resultant blistering of the clad away from the fuel meat. Data were presented that indicate out-gassing becomes a problem when the fuel temperature exceeds 350°C for extended periods. Unfortunately, the data available did not establish if significant outgassing will occur if these temperature limits are exceeded for short periods of time during transients. The margin to flow induced (hydraulic) buckling of the ANSR fuel plates during normal operation is large. However, hydraulic buckling of the ANSR fuel plates is possible if the coolant velocity increases or if the material properties or dimensions of the fuel assembly are altered during a transient. The presentations concerning fuel plate stability in the flow field are included in Appendix D.

It was decided that the incipience of fuel damage was best represented by the fuel centerline temperature for the purposes of the PIR. A fuel plate centerline temperature of 500°C was used as the thermalhydraulic parameter associated with incipient fuel damage during the early part of the large break LOCA. This temperature may be somewhat higher than what can actually be allowed. However, the phenomena that influence the fuel plate temperature and their associated ranks remain applicable. Control rod cooling and target cooling were also considered since failure of these components during an accident could lead to damaged fuel.

4. CRITICAL INDICATOR DURING DECAY HEAT REMOVAL

The evaluation of decay heat removal for the PIR focused on the performance of the reactor after the early part of a LBLOCA. The simulation of this event provided to the ANSR PIR Team assumed that all the pony motors had failed. The team suggested that this would be a very low probability event and that excluding the pony motor function from the transient may indicate either that the pony motors are not reliable or that they are not necessary. It was assumed that the pony motors would run for ~30 min after the LBLOCA to make the transient evaluated by the team more credible and general. The secondary side of the system was assumed to be intact. Fuel damage was not considered as the limiting criterion. It is intended that the reactor accomplish decay heat removal after the pony motor shutdown via single phase natural circulation. Credible analysis techniques are needed to insure the reactor is designed such that decay heat can be removed in this fashion. The analysis tools must therefore predict local fluid temperature and pressures accurately. The limiting thermalhydraulic criterion for this portion of the transient simulation was taken as the onset of net vapor generation (ONVG).

5. PIR DEVELOPMENT FOR LBLOCA

The early part of the LBLOCA was simulated using RELAP5-MOD2.5. The model and results for the simulation of a 152 mm break in the CPBT at the core inlet are given in Appendix E. Suggestions and comments from the ANSR PIR team during the presentation of the LBLOCA simulation results are given in Appendix F. The component and phenomena lists were selected for the early part of a general LBLOCA when possible, with specific consideration made for the CPBT break at the Fuel Assembly inlet when necessary. Only the initial depressurization and reactor shut-down was considered in the early part of the LBLOCA. The components and associated phenomena selected by the ANSR PIR Team are given in Table 5.1. Definitions for some of the components and phenomena are given in Appendix C where clarification seemed necessary.

Table 5.1. Components and phenomena considered for the early part of the LBLOCA

---

1. Fuel Assembly (FA)
   - A. Single phase heat transfer
   - B. Single phase friction factor
   - C. Form factors (local pressure loss coefficients)
   - D. Temporal and spatial power density
     - a. flux profile and total power
     - b. manufacturing defects (hot spots)
     - c. decay heat and scram power versus time
   - E. Oxide growth and spallation
   - F. Plate spacing variations
   - G. Conduction in fuel (2-D or 3-D needed)
   - H. Heavy water properties
   - I. Vapor generation
     - a. Incipience of boiling
     - b. Onset of net vapor generation (ONVG)
     - c. Interfacial terms (area, heat, mass, and momentum)
   - J. Two phase pressure drop
     - a. due to momentum flux (related to vapor generation)
     - b. viscous losses
   - K. Two phase heat transfer
   - L. Departure from nucleate boiling (DNB)
   - M. Multidimensional flow (2-D or 3-D)
   - N. Critical flow
   - O. Thermal strain
   - P. Hydraulic loads
   - Q. Inlet temperature and velocity distribution

2. Control Rods (CR)
   - A. Single phase heat transfer
   - B. Two phase heat transfer
   - C. Flow resistance

D. Heat Load (Note that the control rod worth vs. time was initially considered as a separate phenomenon but was later dropped. This is considered a boundary condition that must be known to describe the problem. The heat load as a function of time for the individual components includes the rod worth vs. time and associated physics calculations. Note also that the heat transfer phenomena associated with the control rods are potentially an issue to fuel damage under LBLOCA conditions. Control rod damage is considered unacceptable in most situations. The ranking of phenomena was evaluated with and without the control rod heat transfer and heat load items. The team elected to keep the control rod heat transfer items in the final results).

3. Targets (note the targets include the transuranium production rods) (T)
   A. Single phase heat transfer
   B. Two phase heat transfer
   C. Flow resistance
   D. Heat load
   E. Multi-Dimensional conduction

(Note that the phenomena listed for the targets were developed with the perspective that a damaged target may lodge on the fuel assembly inlet and cause fuel damage.)

4. Core Pressure Boundary Tube (CPBT)
   A. Single phase heat transfer
   B. Two phase heat transfer
   C. Flow resistance
   D. Heat load

5. Bypass (Note the bypass was defined as the flow passage around the outside of the outer fuel annulus. The objective of the Team was to account for and properly model all the primary flow within the CPBT.) (B)
   A. Flow resistance

6. Hot Leg Pipes (HL)
   A. Heat transfer
   B. Flow resistance

(Note that sound speed was included in the original phenomena listing for the hot leg, but was dropped during the ranking process.)

7. Pressurization System (PR)
   A. Mass flow into primary versus time

8. Break (BR)
   A. Break area versus time
   B. Mass flux versus time
   C. Shape versus time

(Note that break location was included in the original phenomena listing for the break, but was later dropped.)

9. Accumulator (A)
   A. Gas process line
   B. Gas evolution

(Note that inventory was originally included as a phenomena, but was later dropped. It was decided that the original inventory of the accumulators is part of the problem boundary conditions.

Attention was drawn to the accumulators and the need to be careful in the control of their liquid levels in the first ANSR PIR Team meeting.)

10. Cold Leg Pipes (CL)
    A. Flow resistance
    B. Heat transfer

(Note that sound speed was originally included in the phenomena listing for the cold leg, but was dropped during the phenomena ranking process.)

11. Primary Heat Exchanger (PHX)
    A. Heat transfer
    B. Flow resistance

12. Emergency Heat Exchanger (EHX)
    A. Flow resistance
    B. Heat transfer

13. Pump (P)
    A. Performance
    B. Trip time

14. Secondary Side (Note it was felt that specific components on the secondary side do not significantly influence the early part of a LBLOCA). (S)

The components were ranked for the LBLOCA and are given in Table 5.2. The phenomena were ranked and are given in Table 5.3. The five tiered ranking system was used for the initial pairwise ranking by the PIR team as discussed in the section entitled *The PIR Process*. However, the AHP software converts the pairwise input ranks to composite output ranks on a scale of 1 to 9, with 9 as most important and 1 as least important, as shown in Tables 5.2 and 5.3.

Table 5.2. The component rankings for LBLOCA

Composite ranking from AHP	Priority
Break (BR)	(9)
Fuel assembly (FA)	(9)
Accumulator (A)	(6)
Control rods (CR)	(6)
Targets (T)	(3)
Pumps (P)	(3)
CPBT	(3)
Bypass (B)	(2)
Cold leg (CL)	(2)
Hot leg (HL)	(2)
Primary heat exchanger (PHX)	(2)
Pressurizer (PR)	(1)
Emergency heat exchanger (EHX)	(1)
Secondary side (S)	(1)

A sensitivity study of the PIR Process published by Gary Wilson, 1992, quantifies the variation in the composite rankings relative to plausible uncertainties in the pairwise rankings. This study is based on the results of this PIR effort for the ANSR and is helpful in interpreting the significance of the composite rankings.

The composite ranking of all the phenomena under all the components is given in Appendix G. Comments noted during the pairwise ranking of components and phenomena are included in Appendix H.

Table 5.3. The phenomena rankings for LBLOCA

Composite ranking from AHP	Priority
Area vs. time (BR)	(9)
Vapor generation (FA)	(9)
DNB (FA)	(8)
Power density (FA)	(8)
Gas process line (A)	(7)
Two phase heat transfer (CR)	(6)
Heat load (CR)	(6)
Plate spacing (FA)	(5)
Inlet temperature & velocity dist. (FA)	(5)
Single phase heat transfer (FA)	(4)
Mass flux vs. time (BR)	(4)
Two phase pressure drop (FA)	(4)
Flow resistance (CR)	(4)
Single phase friction factor (FA)	(4)
Multi-dimensional flow (FA)	(4)
Form factor (FA)	(4)
Hydraulic loads (FA)	(4)
Heat load (T)	(3)
Head vs. mass flow and time (P)	(3)
Heat load (CPBT)	(3)
Thermal strain (FA)	(3)
Flow resistance (B)	(3)
Gas evolution (A)	(3)
Critical flow (FA)	(3)
Shape vs. time (BR)	(2)
Flow resistance (CL)	(2)
Flow resistance (HL)	(2)
Two phase heat transfer (FA)	(2)
Oxide growth & spallation (FA)	(2)
Flow resistance (T)	(2)
2D conduction (FA)	(2)
Flow resistance (PHX)	(2)
Two phase heat transfer (T)	(2)
Flow resistance (CPBT)	(2)
Single phase heat transfer (CR)	(2)
Mass flow into primary (PR)	(2)
Trip time (P)	(2)
Flow resistance (EHX)	(1)
D2O properties (FA)	(1)
Conduction (T)	(1)
Secondary side	(1)
Single phase heat transfer (CPBT)	(1)
Heat transfer (CL)	(1)
Heat transfer (HL)	(1)
Single phase heat transfer (T)	(1)
Heat transfer (PHX)	(1)
Two phase heat transfer (CPBT)	(1)
Heat transfer (EHX)	(1)

6. PIR DEVELOPMENT FOR DECAY HEAT REMOVAL

The results of the ACSL based simulation of the ANSR during the late part of the LBLOCA (i.e., decay heat removal) are given in Appendix I. This simulation was reviewed by the PIR Team prior to conducting the PIR process for decay heat removal. The ANSR PIR Team ranked phenomena and components according to their importance in predicting bulk fluid temperatures and pressures during the decay heat removal portion of the transient. It was agreed that air ingestion would not be possible based on the current design philosophy for the accumulators, submerged primary pipes, and limited volume cells. It was also agreed that any gas that might evolve after depressurization would end up in the top of the shell side of the primary heat exchanger (i.e., the highest point in the primary loop) and therefore not create a so-called loop-seal to natural circulation. It was assumed that the primary pipes would not be insulated.

The components and associated phenomena considered important during decay heat removal are given in Table 6.1. Definitions are given for some of the components and phenomena in Appendix C where clarification seemed necessary.

Table 6.1. Components and phenomena associated with decay heat removal

---

1. Fuel Assembly (FA)
   A. Power versus time
   B. Flow resistance
   C. Power versus position
   D. Parallel channel effects
   E. Inlet velocity and temperature

2. Control Rods (CR)
   A. Heat load in control rods
   B. Flow resistance

3. Targets (includes the transuranium production rods) (T)
   A. Heat load in targets
   B. Flow resistance

4. Core Pressure Boundary Tube (CPBT)
   A. Heat load in the CPBT
   B. Heat transfer to reflector

5. Bypass (defined as the flow passage around the outside of the outer fuel annulus. The objective of the Team was to account for and properly model all the primary flow within the CPBT.) (B)
   A. Flow resistance

6. Hot Leg Pipes (HL)
   A. Heat transfer
   B. Flow resistance
   C. Stratification

7. Pressurization System (PR)
   A. Mass flow into primary versus time

Table 6.1 (continued)

8. Break (BR)
   A. Location

9. Cold Leg Pipes (CL)
   A. Heat transfer
   B. Flow resistance
   C. Stratification (all pipes should be sloped 1/2" to the foot)

10. Primary Heat Exchanger (PHX)
    A. Heat transfer to secondary
    B. Heat transfer to pool
    C. Flow resistance

11. Emergency Heat Exchanger (EHX)
    A. Flow resistance
    B. Heat transfer

12. Pump (P)
    A. Performance (Head or resistance versus time and/or flow)

13. Reactor pool (RP)
    A. Stratification
    B. Initial temperature

14. Hot and cold leg of secondary side (SHL & SCL)
    A. Flow resistance
    B. Heat transfer to air

15. Cooling tower basin (CTB)
    A. Inventory
    B. Initial temperature
    C. Stratification
    D. Heat transfer to air

16. Reflector tank (RT)
    A. Heat load
    B. Initial temperature

The components were ranked for decay heat removal simulation and are given in Table 6.2. The phenomena are ranked for decay heat removal in Table 6.3. The five tiered ranking system was used for the initial pairwise ranking by the PIR team as discussed in the section entitled *The PIR Process.* However, the AHP software converts the pairwise input ranks to composite output ranks on a scale of 1 to 9 with high ranked as 9, and low ranked as 1, as shown in Tables 6.2 and 6.3.

Table 6.2. Component ranking for decay heat removal

Composite ranking from AHP	Priority
Fuel assembly (FA)	(9)
Primary heat exchanger (PHX)	(7)
Cooling tower basin (CTB)	(5)
Pumps (P)	(5)
Hot leg (HL)	(3)
Cold leg (CL)	(3)
Reactor pool (RP)	(3)
Emergency heat exchanger (EHX)	(2)
Control rods (CR)	(2)
CPBT	(2)
Hot & cold legs of secondary (SHL & SCL)	(2)
Bypass (B)	(2)
Reflector tank (RT)	(1)
Targets (T)	(1)
Pressurizer (PR)	(1)
Break (BR)	(1)

Table 6.3. Phenomena ranking for decay heat removal

Composite ranking from AHP	Priority
Power vs. time (FA)	(9)
Flow resistance (FA)	(8)
Heat transfer to secondary (PHX)	(7)
Heat transfer to air (CTB)	(6)
Head vs. mass flow and time (P)	(5)
Initial temperature (CTB)	(5)
Stratification (CTB)	(5)
Heat transfer (HL)	(4)
Heat transfer (CL)	(4)
Power density (FA)	(4)
Initial temperature (RP)	(3)
Inventory (CTB)	(3)
Heat transfer (EHX)	(3)
Heat transfer to pool (PHX)	(3)
Flow resistance (CR)	(3)
Parallel channel effects (FA)	(2)
Stratification (HL)	(2)
Heat load (CPBT)	(2)
Flow resistance (PHX)	(2)
Flow resistance (SHL & SCL)	(2)
Flow resistance (B)	(2)
Stratification (RP)	(2)
Inlet velocity and temperature (FA)	(2)
Initial temperature (RT)	(2)
Flow resistance (CL)	(2)
Stratification (CL)	(2)
Heat load (T)	(2)
Mass flow vs. time (PR)	(2)
Location (BR)	(2)
Flow resistance (HL)	(2)
Flow resistance (EHX)	(1)
Heat load (CR)	(1)
Heat vs. time (RT)	(1)
Heat transfer to RT (CPBT)	(1)
Heat transfer to air (SHL & SCL)	(1)
Flow resistance (T)	(1)

The input to AHP and the ranking output from AHP is given in Appendix J. Comments and discussion from ANSR PIR Team members during the ranking process for decay heat removal are given in Appendix K.

7. APPLICABILITY OF THE ANSR PIR TEAM RESULTS

A list of potential transients prepared by Mike Harrington, the ANSR Safety Analysis Manager, was reviewed by the ANSR PIR team to establish if the PIR results for the LBLOCA and decay heat removal were applicable. The results of the ANSR PIR were judged to be either Applicable (AP); Partially Applicable (PA); or Not Applicable (NA). The listing of transients considered and the judgment of the ANSR PIR Team concerning the applicability of the PIR results are given in Appendix L. The transients judged to be applicable or partially applicable are listed in Table 7.1.

Table 7.1. Transients for which PIR results apply

Items (1) through (4) were covered as part of the decay heat removal PIR so long as the system remains single phase:

(1) Reactor Natural Circulation Cooling Test
(2) Loss of offsite power
(3) Station blackout
(4) Loss of all non-1E power

Loss of Coolant Pressure Control (Pressure Decrease)

Items (5) through (8) look like LOCA events from the perspective of thermalhydraulics. Those events that cause a rapid depressurization of the facility are well represented by the large break LOCA PIR. All those events that cause an early reactor trip and pump trip will be covered in their later phases by the long term decay heat removal PIR.

(5) One letdown valve goes fully open
(6) All letdown valves go fully open
(7) Pressurizer pump shutdown
(8) Overpressure relief valve fails fully open

Loss of Primary Coolant Flow

(9) All pumps coast down to natural circulation flow (Pony motors fail)

The long term decay heat removal PIR will apply to the simulation after the scram.

Loss of Primary Coolant

The large break LOCA was exactly the transient considered during the ANSR PIR Team meetings. The long term decay heat removal portion of the large break will apply to the medium break, and to the small break after scram and pump trip has occurred.

(10) Small break (no immediate scram)
(11) Medium break
(12) Large break

Table 7.1. (continued)

External events

(13) Tornado
(14) Seismic
(15) Coolant off-gas as a result of primary coolant depressurization

Items (13) and (14) may initiate a power interruption similar to Items (2), (3) or (4). Item (14) may initiate a system break as covered in items (10), (11), and (12). Item (15) is actually a phenomenon associated with system depressurization and was included in the ANSR PIR Team evaluation of the large break LOCA.

8. RELAP5-MOD3 APPLICABILITY AND EXPERIMENTS NEEDED TO SUPPORT ANSR SAFETY ANALYSIS

Highly ranked phenomena were reviewed individually to determine if there was a need for experiments to establish models for their behavior. The lower ranked phenomena (i.e., priority 4 or less on the AHP generated Phenomena Rankings) were simply categorized by the team in Table 8.1 as either being (A) adequately modeled by RELAP5-MOD3, (B) possibly well modeled by RELAP5-MOD3 (i.e., needs a careful evaluation), (C) not well modeled by RELAP5-MOD3, or (D) not included in RELAP5-MOD3. The category of "possibly well modeled" is a result of limitations in the knowledge of RELAP5-MOD3 within the ANSR PIR team. The "I" preceding some of the categories in Table 8.1 indicates RELAP5-MOD3 can model these phenomena via input.

Discussion of experiments began with some recommendations concerning fuel plate buckling tests. Tests are underway and planned to establish the circumstances when plate buckling occurs in the fuel assembly. Peter Griffith cited experience he had with modeling natural frequencies of boiler tubes in crossflow. He found that an assumption of a non moment bearing connection was appropriate where the boiler tubes met the tube sheet. This assumption was appropriate even though the tube sheet was massive and the tubes were close fitting and welded into the sheet. Apparently groups of tubes act together to cause significant deflections in the tube sheet. A similar effect seems likely in the ANSR fuel assembly where the fuel plates attach to the inner and outer barrels. The team recommends that the boundary conditions imposed on the simulated plates in the buckling tests be consistent with those in the reactor in order to capture these effects.

The remainder of the discussion of phenomena were organized according to the ranking of the phenomena. The highly ranked phenomena associated with the early part of the LBLOCA were considered first.

Area versus time for the break: The team felt that the modeling of the mass flow versus time is reasonably well understood. Therefore the actual break opening mechanism is the topic needing attention. A best estimate of the break opening is needed to be consistent with the overall philosophy of transient simulations for safety analysis. This would involve a burst test of a ductile pipe with a realistic initial flaw. The size and nature of the initial flaw should be consistent with the "best estimate" philosophy incorporated in the remainder of the thermalhydraulic simulation. An experiment establishing credible break opening dynamics is essential to the fidelity of simulation in the early part of the LBLOCA.

Vapor Generation: The Saha-Zuber model for ONVG is employed in RELAP5-MOD3. An experiment is needed to establish a model for ONVG in the fueled region due to the unusually high mass flux, heat flux, and subcooling. Most of the data in the literature establishing ONVG were developed from heat flux transients. The ANSR experiences a pressure transient during the early part of the LBLOCA. An experiment evaluating the performance of the system during a pressure transient was suggested. These comments also apply to vapor generation subsequent to ONVG. A "tube in glass" experiment was suggested as a way to get vapor generation data directly.

Note that the phenomena listing includes vapor generation and two-phase flow resistance separately. The ANSR PIR team agreed that the rate of vapor generation controls the change in the momentum flux and associated acceleration pressure drop. The two-phase flow resistance listing only includes the viscous losses associated with a boiling two-phase flow.

Table 8.1. RELAP5-MOD3 performance relative to low ranked phenomena for the early part of the LBLOCA

RELAP5 Performance	Phenomena	Priority
A	Single phase heat transfer (FA)	(4)
A	Mass flux vs. time (BR)	(4)
A	Two phase flow resistance (FA)	(4)
A	Flow resistance (CR)	(4)
A	Single phase friction factor (FA)	(4)
B	2-D flow (FA)	(4)
I-A	Form factors (FA)	(4)
D	Hydraulic loads (FA)	(4)
I-A	Heat load (T)	(3)
A	Performance (P)	(3)
I-A	Heat load (CPBT)	(3)
D	Thermal strain (FA)	(3)
A	Flow resistance (B)	(3)
D	Gas evolution (A)	(3)
B	Critical flow (FA)	(3)
D	Shape vs. time (BR)	(2)
A	Flow resistance (CL)	(2)
A	Flow resistance (HL)	(2)
B	Two phase heat transfer (FA)	(2)
D	Oxide growth (FA)	(2)
A	Flow resistance (T)	(2)
D	2-D conduction (FA)	(2)
A	Flow resist (PHX)	(2)
B	Two phase heat transfer (T)	(2)
A	Flow resistance (CPBT)	(2)
A	Single phase heat transfer (CF)	(2)
I-A	Mass flow into primary (PR)	(2)
I-A	Trip (P)	(2)
A	Flow resistance (EHX)	(1)
A	D20 properties (FA)	(1)
A	Conduction (T)	(1)
A	Secondary	(1)
A	Single phase heat transfer (CPBT)	(1)
A	Heat transfer (CL)	(1)
A	Heat transfer (HL)	(1)
A	Single phase heat transfer (T)	(1)
A	Heat transfer (PHX)	(1)
B	Heat transfer (CPBT)	(1)
A	Heat transfer (EHX)	(1)

<u>**Legend**</u>

A: Adequately modeled by RELAP5-MOD3
B: Possibly well modeled by RELAP5-MOD3 (needs evaluation)
C: Not well modeled by RELAP5-MOD3
D: Not included in RELAP5-MOD3
I: Can be treated as input to RELAP5-MOD3

Departure from Nucleate Boiling: Departure from Nucleate Boiling (DNB) data are needed. The ANSR PIR team visited the experimental facility designed to measure DNB and pressure demand characteristics for the fuel assembly. It was noted that an understanding of the flow resistance associated with the distance from the incipience of boiling to ONVG may be important due to the very high heat flux and subcooling in the fuel cooling channels.

Peter Griffith suggested that setting the thermal limit when the wall temperature was equal to the local saturation temperature would make the thermalhydraulic modeling easy and reduce the need for experiments. He suggested examining the impact of using this limit on the performance of the reactor.

Power Distribution: The PIR team felt that the spatial power distribution in the reactor should be adequately modeled with point kinetics. It was noted that the power shape versus time in the fuel assembly can be input to RELAP5-MOD3. This has been done to modify the spatial power distribution in fuel during control rod insertion for other reactor simulations using RELAP5.

Process Line for the Accumulator: The ANSR PIR team believes the process line of the accumulator is not adequately modeled in RELAP5-MOD3. A significant amount of data exists. It was recommended that this data be reviewed and used as the basis for a simple and effective model. It was noted that a different process line may be needed for expansion than for compression (i.e., expansion condenses vapor while compression may extend vapor into the superheat region).

Two-Phase Heat Transfer in the Control Rods: The two-phase heat transfer models in RELAP5-MOD3 are expected to be adequate for the control rods (i.e., assuming the coolant flows are essentially one dimensional).

Heat Load in the Control Rods: The heat load in the control rods should be calculated using a code designed to do space-time neutronics simulations.

Fuel Coolant Gap: The coolant gap should be varied over the range of possible values. RELAP5-MOD3 should properly model the effect of these variations.

Fuel Assembly Inlet Conditions: The inlet conditions to the fuel assembly should also be handled by varying the inlet conditions to the fuel assembly over the range of possible values. RELAP5-MOD3 should properly model the effect of these variations.

The ability for RELAP5-MOD3 to model the low ranked phenomena for decay heat removal is evaluated in Table 8.2. The same evaluation indices are used in Table 8.2 as were used in Table 8.1.

Table 8.2. RELAP5-MOD3 performance relative to low ranked phenomena for decay heat removal

RELAP5 Performance	Phenomena	Priority
I-D	Initial temperature (RP)	(3)
I-D	Inventory (CTB)	(3)
A	Heat transfer (EHX)	(3)
A	Heat transfer (PHX)	(3)
I-A	Flow resistance (CR)	(3)
A	Parallel channel effects (FA)	(2)
D	Stratification (HL)	(2)
I-A	Heat load (CPBT)	(2)
A	Flow resistance (PHX)	(2)
A	Flow resistance (SHL & SCL)	(2)
A	Flow resistance (B)	(2)
D	Stratification (RP)	(2)
D	Inlet velocity and temperature distribution (FA)	(2)
I-A	Initial temperature (RT)	(2)
A	Flow resistance (CL)	(2)
D	Stratification (CL)	(2)
I-A	Heat load (T)	(2)
I-A	Mass flow into primary (PR)	(2)
I-A	Position (BR)	(2)
A	Flow resistance (HL)	(2)
A	Flow resistance (EHX)	(1)
I-A	Heat load (CR)	(1)
I-A	Heat vs. time (R)	(1)
A	Heat transfer (CPBT)	(1)
A	Heat transfer (SHL & SCL)	(1)
A	Flow resistance (T)	(1)

Legend
A: Adequately modeled by RELAP5-MOD3
B: Possibly well modeled by RELAP5-MOD3 (needs evaluation)
C: Not well modeled by RELAP5-MOD3
D: Not included in RELAP5-MOD3
I: Can be treated as input to RELAP5-MOD3

The highly ranked phenomena for decay heat removal were considered separately as follows:

Power versus Time: The power versus time during decay heat removal is input to RELAP5-MOD3 from calculations using other codes. RELAP5-MOD3 is able to handle the power versus time during decay heat removal.

Flow Resistance in the Fuel Assembly: RELAP5-MOD3 is able to handle the single phase flow resistance in the fuel assembly if some data are used to tune either the wall roughness or friction factor.

Heat Transfer to the Secondary Cooling System through the Primary Heat Exchanger: RELAP5-MOD3 will be able to model this heat transfer if it is calibrated and benchmarked with some performance data.

Heat Transfer in the Cooling Tower Basin: Data are available to build a model for RELAP5-MOD3 for heat transfer in the cooling tower basin. RELAP5-MOD3 will not currently model heat transfer between the cooling tower basin and ambient. However, the RELAP5-MOD3 can be modified to do an adequate calculation if data are available for calibration.

Head versus Mass Flow and Time for the Pump(s): RELAP5-MOD3 can simulate the pump performance if adequate pump data are available.

Initial Temperature of the Cooling Tower Basin: RELAP5-MOD3 can handle the initial temperature of the cooling tower basin as an input.

Stratification in the Cooling Tower Basin: RELAP5-MOD3 would need to be calibrated to handle stratification in the cooling tower basin. It was suggested that the cooling tower basin be designed such that stratification is not possible.

Heat Transfer in the Hot Leg: RELAP5-MOD3 is able to handle heat transfer between the hot leg and its surroundings.

Heat Transfer in the Cold Leg: RELAP5-MOD3 is able to handle heat transfer between the cold leg and its surroundings.

Power versus Position in the Fuel: RELAP5-MOD3 is able to handle this as input from physics calculations.

9. REVIEW OF EXPERIMENTS

Several experiments had been tentatively planned by the ANSR program prior to the ANSR PIR meetings. Additional experiments were suggested during the review of RELAP5-MOD3 applicability. Those experiments pertinent to the PIR results were reviewed and their benefit assessed by identifying phenomena in the phenomena rankings that the experiment would help to quantify.

Pipe Break (PB): Pressurize a pipe and cause it to break. This could be an aluminum pipe to simulate the CPBT. Design the experiment to produce best estimate results plus uncertainties. This is an important point. Structural design and experimental philosophies are very conservative. It is not sensible to use very conservative materials models and best estimate thermalhydraulic models for the same simulation.

Find or obtain data on radiation damage. This will allow the condition and the stress history of the CPBT to be accurately determined.

Thermalhydraulic Test Loop (THTL): The ANSR program constructed the THTL to establish thermal limits in the fueled region of the reactor. The THTL was near completion during the PIR meetings. The THTL was designed to establish thermal limits for steady state and safety related transient situations. The THTL facility allows the mass flow or pressure drop across the heated channel to be controlled. Some information on vapor generation (e.g., gathered indirectly from pressure drop data) and on DNB can be gathered from this facility to support RELAP5-MOD3 validation or model development. The so-called tube-in-glass experiment suggested by Peter Griffith during the discussion of the applicability of RELAP5-MOD3 (from the section entitled "RELAP5-MOD3 Applicability and Experiments Needed to Support ANSR Safety Analysis") would also supply data for building models for vapor generation.

Flow Blockage Experiment (FBE): One of the more likely events leading to fuel damage is a flow blockage of the fuel assembly inlet. An experiment has been planned by the ANSR program to examine the affect of an inlet flow blockage on the heat transfer coefficient downstream in the fuel cooling channel. The objective is to establish how large a blockage must be to initiate fuel damage. A flow strainer or some other design measure may then be fashioned to minimize the probability of an inlet blockage large enough to cause damage to the fuel. The experimental results were to be modeled using a Computational Fluid Dynamics (CFD) code to allow the extrapolation of the experimental results to related flow situations. Peter Griffith suggested that this would not work because real turbulence depends on upstream behavior while current CFD models for turbulence are local.

Natural Circulation Test (NCT): The ANSR program had planned a small scale natural circulation test to verify the performance of the passive decay heat removal systems. The results of the PIR did not strongly motivate the natural circulation test. Some concern of how the system would perform with noncondensible gas was discussed. However, the orientation of the primary heat exchangers at the high point in the system minimizes the affect of gas in the system. The PHX's are oriented on their sides with a vertical division of the shell side (i.e., the primary side) in the middle so that gas in the system will end up in the PHX but will not disrupt flow).

Accumulator and Pump Tests (APT): A controls experiment was suggested to allow the method of maintaining the level in the accumulators to be examined. This suggestion came out of the ANSR Design briefing.

A combined test of the accumulator and a pump was suggested since both the accumulator process line and the pump performance are highly ranked phenomena for the early part of the LBLOCA.

The pipe break and THTL experiments were considered relative to the phenomena ranked for the early part of the LBLOCA and for decay heat removal. The phenomena addressed by these experiments associated with the early part of the LBLOCA are given in Table 9.1. Flow resistance in the fuel assembly and the effect of inlet velocity and temperature are the only decay heat removal phenomena examined by the THTL experiment.

Table 9.1. LBLOCA phenomena addressed by experiments discussed by the ANSR PIR team

Phenomena	Experiment
Area vs. time (BR)	PB
Vapor generation (FA)	THTL
DNB	THTL
Power density (FA)	
Gas process line (A)	PB
Two phase heat transfer (CR)	
Heat load (CR)	
Plate spacing (FA)	
Inlet temperature & velocity dist. (FA)	THTL
Single phase heat transfer (FA)	THTL
Mass flux vs. time (BR)	PB
Two phase pressure drop (FA)	THTL
Flow resistance (CR)	
Single phase friction factor (FA)	THTL
Multi-dimensional flow (FA)	
Form factor (FA)	
Hydraulic loads (FA)	
Heat load (T)	
Performance (P)	
Heat load (CPBT)	
Thermal strain (FA)	
Flow resistance (B)	
Gas evolution (A)	PB
Critical flow (FA)	THTL
Shape vs. time (BR)	PB
Flow resistance (CL)	
Flow resistance (HL)	
Two phase heat transfer (FA)	THTL
Oxide growth & spallation (FA)	
Flow resistance (T)	
2D conduction (FA)	
Flow resistance (PHX)	
Two phase heat transfer (T)	
Flow resistance (CPBT)	
Single phase heat transfer (CR)	
Mass flow into primary (PR)	PB
Trip time (P)	
Flow resistance (EHX)	
D2O properties (FA)	THTL
Conduction (T)	

Table 9.1. (Continued)

Secondary side
Single phase heat transfer (CPBT)
Heat transfer (CL)
Heat transfer (HL)
Single phase heat transfer (T)
Heat transfer (PHX)
Two phase heat transfer (CPBT)
Heat transfer (EHX)

Legend
PB: Pipe break
THTL: Thermal Hydraulic Test Loop

10. CONCLUSIONS

Most of the results of the phenomena ranking process were consistent with current priorities in the safety analysis group for the ANSR. However, the emphasis of the PIR results on the dynamics of the break opening was high. It was felt that an experiment is needed to quantify the break opening in the primary before credible simulations would be possible.

The results of the ANSR PIR are applicable or partially applicable to a large number of transients that will need to be evaluated in the Safety Analysis Report of the ANSR. The ranking of phenomena will be used continuously by the ANSR program in the development of the experimental plan to support the ANSR design and analysis. The direct link between highly ranked phenomena and experiments will greatly improve the quality and credibility of the ANSR analyses.

The PIR process is very effective in formulating a clear consensus from a team of technical experts. The results carry the weight of the professional experience and consideration of the entire PIR team. This is a very useful tool for establishing priorities to use as a partial basis for planning and cost benefit analyses.

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Appendix A: ANSR SYSTEM DESIGN

ADVANCED NEUTRON SOURCE PROJECT

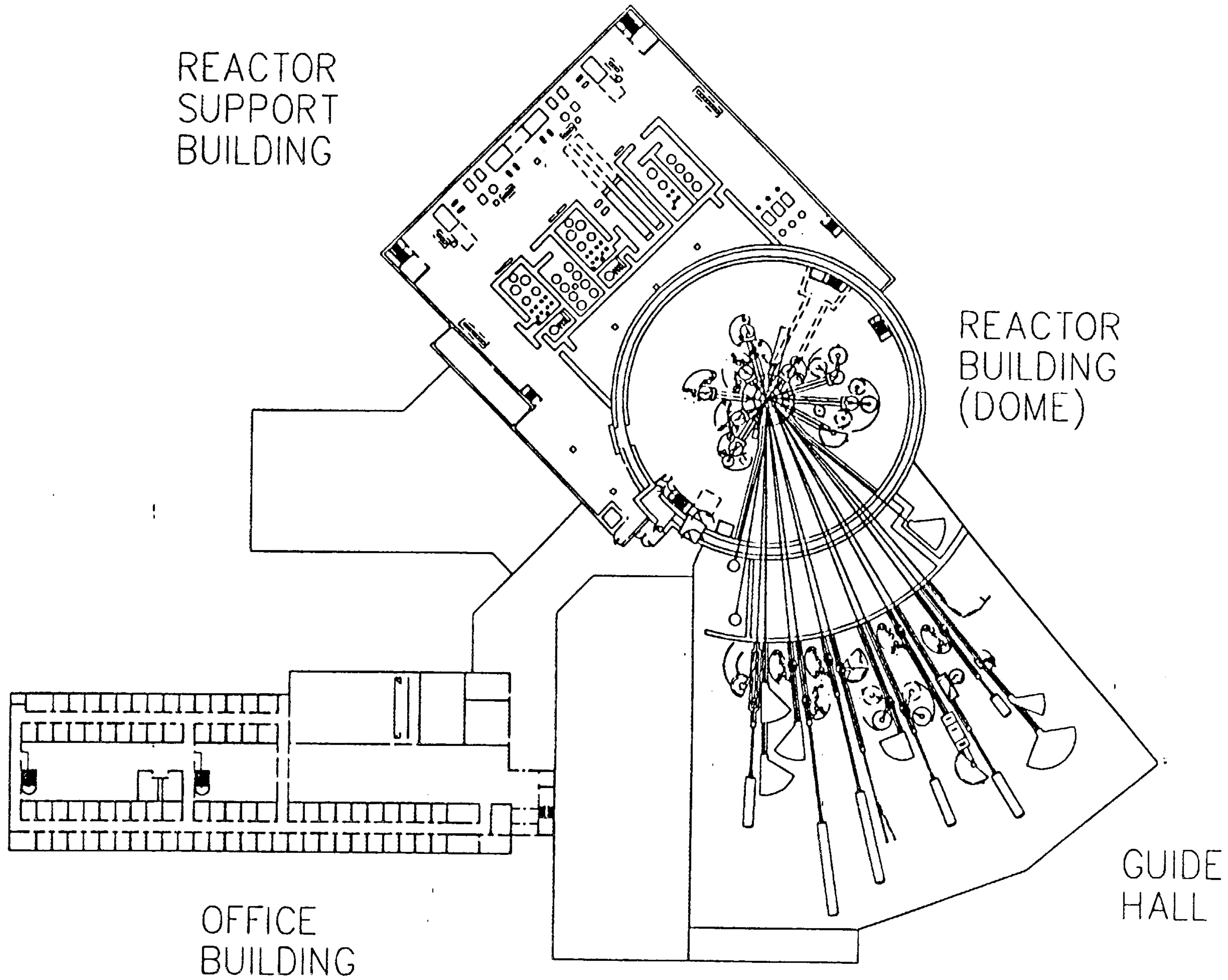
A-E KICKOFF MEETING

THE ANS AS A COHERENT FACILITY: RESEARCH COMPLEX, REACTOR SYSTEM, AND BALANCE OF PLANT

FRED PERETZ
(615) 576-5516

JULY 16, 1991

PLAN VIEW OF THE GROUND FLOOR



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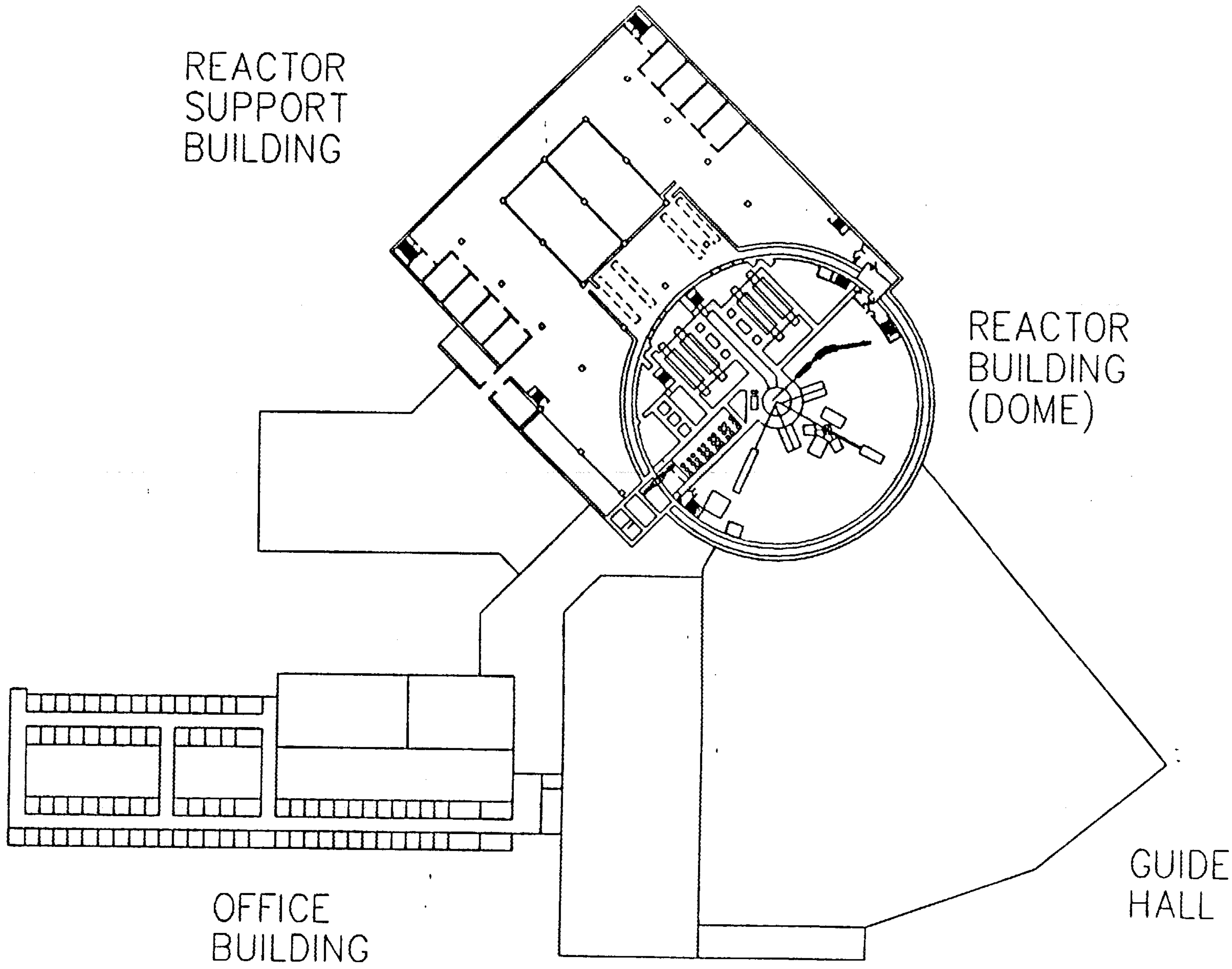
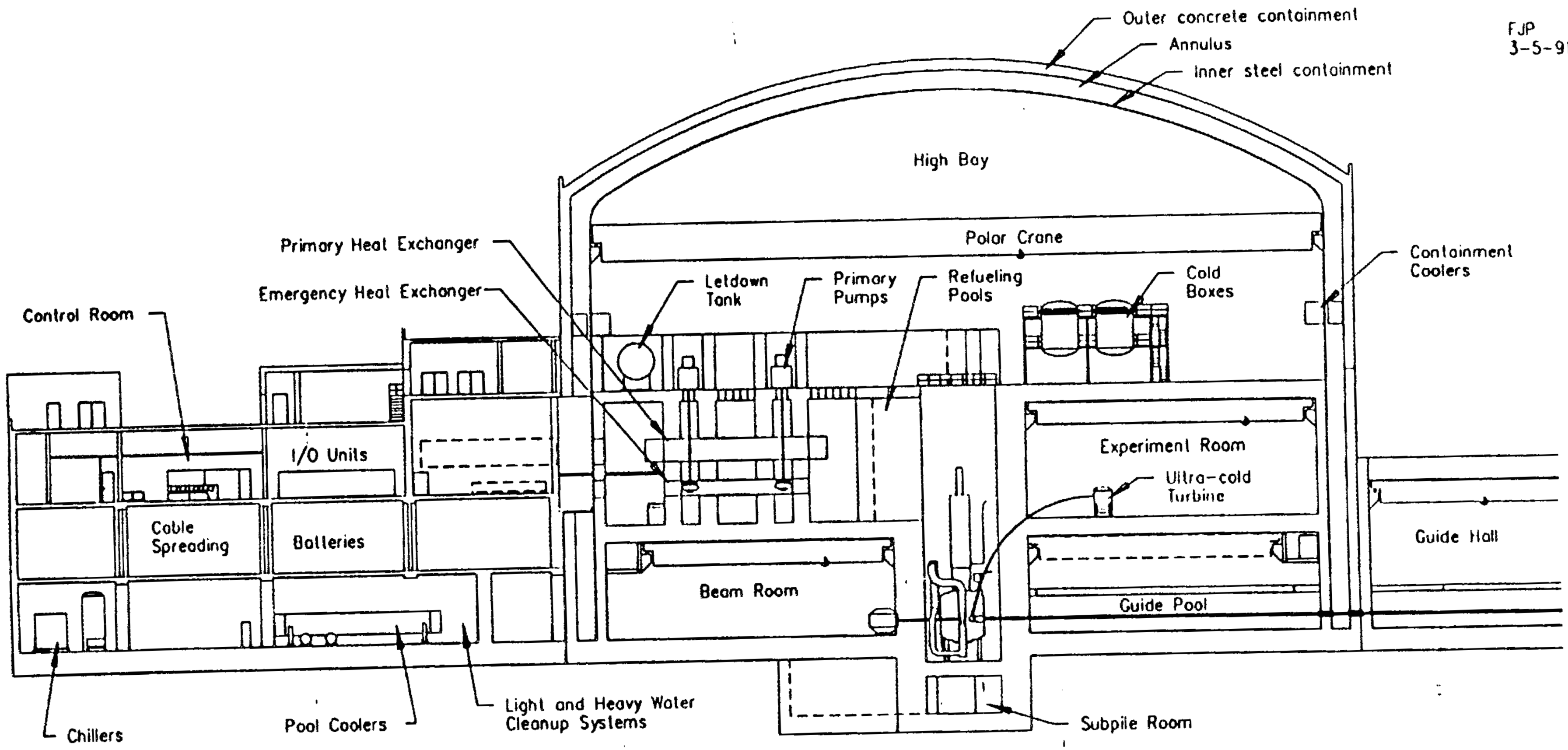
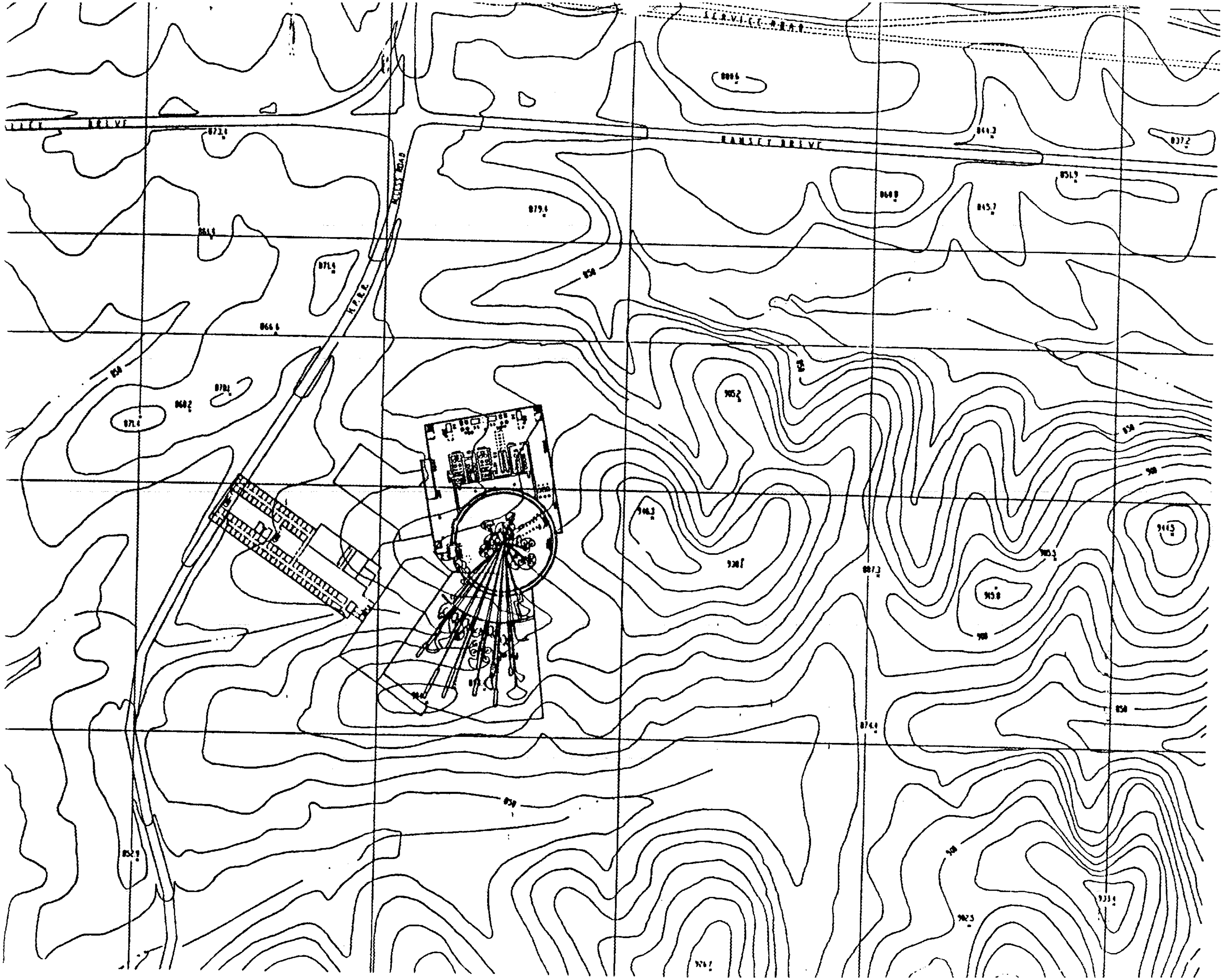



Outer concrete containment
Annulus
Inner steel containment
FJP
3-5-91
High Bay
Polar Crane
Primary Heat Exchanger
Emergency Heat Exchanger
Control Room
Letdown Tank
Primary Pumps
Refueling Pools
Cold Boxes
Containment Coolers
I/O Units
Experiment Room
Ultra-cold Turbine
Cable Spreading
Batteries
Guide Hall
Beam Room
Guide Pool
Chillers
Pool Coolers
Light and Heavy Water Cleanup Systems
Subpile Room


SERVICE ROAD
RAMSEY DRIVE
ACCESS ROAD

ANS REACTOR ASSEMBLY

FJP
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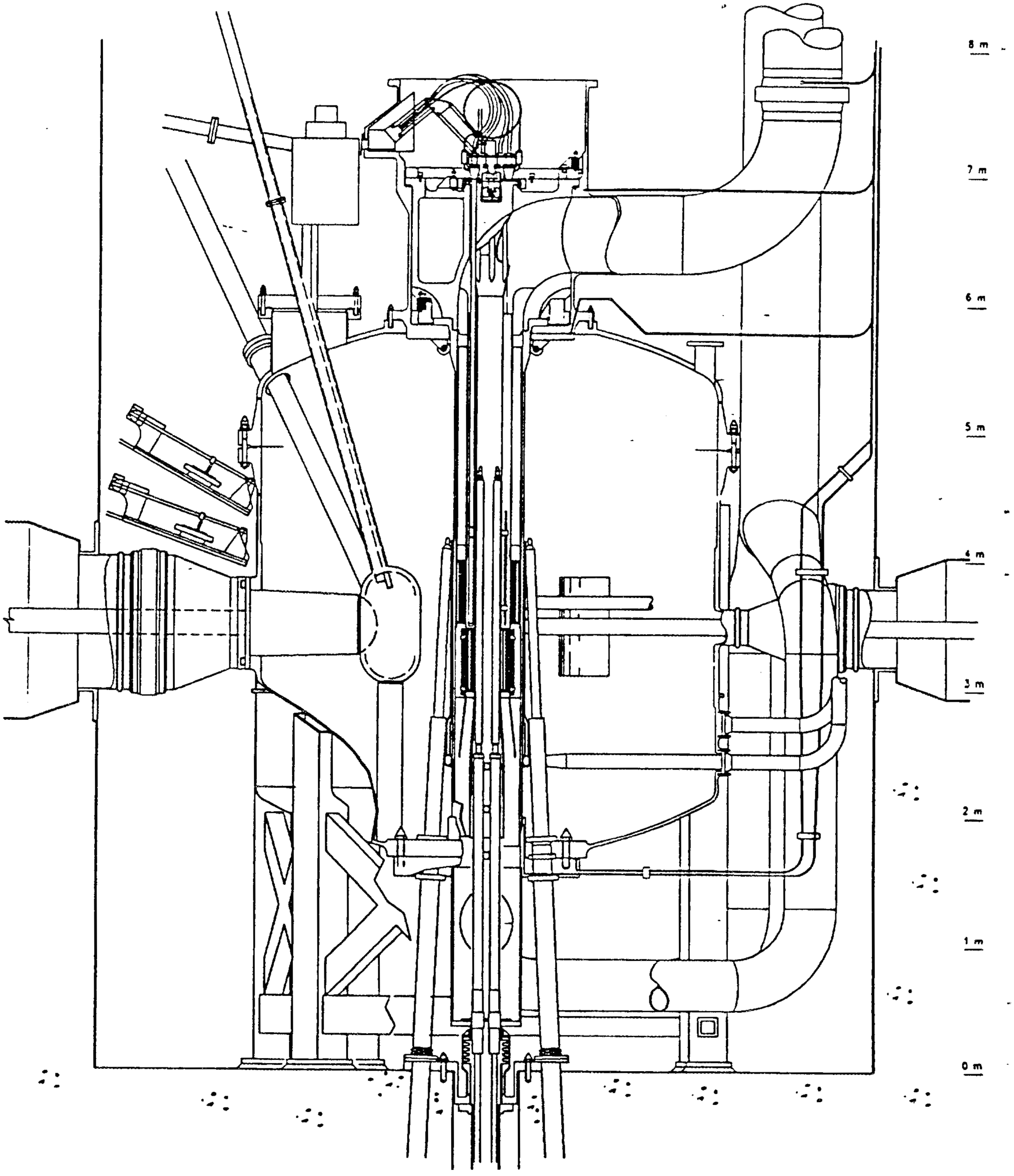
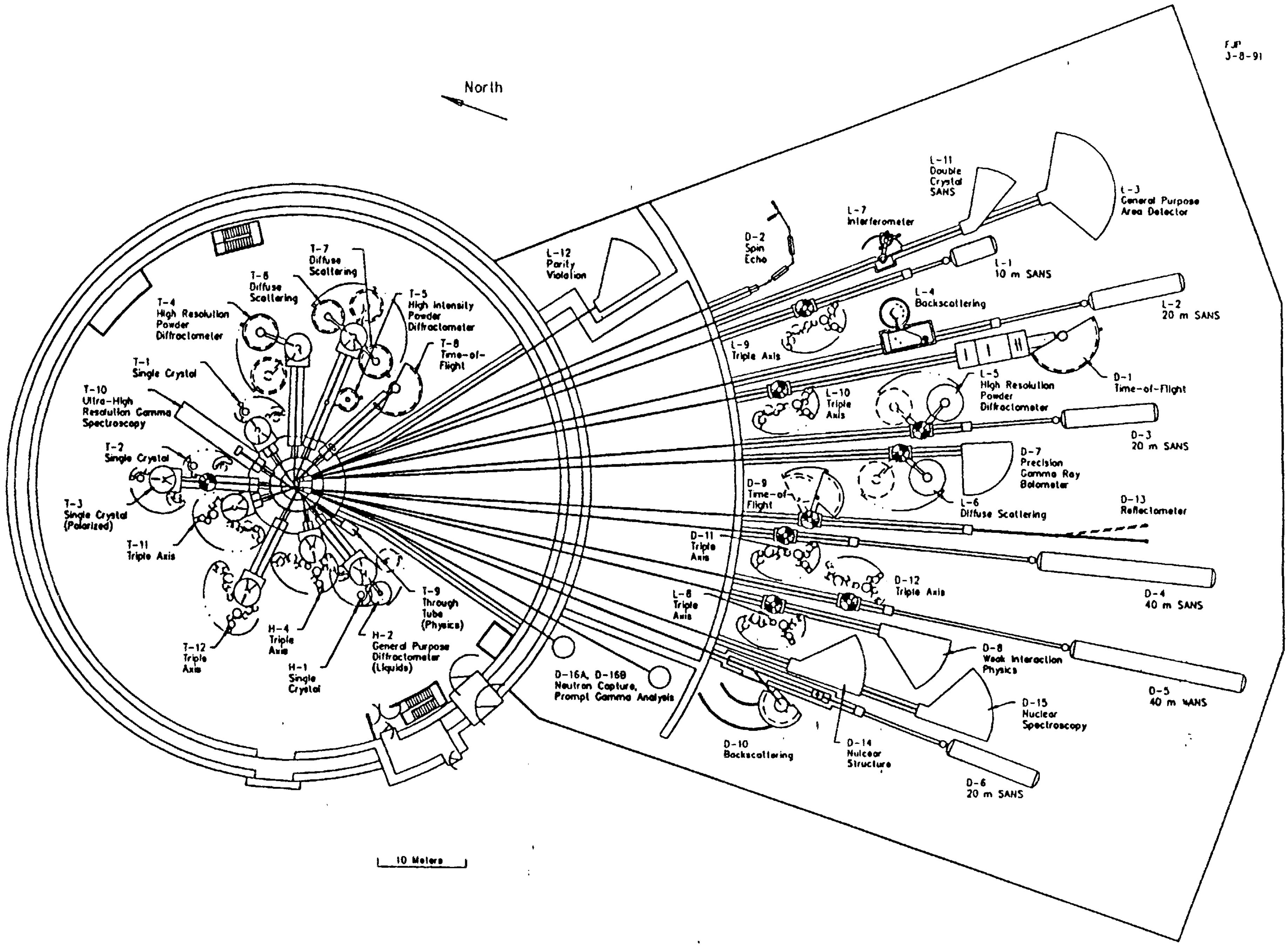
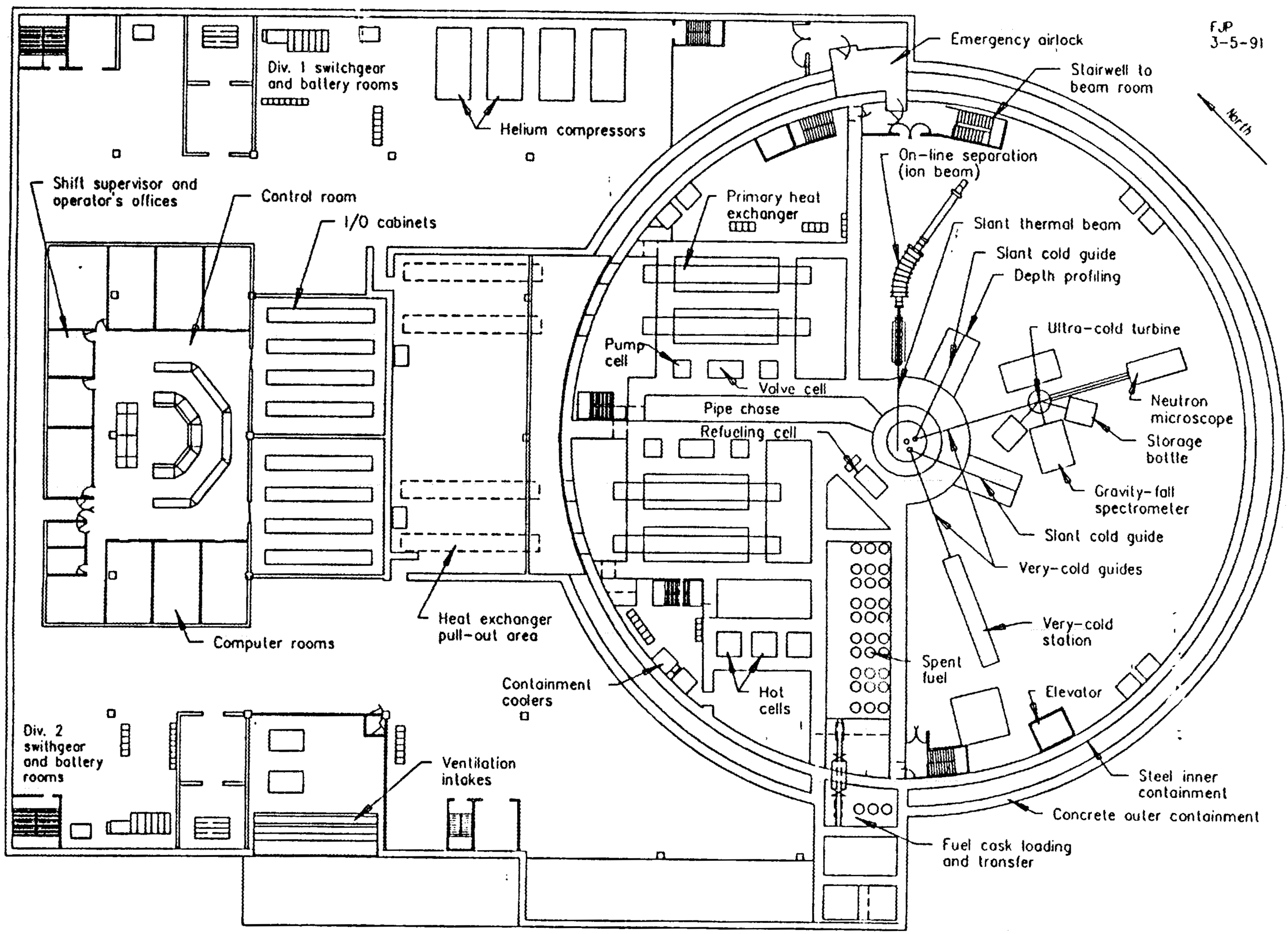


RESEARCHERS' PERSPECTIVE

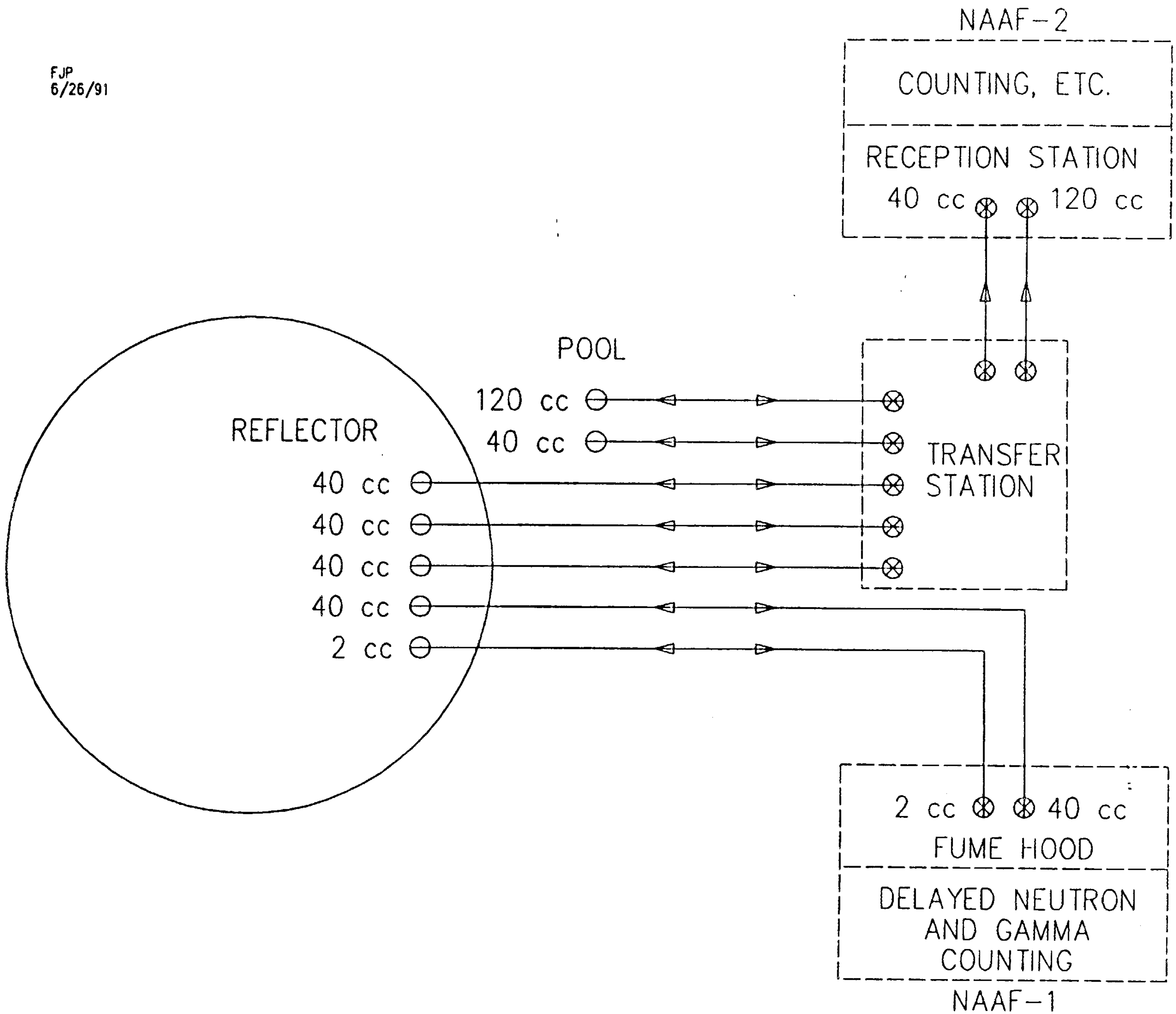


FJP
J-8-91
North
L-11
Double
Crystal
SANS
L-3
General Purpose
Area Detector
L-7
Interferometer
D-2
Spin
Echo
L-12
Parity
Violation
L-1
10 m SANS
L-4
Backscattering
L-2
20 m SANS
L-9
Triple Axis
D-1
Time-of-Flight
L-5
High Resolution
Powder
Diffractometer
L-10
Triple
Axis
D-3
20 m SANS
D-7
Precision
Gamma Ray
Bolometer
D-9
Time-of-
Flight
L-6
Diffuse Scattering
D-13
Reflectometer
D-11
Triple
Axis
D-12
Triple Axis
D-4
40 m SANS
L-8
Triple
Axis
D-8
Weak Interaction
Physics
D-5
40 m SANS
D-15
Nuclear
Spectroscopy
D-16A, D-16B
Neutron Capture,
Prompt Gamma Analysis
D-10
Backscattering
D-14
Nuclear
Structure
D-6
20 m SANS
T-7
Diffuse
Scattering
T-6
Diffuse
Scattering
T-5
High Intensity
Powder
Diffractometer
T-4
High Resolution
Powder
Diffractometer
T-8
Time-of-
Flight
T-1
Single Crystal
T-10
Ultra-High
Resolution Gamma
Spectroscopy
T-2
Single Crystal
T-3
Single Crystal
(Polarized)
T-11
Triple Axis
T-9
Through
Tube
(Physics)
H-2
General Purpose
Diffractometer
(Liquids)
H-4
Triple
Axis
T-12
Triple
Axis
H-1
Single
Crystal
10 Meters



FJP
3-5-91
North
Emergency airlock
Stairwell to beam room
Div. 1 switchgear and battery rooms
Helium compressors
On-line separation (ion beam)
Shift supervisor and operator's offices
Control room
I/O cabinets
Primary heat exchanger
Slant thermal beam
Slant cold guide
Depth profiling
Ultra-cold turbine
Pump cell
Valve cell
Pipe chase
Refueling cell
Neutron microscope
Storage bottle
Gravity-fall spectrometer
Slant cold guide
Very-cold guides
Very-cold station
Spent fuel
Computer rooms
Heat exchanger pull-out area
Containment coolers
Hot cells
Elevator
Div. 2 swithgear and battery rooms
Ventilation intakes
Steel inner containment
Concrete outer containment
Fuel cask loading and transfer

FJP
6/26/91

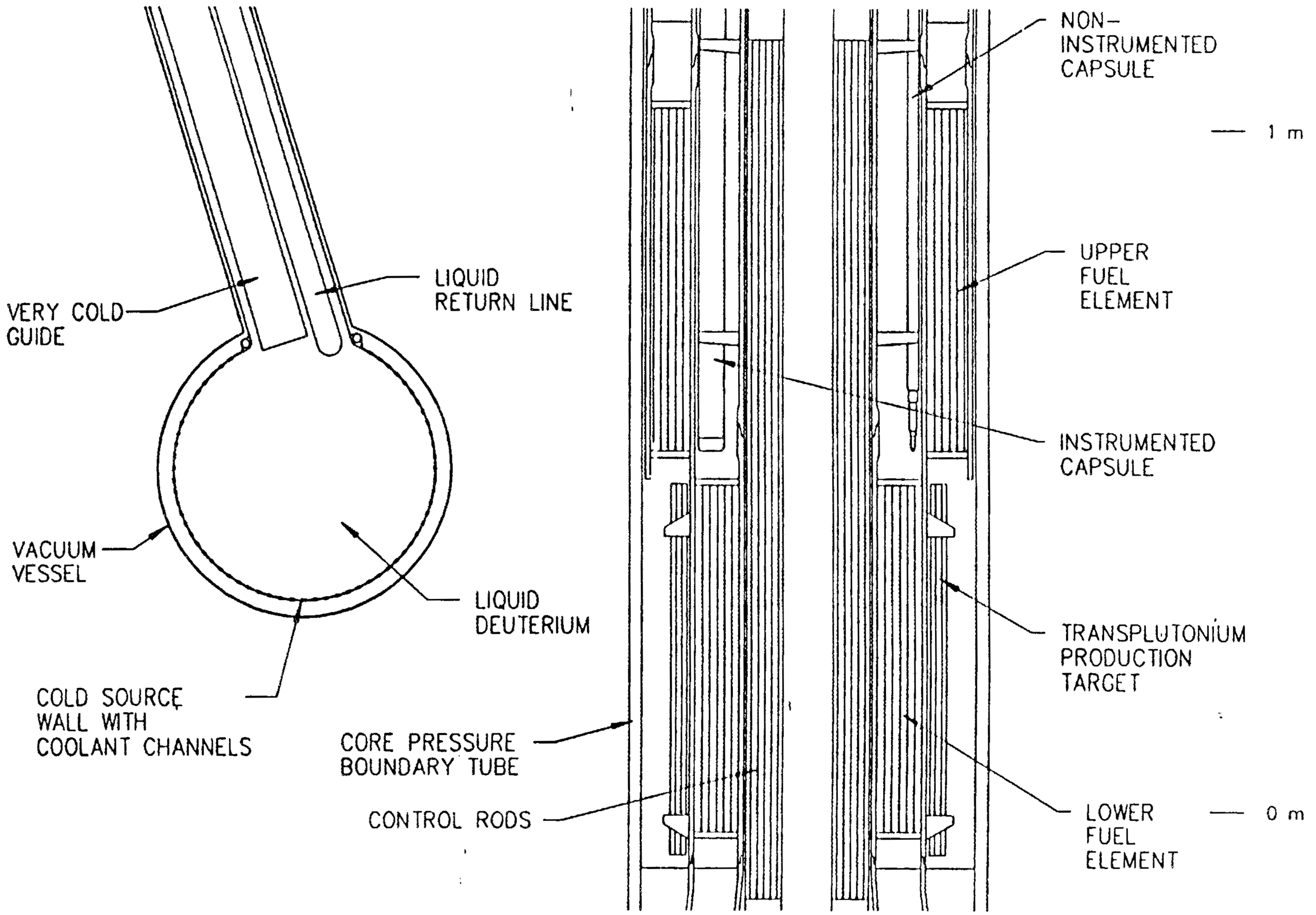

NAAF-2
COUNTING, ETC.
RECEPTION STATION
40 cc
120 cc
POOL
120 cc
40 cc
REFLECTOR
40 cc
40 cc
40 cc
40 cc
2 cc
TRANSFER STATION
2 cc
40 cc
FUME HOOD
DELAYED NEUTRON AND GAMMA COUNTING
NAAF-1

ISOTOPES PRODUCTION - GENERAL

- Transuranic Isotopes
  - Produce 1.5 g of ^{252}Cf and 40 μg of ^{254}Es per year
  - Produce small amounts of many other transuranic isotopes
- Other Isotopes
  - Produce isotopes for medical, industrial and other applications that cannot be produced commercially
  - Program inherently ill-defined
- Isotopes Production Facilities
  - In-core and reflector irradiation positions
  - Use of rabbit tubes and other facilities
  - Hot cells for unloading and shipping

CORE, COLD SOURCE, AND IRRADIATION FACILITIES

FJP
5/21/90



MATERIALS IRRADIATION

- Serve as a Key Element of the National Irradiation Program
  - Replace the irradiation facilities that currently exist at the HFIR
  - Maximize the use of the unique flux characteristics of the ANS
  - Minimize impact on scattering and on reactor availability
- Materials Irradiation Facilities
  - In-core fast neutron irradiation positions
  - Slant and rabbit tubes in the reflector
  - Hot cells for unloading, segmenting, and shipping

RESEARCH SUPPORT

- Laboratories
  - Sample preparation laboratories for scattering and other beam research
  - Final target preparation for irradiation programs
  - Analytical chemistry counting rooms and laboratories
- Shops and Assembly Areas
  - Instrument and sample environment chamber assembly and checkout areas
  - Irradiation capsule receiving facilities
  - Electronics and other repair shops
- Personnel and Computing Facilities
  - Offices, conference rooms, auditorium, food, possibly an overnight dormitory
  - Computer network (internal and external)

OPERATORS' PERSPECTIVE: OPERABLE REACTOR COMPLEX

- Operation of the Reactor Assembly
  - Neutronic startup and operation
  - Thermal and hydraulic startup and operation
  - Response to upsets
  - Refueling and maintenance activities
- Operation of the Overall Complex
  - Operation of plant machinery (motors, cranes)
  - Electrical, water, and other service systems
  - Heating, ventilation and air conditioning
  - Testing and in-service inspection
  - Deliveries, maintenance, consumables
- Control Room as the Plant Nerve Center

REGULATORS' PERSPECTIVE:
INTEGRATION INTO REGULATORY STRUCTURE

- Department of Energy
  - DOE 5480.6, Safety of DOE-Owned Reactors
  - DOE 6430.1A, General Design Criteria
  - Formal approval process
- Nuclear Regulatory Commission
  - Invoked by DOE 5480.6
  - 10 CFR 50, especially Appendix A
  - Research and power reactor guidance documents
- State of Tennessee, Environmental Protection Agency
  - Federal facilities agreements
- OSHA and Others

DESIGNERS' PERSPECTIVE: KEY DESIGN OBJECTIVES

- Create a World-Class Research Facility
  - Not an industrial facility

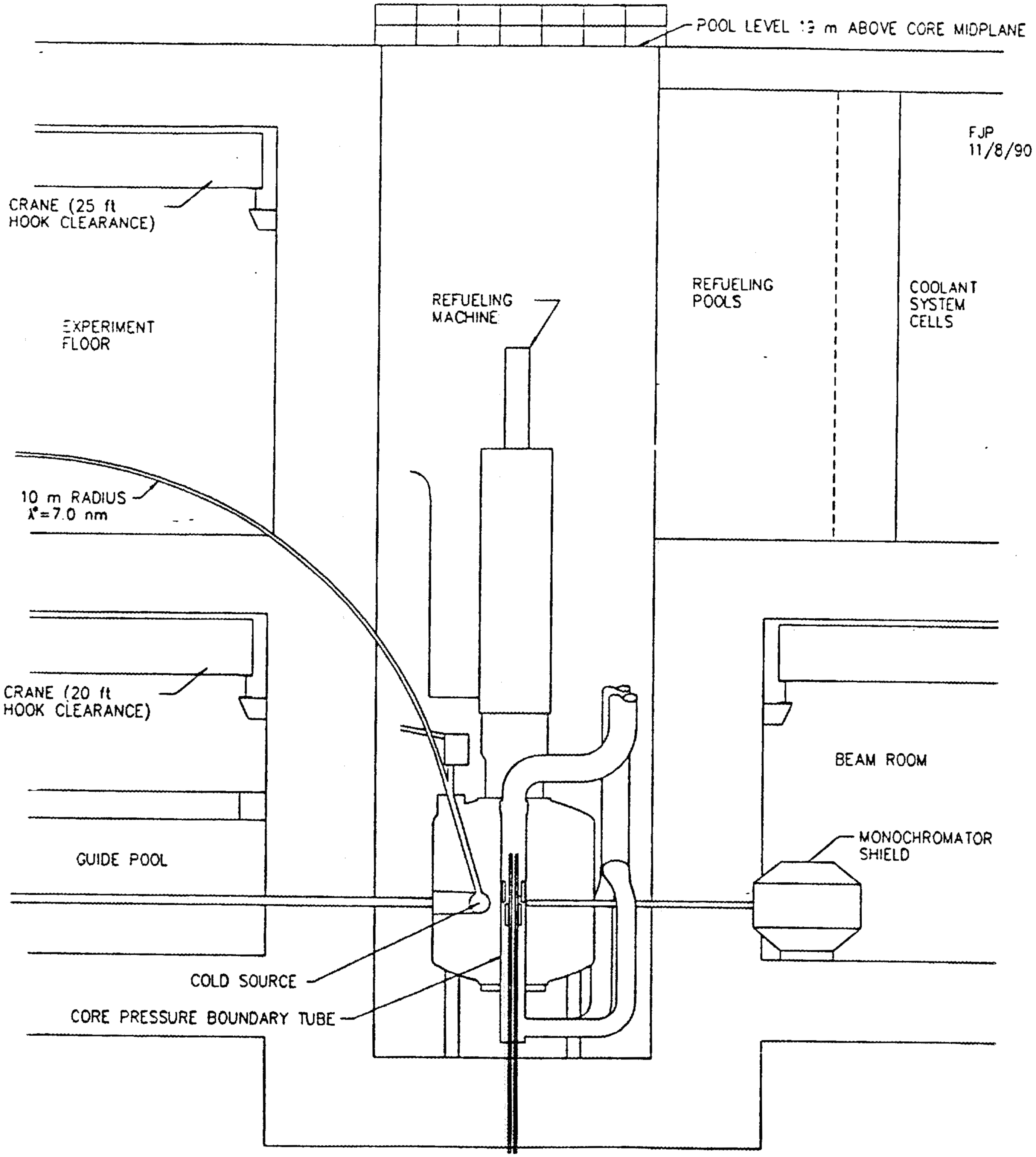
- Construct an Operable Reactor
  - And a unique one!

- Provide For All Supporting Functions
  - Support to research and reactor operation
  - As part of the construction package, and by integration with the existing infrastructure

- Understand, Clarify and Meet Regulatory Requirements
  - Not by rote application of standards established for other systems

BALANCE OF PLANT
AS
COMPLETION OF THE REACTOR

REACTOR POOL AND ASSOCIATED AREAS



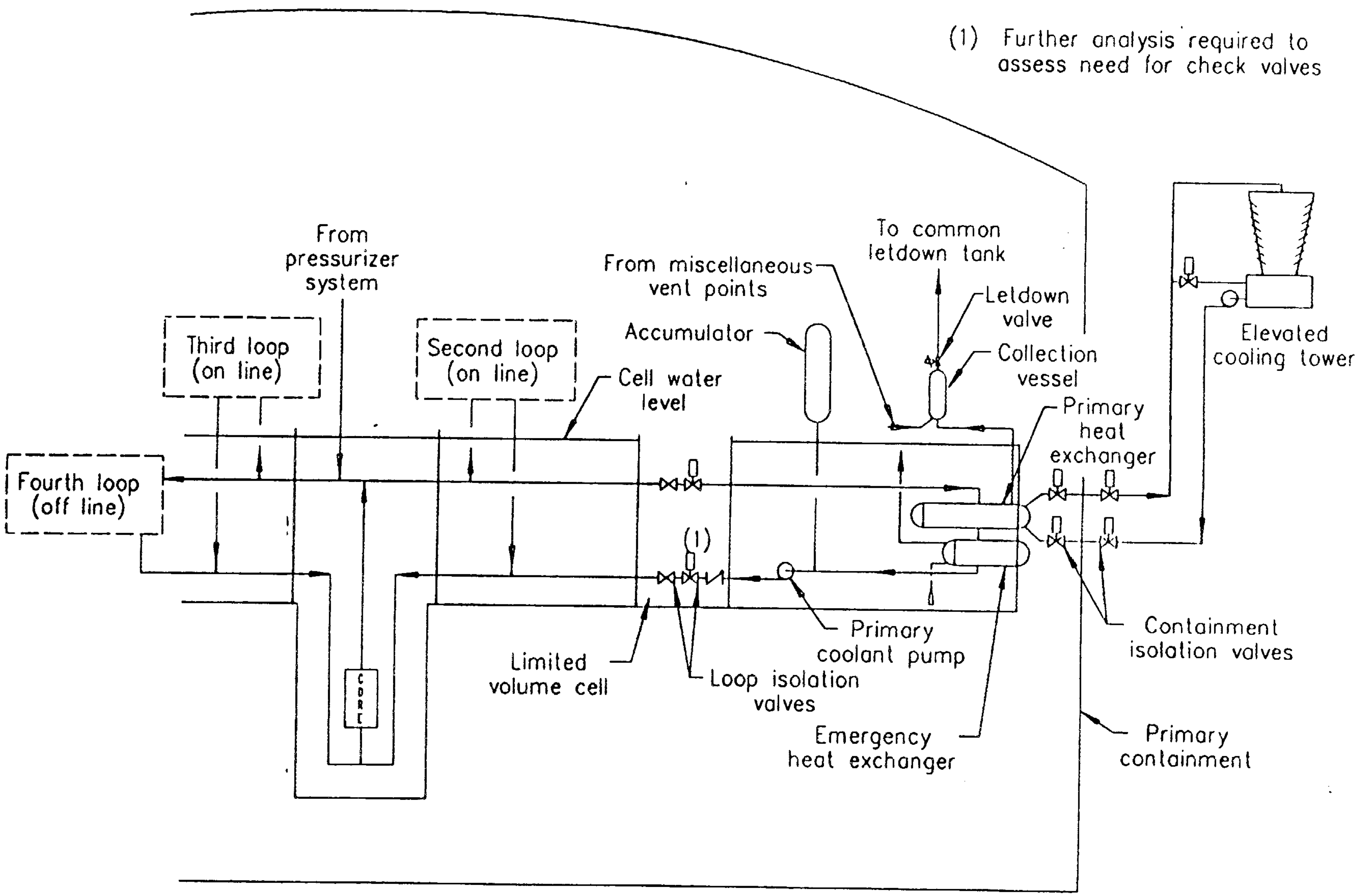
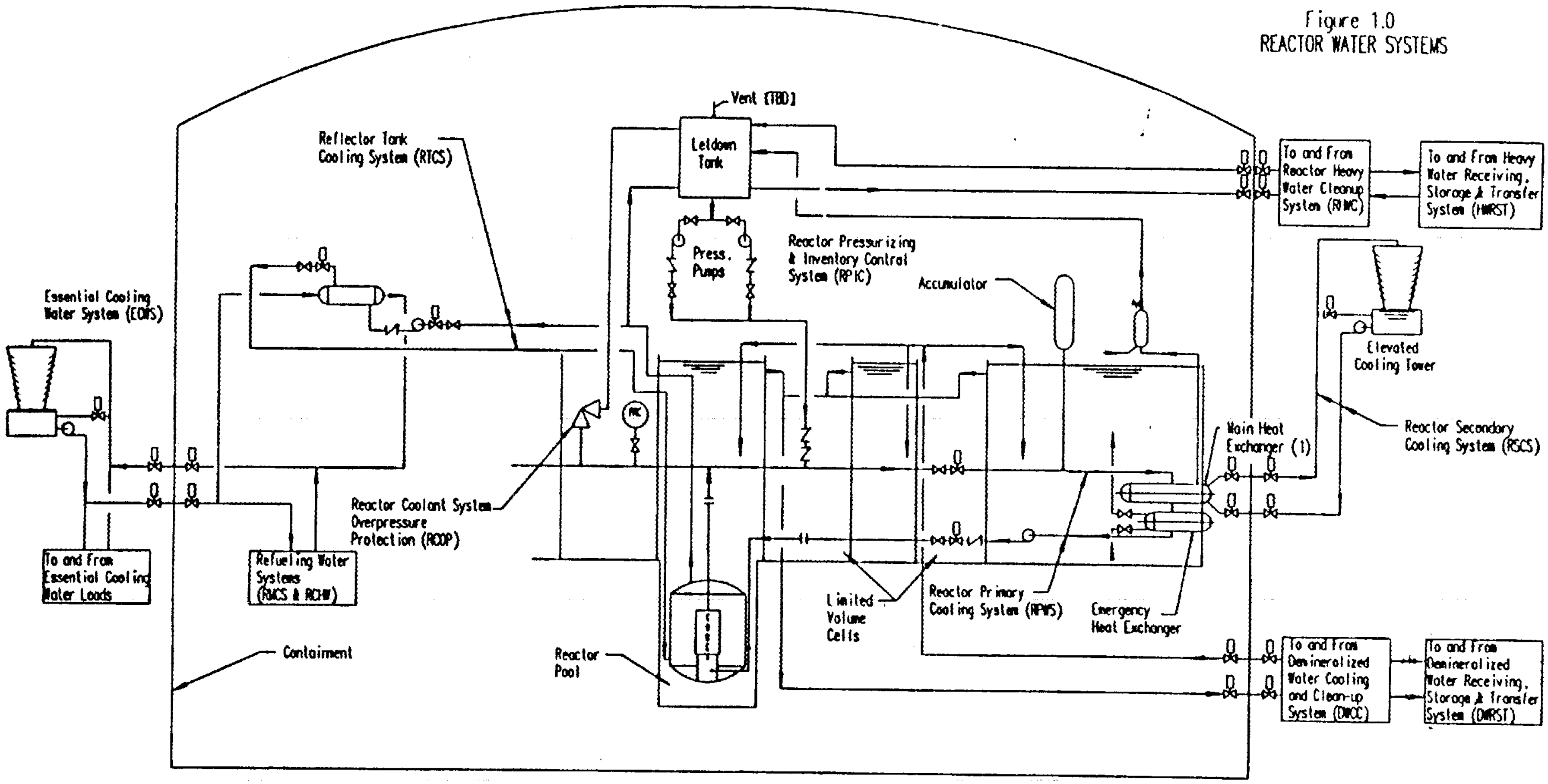

(1) Further analysis required to assess need for check valves
From pressurizer system
Third loop (on line)
Second loop (on line)
Fourth loop (off line)
Cell water level
From miscellaneous vent points
Accumulator
To common letdown tank
Letdown valve
Collection vessel
Primary heat exchanger
Elevated cooling tower
(1)
Limited volume cell
Loop isolation valves
Primary coolant pump
Emergency heat exchanger
Containment isolation valves
Primary containment
CORE

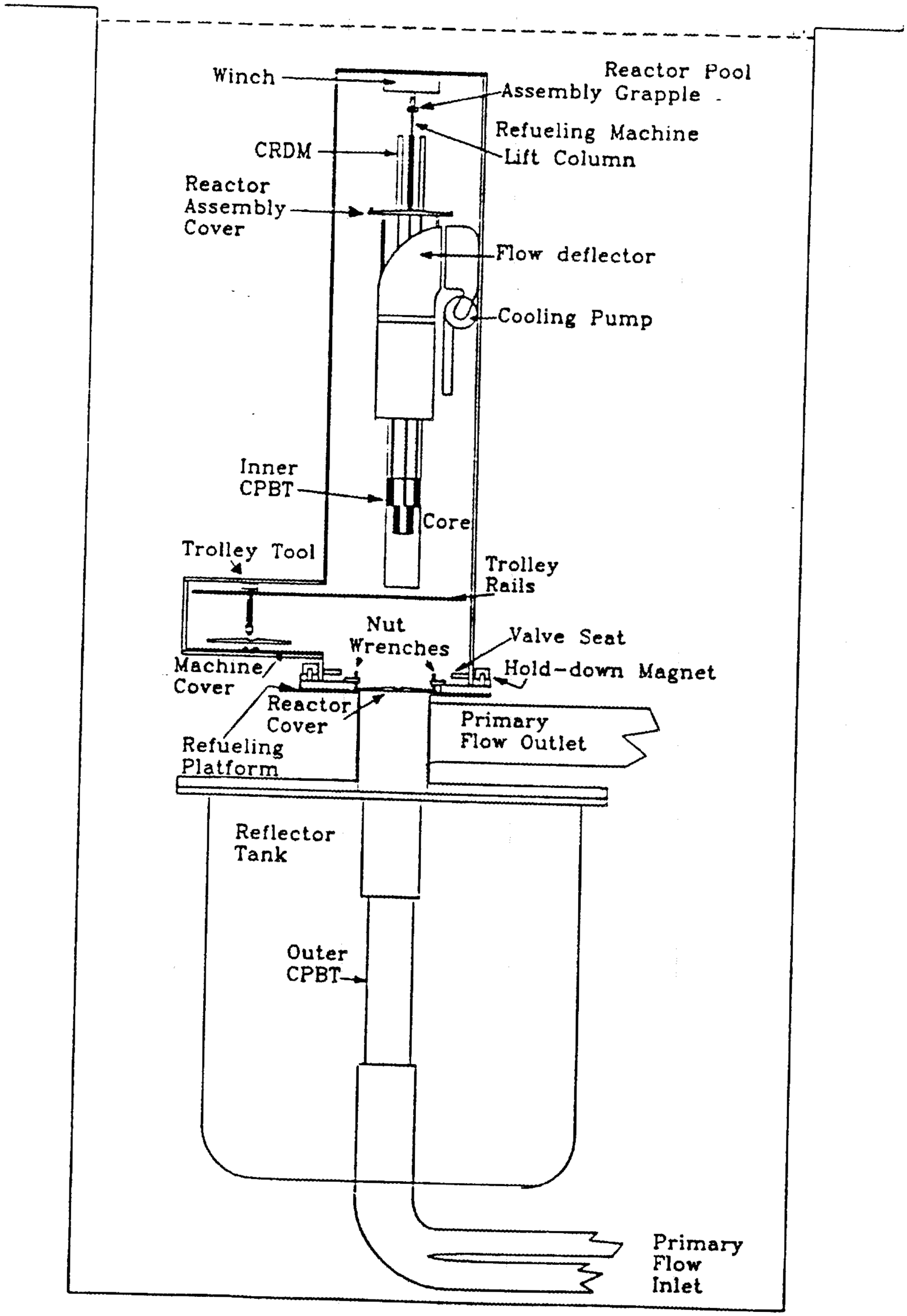
Figure 1.0
REACTOR WATER SYSTEMS



Redundant loops and components are not shown

REACTOR CONTROL

- Neutronic Operation of the Reactor
  - Collection and analysis of detector and heat-power signals
  - Startup, shutdown, and power adjustments
  - Protection system under reactor systems SDD
- Thermal Operation of the Reactor
  - Collection and analysis of reactor temperature and flow data
  - Collection and analysis of coolant loop data
  - Adjustment of temperature and flow in all loops (primary loop always flows open)
- Because of Impacts on Reactor Design and Operation, ANS Project Retains Much of the Plant I&C SDD


Winch
Reactor Pool
Assembly Grapple
Refueling Machine
Lift Column
CRDM
Reactor
Assembly
Cover
Flow deflector
Cooling Pump
Inner
CPBT
Core
Trolley Tool
Trolley
Rails
Nut
Wrenches
Valve Seat
Machine
Cover
Hold-down Magnet
Reactor
Cover
Primary
Flow Outlet
Refueling
Platform
Reflector
Tank
Outer
CPBT
Primary
Flow
Inlet

COLD SOURCE SUPPORT

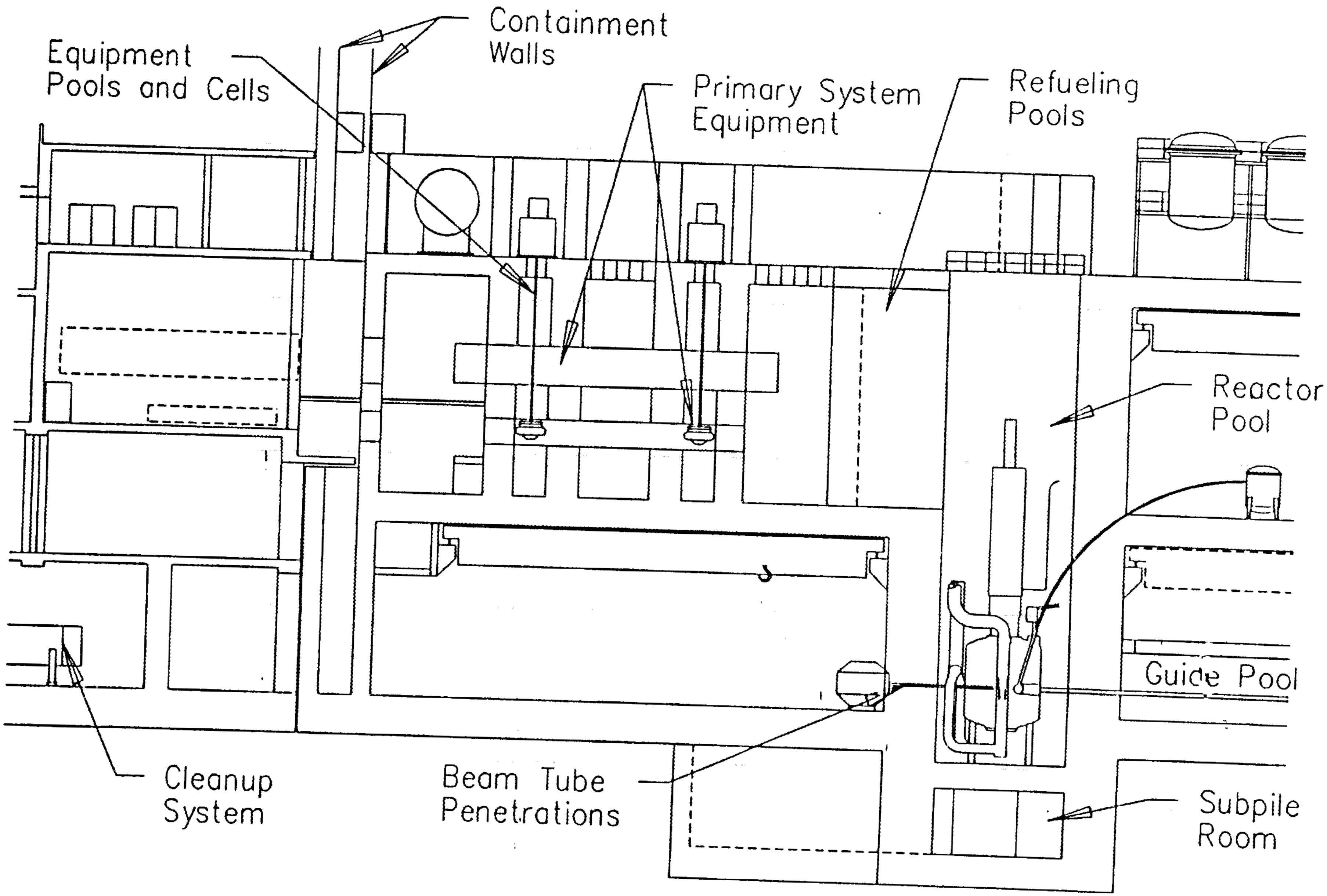
- Cold Gaseous Helium System
  - Compressors
  - Cold boxes
  - Cryogenic piping
- Deuterium Fill and Relief System
  - Deuterium supply
  - Relief tank
  - Fire protection (support area)
- Equivalent Support Needs for the Hot Source

CONTAINMENT

- ANS Reactor Containment System
  - Limits the consequences of very low probability severe accidents
  - Protects the public, on-site and other ORNL personnel, and the environment
- Features of Containment Concept
  - Inner, low leakage steel containment vessel
  - Outer, concrete secondary containment structure
  - Filtered exhaust from annulus region
- Components of the Containment System
  - Reactor building structure and containment vessel
  - Fans, filters, ducts, penetrations, valves, controls
  - Requirements allocated in an integrating SDD

CONTAINMENT PERFORMANCE AND SITING

- Siting and Design Factors Impact Potential Doses
  - Source material escape rates for fuel, piping, pools, cells, containment barriers, filters
  - Dispersion of source as a function of distance
  - Reaction time as a function of distance
- Close Relationship Exists Between Siting, Plant Design, and Containment Analysis Tasks
  - Severe accident analyses define challenge to containment systems (temperature, pressure, kinetics)
  - Plant features impact accident progression
  - Siting considerations impact control zones, dispersion factors


Containment
Walls
Equipment
Pools and Cells
Primary System
Equipment
Refueling
Pools
Reactor
Pool
Guide Pool
Cleanup
System
Beam Tube
Penetrations
Subpile
Room

CONFINEMENT ZONES

- Reactor Building Enclosed by Containment
  - Zones within containment for protection of personnel, and to limit spread of activity

- Other Buildings Include Confinement Areas
  - Process and waste systems in reactor support building and other structures
  - Limited research zones
  - Confinement areas use tight enclosures and controlled ventilation sweep, but are not capable of holding pressure

- Confinement Systems Composed of Similar Components
  - Walls, barriers, ventilation systems, filters, controls

RADIATION PROTECTION AND ALARA

- The ANS Presents Unusual Challenges to Radiation Protection

  - Desire for instruments close to the reactor source, but with acceptable dose rates and background

  - Dose rates and background levels around very-cold guides, and in guide halls

  - Tritiated heavy water leads to pervasive tritium dose issues

- The ANS Also Presents the Usual Radiation Protection Issues

  - Design goals well below legal limits (1/10?)

  - Shielding of radioactive fluids (^{16}N, crud, etc)

  - Spent fuel and target storage and handling (neutron dose with Cf targets)

- Radiation Protection Measures Should Accommodate Potential Accident Conditions

Appendix B: SUGGESTIONS FROM ANSR PIR TEAM MEMBERS RELATING TO THE ANSR DESIGN

Appendix B: SUGGESTIONS FROM ANSR PIR TEAM MEMBERS RELATING TO THE ANSR DESIGN

(1) It may be desirable to connect the gas volumes at the top of the three accumulators. This may ease control of accumulator air volumes. The use of liquid pressurizer pump and liquid bleed strategy to control system pressure may be difficult with accumulators in place. Liquid adjacent to the gas in the accumulators will be saturated with dissolved gas. Gas will come out of solution when the system pressure is lowered. Gas will go into solution when the system pressure is raised. The gas takes time to go into and come out of solution, which may complicate pressure control. Bob Graham has looked into these types of systems.

(2) Prefer to see a single pressure relief valve on the system hot side prior to splitting the flow to the four independent primary loops (i.e., on the current hot leg header).

(3) The Core Pressure Boundary Tube (CPBT) can be exposed to compressive stress if the primary system pressure in the CPBT falls below the pressure in the reflector tank. This may collapse the tube in the present design. Also, the strength of the CPBT may be increased by adding hoops to help with the circumferential stresses. These may increase strength without adding as much material. This may minimize associated penalties in neutronic performance. A combination of circumferential hoops and axial reinforcements may be used to stop crack growth (a strategy frequently used in aerospace structures). This may allow a maximum break size to be postulated. Similar ideas can be employed in the double walled CPBT concept.

(4) It may be desirable to control the flow to the upper and lower fuel assembly halves during the fuel cycle. This would be done to maintain the power to flow ratio roughly constant for each fuel annulus during the fuel cycle. This trimming of the flow may be done integral to the control rod drive, or with an independent drive. A broad range of flow control is not necessary.

Appendix C: COMPONENT AND PHENOMENA DEFINITIONS

Appendix C: COMPONENT AND PHENOMENA DEFINITIONS

Accumulator (A): A gas volume over a large liquid volume that follows the pressure of the primary system. The gas volume, water volume, tank and pipes connecting the assembly to the primary loop are all included in the accumulator component definition.

Area vs. time (BR): The manner in which a break is formed in the primary coolant pressure boundary as described by a cross-sectional area for flow as a function of time.

Break (BR): An opening in the primary coolant circuit that allows coolant to escape.

Bypass: The flow that goes between the fuel assembly and the CPBT.

Cold leg (CL): Pipes in the primary loops downstream of the primary heat exchanger including the junction of the primary loops at the bottom of the core pressure boundary tube.

Control Rods (CR): Rods used to trim the reactor power during fuel burn and used to runback the reactor during station blackout. These rods are also used to scram the reactor.

Cooling tower basin (CTB): Basin containing secondary water at the base of the cooling tower.

Core Pressure Boundary Tube (CPBT): An aluminum pressure boundary between the primary system coolant flow and the reflector tank.

Critical flow (FA): A flow traveling at a velocity sufficient to preclude propagation of pressure information upstream.

Emergency Heat Exchanger (EHX): Heat exchanger using the reactor light water pool for decay heat removal from the primary flow using natural circulation. Primary flow is on the shell side and the heat exchanger is horizontal, with the shell side split vertically along the tube axis.

Flow resistance (B): Resistance to coolant flow in the region of the bypass.

Flow resistance (CR): Resistance to coolant flow in the region of the control rods.

Flow resistance (FA): Resistance to coolant flow in the fuel cooling channels.

Form factors (FA): Coefficients to model pressure losses associated with changes in flow cross-section, direction or geometry.

Fuel Assembly (FA): Upper and lower fuel annuli including the fuel plates and supporting aluminum rings.

Gas evolution (A): Dissolved gas coming out of solution as the accumulator pressure changes.

Gas process line (A): The thermodynamic process line that describes the behavior of the lumped state variables for the gas in the accumulator during its operation.

Head vs. mass flow and time (P): The head developed by the pumps in the primary cooling system as a function of the mass flow and of time. The time variable is coordinated with other variables in the transient such as power or system pressure.

Heat load (CR): The thermal energy that is deposited in the control rods.

Heat load (CPBT): The thermal energy that is deposited in the CPBT.

Heat load (T): The thermal energy that is deposited in the targets.

Heat transfer to secondary (PHX): The transfer of thermal energy to the secondary cooling system through the primary heat exchanger.

Heat transfer (HL): Thermal energy transferred from the primary coolant by the hot leg pipes. This would include heat transferred to the pool and to other regions in contact with the hot leg pipes.

Heat transfer (CL): Thermal energy transferred from the primary coolant by the cold leg pipes. This would include heat transferred to the pool and to other regions in contact with the cold leg pipes.

Heat transfer (EHX): Heat transfer from the primary coolant to the reactor pool by the emergency heat exchanger.

Heat Transfer to Air (CTB): The transfer of thermal energy to the atmosphere by the cooling tower basin.

Heat transfer to pool (PHX): Heat transfer to the reactor pool by the primary heat exchanger shell.

Hot leg (HL): Pipes in the primary loops upstream of the primary heat exchanger including the junction of the primary loops at the top of the core pressure boundary tube.

Hydraulic loads (FA): The loads on the fuel plates associated with loads induced by the coolant flow.

Initial temperature (CTB): The temperature of the cooling tower basin at the beginning of the transient.

Initial temperature (RP): The temperature of the reactor pool at the beginning of the transient.

Inlet temperature and velocity distribution (FA): The velocity and temperature distribution of the coolant entering the fuel assembly.

Inventory (CTB): The initial water content in the cooling tower basin.

Mass flux vs. time (BR): Mass flux of liquid out of the break as a function of time.

Multi-dimensional flow (FA): Spatial variations in the flow in the fuel cooling channels. This flow is frequently assumed to be one dimensional (axial).

Oxide growth and spallation (FA): The formation of corrosion products on the surfaces of the reactor fuel plates. This layer of "oxide" can flake off the fuel plate into the primary coolant flow (i.e., spall). This process has defined a thermal limit for the fuel assembly since the spalling may lead to a breach in the fuel cladding.

Plate spacing (FA): The spacing between the fuel plates in the fuel assembly that determine the hydraulic diameter of the fuel cooling channels.

Power density (FA): The spatial distribution of power in the fuel assembly.

Power vs. time (FA): Power produced in the fuel assembly as a function of time.

Pressurizer (PR): System providing flow into primary pressure boundary for pressurization as determined by performance of pressurizer pumps (both on-line and standby) and heavy water inventory available for pressurization. The behavior of the liquid bleed valve and associated control systems are also included.

Primary departure from nucleate boiling (DNB), (FA): The transition from nucleate boiling heat transfer to transition boiling (and eventually to film boiling). The point of DNB is associated with known mass flux, heat flux and fluid conditions.

Primary Heat Exchanger (PHX): Heat exchanger in the light water pool with the primary flow on the shell side and the secondary flow on the tube side. The heat exchanger is oriented horizontal, with the shell side split vertically along the tube axis.

Pumps (P): Primary coolant pump including the primary motor, pony motor and associated power supplies.

Reactor pool (RP): The pool of water surrounding the reflector tank.

Single phase friction factor (FA): Friction factor for determining the pressure losses for single phase flow in the fuel cooling channels.

Single phase heat transfer (FA): The heat transfer when the coolant in the fuel assembly is single phase liquid.

Stratification (CTB): Thermal stratification in the cooling tower basin during decay heat removal.

Targets (T): Isotope production rods with cooling shrouds are positioned upstream of the upper fuel annulus. Various target assemblies are positioned inside the upper fuel annulus.

Thermal strain (FA): The distortion of the fuel plates due to thermal gradients.

Two phase pressure drop (FA): The pressure drop in a two-phase diabatic flow includes components due to the change in momentum flux and due to viscous losses. The change in momentum flux should be modeled as a result of modeling vapor generation and using that information in the two-fluid conservation equations.

Two phase heat transfer (FA): All heat transfer modeling between the wall and fluid(s) occurring after the incipience of boiling.

Vapor generation (FA): The initial formation of vapor and its subsequent development is usually modeled by assembling models for several fundamental phenomena:

a) Incipience of boiling
b) Onset of significant vapor generation or bubble detachment
c) Balance of vapor generation, condensation and transport terms which depend on interfacial area, heat, mass and momentum transfer models.

Appendix D: FUEL DAMAGE MECHANISMS ADDITIONAL TO MELTING

FUEL-PLATE STABILITY EXPERIMENTS AND ANALYSES FOR THE ADVANCED NEUTRON SOURCE

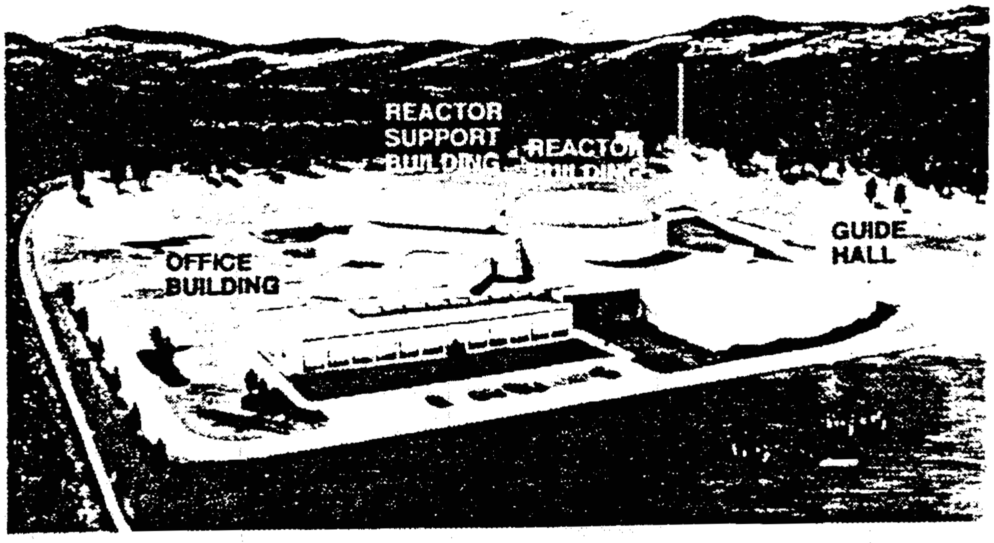
W. K. Sartory, W. F. Swinson, and G. T. Yahr

One of the missions of the Oak Ridge National Laboratory is to design, construct, and operate an advanced neutron source (ANS) facility for research. The ANS is to be a new experimental facility that will provide an intense steady-state source of neutrons from a reactor of unprecedented flux. The user facility will serve scientific research from across the nation in the fields of chemistry, physics, biology, materials, and nuclear science.

Assessing the structural performance of the reactor fuel plates is the responsibility of a group within the Structural Mechanics Section. Past experience has shown that fuel-plate failures can occur when the coolant flow causes the closely spaced plates to deflect and touch, causing burnouts. Because the ANS has a very high power density that requires a higher coolant flow velocity than previous reactors, there is a higher potential for plate stability problems. Classical theory indicates that at some coolant velocity the plates will become unstable and collapse. This potential stability problem is being examined by extending the classical theory to include curved (involute) plates and coupling the plate equations with coolant flow equations containing friction and entrance/exit conditions. In addition, limiting design analysis, based on the dynamic pressure of the coolant, is being proposed and developed for predicting the plate deflection and structural failure.

Hydraulic experiments are being conducted by testing epoxy plate models to failure, and thereby assessing the analyses and the applicability of the theories. Plans are also being made to test the response to coolant flow of dummy aluminum plates to verify their stability under reactor conditions. In addition, the vibrational characteristics of the plates are to be determined experimentally.

Building This Advanced Neutron Source (ANS) at the Oak Ridge National Laboratory is a Very Important Project



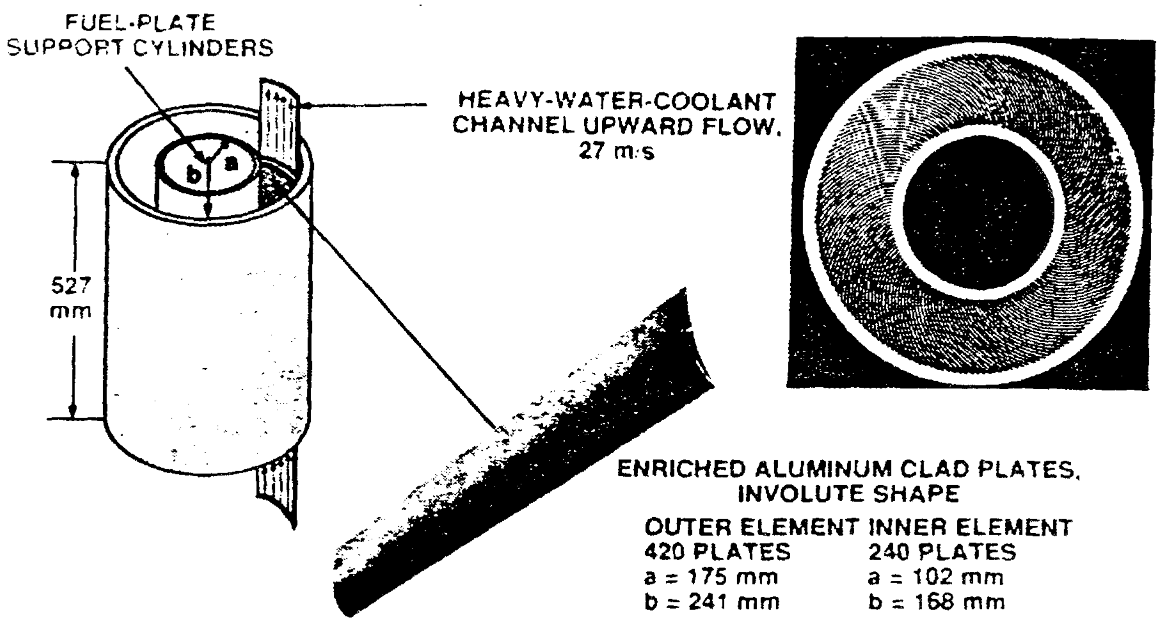
ET/ornl

This Presentation Has Two Objectives

- Outline the fuel-plate stability problem as it relates to the design of the ANS
- Review how the fuel-plate stability problem is being addressed and solved through
  - Analytical models to predict the fuel-plate response to coolant flow
  - Experiments to validate the analytical models and to describe the fuel-plate response to coolant flow

ET/ornl

The Enriched Plates Are Aluminum Clad, Involute in Shape, Supported by Concentric Cylinders, and Cooled with an Upward Coolant Flow



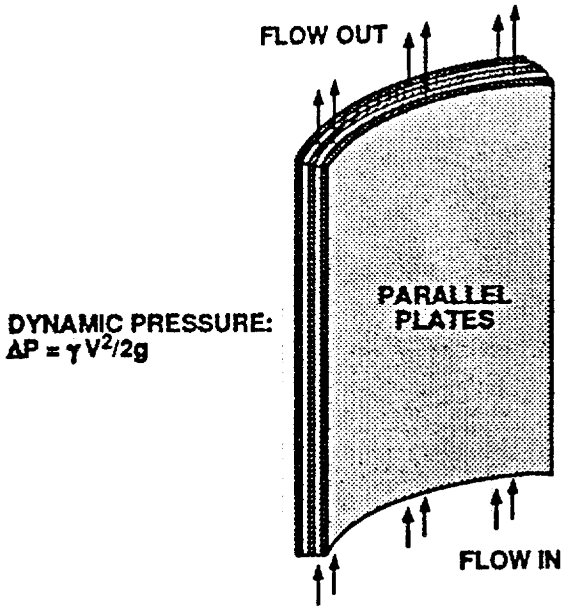
ET/ornl

To Appreciate Just How High the ANS Coolant Velocity Is, Compare the HFIR, ILL, and ANS Coolant Velocities

Description	HFIR	ILL	ANS
Plate shape	Involute	Involute	Involute
Plate thickness	1.27 mm	1.27 mm	1.27 mm
Coolant channel	1.27 mm	1.80 mm	1.27 mm
Coolant velocity	15.5 m/s	15.5 m/s	27.4 m/s

ET/ornl

A Model Based on Parallel and Unequal Mass Flow Shows the Limiting Pressure between Plates to be the Dynamic Pressure That Can be Used to Calculate Plate Deformation and Stress



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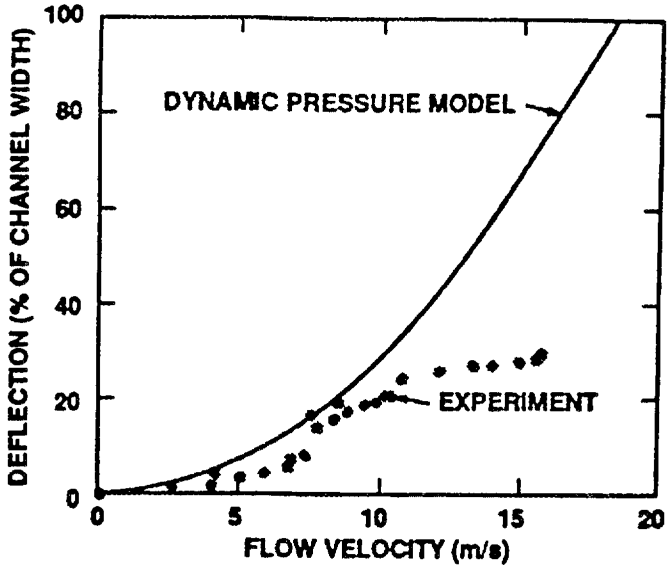
Data from Hydraulic Flow Experiments Are Required to Validate Analyses and to Quantify the Plate Response Directly

- Tests were run and data collected from a single epoxy model of a HFIR involute plate for comparing with the developed analytical models
- A flow loop for testing multiplate ANS involute plates is in construction
  - — Epoxy models of the ANS plates are to be tested to failure
  - — Dummy aluminum plates are to be tested at the ANS operating coolant velocity

ET/ornl

Dynamic Pressure Model Gives Reasonable Bound on Deflection of Single Involute Plate



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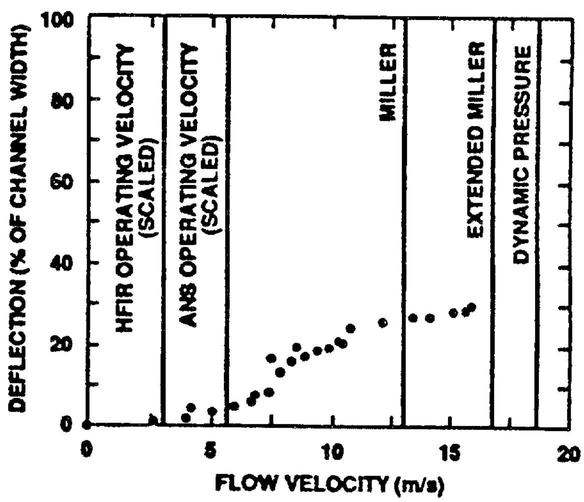
Single Epoxy Involute Plate Test Suggest the Following Points

- Negligible deflection at equivalent HFIR velocity
- Small deflection at equivalent ANS velocity
- No instability at 94% of extended Miller prediction
- Dynamic pressure prediction 109% of extended Miller prediction



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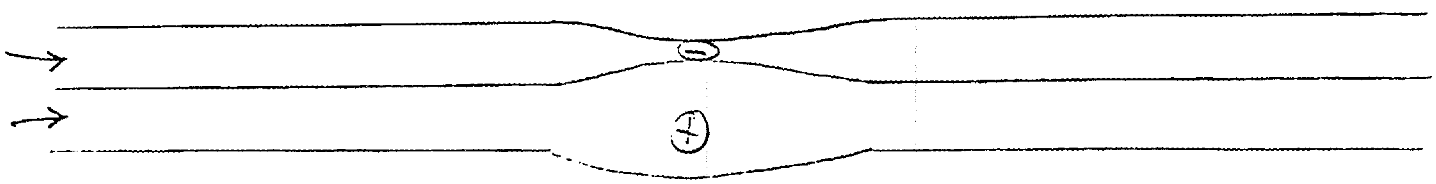
ANS Involute Plate Hydraulic Stability

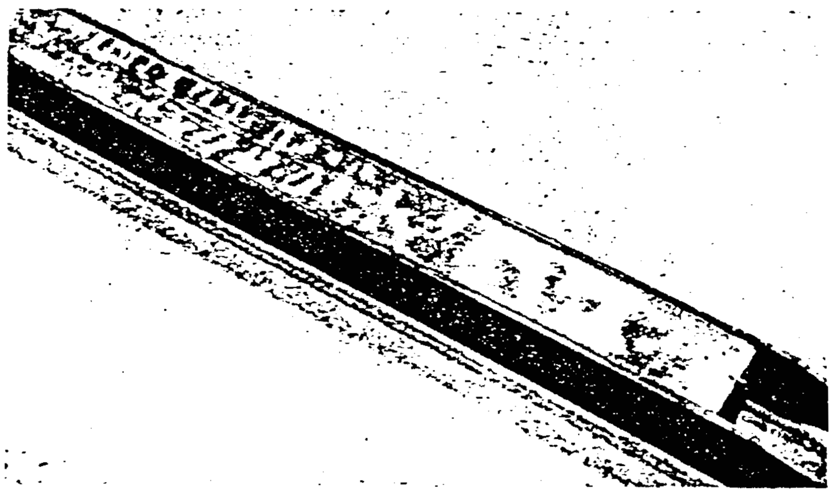
W. K. Sartory

October 29, 1991



Sketch of Miller's Plate Deflection Leading to Unstable Bernoulli Pressure

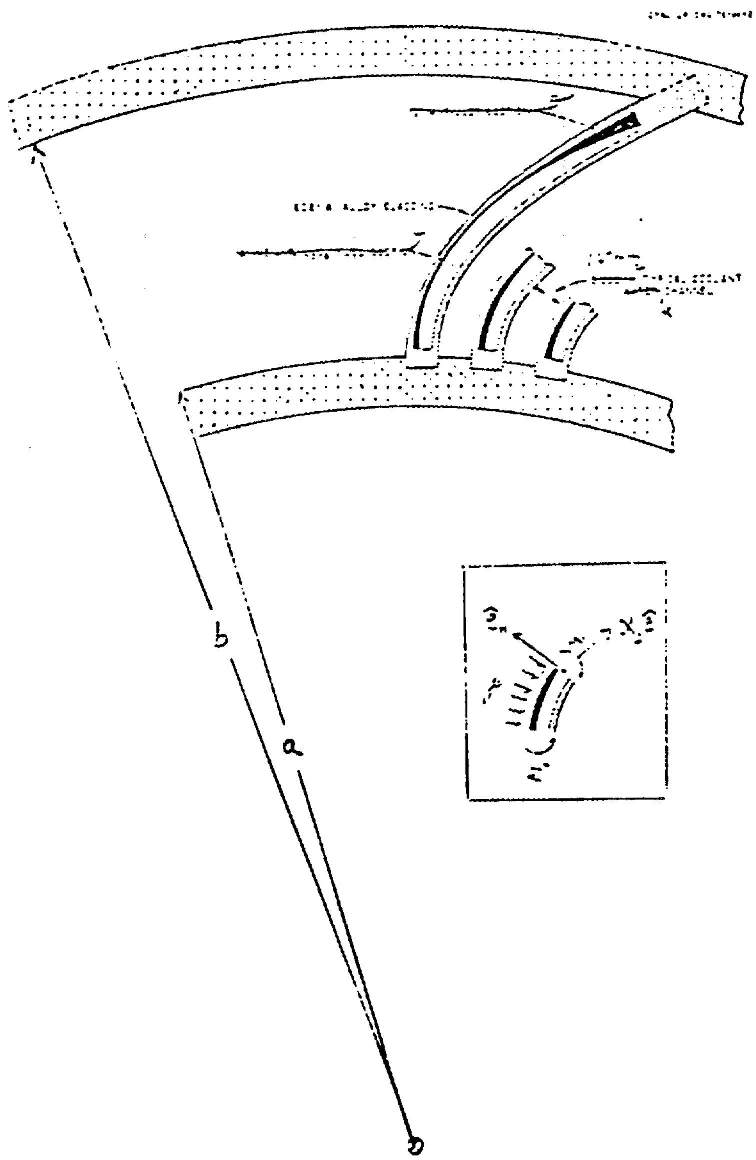
Photo of Buckled Reactor Fuel Element



One of ETR fuel elements that buckled during in-core test at design flow

Sketch of Hundreds of Involute Plates

Mounted in Cylindrical Sidewalls



Computer Modeling

- Plate Modeling--ABAQUS 9-node Quad
- Fluid Modeling--User Element
  - 2-D Thin Flat Channel
  - Two Momentum Equations
    - Plus One Continuity Equation
  - Fanning Friction Factor
  - Linear Perturbation Theory
- Infinite Array of Plates & Channels
  - Only one plate & one channel calc'd

Fluid Channel Flow Equations

$$\frac{\partial \rho h v_1}{\partial \alpha} + \frac{\partial \rho h v_2}{\partial z} = 0$$

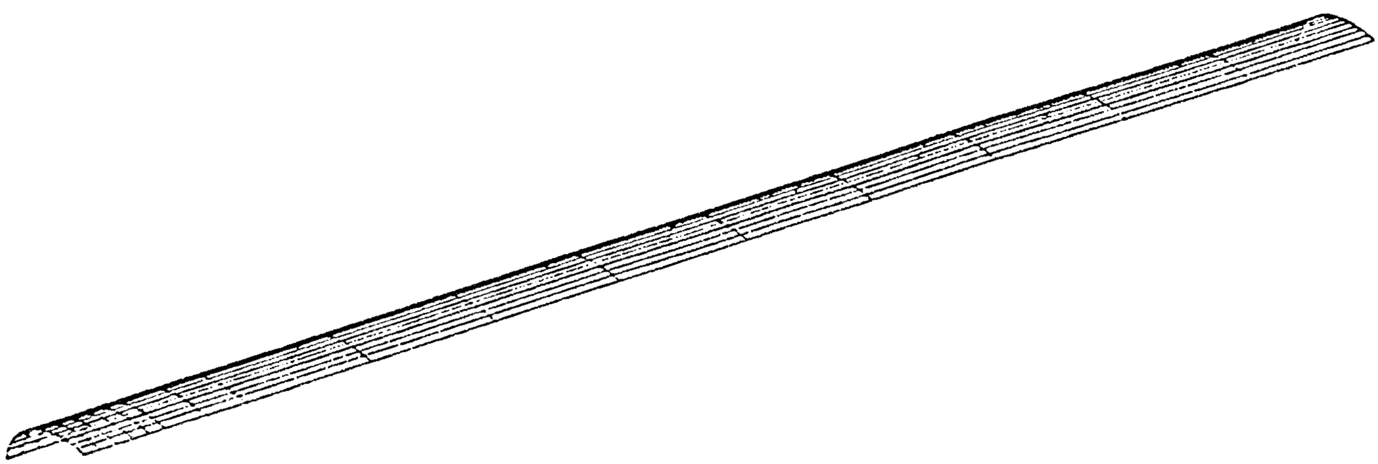
$$\frac{\partial \rho h v_1^2}{\partial \alpha} + \frac{\partial \rho h v_1 v_2}{\partial z}$$

$$= - h\frac{\partial p}{\partial \alpha} - f\rho[v_1^2 + v_2^2]^{1/2}v_1$$

$$\frac{\partial \rho h v_1 v_2}{\partial \alpha} + \frac{\partial \rho h v_2^2}{\partial z}$$

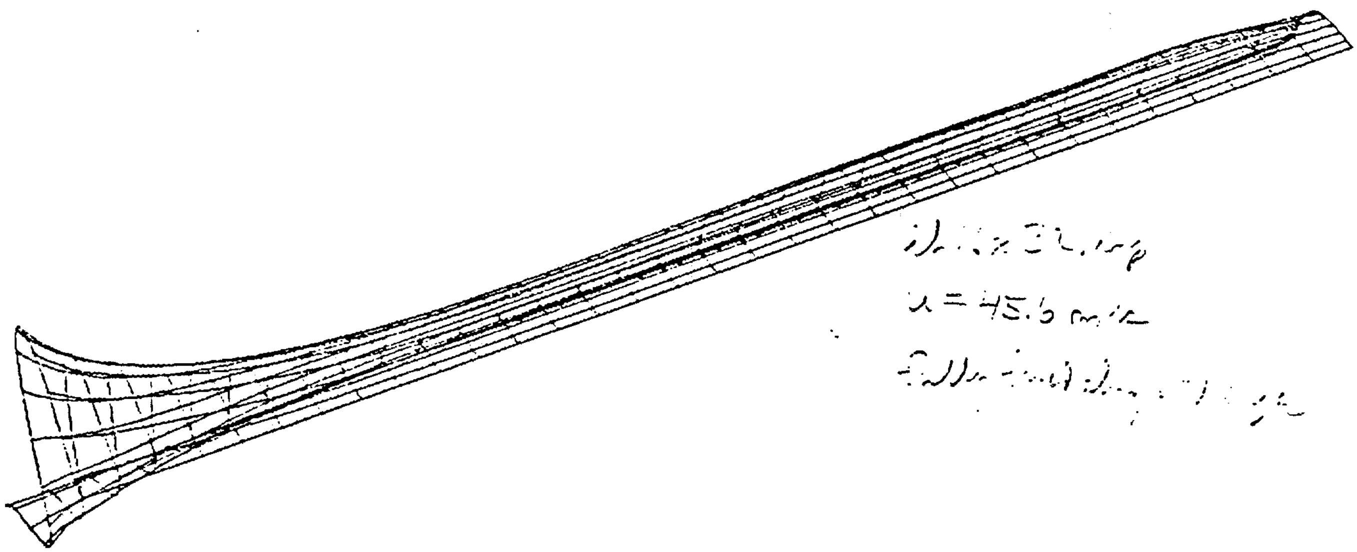
$$= - h\frac{\partial p}{\partial z} - f\rho[v_1^2 + v_2^2]^{1/2}v_2$$

Undeformed Involute Grid



- Only Plate Elements Shown
  One Fluid Element for Each Plate Elm
- Entrance at Left
- Grid Refined at Entrance

Deformed Grid Plot



- Calculated Buckling Mode Shape
- Calculate Buckling for 45.6 m/s
- Deflection Exaggerated for Visibility
- Deflection Much Larger near Entrance

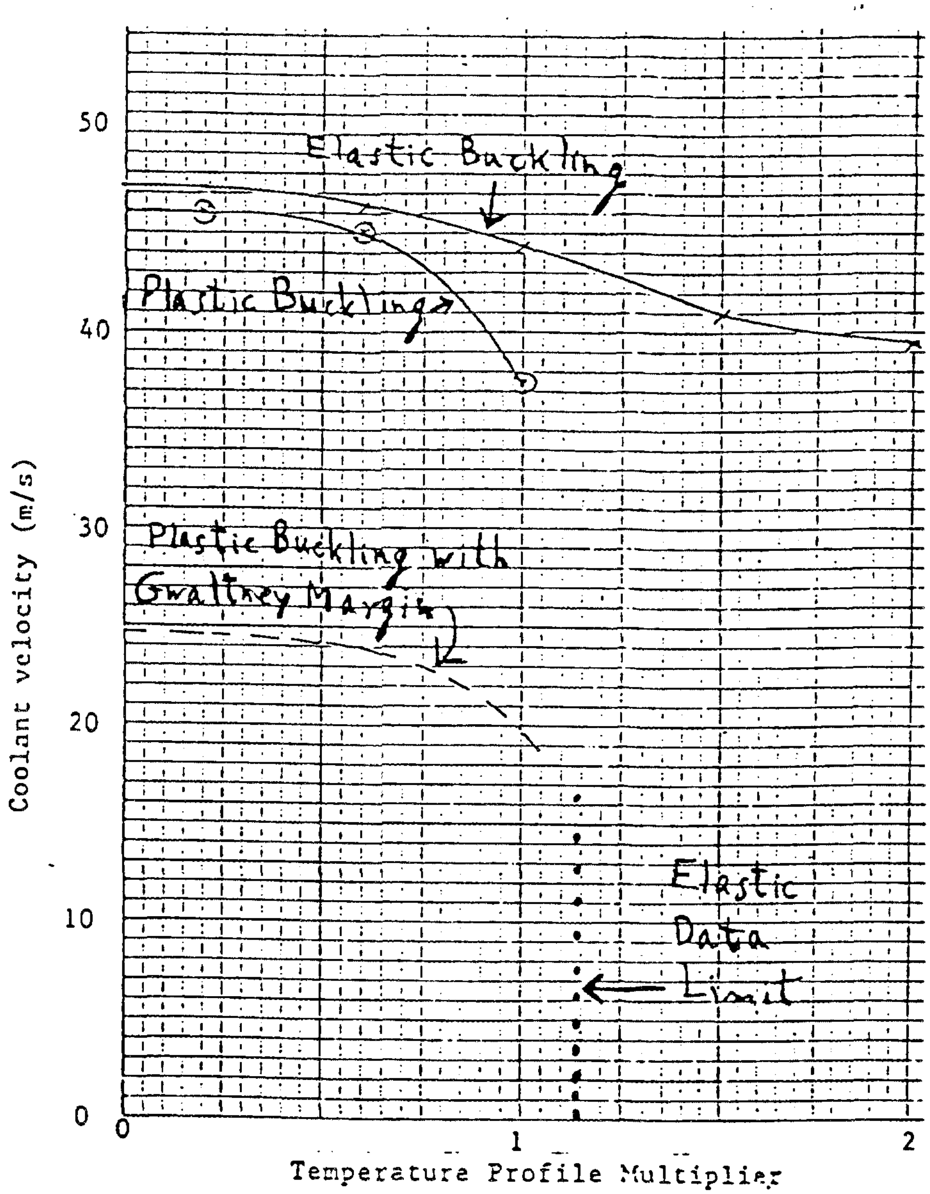
Fuel Plate Temperature Limit

- Raising the Fuel PLate Temperature:
  - Lowers the Elastic Modulus
  - Lowers the Yield Stress
  - Lowers the Tensile Strength
  - Raises the Thermal Stress
- Might Reduce Structural Integrity
- Thermal Stress & Plasticity Added to ABAQUS Plate Model

Thermal-Elastic Buckling

- No Fluid Motion
- No Plate Plasticity
- Result: Calculated Buckling at 9.1 times Estimated Maximum Plate Temp.
- Conclusion: Thermal-Elastic Buckling Not a Problem in ANS

Hydraulic-Thermal-Plastic Buckling



Effect of Relative Temperature on Allowable Coolant Velocity

≥19% Reduction in Allowable Coolant Velocity at Estimated Maximum Plate Temperature

Present Limits Are Preliminary

Revised Calculations Are Planned

Revised Temperatures Are Expected

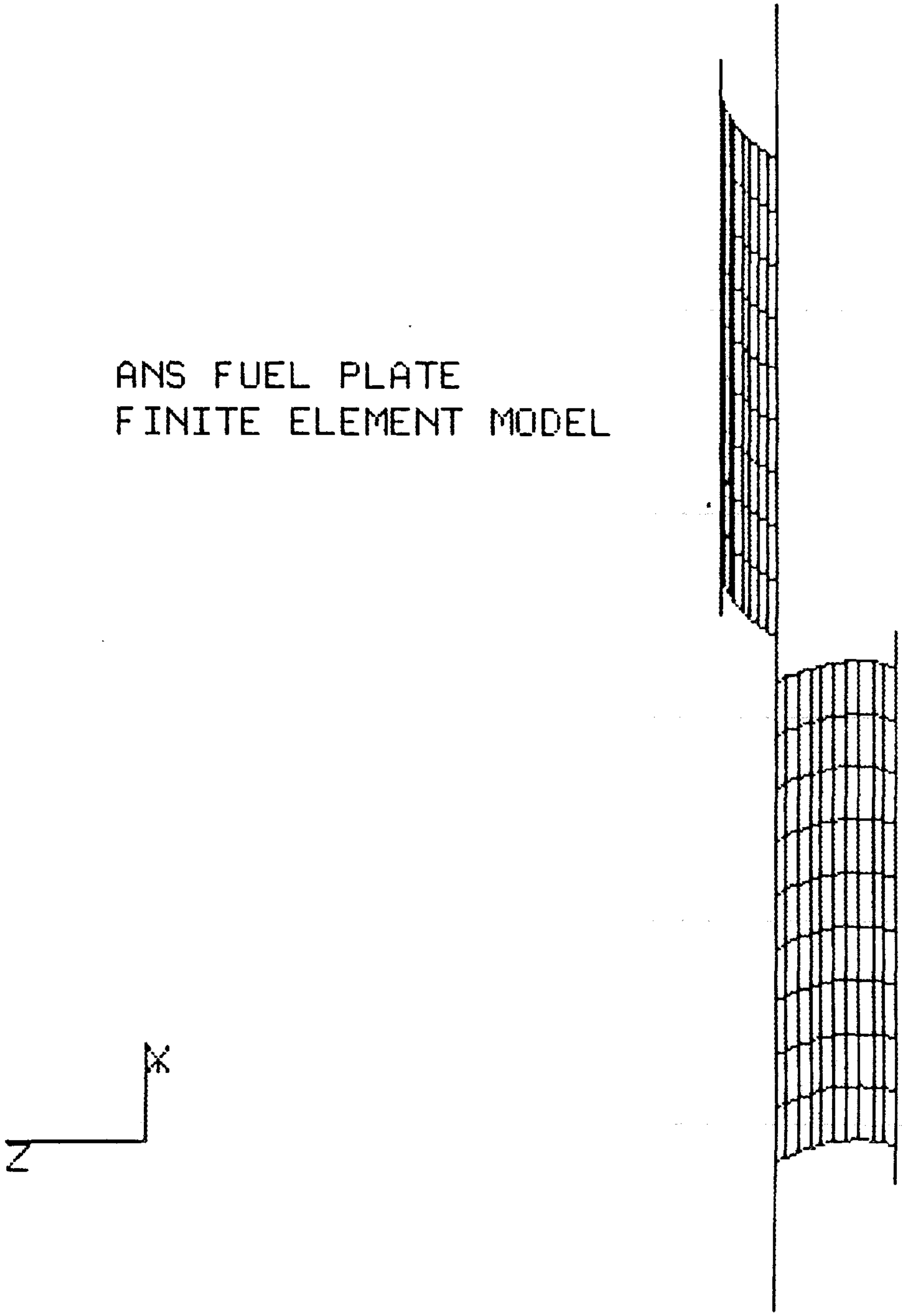
Need More Data on Alum. Plasticity

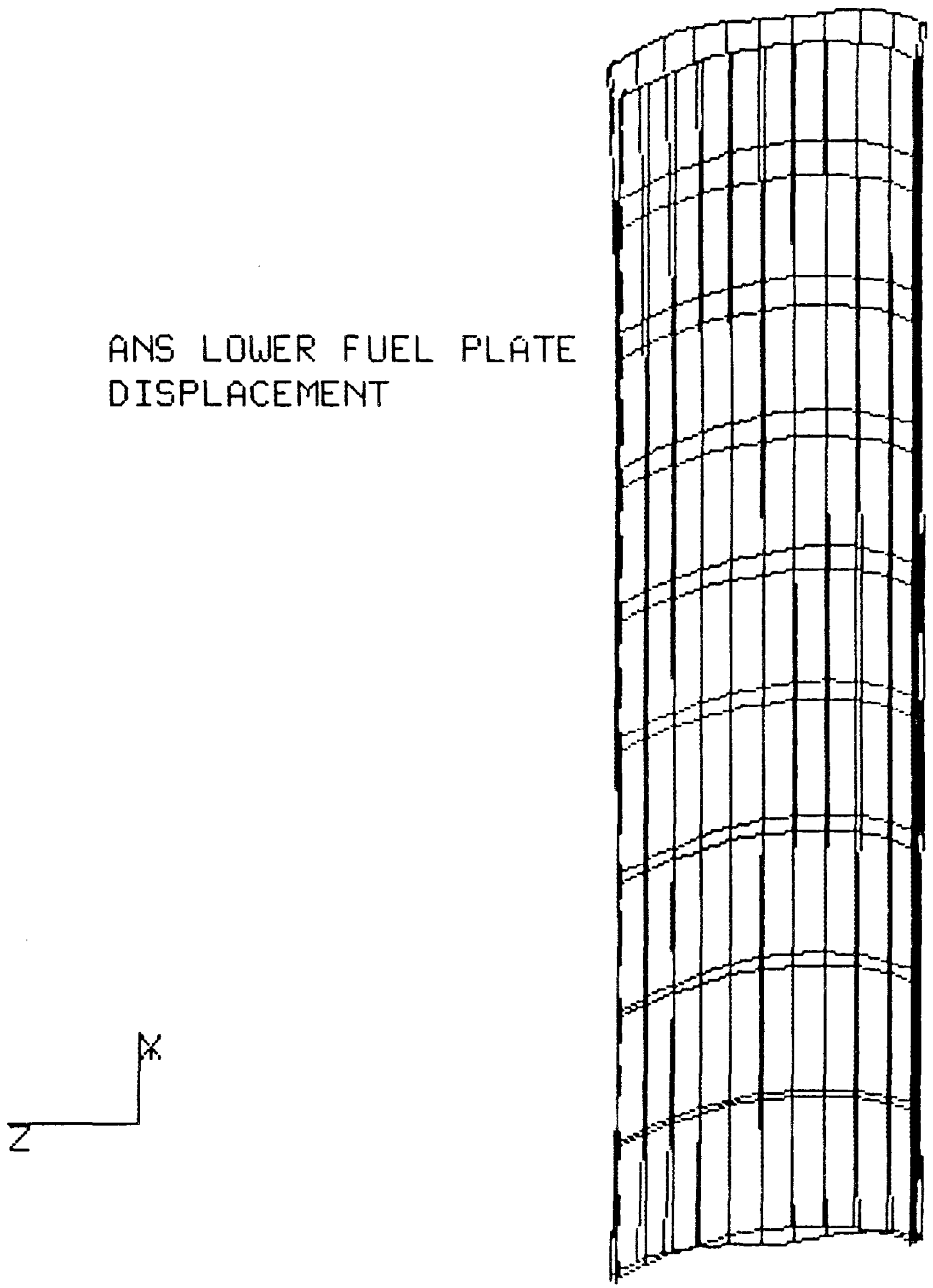
- We Used Data on 6061-O Temper
- ANS Plates Are Still Softer (?)
- Creep of Aluminum?

Elastic Thermal Deflection Calculatior

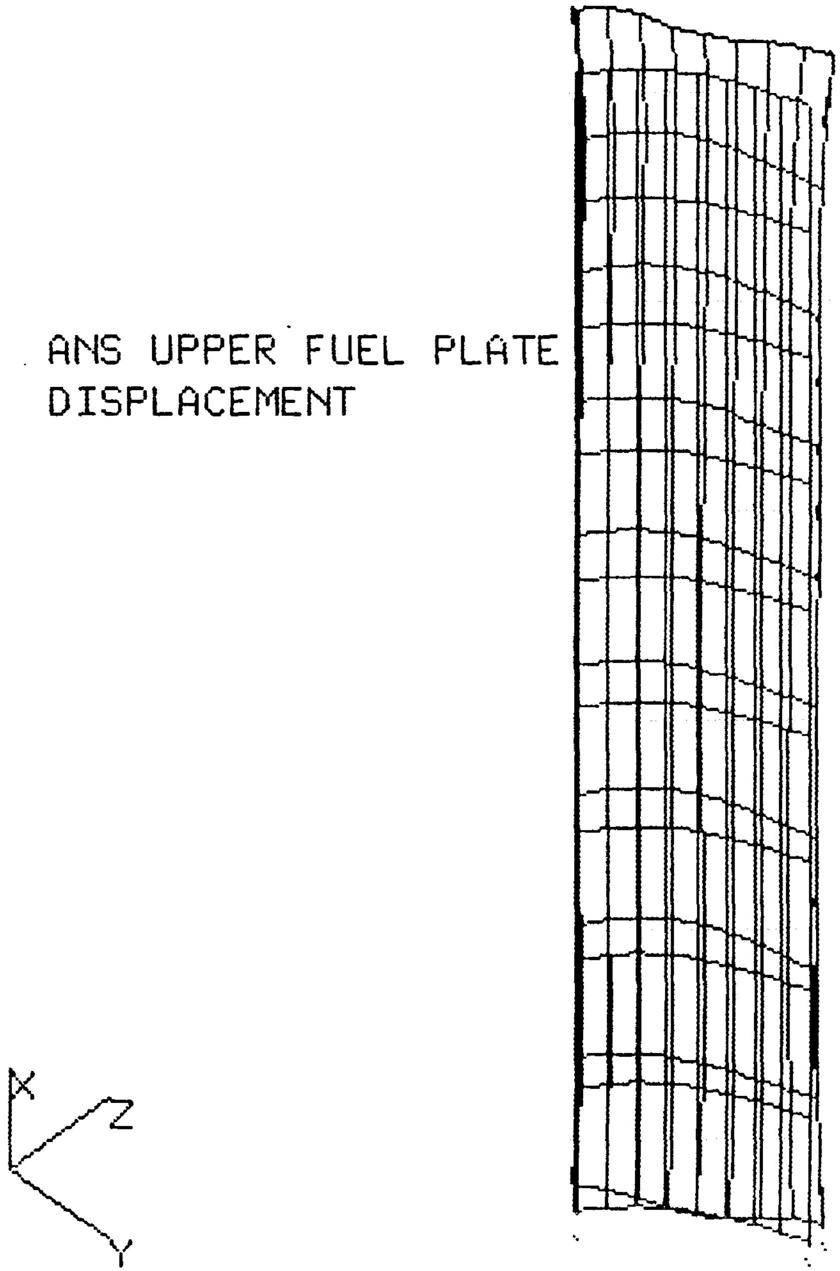
Analyses by C. R. Luttrell

These calculations predict the distortion of the involutes due to heating, rather than a buckling or instability threshold.

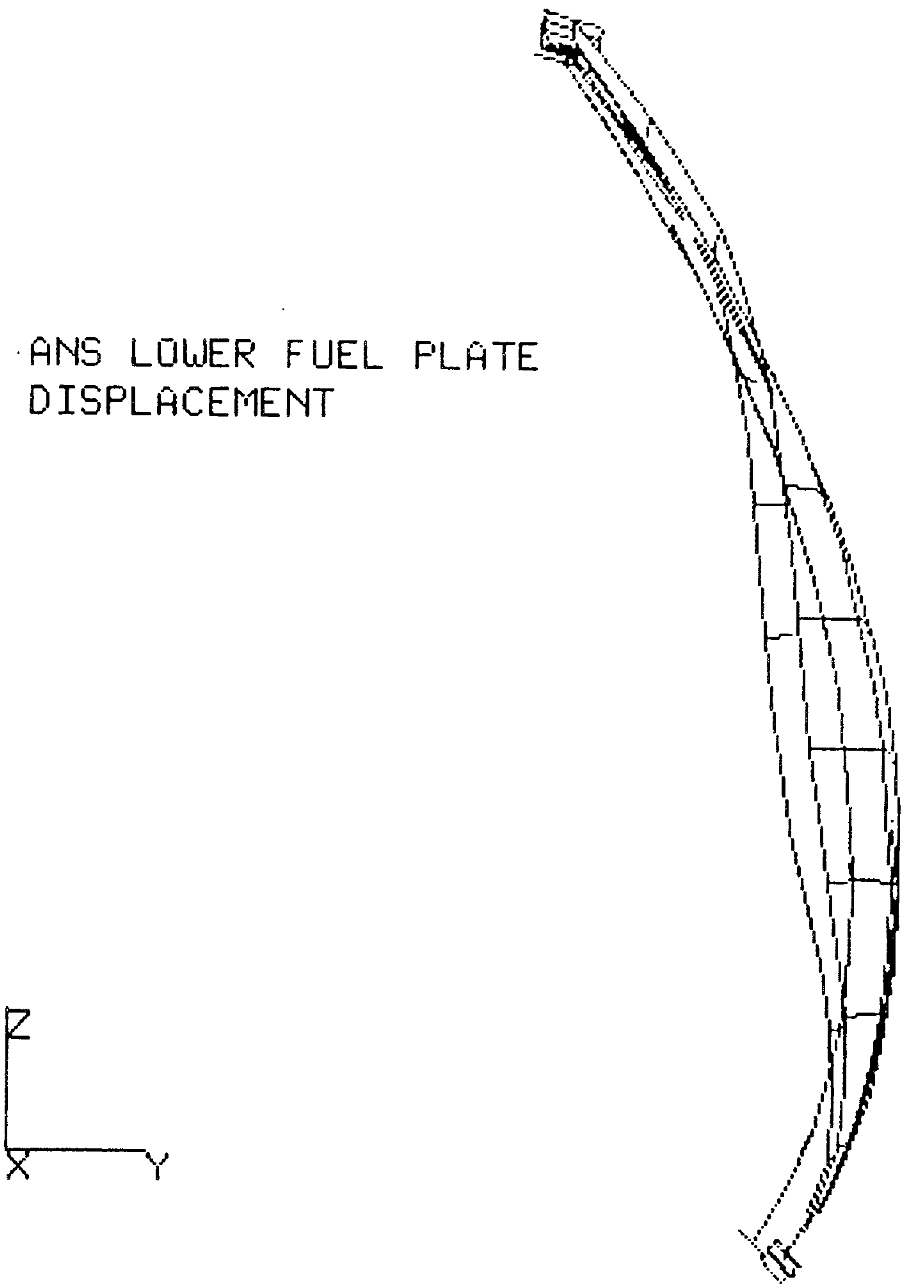

ANS FUEL PLATE
FINITE ELEMENT MODEL
X
Z


ANS LOWER FUEL PLATE
DISPLACEMENT
X
Z

ANS UPPER FUEL PLATE
DISPLACEMENT



ANS LOWER FUEL PLATE
DISPLACEMENT



Appendix E: SIMULATION RESULTS FOR THE EARLY PART OF THE LBLOCA

RELAP5 ADVANCED NEUTRON SOURCE REACTOR (ANSR) SYSTEM MODEL

Norbert Chen

Presented for
ANSR Phenomena Identification and Ranking (PIR) Working Group

September 17, 1991
Fusion Energy Design Center

PURPOSE OF RELAP5 ANSR SYSTEM MODEL DEVELOPMENT IS

- To provide early input to design process

- To perform transient calculations for the ANSR safety analysis report

  - Loss-of-Coolant Accidents (LOCA)
  - Station Blackout Accidents
  - Reactivity Insertion Accidents
  - Natural circulation characteristics

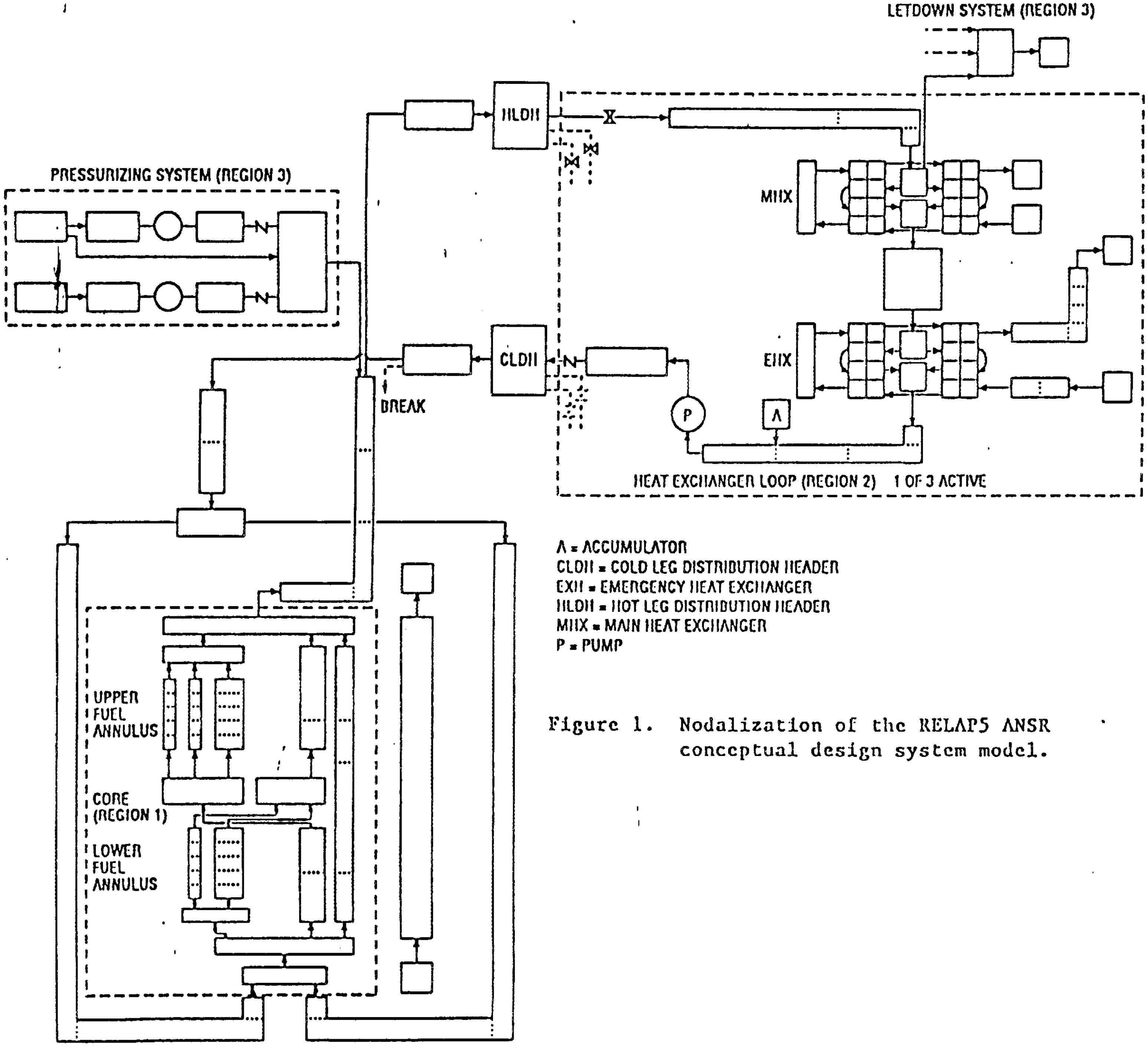
BACKGROUND

- Completed the RELAP5 ANSR preconceptual system model in May 1989
  - Model developed by D. Fletcher of INEL
  - Model reviewed by N. Chen of ORNL
  - Code modified by A. Ruggles of ORNL
- Completed the RELAP5 ANSR conceptual system model in January 1991
  - Preliminary model reviewed (September 1990)
  - Model developed by N. Chen
  - Model reviewed by D. Fletcher
  - Completed a Martin Marietta Award Fee Milestone

THE RELAP5 ANSR CONCEPTUAL DESIGN SYSTEM MODEL CONSISTS OF THREE MAJOR REGIONS

- Core Region
- Heat Exchanger Loop Region
- Pressurizing and Letdown System Region

ORNL-DWG 91M-2554R ETD



A = ACCUMULATOR
CLDH = COLD LEG DISTRIBUTION HEADER
EXH = EMERGENCY HEAT EXCHANGER
HLDH = HOT LEG DISTRIBUTION HEADER
MHX = MAIN HEAT EXCHANGER
P = PUMP

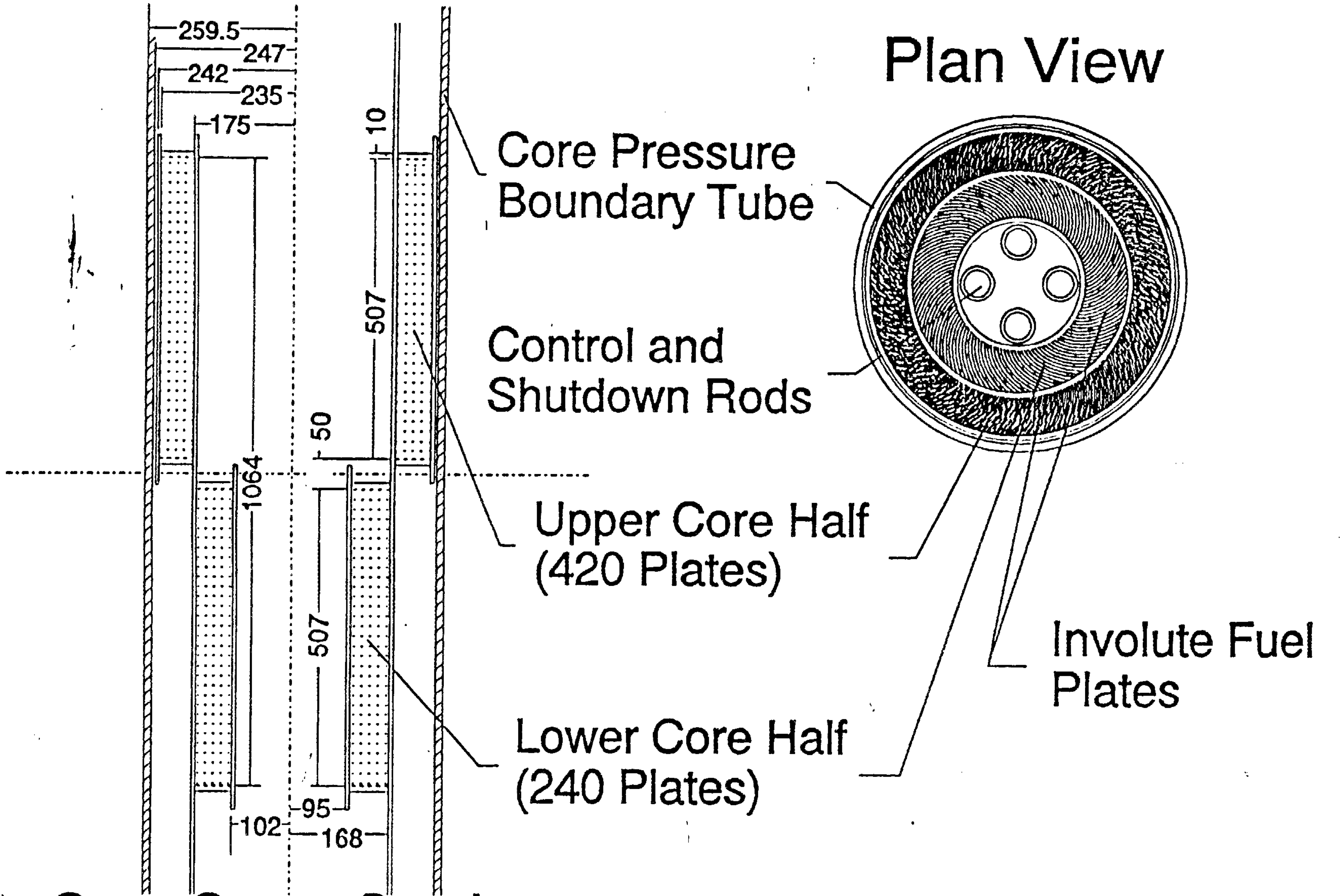
Figure 1. Nodalization of the RELAP5 ANSR conceptual design system model.

CORE MODEL

- Incorporated geometry and parameters based on the conceptual core design of April 1990 (C.D.West)

- Adapted the gamma heating fractions for the structures and fluids and peaking factors for hot streak/stripe as that of INT-1

- Used power density distributions based on the I3 fuel grading at the end of cycle where the limiting condition occurred under steady state calculations



Core Cross Section
(Vertical Midplane)

Dimensions in mm

CORE POWER DISTRIBUTION AMONG VARIOUS METAL AND FLUID REGIONS

95 % to

Fuel meat	85.5 %
Clad	7.6 %
Core coolant	1.9 %

5 % to

Side plate	1.0 %
Center control rods	0.15 %
CPBT wall	0.75 %
Moderator tank coolant	0.3 %
Bypass coolant	1.1 %
Control rod coolant	1.7 %

Table 1. Normalized Power Density of 13 Fuel Grading at the End of Cycle.

Upper Core

Zone	Average Channel	Hot Channel (with Mult. 1.14)	Hot Stripe (with mult. 1.31)
5	1.243	1.517	1.991
4	1.252	1.517	1.886
3	1.261	1.566	1.886
2	1.222	1.614	1.965
1	0.991	1.362	1.900

Lower Core

Zone	Average Channel	Hot Channel (with Mult. 1.14)	Hot Stripe (with mult. 1.31)
5	0.641	0.864	1.074
4	0.651	0.901	1.218
3	0.686	0.963	1.323
2	0.749	1.034	1.454
1	0.780	1.109	1.546

Power Split at EOC

Lower Core 0.294
Upper Core 0.702

CORE MODEL DETAILS AND RATIONAL

Three parallel flow paths represented:

- Lower core

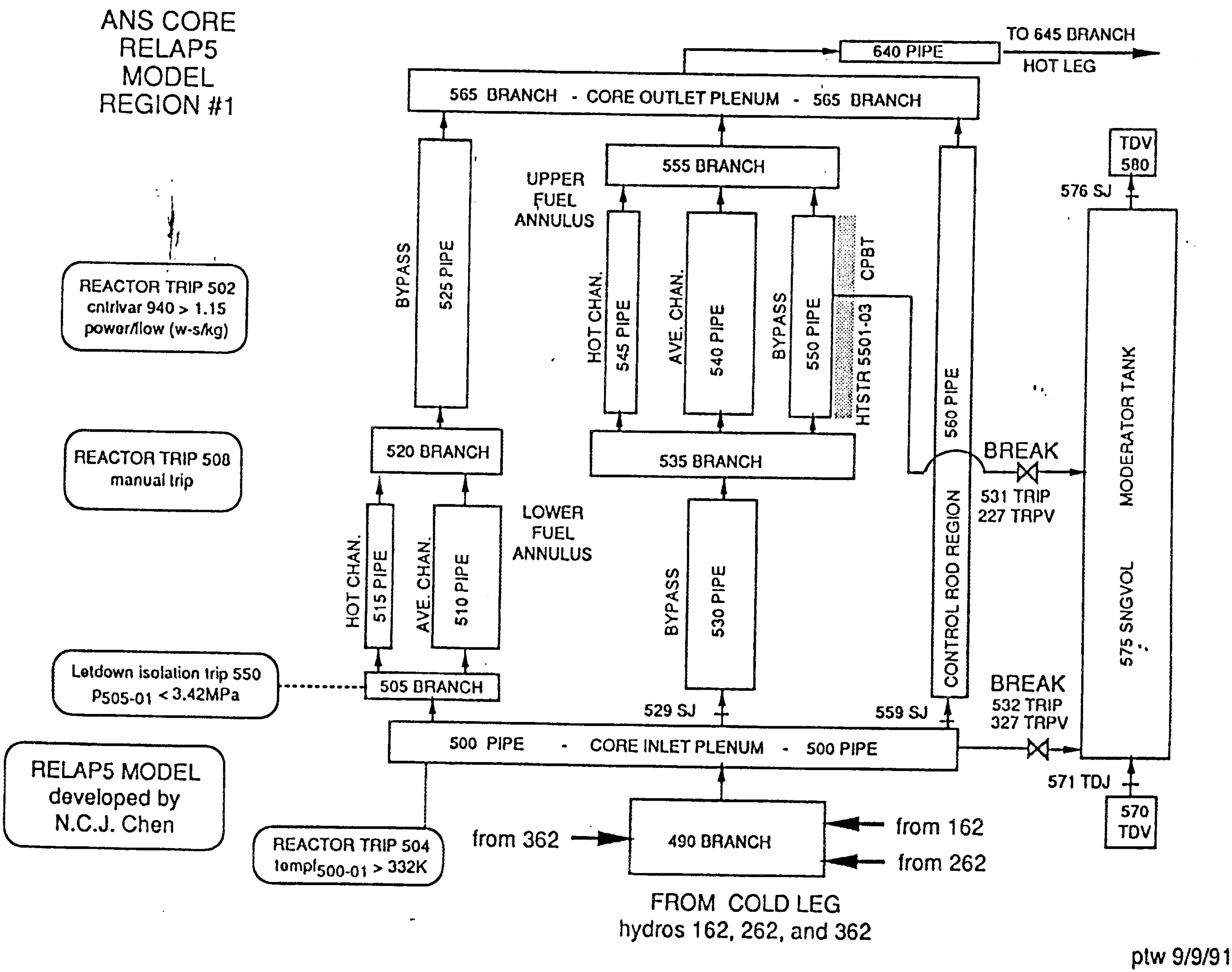
  1 Hot channel to determine maximum bulk temperature rise
  1 Hot stripe to capture flow excursion and critical heat flux (CHF)
  239-channel lumped to determine lower core average coolant behavior
  5-cell a compromise between cost and accuracy

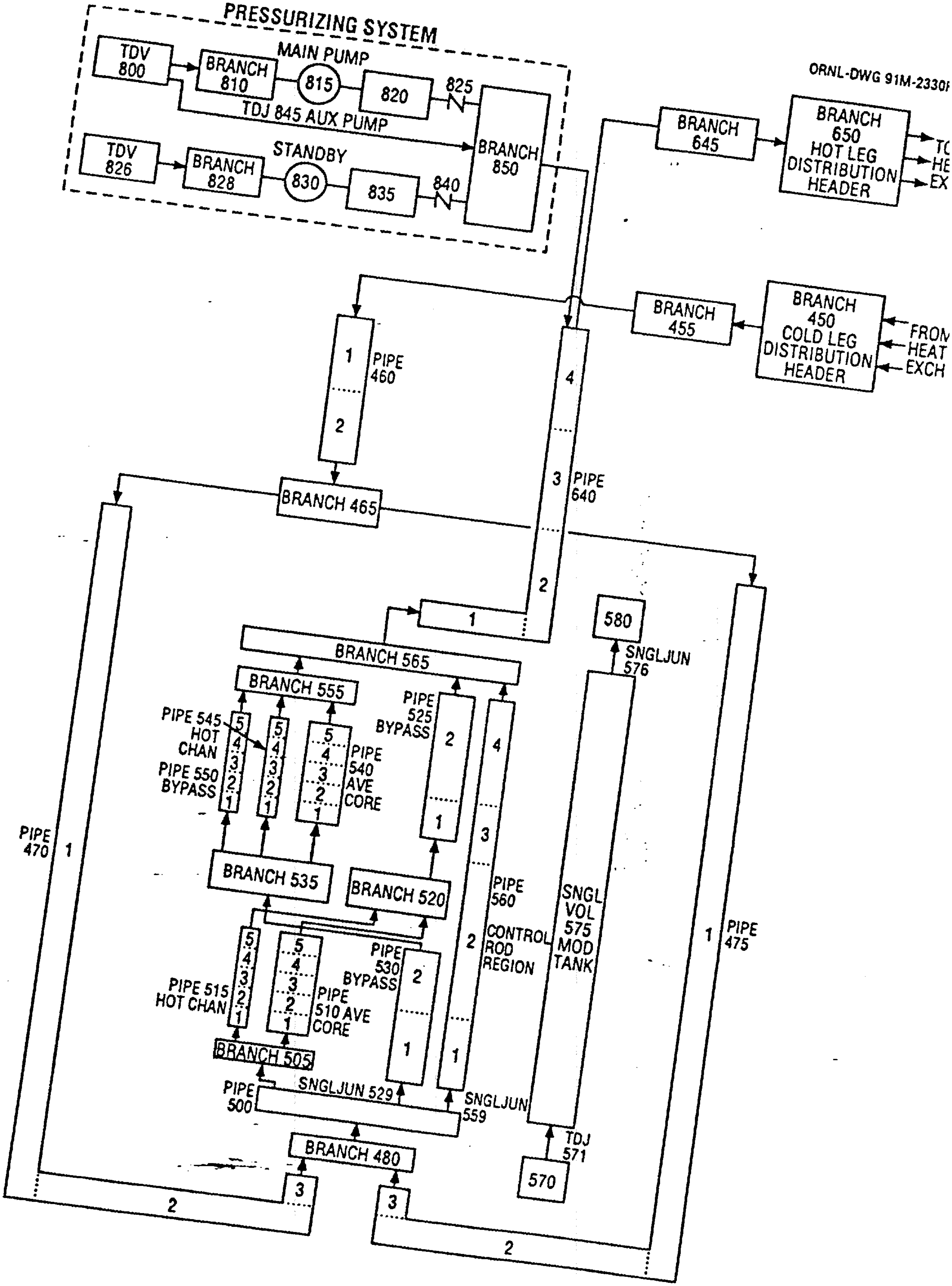
- Upper core

  1 Hot channel
  1 Hot stripe
  419-channel lumped to determine upper core average coolant behavior
  5-cell

- Central control rod channel

  4-cell
  4 Hollow cylindrical rods
  Orificing resistance at channel exit to avoid boiling and maintain the desired velocity


ANS CORE
RELAP5
MODEL
REGION #1
640 PIPE
TO 645 BRANCH
HOT LEG
565 BRANCH - CORE OUTLET PLENUM - 565 BRANCH
555 BRANCH
UPPER FUEL ANNULUS
TDV 580
576 SJ
BYPASS
525 PIPE
HOT CHAN.
545 PIPE
AVE. CHAN.
540 PIPE
BYPASS
550 PIPE
CPBT
HTSTR 5501-03
560 PIPE
MODERATOR TANK
REACTOR TRIP 502
cntrlvar 940 > 1.15
power/flow (w-s/kg)
520 BRANCH
535 BRANCH
BREAK
531 TRIP
227 TRPV
REACTOR TRIP 508
manual trip
LOWER FUEL ANNULUS
HOT CHAN.
515 PIPE
AVE. CHAN.
510 PIPE
BYPASS
530 PIPE
CONTROL ROD REGION
575 SNGVOL
Letdown isolation trip 550
P505-01 < 3.42MPa
505 BRANCH
529 SJ
559 SJ
BREAK
532 TRIP
327 TRPV
500 PIPE - CORE INLET PLENUM - 500 PIPE
571 TDJ
RELAP5 MODEL
developed by
N.C.J. Chen
REACTOR TRIP 504
tempf500-01 > 332K
from 362
490 BRANCH
from 162
from 262
570 TDV
FROM COLD LEG
hydros 162, 262, and 362
ptw 9/9/91

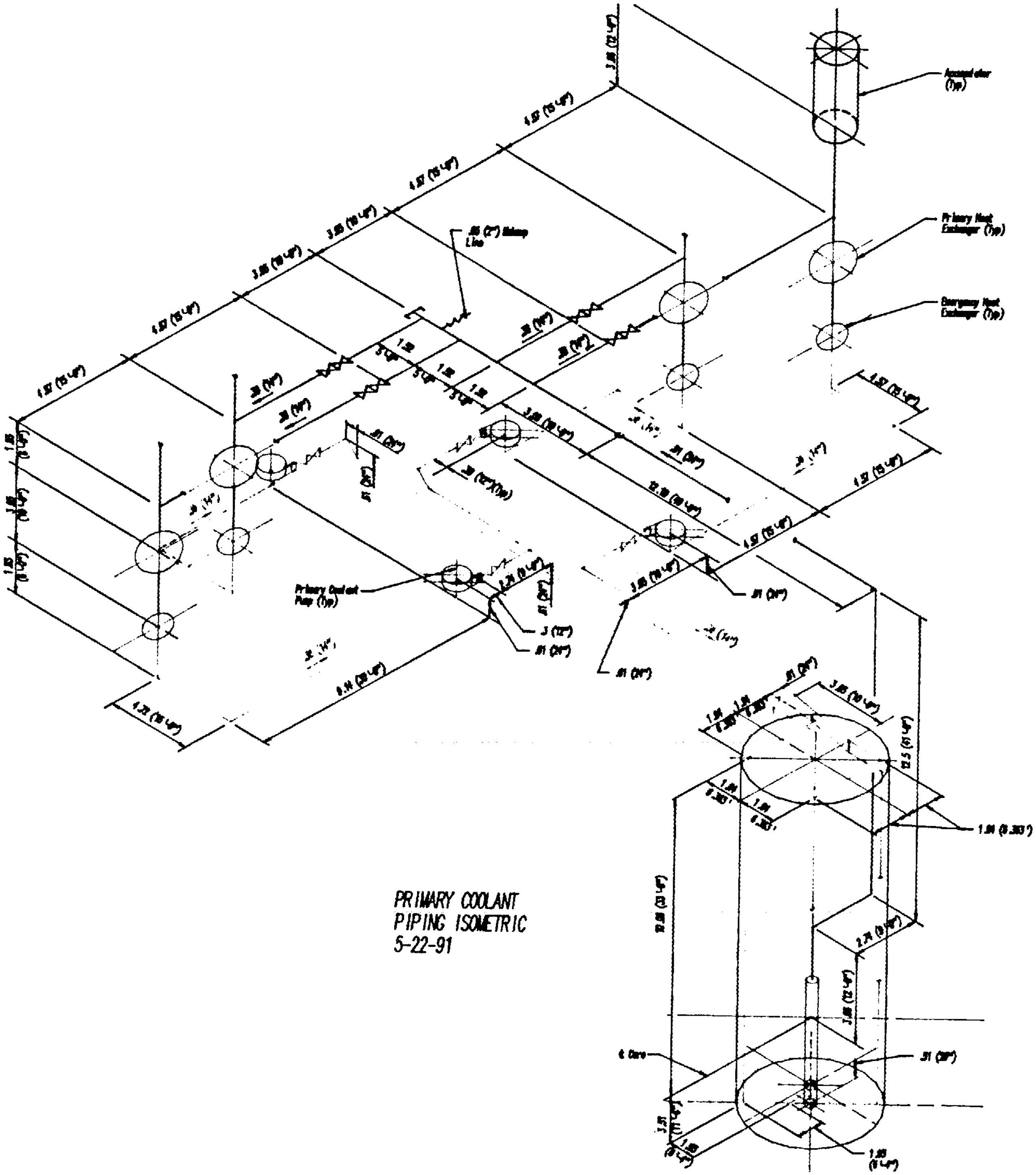


ORNL-DWG 91M-2330
PRESSURIZING SYSTEM
TDV 800
BRANCH 810
MAIN PUMP
815
820
825
TDJ 845 AUX PUMP
TDV 826
BRANCH 828
STANDBY
830
835
840
BRANCH 850
BRANCH 645
BRANCH 650 HOT LEG DISTRIBUTION HEADER
BRANCH 455
BRANCH 450 COLD LEG DISTRIBUTION HEADER
FROM HEAT EXCH
PIPE 460
PIPE 640
BRANCH 465
580
SNGLJUN 576
BRANCH 565
BRANCH 555
PIPE 545 HOT CHAN
PIPE 550 BYPASS
PIPE 540 AVE CORE
PIPE 525 BYPASS
PIPE 470
PIPE 475
BRANCH 535
BRANCH 520
PIPE 560 CONTROL ROD REGION
SNGL VOL 575 MOD TANK
PIPE 530 BYPASS
PIPE 515 HOT CHAN
PIPE 510 AVE CORE
BRANCH 505
PIPE 500
SNGLJUN 529
SNGLJUN 559
BRANCH 480
TDJ 571
570

LOOP MODEL

Incorporated component and piping configuration based on the reference coolant system of May 1991 (G.R.McNutt)

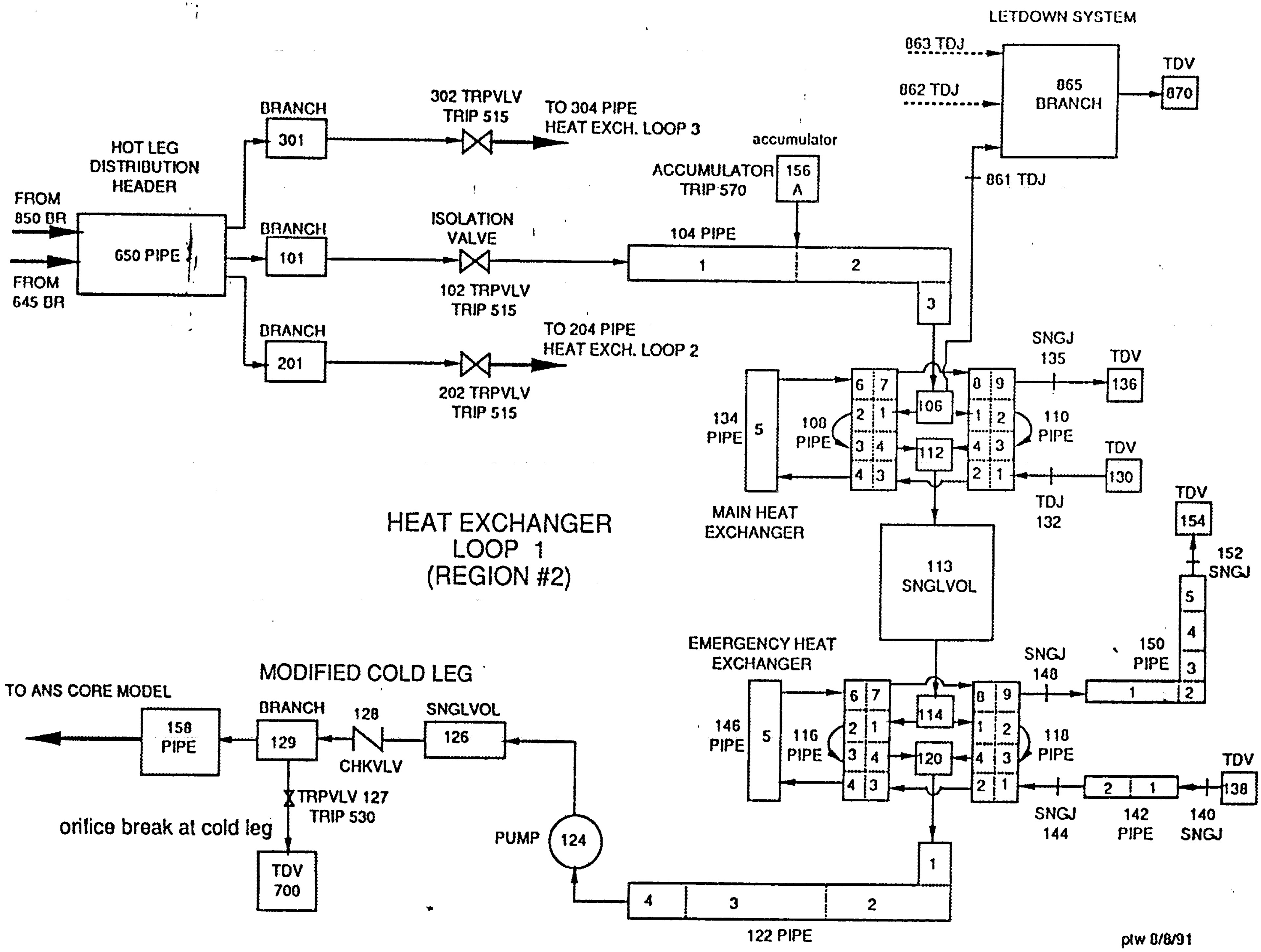
- Elimination of the cold leg distribution header
- Main heat exchanger operated in series with emergency heat exchanger
- Emergency heat exchanger cooled by natural convection
- Accumulator installed upstream of the main heat exchanger
- Flow diode located near the core inlet



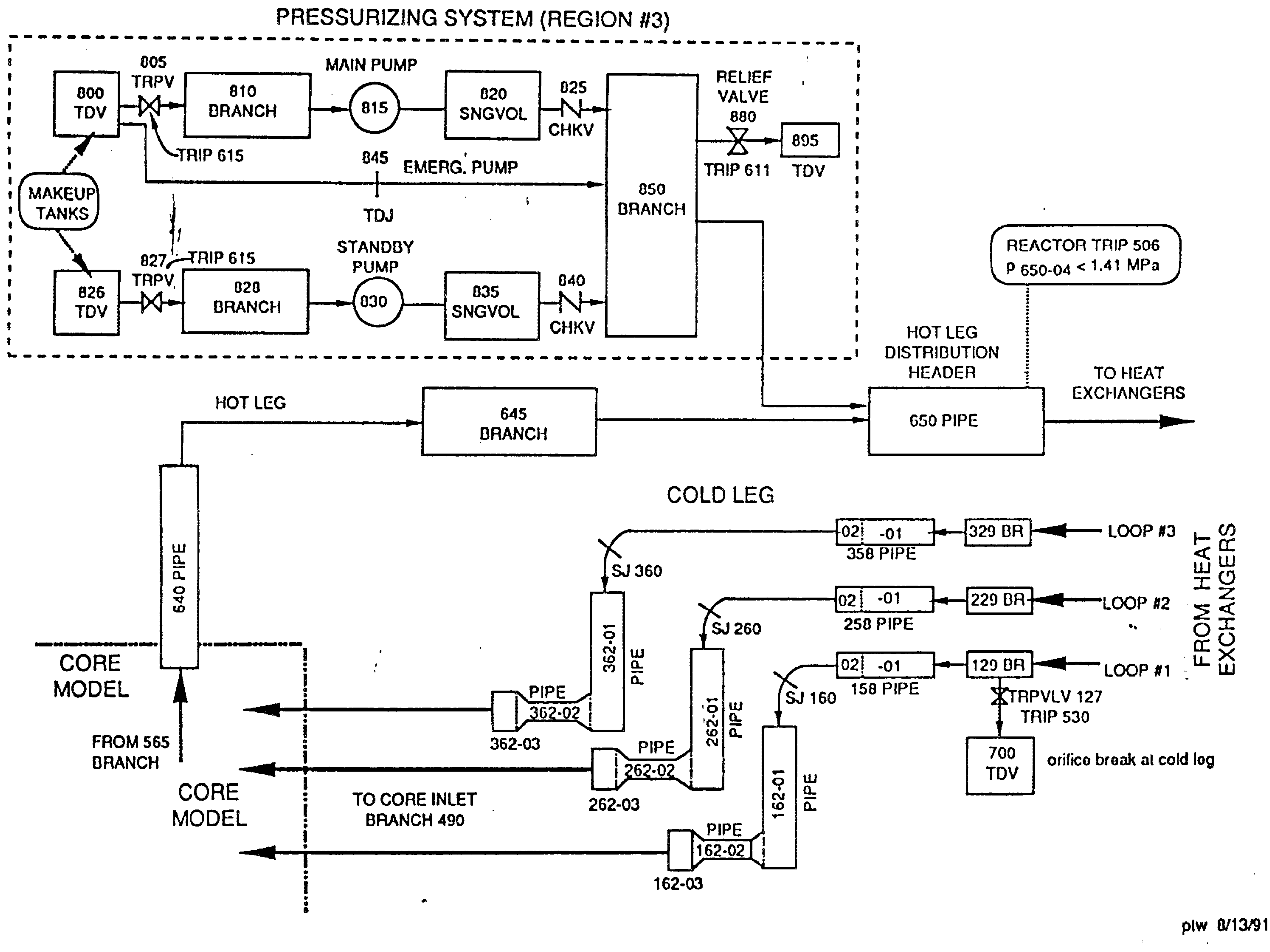
PRIMARY COOLANT
PIPING ISOMETRIC
5-22-91

LOOP MODEL DETAILS AND RATIONAL

- Followed standard INEL RELAP5 nodalization for PWR
- Modeled three normal-operation coolant loops separately to provide flexibility in modeling accident scenarios
- Excluded the fourth loop (standby) in the model
- Increased number of cells in components where the fluid stratification becomes important



LETDOWN SYSTEM
863 TDJ
862 TDJ
865 BRANCH
TDV
870
BRANCH
301
302 TRPVLV
TRIP 515
TO 304 PIPE
HEAT EXCH. LOOP 3
HOT LEG
DISTRIBUTION
HEADER
FROM
850 BR
FROM
645 BR
650 PIPE
accumulator
ACCUMULATOR
TRIP 570
156
A
861 TDJ
BRANCH
101
ISOLATION
VALVE
102 TRPVLV
TRIP 515
104 PIPE
1
2
3
BRANCH
201
202 TRPVLV
TRIP 515
TO 204 PIPE
HEAT EXCH. LOOP 2
SNGJ
135
TDV
136
134
PIPE
5
108
PIPE
106
112
110
PIPE
TDV
130
TDJ
132
MAIN HEAT
EXCHANGER
HEAT EXCHANGER
LOOP 1
(REGION #2)
113
SNGLVOL
TDV
154
152
SNGJ
150
PIPE
SNGJ
148
EMERGENCY HEAT
EXCHANGER
146
PIPE
116
PIPE
114
120
118
PIPE
TDV
138
SNGJ
144
142
PIPE
140
SNGJ
MODIFIED COLD LEG
TO ANS CORE MODEL
158
PIPE
BRANCH
129
128
CHKVLV
SNGLVOL
126
TRPVLV 127
TRIP 530
orifice break at cold leg
TDV
700
PUMP
124
122 PIPE
plw 8/8/91



PRESSURIZING SYSTEM (REGION #3)
800 TDV
805 TRPV
TRIP 615
810 BRANCH
MAIN PUMP
815
820 SNGVOL
825 CHKV
850 BRANCH
RELIEF VALVE 880
TRIP 611
895 TDV
845 EMERG. PUMP
TDJ
MAKEUP TANKS
826 TDV
827 TRPV
TRIP 615
828 BRANCH
STANDBY PUMP
830
835 SNGVOL
840 CHKV
REACTOR TRIP 506
P 650-04 < 1.41 MPa
HOT LEG DISTRIBUTION HEADER
TO HEAT EXCHANGERS
HOT LEG
645 BRANCH
650 PIPE
640 PIPE
CORE MODEL
FROM 565 BRANCH
CORE MODEL
COLD LEG
02 -01
358 PIPE
329 BR
LOOP #3
SJ 360
02 -01
258 PIPE
229 BR
LOOP #2
SJ 260
02 -01
158 PIPE
129 BR
LOOP #1
SJ 160
FROM HEAT EXCHANGERS
TRPVLV 127
TRIP 530
700 TDV
orifice break at cold leg
362-01 PIPE
PIPE 362-02
362-03
262-01 PIPE
PIPE 262-02
262-03
162-01 PIPE
PIPE 162-02
162-03
TO CORE INLET BRANCH 490
plw 8/13/91

ACCUMULATOR MODEL ASSUMPTIONS

- Nitrogen Charged
- Heavy water filled
- Fluid at water pool temperature (308 K)
- Total tank volume = 5 m^3
- A tank length-to-diameter ratio of 3
- Initial gas space chosen such that an isentropic expansion to atmospheric pressure will not drain the tank

COMMON CHARACTERISTICS OF THE MAIN AND EMERGENCY HEAT EXCHANGER

- Shell and tube type
- Primary coolant on shell side, secondary on tube side
- Split flow two-pass
- Mounted horizontally
- 1/3 flow per exchanger

DIMENSIONS OF THE MAIN HEAT EXCHANGER

- Tube dimensions

number of tube	7304
OD	19 mm (3/4 ")
length	9.15 m (30 ')

- Shell dimensions

OD	2.235 m (88")
ID	2.185 m (86")

CHARACTERISTICS OF THE EMERGENCY HEAT EXCHANGER

- Operated in series with the main heat exchanger
- Cooled by natural convection
- Tube dimension

number of tube	250
OD	50 mm (2 ")
length	6.1 m (20 ')

- Shell dimensions

OD	1.168 m (46 ")
ID	1.1398 m (44.875")

CENTRIFUGAL MAIN CIRCULATION PUMP MODELED BASED ON HFIR

- The single-phase homologous curves generated from new pump design data (i.e. three-quadrant Byron Jackson design curves)
- Two-phase corrections based on Semiscale data
- Coastdown curve to battery-operated pony motor similar to that of the HFIR
- Pump cavitation model developed by M. Wendel of ORNL implemented

CHARACTERISTICS OF THE HFIR AND ANSR MAIN CIRCULATION PUMPS

Vertical shaft centrifugal pump

	HFIR	ANSR
Capacity (gpm)	5000	12,000 (821 kg/s)
Head (ft)	365 ft of H20	842 ft of D20 (257 m)
Speed (rpm)	1780 (187 rad/s)	2021 (212 rad/s)
Pony Motor		
speed	270 rpm (15% of nominal)	

MAIN CIRCULATION PUMP COASTDOWN TIMES

	HFIR	ANSR
50% of nominal speed	2 s	2 s
15% of nominal speed (pony motor)	9 s	15 s

PRESSURIZING SYSTEM MODEL

- Main and standby pump characteristics and rated conditions scaled up from that of the HFIR
- Injection flow drawn from a constant pressure and temperature heavy-water tank
- The standby pump started and ramped up linearly after letdown isolated
- Pumps tripped when tank drained
- Check valves to prevent backflow of the primary coolant into the pressurizing system

PRIMARY SYSTEM PRESSURE CONTROLLED BY THROTTLING THE LETDOWN VALVES

- Letdown flow extracted from the inlet plenum of the three main heat exchangers
- Letdown modeled using Time-Dependent Junction components
- Letdown assumed to be isolated by closure of block valves upon reaching the core inlet pressure setpoint

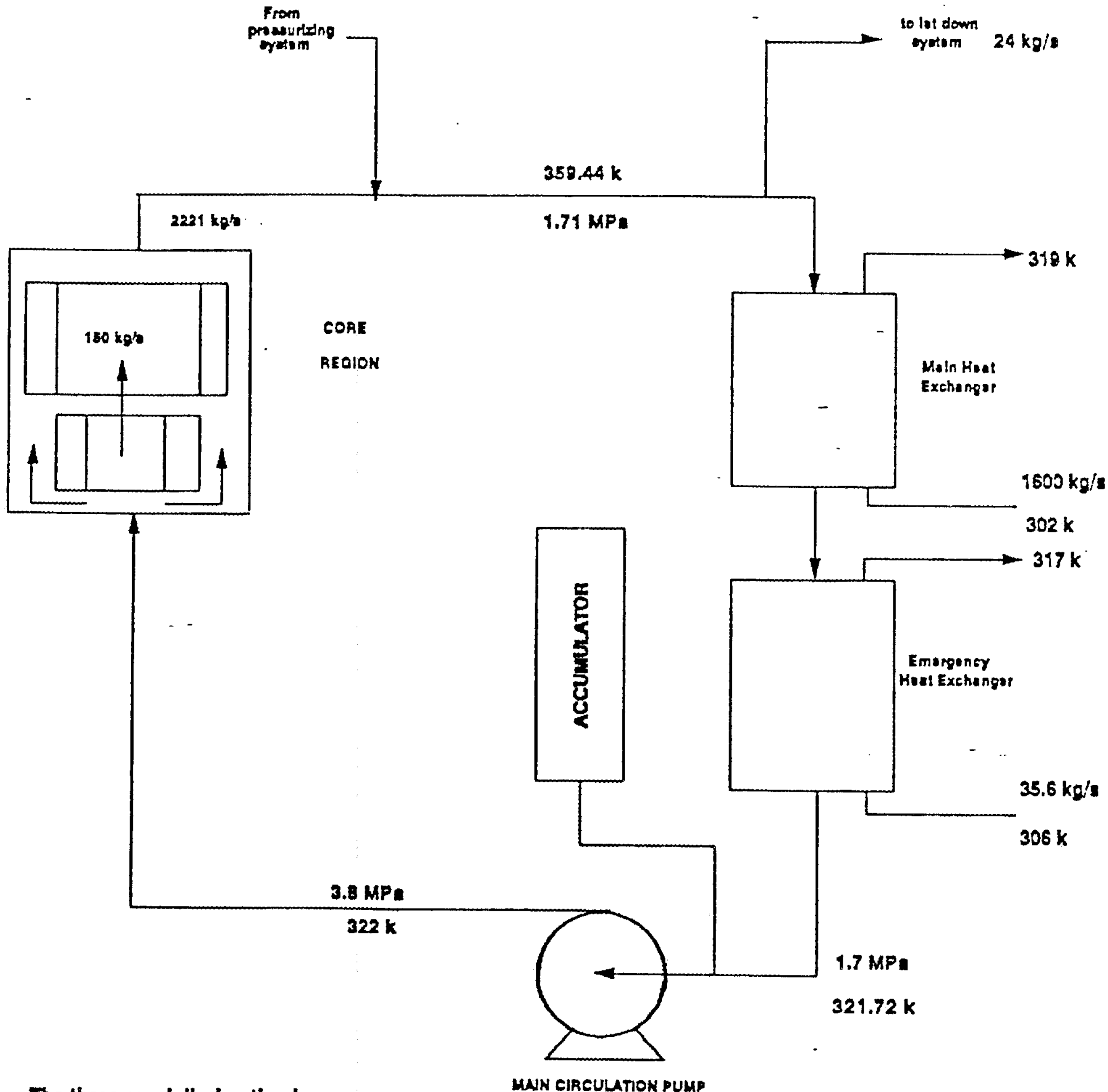
PROTECTION SYSTEM SETPOINTS AND RESPONSE

Modeled based on J.A.Anderson memo of January 1991

- Reactor scrammed by any one the following conditions:

Sensor	Setpoints	Response time (ms)	Location
Reactor outlet pressure	80% (1.68 Mpa)	30	top of the hot leg riser
Reactor inlet temperature	120% (58.8 %)	1250	the EHX outlet
Flux Flow	flux 1.15 5 flow 750		

- Main circulation pump tripped to pony motor when the suction pressure drops below one atmosphere



The three modelled active loops
are shown here as combined into one.

SUMMARY

The RELAP5 ANSR system model has been documented, externally reviewed, and used as a reference model for ORNL and INEL in steady-state and transient analyses

- ORNL studies

  Maximum permissible power calculations
  Pipe breaks at pump discharge
  CPBT inlet and outlet breaks
  Breaks upstream and downstream of the flow diodes

- INEL studies

  Sensitivity of nodalization scheme
  Letdown isolation model improvement
  Time-dependent heat generation used in CPBT and control rod
  Multiple-accumulator interactions

ANS RELAP5 Mod 2.5
THERMAL-HYDRAULICS MODEL

model developed by
N. C. J. Chen

LOCA simulations by
N. C. J. Chen and P. T. Williams
RELAP5SRL-C/V3e

ANS RELAP5 MODEL
LOCA SIMULATIONS

- CURRENTLY INVESTIGATING SHARP-EDGED ORIFICE BREAKS AT THE CORE INLET
- MEDIUM SIZED BREAK (6 in. diam.) FROM CORE INLET PLENUM TO REFLECTOR TANK
- SIMULATION INCLUDES RELAP CROSS FLOW MODEL WITH SHARP EDGED ORIFICE LOSS COEFFICIENTS BASED ON STANDARD HYDRAULIC DATA

ptw 9/17/91

ANS RELAP5 MODEL
LOCA SIMULATIONS

SUMMARY OF PRELIMINARY RESULTS FOR MEDIUM-SIZED (6 in. diam.) ORIFICE BREAKS AT THE CORE INLET PLENUM

- HIGH FREQUENCY PRESSURE OSCILLATIONS OCCUR AT BREAK SITE FOR APPROXIMATELY 20 msec AFTER BREAK.
- THESE PRESSURE OSCILLATIONS CAUSE A FLOW EXCURSION AT THE OUTLET OF THE UPPER FUEL HOT STRIPE WITHIN 5 msec OF THE BREAK.
- CURRENT ANALYSIS EFFORTS ARE FOCUSED ON ATTEMPTING TO DETERMINE IF THESE OSCILLATIONS AND THE MODEL'S RESPONSE TO THEM ARE PHYSICALLY REALISTIC.

ptw 9/17/91

ANS RELAP5 MODEL: 6 in. diam. break
SEQUENCE OF EVENTS TABLE

time	event description	condition for trip
20.00100	break opens	
20.00152	letdown isolation	p550<3.42MPa
20.00172	upper hot stripe	
20.00187	lower hot stripe	
20.02288	reactor trip signal	p650<1.41MPa
20.04888	pump 1 trip	p120<0.1MPa
20.08300	reactor SCRAM	
20.54036	pump 3 trip	p320<0.1MPa
20.62265	pump 2 trip	p220<0.1MPa

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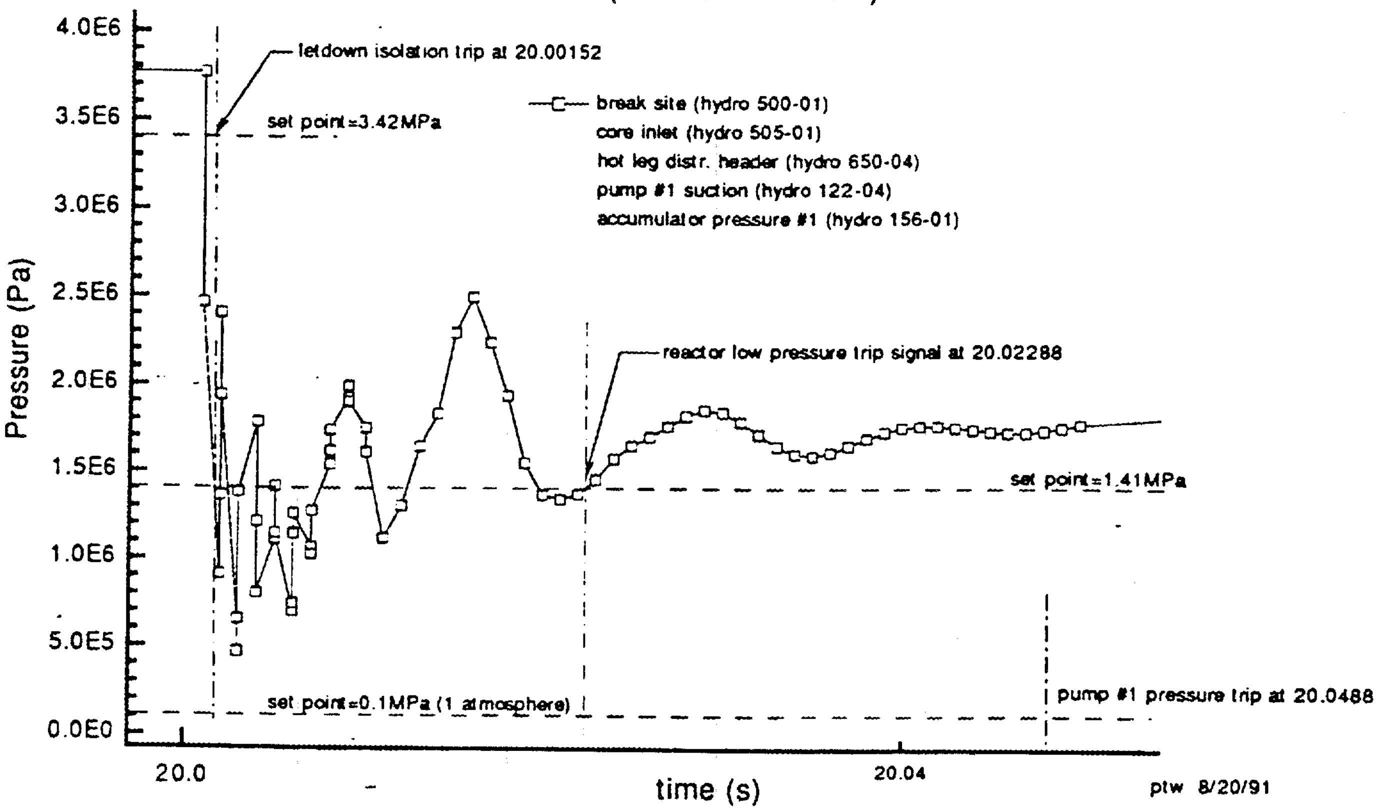
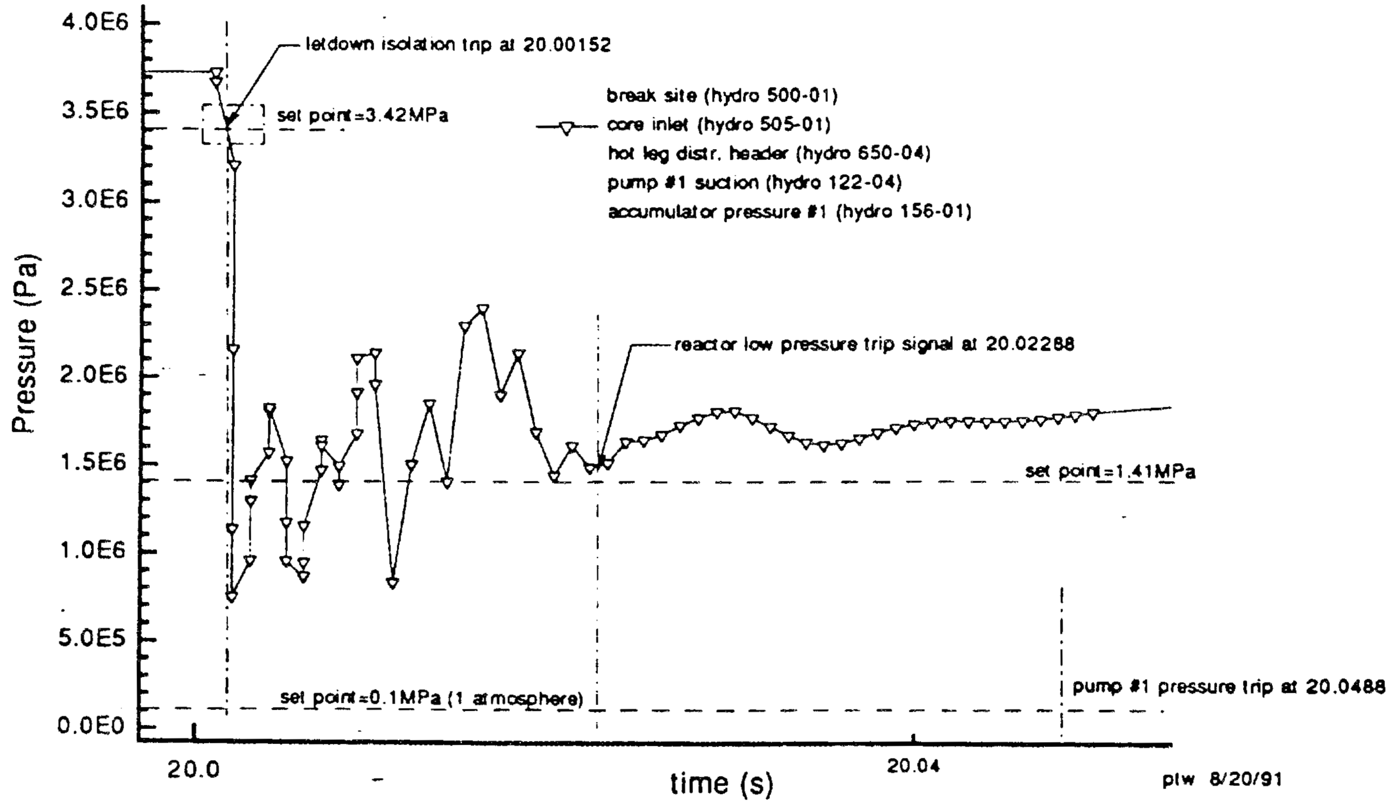
ANS RELAP5 Model: RELAP5SRL-C/V3e
Orifice Break from Core Inlet (at CPBT) to Moderator Tank
(6 in. diam. break)



ANS RELAP5 Model: RELAP5SRL-C/V3e
Orifice Break from Core Inlet (at CPBT) to Moderator Tank
(6 in. diam. break)



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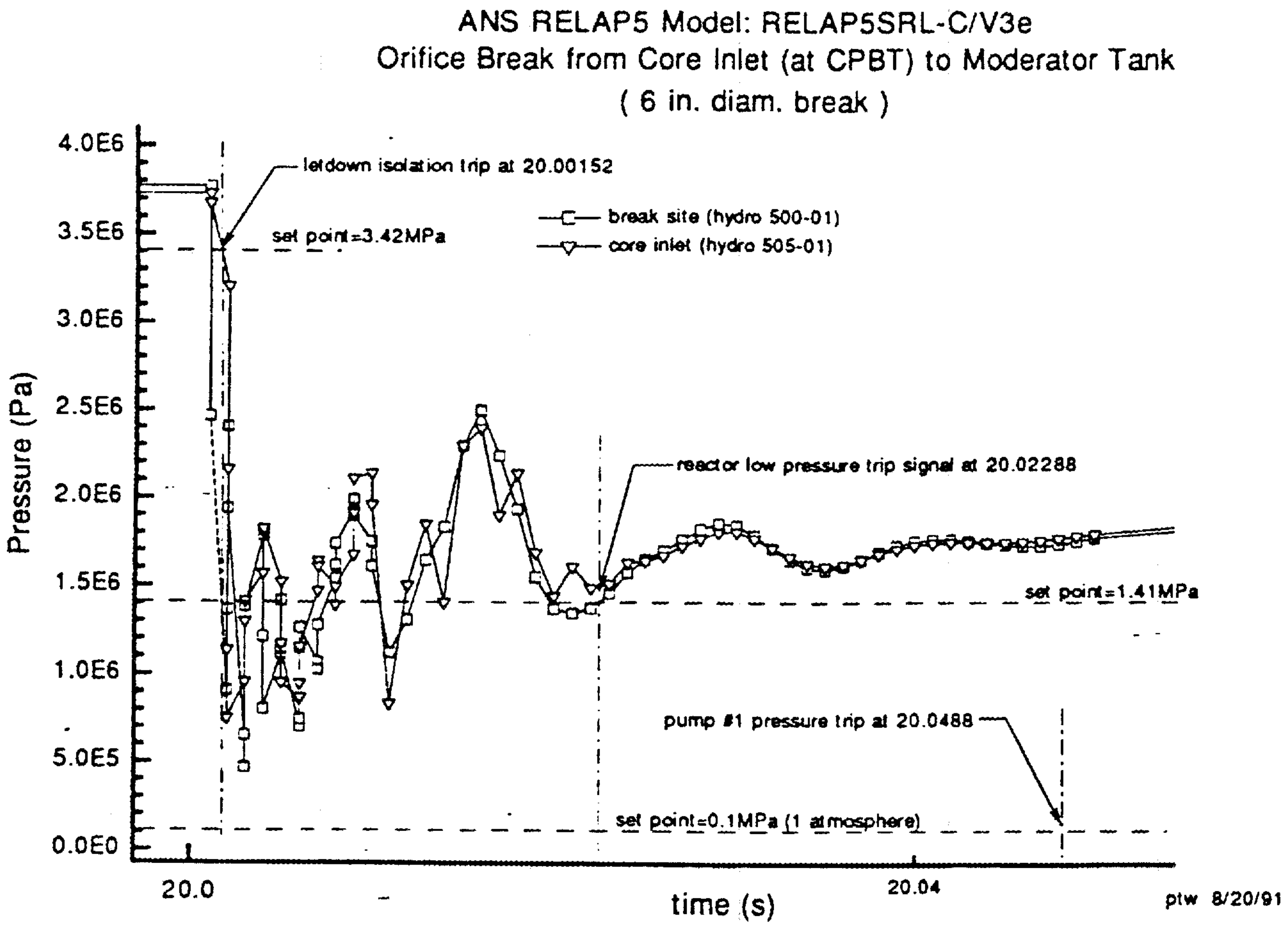
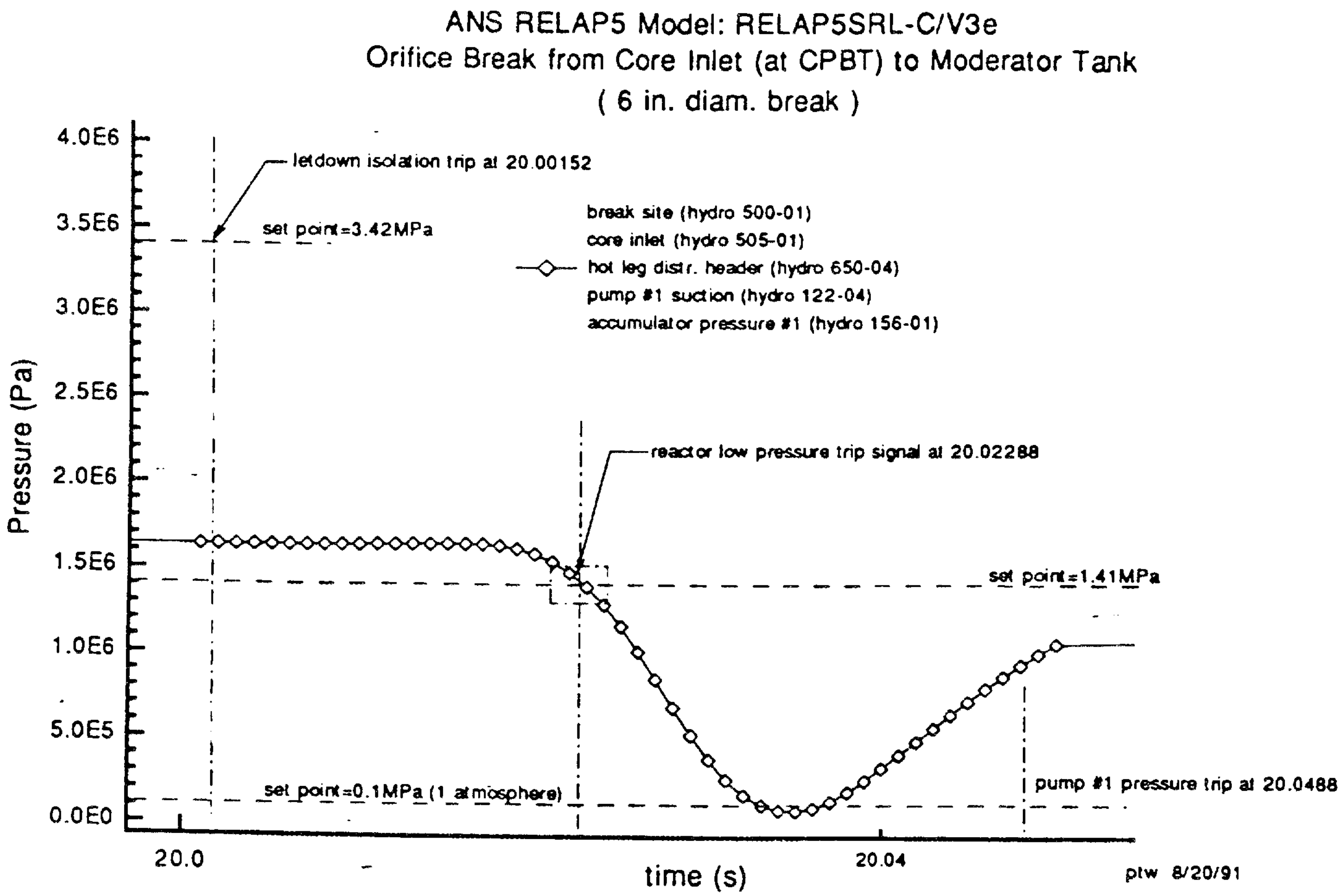
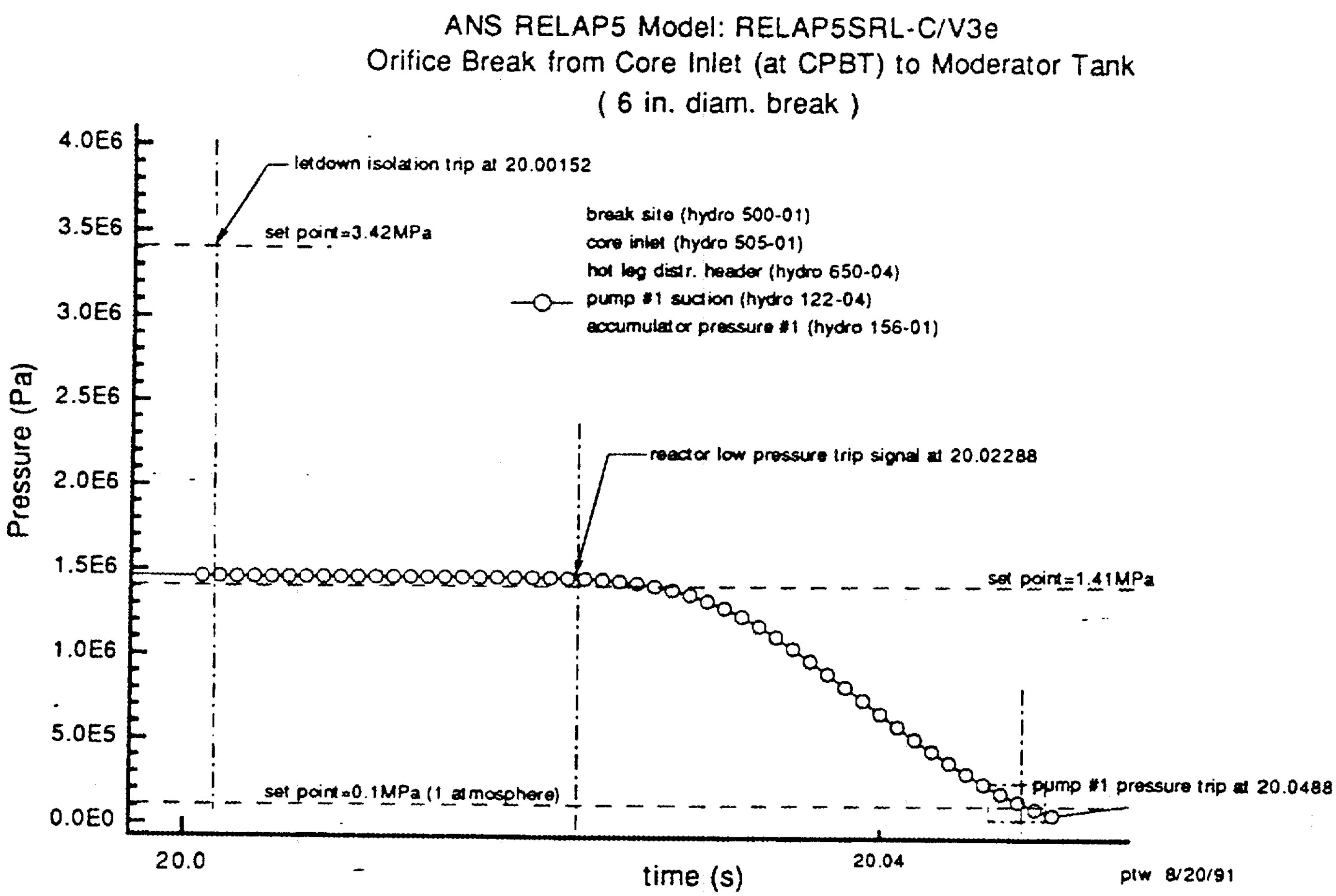
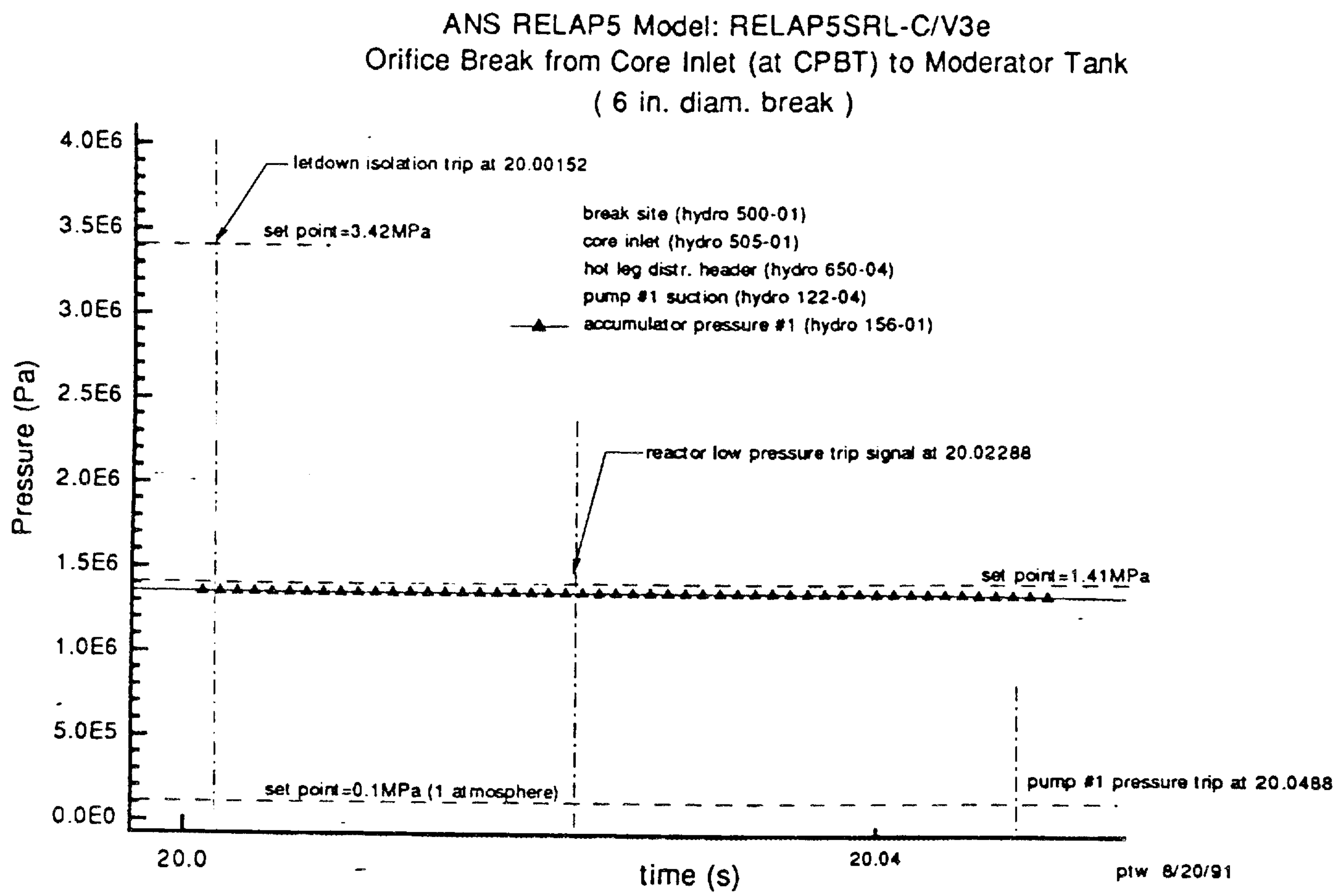

ANS RELAP5 Model: RELAP5SRL-C/V3e
Orifice Break from Core Inlet (at CPBT) to Moderator Tank
(6 in. diam. break)
letdown isolation trip at 20.00152
set point=3.42MPa
break site (hydro 500-01)
core inlet (hydro 505-01)
reactor low pressure trip signal at 20.02288
set point=1.41MPa
pump #1 pressure trip at 20.0488
set point=0.1MPa (1 atmosphere)
Pressure (Pa)
4.0E6
3.5E6
3.0E6
2.5E6
2.0E6
1.5E6
1.0E6
5.0E5
0.0E0
20.0
20.04
time (s)
ptw 8/20/91



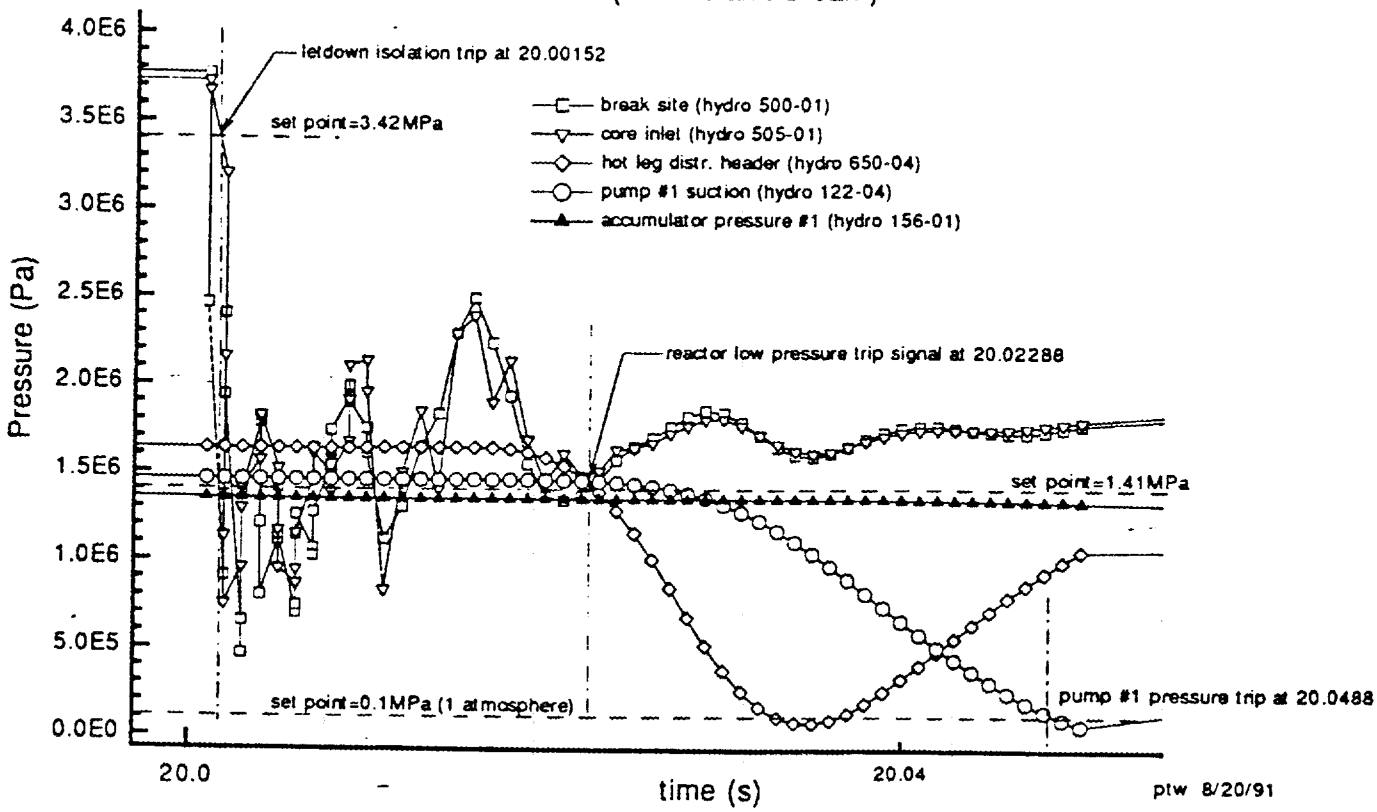
ANS RELAP5 Model: RELAP5SRL-C/V3e
Orifice Break from Core Inlet (at CPBT) to Moderator Tank
(6 in. diam. break)
letdown isolation trip at 20.00152
set point=3.42MPa
break site (hydro 500-01)
core inlet (hydro 505-01)
hot leg distr. header (hydro 650-04)
pump #1 suction (hydro 122-04)
accumulator pressure #1 (hydro 156-01)
reactor low pressure trip signal at 20.02288
set point=1.41MPa
set point=0.1MPa (1 atmosphere)
pump #1 pressure trip at 20.0488
Pressure (Pa)
4.0E6
3.5E6
3.0E6
2.5E6
2.0E6
1.5E6
1.0E6
5.0E5
0.0E0
20.0
20.04
time (s)
ptw 8/20/91


ANS RELAP5 Model: RELAP5SRL-C/V3e
Orifice Break from Core Inlet (at CPBT) to Moderator Tank
(6 in. diam. break)
letdown isolation trip at 20.00152
set point=3.42MPa
break site (hydro 500-01)
core inlet (hydro 505-01)
hot leg distr. header (hydro 650-04)
pump #1 suction (hydro 122-04)
accumulator pressure #1 (hydro 156-01)
reactor low pressure trip signal at 20.02288
set point=1.41MPa
set point=0.1MPa (1 atmosphere)
pump #1 pressure trip at 20.0488
Pressure (Pa)
4.0E6
3.5E6
3.0E6
2.5E6
2.0E6
1.5E6
1.0E6
5.0E5
0.0E0
20.0
20.04
time (s)
ptw 8/20/91



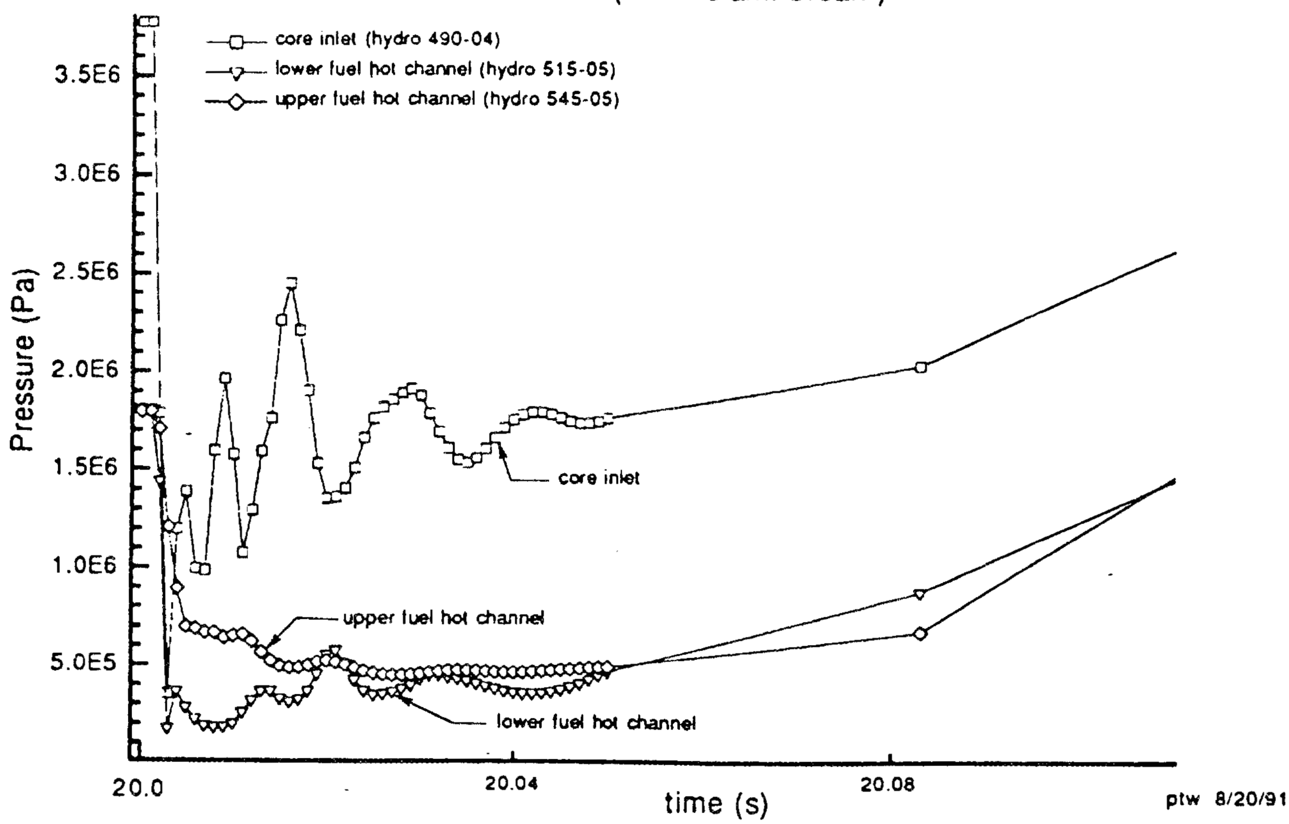
ANS RELAP5 Model: RELAP5SRL-C/V3e
Orifice Break from Core Inlet (at CPBT) to Moderator Tank
(6 in. diam. break)
4.0E6
3.5E6
3.0E6
2.5E6
2.0E6
1.5E6
1.0E6
5.0E5
0.0E0
Pressure (Pa)
letdown isolation trip at 20.00152
set point=3.42MPa
break site (hydro 500-01)
core inlet (hydro 505-01)
hot leg distr. header (hydro 650-04)
pump #1 suction (hydro 122-04)
accumulator pressure #1 (hydro 156-01)
reactor low pressure trip signal at 20.02288
set point=1.41MPa
set point=0.1MPa (1 atmosphere)
pump #1 pressure trip at 20.0488
20.0
20.04
time (s)
ptw 8/20/91

ANS RELAP5 Model: RELAP5SRL-C/V3e
Orifice Break from Core Inlet (at CPBT) to Moderator Tank
(6 in. diam. break)

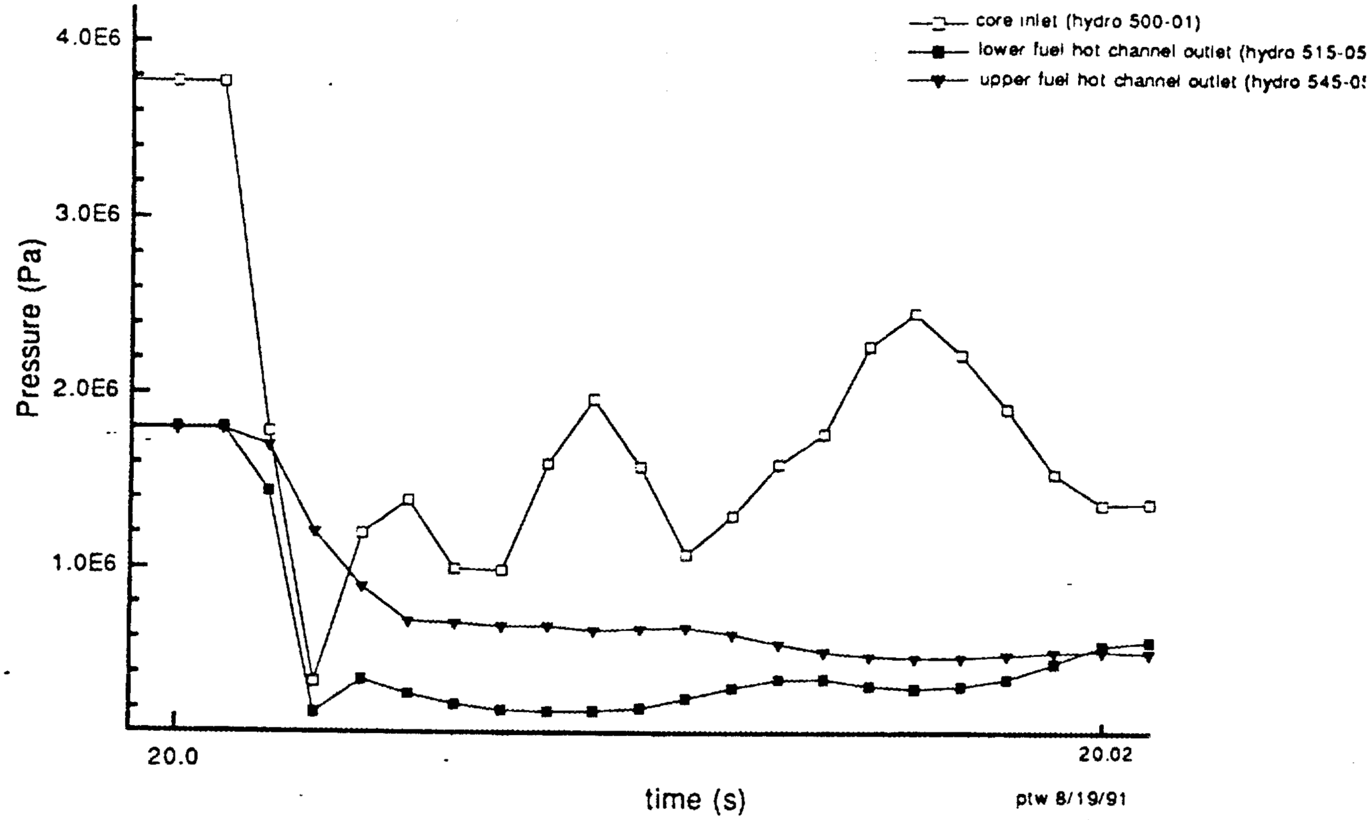


ptw 8/20/91

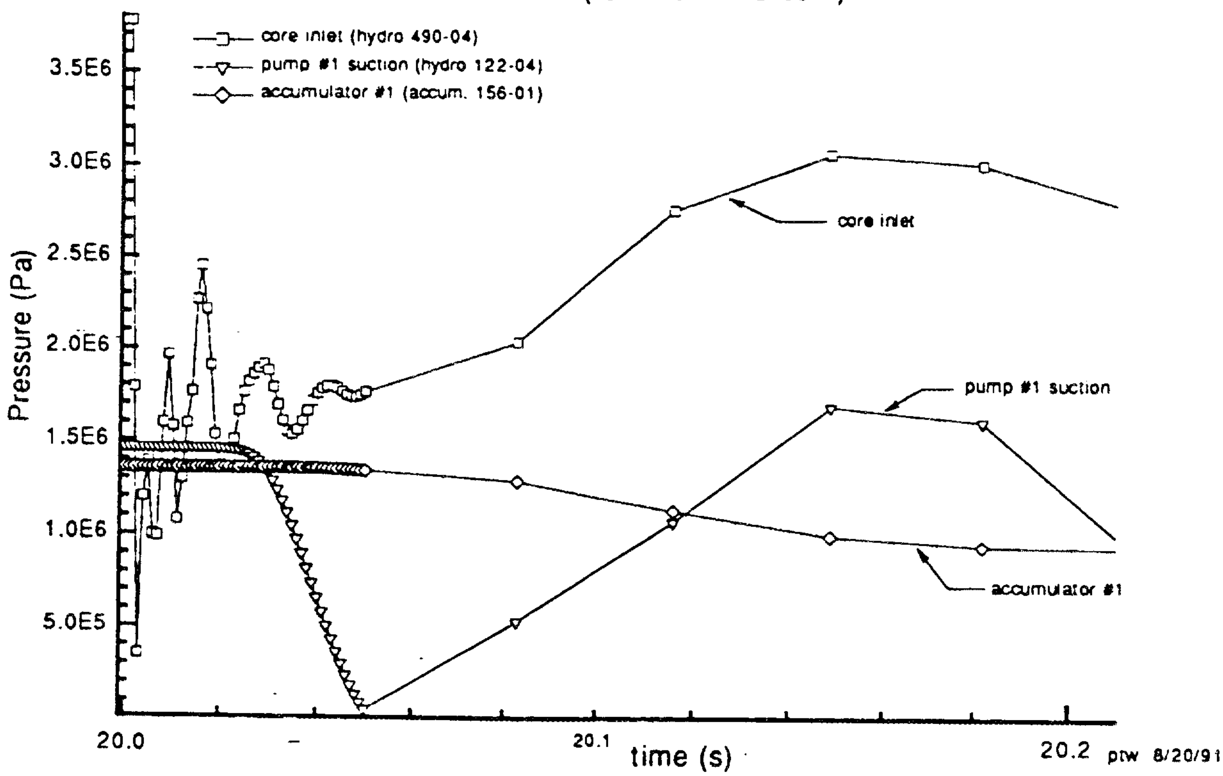
ANS RELAP5 Model: RELAP5SRL-C/V3e
Orifice Break from Core Inlet (at CPBT) to Moderator Tank
(6 in. diam. break)

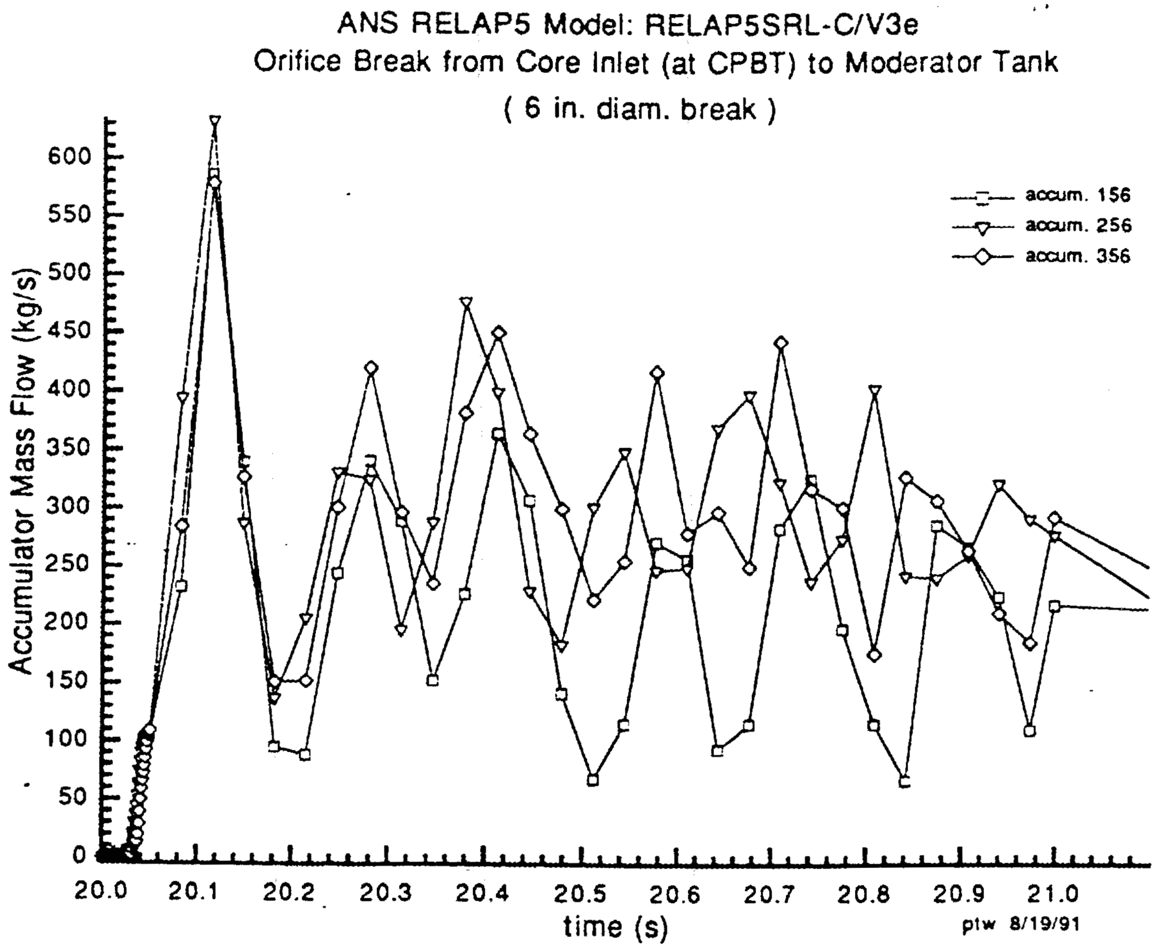


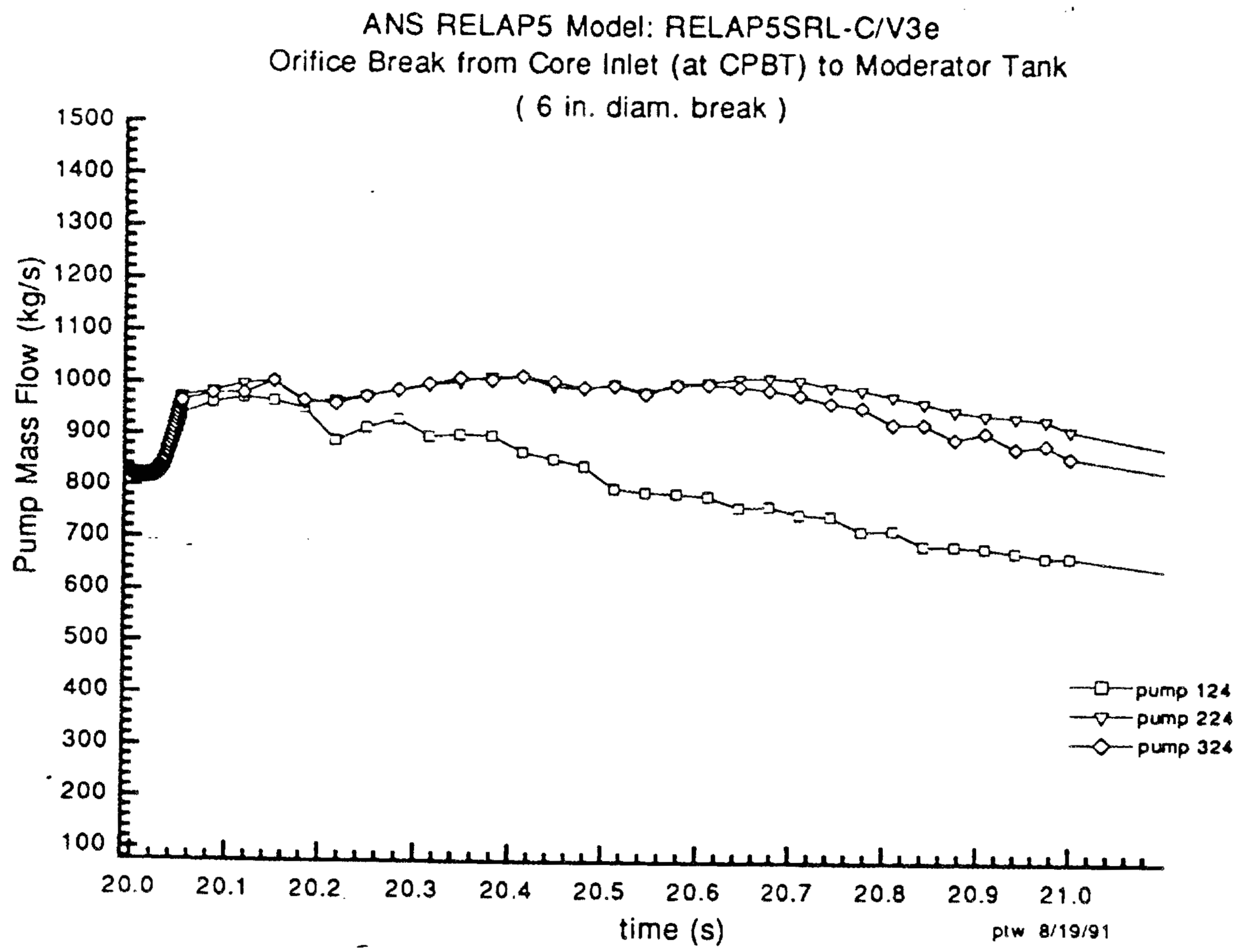
ANS RELAP5 Model: RELAP5SRL-C/V3e
Orifice Break from Core Inlet (at CPBT) to Moderator Tank
(6 in. diam. break)



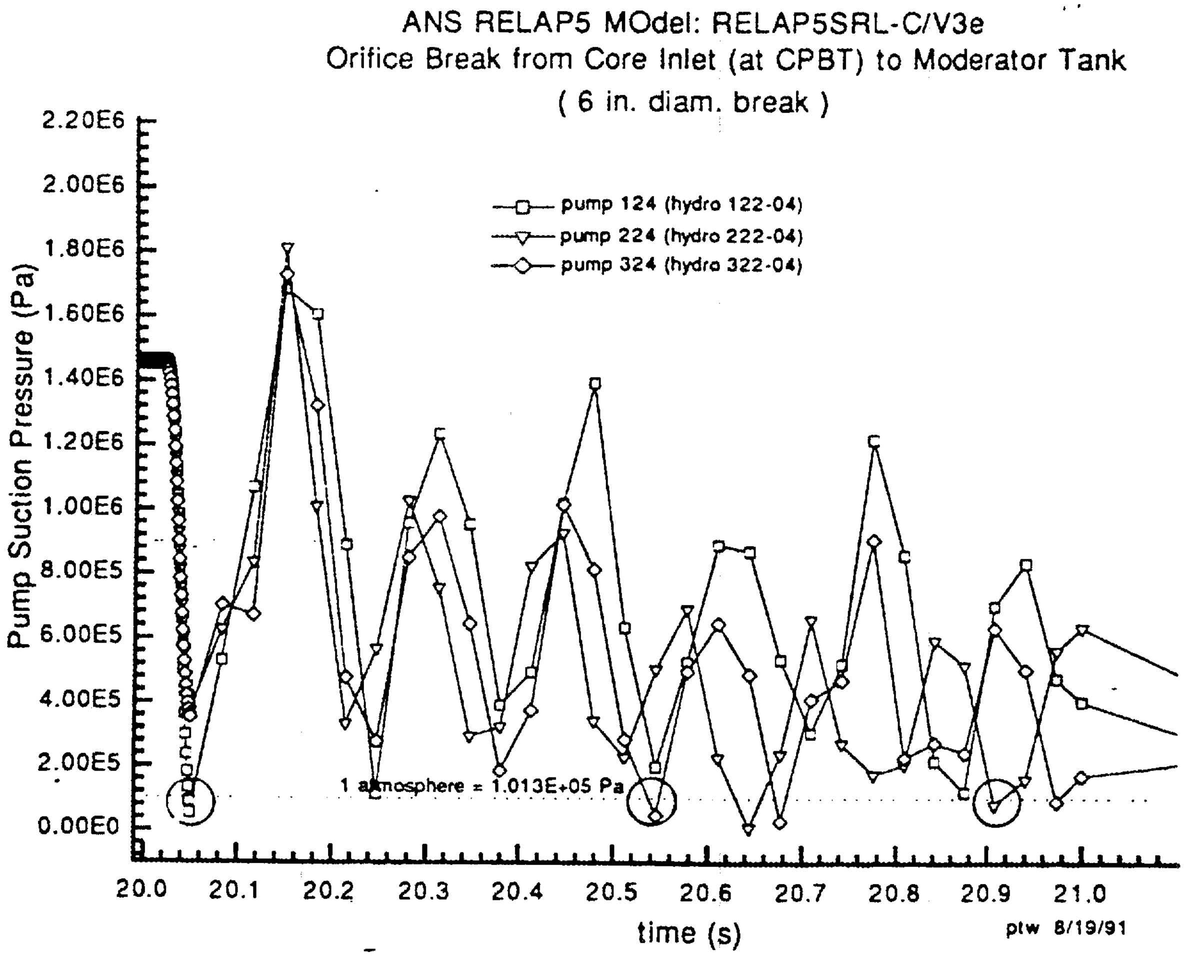
ANS RELAP5 Model: RELAP5SRL-C/V3e
Orifice Break from Core Inlet (at CPBT) to Moderator Tank
(6 in. diam. break)

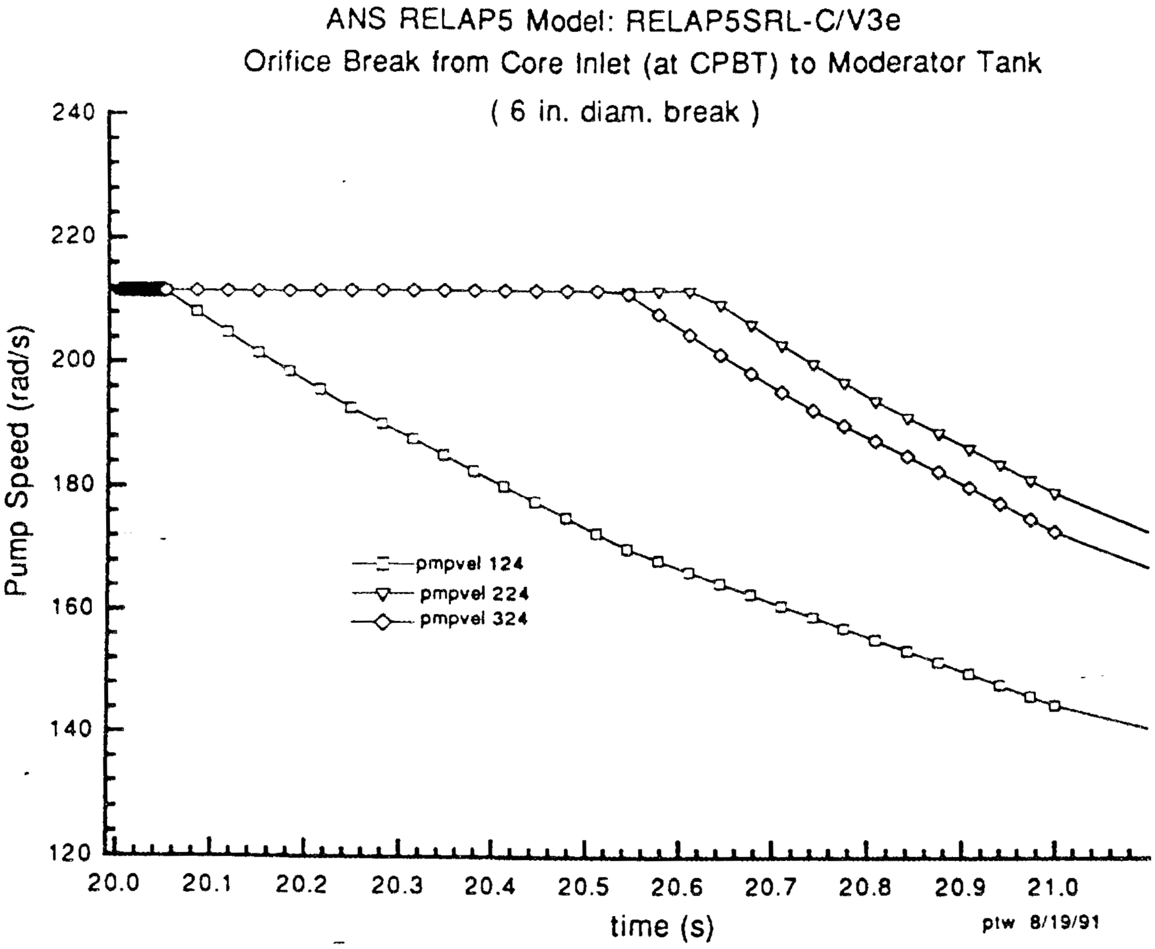



ANS RELAP5 Model: RELAP5SRL-C/V3e
Orifice Break from Core Inlet (at CPBT) to Moderator Tank
(6 in. diam. break)
accum. 156
accum. 256
accum. 356
Accumulator Mass Flow (kg/s)
0
50
100
150
200
250
300
350
400
450
500
550
600
20.0
20.1
20.2
20.3
20.4
20.5
20.6
20.7
20.8
20.9
21.0
time (s)
ptw 8/19/91

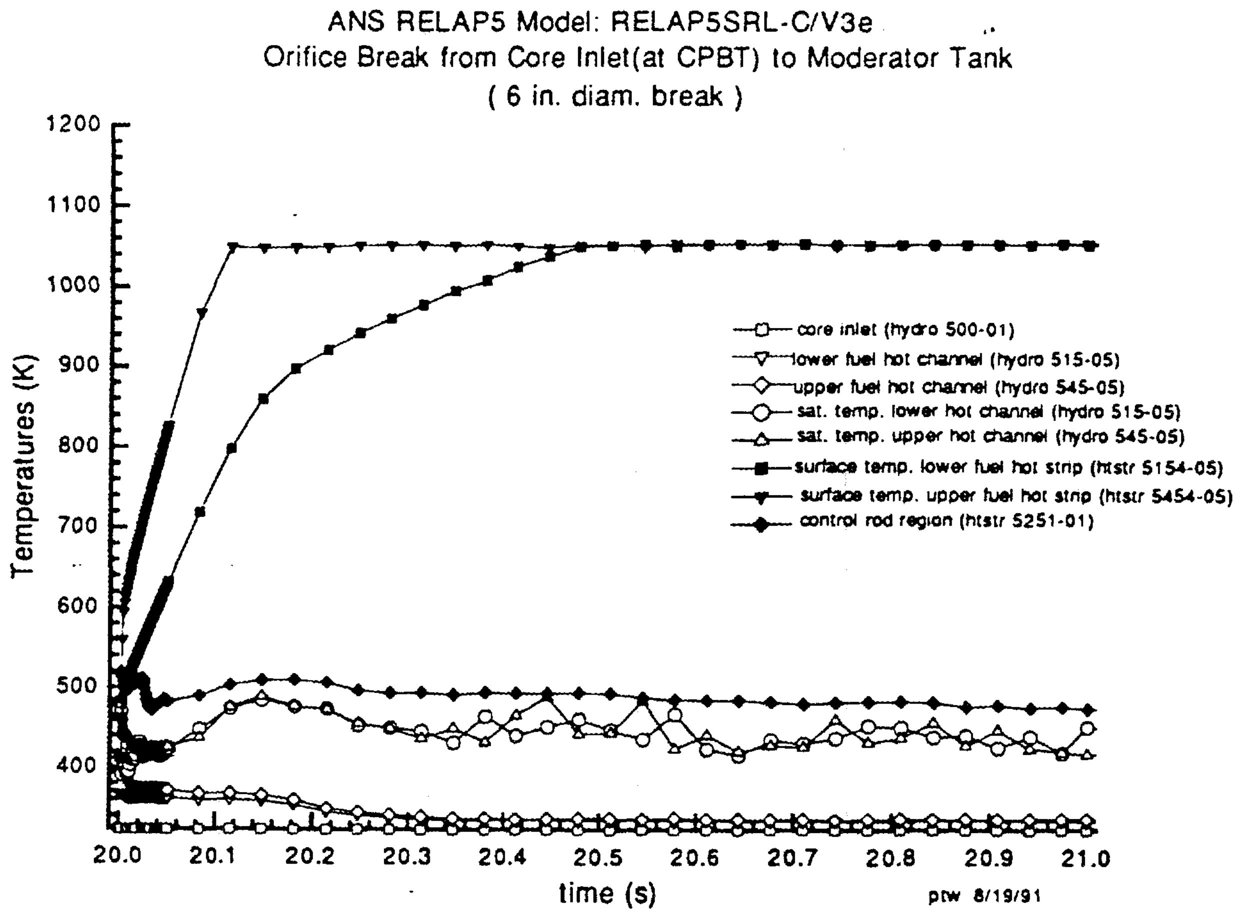


ANS RELAP5 Model: RELAP5SRL-C/V3e
Orifice Break from Core Inlet (at CPBT) to Moderator Tank
(6 in. diam. break)
Pump Mass Flow (kg/s)
1500
1400
1300
1200
1100
1000
900
800
700
600
500
400
300
200
100
20.0
20.1
20.2
20.3
20.4
20.5
20.6
20.7
20.8
20.9
21.0
time (s)
pump 124
pump 224
pump 324
ptw 8/19/91


ANS RELAP5 MOdel: RELAP5SRL-C/V3e
Orifice Break from Core Inlet (at CPBT) to Moderator Tank
(6 in. diam. break)
pump 124 (hydro 122-04)
pump 224 (hydro 222-04)
pump 324 (hydro 322-04)
1 atmosphere = 1.013E+05 Pa
Pump Suction Pressure (Pa)
2.20E6
2.00E6
1.80E6
1.60E6
1.40E6
1.20E6
1.00E6
8.00E5
6.00E5
4.00E5
2.00E5
0.00E0
20.0
20.1
20.2
20.3
20.4
20.5
20.6
20.7
20.8
20.9
21.0
time (s)
ptw 8/19/91

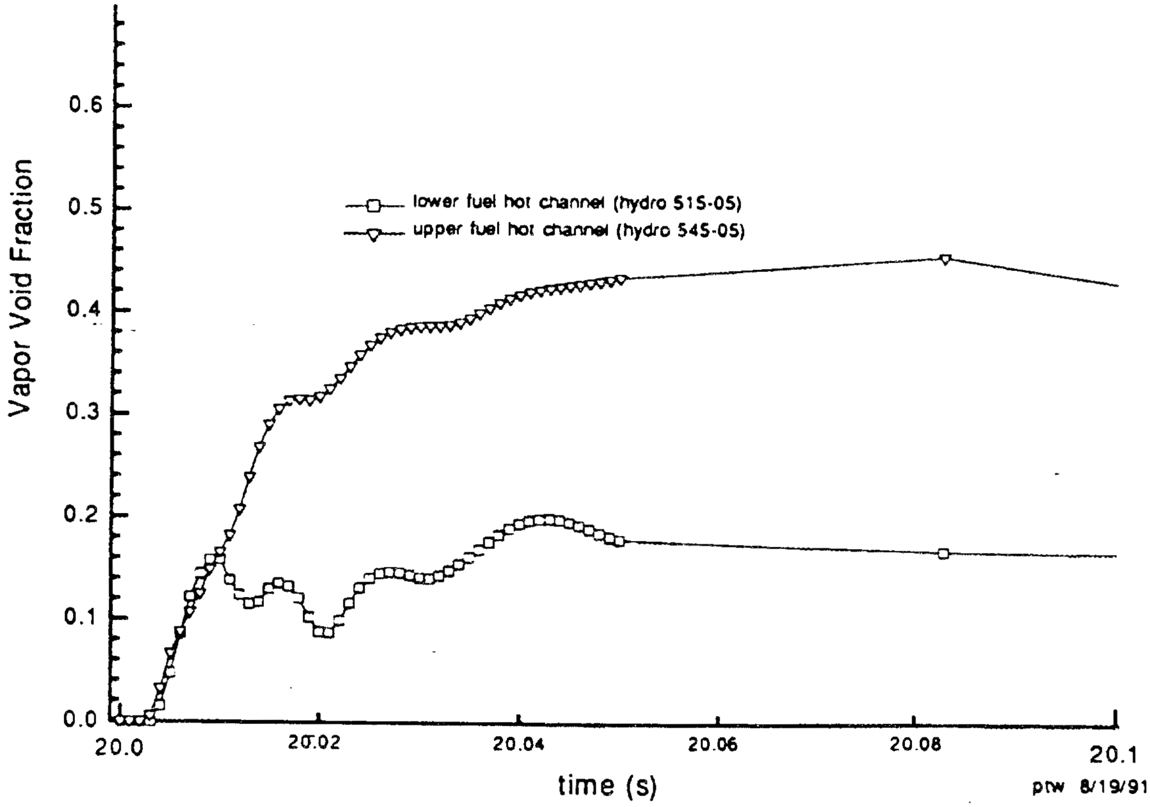


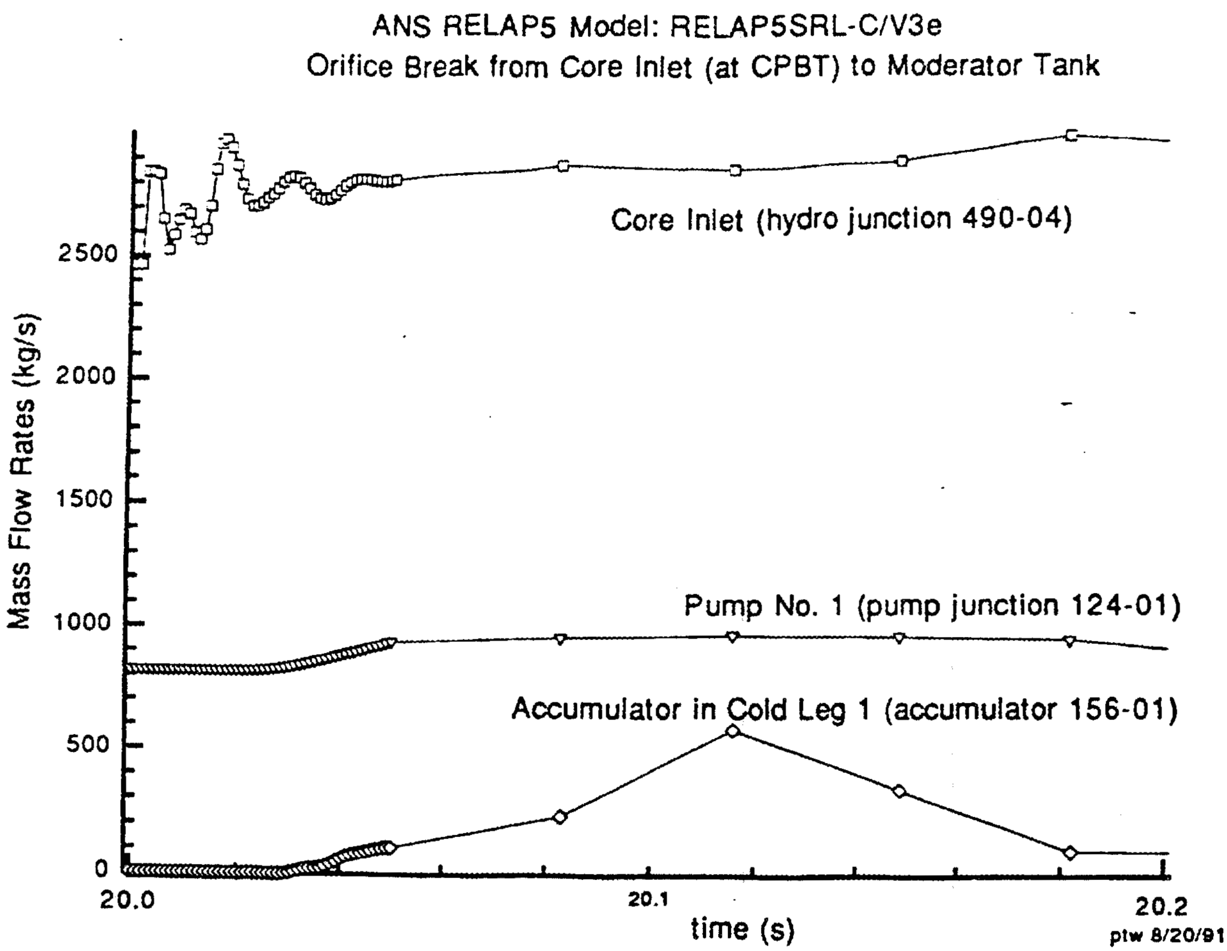
ANS RELAP5 Model: RELAP5SRL-C/V3e
Orifice Break from Core Inlet (at CPBT) to Moderator Tank
(6 in. diam. break)
Pump Speed (rad/s)
240
220
200
180
160
140
120
20.0
20.1
20.2
20.3
20.4
20.5
20.6
20.7
20.8
20.9
21.0
time (s)
pmpvel 124
pmpvel 224
pmpvel 324
ptw 8/19/91



ANS RELAP5 Model: RELAP5SRL-C/V3e
Orifice Break from Core Inlet(at CPBT) to Moderator Tank
(6 in. diam. break)
Temperatures (K)
1200
1100
1000
900
800
700
600
500
400
20.0
20.1
20.2
20.3
20.4
20.5
20.6
20.7
20.8
20.9
21.0
time (s)
core inlet (hydro 500-01)
lower fuel hot channel (hydro 515-05)
upper fuel hot channel (hydro 545-05)
sat. temp. lower hot channel (hydro 515-05)
sat. temp. upper hot channel (hydro 545-05)
surface temp. lower fuel hot strip (htstr 5154-05)
surface temp. upper fuel hot strip (htstr 5454-05)
control rod region (htstr 5251-01)
ptw 8/19/91

ANS RELAP5 Model: RELAP5SRL/C-V3e
Orifice Break from Core Inlet (at CPBT) to Moderator Tank
(6 in. diam. break)




ANS RELAP5 Model: RELAP5SRL-C/V3e
Orifice Break from Core Inlet (at CPBT) to Moderator Tank
Core Inlet (hydro junction 490-04)
Pump No. 1 (pump junction 124-01)
Accumulator in Cold Leg 1 (accumulator 156-01)
Mass Flow Rates (kg/s)
2500
2000
1500
1000
500
0
20.0
20.1
20.2
time (s)
ptw 8/20/91

ANS RELAP5 MODEL
LOCA SIMULATIONS

TASK LIST

- REMESH AT BREAK SITE
- RAMP THE BREAK OPENING OVER APPROXIMATELY 1-2 msec.
- CHECK SENSITIVITY OF LOSS COEFF.

ptw 9/17/91

Appendix F: SUGGESTIONS FROM THE ANSR PIR TEAM DURING PRESENTATION OF LBLOCA ANALYSIS

Appendix F: SUGGESTIONS FROM THE ANSR PIR TEAM DURING PRESENTATION OF LBLOCA ANALYSIS

Technical:

(1) In systems that depressurize very rapidly it makes sense to use a finite break opening time. Models may exist for the opening of diaphragms used in shock tube experiments. Cracks propagate at velocities less than the sound speed in the material.

(2) Pressure/shock wave propagation may not be well modeled in RELAP5. Commercial power system vendors frequently use a code called WHAM (Water Hammer Analysis Model) which may perform better for this part of the Loss of Coolant Accident (LOCA) analysis. Oil pipeline blowdown data may exist that would help here (Contact Alan Bilanian). Note also that sound propagation is less in elastic walled pipes.

(3) Isothermal gas expansion is the conservative assumption with respect to sizing the accumulator to prevent gas entering the primary system.

(4) Much experimental data relevant to the ANSR exists in Bettis and KAPL reports. We should press (from high levels) to gain access to this information. Stan Greene (EPRI) may know what reports we need and who to contact.

(5) Collect a large amount of data at conditions below the thermal limit prior to destroying a channel in a destructive test.

(6) The accumulator liquid flow rate seems to accelerate too quickly during the CPBT break simulation. This result could be checked.

(7) Examine accidents that are initiated at low power operation and look at how protection systems will function. Look at possibility of tripping the reactor on period (i.e., the rate of change in parameters) to allow rapid mitigation of accidents during startup, loss of AC power, or during other situations when the present reactor protection system may not function. Ingress of a light water slug to the fuel assembly (i.e., a positive reactivity insertion) during startup is an example of a situation for which this type of protection may be desirable.

(8) Critical flow velocity in subcooled boiling flows may be treated in Fred Moody's book.

Strategic:

(1) It is best to base simulation of the reactor performance on single phase models where possible. This is especially true for design basis events. Only present analyses with two-phase phenomena when absolutely necessary.

(2) May want to formalize a periodic review of thermal-hydraulic analysis by neutronic analysis experts.

(3) What is the success criterion for the planned severe accident analyses? Implement robust design features to address severe accident conditions (e.g., put the fuel assembly at the bottom of a pool and show it can never be uncovered).

(4) One reason US-NRC has not strongly endorsed the leak before break analysis strategy may be that large pipe breaks in power reactors can be initiated by water hammer events. This happened at

Indian Point. However, it will be difficult to get credit for leak detection capability in the ANSR even if the possibility for a water hammer event is discredited.

(5) May want to establish power limits for the ANSR with explicit opportunities to redefine the operating power in the future. Improvements in fuel manufacturing capabilities and more accurate knowledge of thermal limits in the fuel assembly may then be used to support improvement in the ANSR performance.

(6) Try to be flexible in the approach to experiments and design. Small inexpensive experiments should be performed to aid in the design of larger expensive experiments. Design equipment, budgets and schedules to allow future modifications.

Appendix G: AHP INPUT AND OUTPUT FOR THE EARLY PART OF THE LBLOCA

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@@@@@@@@@@@@@@@@@@@@@@@@@@@@@@@@@@@@@@@@@@@@@@@@@@@@@@@@@@@@@@
@@@@@@@@@@@@@@@@@@@@@@@@@@@@@@@@@@@@@@@@@@@@@@@@@@@@@@@@@@@@@@
@@@@      Analytical Hierarchy Process - output file      @@@@
@@@@                 Program version 5e                   @@@@
@@@@                                                      @@@@
@@@@                      LBLOCA ANSR                     @@@@
@@@@                                                      @@@@
@@@@      date: 10-27-1991           time: 11:17:01 pm    @@@@
@@@@@@@@@@@@@@@@@@@@@@@@@@@@@@@@@@@@@@@@@@@@@@@@@@@@@@@@@@@@@@
@@@@@@@@@@@@@@@@@@@@@@@@@@@@@@@@@@@@@@@@@@@@@@@@@@@@@@@@@@@@@@
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INPUT + OUTPUT WITH CONTROL ROD HT ITEMS

HIERARCHY RELATIONSHIPS AND FACTOR RANKS
(a negative rank indicates a reciprocal; e.g., -3 implies 1/3)

ANSR LBLOCA/COMPONENTS data arrays

ANSR LBLOCA		1.	2.	3.	4.	5.	6.	7.	8.	9.	10.	11	12.	13.	14.
Fuel assy	1.	1	3	5	5	5	5	5	5	5	3	5	5	1	5
control rods	2.		1	3	5	5	5	3	5	5	1	5	5	-3	5
targets	3.			1	2	1	3	1	5	3	-5	3	5	-5	5
CPBT	4.				1	1	3	1	5	3	-5	3	5	-5	5
bypass	5.					1	1	1	5	3	-5	1	3	-5	5
cold leg	6.						1	-3	5	2	-5	1	3	-5	5
pumps	7.							1	5	3	-3	3	5	-5	5
emer HX	8.								1	-5	-5	-3	1	-5	1
prime HX	9.									1	-5	-3	3	-5	5
accumulator	10.										1	5	5	-3	5
hot leg	11.											1	3	-5	5
pressurizer	12.												1	-5	5
break	13.													1	5
secondary	14.														1

COMPONENTS/PHENOMENA data arrays

Fuel assy		1.	2.	3.	4.	5.	6.	7.	8.	9.	10.	11	12.	13.	14.	15.	16.	17.
1 phase HT	1.	1	-3	-3	-5	1	-3	2	5	-5	-3	5	1	1	1	1	-3	3
1 phase FF	2.		1	4	-3	3	-3	3	5	-5	1	-5	1	1	1	1	-3	-3
form factor	3.			1	-3	3	-2	3	5	-5	1	-5	1	-2	3	1	-3	3
power dist	4.				1	5	3	5	5	1	2	1	2	3	5	4	3	3
oxide	5.					1	-3	1	3	-5	-3	-5	-3	1	2	-3	-3	2
coolant gap	6.						1	3	5	-3	3	-3	2	3	3	2	1	3
2D conduction	7.							1	3	-3	-3	-5	-3	-2	1	-3	-2	1
D2O prop	8.								1	-5	-5	-5	-5	-5	-3	-5	-5	-3
Vapor gen	9.									1	2	1	3	5	5	3	3	5
2 phase resist	10.										1	-3	1	3	3	1	1	3
DNB	11.											1	3	5	5	3	3	5
2D flow	12.												1	2	2	-2	1	3
crit. flow	13.													1	-2	-3	-3	2
thermal strain	14.														1	2	1	2
hydraulic loads	15.															1	-3	3
inlet cond.	16.																1	3
2 phase HT	17.																	1

---

control rods		1.	2.	3.	4.
1P HT CR	1.	1	-5	-5	-5
2P HT CR	2.		1	2	1
flow resist CR	3.			1	-2
Heat Load CR	4.				1

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targets		1.	2.	3.	4.	5.
1P HT T	1.	1	-3	-3	-5	1
2P HT T	2.		1	1	-2	1
flow resist T	3.			1	-2	3
Heat load T	4.				1	5
Conduction T	5.					1

CPBT		1.	2.	3.	4.

1P HT CPBT	1.	1	3	-3	-5
2P HT CPBT	2.		1	-3	-5
flow Resis CPBT	3.			1	-3
Heat Load CPBT	4.				1

bypass		1.	
Flow resist BYP	1.	1	

cold leg		1.	2.
flow resis CL	1.	1	3
HT CL	2.		1

pumps		1.	2.
performance P	1.	1	3
Trip P	2.		1

emer HX		1.	2.
flow resist EHX	1.	1	3
HT EHX	2.		1

prime HX		1.	2.
HT PHX	1.	1	-3
flow resist PHX	2.		1

accumulator		1.	2.
process ACC	1.	1	3
evolution ACC	2.		1

hot leg		1.	2.
HT HL	1.	1	-3
flow resist HL	2.		1

pressurizer		1.	
Press. mass flo	1.	1	

break		1.	2.	3.
area vs t BR	1.	1	3	4
m flux vs t BR	2.		1	3
shape vs t BR	3.			1

---

secondary		1.
secondary	1.	1

COMPONENTS FACTORS RELATIVE TO ANSR LBLOCA

Factors relative to ANSR LBLOCA:

	weight
Fuel assy	1.0000
control rods	0.6641
targets	0.3226
CPBT	0.2887
bypass	0.2379
cold leg	0.1894
pumps	0.3103
emer HX	0.0818
prime HX	0.1558
accumulator	0.7047
hot leg	0.1889
pressurizer	0.1056
break	1.0000
secondary	0.0749

lambda (maximum) = 15.8862
consistency index = 0.1451
consistency ratio = 0.0967

Composite priorities:

	weight	priority
break	0.1878	(9)
Fuel assy	0.1878	(9)
accumulator	0.1323	(6)
control rods	0.1247	(6)
targets	0.0606	(3)
pumps	0.0583	(3)
CPBT	0.0542	(3)
bypass	0.0447	(2)
cold leg	0.0356	(2)
hot leg	0.0355	(2)
prime HX	0.0293	(2)
pressurizer	0.0198	(1)
emer HX	0.0154	(1)
secondary	0.0141	(1)

PHENOMENA FACTORS RELATIVE TO COMPONENTS

Factors relative to Fuel assy:

	weight
1 phase HT	0.4455
1 phase FF	0.3634
form factor	0.3455
power dist	0.9017
oxide	0.1839
coolant gap	0.5649
2D conduction	0.1563
D2O prop	0.0816
Vapor gen	1.0000
2 phase resist	0.4297
DNB	0.9296
2D flow	0.3488
crit. flow	0.2321
thermal strain	0.2396
hydraulic loads	0.3401
inlet cond.	0.5065
2 phase HT	0.1859

lambda (maximum) = 19.8287
consistency index = 0.1768
consistency ratio = 0.1179 (See footnote below)

Factors relative to control rods:

	weight
1P HT CR	0.1702
2P HT CR	1.0000
flow resist CR	0.6048
Heat Load CR	1.0000

lambda (maximum) = 4.0606
consistency index = 0.0202
consistency ratio = 0.0225

Factors relative to targets:

	weight
1P HT T	0.1872
2P HT T	0.4467
flow resist T	0.5421
Heat load T	1.0000
Conduction T	0.2451

lambda (maximum) = 5.1389
consistency index = 0.0347
consistency ratio = 0.0310

Factors relative to CPBT:

	weight
1P HT CPBT	0.2352
2P HT CPBT	0.1340
flow Resis CPBT	0.4506
Heat Load CPBT	1.0000

lambda (maximum) = 4.1981
consistency index = 0.0660
consistency ratio = 0.0734

Factors relative to bypass:

	weight
Flow resist BYP	1.0000

lambda (maximum) = 1.0000
consistency index = 0.0000
consistency ratio = 0.0000

Factors relative to cold leg:

	weight
flow resis CL	1.0000
HT CL	0.3333

lambda (maximum) = 2.0000
consistency index = 0.0000
consistency ratio = 0.0000

Factors relative to pumps:

	weight
performance P	1.0000
Trip P	0.3333

lambda (maximum) = 2.0000
consistency index = 0.0000
consistency ratio = 0.0000

Factors relative to emer HX:

	weight
flow resist EHX	1.0000
HT EHX	0.3333

lambda (maximum) = 2.0000
consistency index = 0.0000
consistency ratio = 0.0000

Factors relative to prime HX:

	weight
HT PHX	0.3333
flow resist PHX	1.0000

lambda (maximum) = 2.0000
consistency index = 0.0000
consistency ratio = 0.0000

Factors relative to accumulator:

	weight
process ACC	1.0000
evolution ACC	0.3333

lambda (maximum) = 2.0000
consistency index = 0.0000
consistency ratio = 0.0000

Factors relative to hot leg:

	weight
HT HL	0.3333
flow resist HL	1.0000

lambda (maximum) = 2.0000
consistency index = 0.0000
consistency ratio = 0.0000

Factors relative to pressurizer:

	weight
Press. mass flo	1.0000

lambda (maximum) = 1.0000
consistency index = 0.0000
consistency ratio = 0.0000

Factors relative to break:

	weight
area vs t BR	1.0000
m flux vs t BR	0.4368
shape vs t BR	0.1908

lambda (maximum) = 3.0735
consistency index = 0.0368
consistency ratio = 0.0634

Factors relative to secondary:

	weight
secondary	1.0000

lambda (maximum) = 1.0000
consistency index = 0.0000
consistency ratio = 0.0000

~~[illegible]~~ with CR HT ITEMS

Composite priorities:

	weight	priority
area vs t BR	0.0684	(9)
Vapor gen	0.0684	(9)
DNB	0.0636	(8)
power dist	0.0617	(8)
process ACC	0.0482	(7)
2P HT CR	0.0454	(6)
Heat Load CR	0.0454	(6)
coolant gap	0.0386	(5)
inlet cond.	0.0346	(5)
1 phase HT	0.0305	(4)
m flux vs t BR	0.0299	(4)
2 phase resist	0.0294	(4)
flow resist CR	0.0275	(4)
1 phase FF	0.0249	(4)
2D flow	0.0238	(4)
form factor	0.0236	(4)
hydraulic loads	0.0233	(4)
Heat load T	0.0221	(3)
performance P	0.0212	(3)
Heat Load CPBT	0.0197	(3)
thermal strain	0.0164	(3)
Flow resist BYP	0.0163	(3)
evolution ACC	0.0161	(3)
crit. flow	0.0159	(3)
shape vs t BR	0.0130	(2)
flow resis CL	0.0130	(2)
flow resist HL	0.0129	(2)
2 phase HT	0.0127	(2)
oxide	0.0126	(2)
flow resist T	0.0120	(2)
2D conduction	0.0107	(2)
flow resist PHX	0.0106	(2)
2P HT T	0.0099	(2)
flow Resis CPBT	0.0089	(2)
1P HT CR	0.0077	(2)
Press. mass flo	0.0072	(2)
Trip P	0.0071	(2)
flow resist EHX	0.0056	(1)
D2O prop	0.0056	(1)
Conduction T	0.0054	(1)
secondary	0.0051	(1)
1P HT CPBT	0.0046	(1)
HT CL	0.0043	(1)
HT HL	0.0043	(1)
1P HT T	0.0041	(1)
HT PHX	0.0035	(1)
2P HT CPBT	0.0026	(1)
HT EHX	0.0019	(1)

Factors relative to Fuel assy:

	weight	priority
1 phase HT	0.0305	(4)
1 phase FF	0.0249	(4)
form factor	0.0236	(4)
power dist	0.0617	(8)
oxide	0.0126	(2)
coolant gap	0.0386	(5)
2D conduction	0.0107	(2)
D2O prop	0.0056	(1)
Vapor gen	0.0684	(9)
2 phase resist	0.0294	(4)
DNB	0.0636	(8)
2D flow	0.0238	(4)
crit. flow	0.0159	(3)
thermal strain	0.0164	(3)
hydraulic loads	0.0233	(4)
inlet cond.	0.0346	(5)
2 phase HT	0.0127	(2)

Factors relative to control rods:

	weight	priority
1P HT CR	0.0077	(2)
2P HT CR	0.0454	(6)
flow resist CR	0.0275	(4)
Heat Load CR	0.0454	(6)

Factors relative to targets:

	weight	priority
1P HT T	0.0041	(1)
2P HT T	0.0099	(2)
flow resist T	0.0120	(2)
Heat load T	0.0221	(3)
Conduction T	0.0054	(1)

Factors relative to CPBT:

	weight	priority
1P HT CPBT	0.0046	(1)
2P HT CPBT	0.0026	(1)
flow Resis CPBT	0.0089	(2)
Heat Load CPBT	0.0197	(3)

Factors relative to bypass:

	weight	priority
Flow resist BYP	0.0163	(3)

Factors relative to cold leg:

	weight	priority
flow resis CL	0.0130	(2)
HT CL	0.0043	(1)

Factors relative to pumps:

	weight	priority
performance P	0.0212	(3)
Trip P	0.0071	(2)

Factors relative to emer HX:

	weight	priority
flow resist EHX	0.0056	(1)
HT EHX	0.0019	(1)

Factors relative to prime HX:

	weight	priority
HT PHX	0.0035	(1)
flow resist PHX	0.0106	(2)

Factors relative to accumulator:

	weight	priority
process ACC	0.0482	(7)
evolution ACC	0.0161	(3)

Factors relative to hot leg:

	weight	priority
HT HL	0.0043	(1)
flow resist HL	0.0129	(2)

Factors relative to pressurizer:

	weight	priority
Press. mass flo	0.0072	(2)

Factors relative to break:

	weight	priority
area vs t BR	0.0684	(9)
m flux vs t BR	0.0299	(4)
shape vs t BR	0.0130	(2)

Factors relative to secondary:

	weight	priority
secondary	0.0051	(1)

CONSISTENCY OF THE HIERARCHY = 0.0912

Footnote: The consistency limit has exceeded 10%.
A review of the input assumptions may be necessary.

***** Above results produced using the Dimenna normalization

Appendix H: COMMENTS DURING THE LBLOCA PIR DEVELOPMENT

Appendix H: COMMENTS DURING THE LBLOCA PIR DEVELOPMENT

Comments from PIR Team members during the consideration of pairwise rankings in the development of the PIR for the LBLOCA follow:

Fuel Assembly
- versus control rods: Control rods may begin to move for breaks far from the fuel.
- versus accumulator: accumulator important for some break locations
- versus bypass: Fuel assembly carries +90% of the primary flow.
- versus break: Break timing has a very large impact on power versus pressure performance of transient.

Control Rods versus CPBT: CPBT passive during early LBLOCA depressurization. Break is modeled separately.

Targets versus pressurization system: Pressurization system slow to react to LBLOCA.

Phenomena Rankings under the Fuel Assembly:

Single phase heat transfer
- vs. single phase friction factor: Friction factor influences mass flux which influences bulk temperature.
- vs. form factor: Velocity head is 0.4 MPa, while friction factor is causing 1.9 MPa.
- vs. oxide growth: Oxide thickness influences fuel centerline temperature.
- vs. plate spacing: Plate spacing influences mass flux which influences the bulk temperature.
- vs. two phase flow resistance: Assume the momentum flux change due to vapor generation is picked up in the vapor generation term.
- vs. 2D 3D flow: flow may redistribute, models and data may be available used in COBRA due to Rowe.
- vs. critical flow: Sound speed changes rapidly with vapor generation. May want to use a model developed by Henry and Fauske as a check.
- vs. thermal strain: fuel is designed with thermal strain in mind. Strain may influence the channel gap.
- vs. two phase heat transfer: two phase heat transfer coefficient is so large it does not need to be right.

Single Phase Friction Factor
- vs. D20 properties: The ranking here considers using D20 properties instead of the already widely used H20 properties .

Phenomena Ranking under Accumulator: Assumed we used He due to low disolved volume in water.

A few additional comments of value:
(1) The Canadians have pressure tubes similar to our CPBT and may have some good ideas how to design and analyze them.
(2) Some hot spot and hot channel data may be available from FRIGG data (Rex Shumway, Rouhani, Standervold).

Appendix I: SIMULATION RESULTS FOR DELAY HEAT REMOVAL

MARTIN MARIETTA

Internal Correspondence

MARTIN MARIETTA ENERGY SYSTEMS, INC.

October 17, 1991

Art E. Ruggles MS 8045 (576-3977)

Dear Art,

Enclosed are the results of a 0.15 m (6") break at the CPBT outlet. I have plotted a few of the variables and supply them also in a floppy, just in case somebody cares for the actual numbers (it was too long to simply print it out). The case that I run is as follows:

(1) A break is opened slowly (it takes 250 ms) in the hot leg just outside the CPBT. The break pressure in set to the light water pool pressure. It is not yet clear what is the depth of the pool at that point. I have assumed a pressure of 0.29 MPa (i.e., about 19 m of H_2O)

(2) The reactor is operating with I3 grading, at 350 MW-fis, 3.7 MPa inlet pressure, and 27.6 m/s coolant velocity. This transient assumes multiplicative uncertainties and end of cycle conditions. These are the conditions of Norbert Chen's RELAP model that we have used for the RELAP-Dynamic-model benchmark. The pressure drops are fudged in the dynamic model to agree with those of RELAP (See September monthly report)

(3) Upon detection of low pressure (80% of nominal) at the detector location (30 ms delay), the reactor is scrammed, and the main circulation pumps are tripped. A complete failure of the pony motors is assumed from time zero (i.e., in this transient, pony motors do not exist)

(4) The nominal amount of makeup flow (10 kg/s) continues during all the transient. This is unrealistic, because we would run out of D_2O for injection fairly soon, and we would not want to keep it going anyway. Nevertheless, this flow does not really affect the result much.

(5) The reactor and plant conditions are summarized some how in file ANS.DOC in fairly self explanatory fashion.

(6) The *.DAT files in the enclosed floppy have the transient data for some of the most interesting variables. All other variables are available, but I can not just print you all of them. If you need any more, let me know.

The most interesting result of this analysis is that for the I3 grading, multiplicative uncertainties, and 250 ms break opening time, pony motors are not required for any length of time. The upper core hot channel outlet temperature gets very close to depressurized saturation (see Fig. IC/ANS/F/92-10), but it does not get there. The incipient boiling limit is not reached for either the lower or upper cores. Note that the ponies would be required if there was a loss of circuit integrity.

José March-Leuba

José March-Leuba

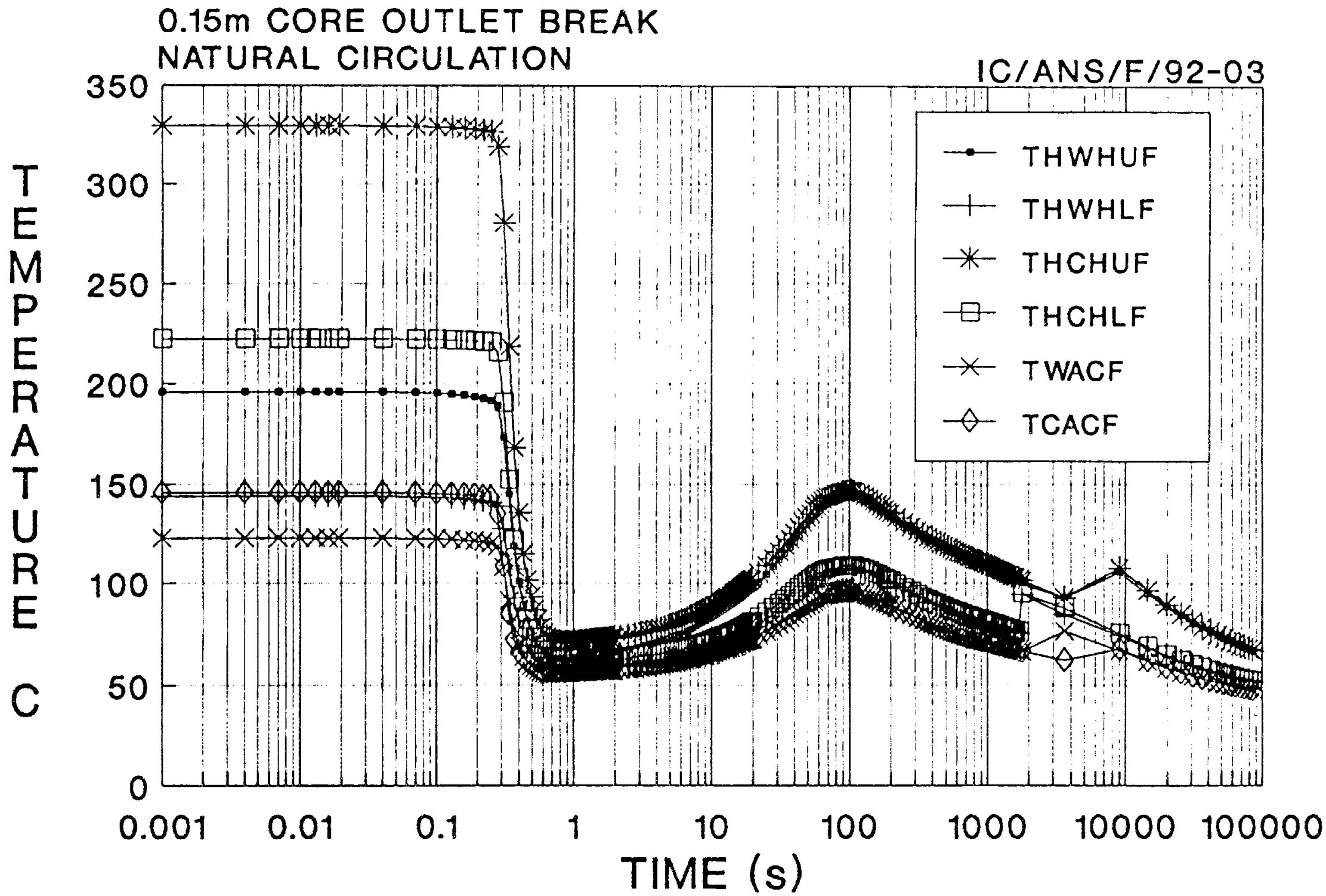
cc: R. M. Harrington
D. L. Selby
C. D. West

Advanced Neutron Source

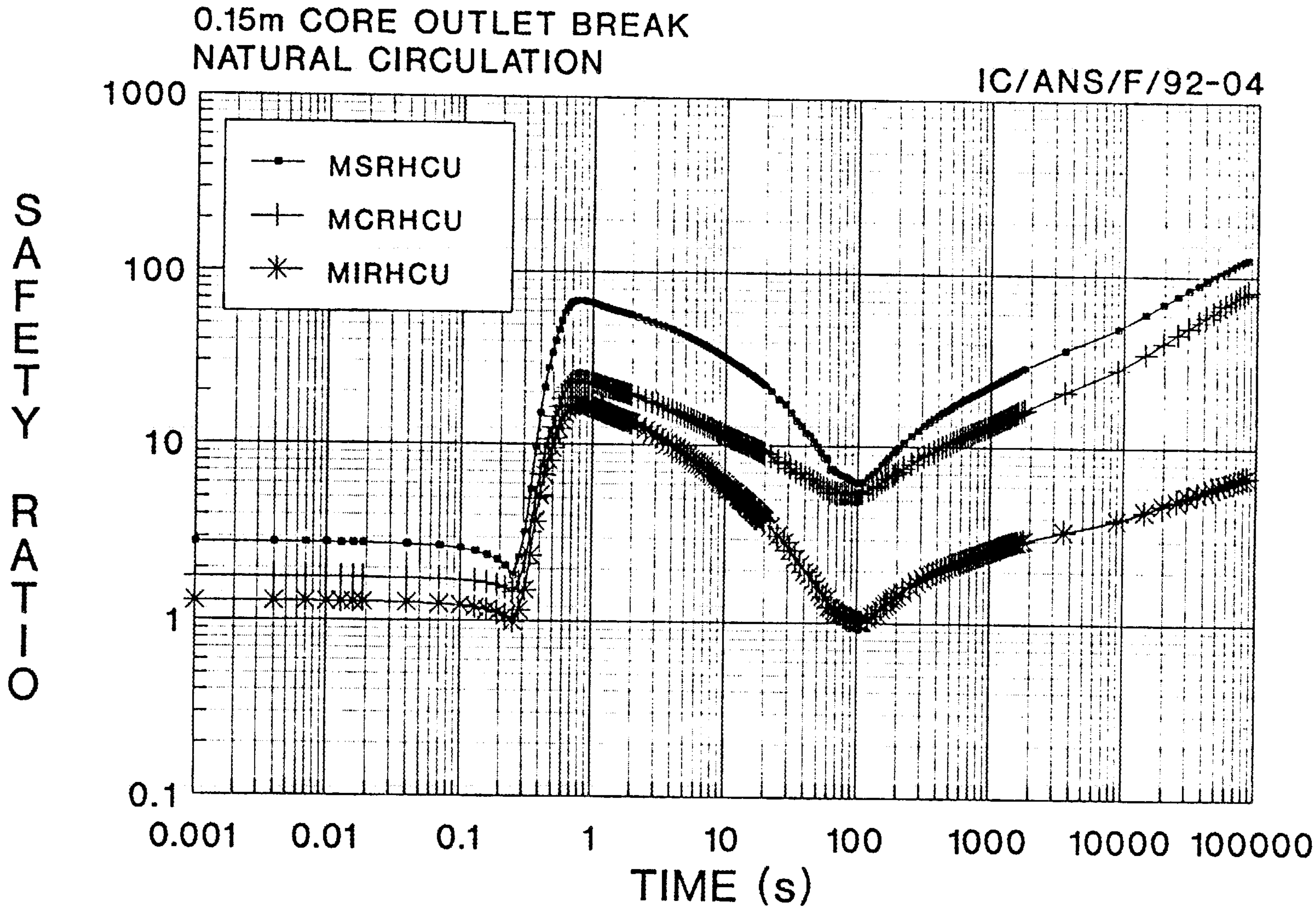
This is extremely important information to understand the enclosed data tables and figures. I did not have time to translate the names of the dynamic code internal variables to nice, easy to understand names. Thus, you must use the following conversion table:

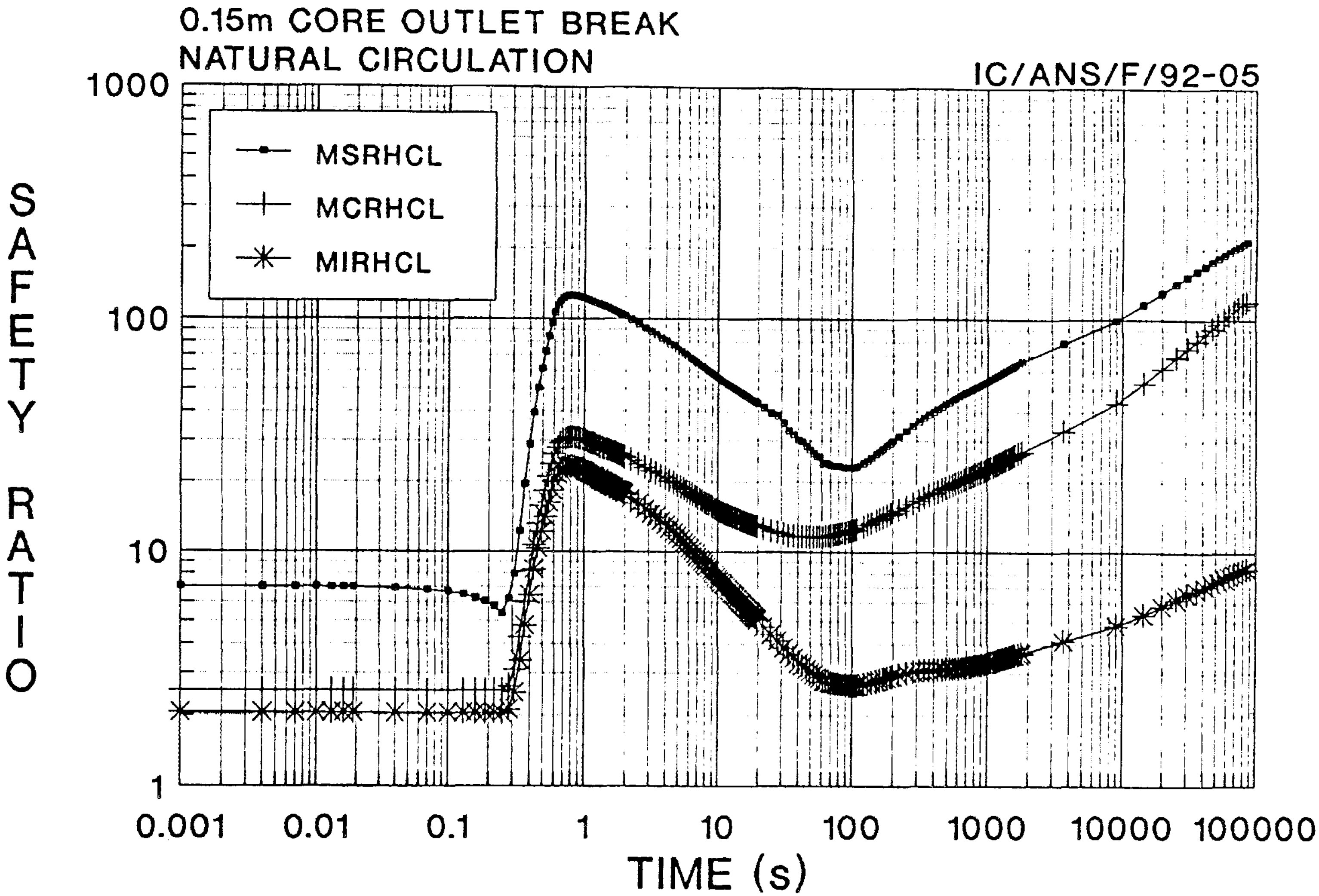
THWHUF	Surface temperature at fuel-D2O interface at the hot spot of the upper core
THWHLF	Surface temperature at fuel-D2O interface at the hot spot of the lower core
THCHUF	Fuel centerline temperature at the hot spot of the upper core. Assumes 2.E-6 m oxide.
THCHLF	Fuel centerline temperature at the hot spot of the lower core. Assumes 2.E-6 m oxide.
TWACF	Surface temperature at fuel-D2O interface at the exit of the average channel of the upper core
TCACF	Fuel centerline temperature at the exit of the average channel of the upper core. Assumes 2.E-6 m oxide.
MSRHCU	Ratio of flow stability critical heat flux (Costa) to actual heat flux at worst point in upper core
MCRHCU	Ratio of critical heat flux (Gambill/Weatherhead) to actual heat flux at worst point in upper core
MIRHCU	Ratio of incipient boiling heat flux (Bergles/Rosenhow) to actual heat flux at worst point in upper core
MSRHCL	Ratio of flow stability critical heat flux (Costa) to actual heat flux at worst point in lower core
MCRHCL	Ratio of critical heat flux (Gambill/Weatherhead) to actual heat flux at worst point in lower core
MIRHCL	Ratio of incipient boiling heat flux (Bergles/Rosenhow) to actual heat flux at worst point in lower core
XICR	Position (m) of inner control rod (0 - fully withdrawn, 1 - fully inserted)
RICR	Reactivity worth of inner control rod (set to zero at initial position)
XOCR	Position (m) of outer control rod (0 - fully withdrawn, 1 - fully inserted)
ROCR	Reactivity worth of outer control rod (set to zero at initial position)
FCNFX	Neutron flux

FDHFX	Decay heat as percent of nominal power
JCPG	Reactor fision power
WVESI	Mass flow rate at CPBT inlet
WHLRI	Mass flow rate at hot leg riser inlet (after the break)
WCOLC	Break (leak) flow from Hot leg inlet (CPBT outlet) to pool
WGAC	Mass flow rate out of all three gas accumulators
WMUSO	Makeup mass flow rate
WLDSI	Letdown mass flow rate
PVESI	Pressure at CPBT inlet. This pressure is actually calculated inside the nozzle of the inertial flow diode, where the flow velocity is close to 25 m/s. Stagnation pressure is almost 0.4 MPa higher
PACHO	Pressure at core outlet, just inside the coolant channel, where the flow velocity is 27.6 m/s. Sagnation pressure is about 0.4 MPa higher
PVESO	Pressure at CPBT outlet (the break location)
PDET	Pressure at the outlet of hot leg riser. This pressure includes a 30 ms fist orcer lag to simulate the pressure sensor. This pressure defines the scram action and control
PLHLO	Pressure at gas accumulator inlet
PMCPI	Pressure at pump suction
TVESI	Temperature at CPBT inlet
TACHO	Average core outlet temperature. This temperature is the mixed bulk temperature of both upper and lower cores
THCUO	Outlet temperature of hot channel in upper core
THCLO	Outlet temperature of hot channel in lower core
TVESO	Temperature at the hot leg riser inlet (CPBT outlet). This temperature is the mixed bulk temeprature once all the flows inside the CPBT are mixed toghether

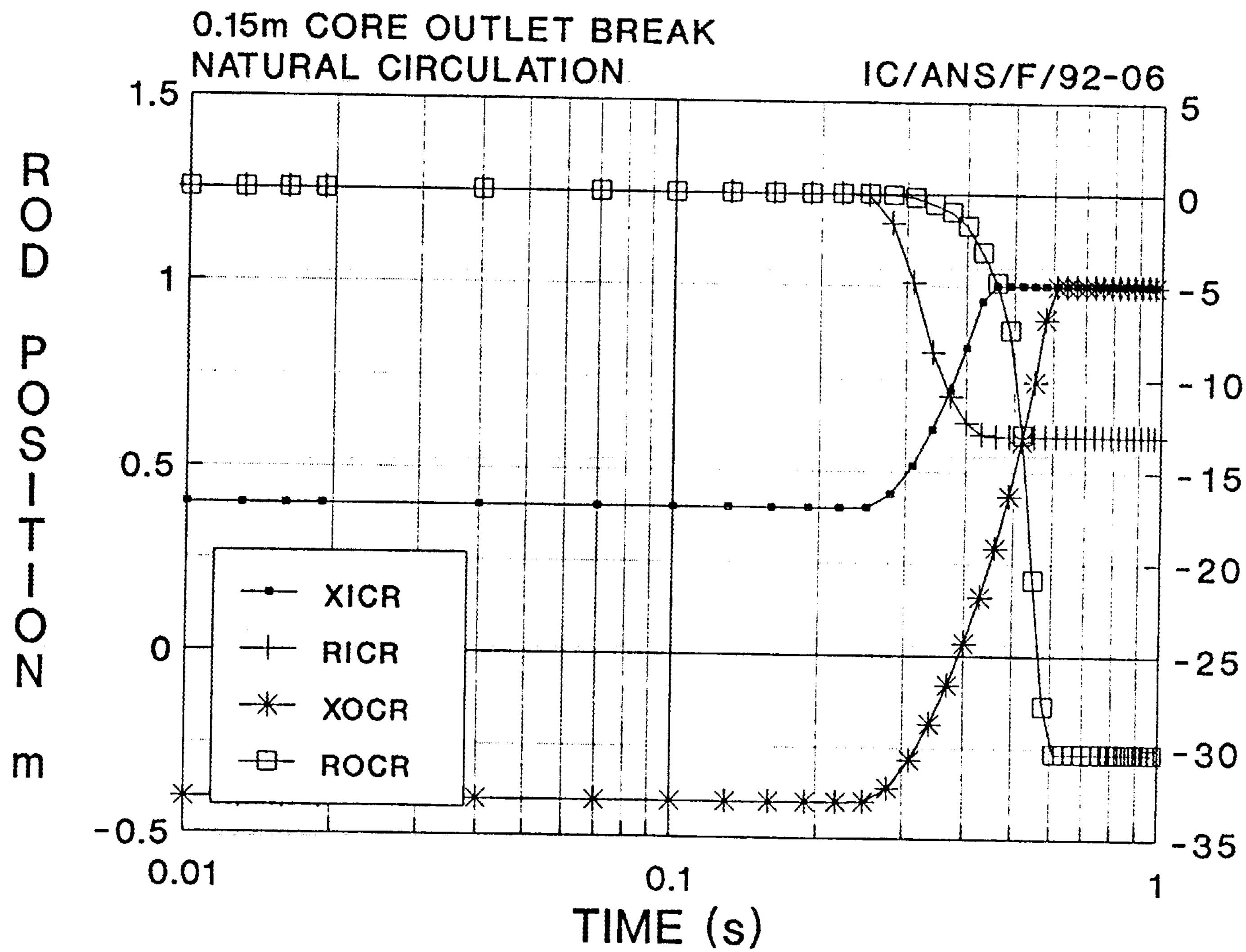


0.15m CORE OUTLET BREAK
NATURAL CIRCULATION
IC/ANS/F/92-03
TEMPERATURE C
TIME (s)
350
300
250
200
150
100
50
0
0.001
0.01
0.1
1
10
100
1000
10000
100000
THWHUF
THWHLF
THCHUF
THCHLF
TWACF
TCACF

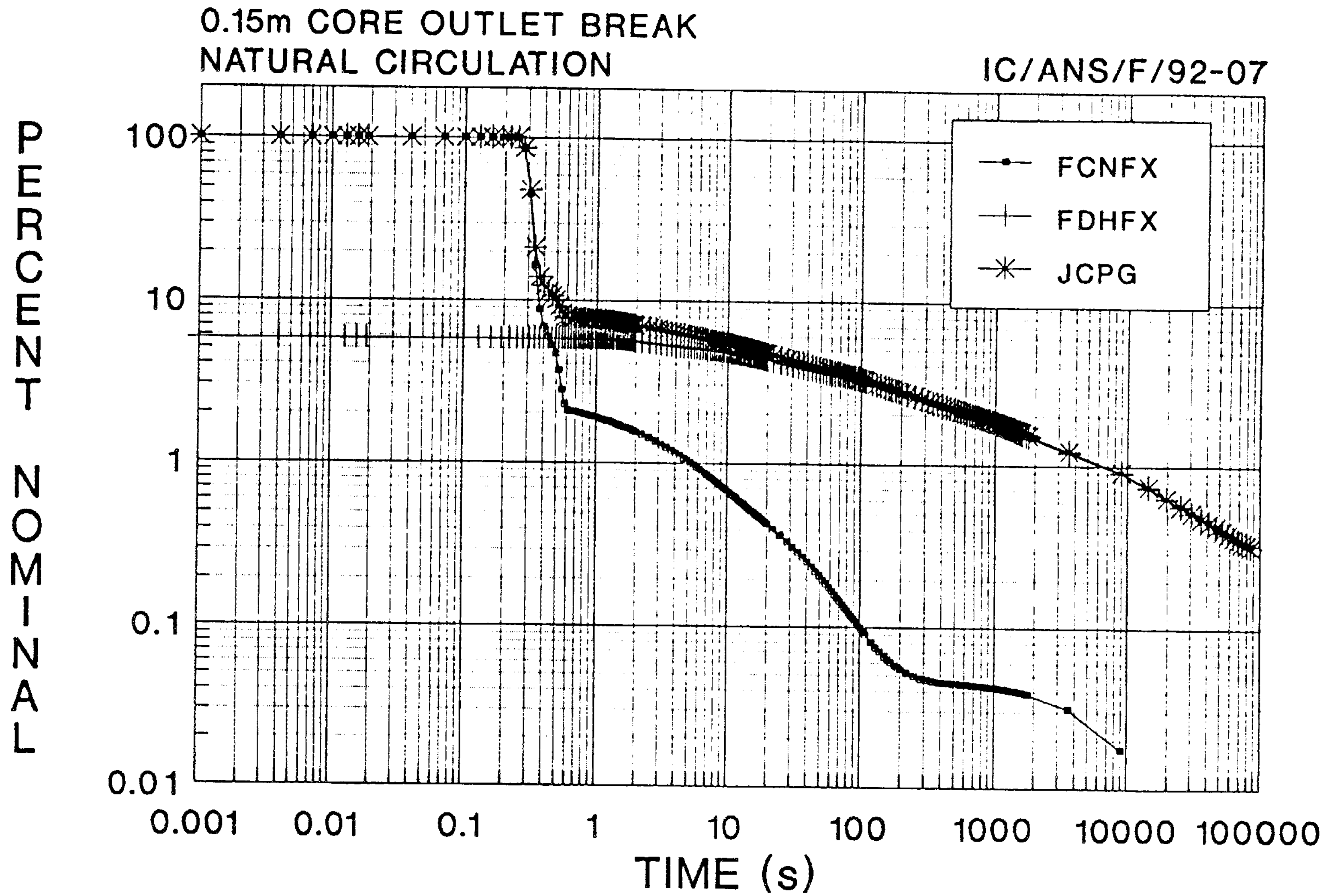

0.15m CORE OUTLET BREAK
NATURAL CIRCULATION
IC/ANS/F/92-04
MSRHCU
MCRHCU
MIRHCU
SAFETY RATIO
1000
100
10
1
0.1
0.001
0.01
0.1
1
10
100
1000
10000
100000
TIME (s)



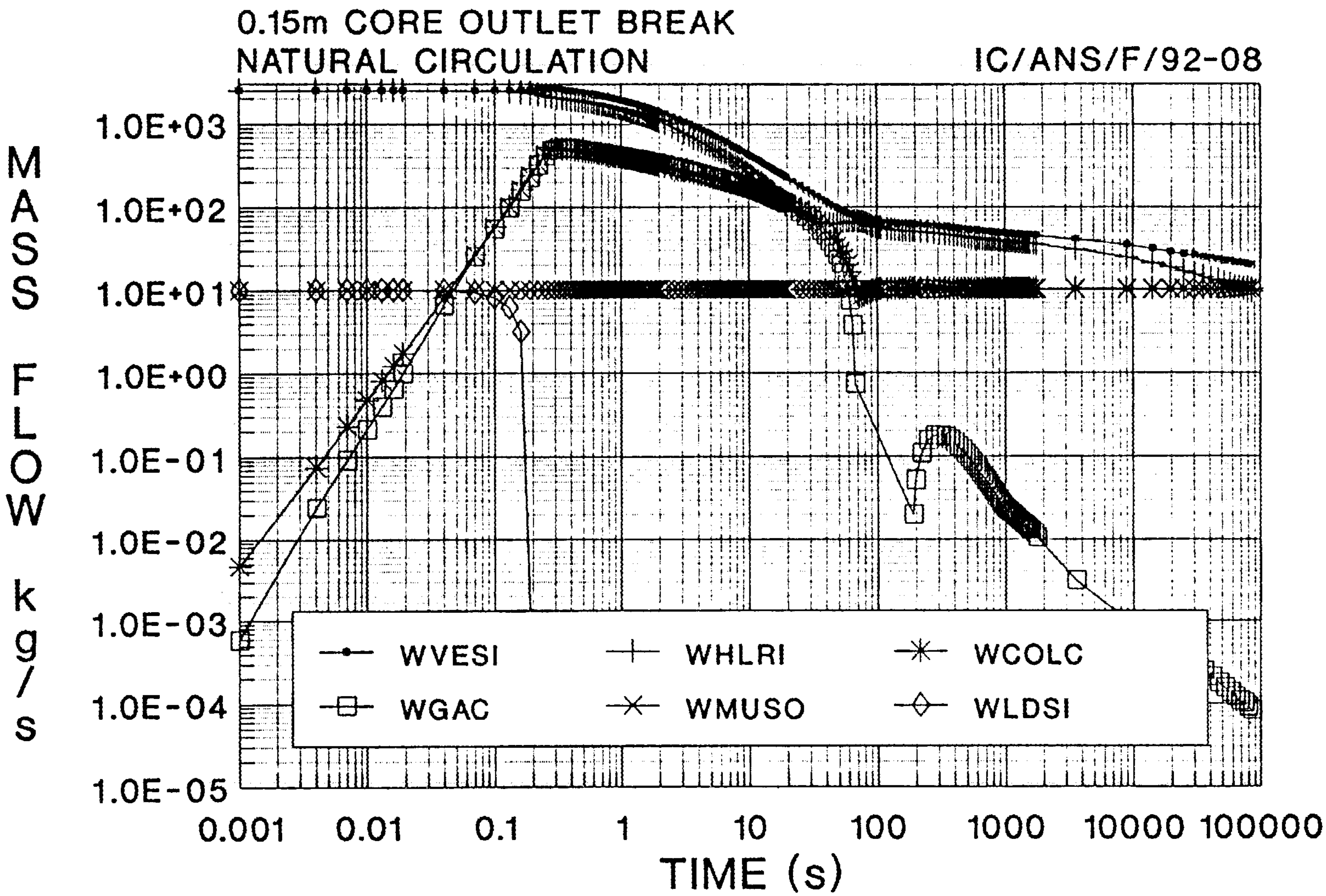
0.15m CORE OUTLET BREAK
NATURAL CIRCULATION
IC/ANS/F/92-05
MSRHCL
MCRHCL
MIRHCL
SAFETY RATIO
1000
100
10
1
0.001
0.01
0.1
1
10
100
1000
10000
100000
TIME (s)

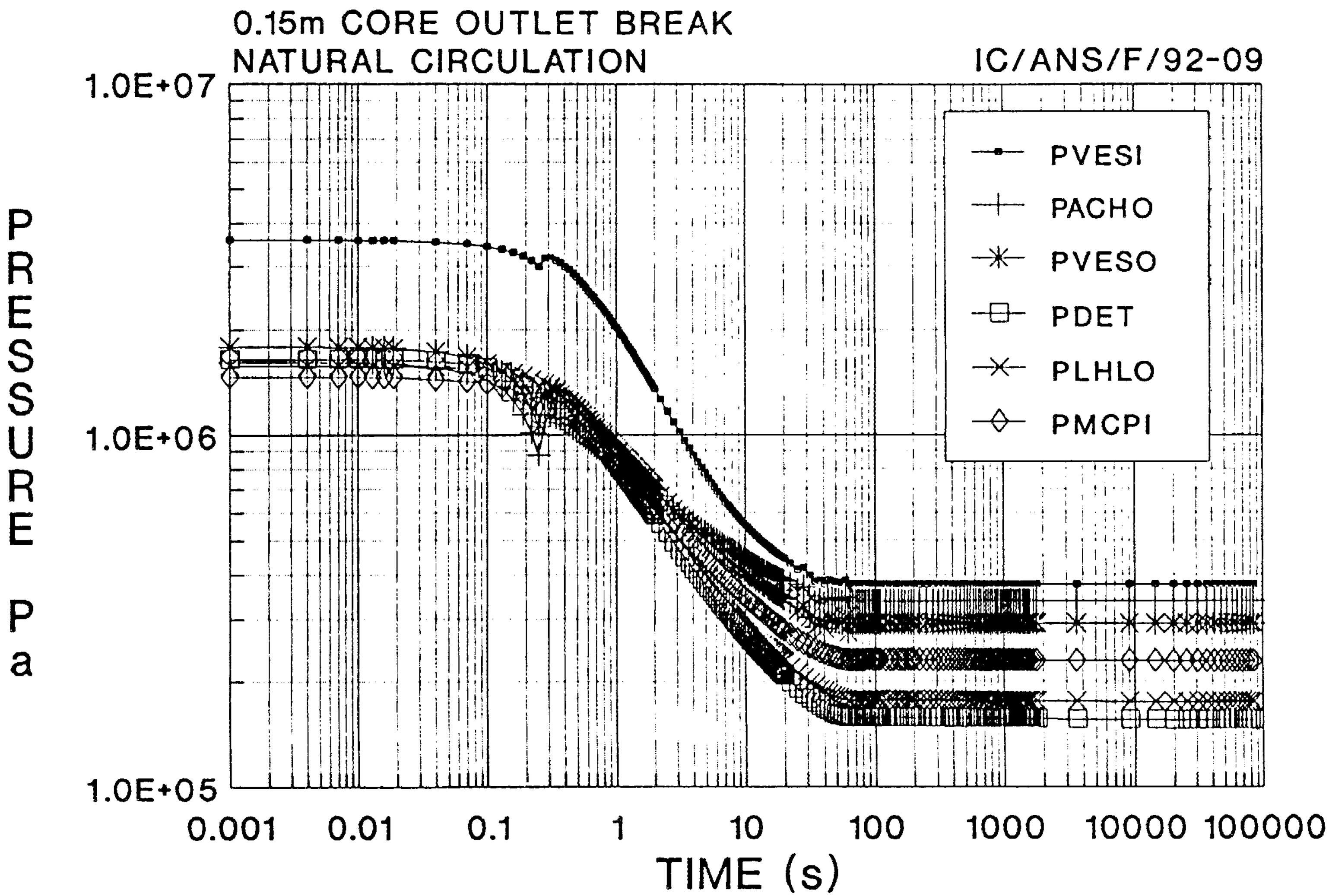


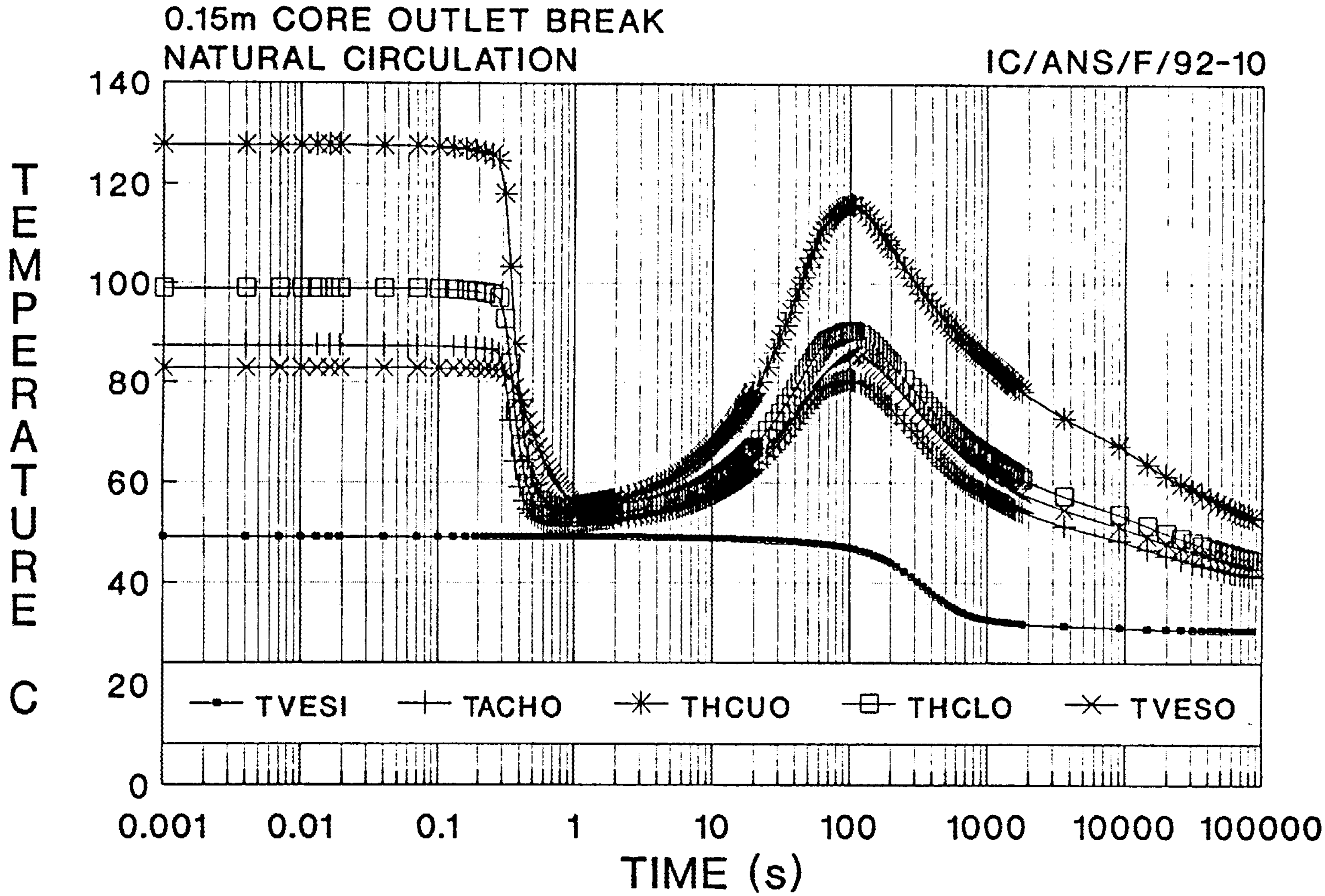
0.15m CORE OUTLET BREAK
NATURAL CIRCULATION
IC/ANS/F/92-06
ROD POSITION m
ROD WORTH $
TIME (s)
1.5
1
0.5
0
-0.5
5
0
-5
-10
-15
-20
-25
-30
-35
0.01
0.1
1
XICR
RICR
XOCR
ROCR



0.15m CORE OUTLET BREAK
NATURAL CIRCULATION
IC/ANS/F/92-07
PERCENT NOMINAL
100
10
1
0.1
0.01
FCNFX
FDHFX
JCPG
0.001
0.01
0.1
1
10
100
1000
10000
100000
TIME (s)


0.15m CORE OUTLET BREAK
NATURAL CIRCULATION
IC/ANS/F/92-08
MASS FLOW kg/s
1.0E+03
1.0E+02
1.0E+01
1.0E+00
1.0E-01
1.0E-02
1.0E-03
1.0E-04
1.0E-05
0.001
0.01
0.1
1
10
100
1000
10000
100000
TIME (s)
WVESI
WHLRI
WCOLC
WGAC
WMUSO
WLDSI


0.15m CORE OUTLET BREAK
NATURAL CIRCULATION
IC/ANS/F/92-09
1.0E+07
1.0E+06
1.0E+05
PRESSURE Pa
PVESI
PACHO
PVESO
PDET
PLHLO
PMCPI
0.001
0.01
0.1
1
10
100
1000
10000
100000
TIME (s)



0.15m CORE OUTLET BREAK
NATURAL CIRCULATION
IC/ANS/F/92-10
TEMPERATURE C
TIME (s)
140
120
100
80
60
40
20
0
0.001
0.01
0.1
1
10
100
1000
10000
100000
TVESI
TACHO
THCUO
THCLO
TVESO

ATTACHMENT 3

ANS DYNAMIC MODEL

A computer model of the Advanced Neutron Source (ANS) reactor and cooling system has been developed in house at the Oak Ridge National Laboratory (ORNL) for dynamic simulation studies. This model is intended as an aid to the designers, to help evaluate the effect of different design options on the transient performance of the ANS reactor under upset conditions. The present model has not been qualified for final safety analyses; however, given its interactive and modular nature, the model can be adapted quite easily to changes in reactor design and several options can then be evaluated quantitatively at design time.

Summary Model Description

The ANS Dynamic model has been programmed in the ACSL simulation language, fact that gives it fairly good flexibility of operation at run time. The model is composed of a collection of modules, most of which (for instance the PIPE module or the PUMP module) are reused through out the model. Figure IC/ANS/F-91/15 is a block diagram showing most of the components modeled. These components include:

(1) Core neutronics, delayed neutrons, and decay heat (based on ANS-specific correlations)

(2) Average channel fuel and coolant dynamics. The average channel determines the average core outlet conditions. A single node is used for these calculations.

(3) Hot channels fuel and coolant dynamics. The dynamics of the hot streak of the upper and lower core are simulated. The lower core is typically limited at beginning of cycle (BOC) and the upper core at end of cycle (EOC). Thus, in our model, we use the BOC axial power shape and hot streak factors for the lower core hot channel and the EOC conditions for the upper core. The hot channels are divided into up to 50 axial nodes (typically 27) where local temperatures, pressures, and heat fluxes are estimated to determine its margin to incipient boiling, critical heat flux, and flow excursion instability.

(4) A bypass region models the flow of heavy water that bypass the fuel elements inside the pressure vessel. This coolant is typically colder than the core outlet coolant, so that when it mixes, the vessel outlet temperature (which is computed dynamically) is lower than the core outlet temperature.

(5) A reflector region is modeled with a very simplified one node approach (to this date, the reflector cooling system is not yet properly defined). The reflector provides some (but not much) reactivity feedback to the core due to the direct neutron and gamma heating.

(6) Cooling system pipes are modeled, and they release heat to the appropriate surrounding light water pools.

(7) In-containment light water pools are modeled. This include the main reactor pool, the pipe chase pool, and the heat exchangers pool. These pools take heat from the reactor piping according to their relative temperature and based on natural convection heat transfer coefficients. The heat exchanger pool also cools the emergency heat exchanger secondary side by natural circulation.

(8) The main heat exchanger is modeled with the primary flow in the shell side and the secondary flow in the tube side. Heat transfer characteristics are adjustable; typically used values include a fouling heat transfer resistance factor.

(9) The emergency heat exchanger is modeled in series with the main heat exchanger. Primary flow is in the shell side and secondary flow is on the tube side. The shell side (primary) assumes "turbulizers" so that the flow is never laminar, regardless of Reynolds number. The tubes diameter is designed to be of the order of 0.05 m (2") so that the Reynolds number will be large enough to assure turbulent flow even at the low natural circulation flow rates. The secondary side of the emergency heat exchanger is connected to the heat exchangers pool and allow to flow by natural circulation.

(10) Main circulation pumps are modeled according to the head-flow characteristic curve. The characteristic curve scales the flow directly proportional to the pump rotational speed; the pump head is proportional to the square of the pump speed; and the power required is proportional to the cube of the speed. Pump coastdown is modeled based on a conservation of angular momentum; the resulting differential equation that is solved by the model is

$$\frac{dn}{dt} = (n^2 - n_0^2) / \tau$$

where n is the pump rotational speed, n_o is the desired equilibrium speed (for instance, n_o = 10% if a reduction to pony flow is desired), and τ is the pump half speed time constant. The coastdown flow and pump head are computed by scaling the characteristic pump curve using the calculated speed, n.

(11) The gas accumulator is assumed to follow the ideal gas law ($P V^\gamma$ = constant). In our model we assume that the accumulators expand isothermally (i.e., $\gamma = 1.0$). The initial gas to liquid ratio is such that the liquid level will not reach the bottom of the accumulator after the gas has expanded to the depressurized condition; for a 2.0 MPa core outlet pressure, the liquid to gas ratio is 20 to 1.

(12) The reactor pressure is maintained high by a makeup flow. The model simulates this flow with a pump module (the pressurizer pump) with a suction in a constant pressure tank (the cleanup system tank). The makeup pump speed is maintained constant unless a coastdown (i.e., loss of off site power) is required. During normal operation, the makeup flow adjusts itself to the system pressure; for instance, as the system pressure lowers, the makeup flow increases. These changes, however, are not sufficient to maintain constant pressure. The pressure regulation is accomplished by modulating the flow through the letdown valves. The letdown valves are modeled as a pressure drop with variable coefficient (according to valve opening); the letdown flow is collected in the letdown tank. The model does not simulate the low pressure cleanup system and this tank is assumed to have an infinite supply of D_2O, so that makeup can always be maintained. Makeup supply problems can be simulated at any time by tripping the makeup pump that is supplied with a perfect (i.e, no reverse flow) check valve.

(13) The secondary side of the ANS cooling system is represented by: (1) the secondary side of the main heat exchanger in the tube side, (2) secondary hot leg, (3) main cooling towers and cooling towers basin, (4) secondary circulation pump, and (5) secondary cold leg. All these components use approximations similar to those in the primary system. Indeed, for most of

them the same modules (for instance PIPE or PUMP) are used.

(14) A preliminary control system is simulated in the model. The control system includes: (1) control rod position based on the measured power-to-flow ratio, (2) pressure control, that actuates the letdown valve based on hot leg pressure measurements, and (3) core inlet temperature control, that actuates on the secondary flow based on the temperature measured at heat exchangers outlet.

(15) Sensor dynamics are modeled as first order lag systems. The required time constants have been determined through simulation of control and plant protection system challenges. The time constant currently in the model are those required to satisfy most design basis events requirements.

Model Limitations

The most important limitations of this model are:

(1) Point kinetics for the neutron dynamics in the core region. The power is distributed among different components (i.e, upper and lower cores, reflector, bypass region, ...) based on steady state power fraction distributions that have been estimated for the specific ANS conditions. This is not such a bad approximation since most transients result in a reactor scram within the first few milliseconds and then the power is determined by a decay heat correlation.

(2) Incompressible flow. The model is limited to liquid phase state; whenever a transient results in saturated boiling, the simulation fails. Note that the core typically is damaged (due to either critical heat flux or flow excursion instabilities) well before saturated boiling can be established and, thus, this approximation is fairly accurate except when acoustic wave propagation is a relevant effect (such as during large break LOCAs).

(3) Single loop flow dynamics. All three loops are simulated by one effective loop. Because of this approximation, the model is not able to simulate imbalances between loops; for instance, we can not model the shutdown of one pump while the other two remain on.

(4) No reverse flow. The model fails if reverse flow is established. Note that the core would be damaged under most conditions if the flow were reversed in any case.

Model Applications

The figures enclosed show an array of example applications related to the ANS cooling system design. Most of these analyses were performed in FY90 with the so called CCD core design, peaking factor assumptions, and uncertainties. Many of the core conditions and model assumptions have changed since then, so that these figures are shown for illustration purposes only.

Figures IC/ANS/F/90-61, 90-60, 90-75, 90-74, 90-66, 90-72, 90-64, 90-73, and 90-68 show pump induced transients. The figure captions are more or less self-explanatory. Based on these al later analyses, we determined to have main circulation pump coastdown times of 2 s (this is the time for the pump to reach 50% speed following a complete loss of power). The main pumps will be supported by battery driven "pony" motors that will maintain a 10% speed; this will result in about

10% flow with ideally about a 0.1% power consumption than at nominal conditions (changes in pump efficiency at low speed will result in increased power consumption, but we are not modeling it in such detail). These figures also show that the ANS cooling system is capable to sustain natural circulation without any incipient boiling a few minutes after shutdown occurs. New analysis using the most recent axial power shapes (i.e., fuel grading) show that critical heat flux is not violated even if natural circulation were to be established immediately after shutdown (allowing for the normal pump coastdown) even if a depressurization occurs at the same time (for instance due to a loss of off site power that knocks down the pressurizer makeup pump). If these analyses are confirmed, this would indicate that the pony motors, although desirable, are not required and do not need to be safety grade.

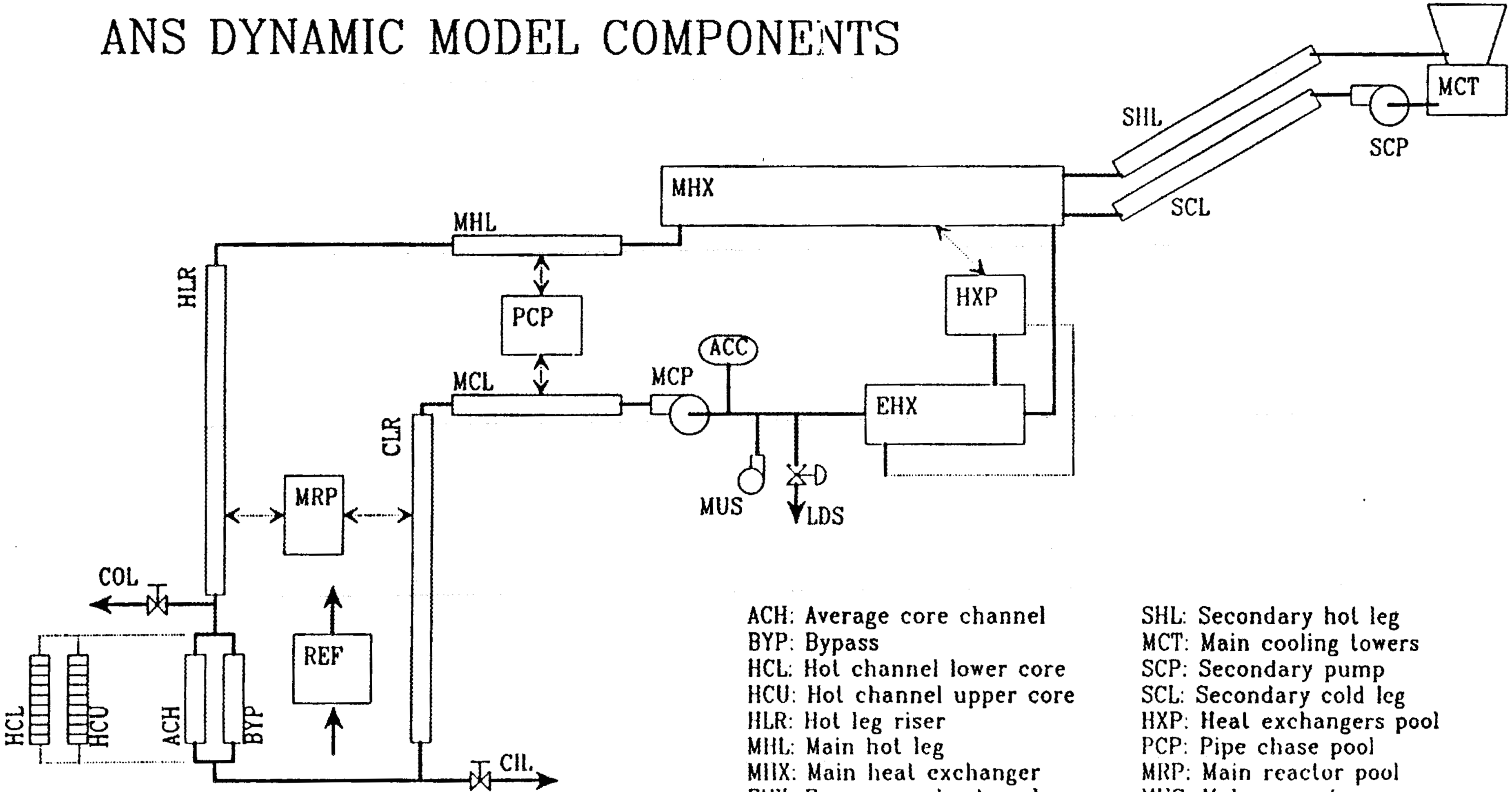
Figures IC/ANS/F/90-29, 90-31, 90-84, and 90-85 show some containment isolation or loss of normal heat sink transients. These analyses were performed to estimate the size of the emergency heat exchangers. These analyses showed that one of the main problems during this upset conditions was the transition to laminar flow in the heat exchanger tubes. The laminar flow heat transfer is orders of magnitude lower than turbulent heat transfer and, thus, large heat transfer areas (of the order of 25% of the main heat exchanger area was required). By increasing the tube diameter of the emergency heat exchanger to 0.05 m (2"), the flow remained turbulent for most of the transient and satisfactory results were obtained. The size of the in-containment pools was also studied. Fig 90-85 shows that a 300 m^3 heat exchanger pool will maintain the bulk coolant temperature below the 100°C goal and avoid saturated boiling at the top of the loop. For these analyses, heat transfer from the bare pipes to the other containment pools is also considered.

Figures IC/ANS/F/90-93, 90-95, 90-98, and 90-99 represent loss of coolant accident (LOCA) scenarios. At that time, our main consideration was to maintain the pump suction pressure above the net positive suction head (that we assumed to be 0.1 MPa or atmospheric) and avoid cavitation that could compromise the establishment of pony flow later during the transient. The accumulators were located in the pump suction side to maintain the maximum possible net positive suction head. The maximum size of the break for which the pump survives the transient depends on the pump coastdown time constant, and that is one reason to maintain it as low as possible.

Concluding Remarks

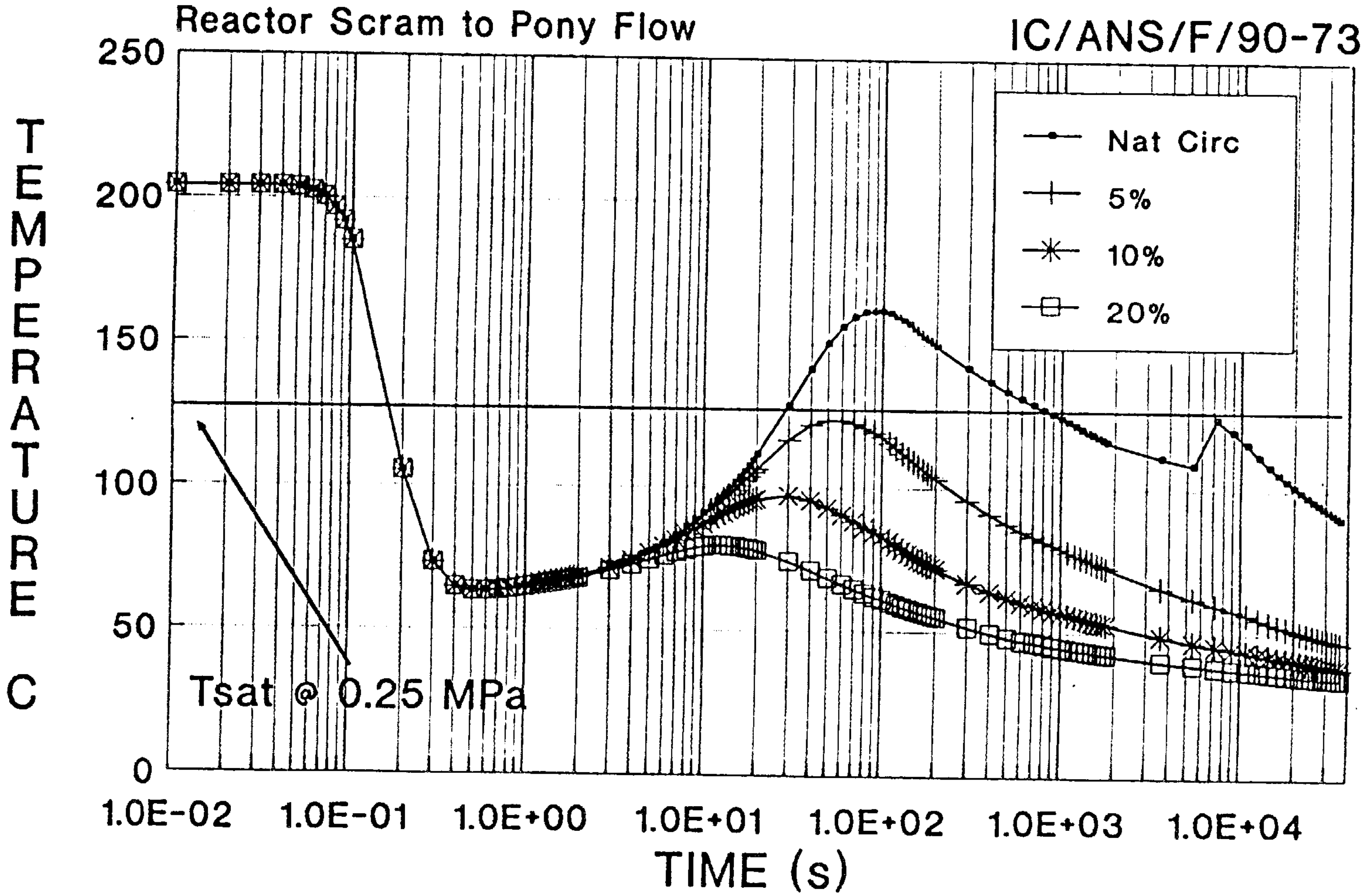
In summary, a model has been developed for dynamic simulations of the Advanced Neutron Source Reactor and its associated cooling systems. The model runs in quasi real time in a desktop workstation and can be easily modified to reflect changes in reactor design; thus, making it ideal for design checks and decisions.

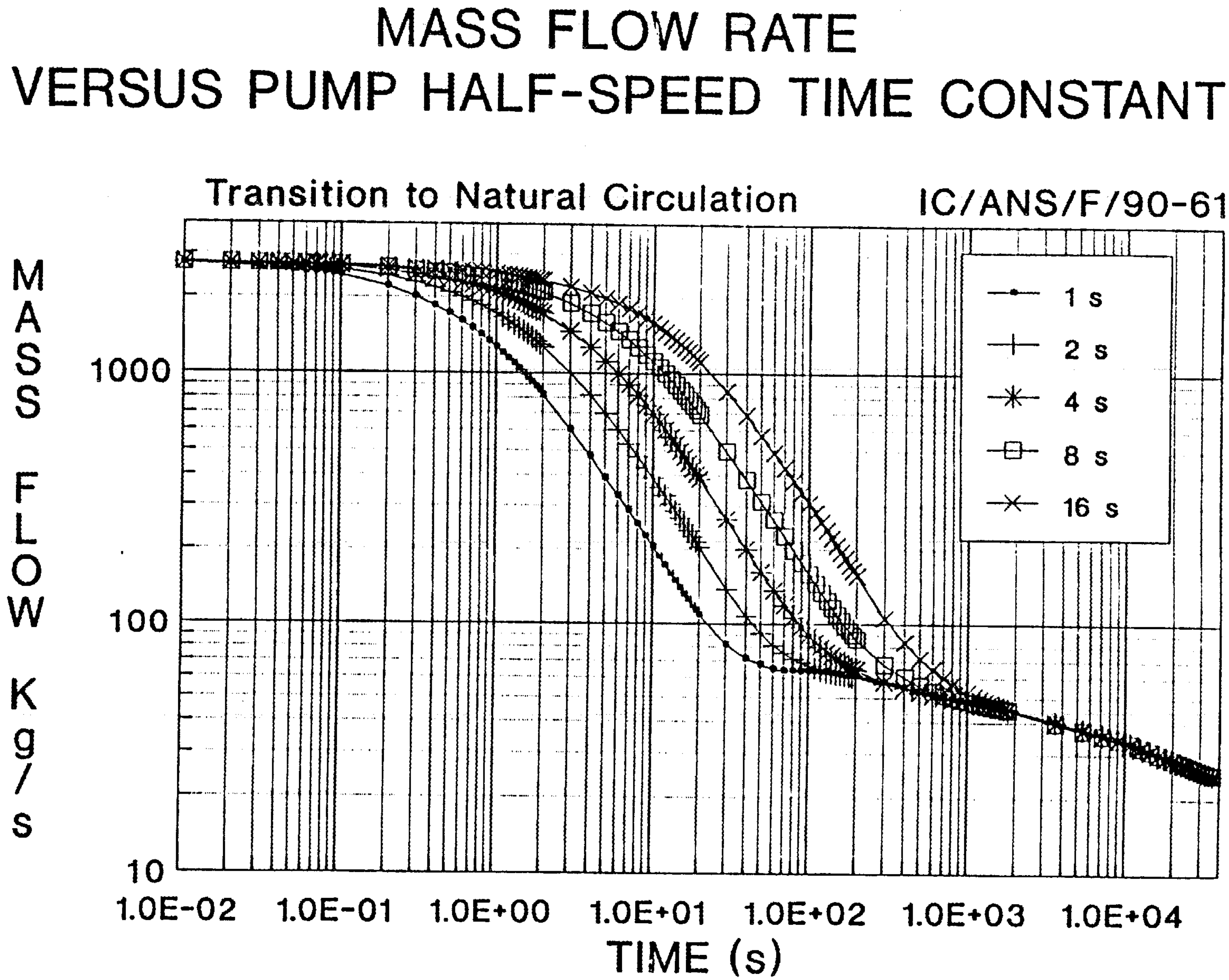
ANS DYNAMIC MODEL COMPONENTS



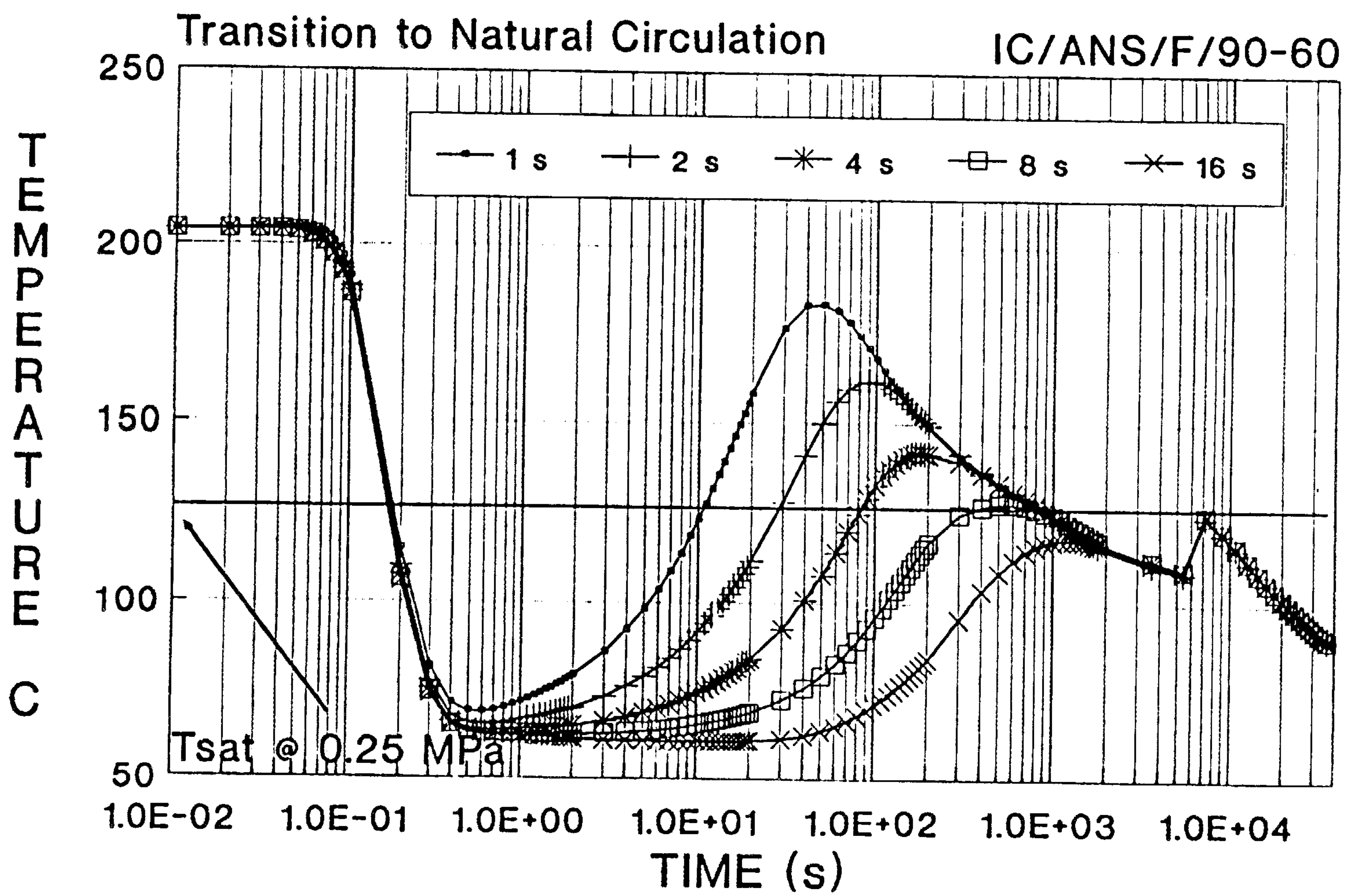
ACH: Average core channel
BYP: Bypass
HCL: Hot channel lower core
HCU: Hot channel upper core
HLR: Hot leg riser
MHL: Main hot leg
MHX: Main heat exchanger
EHX: Emergency heat exchanger
ACC: Gas accumulator
MCP: Main coolant pump
MCL: Main cold leg
CLR: Cold leg riser
SHL: Secondary hot leg
MCT: Main cooling towers
SCP: Secondary pump
SCL: Secondary cold leg
HXP: Heat exchangers pool
PCP: Pipe chase pool
MRP: Main reactor pool
MUS: Makeup system
LDS: Letdown system
CIL: Core inlet leak
COL: Core outlet leak

HOT SPOT SURFACE TEMPERARTURE VERSUS PONY MOTOR SPEED

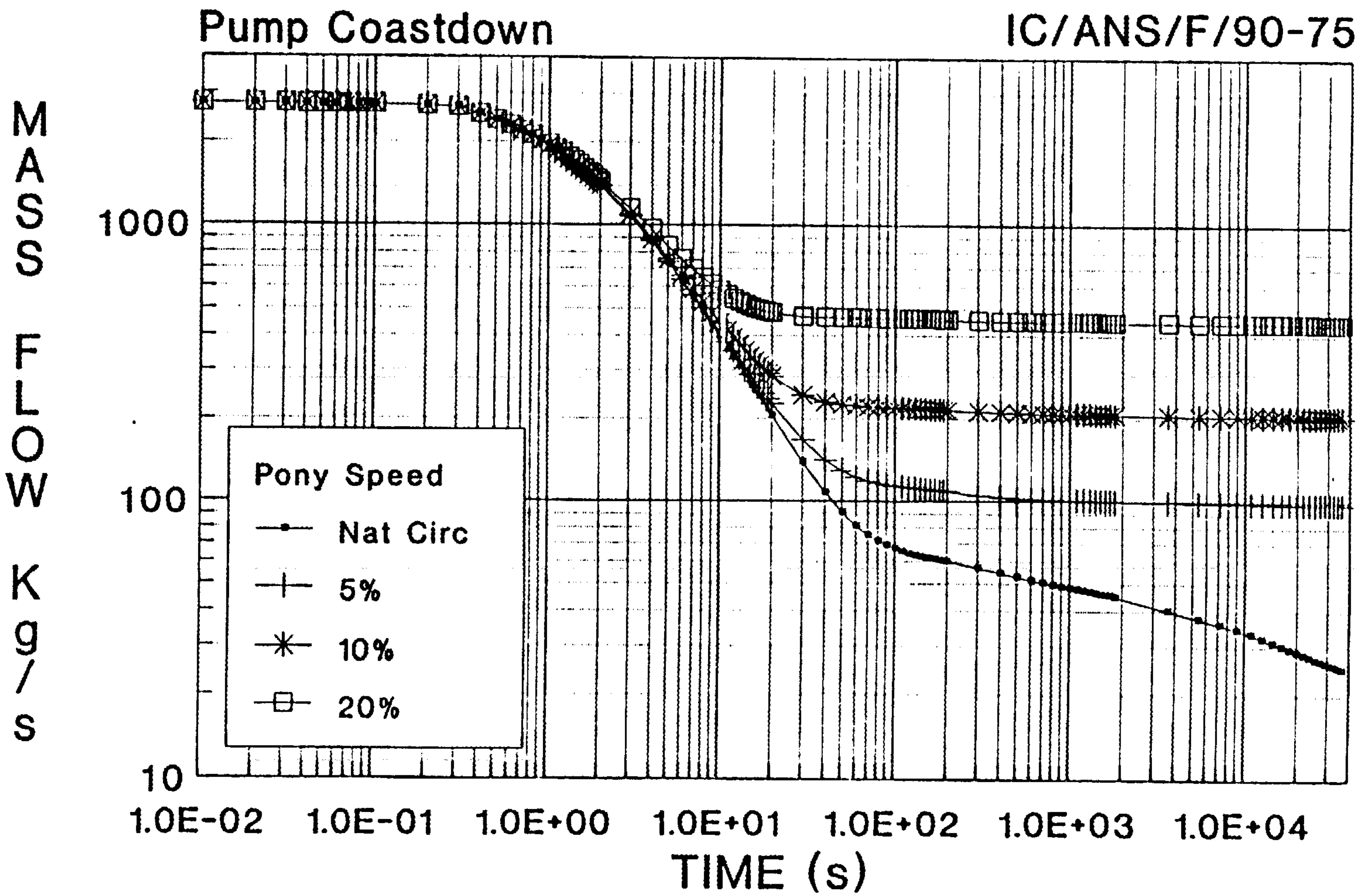



MASS FLOW RATE
VERSUS PUMP HALF-SPEED TIME CONSTANT
Transition to Natural Circulation
IC/ANS/F/90-61
MASS FLOW Kg/s
1000
100
10
1.0E-02
1.0E-01
1.0E+00
1.0E+01
1.0E+02
1.0E+03
1.0E+04
TIME (s)
1 s
2 s
4 s
8 s
16 s

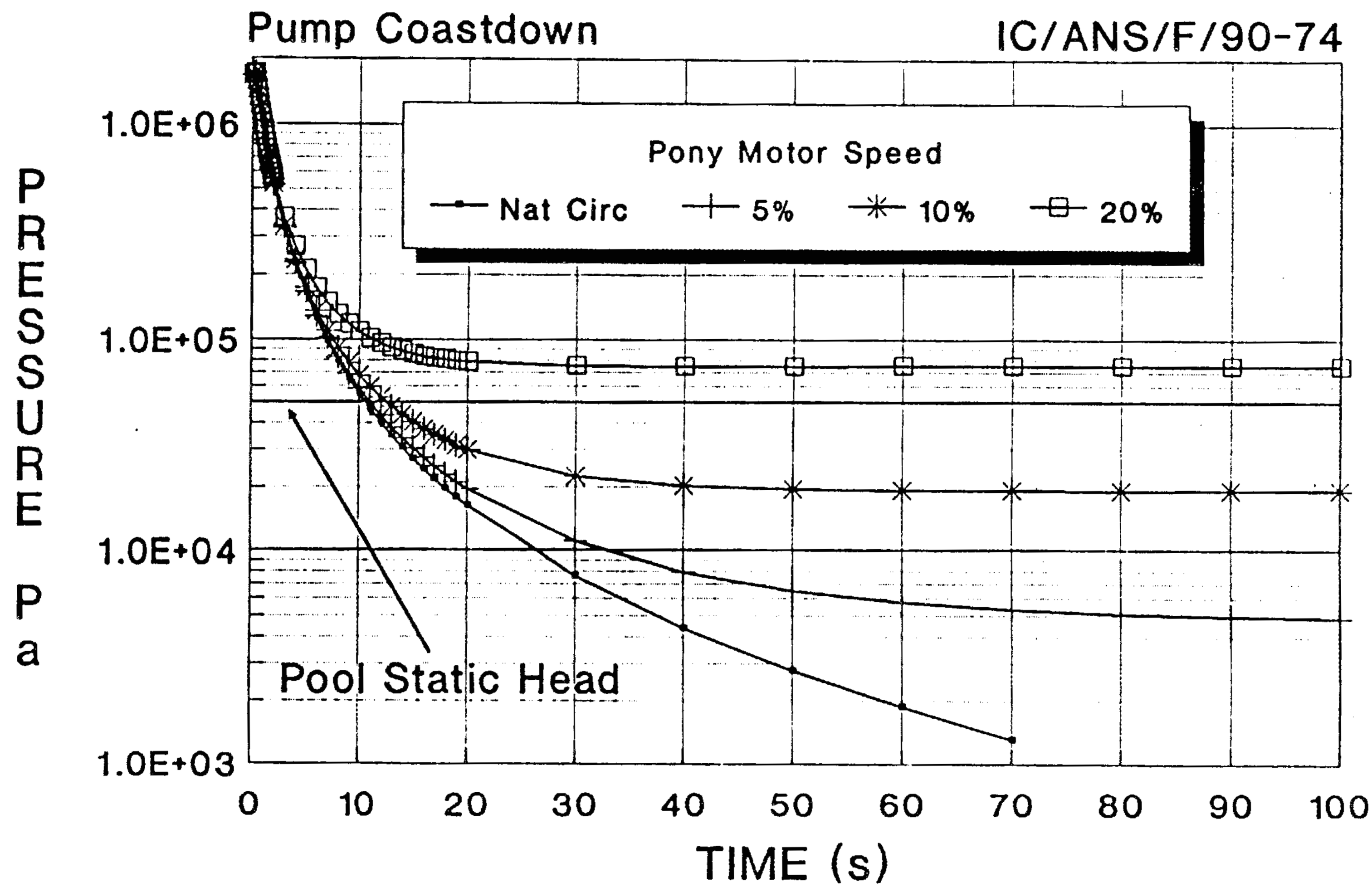
HOT SPOT SURFACE TEMPERARTURE VERSUS PUMP HALF-SPEED TIME CONSTANT



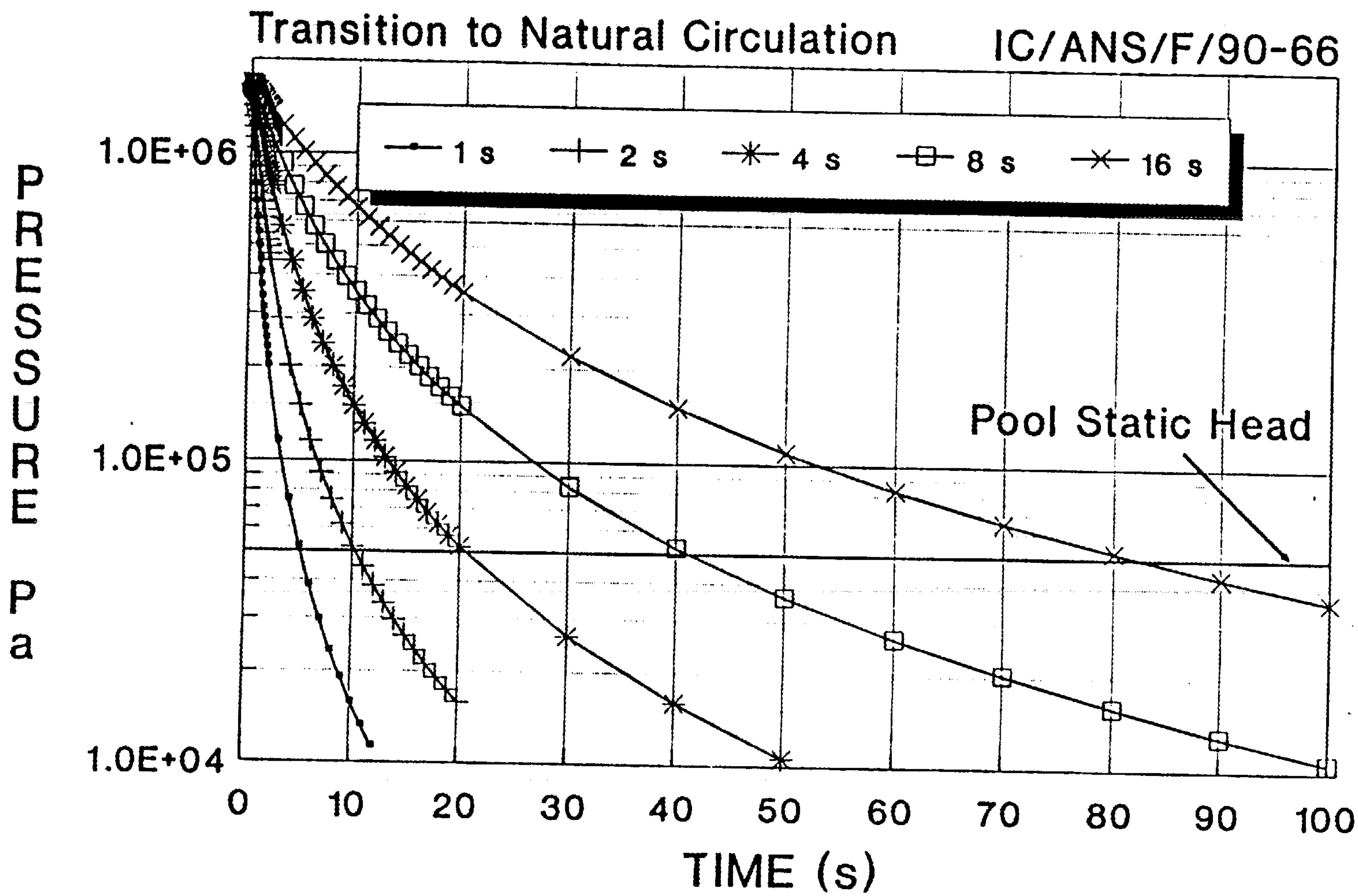
MASS FLOW RATE VERSUS PONY MOTOR SPEED



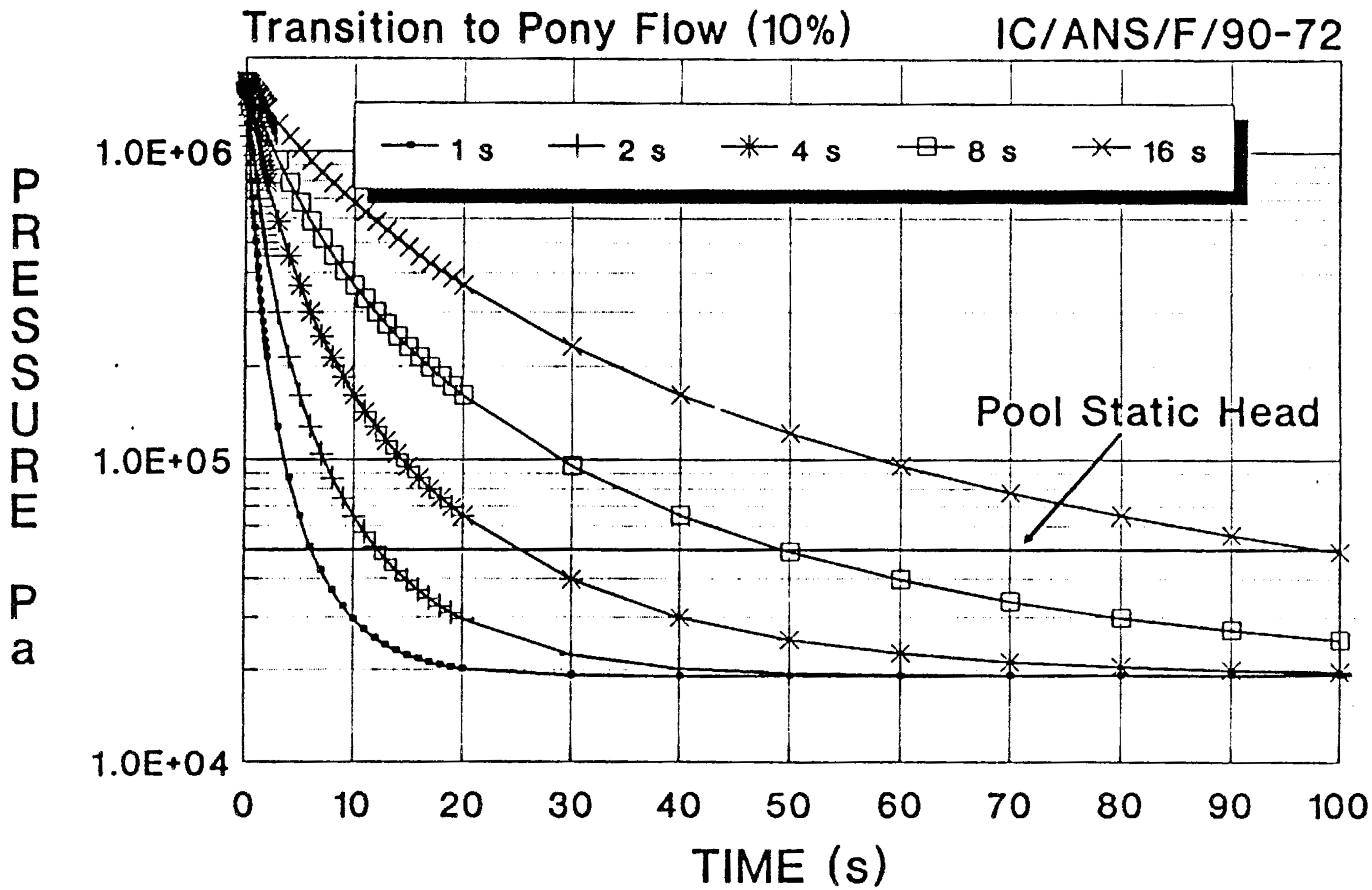
PUMP DRAWDOWN
VERSUS PONY MOTOR SPEED



PUMP DRAWDOWN
VERSUS PUMP HALF-SPEED TIME CONSTANT

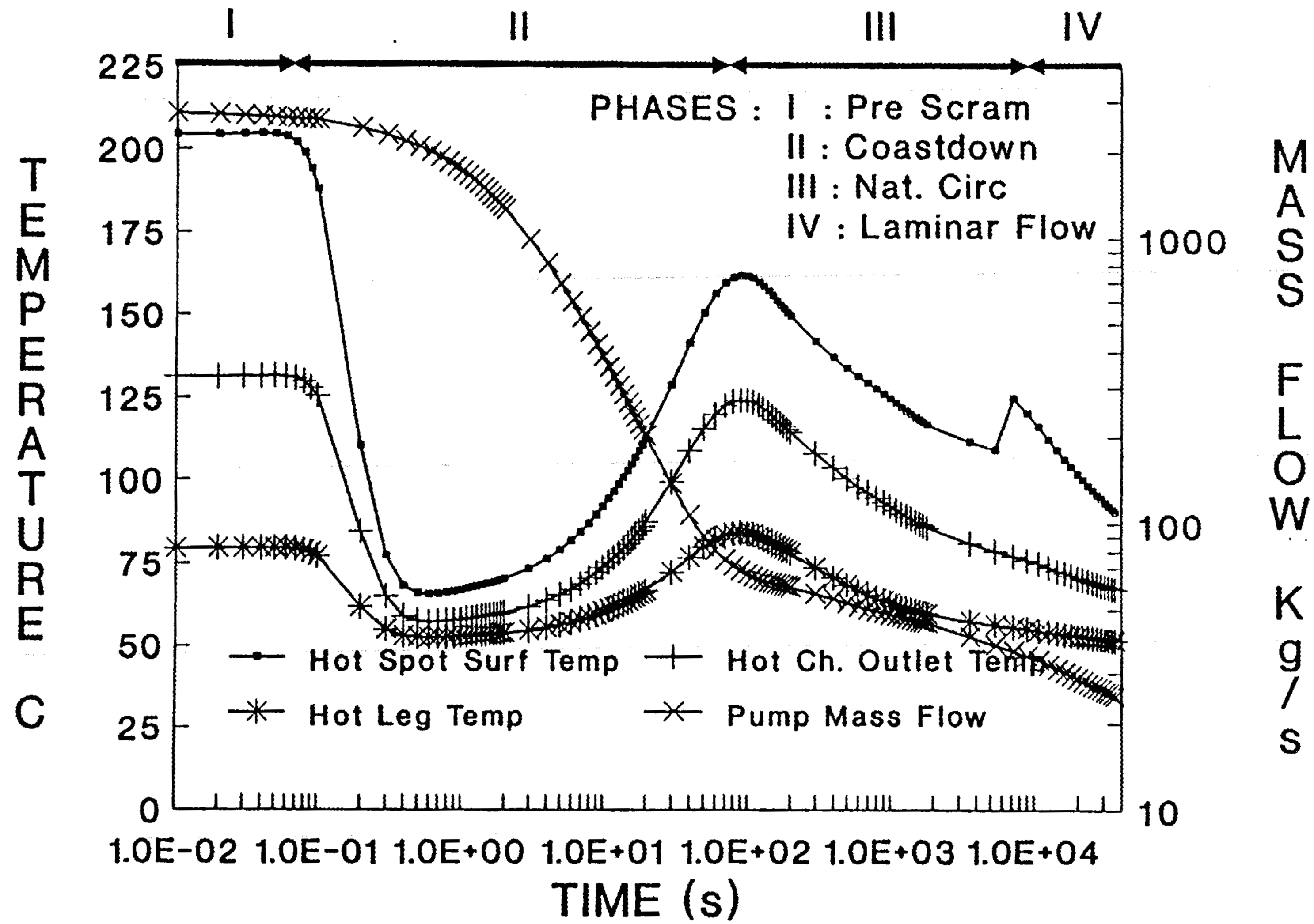


PUMP DRAWDOWN
VERSUS PUMP HALF-SPEED TIME CONSTANT

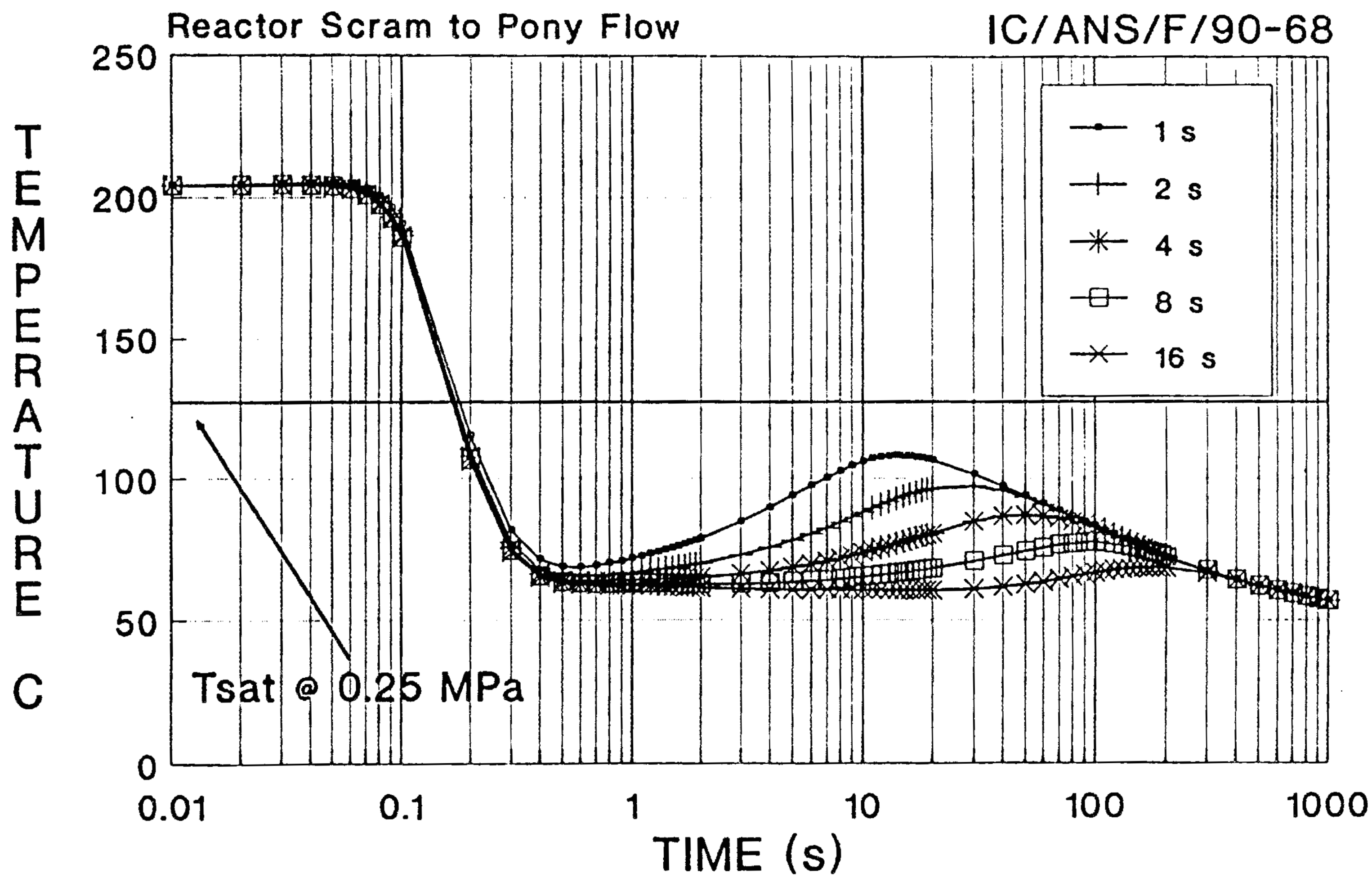


PUMP COASTDOWN TO NATURAL CIRCULATION

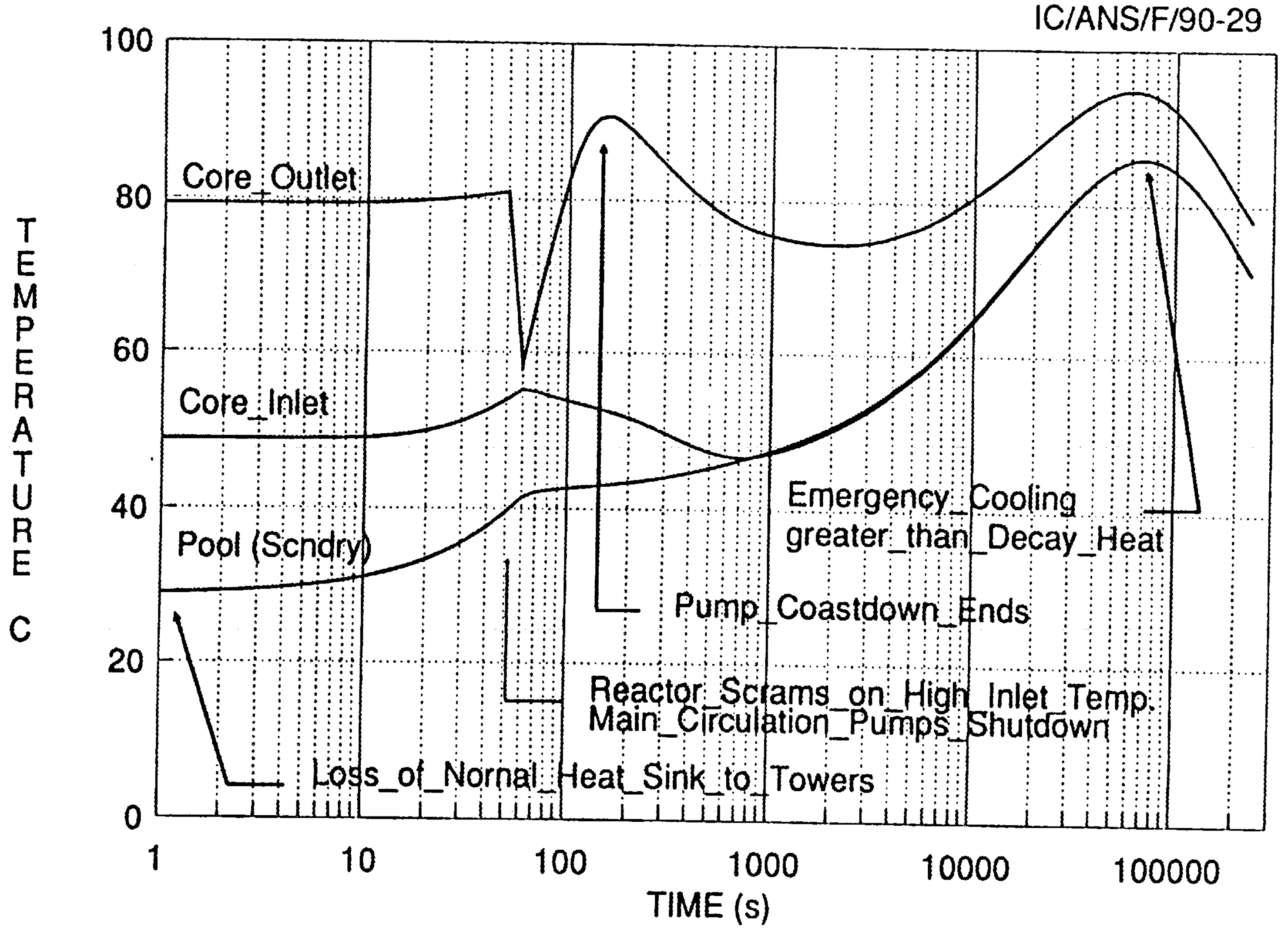
IC/ANS/F/90-64



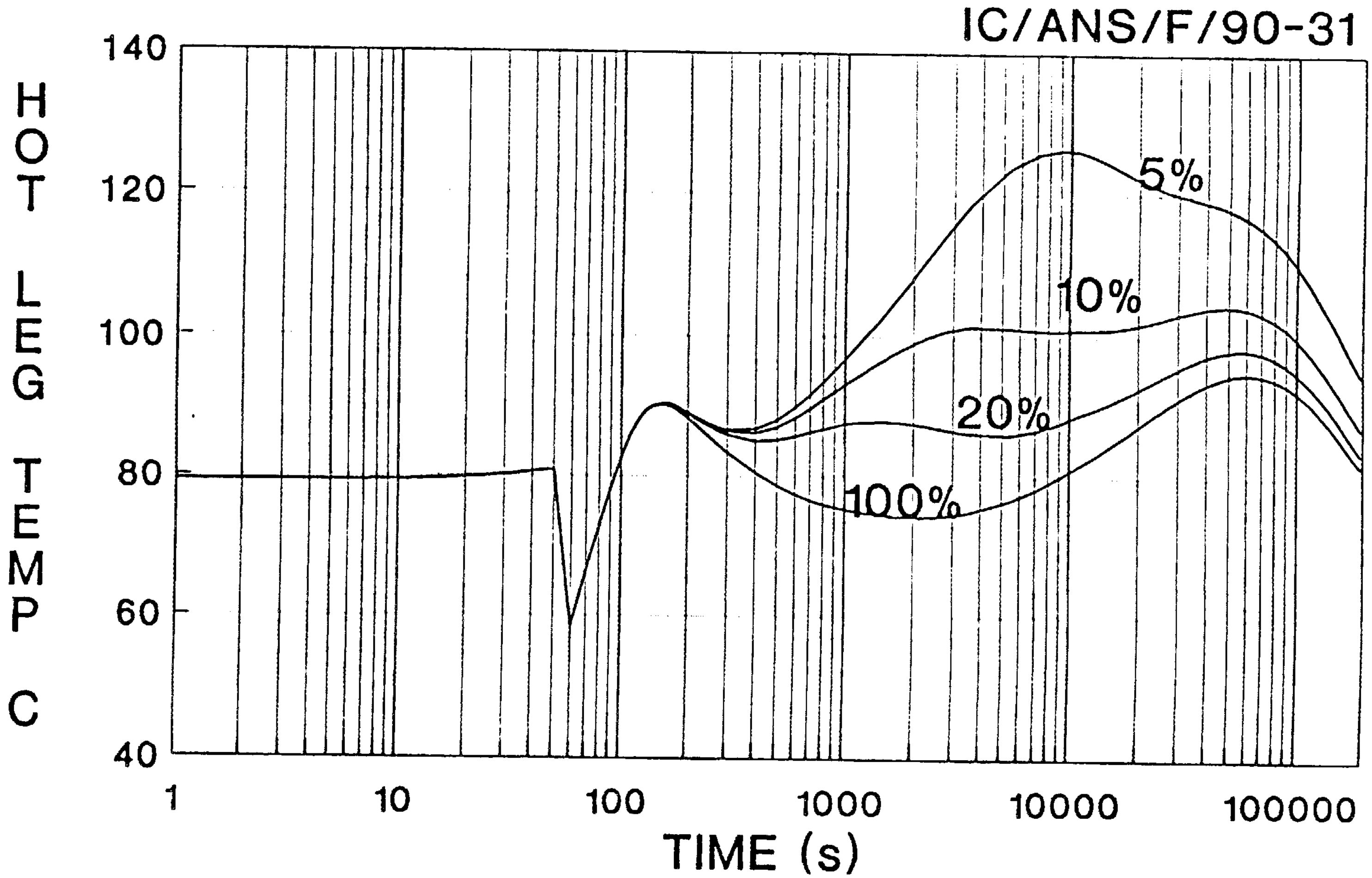
HOT SPOT SURFACE TEMPERARTURE VERSUS PUMP HALF-SPEED TIME CONSTANT

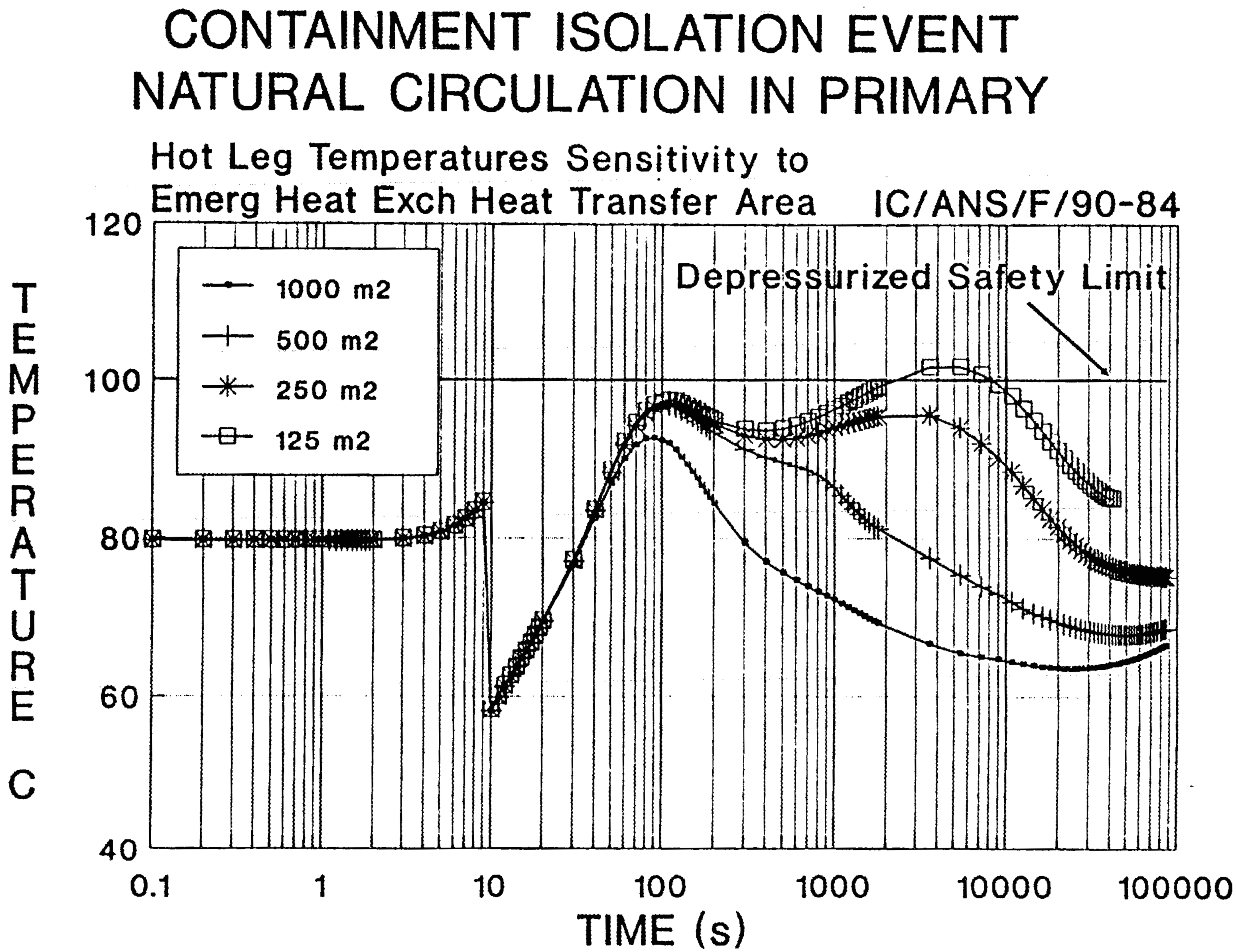


LOSS OF NORMAL HEAT SINK

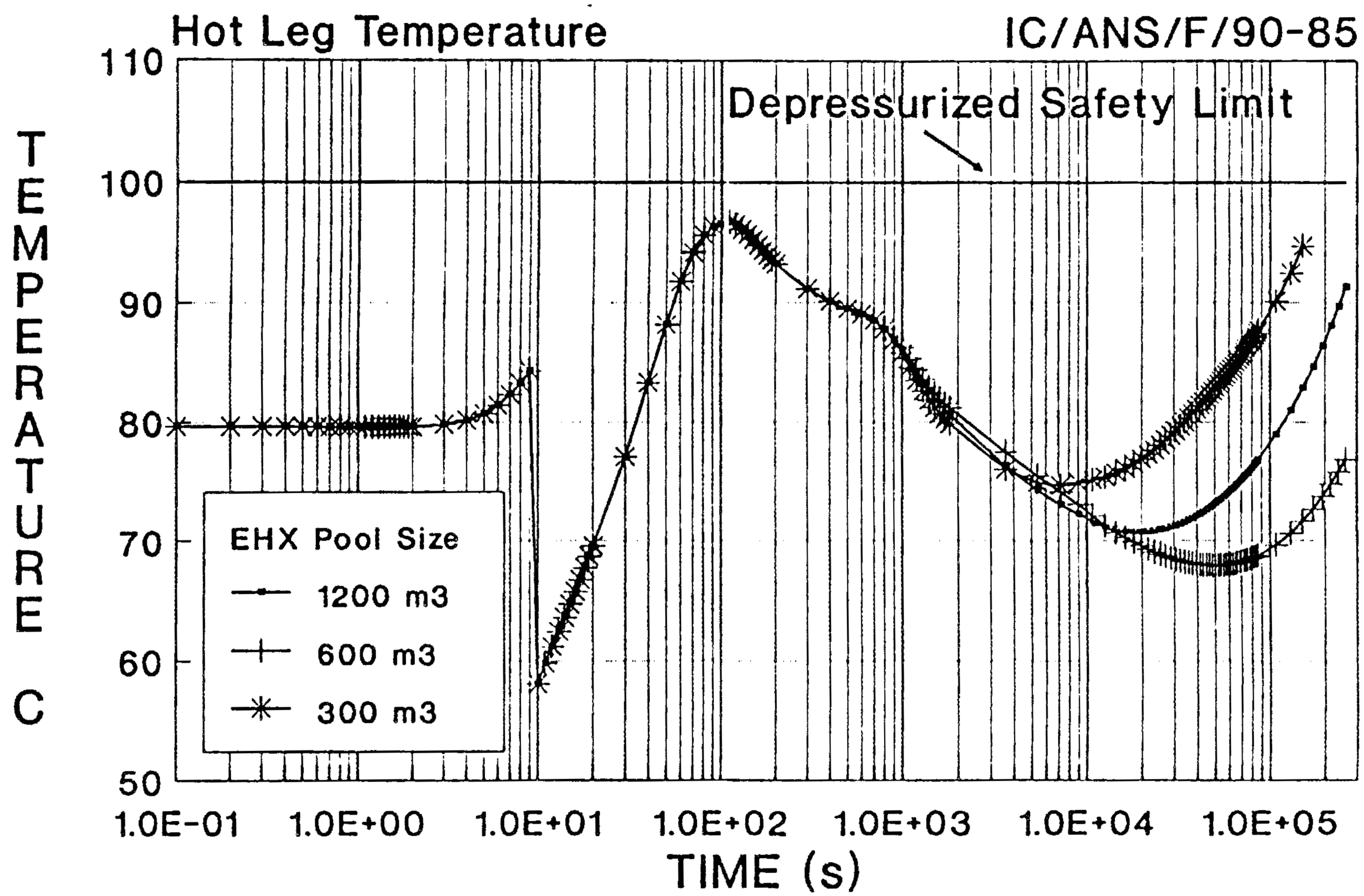


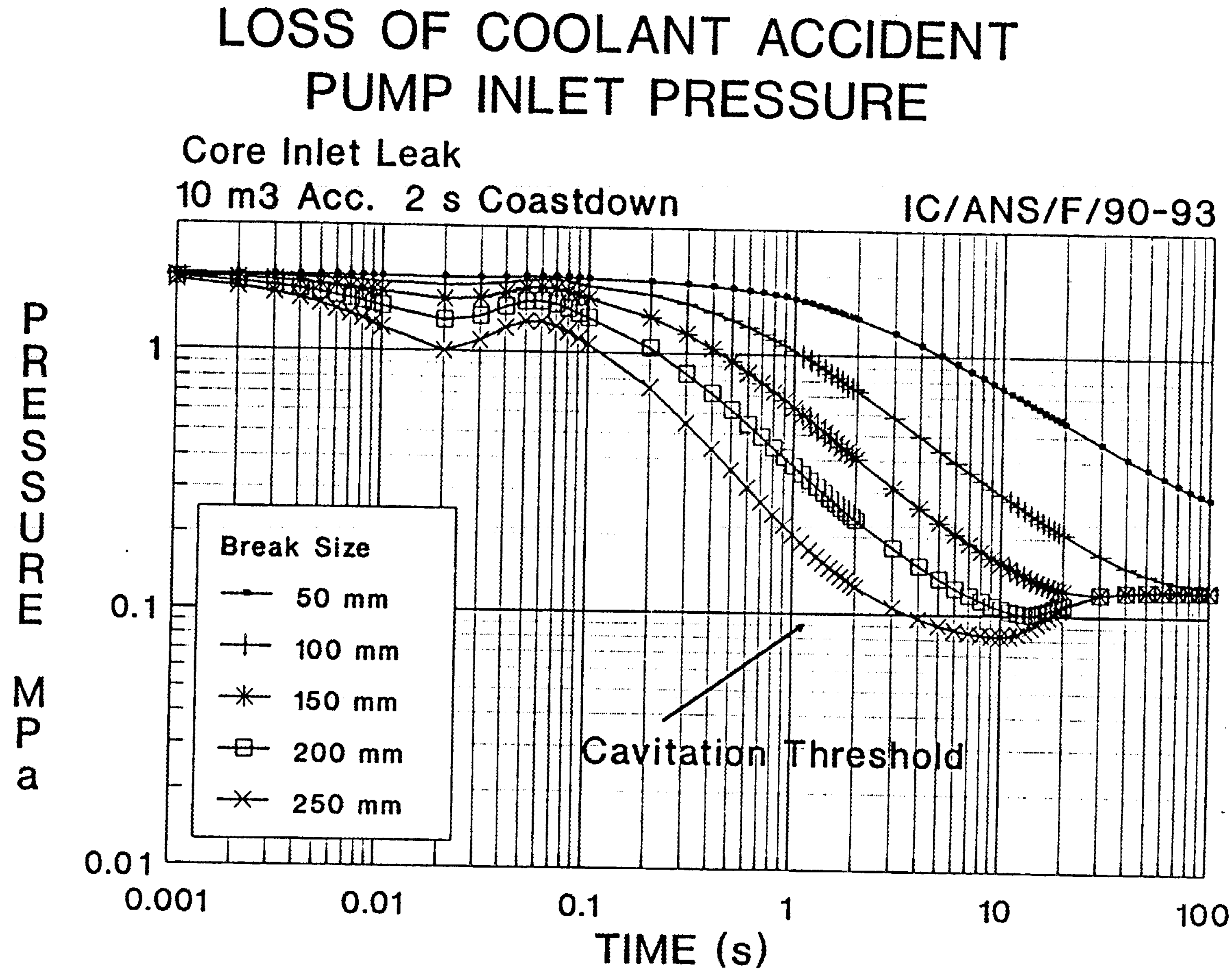
LOSS OF NORMAL HEAT SINK
vs AVAILABLE HEAT EXCHANGER AREA

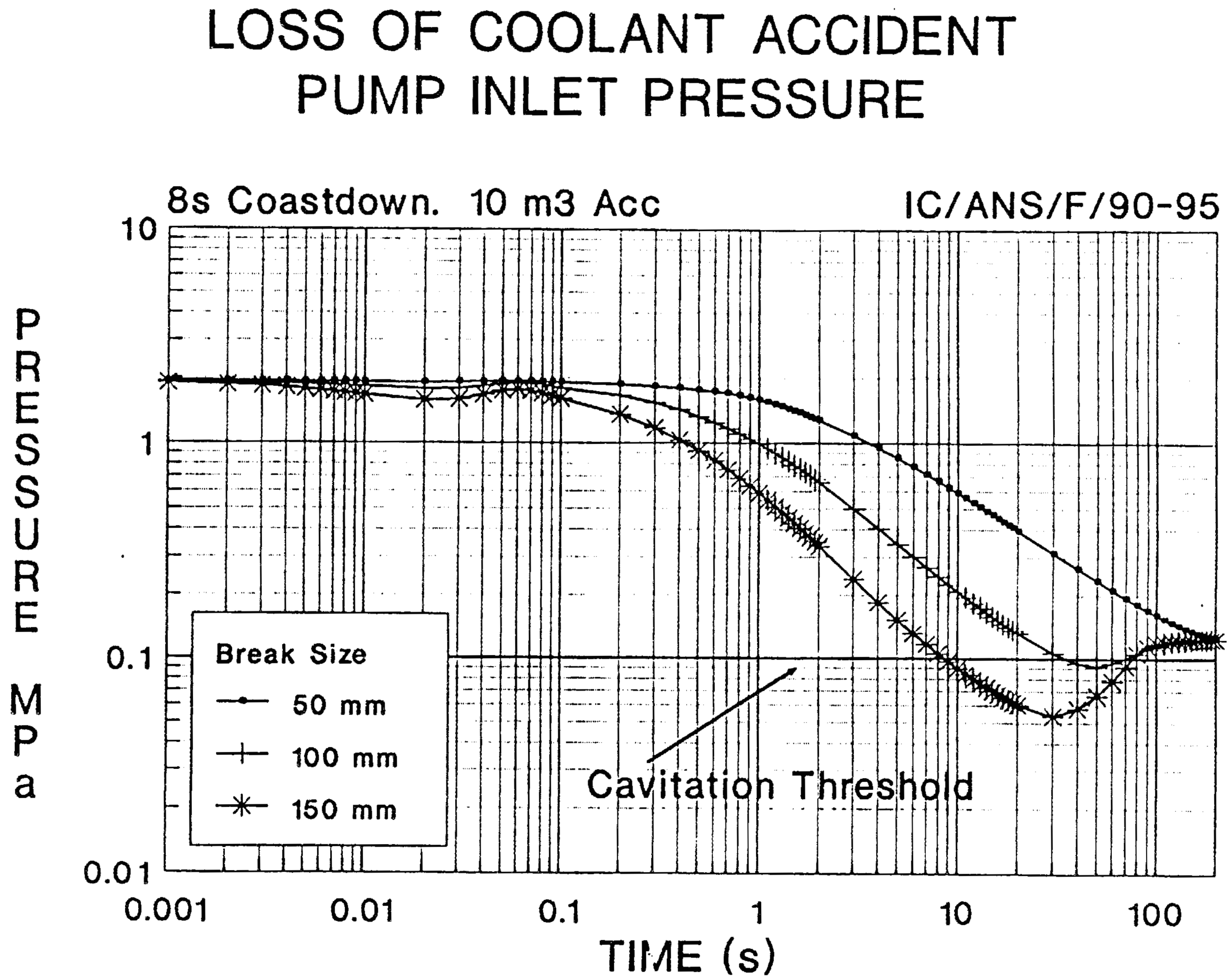



CONTAINMENT ISOLATION EVENT
NATURAL CIRCULATION IN PRIMARY
Hot Leg Temperatures Sensitivity to
Emerg Heat Exch Heat Transfer Area
IC/ANS/F/90-84
Depressurized Safety Limit
1000 m2
500 m2
250 m2
125 m2
TEMPERATURE C
120
100
80
60
40
0.1
1
10
100
1000
10000
100000
TIME (s)

CONTAINMENT ISOLATION EVENT
SENS TO EMERG HEAT EXCH POOL SIZE




LOSS OF COOLANT ACCIDENT
PUMP INLET PRESSURE
Core Inlet Leak
10 m3 Acc. 2 s Coastdown
IC/ANS/F/90-93
PRESSURE MPa
1
0.1
0.01
0.001
0.01
0.1
1
10
100
TIME (s)
Break Size
50 mm
100 mm
150 mm
200 mm
250 mm
Cavitation Threshold


LOSS OF COOLANT ACCIDENT
PUMP INLET PRESSURE
8s Coastdown. 10 m3 Acc
IC/ANS/F/90-95
PRESSURE MPa
TIME (s)
Break Size
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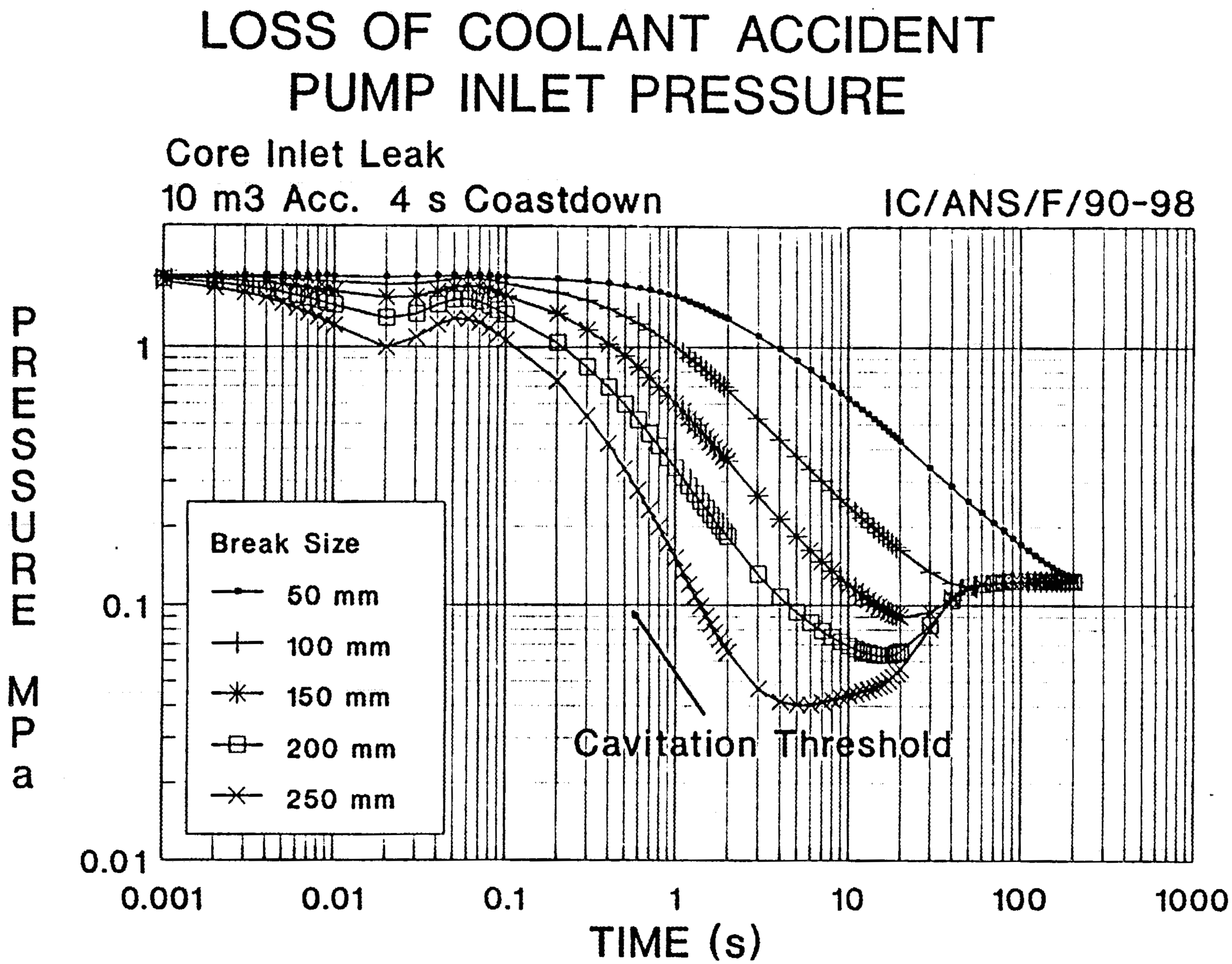
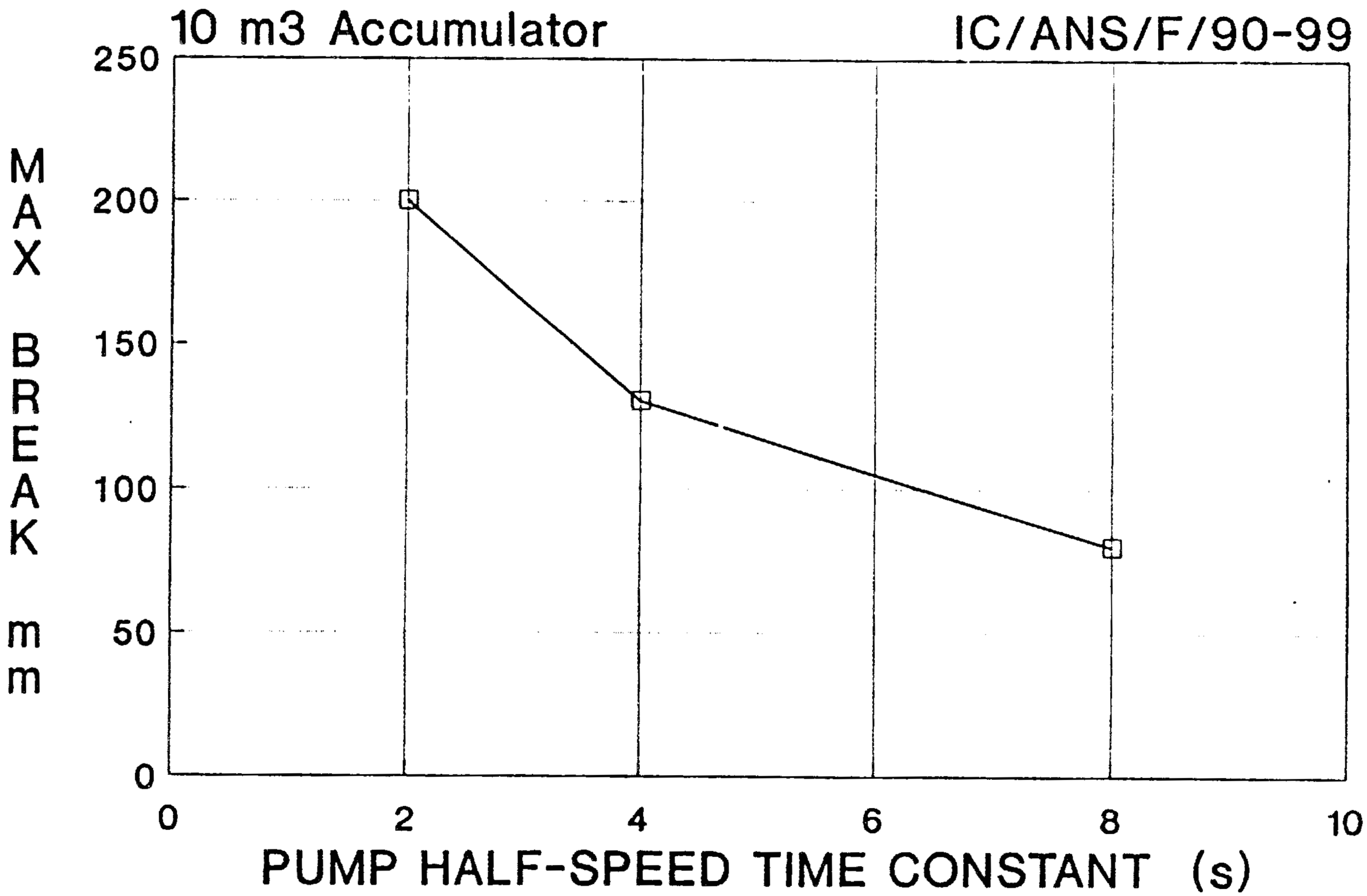
100 mm
150 mm
Cavitation Threshold
10
1
0.1
0.01
0.001
0.01
0.1
1
10
100


LOSS OF COOLANT ACCIDENT
PUMP INLET PRESSURE
Core Inlet Leak
10 m3 Acc. 4 s Coastdown
IC/ANS/F/90-98
PRESSURE MPa
1
0.1
0.01
0.001
0.01
0.1
1
10
100
1000
TIME (s)
Break Size
50 mm
100 mm
150 mm
200 mm
250 mm
Cavitation Threshold

LOSS OF COOLANT EVENT
MAX BREAK WITHOUT PUMP CAVITATION



Internal Correspondence

MARTIN MARIETTA ENERGY SYSTEMS, INC.

January 29, 1990

B. S. Maxon

Potential Recirculation Within the ANSR Primary System During Natural Convection Cooling Following Shutdown

Following certain hypothetical accident scenarios cooling of the Advanced Neutron Source Reactor (ANSR) may occur by natural convection after shutdown. With reference to Fig. 1, head for loop flow is created by the difference in densities of the hot water exiting the core and flowing upward through the hot leg riser(s) and the cold water exiting the heat exchanger and flowing downward through the cold leg risers. Although not indicated in Fig. 1, the reactor, and cold and hot legs are contained within a reactor pool. Hot leg riser heat losses to the pool reduce the loop head, and may also result in adverse recirculation (i.e., countercurrent flows) which cause an increase in the effective piping hydraulic resistance. Heat losses from the horizontal portions of the hot leg(s) which would have otherwise been removed at the heat exchanger do not affect loop head. Cold leg riser heat losses to the pool result in buoyancy effects which assist flow and therefore should not cause adverse recirculation. However, these losses as well as those through horizontal portions of the cold legs are less effective in providing head for loop flow than if that heat was removed at the heat exchanger. Obviously, heat losses from the hot and cold legs reduce the heat exchanger load.

A scoping study was performed to examine loop heat transfer to the reactor pool and assess potential recirculation. Buoyancy driven primary system flows and temperatures after ANSR shutdown were previously calculated by J. A. March-Leuba[1]. However, since these calculations ignored primary-to-reactor pool heat transfer, supplemental calculations accounting for this effect were necessary and are described here. In the hypothetical accident scenario, flow from the secondary pool (containing the primary system heat exchanger) to the cooling tower is blocked. Approximately 60 s later the reactor scrams and the primary coolant pumps shut down. By ~ 100 s the primary flow is buoyancy driven. When the secondary pool temperature reaches 40°C, heat removal by an air cooler initiates providing increased cooling as the pool temperature increases according to:[1]

$$\text{Heat Removal (MW)} = \frac{1\ \text{MW}}{40^\circ\text{C}}\left(T_{pool} - 40^\circ\text{C}\right)$$

Due to the scoping nature of the recirculation calculations, the temperature rise of the reactor pool and fall of the primary system caused by primary-to-reactor pool heat loss was not modelled. Heat losses from horizontal hot and cold leg portions were not calculated, but should be similar to those from the hot and cold leg risers, respectively, on a per unit surface area basis. An assessment of the impact of heat loss from hot and cold leg risers was made by comparing the losses with the heat addition of the core and removal by the heat exchanger. In addition, "forced flow" (i.e., overall loop flow driven by the density difference between hot and cold leg risers) Reynolds numbers were compared to Rayleigh or Grashof numbers characterizing potential adverse recirculation effects. The latter are governed by geometry (i.e., piping diameter and length) and spatial temperature variations. The comparison of these numbers allowed a determination of the significance of "secondary" recirculation. The flow situation at a number of points in time after shutdown was examined.

Potential recirculation within hot leg risers was examined in three geometries, all of which assumed a 10 m length:

1. Two, 0.36-m-diameter pipes,
2. One, 0.60-m-diameter pipe, and
3. One, 0.36-m-diameter pipe.

It was assumed that the forced flow rates as calculated with March-Leuba model were unaffected by hot leg riser geometry variations since the dominant loop pressure drop, that of the core, remains the same. Heat loss was calculated assuming forced convection heat transfer inside the pipe and natural convection outside the pipe with heat transfer coefficients based on standard correlations. Pipe wall conduction thermal resistance was ignored. The calculated pipe wall and core outlet temperatures were used to calculate Ra_D (D/L) and Gr_D. These values and corresponding Re numbers were then mapped onto a "flow chart" and evaluated in a functional relation to determine recirculation significance. In functional form, the onset of significant (10%) effects of buoyancy is given by[2]:

$$(Gr_D/2/Re^{2.7}) \geq 4.9 \times 10^{-5} \quad (1)$$

This relationship indicates that for given pipe mass flow and thermal conditions the onset of significant buoyancy effects scales with $\sim D^6$.

Tables 1 - 3 provide calculation results for five selected time points following shutdown for the three geometries considered. Figures 2 through 4 map calculated Ra_D (D/L) or Gr_D onto an appropriate flow chart.[2,3,4] The chart used for the 0.6-m-diameter pipe is different from that used for the other two cases since it provides a larger needed range for the natural convection parameter (Gr_D).[4] The flow chart and function indicate the onset of significant effects of buoyancy on <u>heat transfer</u>; however, changes in heat transfer should reflect changes in the flow field (i.e., possible onset of adverse recirculation).

Results of the evaluation of Eqn. 1 are presented in Tables 1 - 3 and indicate that recirculation effects may be significant within ~3 h, 17 m, and 3 d, respectively, for the three cases examined. As expected effects become significant latest for the single, 0.36-m-diameter pipe case. Results of analysis employing flow charts presented in Figs. 2-4 indicate the onset of significant recirculation at much later times than those suggested by Eqn. 1 for all cases. Due to significant hot leg-to-reactor pool heat loss beyond ~3 h in all cases, calculated flows after 3 h may be in error significantly, and in consequence, so are recirculation calculations. The true error depends on the relative magnitude of heat losses which depend on and affect the temperature of the reactor pool and primary system after shutdown. Adverse recirculation is reduced with an increase in the reactor pool temperature and/or a decrease in the average primary temperature. It is recommended that the March-Leuba thermal-hydraulic model be extended to include primary-to-reactor pool heat losses and reactor pool heat capacity in order to properly predict system flow behavior. It is estimated that this extension could be accomplished fairly easily.

Cold leg analysis considered two cases, both of which assumed two, 0.36-m-diameter pipes 10 m long. In one case, the outside surface of the pipe was assumed to be shrouded to enhance heat transfer to the reactor pool. The gap between the shroud and pipe surface was optimized to provide maximum heat transfer. The optimal gap was determined to be 6.1 cm and is approximately a factor of 4 less than the turbulent natural convection boundary layer thickness if no shroud were present. With the shroud, forced convection heat transfer was assumed within it as well as within the pipe. In the second case, no shroud was present and forced flow convection was assumed inside the pipe and natural convection outside. As in the hot leg analysis heat loss was calculated as well as Re, Ra_D (D/L), and Gr_D numbers. These values were then mapped onto a flow chart[2,3] and evaluated in the following function:[2]

$$(Gr_D/2/Re^{2.7}) \geq 1 \times 10^{-5} \qquad (2)$$

This is the same function used for the hot leg analysis except that 4.9×10^{-5} is replaced with 1×10^{-5} to reflect buoyancy assisted flow.

Tables 4 and 5 provide calculation results for the shrouded and "bare" pipe cases, respectively, while Figs. 5 and 6 map calculated Ra_D (D/L) and Re numbers onto the flow chart. The shroud provides an enhancement of heat transfer over the bare pipe by a factor of 1.2 to 1.5. Based on Eqn. 2 results, "local" buoyancy effects will become significant within ~3 h after shutdown for both cases while results of analysis employing the flow chart indicate the onset of significant recirculation at much later times. Cold leg-to-reactor pool heat loss is significant beyond ~3 h for both cases, so calculated results after 3 h may be in error significantly. As was the case for the hot legs, recirculation is reduced with an increase in the reactor pool temperature and/or a decrease in the average primary temperature. Again, the March-Leuba thermal hydraulic model should be extended to account for primary-to-reactor pool heat loss and its effects.

DGM 10-16-91

~~Adverse recirculation within horizontal primary coolant piping is probably not a significant concern provided a turbulent "forced flow" exists.[2]~~ The corresponding flow chart for horizontal pipe flow does not span the range of Re and Ra_D (D/L) of interest here[2,3].

In summary, scoping calculations have been performed to assess potential recirculation within the primary system during natural convection circulation cooling of the ANSR following shutdown. Under simplifying assumptions, recirculation may become significant within ~20 m after shutdown or days later for the range of primary system piping geometries considered here. Results depend dramatically on the evaluation criterion selected. Differences between criteria need to be resolved. Primary-to-reactor pool heat loss effects are significant after ~ 3 h and therefore require representation in the March-Leuba thermal-hydraulic model to properly predict system natural circulation flows. Results presented here differ from previous findings[5], at least in part, due to higher average primary system temperatures after shutdown used here.

D. G. Morris, 9204-1, MS-8045 (6-2092)

DGM:dw

cc: N. C. J. Chen
W. G. Craddick
J. A. March-Leuba
D. L. Selby
G. L. Yoder, Jr.

References

1. March-Leuba, J. A., Oak Ridge National Laboratory, personal communication (ANSR decay heat removal calculations) to Morris, D. G., Oak Ridge National Laboratory, January 5, 1990.

2. Aung, Win, "Mixed Convection in Internal Flow," in *Handbook of Single-phase Convective Heat Transfer*, Kakac, Sadik, Shah, Ramesh K., and Aug, Win, editors, John Wiley and Sons, 1987, pp. 15.1-15.51.

3. Metais, B., and Eckert, E. R. G., "Forced, Mixed, and Free Convection Regimes," in *Journal of Heat Transfer,* May 1964, pp. 295-296.

4. Eckert, E. R. C., and Diaguila, A. J., "Convective Heat Transfer for Mixed, Free, and Forced Flow Through Tubes," in *Transactions of the ASME*, Vol. 76, No. 4, May 1954, pp. 497-504.

5. Morris, D. G., internal correspondence to Maxon, B. S., "Analysis of Recirculation During ANS Decay Heat Removal," November 16, 1989.

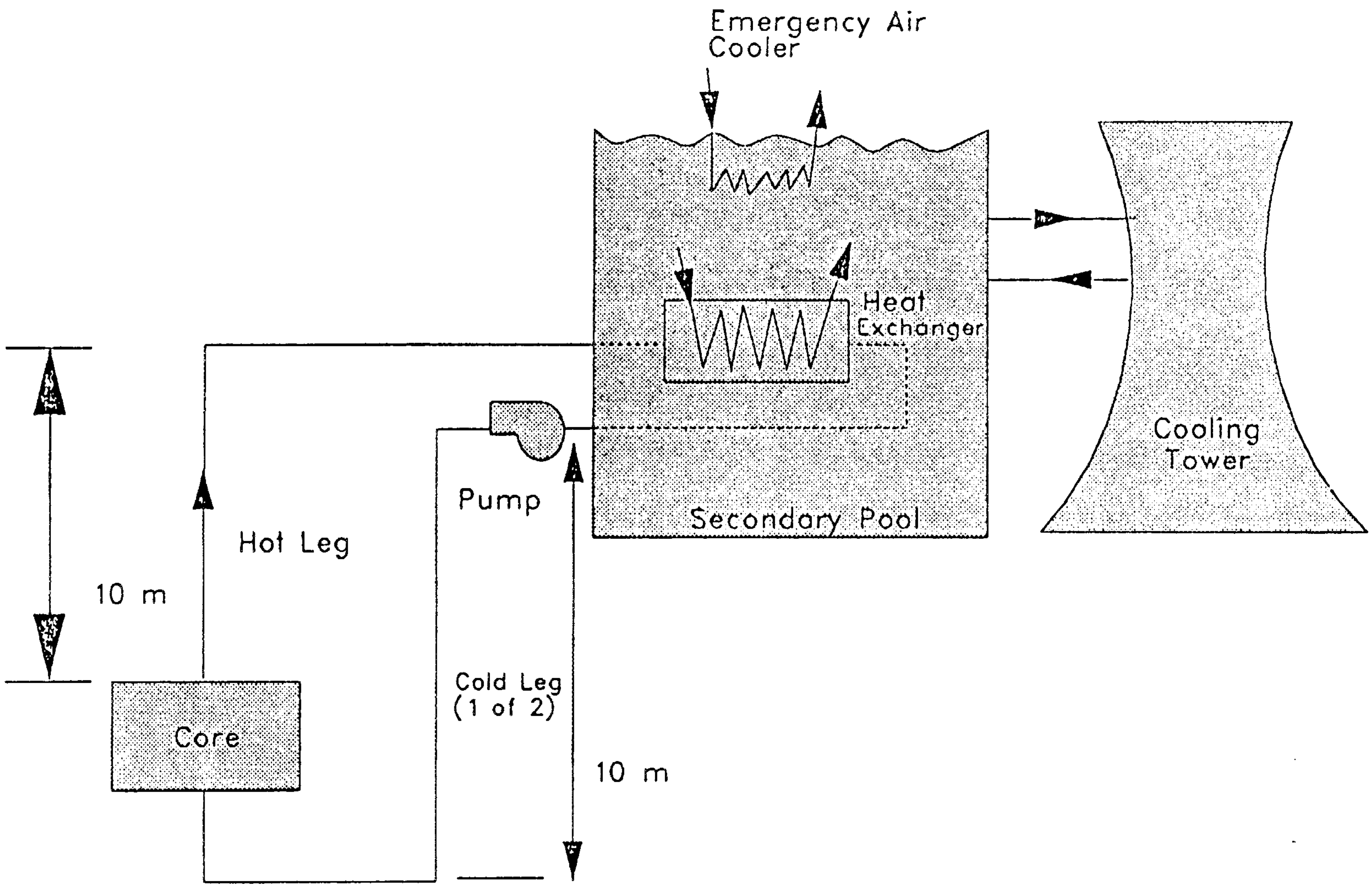


Figure 1. Simplified schematic of ANSR decay heat removal design characteristics.

Table 1. Hot Leg Flow Analysis For Two, 0.36-m-Diameter Pipes 10 m Long

Parameter	Time After Loss of Tower Cooling 100 s	16.7 m	3 h	24 h	3 d
Flow[1] (kg/s)	53.80	24.83	20.47	15.67	10.92
Inlet core T (°C)	53.37	47.16	65.54	85.48	69.37
Outlet core T (°C)	82.51	75.43	81.60	93.52	76.27
Hot leg velocity (m/s)	0.56	0.26	0.21	0.16	0.11
Re	$5.5x10^5$	$2.6x10^5$	$2.1x10^5$	$1.8x10^5$	$1.1x10^5$
Core ΔT (°C)	29.14	28.27	16.06	8.04	6.9
Core decay heat (W)	$9.5x10^6$	$4.1x10^6$	$1.9x10^6$	$7.4x10^5$	$4.4x10^5$
Ra_L	$3.33x10^{15}$	$2.86x10^{15}$	$3.27x10^{15}$	$4.06x10^{15}$	$2.92x10^{15}$
h_s[2] (W/m² °C)	971	923	965	1037	929
R_o[2] (°C/W)	$9.22x10^{-5}$	$9.70x10^{-5}$	$9.28x10^{-5}$	$8.63x10^{-5}$	$9.64x10^{-5}$
h_i[3] (W/m² °C)	1744	939	812	655	482
R_i[3] (°C/W)	$5.13x10^{-5}$	$9.53x10^{-5}$	$1.10x10^{-4}$	$1.37x10^{-4}$	$1.86x10^{-4}$
Heat loss[1,4] (W)	$3.52x10^5$	$2.26x10^5$	$2.45x10^5$	$2.76x10^5$	$1.57x10^5$
ΔT_h[5] (°C)	1.56	2.17	2.85	4.20	3.43
Pipe wall T (°C)	64.5	53.9	54.7	55.9	47.2
$Ra_D(D/L)$	$2.47x10^9$	$2.96x10^9$	$3.69x10^9$	$5.16x10^9$	$3.67x10^9$
Gr_D	$2.62x10^{10}$	$3.14x10^{10}$	$3.91x10^{10}$	$5.47x10^{10}$	$3.67x10^{10}$
$Gr_D/2/Re^{2.7}$	$4.2x10^{-6}$	$3.8x10^{-5}$	$8.3x10^{-5}$	$1.8x10^{-4}$	$4.5x10^{-4}$

[1] one of two hot legs
[2] outer pipe surface heat transfer coefficient and thermal resistance
[3] inner pipe surface heat transfer coefficient and thermal resistance
[4] hot leg-to-reactor pool
[5] temperature change caused by hot leg-to-reactor pool heat loss

Table 2. Hot Leg Flow Analysis For 0.60-m-Diameter Pipe 10 m Long

Parameter	Time After Loss of Tower Cooling				
	100 s	16.7 m	3 h	24 h	3 d
Flow (kg/s)	107.6	49.66	40.94	31.33	21.83
Inlet core T (°C)	53.37	47.16	65.54	85.48	69.37
Outlet core T (°C)	82.51	75.43	81.60	93.52	76.27
Hot leg velocity (m/s)	0.39	0.18	0.15	0.11	0.079
Re	$6.5x10^{5}$	$3.0x10^{5}$	$2.5x10^{5}$	$2.0x10^{5}$	$1.3x10^{5}$
Core ΔT (°C)	29.14	28.27	16.06	8.04	6.9
Core decay heat (W)	$9.5x10^{6}$	$4.1x10^{6}$	$1.9x10^{6}$	$7.4x10^{5}$	$4.4x10^{5}$
Ra_L	$3.33x10^{5}$	$2.86x10^{15}$	$3.27x10^{15}$	$4.06x10^{15}$	$2.92x10^{15}$
h_s[1] (W/m² °C)	971	923	965	1037	929
R_o[1] (°C/W)	$5.47x10^{-5}$	$5.75x10^{-5}$	$5.50x10^{-5}$	$5.12x10^{-5}$	$5.71x10^{-5}$
h_i[2] (W/m² °C)	1575	848	733	591	435
R_i[2] (°C/W)	$3.37x10^{-5}$	$6.26x10^{-5}$	$7.24x10^{-5}$	$8.98x10^{-5}$	$1.22x10^{-4}$
Heat loss[3] (W)	$5.71x10^{5}$	$3.62x10^{5}$	$3.89x10^{5}$	$4.26x10^{5}$	$2.47x10^{5}$
ΔT_b[4] (°C)	1.26	1.74	2.27	3.31	2.70
Pipe wall T(°C)	63.3	52.8	53.4	55.3	46.2
Ra_D(D/L)	$2.03x10^{10}$	$1.93x10^{10}$	$2.98x10^{10}$	$4.05x10^{10}$	$2.57x10^{10}$
Gr_D	$1.29x10^{11}$	$1.23x10^{11}$	$1.90x10^{11}$	$2.58x10^{11}$	$1.42x10^{11}$
$Gr_D/2/Re^{2.7}$	$1.3x10^{-5}$	$1.0x10^{-4}$	$2.5x10^{-4}$	$6.3x10^{-4}$	$1.0x10^{-3}$

[1] outer pipe surface heat transfer coefficient and thermal resistance
[2] inner pipe surface heat transfer coefficient and thermal resistance
[3] hot leg-to-reactor pool
[4] temperature change caused by hot leg-to-reactor pool heat loss

Table 3. Hot Leg Flow Analysis For 0.36-m-Diameter Pipe 10 m Long

Parameter	Time After Loss of Tower Cooling 100 s	16.7 m	3 h	24 h	3 d
Flow (kg/s)	107.6	49.66	40.94	31.33	21.83
Inlet core T (°C)	53.37	47.16	65.54	85.48	69.37
Outlet core T (°C)	82.51	75.43	81.60	93.52	76.27
Hot leg velocity (m/s)	1.12	0.52	0.42	0.32	0.22
Re	$1.1x10^{6}$	$5.2x10^{5}$	$4.2x10^{5}$	$3.6x10^{5}$	$2.2x10^{5}$
Core ΔT (°C)	29.14	28.27	16.06	8.04	6.9
Core decay heat (W)	$9.5x10^{6}$	$4.1x10^{6}$	$1.9x10^{6}$	$7.4x10^{5}$	$4.4x10^{5}$
Ra_L	$3.33x10^{15}$	$2.86x10^{15}$	$3.27x10^{15}$	$4.06x10^{15}$	$2.92x10^{15}$
h_a[1] (W/m² °C)	971	923	965	1037	929
R_o[1] (°C/W)	$9.22x10^{-5}$	$9.70x10^{-5}$	$9.28x10^{-5}$	$8.63x10^{-5}$	$9.64x10^{-5}$
h_i[2] (W/m² °C)	2006	1080	934	753	554
R_i[2] (°C/W)	$4.45x10^{-5}$	$8.27x10^{-5}$	$9.56x10^{-5}$	$1.19x10^{-4}$	$1.61x10^{-4}$
Heat Loss[1,4] (W)	$3.69x10^{5}$	$2.42x10^{5}$	$2.63x10^{5}$	$3.00x10^{5}$	$1.72x10^{5}$
ΔT_b[4] (°C)	0.82	1.16	1.53	2.28	1.88
Pipe wall T(°C)	66.1	55.4	56.5	57.9	48.5
$Ra_D(D/L)$	$2.25x10^{9}$	$2.75x10^{9}$	$3.45x10^{9}$	$6.01x10^{9}$	$3.81x10^{9}$
Gr_D	$2.39x10^{10}$	$2.92x10^{10}$	$3.66x10^{10}$	$6.37x10^{10}$	$3.50x10^{10}$
$Gr_D/2/Re^{27}$	$5.8x10^{-7}$	$5.4x10^{-6}$	$1.2x10^{-5}$	$3.2x10^{-5}$	$6.6x10^{-5}$

[1] outer pipe surface heat transfer coefficient and thermal resistance
[2] inner pipe surface heat transfer coefficient and thermal resistance
[3] hot leg-to-reactor pool
[4] temperature change caused by hot leg-to-reactor pool heat loss

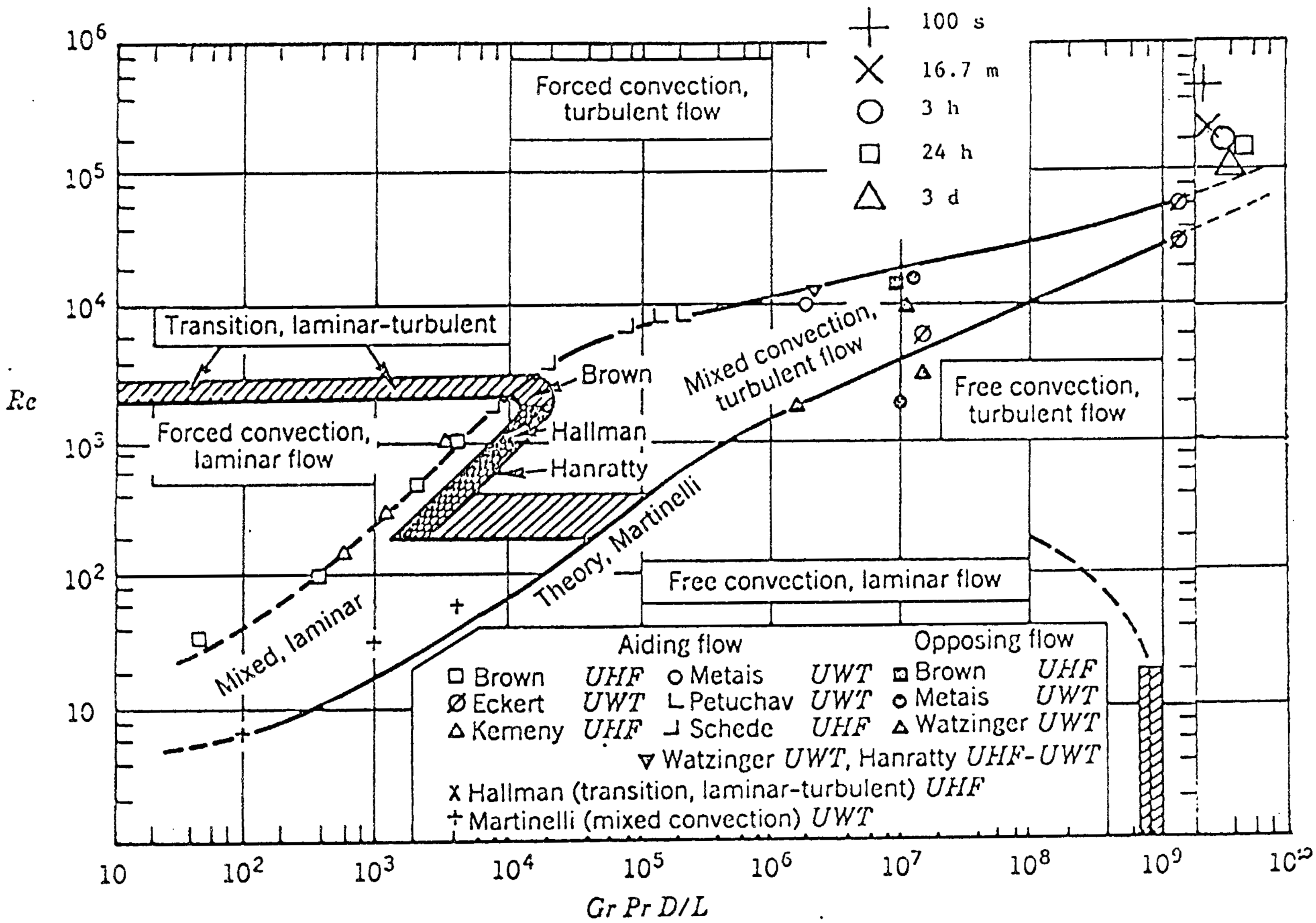


Figure 2. Flow chart results for two, 0.36-m-diameter hot leg pipes.[2]

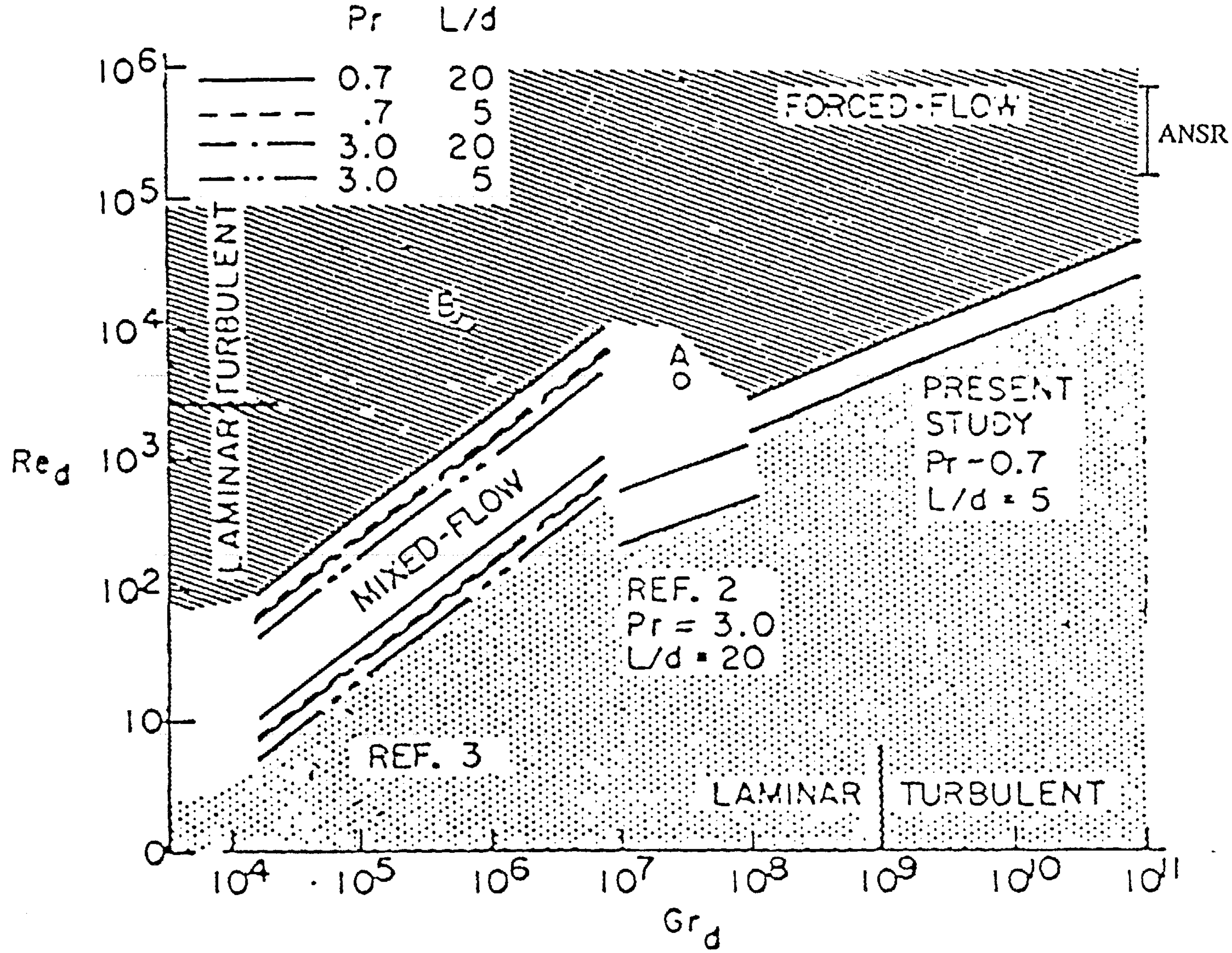


Figure 3. Flow chart results for 0.60-m-diameter hot leg pipe.[4]

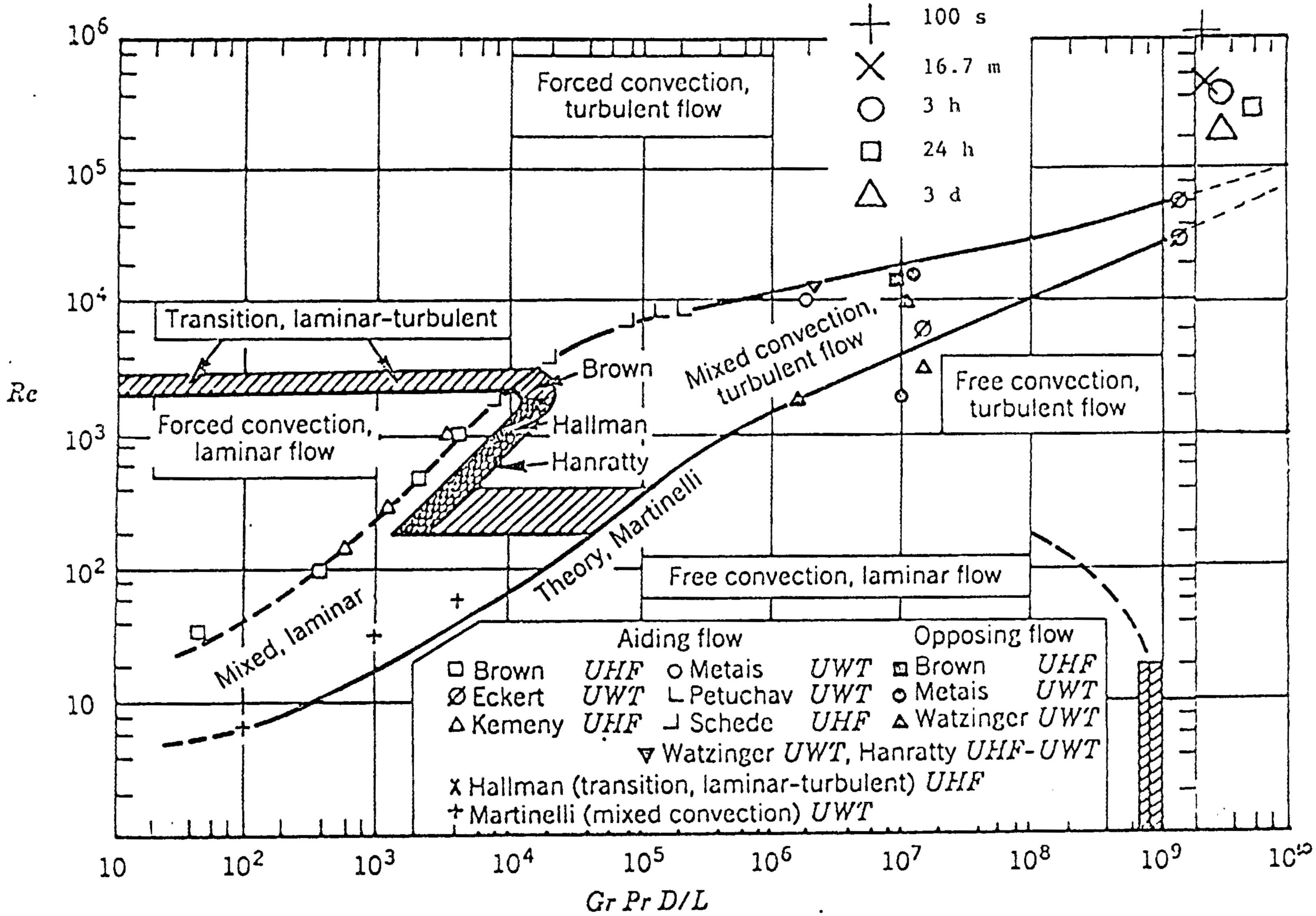


Figure 4. Flow chart results for 0.36-m-diameter hot leg pipe.[2]

Table 4. Cold Leg Flow Analysis For Two, Shrouded 0.36-m-Diameter Pipes 10m Long

Parameter	Time After Loss of Tower Cooling 100 s	16.7 m	3 h	24 h	3 d
Flow[1] (kg/s)	53.80	24.83	20.47	15.67	10.92
T_{be} outlet[2] (°C)	47.59	48.09	65.99	85.45	69.31
Heat exchanger load (W)	$1.5x10^7$	$5.4x10^6$	$2.5x10^6$	$1.1x10^6$	$6.4x10^5$
Cold leg velocity (m/s)	0.56	0.26	0.21	0.16	0.11
Re	$4.2x10^5$	$2.0x10^5$	$1.6x10^5$	$1.2x10^5$	$8.3x10^4$
h_i[3] (W/m^2 °C)	2040	1127	943	749	558
Heat Loss[1,4] (W)	$1.2x10^5$	$9.6x10^4$	$2.1x10^5$	$2.9x10^5$	$1.6x10^5$
ΔT_c[5](°C)	0.53	0.92	2.4	4.4	3.5
Pipe wall T(°C)	42.3	40.5	46.1	50.9	43.7
Ra_D(D/L)	$3.3x10^8$	$4.8x10^8$	$2.2x10^9$	$4.7x10^9$	$2.8x10^9$
Gr_D	$2.1x10^9$	$3.1x10^9$	$2.0x10^{10}$	$5.0x10^{10}$	$2.6x10^{10}$
$Gr_D/2/Re^{2.7}$	$6.9x10^{-7}$	$7.5x10^{-6}$	$8.9x10^{-5}$	$4.8x10^{-4}$	$6.8x10^{-4}$

[1] one of two cold legs
[2] heat exchanger outlet temperature
[3] inner pipe surface heat transfer coefficient
[4] cold leg-to-reactor pool
[5] temperature change caused by cold leg-to-reactor pool heat loss

Table 5. Cold Leg Flow Analysis For Two, Bare 0.36-m-Diameter Pipes 10m Long

Parameter	Time After Loss of Tower Cooling 100 s	16.7 m	3 h	24 h	3 d
Flow[1] (kg/s)	53.80	24.83	20.47	15.67	10.92
T_{he} outlet[2] (°C)	47.59	48.09	65.99	85.45	69.31
Heat exchanger load (W)	$1.5x10^7$	$5.4x10^6$	$2.5x10^6$	$1.1x10^6$	$6.4x10^5$
Cold leg velocity (m/s)	0.56	0.26	0.21	0.16	0.11
Re	$4.2x10^5$	$2.0x10^5$	$1.6x10^5$	$1.2x10^5$	$8.3x10^4$
Ra_L	$7.72x10^{14}$	$7.97x10^{14}$	$1.68x10^{15}$	$2.65x10^{15}$	$1.85x10^{15}$
h_a[3](W/m² °C)	586	592	767	893	792
R_o[3] (°C/W)	$1.52x10^{-4}$	$1.51x10^{-4}$	$1.16x10^{-4}$	$1.00x10^{-4}$	$1.13x10^{-4}$
h_i[4] (W/m² °C)	2040	1127	943	749	558
R_i[4] (°C/W)	$4.38x10^{-5}$	$7.92x10^{-5}$	$9.47x10^{-5}$	$1.19x10^{-4}$	$1.60x10^{-4}$
Heat loss[1,5] (W)	$7.96x10^4$	$6.99x10^4$	$1.61x10^5$	$2.44x10^5$	$1.37x10^5$
ΔT_c[6](°C)	0.35	0.67	1.87	3.71	2.99
Pipe wall T(°C)	44.1	42.6	50.7	56.4	47.4
$Ra_D(D/L)$	$2.9x10^8$	$4.6x10^8$	$1.7x10^9$	$4.0x10^9$	$2.4x10^9$
Gr_D	$2.2x10^9$	$3.5x10^9$	$1.6x10^{10}$	$4.2x10^{10}$	$2.2x10^{10}$
$Gr_D/2/Re^2$[7]	$7.2x10^{-7}$	$8.5x10^{-6}$	$7.1x10^{-5}$	$4.1x10^{-4}$	$5.8x10^{-4}$

[1] one of two cold legs
[2] heat exchanger outlet temperature
[3] outer pipe surface heat transfer coefficient and thermal resistance
[4] inner pipe surface heat transfer coefficient and thermal resistance
[5] cold leg-to-reactor pool
[6] temperature change caused by cold leg-to-reactor pool heat loss

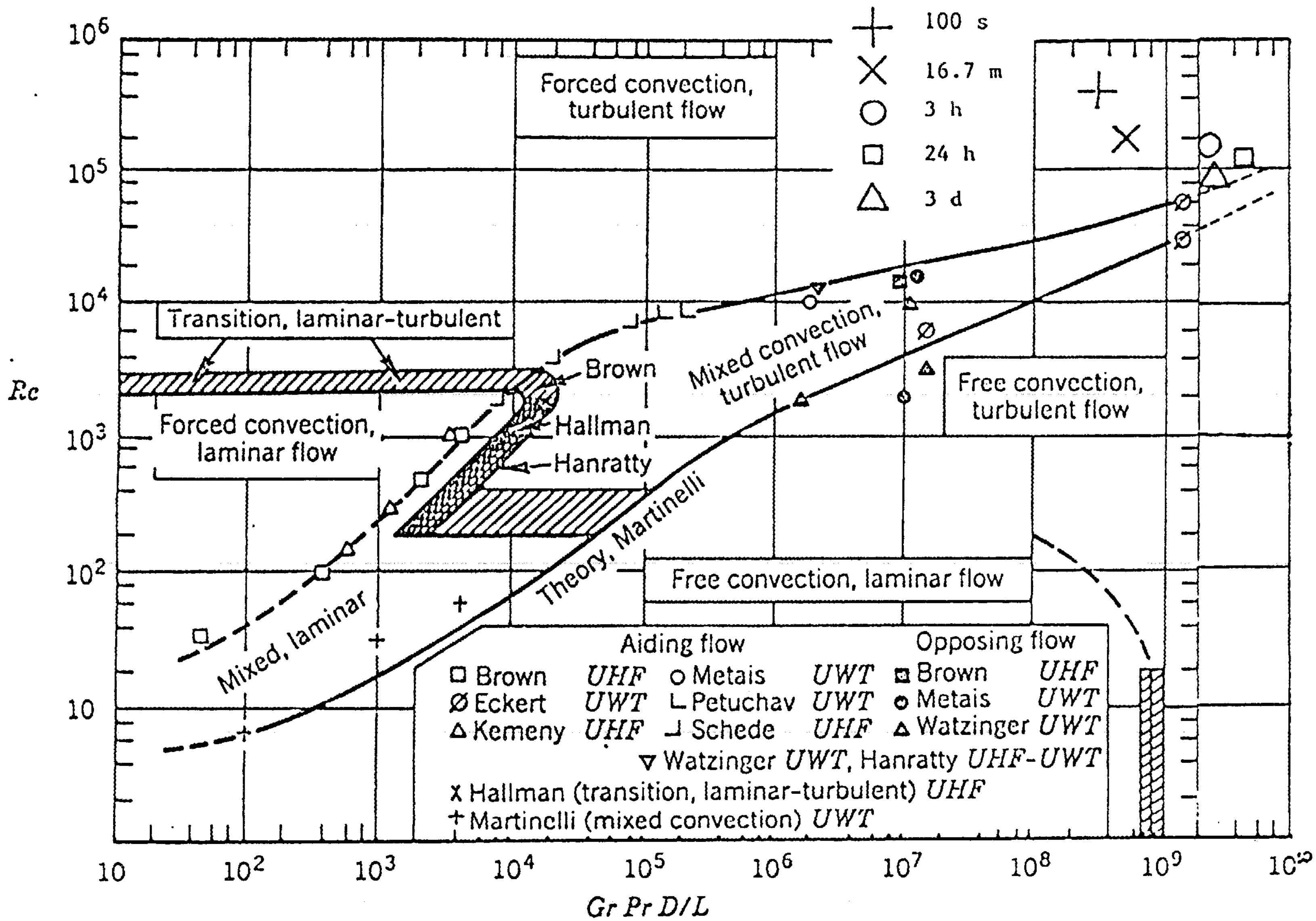


Figure 5. Flow chart results for two, shrouded 0.36-m-diameter cold leg pipes.[2]

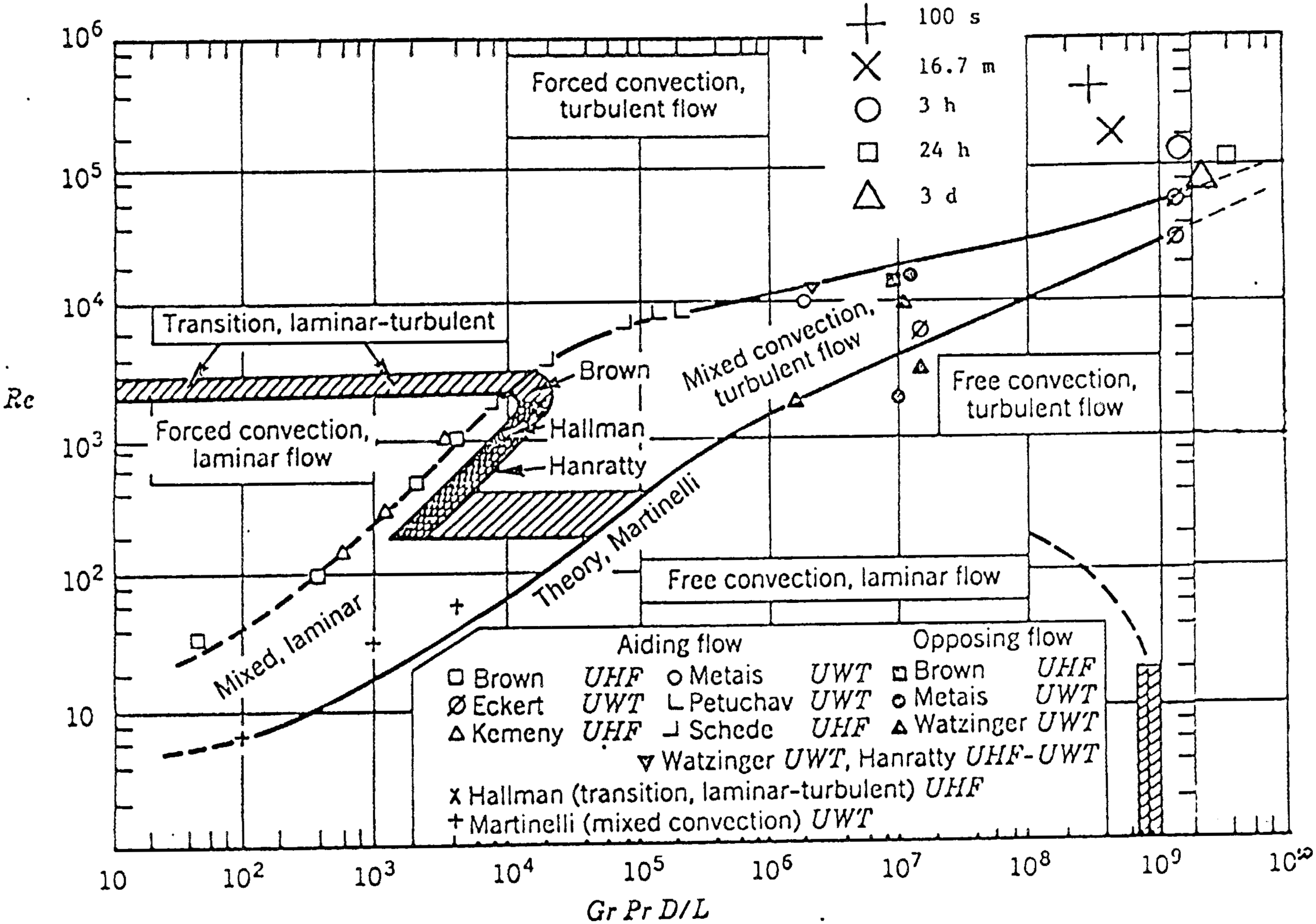


Figure 6. Flow chart results for two, bare 0.36-m-diameter cold leg pipes.[2]

Appendix J: AHP INPUT AND OUTPUT FOR DECAY HEAT REMOVAL

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@@@@@@@@@@@@@@@@@@@@@@@@@@@@@@@@@@@@@@@@@@@@@@@@@@@@@@@@@@@@@@@@@@@@@@
@@@@@@@@@@@@@@@@@@@@@@@@@@@@@@@@@@@@@@@@@@@@@@@@@@@@@@@@@@@@@@@@@@@@@@
@@@@      Analytical Hierarchy Process - output file      @@@@
@@@@                 Program version 5e                   @@@@
@@@@                                                      @@@@
@@@@              ANSR Decay Heat Removal                 @@@@
@@@@                                                      @@@@
@@@@      date: 12-03-1991         time: 10:00:58 pm      @@@@
@@@@@@@@@@@@@@@@@@@@@@@@@@@@@@@@@@@@@@@@@@@@@@@@@@@@@@@@@@@@@@@@@@@@@@
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```

HIERARCHY RELATIONSHIPS AND FACTOR RANKS
(a negative rank indicates a reciprocal; e.g., -3 implies 1/3)

Decay Heat/Components data arrays

Decay Heat		1.	2.	3.	4.	5.	6.	7.	8.	9.	10.	1	12.	13.	14.	15.	16.
Fuel Assy	1.	1	5	5	5	5	3	5	2	5	5		3	5	4	4	5
Control Rods	2.		1	2	2	-3	-3	1	-4	3	3	-	-3	3	-3	-2	3
Targets	3.			1	-2	-4	-5	-2	-5	1	2	-	-4	1	-4	-2	1
Bypass	4.				1	-3	-5	1	-5	2	2		-3	1	-3	1	1
Cold Leg	5.					1	-2	3	-3	2	3	[illegible]	-2	3	-2	2	3
Pumps	6.						1	3	-2	3	4	[illegible]	2	4	-2	3	5
Emerg. HX	7.							1	-5	2	1	-[illegible]	1	3	-2	3	3
Primary HX	8.								1	5	5	[illegible]	3	5	1	5	5
Reflector Tank	9.									1	3	-[illegible]	-2	2	-4	1	-2
Break	10.										1	-[illegible]	-3	1	-5	-3	-4
Pool	11.											[illegible]	1	5	-2	3	3
Hot Leg	12.												1	4	-2	3	3
Pressurizer Sys	13.													1	-5	-2	-3
Cooling Twr Bsn	14.														1	5	5
Hot&Cold Leg Se	15.															1	-3
CPBT	16.																1

Components/Phenomena data arrays

Fuel Assy		1.	2.	3.	4.	5.
power vs time	1.	1	2	3	4	5
flow resist	2.		1	5	5	5
power vs posit.	3.			1	2	4
Parallel chan.	4.				1	1
inlet vel & tem	5.					1

Control Rods		1.	2.
Heat load CR	1.	1	-3
Flow Resist CR	2.		1

Targets		1.	2.
Heat Load T	1.	1	5
Flow Resist T	2.		1

Bypass		1.
Flow Resist BP	1.	1

Cold Leg		1.	2.	3.
Flow Resist CL	1.	1	-3	1
Heat Trans CL	2.		1	3
Strat. CL	3.			1

Pumps		1.
Head vs G&t P	1.	1

Emerg. HX		1.	2.
Heat Trans EHX	1.	1	3
Flow Resist EHX	2.		1

Primary HX

		1.	2.	3.
Flow Resist PHX	1.	1	-2	-3
HT to Pool PHX	2.		1	-5
HT to Sec PHX	3.			1

Reflector Tank

		1.	2.
Heat vs time R	1.	1	-2
Init. Temp. R	2.		1

Break

		1.
Position BR	1.	1

Pool

		1.	2.
Strat. POOL	1.	1	-2
Init. Temp POOL	2.		1

Hot Leg

		1.	2.	3.
Flow Resist HL	1.	1	-3	-3
Heat Trans HL	2.		1	3
Strat. HL	3.			1

Pressurizer Sys

		1.
Mass vs t PS	1.	1

Cooling Twr Bsn

		1.	2.	3.	4.
Inventory CTB	1.	1	-2	-2	-3
Init Temp CTB	2.		1	1	1
Strat. CTB	3.			1	1
Heat Trans CTB	4.				1

Hot&Cold Leg Se		1.	2.
Flow Res H&CLS	1.	1	3
Heat Tran H&CLS	2.		1

CPBT		1.	2.
Heat Load CPBT	1.	1	3
Heat Tran CPBT	2.		1

COMPONENTS FACTORS RELATIVE TO DECAY HEAT

Factors relative to Decay Heat:

	weight
Fuel Assy	1.0000
Control Rods	0.2164
Targets	0.1147
Bypass	0.1640
Cold Leg	0.3562
Pumps	0.5310
Emerg. HX	0.2290
Primary HX	0.7357
Reflector Tank	0.1381
Break	0.0969
Pool	0.3242
Hot Leg	0.3650
Pressurizer Sys	0.0981
Cooling Twr Bsn	0.5754
Hot&Cold Leg Se	0.1641
CPBT	0.1710

lambda (maximum) = 17.3361
consistency index = 0.0891
consistency ratio = 0.0594

Composite priorities:

	weight	priority
Fuel Assy	0.1894	(9)
Primary HX	0.1393	(7)
Cooling Twr Bsn	0.1090	(5)
Pumps	0.1006	(5)
Hot Leg	0.0691	(3)
Cold Leg	0.0675	(3)
Pool	0.0614	(3)
Emerg. HX	0.0434	(2)
Control Rods	0.0410	(2)
CPBT	0.0324	(2)
Hot&Cold Leg Se	0.0311	(2)
Bypass	0.0311	(2)
Reflector Tank	0.0262	(1)
Targets	0.0217	(1)
Pressurizer Sys	0.0186	(1)
Break	0.0183	(1)

PHENOMENA FACTORS RELATIVE TO COMPONENTS

Factors relative to Fuel Assy:

	weight
power vs time	1.0000
flow resist	0.8981
power vs posit.	0.3424
Parallel chan.	0.1756
inlet vel & tem	0.1499

lambda (maximum) = 5.2751
consistency index = 0.0688
consistency ratio = 0.0614

Factors relative to Control Rods:

	weight
Heat load CR	0.3333
Flow Resist CR	1.0000

lambda (maximum) = 2.0000
consistency index = 0.0000
consistency ratio = 0.0000

Factors relative to Targets:

	weight
Heat Load T	1.0000
Flow Resist T	0.2000

lambda (maximum) = 2.0000
consistency index = 0.0000
consistency ratio = 0.0000

Factors relative to Bypass:

	weight
Flow Resist BP	1.0000

lambda (maximum) = 1.0000
consistency index = 0.0000
consistency ratio = 0.0000

Factors relative to Cold Leg:

	weight
Flow Resist CL	0.3333
Heat Trans CL	1.0000
Strat. CL	0.3333

lambda (maximum) = 3.0000
consistency index = 0.0000
consistency ratio = 0.0000

Factors relative to Pumps:

	weight
Head vs G&t P	1.0000

lambda (maximum) = 1.0000
consistency index = 0.0000
consistency ratio = 0.0000

Factors relative to Emerg. HX:

	weight
Heat Trans EHX	1.0000
Flow Resist EHX	0.3333

lambda (maximum) = 2.0000
consistency index = 0.0000
consistency ratio = 0.0000

Factors relative to Primary HX:

	weight
Flow Resist PHX	0.2231
HT to Pool PHX	0.2988
HT to Sec PHX	1.0000

lambda (maximum) = 3.1632
consistency index = 0.0816
consistency ratio = 0.1407 (See footnote below)

Factors relative to Reflector Tank:

	weight
Heat vs time R	0.5000
Init. Temp. R	1.0000

lambda (maximum) = 2.0000
consistency index = 0.0000
consistency ratio = 0.0000

Factors relative to Break:

	weight
Position BR	1.0000

lambda (maximum) = 1.0000
consistency index = 0.0000
consistency ratio = 0.0000

Factors relative to Pool:

	weight
Strat. POOL	0.5000
Init. Temp POOL	1.0000

lambda (maximum) = 2.0000
consistency index = 0.0000
consistency ratio = 0.0000

Factors relative to Hot Leg:

	weight
Flow Resist HL	0.2311
Heat Trans HL	1.0000
Strat. HL	0.4808

lambda (maximum) = 3.1356
consistency index = 0.0678
consistency ratio = 0.1169 (See footnote below)

Factors relative to Pressurizer Sys:

	weight
Mass vs t PS	1.0000

lambda (maximum) = 1.0000
consistency index = 0.0000
consistency ratio = 0.0000

Factors relative to Cooling Twr Bsn:

	weight
Inventory CTB	0.4078
Init Temp CTB	0.8986
Strat. CTB	0.8986
Heat Trans CTB	1.0000

lambda (maximum) = 4.0206
consistency index = 0.0069
consistency ratio = 0.0076

Factors relative to Hot&Cold Leg Se:

	weight
Flow Res H&CLS	1.0000
Heat Tran H&CLS	0.3333

lambda (maximum) = 2.0000
consistency index = 0.0000
consistency ratio = 0.0000

Factors relative to CPBT:

	weight
Heat Load CPBT	1.0000
Heat Tran CPBT	0.3333

lambda (maximum) = 2.0000
consistency index = 0.0000
consistency ratio = 0.0000

Composite priorities:

	weight	priority
power vs time	0.1052	(9)
flow resist	0.0944	(8)
HT to Sec PHX	0.0774	(7)
Heat Trans CTB	0.0605	(6)
Head vs G&t P	0.0558	(5)
Init Temp CTB	0.0544	(5)
Strat. CTB	0.0544	(5)
Heat Trans HL	0.0384	(4)
Heat Trans CL	0.0375	(4)
power vs posit.	0.0360	(4)
Init. Temp POOL	0.0341	(3)
Inventory CTB	0.0247	(3)
Heat Trans EHX	0.0241	(3)
HT to Pool PHX	0.0231	(3)
Flow Resist CR	0.0228	(3)
Parallel chan.	0.0185	(2)
Strat. HL	0.0184	(2)
Heat Load CPBT	0.0180	(2)
Flow Resist PHX	0.0173	(2)
Flow Res H&CLS	0.0173	(2)
Flow Resist BP	0.0172	(2)
Strat. POOL	0.0170	(2)
inlet vel & tem	0.0158	(2)
Init. Temp. R	0.0145	(2)
Flow Resist CL	0.0125	(2)
Strat. CL	0.0125	(2)
Heat Load T	0.0121	(2)
Mass vs t PS	0.0103	(2)
Position BR	0.0102	(2)
Flow Resist HL	0.0089	(2)
Flow Resist EHX	0.0080	(1)
Heat load CR	0.0076	(1)
Heat vs time R	0.0073	(1)
Heat Tran CPBT	0.0060	(1)
Heat Tran H&CLS	0.0058	(1)
Flow Resist T	0.0024	(1)

Factors relative to Fuel Assy:

	weight	priority
power vs time	0.1052	(9)
flow resist	0.0944	(8)
power vs posit.	0.0360	(4)
Parallel chan.	0.0185	(2)
inlet vel & tem	0.0158	(2)

Factors relative to Control Rods:

	weight	priority
Heat load CR	0.0076	(1)
Flow Resist CR	0.0228	(3)

Factors relative to Targets:

	weight	priority
Heat Load T	0.0121	(2)
Flow Resist T	0.0024	(1)

Factors relative to Bypass:

	weight	priority
Flow Resist BP	0.0172	(2)

Factors relative to Cold Leg:

	weight	priority
Flow Resist CL	0.0125	(2)
Heat Trans CL	0.0375	(4)
Strat. CL	0.0125	(2)

Factors relative to Pumps:

	weight	priority
Head vs G&t P	0.0558	(5)

Factors relative to Emerg. HX:

	weight	priority
Heat Trans EHX	0.0241	(3)
Flow Resist EHX	0.0080	(1)

Factors relative to Primary HX:

	weight	priority
Flow Resist PHX	0.0173	(2)
HT to Pool PHX	0.0231	(3)
HT to Sec PHX	0.0774	(7)

Factors relative to Reflector Tank:

	weight	priority
Heat vs time R	0.0073	(1)
Init. Temp. R	0.0145	(2)

Factors relative to Break:

	weight	priority
Position BR	0.0102	(2)

Factors relative to Pool:

	weight	priority
Strat. POOL	0.0170	(2)
Init. Temp POOL	0.0341	(3)

Factors relative to Hot Leg:

	weight	priority
Flow Resist HL	0.0089	(2)
Heat Trans HL	0.0384	(4)
Strat. HL	0.0184	(2)

Factors relative to Pressurizer Sys:

	weight	priority
Mass vs t PS	0.0103	(2)

Factors relative to Cooling Twr Bsn:

	weight	priority
Inventory CTB	0.0247	(3)
Init Temp CTB	0.0544	(5)
Strat. CTB	0.0544	(5)
Heat Trans CTB	0.0605	(6)

Factors relative to Hot&Cold Leg Se:

	weight	priority
Flow Res H&CLS	0.0173	(2)
Heat Tran H&CLS	0.0058	(1)

Factors relative to CPBT:

	weight	priority
Heat Load CPBT	0.0180	(2)
Heat Tran CPBT	0.0060	(1)

CONSISTENCY OF THE HIERARCHY = 0.0604

Footnote: The consistency limit has exceeded 10%.
A review of the input assumptions may be necessary.

***** Above results produced using the Dimenna normalization

Appendix K: COMMENTS AND DISCUSSION FROM THE ANSR PIR TEAM DURING THE RANKING PROCESS FOR DECAY HEAT REMOVAL

Appendix K: COMMENTS AND DISCUSSION FROM THE ANSR PIR TEAM DURING THE RANKING PROCESS FOR DECAY HEAT REMOVAL

Comments from PIR Team members during the consideration of pairwise rankings in the development of the PIR for decay heat removal follow:

Fuel Assembly
vs. Cold Leg
vs. Emergency Heat Exchanger
vs. Primary Heat Exchanger
Comment: Several heat sinks and basicly one heat source, Fuel assembly ranked more important than any one heat sink. Team expects the heat sinks to be grossly oversized.
vs. Reflector
Comment: Not much information regarding reflector tank design available. Pete Griffith indicated that the intentional design of a leaky seal at the bottom of the CPBT into the reflector tank would be difficult to design. He suggested Idel'Chik, 1960, as a source of design information and warned to expect wear/erosion in service.
vs. Cooling tower basin
Comment: Basin temperature will be around 30 degrees Celsius in normal operation. Pete Griffith suggested the design must not allow the inlet or exit pipes to uncover as the pool evaporates. May want to draw return flow from the bottom to insure cool water return if the pool stratifies.

Cold leg vs. Hot leg
Comment: Pool temperature will be around 35 degrees Celsius.
Emergency Heat exchanger should dominate heat transfer to the pool.
Pipe bowing may occur in the hot leg if there is stratification.

FUEL ASSEMBLY

Power vs. Time
vs. flow resistance
Comment: Temperature rise goes as the power, but is proportional to the flow resistance taken to some exponent near 0.5.
vs. Position
Comment: Peak temperature is at the top of the fuel. Local wall superheats will be small.
vs. Parallel channel behavior
Comment: Flow is always up. Even an unheated channel will flow up since the majority of the driving head is due to the elevation in the hot leg.
vs. Inlet velocity and temperature
Comment: Inlet should still be turbulent and well mixed.

CONTROL RODS

Heat load vs. flow resistance
Comment: Driving head and single phase flow resistance are known.
Heat load is more important to determining onset of vapor generation here.

COLD LEG PIPES

Flow resistance vs. heat transfer to the pool
Comment: Big flow resistance is the fuel assembly, not the pipes.

Appendix L: ANSR TRANSIENTS AND APPLICABILITY OF THE ANSR PIR RESULTS

Appendix L: ANSR TRANSIENTS AND APPLICABILITY OF THE ANSR PIR RESULTS

Four basic thermalhydraulic analysis tools are being used to evaluate the reactor design. The list does not include tools used to evaluate severe accidents.

A. A modular Advanced Continuous Simulation Language (ACSL) based simulation code is used for single phase one-dimensional thermal-hydraulic evaluations. This code is flexible and fast running and has been used extensively to evaluate the conceptual design of the reactor.

B. A single phase one-dimensional thermalhydraulic model of a fuel cooling channel is used to examine details of the fuel performance. Two-dimensional power profiles are modeled and oxide growth is simulated for the entire fuel cycle. Fuel surface temperatures, fuel centerline temperatures and thermal limits are calculated by this model.

C. Small FORTRAN programs and engineering notebook calculations performed to bound performance limits and check more complicated models for consistency and credibility.

D. RELAP5/MOD3, a one-dimensional two-phase thermalhydraulic code developed for power reactor transient simulations. A version of this code has been modified to allow more accurate simulation of ANSR behavior during transients.

An introduction to ANSR design basis conditions and events follows. The column labeled "T-H Analysis Tools" in Table L.1 denotes which of the above tools can be used to evaluate each transient. Table L1 lists the design basis events that are applicable to the ANS. The events are grouped by initiating event, cause or consequence, and are classified as Normal, Anticipated, Unlikely, or Extremely Unlikely events in accordance with the ANS 51.1-1983 classification scheme. The grouping of events into frequency categories is based on regulatory requirements as well as on available data from research or power reactors. Data sources expressed in the ANS event category grouping include PRA studies conducted for the ANS, the HFIR Level I PRA, and applicable power reactor experience. In Table L1, this philosophy has been applied to any event involving a well-defined single failure such as the unintended closure of any one valve, or the stoppage or failure of any one pump. Table L1 is a compact, comprehensive listing of all the design basis events that affect more than one plant system.

A fifth event category, Test Conditions, is included to document the special plant level test conditions specified for the reactor at predetermined but infrequent intervals in accordance with ASME code rules or other requirements. The plant must be designed to accommodate these test conditions.

Applicability of ANSR PIR team results given in last column of Table 8-1 according to the following three categories:

AP - Applicable
NA - Not Applicable
PA - Partially Applicable

Table L1: ANSR Transients

Event	Frequency Category[1]	T-H Analysis Tools	ANSR PIR Team Results
NORMAL REACTOR OPERATIONS	Normal		
Fuel loading		A,C,D	NA
Approach to criticality		A	NA
Startup to low power		A	NA
Startup to full power		A	NA
Controlled shutdown to low power		A,D	NA
Fast run back		A,D	NA
Scram		A,D	NA
Fuel Unloading		A,C,D	PA
TEST CONDITIONS	Test		
Primary coolant system hydrostatic pressure test		C	NA
Secondary coolant system hydrostatic pressure test		C	NA
Reflector coolant system hydrostatic pressure test		C	NA
Containment building pressure-leak tests			NA
Integrated		C	NA
Type A		C	NA
Type B		C	NA
Type C		C	NA
Reactor natural circulation cooling test		A,C,D	AP
REACTIVITY EVENTS (RE)			
NEGATIVE REACTIVITY (REN)			
Single control element insertion or drop, partial or full	Anticipated	A,D	NA
Spurious actuation of one shutdown system	Anticipated	A,D	NA
Liquid poison injection (HOLD)	TBD	A,D	NA
Light water injection into reflector tank	Unlikely	A,D	NA
POSITIVE REACTIVITY (REP)			
Shim/safety withdrawal at normal speed from start-up or low power conditions	Anticipated	A	NA
Shim/safety withdrawal at normal speed from full power	Anticipated	A,D	NA
All-rod withdrawal at normal speed	Unlikely	A,D	NA
Rapid expulsion of a single rod (may be precluded by design)	TBD	A,D	NA
Single beam tube flooding	Anticipated	A,D	NA
Cold Source Inventory Change	Anticipated	A,D	NA
Multiple beam tube flooding	Unlikely	A,D	NA
Light water injection via pressurizer pumps	Unlikely	A,D	NA
Light water slug enters core following start of H_2O-contaminated spare loop (may be prevented by design)	Extremely unlikely	A,D	NA

Table L1 (continued)

Event	Frequency Category[1]	T-H Analysis Tools	ANSR PIR Team Results
LOSS OF COOLANT PRESSURE CONTROL (LOPC) PRESSURE DECREASE			
One letdown valve goes fully open[2]	Anticipated	A	PA
All letdown valves go fully open	Anticipated	A	PA
Pressurizer pump shutdown	Anticipated	A	PA
Overpressure relief valve fails open	Unlikely	A,D	PA
PRESSURE INCREASE			
One letdown valve goes closed[2]	Anticipated	A	NA
All letdown valves go closed	Anticipated	A	NA
Inadvertent start of one or more pressurize (charging) pumps	Anticipated	A	NA
Pressurizer pump overspeed (if variable speed pump or speed reduction coupling used)	Anticipated	A	NA
PRIMARY COOLANT FLOW INCREASE (FI)			
Inadvertent start of one or more primary coolant pumps	Anticipated	A	NA
Failure of core by pass flow restrictor	Unlikely	A	NA
LOSS OF PRIMARY COOLANT FLOW (LOF) LOSS OF FORCED FLOW			
Single pump shutdown	Anticipated	A,D	NA
All pumps coastdown to pony motor flow	Anticipated	A,D	NA
Single pump shaft break	Unlikely	A,D	NA
All pumps coastdown to natural circulation flow (all pony motors fail)	Extremely unlikely	A,D	PA
LOSS OF FLOW PATH			
Single isolation valve closed	Anticipated	A,D	NA
Flow strainer in one loop blocked	Unlikely	A,D	NA
Multiple isolation valve closure	Extremely unlikely	A,D	NA
(Multiple isolation valve closure prevented by interlock)			
LOSS OF REFLECTOR COOLANT FLOW (LORF)			
All pumps shutdown	Anticipated	A	NA
All flow control or isolation valves closed	Anticipated	A	NA
CORE FLOW BLOCKAGE (CB)			
Experiment or transuranic target structural failure	Anticipated	B,C	NA
Foreign object in coolant	Anticipated	B,C	NA
Core inlet strainer structural failure	Unlikely	B,C	NA
Major core inlet flow blockage	Extremely unlikely	C	NA

Table L1 (continued)

Event	Frequency Category[1]	T-H Analysis Tools	ANSR PIR Team Results
LOSS OF HEAT SINK (LOHS)			
Loss of one normal heat sink[2]	Anticipated	A,D	NA
Loss of all normal heat sinks outside containment	Anticipated	A,D	NA
LOSS OF COOLANT pD CONTROL (ACID)			
High pD (loss of HNO_3 addition)	Anticipated	B	NA
Low pD (excessive HNO_3 addition)	Anticipated	B	NA
LOSS OF PRIMARY COOLANT (LOC)			
SIZES			
Small (Depressurization not sufficient to cause immediate scram)	Anticipated	A,D	PA
Medium (Rapid depressurization to below scram setpoint, but pressure adequate for AC motor operation of 1 primary coolant pump)	Unlikely	A,D	PA
Large (Immediate depressurization to ambient pressure)	Extremely unlikely	D	AP
LOCATIONS:			
Small, medium, and large leaks and breaks to be examined in variety of possible locations, including			
Reactor to reflector coolant (CPBT)			
Reactor to reactor pool			
Reactor to water cell			
Reactor to limited volume air cell			
Reactor to elevated air cell			
Reactor main heat exchanger tube break			
Reactor emergency heat exchanger tube break			
Reactor to subpile room			
LOSS OF REFLECTOR COOLANT (LORC)			
SIZES			
Small (size insufficient to cause immediate degradation of safety related reflector cooling or moderator functions)	Anticipated	A	NA
Large (immediate degradation of reflector cooling or moderator functions)	Unlikely	A	NA
LOCATIONS:			
Small and large leaks and breaks to be examined in variety of possible locations, including			
Reflector beam tube break			
Reflector to reactor pool			

Table L1 (continued)

Event	Frequency Category[1]	T-H Analysis Tools	ANSR PIR Team Results
Reflector to water/air cell (HOLD)			
Reflector Auxiliary Heat Exchanger tube break			
LOSS OF SECONDARY COOLANT (LOSC)			
SIZES			
Small (insufficient to cause immediate degradation of secondary cooling)	Anticipated	A,D	NA
Large (immediate degradation of secondary coolant)	Unlikely	A,D	NA
LOCATIONS:			
Small and large breaks to be examined in a variety of possible locations, including			
Reactor support building			
Pipe chase			
Basin, pump section			
Basin, discharge			
NON-CONDENSIBLE GAS EVENTS (NCG)			
Coolant off-gas as a result of primary coolant depressurization	Anticipated	D	AP*
Failure of gas-cooled irradiation experiment	Anticipated	C,D	NA
Accumulator excess gas supply	Unlikely	C,D	NA
EVENTS WITH FAILURE OF SCRAM SYSTEM (ATWS)			
Anticipated Event with failure of primary scram system	Unlikely	A,D	NA
Unlikely Event with failure of primary scram system	TBD	A,D	NA
FUEL HANDLING ACCIDENTS			
NEW FUEL STORAGE	TBD	C	NA
FUEL TRANSFER OPERATIONS	TBD	C	NA
FUEL HOT CELL OPERATIONS	TBD	C	NA
SPENT FUEL STORAGE	TBD	C	NA
Loss of criticality control	TBD	C	NA
Loss of spent fuel cooling	TBD	C	NA
Fuel element stuck	TBD	C	NA
Fuel element drop	TBD	C	NA
LOSS OF ELECTRICAL POWER (LOEP)			
Loss of all offsite power	Anticipated	A,D	AP
Station Blackout	Unlikely	A,D	PA
Loss of all non-1E power	Anticipated	A,D	AP

*Considered in LBLOCA PIR.

Table L1 (continued)

Event	Frequency Category[1]	T-H Analysis Tools	ANSR PIR Team Results
EXPERIMENT ACCIDENTS			
Cold Source (CS)			
Loss of cooling	Anticipated	C	NA
Pressure boundary fracture	Unlikely	C	NA
Internal D_2-air explosion	Extremely unlikely	C	NA
Hot Source (HS)			
Loss of temperature control	Anticipated	C	NA
Pressure boundary fracture	Unlikely	C	NA
Transuranic Targets (TRU)			
Pin-hole leak			
Pin-hole leak with water-logging			
Major perforation			
Structural failure, target or mounting hardware			
Loading error (manufacturing, not detected before operation)			
Material Irradiation (IRR)			
Inadequate cooling			
Loss of primary experiment containment boundary integrity			
Loss of experiment containment primary and secondary boundary integrity			
Major structural failure			
RADIATION RELEASE FROM COMPONENTS (RR)			
Radioactivity contained in normal liquid or gaseous process waste streams, not associated with severe fuel damage accidents shall be assumed to be released as a result of subsystem or component failure. Component and subsystem failures considered shall include but not be limited to the following:			
Radioactive waste system component failure, liquid release	Unlikely	C	NA
Radioactive waste system component failure, gaseous release	Unlikely	C	NA
Beam or guide tube rupture, tritium and D_2O release	Unlikely	C	NA
OTHER INTERNAL EVENTS			
Fires	TBD	C	NA
Equipment generated missiles	TBD	C	NA
Flooding	TBD	C	NA
Pools			
Water cells			
Secondary coolant			
Heavy object drop	TBD	C	NA

Table L1 (continued)

Event	Frequency Category[1]	T-H Analysis Tools	ANSR PIR Team Results
EXTERNAL EVENTS			
Tornado	TBD		PA
Seismic	Anticipated		PA
Floods	Unlikely		

[1]The anticipated category includes events or mishaps at frequency greater than 10^{-2}/year. Unlikely includes accidents of frequency between 10^{-4}/year and 10^{-2}/year, and extremely unlikely includes accidents of frequency between 10^{-6}/year and 10^{-4}/year.

[2]These non-limiting events are included for analysis to show that the plant control system is capable of controlling plant parameters in such a manner that the reactor does not scram as a result of the event, and continues to operate at full power or some reduced power after the event.

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