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PRISM™

Preliminary Safety Information Document

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Volume IV

Chapters 15-17 and Appendices A-E

APPLIED TECHNOLOGY

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Letter dated 5/26/93

GENERAL ELECTRIC
ADVANCE NUCLEAR TECHNOLOGY
SAN JOSE, CALIFORNIA

AMENDMENT 9

87-568-04

DISCLAIMER

Department of Energy
Washington, DC 20585

MAY 26 1993

Mr. Dennis M. Crutchfield
Associate Director for Advanced Reactors
and License Renewal
Office of Nuclear Reactor Regulation
U.S. Nuclear Regulatory Commission
Washington, D.C. 20555

Dear Mr. Crutchfield:

As you indicated in your letter, dated April 29, 1993, you are completing the final Preapplication Safety Evaluation Report (PSER) for the "Power Reactor Innovative Small Module" (PRISM) Advanced Liquid Metal Reactor design. You expressed concern about meeting one of the Commission's objectives of public disclosure since the PSER will be based on documents on which the Department of Energy (DOE), Office of Nuclear Energy, placed a restrictive distribution labeled "Applied Technology." We hereby approve your request for public disclosure and you are authorized to remove the "Applied Technology" (AT) distribution limitation from all of the DOE documents titled Preliminary Safety Information Document. The documents are:

"PRISM - Preliminary Safety Information Document" (PSID) - GEFR-00795

Volume I	- December 1987, Chapters 1-4
Volume II	- December 1987, Chapters 5-8
Volume III	- December 1987, Chapters 9-14
Volume IV	- December 1987, Chapters 15-17 and Appendices A-E
Volume V	- February 1988, Amendment to PSID
Volume VI	- March 1990, Appendix G

With regard to the Modular High Temperature Gas-Cooled Reactor (MHTGR), we would like to request that public disclosure of its AT information be delayed until publication of the MHTGR PSER becomes more imminent. We would appreciate your understanding of this situation and assure you that we will release MHTGR AT for public disclosure when needed to support the PSER issuance. We will be happy to meet with you and your staff to discuss this further at your convenience.

Sincerely,

Jerry D Griffith

Jerry D. Griffith
Director
Office of Advanced Reactor Programs
Office of Nuclear Energy

cc:
Salma El-Safwany, DOE/SF
James Quinn, GE
Richard Hardy, GE
Robert Pierson, NRC
Ray Mills, PDCO

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ABSTRACT

This document is a Preliminary Safety Information Document (PSID) for a PRISM (Power Reactor Inherently Safe Module) electric power plant. The PSID is the document in the PRISM licensing plan that provides the description and evaluation of the conceptual design using nine reactor modules. Each module is a compact liquid metal reactor of the pool type design. The reactor module has unique passive safety characteristics that enhance the safety of the design. These include passive shutdown heat removal and passive reactivity shutdown. The document presents design criteria, design description and analyses that demonstrate these favorable safety characteristics. The format is similar to the standard format for safety analysis reports, however, the design description and evaluations are consistent with the conceptual design level. Design basis accidents are described in Chapter 15 and a preliminary PRISM probabilitic risk assessment is included in Appendix A.

CHAPTER 15
ACCIDENT ANALYSIS

CHAPTER 15

ACCIDENT ANALYSIS

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CHAPTER 15 ACCIDENT ANALYSIS

15.1 Introduction

Design basis events (DBE's) impose requirements on the design of components and systems that have safety-related functions, and define the range of conditions under which these functions are to be performed.

Methodology for defining DBE's is evolving from early practice in which safety design based on a single bounding event (e.g., the maximum credible accident) was believed to be adequate. Current practice focuses on the use of systematic and quantitative approaches leading to a number of DBE's covering a range of probabilities and assuring completeness in the identification of challenges to safety. This focus is a natural product of the increased use of probabilistic risk assessment as a safety assurance tool.

The approach to DBE selection for the PRISM design is described here. The method is systematic, drawing on probabilistic risk assessments (PRA) done in the conceptual stage and later stages of design. The PRISM approach blends the structured quantitative approach with pragmatic use of previous experience.

The ultimate objective of safety design is protection of the public against uncontrolled radiological releases. The PRISM design process provides considerable margin to this objective for events within the design basis envelope. This comes about in two ways: first, design acceptance criteria are conservative with respect to the ultimate goal of preventing radiological releases; and second, the analysis of design performance against these criteria allows for uncertainties in a very conservative way so that there are demonstrable margins between calculated performance and acceptance criteria.

The employment of conservative acceptance criteria and demonstrable margins in performance analysis characterizes design basis events and sets them apart from other postulated events. A third characteristic imposed on the selection of DBE's for PRISM is that they have a definable impact on the design. At the conceptual stage, the list of "potential" DBE's tends to be longer than the final list that determines the final design. Previous experience and judgment are relied on to identify those events likely to have design impact, but the final screening cannot be completed until the design is essentially known.

The PRISM approach to safety uses three levels of safety assurance and a defense-in-depth design approach with particular emphasis on inherent passive features. The first level of safety is the inherent and basic design characteristics. This level focuses on reliable normal operation, and accident prevention through features of the plant design, construction, operation and maintainability. This includes reliability enhancement through redundancy, quality assurance, testability, inspectability, and simplified fail safe system designs. The second level of safety prevents accident propagation, recognizing that accidents may occur despite the care taken in design, construction, and operation associated with level one of safety. This second level focuses on the protection against anticipated events and unlikely events. The reactor shutdown system, actuated by the reactor protection system (RPS), and shutdown heat removal systems (SHRS) provide high reliability protection functions. For the anticipated and unlikely events, heat is primarily removed by the normal heat transport path to the condenser or by the auxiliary cooling system (ACS). The third level of safety focuses primarily on events classified as extremely unlikely events. These events are not expected to occur during the plant life. These events are analyzed to establish a conservative design basis. The events above (normal operation, anticipated events, unlikely events and extremely unlikely events) constitute the PRISM DBE's.

An objective in safety design is to assure that there exists an inverse relationship between the frequency of occurrence of events and their consequences. The defense in depth principle provides such assurance and is a major focus of the PRISM approach.

In keeping with the concept of defense-in-depth, design basis events are selected at more than one point along any postulated accident sequence. Thus, in those relatively improbable occasions when the first level fails to terminate the sequence, their are other levels available to do so.

PRA is employed as a design tool in PRISM. A preliminary PRA is used in the conceptual design phase to define a complete set of accident sequences from initiating event to radiological release into the environment. Probabilities have been assigned to these event sequences (initially, these are based on judgment obtained from previous projects). The event tree structure of the PRA provides visibility for safety functions (i.e., shutdown, heat removal and containment) simultaneous with event frequencies and consequences. Event frequency plays a major role in DBE selection (but is not the only measure). As a general rule, events with frequency lower than 10^{-6} per year are excluded from the design basis envelope, however, design selection itself causes the shift of occurrence frequency from one path to another in the event tree structure. The selection of design basis events thus involves consideration of performance and cost trade-offs among design options as well as event frequencies. These trade-offs arise from the fact that features relied upon to terminate event sequences within the design basis event envelope must be designed conservatively with demonstrable margins to safety design limits.

With respect to the complete set of event sequences described in PRA event trees, all events can be categorized as either within the DBE envelope or as beyond design basis events (BDBE). Commitment to use of PRA as a design tool carries with it a commitment to modify the design if PRA results suggest that this should be done. Thus, all events can have impact on the design, even those outside the formality of design basis events. The PRISM project has selected a specific set of BDBE's which have definite impact on the design. This set of events establishes a basis for assuring that inherently safe response characteristics are built into the design. These events are selected to include the most probable core disruptive accident initiators.

Their frequencies of occurrence are smaller than 10^{-6} per year. Acceptance criteria for design performance against these events are less conservative than the acceptance criteria for DBE's, however they still provide significant margin to offsite consequences (i.e., radiological release). Design analysis against these events is also less conservative than that done for events within the DBE envelope. Analysis of BDBE's is carried out using appropriate nominal values without additional allowance for uncertainties.

Each of the three levels of safety of the defense-in-depth approach, the role of beyond design basis events, and the use of risk assessment are described in Section 15.2. The corresponding PRISM safety evaluation procedure is developed in Section 15.3.

15.2

15.2 PRISM Approach to Safety

15.2.1 First Level of Safety - Inherent and Basic Design Characteristics

PRISM, as does any liquid metal reactor (LMR), embodies specific inherent safety features. Some of the inherent safety characteristics common to LMRs are 1) the favorable combination of viscosity, conductivity, and vapor pressure associated with the use of sodium to remove heat, 2) the ability to operate the reactor efficiently at hundreds of degrees below the boiling point of the sodium coolant, 3) the ability to operate at essentially ambient pressure thus reducing the pressure exerted on the coolant system boundaries. As a result of these inherent features, the sodium-cooled reactor can use relatively simple design approaches to maintain adequate coolant inventory if a leak develops in the coolant boundary.

PRISM incorporates additional inherent features in its specific design. The nuclear reactor module is designed and sized so that inherent safety features reduce the quantity and complexity of engineered safety features. These PRISM features include:

1. The small size of PRISM allows factory fabrication of the reactor module with improved quality assurance control.

2. The reduced core size and independence of the reactor modules, with respect to safety functions, reduces the amount of fission products available for potential release.

3. Safety-related functions are confined to each reactor module. There is no dependence on the balance of plant for safety functions.

4. Selection of core materials and restraints provide a net negative temperature coefficient of reactivity thus assuring reliable feedback mechanism enhancing stability during normal operation and limiting reactivity excursions.

5. Reactor fuel and blanket assemblies have pin spacing and material selection that reduce the potential for the reduction in coolant flow due to fuel or clad swelling.

6. Core inlet nozzles are designed with multiple flow inlet passages to prevent flow blockages.

7. PRISM's size allows seismic isolation to mitigate seismic acceleration on the reactor module.

8. In addition to normal heat removal via the balance of plant (BOP), and heat removal via an auxiliary cooling system, PRISM has an inherent passive reactor vessel auxiliary cooling system (RVACS).

9. Use of coolant boundary materials (stainless steel) with high fracture toughness.

10. Use of fuel material with high retention of fission products.

15.2.2 Second Level of Safety - Protection Against Anticipated and Unlikely Events

Recognizing that accidents may occur during the plant lifetime, even with the care taken to assure normal operation (first level of safety), PRISM design has focused on highly reliable and inherent features to prevent propagation of anticipated and unlikely events to more serious accidents. In many events the PRISM plant control system (PCS) will take actions, including reactor trip, if warranted, to prevent propagation of plant events. However, safety-related systems are provided to assure reactor shutdown and cooling for all plant design basis events. The major PRISM protection features are:

1. The RPS provides automatic shutdown of the reactor module. The RPS is independent of the control room and PCS, and located with each reactor module.

2. Plant shutdown heat removal is accomplished via the normal heat removal path, or the ACS for unlikely events. An RVACS continuously operates in a passive heat removal mode and is capable of removing the complete heat load as required.

3. The natural draft air cooling of the reactor module by the RVACS provides an essentially fail proof heat removal system.

4. All safety-related systems and components are protected from or designed to withstand the effect of natural phenomena (floods, earthquakes, etc.) and abnormal environmental conditions.

5. Reactor core heat removal is maintained if primary sodium leaks from the reactor vessel to the containment vessel.

6. Four primary coolant pumps are provided, with flow coastdown ability, so that core coolability is maintained during the transient to natural circulation. Each pump has its own independent synchronous converter powered coastdown system.

7. The containment boundary is designed to withstand the effects of Na/H_2O reaction postulated to occur in the steam generator.

15.2.3 Third Level of Safety - Protection Against Extremely Unlikely Events

The third level of safety provides acceptable plant response to extremely unlikely events. PRISM maintains reliable core cooling, and reactor shutdown, to retain core integrity for these events. The RPS, reactivity control systems, and RVACS identified within the second level of safety provide the necessary reactor protection functions for extremely unlikely events. Containment of radioactivity that may be inadvertently released due to fuel cladding failure is retained by the containment vessel and reactor module closure assembly.

15.2.4 Beyond Design Basis Events for PRISM

The ultimate means of protection of public safety from the consequences of postulated loss-of-cooling and transient overpower events without scram, will be the inherent negative reactivity feedback resulting from reactor system temperature increases. To assure that the design incorporates this inherently safe response capability, BDBE's (combining accident initiators with no control rod actuation) impose requirements on the design. The reactor core design will be modified as needed to strengthen these effects to the point that adequate reliability is achieved, on a nominal basis with appropriate margin.

15.2.5 Risk Assessment

PRA is a continuing, pervasive influence on the safety design process for PRISM. The use of PRA serves several objectives including:

1. Providing a visible structure for selection of DBE's and BDBE's.

2. Providing a basis for assigning reliability requirements to systems and components.

3. Providing the measure of conformance to design objectives stated in the NRC Safety Goal Policy (Ref. 15.2-1).

4. Conformance with the NRC Severe Accident Policy (Ref. 15.2-2), which calls for the performance of PRA for advanced nuclear plants.

5. Identification and prioritization of safety issues throughout the design process.

6. Providing safety insights to support design trade-offs.

DBE's and those BDBE's discussed in Section 15.2.4 establish specific design requirements, and are analyzed in detail. As the result of this approach, events of high probability present no significant risk because they do not involve release of significant radioactivity. Events at the low probability end of the spectrum present no significant risk because of their low probabilities. The defense-in-depth approach ensures low probabilities for these higher consequence events.

Between the extreme high and extreme low ends of the probability scale lie DBEs and BDBE's that directly influence design decisions. Figure 15.2-1 summarizes the relationship of event categorization to occurrence probability.

References - Section 15.2

15.2-1 "Safety Goals for the Operation of Nuclear Power Plants, Policy Statement," 28044 Federal Register, Vol. 51, No. 149, Monday, August 4, 1986.

15.2-2 Policy Statement on Severe Reactor Accidents Regarding Future Designs and Existing Plants, noticed in the Federal Register, Volume 50, No. 153, August 8, 1985.

A. The spectrum of theoretically possible events is extremely broad. None are omitted from the PRA.

B. The highest likelihood events do not present significant risk. They are not evaluated in detail.

C. The lowest likelihood events do not present significant risk. They are not evaluated in detail.

D. A set of design basis events is conservatively selected to bound the events that are sufficiently likely and consequential to control the plant design.

E. Beyond the design basis events include events with failure to insert shutdown rods, to provide the design basis for inherent response characteristics.

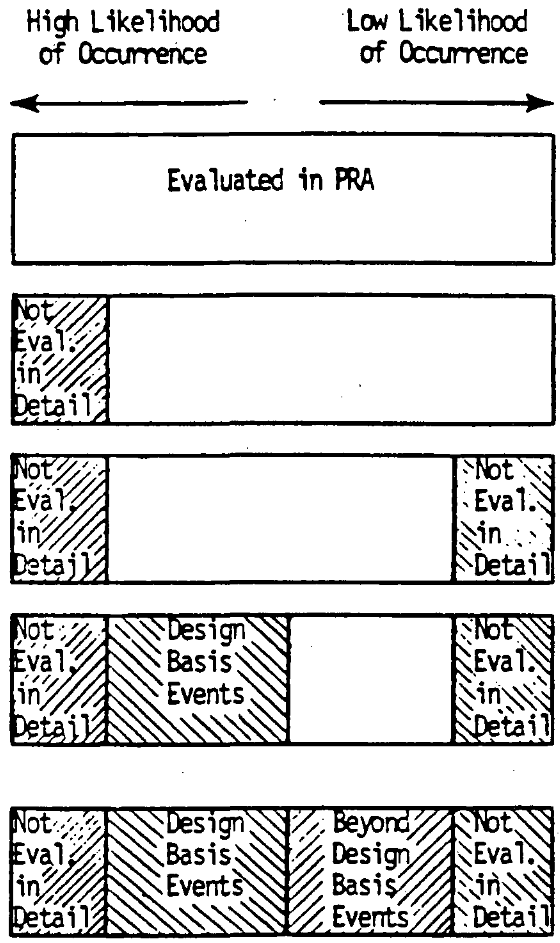

Figure 15.2–1 THE SPECTRUM OF EVENTS ANALYZED IN PRA

15.3

15.3 Safety Evaluation Procedure

The PRISM safety evaluation procedure, summarized in Tables 15.3-1 through 15.3-6, consists of the following steps:

1. Event selection
2. Event categorization
3. Event Analysis
4. Risk assessment

Each of these steps is described in the following sections.

15.3.1 Event Selection

In the PRISM approach to safety design, PRA is an important design tool. It helps to assure completeness in the identification of accident sequences, and to rank the sequences in order of importance based on the combination of occurrence frequency and offsite consequences. It is the intent of the PRISM project to continue using PRA throughout design, so that the attention in safety design remains focused on issues of significance (as measured by their impact on public risk). PRA contributes to design trade-off decisions in the conceptual stage, as well as later stages. Finally, and perhaps more important to the subject of this section, the PRA provides a basic framework for DBE selection.

The initial step toward DBE selection is the establishment of a complete set of event trees. The initial basis for this was the set of generic initiating events and event trees developed in the LMR base technology program over the past twelve years (Ref. 15.3-1). The event trees have subsequently been modified and refined in parallel with more specific development of the PRISM design concept.

Similarly, while the earlier probability assignments were based on previous design studies, the current ones incorporate results from PRISM system reliability analyses and accident analyses specific to the design.

With respect to the event tree structure, event sequences leading to no significant consequences describe the desired responses of safety systems and safety functions. DBE's are therefore selected from the sequences. Specific selection criteria include:

1. Event sequences with frequency greater than 10^{-6} must be within the DBE envelope.

2. Within the DBE envelope, events of greater severity must have lower frequency.

These criteria do not yield a unique set of events. The selection basis incorporates additional factors that are related to non-safety operational objectives and to cost. These considerations arise in the course of trade-off decisions; the trade-off in this case being between making a given system or function conform to stringent safety requirements or accepting as a DBE the postulated failure or non-existence of that system. This process involves the combination of pragmatically applied engineering judgement and the systematic PRA structure. It is an ongoing process involving successive refinements of design definition, PRA model definition, and reconsideration of the proper placement of the design basis event envelope.

All conceivable challenges to safety systems to perform their safety functions are considered candidate DBE's. Specifically, the events of interest are those postulated sequences which challenge and have definable impact on the design of components and/or systems that have safety-related functions. In the initial selection of events there is no attempt to determine if an event is a DBE or an BDBE. This distinction is made as part of the event categorization step (see Section 15.3.2).

Events are identified for systems, components or structures that have key safety functions associated with reactor shutdown, shutdown heat removal and control of radiological releases. Safety functions are identified for each system, component or structure. At the preliminary design stage, events are assessed to identify their potential to impair a safety related function. Event selection is based on engineering experience with analyses of similar events on comparable systems. The selection of events is aided by a systematic review of the following resources:

1. PRISM duty cycle events (Appendix D).

2. Events utilized for the Clinch River Breeder Reactor Plant (Ref. 15.3-2).

3. Light water reactor events identified in the Standard Review Plan (Ref. 15.3-3).

4. Assessment of events unique to PRISM.

5. Events generic to all nuclear reactors including sabotage and other external sources.

6. The U. S. Atomic Energy Commission list of representative types of LMR events (Ref. 15.3-4).

In summary, it is important that all events which challenge and have a definable impact on the safety-related systems associated with reactor shutdown, shutdown heat removal and control of radiological releases be included in the initial list of events. Certain of these events in the initial list may be eliminated from further consideration. For example, one or more events may clearly dominate or envelop other events. A second example is the elimination of an event from further consideration because of its small probability of occurrence. The reduction in the list of events from the initial to final selection occurs as the design and safety evaluation matures.

15.3.2 Event Categorization

Table 15.3-1 provides definitions for the event categories to be used in conjunction with the numerical frequency ranges given in Table 15.3-2. Each event is placed into one of the four DBE categories or the BDBE category using its nominal frequency. These frequency ranges are the same as those currently used by General Electric for its BWR's (Ref. 15.3-5) and similar to those recommended by ANS standards (Refs. 15.3-6, -7) for LWR's. In each case, the division between DBE's and BDBE's is the frequency of 10^{-6} per reactor year. This same figure is also being used for international LMRs (Ref. 15.3-8).

15.3.3 Design Basis Event Analysis

The defense-in-depth approach evaluation is performed as summarized in Table 15.3-2. Conservative calculational bases are used to predict plant performance during the event. Also, for each DBE category a single limiting DBE (or set of DBE's) can be selected which envelops all of the DBE's in that category.*

This evaluation uses four sets of acceptance criteria: reactor shutdown, shutdown heat removal, radiation exposure to plant personnel and offsite radiological dose. The acceptance criteria are based on the premise that if appropriate fuel design and coolable geometry limits are not exceeded and if radiological releases are limited so that the dose guidelines presented in 10 CFR 100 are not exceeded for the postulated site suitability source term then the public health and safety are adequately protected.

15.3.3.1 Reactor Shutdown

The reactor shutdown acceptance criteria are that at least two highly reliable, redundant, and diverse means of shutting down are provided, either one of which is capable of shutting down the core fission power for all DBE's such that the calculated temperature limits of Table 15.3-3 are not exceeded during the event sequence.

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* For the evaluation of radiation exposure to plant personnel and off-site radiological dose, it is necessary to include all DBE's capable of producing radiation exposure to plant personnel or offsite radiological dose.

15.3.3.2 Shutdown Heat Removal Acceptance Criteria

Temperature limits are established for the reactor core cladding, Primary Heat Transport System (PHTS) sodium coolant boundary. Specific temperature limits for the PHTS are based upon the type of materials used, the frequency of occurrence of each event category and the time duration of each event. In the case of the SHRS, these parameters are evaluated using the service levels limits of Table 15.3-4. These service level limits are consistent with current LWR practice (Ref. 15.3-5).

15.3.3.3 Radiation Exposure to Plant Personnel Acceptance Criteria

The radiation exposure to plant personnel acceptance criteria shown in Table 15.3-2 are consistent with current BWR practice (Ref. 15.3-5).

15.3.3.4 Offsite Radiological Dose Acceptance Criteria

The offsite radiological dose acceptance criteria shown in Table 15.3-2 are those recommended by ANS (Refs. 15.3-6, -7).

15.3.4 Beyond Design Basis Events

In PRISM, the ultimate means of protection of public safety from the consequences of postulated loss-of-cooling and transient overpower events without scram, will be the inherent negative reactivity feedback resulting from reactor system temperature increases and the inherent RVACS system. Analyses of BDBE's are conducted to assure that these inherent features are effective in the PRISM design. Appendix E identifies the acceptance criteria for BDBE's and provides an initial evaluation against these criteria.

15.3.5 Risk Assessment

Conformance to NRC Safety Goals is measured by the PRA. Methods developed for LMR risk assessment (Ref. 15.3-1) are used. As called for in the NRC Safety Goal Policy (Ref. 15.3-9), mean risk values are calculated for comparison to design goals. Appendix A provides a preliminary risk assessment of PRISM and compress the results against Reference 15.3-9.

References - Section 15.3

15.3-1 D.E. Hurd, et al., "Single Plant Risk Model Development and Application," GEFR-00573, August 1981.

15.3-2 Clinch River Breeder Reactor Plant Preliminary Safety Analysis Report. Project Management Corporation. Volume 9. Chapter 15.

15.3-3 Standard Review Plan for the Review of Safety Analysis Reports for Nuclear Power Plants - LWR Edition. U. S. Nuclear Regulatory Commission Document NUREG-0800. July 1981.

15.3-4 Standard Formal and Content of Safety Analysis Reports for Nuclear Power Plants - LMFBR Edition. U. S. Atomic Energy Commission. February 1974. (Table 15-1).

15.3-5 "Product Safety Standard," General Electric Company Document 22A8400.

15.3-6 American National Standards Institute/American Nuclear Society Standard - ANSI/ANS-51.1-1983. Nuclear Safety Criteria for the Design of Stationary Pressurized Water Reactor Power Plants.

15.3-7 American National Standards Institute/American Nuclear Society Standard - ANSI/ANS-52.1-1983. Nuclear Safety Criteria for the Design of Stationary Boiling Water Reactor Power Plants.

15.3-8 "Status of Liquid Metal Cooled Fast Breeder Reactors," Technical Report Series No. 246, International Atomic Energy, Vienna, 1985.

15.3-9 "Safety Goals for the Operation of Nuclear Power Plants, Policy Statement," 28044 Federal Register, Vol. 51, No. 149, Monday, August 4, 1986.

TABLE 15.3-1

EVENT CATEGORIES AND DEFINITIONS

EVENT CATEGORY	DEFINITION
DESIGN BASIS EVENTS:	
Normal Operation	Any condition of system startup, design range operations, hot standby or shutdown.
Anticipated Event	An off-normal condition which individually may be expected to occur once or more during the plant's lifetime.
Unlikely Event	An off-normal condition which individually is not expected to occur during the plant's lifetime; however, when integrated over all plant components, events in this category may be expected to occur a number of times.
Extremely Unlikely Event	An off-normal condition of such extremely low probability that no events in this category are expected to occur during the plant's lifetime, but which nevertheless represents extreme or limiting cases of failure which are identified as design bases.
BEYOND DESIGN BASIS EVENTS:	Off-normal conditions of such extremely low probability that no events in this category are credible during the plant's lifetime, but which have such extreme consequences that the risk (probability time consequence) from these events merits their consideration in establishing the design.

TABLE 15.3-2

SUMMARY OF ACCEPTANCE LIMITS FOR DESIGN BASIS EVENTS

DESIGN BASIS EVENT CATEGORY	FREQUENCY[1] RANGE (F) (PER REACTOR YEAR)	ACCEPTANCE LIMITS			
		REACTOR SHUTDOWN[2]	SHUTDOWN HEAT REMOVAL[2]	RADIATION EXPOSURE TO PLANT PERSONNEL	OFFSITE RADIOLOGICAL DOSE[3]
o NORMAL OPERATION	$F \geq 10^{-1}$	TABLE 15.3-4	ASME SERVICE LEVEL "A" LIMITS	10CFR20 LIMITS[5]	10CFR50, APPENDIX I LIMITS
o ANTICIPATED EVENT	$10^{-1} > F \geq 10^{-2}$	TABLE 15.3-4	ASME SERVICE LEVEL "B" LIMITS	10CFR20 LIMITS[5]	10CFR100 LIMITS
o UNLIKELY EVENT	$10^{-2} > F \geq 10^{-4}$	TABLE 15.3-4	ASME SERVICE LEVEL "C" LIMITS	10CFR20 LIMITS[5]	10CFR100 LIMITS
o EXTREMELY UNLIKELY EVENT	$10^{-4} > F \geq 10^{-6}$	TABLE 15.3-4	ASME SERVICE LEVEL "D" LIMITS	NOTE 4	10CFR100 LIMITS

(1) EVENT FREQUENCIES ARE NOMINAL VALUES

(2) LIMITING DBE's CAN BE SELECTED FOR EACH DBE CATEGORY IN EVALUATING REACTOR SHUTDOWN AND SHUTDOWN HEAT REMOVAL

(3) MUST ADDRESS ALL DBE's CONTRIBUTING OFFSITE RADIOLOGICAL DOSE

(4) RADIATION EXPOSURE TO PLANT PERSONNEL IN MAIN CONTROL ROOM NOT TO EXCEED 5 REM WHOLE BODY, 30 REM INHALATION AND 75 REM SKIN FROM ANY ONE EVENT. SSST WILL BE USED TO PROVIDE MARGIN FOR SELECTED BDBE's.

(5) MUST ADDRESS ALL DBE's CONTRIBUTING TO RADIATION EXPOSURE TO PLANT PERSONNEL

TABLE 15.3-3

REACTOR SHUTDOWN ACCEPTANCE
CRITERIA FOR DESIGN BASIS EVENTS(1)
WITH TERNARY ALLOY METAL FUEL

Event Classification	Peak Transient Temperatures, °F		Long Term Temperatures, °F	
	Bulk Coolant	Cladding*	Bulk Coolant	Cladding**
Normal Operation	1200	1200	1200	1200
Anticipated Event	1200	1200	1200	1200
Unlikely Event	1300	1450	1300	1300
Extremely Unlikely Event	1300	1450	1300	1300

(1) Calculations for comparison with these limits must incorporate a 2σ uncertainty margin in parameters having an impact on the listed temperatures.

* Temperature at cladding centerline based on preventing breach by stress rupture.

** Temperature at fuel-cladding interface based on preventing cladding breach by low-melting point formation.

TABLE 15.3-4

SHUTDOWN HEAT REMOVAL ACCEPTANCE
CRITERIA FOR DESIGN BASIS EVENTS

Service Level "A" Limits

Service Level A limits result when the normal heat transfer system (normal SHRS) is operating to remove reactor shutdown decay heat. The resulting temperatures and loadings are considered normal. The reactor core cladding temperature limit shall maintain the fuel life design margin. The PHTS and IHTS temperature limits shall be less than, or equivalent to, their design temperature which results in no damage to systems or components.

Service Level "B" Limits

Service Level B limits apply to anticipated events. The reactor core cladding temperature shall limit fuel damage to a reduction in fuel design margin only and does not affect the fuel design life. The PHTS and IHTS temperatures shall result in no reduction in component/structure design capability and no inspection required for re-operation.

Service Level "C" Limits

Service Level C limits apply to unlikely events. The reactor core cladding temperature shall limit fuel damage to a few failures. A core unload, inspection and some replacements may be required. No permanent damage to the reactor vessel or internals shall result. The PHTS and IHTS temperatures shall not result in coolant boundary failure; however, potential loss in design life may occur and inspection/repair may be required for re-operation.

Service Level "D" Limits

Service Level D limits apply to extremely unlikely events. The reactor core cladding temperature limit shall maintain a coolable core geometry and the bulk sodium temperature shall remain below boiling. The PHTS and IHTS temperature limits shall not result in coolant boundary failure. Structural integrity is maintained to provide the SHRS safety function. Protection of public health and safety is required but plant restart is not mandatory.

15.4

15.4 Reactivity Insertion DBE's

15.4.1 Uncontrolled Rod Withdrawal at 100% Power

15.4.1.1 Event Description

For PRISM reactor control, rod positioning will be accomplished in a "banked" mode, i.e., the plant control system (PCS) will seek the same position for all rods at any given operational state. The design to accomplish this will enable motion of only one rod at a time, hence a change in power level will require small incremental motions of all rods, in sequence.

Because of this design, single rod withdrawal is the most likely rod withdrawal accident. This event also bounds other reactivity insertion events considered as potential PRISM DBE's.

The highest single control rod was assumed for withdrawal at nominal speed. This results in reactivity insertion of 35 cents at the rate of 2 cents per second. The BOC core configuration was selected for the analysis, primarily due to the somewhat smaller reactivity effects from Doppler and bowing and the higher specific power.

Reactor trip was assumed to occur at 115% of full power. Actions of the PCS that would have acted to mitigate this event were also assumed to fail and the other five rods are assumed to remain in their initial position until released by the RPS.

15.4.1.2 Event Analysis

Transient analysis was performed with the ARIES-P code which is a thermal-hydraulic model prepared for PRISM plant transient analysis. The ARIES-P code simulates one, two or more heat transport system (HTS) modules and associated controllers. Key features of the PRISM plant include the reactors, vessels and components of the primary and intermediate heat transport systems, the steam generation system and the BOP. The heat generation and transport in the plant are simulated by a one, two or three reactor-module model. In a two module model, one module, the "S" module, represents a single reactor module and associated heat transfer system and the second module, the "M" module, represents the remainder of the reactor-modules in a lumped fashion.

Thermal power generation is represented by neutron kinetics and decay heat equations. The vessel internals are represented by a lower plenum, a core (which includes the lower axial blanket, the active core, the upper axial blanket and the radial blanket), a bypass channel and an upper plenum with a variable sodium level and a cover gas. The heated sodium leaves the upper hot pool into the intermediate heat exchanger and returns to the cold pool.

The reactor core is divided axially into nine sections. These represent nine core segments without upper and lower axial blankets. Each axial segment is divided radially into three sections which represent two radial fuel sections and the sodium coolant. Four core assemblies are modeled. These represent a peak and an average fuel assembly and an average inner and outer blanket. In addition, the peak power channel is used to calculate a hot channel response. The coolant flow splits between fuel, blanket and bypass are adjusted at each time step to account for friction factor and pressure drop changes. A specified fraction of the total reactor power is generated in fuel, cladding, blanket and sodium. The axial variation of power generation is governed by an input axial power profile.

Each primary heat transport system is represented by the reactor vessel flow passages, the vessel, the radiation shielding, the primary pump, and the shell or tube side of the IHX at the option of the user.

Each intermediate heat transport system is represented by the tube or shell side of the IHX at the option of the user, piping, the shell side of the steam generator and the intermediate pump.

The steam generation system is represented by feedwater control valves, a recirculation pump, a steam drum, piping, and the tube side of the steam generator. The steam leaving all the steam drums enters a common steam header and flows through piping to the turbine throttle valve and a steam dump valve.

The feedwater and main steam system is represented by the turbine control and bypass valves, the turbine generator with extractions for feedwater heating, the feedwater heaters and the feedwater pumps.

The reactor model provides peak and average channel representation in the core as well as inner and outer blanket representations. In addition, the peak power channel is used to calculate a hot channel response. Axial blanket sections (if present) are modeled as part of the fuel channel. The flow splits between the core, blankets and bypass are adjusted at each time step to account for friction factor and pressure drop changes. The different designs are represented by differences in power and flow splits between the core and the blankets and through differences in reactivity coefficients.

Flow rates in ARIES-P are calculated for controlled-speed pumps, constant speed pumps, and natural circulation. Pump pony-motor drive speed is modeled. Friction factors in the hydraulic equations are continuously updated. They account for the transition from turbulent to laminar flow in all parts of the sodium system. Natural circulation also takes into account thermally-driven density changes in all parts of the primary, intermediate and water/steam loops having elevation changes.

The ARIES-P steam generator model provides heat transfer based on subcooled, boiling or superheat conditions. Perfect separation is assumed for fluid leaving the steam drum and feedwater mixing occurs at the recirculation water outlet nozzle only. The main turbine, feedwater heater, feedwater pumps and feedwater hydraulics are included in separate modules to provide a better description of overall plant behavior.

Hot channel factors were used to estimate two-sigma values for the core outlet temperature, the peak fuel temperature and the peak cladding temperature. The ARIES-P core model performs separate thermal hydraulic calculations for four core assemblies: average fuel, average inner and radial blankets, and a peak fuel assembly. To estimate these values a set of factors was taken from a previous design study which is applicable to the coolant and cladding temperatures. To compute the two-sigma fuel temperature for the TOP event, a 13% uncertainty was applied to the fuel temperature drop (surface to inner node). This uncertainty was added to the fuel surface temperature which reflects the cladding hot channel factors and a nominal fuel cladding gap temperature increment.

Table 15.4-1 presents the hot channel factors used in this analysis, which were applied to the peak fuel assembly. Prior to their application, however, an additional factor was used to reflect the radial coolant temperature peaking within an assembly. This value was derived using batch averaged values, for the peak power to flow assembly. This intra-assembly radial peaking was determined to be 1.036 times the bulk average local temperature.

The core outlet temperature 2σ value was derived by applying a constant factor of 1.12 to the total nominal temperature rise across each assembly and then flow weighting this value. The bypass temperature was increased by the uncertainty in inlet temperature.

15.4.1.3 Analysis Results

Results from the single rod withdrawal accident analysis are plotted in Figures 15.4-1 through 15.4-4

As seen in Figure 15.4-1, core power reaches 115% at 5.5 seconds, initiating reactor trip. Figure 15.4-2 shows the power to flow ratio as a function of time. Figure 15.4-3 plots clad and coolant temperatures vs time. The two uppermost curves are 2σ values for clad and coolant temperatures, respectively. Corresponding nominal values are also shown. Peak temperatures are seen to remain well below the design limits of 1300°F for unlikely events (see Section 15.3), with allowance for uncertainties at the 2σ level. Fuel temperatures are shown in Figure 15.4-4.

TABLE 15.4-1

VALUES USED IN TWO-SIGMA TEMPERATURE DETERMINATIONS

Contributor	Value Input
Sodium Temperature Rise from Inlet Factor	1.21
Sodium Inlet Temperature Uncertainty, °F	4.90
Film Temperature Drop Factor	2.68
Cladding Temperature Drop Factor	1.11
Fuel Surface to Inner Node Temperature Factor	1.13
Fuel Cladding Gap Temperature Factor	1.00
Intra-Assembly Sodium Radial Peaking Factor	1.036
Core Wide Outlet Temperature Increase Factor	1.12

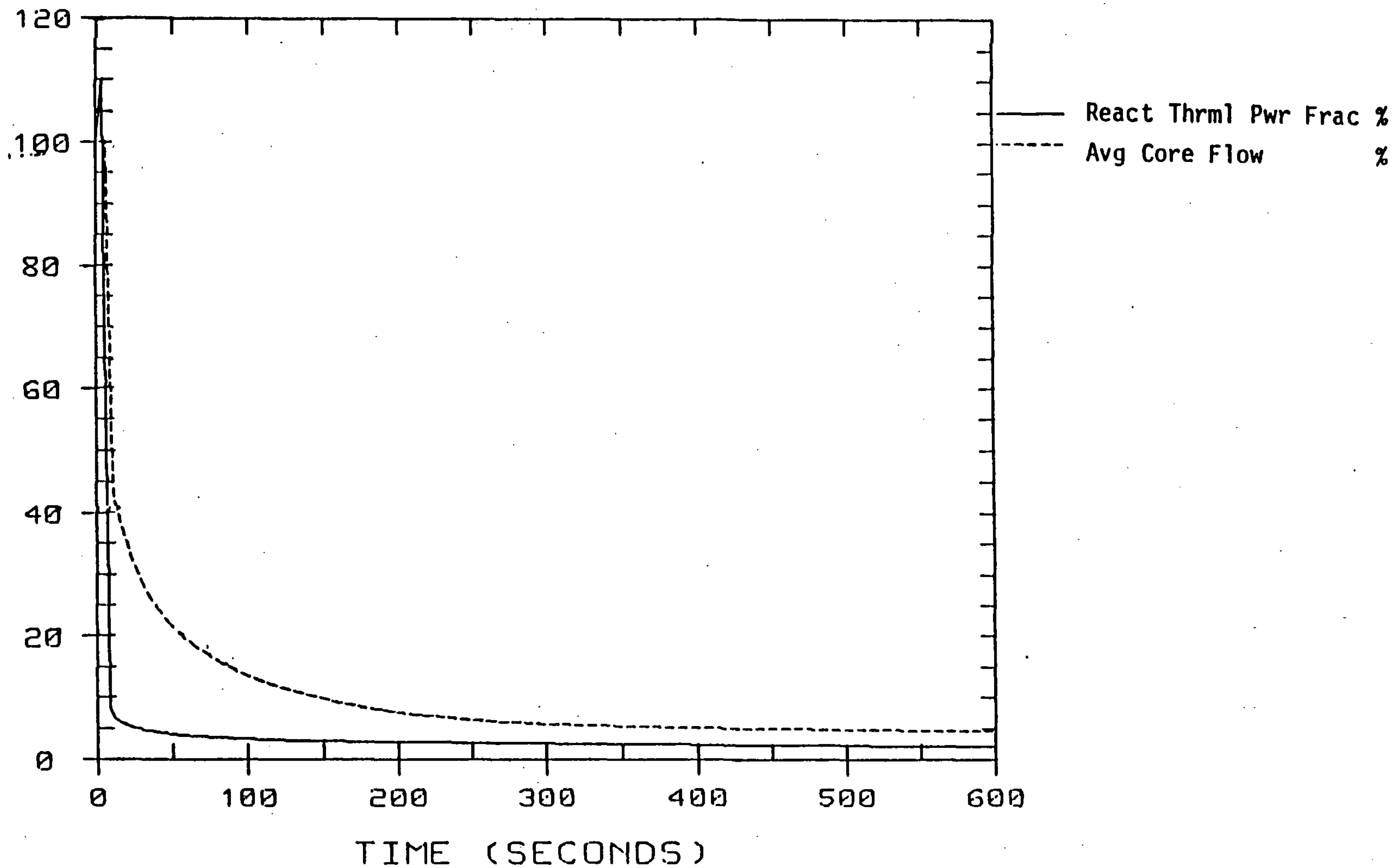

Figure 15.4–1 POWER AND FLOW TRANSIENTS SINGLE ROD WITHDRAWAL

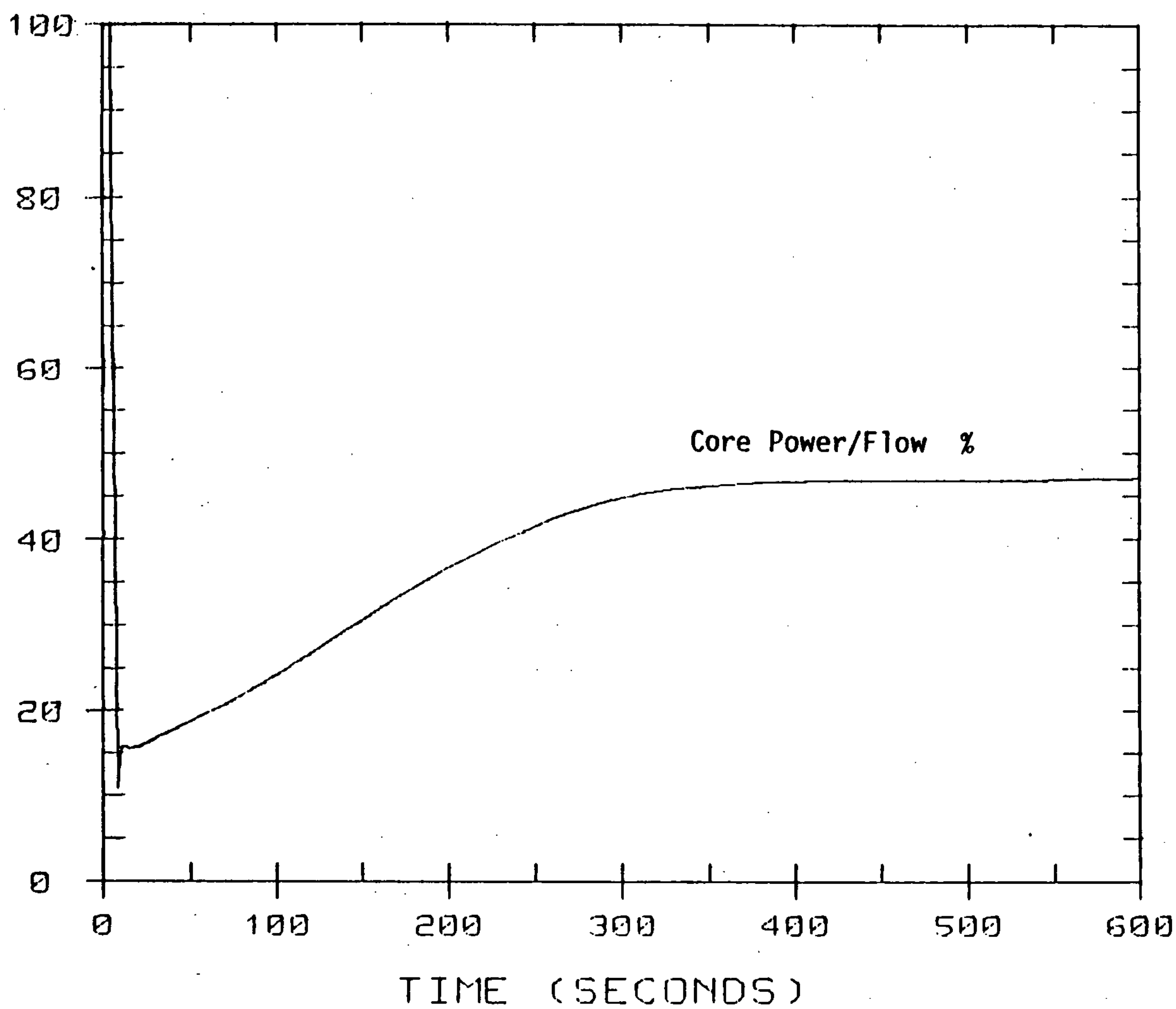

Figure 15.4–2 POWER/FLOW TRANSIENT SINGLE ROD WITHDRAWAL

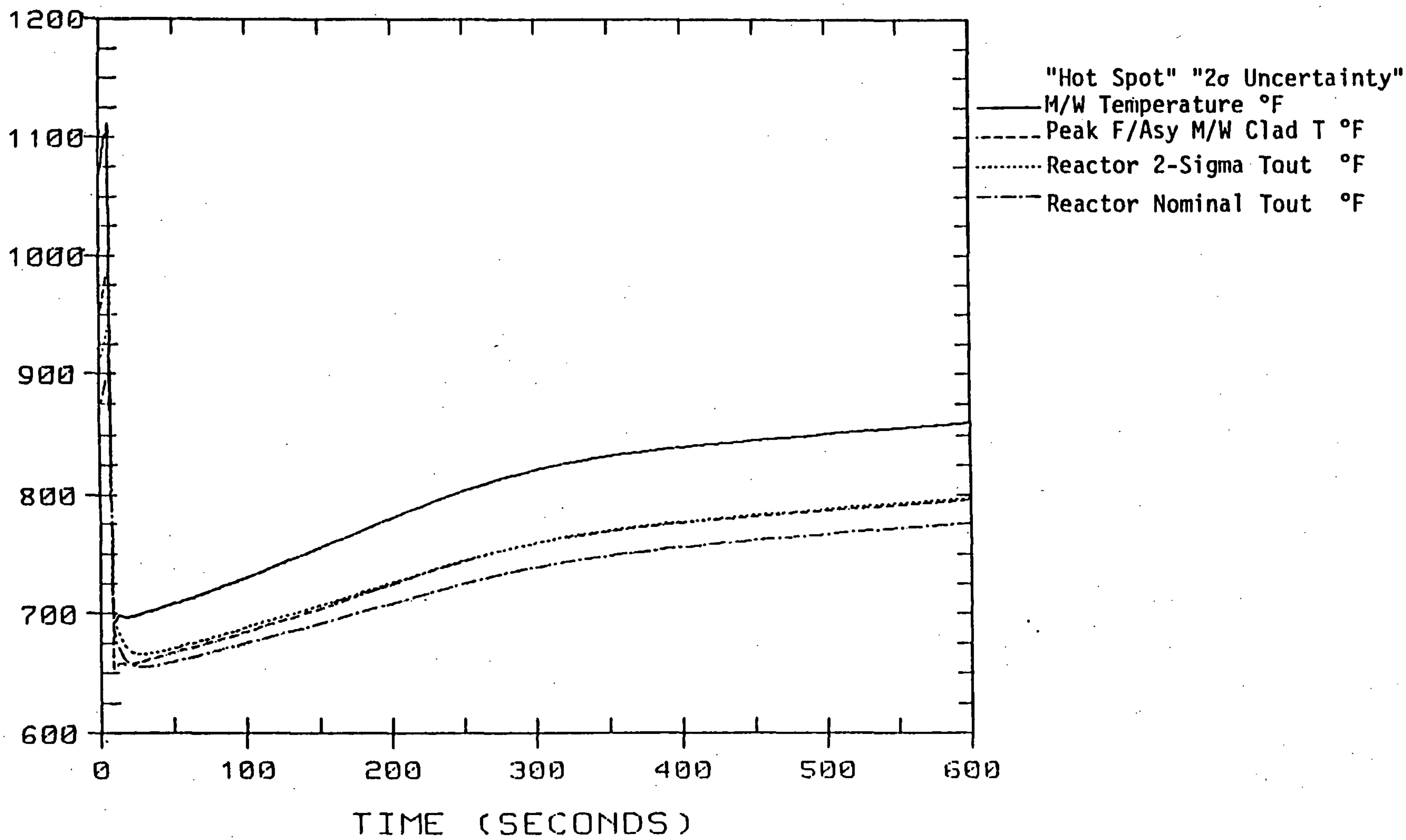

Figure 15.4–3 CLAD AND COOLANT TEMPERATURES SINGLE ROD WITHDRAWAL

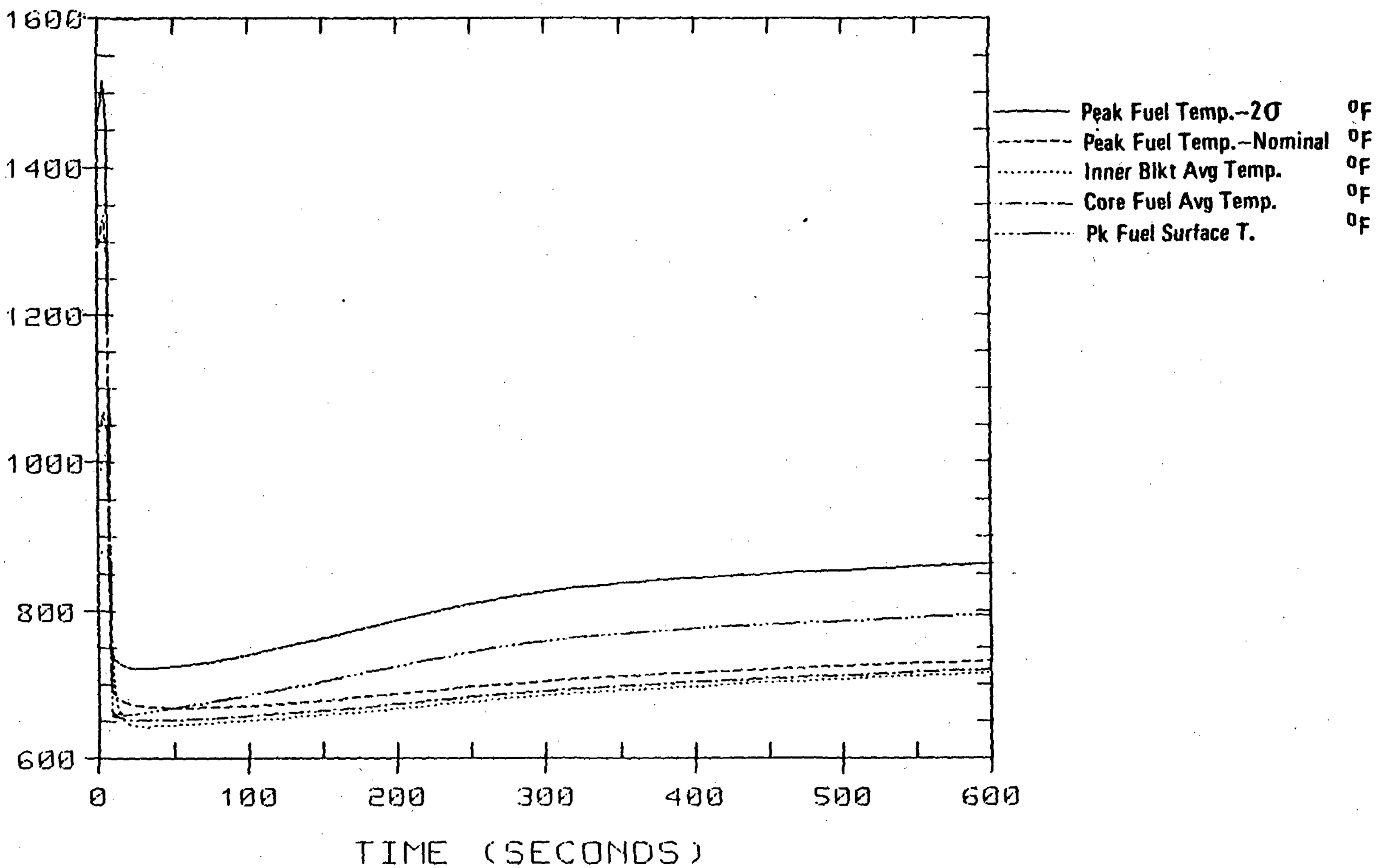

Figure 15.4–4 FUEL TEMPERATURE TRANSIENT SINGLE ROD WITHDRAWAL

15.5 Undercooling DBE's

15.5.1 Loss of Normal Shutdown Cooling

15.5.1.1 Event Description

Normally shutdown heat removal is by condenser cooling. Failing that, shutdown heat is removed from the steam generators by the ACS augmented by some initial steam venting while water is still available. If the water has been lost from the steam generator, ACS will work in conjunction with RVACS to reduce reactor and heat transport component temperatures and cool down the plant. If the ACS is not available, RVACS will remove heat directly from the reactor vessel by natural air circulation flow. The estimated usage frequency of RVACS alone is less than one time per module lifetime.

The RVACS transient is characterized by a reactor trip followed by primary and intermediate pump coastdowns. Sodium flow in the IHTS is assumed to drop to zero in a 2-second time period resulting in no heat removal through the IHX at later times. Subsequent heat removal is by the RVACS only. This transient represents the worst condition for establishing natural circulation through the core since no heat removal takes place in the IHX to aid in establishing natural circulation.

15.5.1.2 Event Analysis

The PRISM thermal-hydraulic model is a relatively simple representation of the reactor system which includes the major elements affecting shutdown heat removal. Figures 15.5-1 and 15.5-2 show sketches of the nodal network developed for this model. The network is dominated by two fluid flow systems: the primary sodium and the RVACS air. Nodes are included to represent the heat capacitance of the various reactor system components and the heat input from the core. The calculated core exit temperature represents the average temperature.

Two base thermal performance cases with different input assumptions, shown in Table 15.5-1, were considered. In the "expected" case, nominal values of the important parameters such as decay heat generation, thermal emissivity, and air-side heat transfer coefficient were used whereas in the "conservative" case, conservative values of these parameters were used. The conservative case analysis result is estimated to provide a 2σ probability level that the reactor temperatures will not exceed the calculated values and are used as a design basis for structural evaluations.

15.5.1.3 Analysis Results

RVACS performance is characterized by the average core outlet temperature during the transient. Figure 15.5-3 shows this function for the expected and conservative cases with RVACS cooling only.

The maximum average core outlet temperature for the conservative case with RVACS only is 1182°F which is less than the reactor vessel design limit of 1200°F for Service Level C. The maximum for the expected case with RVACS cooling only is 1133°F. The time at which the sodium temperature reaches its maximum is approximately 30 hours; and for both cases this corresponds to the time when the RVACS cooling rate becomes greater than the core decay heat rate as shown on Figure 15.5-4.

TABLE 15.5-1

INPUT PARAMETER ASSUMPTION FOR THE EXPECTED AND CONSERVATIVE CASES

Parameter	Expected Case	Conservative Case
Decay Heat Curve(1)	Nominal(2)	Nominal + ~5 percent
Heat Transfer Coefficient	IDS(3)	IDS(3)
Thermal Emissivity	0.77(4)	0.70(4)
Bottom Head Heat Loss	Projected surface area effective	Projected surface area effective

(1) Conservative higher oxide core values.
(2) Calculated for EOEC conditions.
(3) IDS = Interim Decay Storage Test at HEDL.
(4) Value at 1000°F.

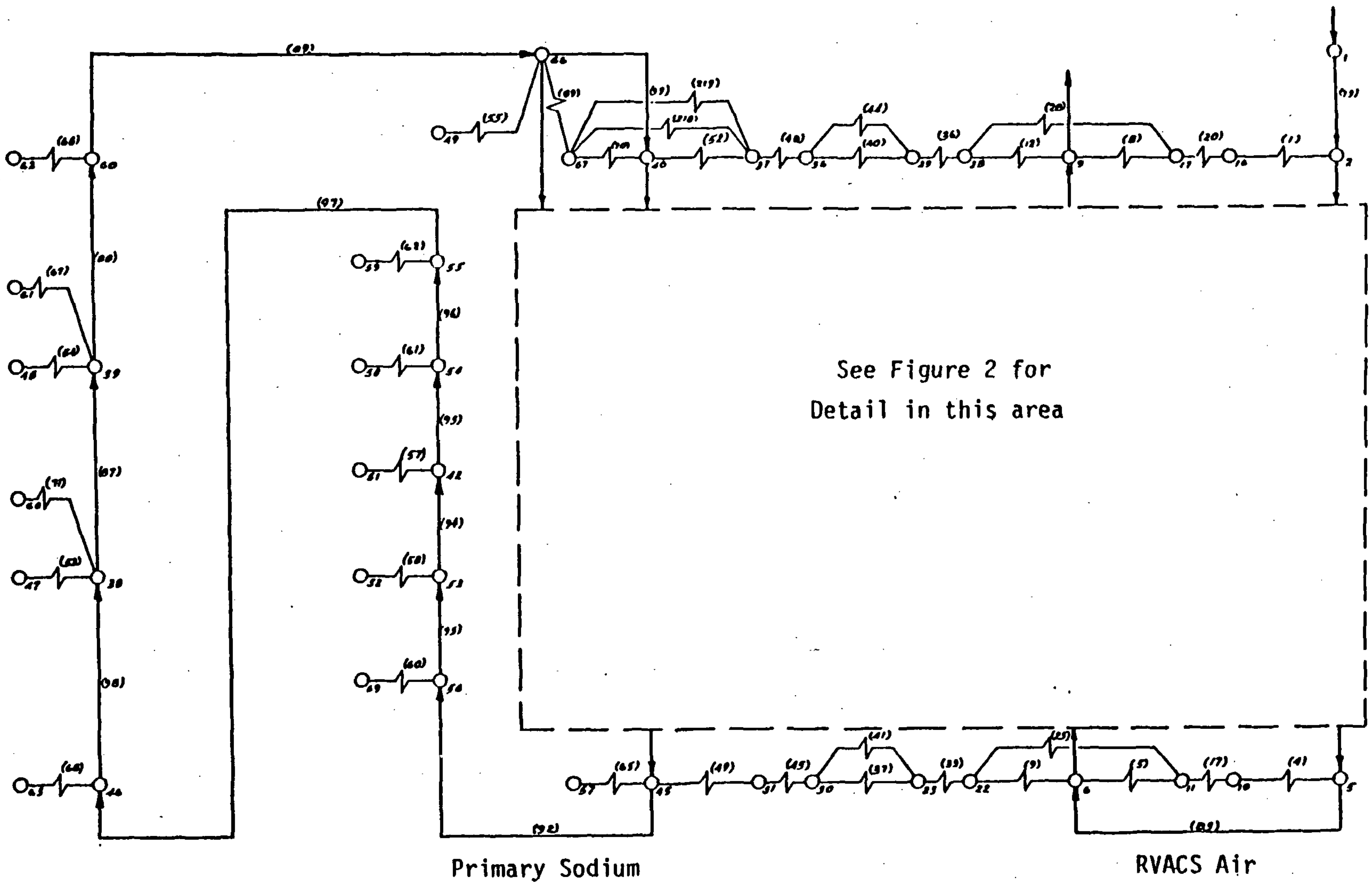

Figure 15.5–1 NODE NETWORK FOR PRISM THERMAL HYDRAULIC MODEL

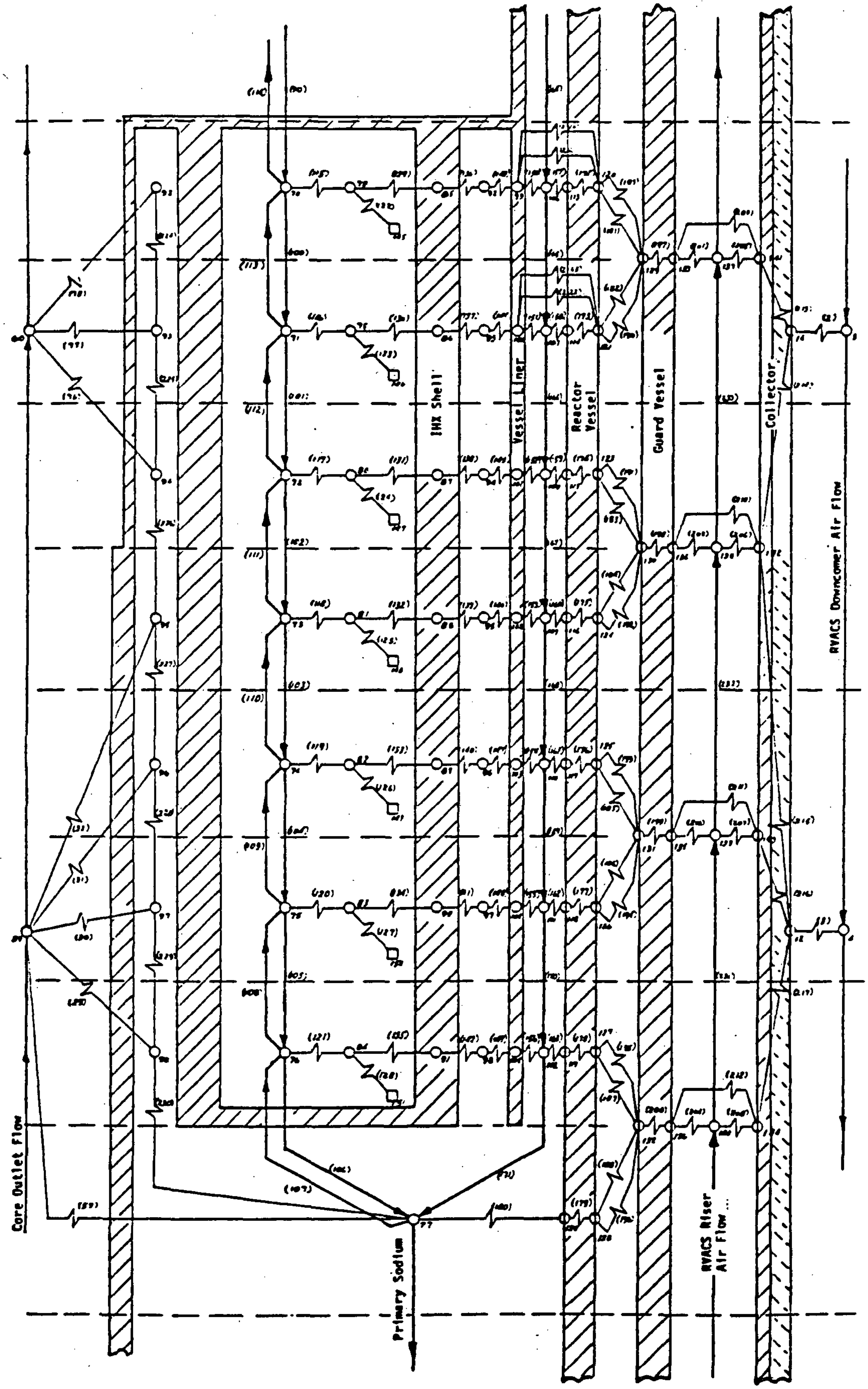

Figure 15.5–2 DETAIL OF NODE NETWORK IN ANNULAR REGION AROUND IHX

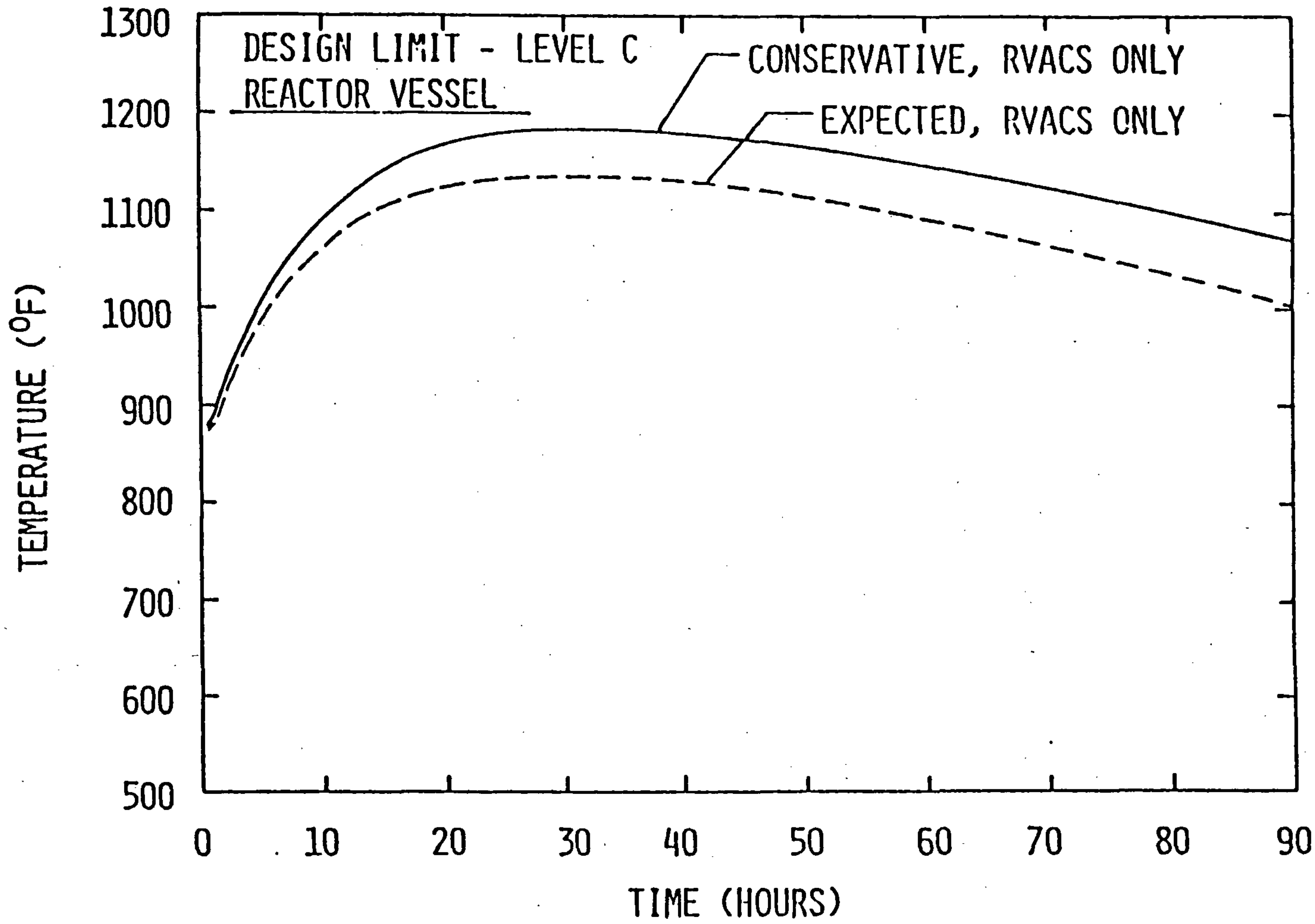

Figure 15.5–3 RVACS PERFORMANCE – AVERAGE CORE OUTLET TEMPERATURE AS A FUNCTION OF TIME

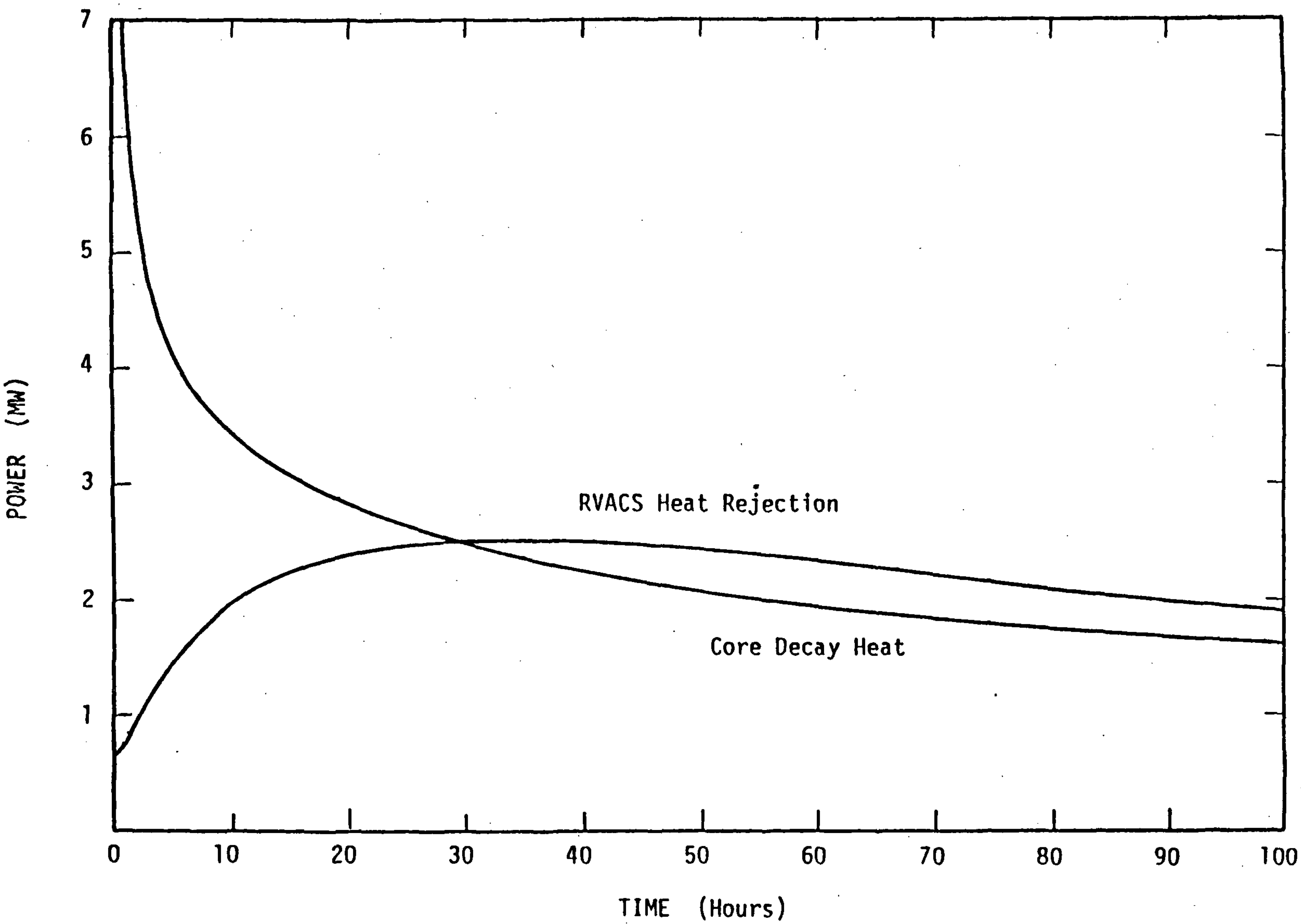

Figure 15.5–4 COMPARISON OF CORE DECAY HEAT TO RVACS COOLING

15.6 Local Fault Tolerance

15.6.1 Introduction

Local faults are fuel failures that result from heat removal imbalance within a single subassembly. Certain features of the PRISM core design serve to prevent such failures. Other features limit the propagation of local faults beyond the immediately affected subassembly. The latter features include the means for detection of local fuel failures and provision for reactor shutdown when the number of local failures reaches a specified limit.

Extensive experience with oxide fuel shows that the rate of failure propagation is slow enough to allow ample time for detection and shutdown. Metal fuel experience gives confidence that its performance with respect to local fault tolerance is superior to that of oxide fuel.

The PRISM metal fuel is expected to be very reliable. The expected value for fuel pin failures in a given core load is less than one. The reactor will be shutdown upon detection of three pin failures. The PRISM fuel failure monitoring system will reliably detect that number of pin failures. This is a conservative limit. The reactor can safely be operated with a greater number of failed pins, based on experience with oxide core design and development and metal fuel operation and testing.

During the licensing process for the Clinch River Breeder Reactor Plant (CRBRP) it was demonstrated for oxide fuel that element to element failure propagation was a very remote event and it was even more difficult to identify a potentially realistic sequence of events which could lead to damage of a neighboring fuel assembly. The Nuclear Regulatory Commission (NRC) staff accepted the arguments presented in the Preliminary Safety Analysis Report (PSAR) and supporting presentations subject to definition of the operating technical specifications which were to be based on the results of the on-going run beyond cladding breach (RBCB) test program.

Although the reference fuel for PRISM is metal fuel, the above position would be the starting point for future licensing activities in this area for the General Electric Power Reactor Inherently Safe Module (PRISM). This section focuses on metal fuel behavior and performance as demonstrated by test and operation in EBR-II. Many of the details of thermal-hydraulics safety disturbances based in the coolant are still applicable to the local faults safety position and are not examined here. Brief descriptions of the plant's systems for failed element monitoring, and cover gas and sodium cleanup are also noted and discussed because of their impact on safety philosophy and potential failure sequences.

15.6.2 Reactor Design

15.6.2.1 Core Design

The PRISM core design is described in Chapter 4. The core and associated systems are designed to prevent mispositioning of core assemblies that could result in abnormal heat generation. Features to prevent mispositioning are:

1. mechanical discriminators
2. identification notches
3. an inventory system
4. low-level range flux monitors

The PRISM pool reactor features that limit the potential blockage of coolant flow to a fuel assembly are:

1. primary pumps without moving parts
2. pump suction from large pool (debris settles)
3. core assembly receptacle multiple flow paths
4. orifice stack plates (filter particles)
5. fuel assembly multiple inlet geometry
6. wire wrapped rod bundle

These features of core design will rely to a large extent on prior reactor experience and will provide confidence that local heat removal imbalance will not result from flow blockage or mispositioning of core assemblies.

15.6.2.2 Fuel Design

The detailed design of the PRISM reactor metallic fuel core uses experience developed at Argonne National Laboratory. The core layout and assembly geometry are described in Chapter 4.

Based on extensive metal fuel operating experience with the driver fuel in EBR-II, metal fuel is expected to have a tolerance to local faults at least equal to or better than oxide fuel. The basis for this is described below as extracted from Ref. 15.6-1. With respect to metal fuel consideration has been given to: 1) fabrication errors, 2) blockages, 3) fission gas release, 4) fuel performance with cladding breaches, and 5) performance during the recent Operational Transient Testing in EBR-II.

The dominant fraction of EBR-II experience is with uranium-fissium alloy. The reasons for expecting similar performance with the U-Pr-Zr alloy are: 1) anticipated similar structural properties, 2) a higher clad-fuel eutectic temperature, and 3) similar fission gas release and fuel swelling characteristics. Before initiating a discussion of metal fuel performance there are several features of metal fuel that have a significant impact upon the excellent tolerance of metal fuel to local fuel failure events. These features are;

1. A high thermal conductivity of metal fuel results in very low fuel centerline temperatures. This also implies reduced hot-spot temperatures for distorted geometries.

2. For EBR-II Mark II fuel elements fabricated with a 75% smear density, the dominant fraction of the generated fission gas is retained within the fuel structure for low burnup fuel elements (less - 2 at %). The fission gas in closed porosity causes volumetric expansion of the fuel. At a volumetric expansion of - 25% a fraction of the induced fission gas porosity becomes open porosity and the generated fission gas is transferred to the fuel element gas plenum.

3. The structural strength of the uranium is very low. At low burnup with closed fission porosity the compression strength of the material at nominal fuel temperatures is dominated by the gas pressure in the closed porosity with high clad loading stresses possible if fuel clad contact occurs. At high burnup, after the formation of open porosity the compressive strength of the fuel is severely reduced and correspondingly the clad loading stresses.

4. The bond sodium is added to the fuel element to provide a low heat transfer resistance path from the fuel to the clad during the low burnup stage when a large gap exists. As the fuel expands as a result of fission gas generation a large fraction of the bond sodium is displaced into the fuel element gas plenum region.

Sections 15.6.4 and 15.6.5 represents a summary of metal fuel tolerance to local failure events. Additional details may be obtained from Ref. 15.6-4.

15.6.3 Failed Fuel Pin Monitoring

The basic components of the failed fuel monitoring system include subsystems to monitor delayed neutrons in the sodium and fission gas activity in the cover gas (see Section 7.6.6). The cover gas monitoring system (CGMS) detects the presence of fission gas from failed fuel elements in the cover gas. The fission gas detection system in the PRISM reference design is not a rapid acting device but is compatible with anticipated failure rates. Under these conditions one failure event is complete before another failure occurs and therefore the CGMS can be used to count fuel element failures. The delayed neutron monitoring system (DNMS) detects neutron-emitting precursors in the sodium coolant and indicates the degree of fuel exposure to the bulk coolant. It therefore provides an indication of failed fuel degradation. The use of several fission counters at each DNMS location permits the extraction of much more information from the coolant. However, due to the variable mixing in the upper plenum, the use of more than one set of DNMS locations is required to achieve full core coverage.

This may permit some limited localization of failures, say to a core quadrant or perhaps better, depending upon plenum fluid hydraulics.

The use of gas tags provides one approach to the economic problem of minimizing down-time for removal of failed elements. The capabilities and limitations of each of these systems is noted in Table 15.6-1.

The purpose of each component of the failed fuel monitoring scheme is shown in Table 15.6-1. The basic purpose of each component can thus be summarized as follows:

1. <u>Cover Gas Monitor</u> - Detection of element failures to allow for determination of number of element failures.

2. <u>Delayed Neutron Monitor</u> - Determination of core status following fuel element failure. A delayed neutron detector cannot count failures.

3. <u>Tag Gas</u> - Tag gases are not added for safety reasons but are sampled only at the time of refueling to allow for determination of the location of failed elements to assist in the removal of the element from the reactor. Engineering judgment indicates that up to five failures can be separated.

The use of a cover gas detector and a delayed neutron detector will allow continuous monitoring of the core status for up to 3-5 pin failures per year. The anticipated failure rate is less than one failure per year. There are still some aspects of fission gas and delayed neutron release behavior which require further study for a sodium bonded fuel element, but are not expected to alter the overall approach to failed fuel monitoring.

15.6.4 Local Heat Removal Imbalance

15.6.4.1 Increased Heat Generation

Enrichment Error

An enrichment error can result in higher than anticipated fuel pin centerline and fuel/clad interface temperatures. Enrichment errors can occur in several ways including: 1) Error in alloying of fuel slugs during fuel reprocessing; and 2) placement of a high enrichment fuel element in a high power zone subassembly. The quality assurance features that can prevent enrichment errors include fuel slug samples, NDE of completed pins using gamma scans or neutron integrator techniques. Subassembly quality control methods during fabrication have been developed and demonstrated on EBR-II and FFTF fuel lines.

The numerous checks and design features reduce the likelihood of enrichment errors. The metal fuel reactor is, however, quite insensitive to enrichment errors. The dominant reason for this is the high thermal conductivity of metal fuel relative to oxide fuel. A large enrichment error (greater than a factor of 2) is required for the fuel centerline to approach the solidus temperature of the fuel. Even larger errors in enrichment are required to raise the clad fuel interface temperature to the fuel clad eutectic temperature.

Thus, enrichment errors for the metal fuel in reprocessing, fabrication, pin placement, and subassembly placement will not have a significant safety impact. Earlier clad failure owing to higher temperatures can be anticipated but with consequences similar to clad failure of nominal fuel elements.

Oversized Fuel

The nominal metal fuel pin is fabricated with a smear density of 75%. The fuel is fabricated by injection casting into glass molds. A potential error to be considered is the use of oversized (larger ID) glass molds resulting in a oversized fuel element. The quality assurance program limits the likelihood of this error; however, the consequences of an error is also very low. A nominal pin has a smear density of 75%; thus the maximum oversized pin that would fit in the clad would contain 33% more material than the nominal pin. The fuel and clad temperatures would be slightly higher but acceptable.

If the available volumetric expansion is less than 25%, the retained fission gas within the fuel pin would result in high clad loading and clad failures similar to that experienced with EBR-II Mark IA fuel. All Mark IA fuel failures were readily detected by the cover gas system and/or the delayed neutron detector system with no observed propagation. Fuel elements with a smear density to - 80% (1.0/0.8 = 1.25) will result in interconnected porosity, low clad loading and the performance characteristic of 75% smear density fuel elements.

15.6.4.2 Reduced Heat Removal

Blockages: Inlet, Exit, Non-Heat and Heat Generating

Modern reactors, including PRISM and EBR-II, have been designed to prevent inlet and exit blockages. An additional feature of a pool reactor, including PRISM and EBR-II, is that the coolant inlet to the pump is from a large pool of sodium with very low coolant velocities. This acts as an effective filter settling and preventing all but the very smallest of particles from being transported to the core.

Bond Defect

The potential bond defects that need to be considered are: 1) Complete failure to add bond sodium to fuel pin, 2) failure of bonding cycles to create complete sodium bond between the fuel and the clad, and 3) early clad failure allowing loss of the bond sodium. The quality assurance program will limit the likelihood of these defects. The likelihood of a fuel pin with a sodium bond defect being inserted into the reactor is very low. However, the consequences of bond defects are also minimal.

Clad failure is unlikely to cause a loss of bond sodium because: 1) the induced fission gas is retained in the fuel matrix until the gap is closed resulting in low fuel element plenum gas pressure, and 2) the static pressure of the primary sodium in the coolant channel is in most cases greater than the gas pressure inside during this period.

Analysis has shown (Ref. 15.6-2) with Mark IA fuel that a sodium bond defect of up to 150° over the full length of the fuel element with the reactor at 1.6 times nominal power can be permitted without fuel melting with argon fill gas. This has been verified with TREAT (Ref. 15.6-3) and EBR-II experiments (Ref. 15.6-2) in which the fuel neither melted or slumped.

Longer term testing in EBR-II was performed with Mark IA element (85% smear density) with bond sodium in the lower half of the element only. The tests (Ref. 15.6-4 & 15.6-5) showed that although some melting occurred this permitted fuel redistribution with periodic contact with the cladding and in situ freezing. Irradiation to 2.2 at.% burnup showed a fuel structure very similar to a nominal EBR-II Mark IA element. We anticipate similar results with the U/Pu/Zr fuel and the 75% smear density fuel.

15.6.5 Local Fault Accommodation

15.6.5.1 Fission Gas Release

Extensive research was performed on the potential for pin to pin propagation as a result of fission gas release during the early seventies for oxide fuel elements (Ref. 15.6-6). The conclusion of this work was that, although random pin failures can be expected to occur considering the large number of elements, rapid and extensive pin-to-pin propagation is unlikely for oxide pins. One reason for this is that the large gap between the fuel and the clad permitting high gas flows occurs only when the driving pressure is low and that when there is a high plenum pressure the gap is closed. A second reason is the good thermal-hydraulic characteristics of the sodium coolant. Both of these conditions exist in the metal fueled core. Furthermore, the high fuel conductivity is a further safety margin whose benefit has not yet been fully evaluated.

In the Mark II elements with Type 316 clad, a large fraction of the failures are in the dimple region (Ref. 15.6-7). No fuel is present in the dimple region. No rapid release of gases has been observed, as well as no evidence of breach propagation. In the ternary alloy metal fuel, the dimple is being eliminated. This will cause the failure location to move into the fuel zone for high burnup fuel elements. Rapid gas release is not expected because of failures in the dimple region but the consequences of a rapid gas release at the dimple region are also benign.

15.6.5.2 Performance of Metal Fuel Following Cladding Failure

Significant information exists concerning the performance of metallic fuel following clad failure. The following is a summary of reactor data for metallic fuel from Ref. 15.6-8.

> "These fuels are obviously compatible with the sodium coolant, and thus the questions regarding RBCB operation are related to breach propagation and movement of fuel from the breach site. To date the reported experience gained with RBCB operation of metallic fuel elements can be narrowed to two subassemblies. The first subassembly was an instrumented subassembly that contained EBR-II Mark IA elements at peak burnups near 3 at .%. Six cladding breaches occurred over a period of about 10 days, with the breaches being separated by at least 12 hours of operation (Ref. 15.6-9). The location and orientation of the breach sites allowed the conclusion that no breach propagation occurred. Fuel was not extruded through the cladding breaches adjacent to the fuel, and thus fuel movement or flow-channel blockage did not occur.
>
> The second subassembly contained EBR-II Mark II elements at peak burnups slightly above 10 at .% (Ref. 15.6-10). Seven elements breached in this subassembly by normal end-of-life breaches over a period of about 12 days. The breaches were all in the restrainer dimple, the expected location of breach for the reference Mark II fuel elements. Again, the location and orientation of the breaches did not suggest a propagation mechanism, and furthermore no fuel movement occurred through the breaches."

An additional noteworthy test involved the irradiation of an exposed fuel pin. The cladding was removed from a 1 inch section of an EBR-II driver fuel element that had been irradiated to 7 at % burnup. The element with the exposed fuel was irradiated for four full-power days. No loss of fuel was detected (Ref. 15.6-11).

The cladding failures in the Mark IA fuel (85% smear density) were in the fuel zone, whereas a large fraction of the failures in the Mark II fuel (75% smear density) were at the dimple. No fuel extrusion at the breach site was observed in the Mark IA fuel breaches; however, they were removed from the reactor shortly after the breaches occurred. These experiments indicate no rapid fuel regress leading to rapid propagation. Long term extrusion of fuel out clad breaches in the fuel zone with Mark IA fuel is likely because insufficient volumetric expansion room was allowed in the elements and the induced fission gas still is in closed porosity. The possibility of extrusion was demonstrated in the experiments in early irradiation tests on Mark IA fuel (Ref. 15.6-12). If a volumetric expansion of ∿25% is allowed, the induced fission gas porosity is connected and much lower clad loading occurs. This was demonstrated with encapsulated Mark II fuel elements irradiated to 6-16 at % burnup in which clad failure occurred in the fuel zone in 20 elements (Ref. 15.6-13). No extrusion of the fuel was observed in the elements which were irradiated for times up to 60 days beyond clad failure. Thus, the extrusion of the fuel is not anticipated for Mark II or ternary alloy fuel elements.

15.6.5.3 Operational Safety

In the last few years an extensive series of transient tests have been performed in EBR-II. Fifty-six low ramp-rate experiments have been performed. In addition, 13 high ramp-rate (16% power increase per second) have been completed. No metallic fuel failure occurred in these experiments. These experiments support the integrity of metallic fuel during operation transients.

References - Section 15.6

15.6-1 R. W. Tilbrook et al., "Local Fault Tolerance of Metal Fuel," ANL-IFR-37, February 1986.

15.6-2 R. R. Smith et al., "The Effects of Driver-Fuel Cladding Defects on the Operation of EBR-II," Argonne National Laboratory Report ANL-77-87, February, 1972.

15.6-3 C. E. Dickerman et al., "TREAT Sodium Loop Experiments on Performance of Unbonded Unirradiated EBR-II Mark I Fuel Elements," Nuclear Engineering and Design, Vol. 12, pp. 381-390 (1970).

15.6-4 J. F. Koenig at al., "Irradiation of Mark IA Fuel Element with Bond Sodium Effects," Argonne National Laboratory Progress Report ANL-RDP-30, July 1974.

15.6-5 J. F. Koenig et al., "Irradiation of Mark IA Fuel Element with Bond Sodium Defects," Argonne National Laboratory Progress Report ANL-RDP-45, November 1975.

15.6-6 J. B. vanErp et al., "Pin-to-Pin Failure Propagation in Liquid-Metal-Cooled Fast Breeder Reactor Fuel Subassemblies," Nuclear Safety, Vol. 16, No. 3, May-June 1975.

15.6-7 B. R. Seidel, "Breach Statistics and Reliability Comparison of Dimple and Plenum-Breach Modes for Mark II Elements Clad with 316 Stainless Steel," Argonne National Laboratory Report ANL-RDP-91, p. 6.10-6.15, January 1980.

15.6-8 L. C. Walters, B. R. Seidel and J. H. Kittel, "Performance of Metallic Fuels in Liquid-Metal Fast Reactors," ANS International Meeting, Washington, DC, November 11-16, 1984.

15.6-9 B. R. Seidel and R. E. Einziger, "In-Reactor Cladding Breach of EBR-II Driver Fuel Elements," Radiation Effects in Breeder Reactor Structural Materials, p. 139, Metallurgical Society of AIME, New York (1977).

15.6-10 R. M. Fryer, E. M. Dean, J. B. D. Lambert, and L. A. Keyes, "Efficacy of the EBR-II FERD System as an Automatic Trip Device," ANL-76-94, Argonne National Laboratory (1977).

15.6-11 W. K. Lehto, "Final Safety Analysis Addendum to Hazard Summary Report, Experimental Reactor II (EBR-II): EBR Operation in Limited Transient Mode," April 1984.

15.6-12 J. H. Kittel et al., "Irradiation Behavior of Uranium Fission Alloys," Argonne National Laboratory, ANL-67-95, October 1971.

15.6-13 G. L. Hoffman, "Irradiation Behavior of Experimental Mark II EBR-II Driver Fuel," Nuclear Technology, Vol. 47, p. 7, 1980.

Table 15.6-1

CAPABILITIES AND LIMITATIONS
OF FAILED FUEL DIAGNOSTIC SYSTEMS

System	Does	Does Not
Cover Gas Monitoring System	Identifies each fuel failure and counts failures.	Does NOT locate NOR monitor status of failure degradation.
Delayed Neutron Monitoring System	Indicates fuel exposure to bulk coolant. May approximate location based on fluid dynamics if more than one DNMS used.	Does NOT identify fuel pin and may not indicate a failure for a considerable time if ever.
DNMS (tri-detector)	Indicates recoil area, neutron age, transit time and some range of source temperature.	Does NOT identify fuel pin and may not indicate a failure for a considerable time if ever.
Gas Tags	Identifies failed element and by administrative procedures locates the failure.	For PRISM using post-shutdown sampling, does not provide any indication of failure, nor indicate fuel status.

15.7 Sodium Spills

15.7.1 Primary Sodium Cold Trap Leak

15.7.1.1 Event Description

One primary sodium service clean up system is permanently installed in each power block (three reactor modules). The sodium service system services one reactor at a time and cannot be activated unless the reactor is shut down for at least three days.

The design basis accident assumes that the entire cold trap primary sodium inventory (1000 gallons) is spilled on the floor of the vault, which contains catch pans to mitigate sodium fires.

The quantity of fuel which circulates in the primary coolant is expected to be of an insignificant magnitude due to the compatibility of the sodium-bonded metal fuel with the coolant. However, the reference approach is to permit operation with as many as two pin failures. Therefore, it is assumed that all of the fission products and 0.01 percent of the transuranics leak into the sodium and become uniformly dispersed in the coolant.

15.7.1.2 Event Analysis

For this accident, a leakage rate from the fuel equal to 1.3×10^{-8} sec^{-1} for iodines and particulates, based on LWR experience, is used.

Primary sodium activity concentrations, at the time of the accident, equal to 4.7×10^{-6} Ci/cc for Na22 and 0.031 Ci/cc for Na24 are used. The sodium activity is calculated by multiplying each concentration by the primary sodium volume (8863 ft∆).

It is assumed that 1000 gallons of primary sodium are in the cold trap when the spill occurs. Since catch pans are available to mitigate the consequences from the sodium fire, approximately 9.75% becomes airborne assuming that 5% burns as an aerosol before it hits the catch pan, 20% of the sodium caught by the catch pan burns, and 25% of the burning sodium becomes airborne.

15.7.1.3 Results

The activity in the cold trap and the activity becoming airborne are shown on Table 15.7-1 and the resultant doses are shown on Table 15.7-2. All doses are well below the 10CFR100 dose limits.

TABLE 15.7-1

ACTIVITY RELEASED FROM A COLD TRAP ACCIDENT

Isotope	Primary Sodium Activity (Curies)	Cold Trap Activity (Curies)	Activity Becoming Airborne (Curies)
NA--22	1.18E+3	1.78E+1	1.73E+0
NA--24	7.78E+6	4.24E+3	4.13E+2
BR--83	2.17E-2	2.81E-13	2.74E-14
BR--84	7.11E-3	0.00E+0	0.00E+0
BR--85	7.28E-4	0.00E+0	0.00E+0
BR--87	2.71E-4	0.00E+0	0.00E+0
BR--88	7.08E-5	0.00E+0	0.00E+0
RB--88	1.09E-2	0.00E+0	0.00E+0
RB--89	1.20E-2	0.00E+0	0.00E+0
RB--90	1.73E-3	0.00E+0	0.00E+0
SR--89	5.78E+1	8.37E-1	8.16E-2
SR--91	6.17E-1	4.81E-5	4.69E-6
SR--92	2.09E-1	3.19E-11	3.11E-12
SR--93	1.15E-2	0.00E+0	0.00E+0
SR--94	1.83E-3	0.00E+0	0.00E+0
SR--95	5.41E-4	0.00E+0	0.00E+0
Y--91M	3.13E-2	3.26E-30	3.18E-31
Y---91	9.08E+1	1.32E+0	1.29E-1
Y---92	2.76E-1	3.15E-9	3.07E-10
Y---93	1.00E+0	1.19E-4	1.16E-5
Y---94	3.23E-2	0.00E+0	0.00E+0
Y---95	1.99E-2	0.00E+0	0.00E+0
ZR--95	1.82E+2	2.66E+0	2.59E-1
ZR--97	2.20E+0	1.73E-3	1.69E-4
NB--95	9.75E+1	1.39E+0	1.35E-1
NB--97	1.58E-1	2.20E-21	2.15E-22
MO--99	9.88E+0	7.00E-2	6.83E-3
TC-101	3.89E-2	0.00E+0	0.00E+0
RU-103	1.62E+2	2.32E+0	2.27E-1
RU-105	5.70E-1	1.13E-7	1.10E-8
RU-106	5.43E+2	8.15E+0	7.95E-1
RH-105	4.54E+0	1.68E-2	1.63E-3
PD-109	5.50E-1	2.02E-4	1.97E-5
AG-111	1.81E+0	2.06E-2	2.01E-3
SB-127	1.32E+0	1.17E-2	1.14E-3
SB-129	1.53E-1	2.81E-8	2.74E-9
TE-127	1.31E-1	9.47E-6	9.24E-7
TE-129	4.02E-2	1.18E-22	1.15E-23
TE131M	4.49E-1	1.28E-3	1.25E-4
TE-131	3.56E-2	0.00E+0	0.00E+0
TE-132	9.99E+0	7.96E-2	7.76E-3

TABLE 15.7-1 (Continued)

ACTIVITY RELEASED FROM A COLD TRAP ACCIDENT

Isotope	Primary Sodium Activity (Curies)	Cold Trap Activity (Curies)	Activity Becoming Airborne (Curies)
TE133M	5.14E-2	2.63E-27	2.56E-28
TE-133	2.06E-2	0.00E+0	0.00E+0
TE-134	8.65E-2	1.03E-34	1.00E-35
I--131	1.87E+1	2.18E-1	2.13E-2
I--132	2.98E-1	1.48E-12	1.44E-13
I--133	3.45E+0	4.79E-3	4.67E-4
I--134	1.53E-1	3.96E-28	3.86E-29
I--135	1.05E+0	8.38E-6	8.17E-7
I--136	1.64E-3	0.00E+0	0.00E+0
I--137	4.30E-4	0.00E+0	0.00E+0
I--138	5.33E-5	0.00E+0	0.00E+0
CS-134	6.30E+1	9.47E-1	9.23E-2
CS-136	1.97E+0	2.53E-2	2.47E-3
CS-137	1.20E+2	1.82E+0	1.77E-1
CS-138	8.16E-2	0.00E+0	0.00E+0
CS-139	2.21E-2	0.00E+0	0.00E+0
BA137M	4.96E-4	0.00E+0	0.00E+0
BA-139	1.97E-1	6.21E-19	6.05E-20
BA-140	4.06E+1	5.20E-1	5.07E-2
BA-141	4.10E-2	0.00E+0	0.00E+0
LA-140	5.41E+0	2.36E-2	2.31E-3
LA-141	5.29E-1	2.22E-8	2.16E-9
LA-142	1.85E-1	2.43E-17	2.37E-18
LA-143	2.54E-2	0.00E+0	0.00E+0
CE-141	1.07E+2	1.51E+0	1.47E-1
CE-143	3.64E+0	1.21E-2	1.18E-3
CE-144	4.30E+2	6.43E+0	6.27E-1
PR-143	3.52E+1	4.55E-1	4.44E-2
PR-144	2.37E-2	0.00E+0	0.00E+0
ND-147	1.44E+1	1.80E-1	1.75E-2
PM-147	1.93E+2	2.90E+0	2.83E-1
PM-148	1.11E+0	1.14E-2	1.11E-3
PM-149	1.89E+0	1.11E-2	1.09E-3
PM-151	6.20E-1	1.61E-3	1.57E-4
SM-153	5.76E-1	2.99E-3	2.91E-4
U--237	1.60E-2	1.78E-4	1.73E-5
NP-238	1.43E-3	8.10E-6	7.89E-7
NP-239	5.90E-1	3.67E-3	3.58E-4
PU-241	1.58E+0	2.38E-2	2.32E-3
CM-242	1.03E-1	1.53E-3	1.49E-4
Total	7.78E+6	4.29E+3	4.18E+2

TABLE 15.7-2

SITE BOUNDARY DOSES IN REM FOR A
PRIMARY COLD TRAP LEAKAGE ACCIDENT

Inhalation Pathways	
- Thyroid	0.19
- Lung	0.69
- Bone	0.31
- Red Bone Marrow	0.095
- Bone Surface	0.29
- Liver	0.74
- Whole Body	0.18
Cloud Immersion Pathways	
- Whole Body	0.36
- Skin	0.044
Whole Body Risk Equivalent Dose	0.69

15.8 Fuel Handling and Storage Accidents

15.8.1 Fuel Transfer Cask Cover Gas Release

15.8.1.1 Event Description

On-site fuel transfer is accomplished within a portable passively cooled cask which is permanently attached to a rail cask transporter. The cask transporter can raise and lower the vertically held cask with its integral gate valve enough to allow it to be sealed to either the reactor vessel fuel transfer port or the adaptor at the fuel cycle facility (FCF). The on-site fuel self propelled cask transporter is moved back and forth on tracks between the reactors and the FCF. The cask is designed to withstand environmental events such as tornado generated missiles and the SSE, therefore, the worst accident involves a leaking cask combined with failed fuel pins which leak fission gases into the cask. A loss of coolant accident is not credible since the three-element cask is passively cooled. However, during transfer, the maximum fuel pin cladding temperature can reach 750°F within the heavily shielded cask.

15.8.1.2 Event Analysis

Three spent subassemblies that have been decaying for one refueling cycle (20 months) are removed from the reactor and placed in the transfer cask. The accident assumes that five fuel pins within the three spent subassemblies fail as their cladding temperature climbs from the 400°F refueling temperature to 750°F in the transfer cask. The failed pins leak their fission gas and volatile inventory into the transfer cask. The cask gate valves fail to seal allowing a leakage from the cask which has become slightly pressurized as its He inventory is heated from ambient to an average temperature of 750°F. Conservatively, a leak of five percent per day is assumed.

15.8.1.3 Results

The resultant activity releases are shown on Table 15.8-1 and the calculated doses are shown on Table 15.8-2. These doses are well within the 10CFR100 dose criteria.

TABLE 15.8-1

ACTIVITY RELEASED FROM A FUEL TRANSFER CASK LEAKAGE ACCIDENT

Isotope	Activity In 5 Failed Pins	0 to 2 Hrs	2 to 8 Hrs	8 to 24 Hrs	1 to 4 Days	4 to 30 Days
KR--85	1.46E+1	6.08E-2	1.81E-1	4.71E-1	1.94+0	8.69+0
I--131	1.24E-19	5.17E-22	1.52E-21	3.80E-21	1.34E-20	2.58E-20
XE-133	2.482E-31	1.03E-33	2.99E-33	7.33E-33	2.39E-32	3.26E-32
TOTAL	1.46E+1	6.08E-2	1.81E-1	4.71E-1	1.94E+0	8.69E+0

TABLE 15.8-2

DOSES IN REM FROM A
FUEL TRANSFER CASK LEAKAGE ACCIDENT

	2 Hr EAB	30 Day LPZ
Inhalation Lung	4.5×10^{-8}	4.3×10^{-8}
Cloud Immersion Pathways		
- Whole Body	2.7×10^{-8}	3.5×10^{-8}
- Skin	2.3×10^{-6}	2.9×10^{-6}
Whole Body Risk Equivalent Dose	3.3×10^{-8}	4.0×10^{-8}

15.9 Other Design Basis Events

15.9.1 Cover Gas Release Accident

15.9.1.1 Event Description

The portable cover gas system services one reactor at a time. In each case the system is connected to the reactor only after the reactor has been shut down and cooled to the refueling temperature (400°F). Its first operation is to remove most of the cover gas from the reactor via vacuum pumps and compressors which transfer the contaminated He cover gas to a portable transfer tank prior to filling the evacuated cover gas space with clean He. This operation loads 98% of the activated cover gas into the portable high pressure storage tank. Following the cover gas evacuation and refilling operation, refueling operations proceed. The activated cover gas is transferred to the fuel cycle facility for processing prior to reuse or release.

15.9.1.2 Event Analysis

The postulated cover gas release accident is the non-mechanistic failure of a pipe or valve such that the radioactive cover gas is released directly to the environment. No operator actions or system functions are assumed.

The plant is assumed to have been operating for 20 months (time between refuelings and cover gas clean-up) prior to the accident at the technical specification limit of two fuel pin failures. It is further assumed that an additional fuel pin fails at shutdown releasing all of its activity. The activity released from the fuel is assumed to be held in the sodium coolant except for the noble gas isotopes. These are assumed to accumulate in the cover gas above the sodium pool. The cover gas system services the subject reactor five days after refueling shutdown.

The equilibrium activity in the cover gas from two failed fuel pins was calculated using the following expression:

$$dA_c/dt = A_f \times L_f - L_d \times A_c$$

where:

A_c = Cover gas activity in Curies
t = Time in seconds
A_f = Core activity for one fuel pin in Curies
L_f = Leakage rate from the fuel to the primary sodium in sec^{-1}
L_d = Isotopic decay constant in sec^{-1}

The time (t) in the above expression is the reactor operating time (20 months). The release rate from the fuel is determined from the core inventory times the leakage rate of the failed fuel.

The leakage rate from the fuel is a function of the fuel quality and the operating history. The leakage rate for noble gases is assumed to be 6.5×10^{-8} sec^{-1}.

In addition to the equilibrium cover gas activity, the activity from one failed pin is also assumed. All particulate activity is assumed to remain in the sodium, and the noble gas inventory is assumed to be released to the cover gas space.

15.9.1.3 Results

The equilibrium cover gas activity, the activity from one additional failed pin and the total activity released from the accident are shown on Table 15.9-1. The resultant exclusion area boundary doses, shown on Table 15.9-2, are well below the 10CFR100 limits.

TABLE 15.9-1

COVER GAS ACTIVITY AND ACTIVITY RELEASED

Isotope	Equilibrium Cover Gas Activity (Curies)	1 Pin Failed At Shutdown (Curies)	Total Activity Released* (Curies)
KR-83M	8.303E-2	1.231E-18	1.232E-18
KR-85M	3.455E-1	9.906E-7	9.936E-7
KR-85	2.079E+1	3.249E+0	2.402E+1
KR-87	1.608E-1	7.373E-27	7.379E-27
KR-88	5.028E-1	6.126E-11	6.138E-11
KR-89	1.001E-2	0.000E+0	0.000E+0
XE-133	1.064E+2	6.460E+2	7.010E+2
XE-135M	4.710E-2	0.000E+0	0.000E+0
XE-135	8.228E+0	1.441E-1	1.450E-1
XE-137	4.511E-2	0.000E+0	0.000E+0
XE-138	1.363E-1	0.000E+0	0.000E+0
TOTAL	1.367E+2	6.494E+2	7.252E+2

* Five days after shutdown

TABLE 15.9-2

SITE BOUNDARY DOSES IN REM
FOR A PORTABLE COVER GAS SYSTEM LEAKAGE ACCIDENT

Inhalation Lung	3.5×10^{-4}
Cloud Immersion Pathways	
- Whole Body	5.7×10^{-3}
- Skin	6.9×10^{-3}
Whole Body Risk Equivalent Dose	5.8×10^{-3}

CHAPTER 16
LIMITING CONDITIONS FOR OPERATION

CHAPTER 16

LIMITING CONDITIONS FOR OPERATION

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Chapter 16 LIMITING CONDITIONS FOR OPERATION

16.1 Reactor Operating Conditions

Applicability

Applies to the reactor core and upper internal structures.

Objective

To assure that core parameters remain within the acceptable range.

Specification

1. The reactor power shall not be allowed to exceed the limiting curve of Figure 16.1-1.

2. The initial core of PRISM shall not be operated with fuel assemblies whose peak burn-up exceeds (TBD) MWD/MT.

3. The reactor shall not be made critical unless each core assembly position is occupied with an assembly which has been tested and approved for proper flow characteristics.

If the reactor is critical and any part of the above specification is not met, it shall be placed in shutdown in an orderly fashion. The reactor shall not be taken critical again until a review has determined that continued operation shall represent no danger to the health and safety of the public.

Basis

By restricting the maximum combination of power and flow given in Specification 1, the plant protection system will be able to mitigate the effects of the normal, upset, and emergency transients described in Appendix D of this PSID.

The peak burnup limit of Specification 2 to ensure fuel cladding integrity. As described in Chapter 15, the fuel cladding integrity is affected by the peak burnup.

Specification 3 is intended to ensure that core coolant flow is not bypassed through an empty grid location or that an assembly with improper flow characteristics is not loaded into the core.

FIGURE 16.1-1

LIMITING CURVE FOR REACTOR POWER

(TBD)

16.2

16.2 Primary Heat Transport System

16.2.1 System Components

Applicability

Applies to the operational limits of the primary heat transport system (PHTS).

Objective

To specify the operational limits of the PHTS components to assure continued power operation of the PHTS over the service life of the plant. The PHTS components are the primary sodium pump and the shell side of the IHX. The definition of the PHTS is extended to also include portions of the reactor module internals that are directly in the primary sodium flow path. These are the hot pool region, the pump discharge manifold and piping, and the core inlet plenum, and the reactor enclosure.

Specification

1. Operational limitations of the PHTS components:

   a. The PHTS shall not be operated at conditions (pressure, temperature and level) exceeding those of Table 16.2-1.

   b. In the event of a PHTS component boundary failure the reactor will be immediately shut down.

   c. In the event of a containment vessel, reactor vessel or reactor closure boundary failure the reactor will be immediately shut down.

2. Following violation of the conditions specified in Table 16.2-1 or indication of a sodium leak from any point in the reactor enclosure or in the PHTS, the following action is required:

a. System and/or component check-out, inspection, and incident evaluation and reporting shall be performed in accordance with approved procedures.

Basis

Specification of the reactor cover gas pressure limit and the maximum sodium level is to ensure that the pressure in the reactor system will at all times be within the specified design limits.

Specification of the minimum sodium level in the reactor system ensures that, in the event of a reactor enclosure leak, the sodium inventory within the reactor vessel will always be above the IHX inlets and that the capability to cool the core and remove decay heat from the system is not challenged.

Specification of the maximum allowable hot pool temperature is intended to assure that the design flow rates through the core are being maintained. It provides a positive indication that the primary flow circuit has not been blocked or interrupted and that the pump flow rates and developed heads are above the allowable minimums.

Specification of the maximum pump discharge pressure assures that the pump control system is functioning properly and there are no blockages in the primary circuit.

Specification of the maximum temperature at the pump inlet assures that the IHX and the IHTS are functioning within design limits in terms of their steady state heat transfer from the reactor system.

Requiring that the reactor be shut down in the event that the above specifications are violated maintains the required level of safety and ensures that the reactor is not operated under possibly ambiguous conditions.

16.2.2 Startup and Shutdown

Applicability

Applies to the operating status of the PHTS during startup and shutdown operations.

Objective

To specify those limiting conditions to ensure continued reliable cooling of the reactor core and to limit potential radioactivity releases from the primary sodium system during plant startup and shutdown operations.

Specification

1. The reactor shall not be made critical unless the primary sodium system has been filled with sodium coolant to the normal level.

2. The maximum heat transport system heatup/cooldown rate between refueling and 550°F shall not exceed an average of 50°F/hr.

3. The maximum rate of change of the temperature of the primary heat transport system hot leg shall not exceed an average of 180°F/hr between hot standby and 25% thermal power operating conditions.

Basis

The precautions listed in Specification 1 ensure adequate sodium inventory for reactor core cooling and to reduce the possibility of anomalous reactivity fluctuations due to gas entrainment.

Specifications 2 and 3 ensure that the heat transport system piping and component structural design heatup/cooldown rate limits are not exceeded.

16.2.3 Cover Gas Activity

Applicability

Applies to the maximum concentration of radioisotopes in the reactor cover gas.

Objective

To specify the limiting concentration of radioisotopes in the cover gas for continued reactor operation.

Specification

The radioactive inventory of the reactor cover gas shall not exceed (TBD) Ci. If this limit is exceeded, an orderly shutdown of the module shall be initiated within (TBD) hours after this has been determined.

Basis

The specification is designed to limit the radioactive inventory to be no greater than used in the Chapter 15 analyses.

16.2.4 Impurities in Reactor Coolant

Applicability

Applies to the sodium purity requirements for the primary heat transport system (PHTS).

Specification

1. The PHTS shall be normally operated with the plugging temperature at least 50^{o}F below the temperature of the coldest part of the sodium system.

2. The plugging temperature shall not exceed $400^{o}F$ when any part of the heat transport system is above $600^{o}F$.

If the above specifications are not met, or cannot be complied with by the corrective action delineated in the appropriate operating manuals in 24 hours, an orderly shutdown of the plant shall be initiated.

Basis

To ensure reliable operation of the PHTS to prevent the plugging of system components, and to minimize corrosion.

TABLE 16.2-1

LIMITING CONDITIONS FOR PHTS NORMAL OPERATION

COMPONENT OR REGION	LIMITING CONDITION
Reactor Cover Gas	20.0 psig max
Hot Pool Sodium	900^{o}F max
	8' 11" $\pm$ 6" Sodium Level
Pump Discharge	150 psig max 630^{o}F max

16.3

16.3 Intermediate Heat Transport System

16.3.1 System Components

Applicability

Applies to the intermediate heat transport system (IHTS) which connects the steam generator system (SGS) to the intermediate heat exchanger (IHX).

Objective

To specify the operational limitation of the IHTS components to assure continued power operation of the IHTS over the service life of the plant.

Specification

1. The argon cover gas pressure in the intermediate sodium pump and in the intermediate sodium tank shall not exceed 50 psig.

2. The IHTS temperatures and pressure, as determined at the various instrumented locations, shall be maintained at or below the values shown in Table 16.3-1.

3. The intermediate heat exchanger must be maintained with a positive intermediate-to-primary pressure differential.

If any of the above specifications are not met, or cannot be complied with by the corrective action delineated in the appropriate operating manuals, an orderly shutdown of the plant shall be initiated. Follow-up action such as system/component check-out, inspection and incident evaluation shall be performed in accordance with the approved procedures.

Basis

The maximum argon cover gas pressure of Specification 1 combined with the pump head shall not be allowed to exceed the structural design limit of 300 psig. Limiting the cover gas pressure to 50 psig provides this assurance with a suitable margin.

The values in Table 16.3-1 of Specification 2 represent the structural design parameters of the IHTS.

Specification 3 ensures that radioactive sodium does not enter the IHTS from the primary system.

16.3.2 Sodium Water Reaction Pressure Relief Subsystem

Applicability

Applies to the sodium water reaction pressure relief subsystem (SWRPRS) which is part of the IHTS.

Objective

To assure overpressure protection for the IHTS intermediate heat exchanger, and sodium side of the steam generator system, and to limit the consequences of a sodium-water reaction by removing the sodium reaction products, water, and steam from the affected components.

Specification

Any time there is water/steam on the tube side of the steam generator and sodium on the shell side of the steam generator, the SWRPRS and sodium leak detection subsystem (2 out of 3) shall be operational.

If the above specification is not met, or cannot be complied with by corrective action within 4 hours, an orderly shutdown of the plant shall be initiated, and the IHTS and/or SGS, shall be placed in a condition such as to prevent a sodium-water reaction.

Basis

During all modes of module operation, the sodium side pressure relief systems must be fully operable. These systems are required to limit the consequences of a water-to-sodium leak in the SGS.

16.3.3 Impurities in Intermediate Coolant

Applicability

Applies to sodium purity requirements for the intermediate heat transport system (IHTS).

Objective

To specify the sodium purity requirements for operating the IHTS.

Specification

1. The IHTS shall be normally operated with the plugging temperature at least 50°F below the temperature of the coldest part of the sodium system.

2. The plugging temperature shall not exceed 350°F when the temperature of any part of the heat transport system is above 600°F.

If the above specifications are not met, or cannot be complied with by corrective action delineated in the appropriate operating manuals in 48 hours, an orderly shutdown of the plant shall be initiated.

Basis

To ensure reliable operation of the IHTS during high temperature operation and prevent the plugging of system compounds.

TABLE 16.3-1

INTERMEDIATE HEAT TRANSPORT SYSTEM LIMITING

TEMPERATURES AND PRESSURES

System/Section/Component	Pressure (psig)	Temperature (^oF)
Hot Leg Piping	300	850
Cold Leg Piping	300	650
IHX Tubes	300	900
Flowmeter	300	650
Pump	300	650
Expansion Tank	300	650

16.4

16.4 Steam Generator System

Applicability

Applies to the steam generator system (SGS) which provides independent steam generation capability for PRISM module.

Objective

To assure reliable and adequate cooling to maintain the IHTS sodium cold leg temperature at a value which will assure proper core cooling.

Specification

1. During operation, the SGS cooling system shall be operable.

2. During reactor power operation, the water level in the steam drums shall not be below (TBD) inches.

3. During reactor power operation, all power/safety relief valves on the SGS shall be operable for the SGS circuits operating in conjunction with PHTS and IHTS loops.

   If the above specifications are not met, or cannot be complied with by corrective action with (TBD) hours, an orderly shutdown of the plant shall be initiated, and the module shall be placed in shutdown condition in (TBD) hours.

4. During module shutdown, the recirculation water temperature in its SGS loop shall not drop below (TBD) ^{o}F.

If Specification 4 is not met, immediate corrective action shall be taken to bring the plant within this specification within 24 hours.

Basis

During plant operation, the single non-safety-related IHTS/SG systems connected to the reactor module must be fully functional.

To assure adequate operation, the water level in the steam drum will not be below (TBD) inches. If the water level drops below the low limit, there is a possibility that the steam drum may dry out. This event could result in loss of the steam generator heat removal capability and a thermal shock.

All power/safety relief valves included in the SGS must be operable to provide adequate relief during overpressurization.

During module shutdowns for periods longer than about five hours, the reactor decay heat is transferred to the atmosphere by RVACS to avoid cooling the primary and secondary sodium below their plugging temperature.

16.5

16.5 Sodium Purification System

Applicability

Applies to radioactivity limits for operation of the sodium purification system.

Objective

To define radioactivity limits for normal operation and maintenance of the intermediate sodium processing and primary sodium processing subsystems.

Specification

1. The activity of the intermediate sodium processing subsystem shall not exceed the following:

   a. total plutonium activity - (TBD) curies

   b. total gross - activity - (TBD) curies

2. The activity of the primary sodium processing subsystem shall not exceed the following:

   a. total plutonium activity - (TBD) curies

   b. total gross - activity - (TBD) curies

Bases

The bases for these specifications are the events analyzed in Chapter 15. In all cases, the limits are in compliance with 10CFR100.

16.6 Inert Gas Receiving and Distribution System

16.6.1 Purity of Gas

Applicability

Applies to the purity of helium, argon and nitrogen.

Objective

To define the minimum allowable purity for each of the inert gases.

Specification

1. The minimum purities in the inert gas receiving and distribution system are:

   a. Helium - 99.9945% (by volume)

   b. Argon - 99.996% (by volume)

   c. Nitrogen - 99.998% (by volume)

2. The minimum purities in the inert gases for continued operation are:

   a. Helium - (TBD)

   b. Argon - (TBD)

   c. Nitrogen - (TBD)

Basis

Both specifications are designed to ensure that inert gas properties are within the values assumed for design.

16.6.2 Cell Atmosphere-Oxygen Control

Applicability

Applies to the fuel handling cell.

Objective

To assure that accident design limits in inerted cells are not exceeded in the event of a large sodium spill because of a high oxygen concentration in the cell atmosphere.

Specification

1. If the oxygen level in the inerted cell atmosphere is greater than 2% or less than 0.5%, corrective action shall be implemented to bring the level to within the specification.

2. If, after (TBD) hours of corrective action, the oxygen level in the inerted cells is not within specification, an orderly isolation, drain, or cooldown of alkali metal inventory in the cell shall be initiated.

Basis

The upper limit of 2% oxygen is based on the allowable level developed in the accidents analyzed in Chapter 15. The lower level of 0.5% is established to prevent nitriding.

16.7 Residual Heat Transport System

16.7.1 Reactor Vessel Auxiliary Cooling System

Applicability

Applies to the operation of the reactor vessel auxiliary cooling system (RVACS).

Objective

To provide adequate long term removal of reactor decay and sensible heat following reactor shutdown when the normal heat rejection path through the steam generator and heat rejection through the secondary auxiliary cooling system are inoperable.

Specification

If the temperatures in the RVACS ducts exceeds the limiting curve of Figure 16.7-1 during periods of hot standby or refueling shutdown, all suspect RVACS components shall be examined and evaluated for suitability for return to power operation.

Basis

Since RVACS is passive and operates continuously (functions at its intended high heat removal rate only, when all other reactor heat removal systems are inoperative), no specification is required for its actuation. However, the limiting curve of Figure 16.7-1 is necessary to ensure the temperature limits for the RVACS components have not been exceeded. Component temperatures are a function of duct temperature and air flow rate.

16.7.2 Auxiliary Cooling System

Applicability

Applies to operation of the auxiliary cooling system (ACS).

Objective

Assure adequate redundancy and diversity of shutdown heat removal system in support of plant availability.

Specification

The reactor shall not be operated at temperatures above 550°F unless the ACS is available.

Basis

The ACS is provided to improve plant availability by shortening the time (approximately 25 to 5 days) required to cool the plant down to a level which will allow refilling the steam generator and plant re-start.

FIGURE 16.7-1

LIMITING CURVE FOR RVACS OPERATION

(TBD)

16.8 Containment Integrity

Applicability

Applies to the limiting conditions under which containment integrity* can be violated.

Objective

To define the status of the containment required to ensure no undue risk to the health and safety of the public.

Specification

Containment integrity shall be maintained unless the reactor is sub-critical by at least (TBD) Δ k/k, and there is no possibility of a primary sodium fire.

Basis

The circumstances under which a violation of containment is permissible are chosen such that the remaining provisions available to prevent a release of radioactivity can be relied upon to perform their function. Thus, by maintaining the reactor in a shutdown condition, the control system will provide sufficient assurance that excessive radioactivity releases can be prevented during refueling or component (i.e., EM pump or CRDM drive line) replacement operations which involve shielded transfer casks and single gate valves. The value of (TBD) Δ k/k is consistent with the disucssion in Section 16.10.

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* Containment integrity is defined as the condition when all isolation valves to other systems are operable, or secured in the closed position or isolated by closed manual valves or flanges.

16.9 Reactor Protection System

Applicability

Applies to the equipment included as part of the reactor protection system (RPS) for each PRISM module.

Objective

To assure operability of the RPS.

Specification

During all operations requiring RPS action, the following conditions for operability of the RPS shall be met:

1. At least 3 instrument channels* of each subsystem shall be operational. If one channel is inoperative, its voter output shall be in the tripped state.

2. Where maintenance and/or calibration disrupts the capability of a channel to initiate trip, its voter output shall be placed in the tripped state.

Basis

For all operating conditions, the RPS provides sufficient redundancy to tolerate a single failure without affecting the ability of the RPS to initiate appropriate protective action. Specifications 1 and 2 assure that suitable redundancy is preserved even if single element failures occur during test operations. Since certain bypasses are provided for refueling operations, which are not automatically taken out, it is necessary to assure that these bypasses are configured properly for on-line operations.

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* Each channel consists of a sensor, data processor and voter.

16.10 Refueling

Applicability

Applies to the limiting conditions for operation of the reactor refueling system (RRS) equipment and facilities, and to refueling operations.

Objective

To ensure that during refueling operations, core reactivity is within controlled limits and to ensure that the release of radioactivity from the containment or RSB in the event of a fuel handling accident is within the limits of 10CFR20 and 10CFR100.

Specification

1. Each irradiated fuel assembly shall be stored in the reactor vessel until the calculated decay heat is no greater than 1.9KW.

2. The following conditions shall be met before initiating refueling operations involving the reactor.

   a. The reactor shall be maintained in the refueling shutdown condition.

   b. The primary pump main circuit breakers shall be racked out and tagged.

   c. During any movement of fuel within the core, a licensed operator shall be available to monitor the activities.

   d. All refueling system equipment required for the refueling operations shall be checked out and verified to be operational.

e. The control rod drive mechanisms shall be disconnected from the control assemblies and the rotatable plug raised. Prior to movement of the large rotatable plug, a verification shall be made that all control rods are disconnected from their drive line assemblies.

f. The reactor cover gas activity shall be less than (TBD) Ci/cc.

g. The reactor core gas pressure shall be maintained at atmospheric pressure or less when the cap is removed from fuel transfer port.

h. The IVTM limit switch which precludes premature release of fuel and blanket assemblies shall be set less than (TBD) inches above the fully seated position.

If any of the above specified limiting conditions are not met, the refueling shall not be initiated.

3. The following conditions shall be met during refueling operations involving the reactor.

a. Communication links between the plant control center, at the IVTM control console, and the reactor servicing coordination center, shall exist whenever changes in core geometry or fuel transfers are taking place.

b. All three source range flux monitor (SRFM) channels shall be operating with any fuel assemblies in the core. If any one of the channels fails, operations in progress to transfer fuel into or out of the reactor core shall be stopped or reversed to place the reactor in a safe hold point configuration until the defective channel is restored to operation.

The SRFM system trip points will be set at signal levels equivalent to a subcriticality of (TBD) for the first core and (TBD) for the equilibrium core. If the trip points are exceeded, the refueling operation must be stopped immediately and a determination made as to the cause of the reactivity anomaly.

c. During refueling operations, not more than two vacant positions in the core may exist at any one time. These vacant positions may not be adjacent to each other.

If any of the specified limiting conditions for refueling are not met, refueling shall cease until the specified limits are met, and no operations will be initiated which may increase the reactivity of the core beyond the reactivity resulting from normal temperature fluctuations within the refueling temperature dead band.

4. Following refueling operations involving the reactor, the following conditions shall be met prior to reactor startup.

   a. The reactor rotatable plug shall be secured and its drive power sources physically disconnected.

   b. The fuel transfer port shall be capped and leak tested.

Basis

Specification 1 ensures that passive cooling by the fuel transfer cask is adequate to meet cladding temperature limits for fuel assmeblies for an indefinite period of time. Control rods and blanket assemblies can be transported without delayed storage because of their low decay power.

Immmediately prior to refueling, Specification 2 lists the conditions which must be satisfied. Item (a) is based on permissible core shutdown levels. Item (b) is written to prevent the operation of the primary pumps during refueling, and Item (c) is intended to assure that proper supervision will exist during movement of fuel within the core. Items (d) and (e) are written to prevent unexpected movement of core components during refueling which could affect core reactivity. Items (f) and (g) are intended to control the release of radioactivity. The level specified in Item (f) is based on the premise that if this amount of activity was all released instantaneously to the atmosphere, the radiation dose at the site boundary would not exceed the limits of 10CFR20 (Annual) and the airborne radiation dose in the refueling enclosure would be below the quarterly 10CFR20 limits for restricted areas. Item (h) is intended to prevent dropping of a core assembly or insertion of a core assembly into an incorrect position.

Specification 3 establishes the control of the operation during refueling. During any subcritical operation other than the intentional approach to the critical, the SRFM must provide a warning to the operator and thereby assure that the reactor does not approach criticality any closer than that level from which criticality could be attained by the worst single refueling error with adequate margin for the associated uncertainties. The minimum shutdown reactivity requirement during refueling is based on this criterion. An alarm will sound in the control center if the minimum shutdown requirement, described above, is violated.

Shuffling of blanket assemblies cannot be done without temporarily leaving open two core positions. If two adjacent core assemblies are removed, the resulting misalignment could exceed the design value, so that a new core assembly or an assembly to be reinserted could either not be inserted or be inserted in the wrong position. Item (c) of Specification 3 is written to prevent this event. Note, however, that shuffling is not part of the current fuel management scheme, but is only a capability provided for any future fuel management scheme.

Specification 4 assures that modifications made to accommodate the refueling are corrected before reactor startup.

16.11 Effluent Release

16.11.1 Liquid Waste

Applicability

Applies to the liquid radioactive effluents from the radioactive waste system to the environment.

Objective

To assure that liquid radioactive material released to the environment is kept as low as practicable and, in any event, is within the limits of 10CFR20.

Specification

1. If the experienced release of radioactive materials in the liquid waste, within a calendar quarter period, is such that these quantities, if continued for a year, would exceed twice the design objectives, the following actions will be taken:

   a. An investigation shall be made to identify the causes for such releases.

   b. A program shall be defined and initiated to reduce such releases to within the design values.

2. The release rate of radioactive materials in liquid waste from the plant shall be controlled, by in-line monitoring, such that the concentration in the cooling tower blowdown will not exceed the concentrations specified in 10CFR20.106.

3. All radioactivity liquid effluents released from the plant shall be reported to the NRC.

Basis

Liquid effluent release rate will be controlled in terms of the concentration in the discharge tunnel containing cooling tower blowdown. This basis assures that even if a person obtained all of his daily water from such a source, the resultant dose would not exceed that specified in 10CFR20. Since no such use of the discharge tunnel is made and considerable natural dilution occurs prior to any location where such water usage could occur, this assures that offsite doses from this source will be far less than the limits specified in 10CFR20.

In addition to the sampling and analysis of eah batch prior to discharge, a radiation monitor on the radioactive waste discharge line and a sampler in the discharge tunnel give further assurance that annual average discharge concentration is kept within the specified limits.

16.11.2 Gaseous Waste

Applicability

Applies to the release of radioactive gaseous effluents from design release points.

Objective

To assure that the amount of radioactivity released as low as is reasonably achievable and will result in site boundary doses which are below 10CFR50, Appendix I limits.

Specification

1. Radioactive gases released from design release points shall be continuously monitored and/or sampled such that the total release can be quantified.

2. The effluent monitor for undefined mixtures from the exhaust of radwaste area of the reactor service building shall be operable and capable of alarming when radioactivity is detected at a level corresponding to (TBD) percent of the maximum permissible radionuclide concentrations given in 10CFR20.

3. The effluent monitor for undefined mixtures from the reactor exhaust of each head access area (HAA) shall be operable and capable of alarming when radioactivity is detected at a level corresponding to (TBD) percent of the maximum permissible radionuclide concentrations given in 10CFR20.

4. The effluent monitor for undefined mixtures from the exhaust of the fuel cycle facility (FCF) shall be operable and capable of alarming when radioactivity is detected at a level corresponding to (TBD) percent of the maximum permissible radionuclide concentrations given in 10CFR20.

5. The effluent monitor for undefined mixtures from the exhaust of each turbine generator building shall be operable and capable of alarming when tritium activity is detected at a level corresponding to (TBD) percent of the maximum permissible concentration given in 10CFR20 for unrestricted areas.

6. In the event of an alarm due to high radioactivity in the effluent of a design discharge point, appropriate action will be taken.

7. If an effluent monitor is inoperable, appropriate action will be initiated and be in effect until the monitor is restored to operational status.

8. If the quantities of radioactive material released during any semi-annual period are significantly above design objectives, the following action will be taken:

   a. Make an investigation to identify the causes of such releases.

   b. Define and initiate a program of corrective action.

<u>Basis</u>

Dose rate estimates have been made for the PRISM design release points for off-normal occurrences. Based on these calculations, release of activity at the alarm limits will result in an off-site annual dose rate which will not exceed (TBD) mr/yr, well below 10CFR20 limits.

16.11.3 HVAC and Radioactive Effluents

<u>Applicability</u>

Applies to the release of radioactive effluents through the HVAC exhausts.

<u>Objective</u>

To assure that radioactivity released to the environment is kept as low as practiable and, in any event, is within the limits of 10CFR20 guidelines.

To assure that the release of radioactivity to unrestricted areas meet the "as low as practicable" concept, the following design objective applies:

The release rate of radioactive isotopes, averaged over a yearly interval except for halogens and particulate radioisotopes with half-lives greater than eight days, discharged from the plant, should not exceed:

$$\sum_i \frac{Q_i}{(MPC)_i} \leq \text{(TBD)}$$

where Q_i is the annual average release rate (Ci/sec) of radioisotope i and $(MPC)_i$ in Ci/cc is defined for isotope i in column 1, Table II of Appendix B to 10CFR20.

Specification

1. The instantaneous release rate of radioactive isotopes, discharged from the plant, shall not exceed:

$$\sum_i \frac{Q_i}{(MPC)_i} \leq \text{(TBD)}$$

where Q_i and $(MPC)_i$ are as defined above.

2. The gaseous and particulate activity of the potentially contaminated HVAC discharge paths shall be monitored and recorded along with the corresponding effluent flow rates.

3. Radiation monitors as required in Specification 2 above shall be operable and capable of detecting a composite radioactivity release rate less than the design objective rate.

4. Whenever any of the radiation monitors are inoperable, grab samples shall be taken in the affected discharge path and analyzed.

5. When the annual projected release rate of radioactivity, averaged over a calendar quarter, exceeds the annual objective, corrective action shall be taken to reduce such release rates to below the objective rate and/or orderly shutdown of the reactor shall be initiated.

6. When the instantaneous release rate or radioactivity exceeds twice the design objective rate, the licensee shall identify the cause of such release rates, initiate action to reduce such release rates to below the objective rate.

Basis

The specifications provide reasonable assurance that the resulting annual exposure rate from noble gases at any location at the site boundary will not exceed (TBD) millirems per year. At the same time, these specifications permit the flexibility of operation, under unusual operating conditions, which may temporarily result in releases higher than the design levels but well below the concentration limits of 10CFR20.

The release rate stated in the objective sets the concentration of radioisotopes, except for halogens and particulate radioisotopes with half-lives greater than 8 days, at less than (TBD) of 10CFR20 requirements at the site boundary (<10 mrem per year).

Specification 1 requires the licensee to limit the release of all radioisotopes such that concentrations at the site boundary are less than the levels specified in 10CFR20.

Specifications 2 through 4 require that suitable equipment to monitor radioactive releases are operating during any period these releases are taking place.

Specification 3 establishes an upper limit for the quarterly average release rate for noble gases equal to the annual design rate. The intent of this specification is to permit the licensee the flexibility of operation under unusual operating conditions which may result in short-term release higher than the annual objective rate.

Specification 4 requires the licensee to initiate action to reduce instantaneous release rates to the annual design level whenever the measured release rate exceeds twice the annual design rate. The intent of this specification is to require the licensee to control and report short-term releases that exceed the annual design rate.

16.12 Reactivity and Control Rod Limits

16.12.1 Shutdown Reactivity

Applicability

Applies to the minimum control rod reactivity worth of the control rod system.

Objective

To ensure reactor shutdown from any operating power condition to zero power following reactor scram.

Specification

The control rod bank insertion limit is (TBD). If this limit is not met, an orderly shutdown of the plant shall be initiated.

Basis

The control rod bank limit assures sufficient worth at all times in the reactor cycle, assuming the failure of any single active component (i.e., a stuck rod), to shutdown the reactor from any operating condition to zero power and to maintain shutdown over the full range of design coolant temperatures. Allowance has been made for the maximum reactivity fault associated with any anticipated occurrence.

The reactivity fault allowance is included in the requirements of the control rod system. The maximum reactivity fault is postulated to occur upon the accidental uncontrolled withdrawal (not ejection) of the highest worth control rod in the reactor from its banked position.

The control system worth is being designed such that a single rod scram from the normal hot, full power condition with the remaining rods unmoved, is sufficient to achieve cold, zero-power critical condition.

16.12.2 Rod Axial Misalignment Limitations

Applicability

Applies to the limits on the deviation of an individual control rod in a bank from the average bank position.

Objective

To ensure that the minimum scram performance requirements are met and to prevent distortions in the core power distributions due to the axial misalignment of control rods in a bank.

Specification

If any operable control rod is axially misaligned from its bank, as indicated by the control rod position displays, by more than 1.5 inches, the PCS shall automatically initiate a reactor module shutdown using a controlled reduction of reactor power.

Basis

The rod axial misalignment specification is intended to preclude operation with instruments, logic, or control drives which are exhibiting detectable degradation of performance. Though undesirable and inefficient, with maldistribution of power and an inoperative control drive, this system will not damage or endanger public health or safety.

16.12.3 Inoperable Rod Position Indicator

Applicability

Applies to the rod position indicating system.

Objective

To provide indication of rod position to the operator during plant operations.

Specification

During operation of the reactor, either the absolute or the relative rod positions indication system for each rod that is maneuvered during operation must be operational. Failure of both systems requires reactor shutdown (not scram). Restart can be undertaken only after the absolute rod position indication system is restored to operational status.

Basis

Rod position indication is required to provide information on correct banking of the control rods. Correct banking assures that the appropriate scram reactivity characteristics are met. The rod position indication systems provide the basic input to this banking determination. Sustained operation with both relative and absolute position indication systems inoperable for any rod that is to be maneuvered is not permissible.

16.12.4 Inoperable Rod Limitations

Applicability

Applies to the limits of operation for an inoperable control rod.

Objective

To assure safe shutdown and control capability at all times for the reactor.

Specification

1. A rod is defined to be inoperable if, in the course of normal operations, the rod fails to respond normally to a design command.

2. If the inoperable rod is located within 0.5 inches of the remaining rods, corrective action shall be taken to determine the cause of the malfunction and correct it. If after (TBD) hours, the inoperable rod has not been restored to an operating status, an orderly shutdown of the reactor shall be initiated.

Basis

Operation of the reactor with a rod within 0.5 inches of the average bank insertion does not compromise the operational capability of the reactor during a scram.

For a rod inoperable at a greater misalignment, there is a local and general power maldistribution effect. Since a single rod is capable of shutting down the reactor, the control system has the capability to safely shut down the reactor with a single stuck rod, plus an inoperable rod. However, this capability is provided to accommodate the unexpected event, and is not intended as an operating condition. Time is provided to repair an inoperable rod condition to avoid unnecessary plant shutdown. However, if the condition cannot be relieved promptly, the plant must be shut down.

16.12.5 Rod Drop Time

Applicability

Applies to all control rods at all operating temperatures.

Objective

To assure prompt operation of all control rods.

Specification

For all operating temperatures and flow rates, the drop time of each control rod shall be less than 2 seconds from tripping of the reactor protection system (RPS) logic to dashpot or damper entry.

Basis

The allowable control rod insertion times from start of rod motion for all operating conditions are presented in Chapter 4 and are consistent with safe operation of the plant. The delay between tripping of RPS logic and start of rod motion is required to be less than 0.1 seconds, consistent with plant investment protection.

This requirement represents practical achievable insertion times which do not approach allowed damage threshold, to assure that the allowable damage severity limits are not exceeded for any design basis event. Iterative transient evaluations have led to the specified minimum insertion rates.

This requirement is to be satisfied under all potential control rod positions within the design limits established and within worst case positional uncertainties for banked control rods. The delay time of 0.3 sec. is specified for consistency with the insertion speeds. Potential tradeoffs between the delay time and insertion speed requirements may be made while assuring that the overall insertion speed requirements may be made while assuring that the overall insertion speed requirements are met. This specification is not intended to require rod drop testing during power operation.

CHAPTER 17
QUALITY ASSURANCE

CHAPTER 17

QUALITY ASSURANCE

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Chapter 17 QUALITY ASSURANCE

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REFERENCES - Chapter 17

17.1-1 "Nuclear Energy Business Group BWR Quality Assurance Program Description," NEDO-11209-04A, Revision 5, March 1985.

APPENDIX A
PROBABILISTIC RISK ASSESSMENT

APPENDIX A

PRISM PRELIMINARY
PROBABILISTIC RISK ASSESSMENT

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A 1.0

A1.0 INTRODUCTION

One of the design requirements of the PRISM plant is to apply probabilistic risk assessment (PRA) to the design process. Specifically, it is required that:

(1) PRA techniques shall be applied to the design process to ensure that public health and safety risk, including that due to beyond design basis accidents (BDBA) is acceptably low; and

(2) Numerical risk limits shall be used to guide judgment of the design adequacy with respect to public risk.

The numerical measures of risk adopted to carry out these requirements have been derived from the NRC safety goal policy statement (Ref. A1-1). The risk measures are given in Table A1-1.

Consistent with the PRISM design requirements and the intent of the above NRC policy statement, this preliminary probabilistic risk assessment has been conducted with the following objectives:

(1) To evaluate the extent to which the PRISM power plant meets the quantitative goals of Table A1-1

(2) To delineate system relationships which must be understood for risk management. This includes:

(a) Identifying major contributors to risk.

(b) Estimating the sensitivity of the risk to uncertainty in input data.

(c) Characterizing the radioactivity release patterns to asses potential for post-accident risk management.

It should be noted that this PRA analysis is a part of an iterative process involving the interaction between design and PRA activities. It is the objective of PRA activities to seek accurate estimates of probabilities and consequences. However, conservative assumptions had to be used in this analysis when necessary to expedite the feedback of PRA analysis to the designers. Using more realistic assumptions and developing more firm design and data base could very well show that the results in this appendix are unduly conservative.

The scope of this PRA study is defined in Section A1.1. Section A1.2 describes how the study was organized between GE and other DOE contractors. Section A1.3 highlights the contents of the remainder of this appendix.

A1.1 Scope

The PRISM power plant analyzed in this study is the reference (metal-core) design described in the main body of this PSID. The plant has been assumed to be located on a GESSAR-II site. The study has been confined to the following scope:

(1) The study does not include risks from acts of sabotage or normal plant effluent releases.

(2) The study has been confined to accidents in a single module. The module affected is assumed to be operating at full power when an accident is assumed to occur. In particular, the study has not considered startup accidents, partial power operation, or situations where one module in the same power block is out for refueling or for other reasons.

(3) The study has been confined to core-related accidents. In particular, accidents related to radioactivity sources outside the reactor vessel, e.g., radwaste systems, have been excluded. However, the radioactivity sources in this study include:

(a) driver fuel;

(b) inner and radial blankets;

(c) activated primary sodium;

(d) spent fuel stored in-vessel.

To simplify the study, end of equilibrium cycle radioactivity inventory has been conservatively assumed at the time of accident.

A1.2 Study Organization

This PRA has been developed in accordance with the following guidelines:

(1) To use state-of-the-art methods and data.

(2) To use the mean values as estimates for the risk measures.

(3) To incorporate uncertainties in important phenomena, key assumptions, and input data in the risk assessment.

Towards these objectives, the General Electric Company (GE) has sought and obtained (as much as possible) analyses and experts' opinions from other DOE contractors, namely, Argonne National Laboratory (ANL), Westinghouse Hanford Company (WHC), and Sandia National Laboratory. These organizations have conducted the following tasks:

(1) General Electric:

- Study organization and coordination.

- Overall risk model development including the definition of initiating events, and event trees describing potential accident scenarios.

- Reliability assessment of plant systems.

- Accident analysis involving reactor transients.

- Assignment of probabilities of accident scenarios events based on input from GE and other organizations.

- Estimation of the risk measures using the RISKSP code.

- Ranking of contributors to the risk.

(2) Argonne National Laboratory:

- Accident analyses involving the reactor core to assess core damage categories.

- Provide experts' opinion on core damage categories for accident scenarios not analyzed.

- Provide experts' judgment on the probability of various core accident events.

- Provide experts' judgment on fuel debris coolability, vessel integrity, accidents involving energetics, fuel and fission product retention in vessel and ex-vessel, and fission product release mechanisms.

(3) Westinghouse Hanford Company:

- Accident analyses involving the primary Na coolant and damaged core to assess release categories of radioactive material to the environment.

- Provide experts' judgment on release categories for accident scenarios not analyzed.

- Provide experts' judgment on the prabability of various release events.

- Provide experts opinion on post accident heat removal, timing of melt-through for vessel and in-vessel structures, mechanisms for radioactive material release and attenuation, Na boiling and burning.

(4) Sandia National Laboratory:

- Consequence analyses to assess acute and latent fatalities for various radionuclide releases and emergency plan assumptions.

- Provide experts' judgment on consequences for release cases not analyzed.

- Provide experts' opinion on the sensitivity of consequences to to release and emergency plan parameters.

A1.3 Appendix Organization

Section A2 presents the risk results and compares them to the NRC safety goals. The section also presents the lessons learned from this PRA which could be useful for design and operation trade offs.

Section A3 provides a summary of the risk assessment model and quantification procedures. This section also discusses the particular characteristics of the PRISM plant which have a significant effect on the risk model structure and results.

Section A4 contains detailed assessment of the initiating events, system response event trees, core response event tres, vessel and containment response event trees, institutional decisions, and public consequences. The rationale for defining the events in the risk model, procedures, data, data sources, and results of assessing the probabilities of these events are discussed in this section.

REFERENCES - SECTION A1.0

A1-1 "Safety Goals for the Operations of Nuclear Power Plants, Policy Statement" 28044 Federal Register, Vol. 51, No. 149, Monday, Aug. 4, 1986.

TABLE A1-1

SAFETY GOALS

Safety Measure	Goal Safety Measure Must Not Exceed
1. <u>Individual Risk:</u> Probability of prompt fatality (per one year of a nuclear plant operation) for an average individual residing within one mile from the plant site boundary.	5×10^{-7}
2. <u>Societal Risk:</u> Probability of cancer fatality (per one year of nuclear plant operation) for population residing within 10 miles of the plant site.	1.9×10^{-6}

A 2.0

A2.0 SUMMARY AND CONCLUSIONS

1. Individual and societal risks have been evaluated using the risk model and quantification procedures summarized in Section A3 and discussed in detail in Section A4. The risk model contains an exhaustive set of accident sequences which may lead to radioactive material release from a PRISM module. Cases with and without evacuation of the population around the site have been assessed.

2. The estimated individual and societal risk measures are presented in Table A2-1. The table shows that the risk from a PRISM module is substantially less than the NRC goal. Specifically,

   a) The societal risk (probability of latent cancer fatality) is less than the NRC goal by a factor of 200,000 with evacuation and a factor of 146,000 without evacuation.

   b) The individual risk (probability of prompt fatality) is negligible with evacuation. Without evacuation, the individual risk is less than the NRC goal by a factor 5,400.

3. The societal and individual risks are dominated by the following accident sequences.

   a) A large earthquake (>0.825 g ground acceleration) which results in a reactivity insertion due to core compaction and relative core-control rod motion, and causes failure of the reactor shutdown system and flow coastdown system. This sequence leads to an energetic core disruption and subsequent release of radioactivity. This accident sequence accounts for 48% of the societal risk and 49% of the individual risk.

   b) A large earthquake (>0.825 g ground acceleration) which results in a reactivity insertion and failure of the reactor shutdown system as above, and causes in-vessel structural damage which prevents proper thermal expansion which nominally provides the inherent reactivity feedback. This sequence leads also to an energetic core disruption and subsequent release of radioactivity.

This accident sequence accounts for 35% of the societal and individual risks.

c) A failure of one or two primary electromagnetic pumps accompanied by failure of the shutdown system in such a way that credit of the control rod thermal expansion cannot be relied upon as an inherent reactivity feedback mechanism. This sequence may lead to an energetic core disruption and subsequent release of radioactivity. This accident sequence accounts for 16% of the societal risk and 11% of the individual risk.

d) A large earthquake (>0.825 g ground acceleration) which causes failure of the seismic isolators and subsequent reactivity insertion, loss of the shutdown heat removal system, and loss of the reactor shutdown system. This sequence leads to an energetic core disruption and subsequent release of radioactivity. This accident sequence accounts for 4% of the individual risk but a negligible fraction of the societal risk.

4. The PRISM risk is of such small magnitude that it is dominated only by the residue of structural failures and severe accidents which have extremely low probability of occurrence. This is attributed to the safety philosophy of the PRISM reactor, which resulted in:

a) limited hazard potential due to the small-size reactor core, small control rod reactivity worth, and seismic isolation;

b) highly reliable systems for control of power, flow, and heat removal, with very little reliance on active systems for safe shutdown;

c) limited radioactivity release potential due to inherent safety characteristics and the large thermal capacity and low pressure of the primary coolant.

5. Although conservative assumptions have been used in assigning the probability of structural failures and paths leading to severe accidents, further analysis is required in the following areas to develop an information base for a more realistic assessment:

a) The risk model should be expanded to include a detailed systematic analysis of man/machine interactions following the occurrence of an initiating event. In particular, assurance that potential accident paths have been conservatively accounted for will be enhanced with explicit modeling of the effect of accident sequences on the operator's cognitive behavior and on the capability of post-accident monitoring and recovery.

b) Detailed failure modes and effects analysis (FMEA) of the reactor core, in-vessel structures, reactor and guard vessels, and other structures is recommended to uncover potential paths for loss of the inherent reactivity feedback features, loss of the heat removal functions, and dependent failures.

c) Fragility analysis is required to assess the probability of the critical failure modes identified in the above FMEA. Seismic analysis to assess the probability of failure propagation and combinations which may lead to loss of the shutdown heat removal function is also required.

d) Man-structure interface during manufacturing, repair, inspection, and operation, and the quality assurance program for these operations should be analyzed to assess the possibility of structural defects which may propagate to serious failures due to applied stresses or man-structure interaction.

e) Detailed common cause failure analysis is needed to replace the conservative beta factor approach used in the PRA and to identify types of dependencies which may be removed by design or operating procedures.

TABLE A2-1

PUBLIC RISK FROM THE OPERATION
OF A PRISM MODULE

Risk Measure	NRC Goal (less than)	PRISM Performance With Evacuation	PRISM Performance Without Evacuation
Societal Risk (probability of latent cancer fatality per one year of operation, 0-10 mi)	1.9×10^{-6}	9.0×10^{-12}	1.2×10^{-11}
Individual Risk (probability of prompt fatality per one year of operation, 0-1 mi)	5×10^{-7}	$<10^{-13}$	2.7×10^{-10}

A 3.0

A3.0 APPROACH OVERVIEW

This section presents a summary of the PRA approach and the quantification procedures and data bases. The procedures and actual data used in the assessment are discussed in detail in Section A4.

The risk from operating a nuclear power plant results from sequences of events which lead to the release of radioactive material to the environment. The definition of these events and the extent to which each event is analyzed could significantly affect the accuracy of the risk results. In principle, it is desirable to use a fine classification of these events if the risk contribution is significant or uncertainty in the risk contribution is large. Conversely, events which have insignificant impact on the risk or lead to comparable risk contribution may be grouped without much loss of accuracy. The net result of applying these principles is a risk model which highlights major risk contributors with minimum uncertainties introduced by inadequate event definitions.

Section A3.1 presents the specific characteristics of the PRISM power plant which have been considered for event definition in the risk model. Section A3.2 provides a summary of the risk model structure. Section A3.3 presents the procedures and data sources used for quantifying the risk.

A3.1 Nature of the PRISM Risk

The PRISM power plant has distinctive features for preventing accidents, for limiting the extent and speed of accident progression should an accident occur, and for retaining the fission products. As discussed below, the effect of these features on the probability and consequences of accidents has been to reduce the relative significance of independent failures and slow transients. Consequently, the risk model developed for this PRA highlights dependent, concurrent, and coherent failures.

A3.1.1 Accident Prevention

The PRISM design places strong emphasis on the reliability principles of redundancy (and diversity), testing, use of passive concepts for power control and heat removal, and fail-safe or self-correcting failure provisions. The following examples illustrate this emphasis:

1. <u>Redundancy and Diversity</u>

   The PRISM Reactor Shutdown System (RSS) uses six control rods, although one rod is adequate to shut down the reactor. Another example of utilizing redundancy is the use of quadruply redundant channels for process data handling and transmission. The application of diversity is illustrated by the use of in-vessel instruments which measure different process parameters, are placed in different locations, and are exposed to different environments.

2. <u>Testing</u>

   The PRISM reactor uses continuous monitoring of its Reactor Protection Systems (RPS) channels. Continuous monitoring is also used at the interface of the RPS and other systems, e.g., Plant Control System (PCS). The PRISM reactor also uses frequent testing by operation of some of the critical components, such as the control rod drive motors and control rods.

3. <u>Use of Passive Concepts for Power Control and Heat Removal</u>

   An example of the first concept is the thermal expansion of the control rod drivelines and core load pads in response to increase in primary coolant temperature. The reactivity resulting from this expansion would offset the positive reactivity resulting from coolant density changes caused by coolant heating. This results in a net negative temperature coefficient of reactivity for the primary coolant temperature. An example of the second concept is the Reactor Vessel

Auxiliary Cooling System (RVACS) which removes decay heat by natural convection.

4. <u>Fail-Safe and Self-Correcting Failure Provisions</u>

These provisions utilize two different principles to respond to a failure.

a. Transfer to a more reliable or at least as reliable configuration. An example of applying this principle is the fault-tolerant quad-redundant logic used in the PRISM RPS. The system uses a 2-out-of-3 logic when all channels are operable (one channel on rotating standby), transfers to the equally reliable configuration of 2-out-of-3 logic when one channel fails (the failed channel is excluded until repaired), and transfers to the more reliable configuration of 1-out-of-2 logic when two channels fail (the failed channels are excluded until repaired).

b. Safe transfer to desired state with appropriate use of stored energy. An example of applying this principle is the use of stored energy to control flow coastdown.

The above characteristics have resulted in high reliabilities of the PRISM systems. In particular, the probability of independent failures under nominal design conditions has been estimated to be extremely low. This raises the relative importance of dependent failures and of operating conditions outside the design envelope as risk contributors.

A3.1.2 Limiting Extent and Speed of Accident Progression

Threshold phenomena such as fuel melting, coolant boiling, clad rupture and structural failure could significantly impact the course of an accident and the resulting consequences. The phenomena may result from excessive energy generation or inappropriate energy distribution between different parts of the system. In particular, when the rate of energy transfer into a part of the system exceeds the rate at which it is transferred out, the deposited energy may result in melting, boiling, creep rupture,...etc. Two types of system time parameters characterize these rates: 1) the reactor period, or equivalently the net reactivity rate and magnitude, which characterizes the rate of nuclear energy generation, and 2) the time constant of the fuel, clad, core, coolant,...etc., which characterizes the rate of heat transfer out.

The PRISM design uses the following provisions to maintain a long reactor period (slow rate of energy generation):

1) small reactivity additions (~20¢) if a control rod is inadvertently withdrawn;

2) inherent negative reactivity feedback if the fuel or the primary coolant temperature increases.

The reference PRISM core uses a U-Pu-Zr metal fuel with HT9 cladding. The core time constant is characteristically short (~0.3 sec or ~10% of a typical oxide core). This results in a fast heat transfer from the fuel to the coolant, or equivalently, the deposition of only a small fraction of the heat in the fuel with a corresponding low fuel temperature. Despite this advantage, the possibility of threshold phenomena cannot be excluded due to the following conditions:

1) relatively low melting point of the metal fuel (~1150°C)

2) formation of fuel/clad eutectic alloy which may begin at ~725°C

3) primary sodium boiling (~880°C) or voiding due to the fast rate of energy transfer in.

The threshold temperatures shown above indicate the potential vulnerability of cladding being the first to fail as a result of eutectic formation. This is different from typical accident scenarios in LMRS using oxide fuels. In such cases, unprotected loss of flow accidents (LOF) lead to sodium voiding, followed by clad melting, then fuel melting. On the other hand, unprotected transient over power (TOP) accidents lead to the opposite order of fuel melting, then clad melting.

Sweepout of the molten clad from the core in the above oxide-core scenarios results in positive reactivity additions. On the other hand, fuel/clad eutectic melt sweepout of the metal core could lead to negative reactivity additions and shutdown due to fuel removal from the core region. In this regard, then, the eutectic formation can be viewed as another mechanism for limiting the extent of accident progression.

The above characteristics reduce the relative importance of slow transients as public risk contributors, or equivalently, highlight the relative importance of fast transients involving rapid reactivity additions or rapid loss of flow as potential risk contributors.

A3.1.3 Radioactive Material Retention

The reference PRISM metal fuel has the following characteristics which could significantly affect the timing and mix of radioactive material releases in case of an accident.

1) For burnups greater than 2 at %, the fuel is virtually transparent to fission gas, i.e., the fission gas generated by fission is transmitted to the fission gas plenum at the rate of generation.

2) Accidents may be terminated at relatively low temperatures ($\leq$1150°C) involving molten fuel/clad eutectic alloy or molten fuel. The low temperature presents a strong potential for retention of strontium (Sr) and tellurium (Te) isotopes (which may be major risk contributors in other fuels) in the fuel body. Therefore, if the fuel is retained inside the vessel, these radionuclides will not be released.

These considerations have been reflected in the definitions of accident scenarios and release categories of the risk model described below.

A3.2 Risk Model

The risk analysis of a PRISM module starts with the initial condition that the module is operating at full power. An event for which the reactor would be or should be shut down is assumed to occur. Such an event is called an initiating event (IE). In response to the initiating event, the module is expected to control the nuclear power generation, coolant flow, and heat removal processes to bring the reactor to a safe shutdown until the cause of shutdown is removed. In the course of this transition to shutdown or following the nuclear shutdown, imbalance between these processes may occur. If such imbalances do not cause clad or core damage, the module resumes operation after the cause of shutdown is removed. Otherwise, the situation is termed an "accident", and the module is assumed not to resume operation.

The risk model defines the events, event sequences or scenarios, and the statistical relationships and dependencies between them, which are required for estimating the probabilities and consequences. An overview of the risk model structure is delineated in Figure A3.2.1. The structure contains the following major elements:

1) Initiating Events: Twenty-one mutually exclusive and collectively exhaustive events have been defined in the risk model. These include normal shutdown for refueling, spurious shutdown signal, forced shutdown, malfunctions leading to three ranges of reactivity additions, partial or complete loss of forced flow, partial or complete loss of heat removal capability, station blackout, partial core blockage, core support and vessel failures, and three levels of earthquake events.

2) System Event Sequences and Accident Types: For each initiating event, a system event tree is developed to identify possible sequences which lead to safe shutdown and those which lead to accidents. The event trees include response of the power control systems (plant control systems [PCS], reactor protection system [RPS], reactor shutdown system [RSS] and inherent reactivity

feedback features), flow control systems (pump trip and flow coastdown system), and heat removal systems (via balance of plant [BOP], intermediate heat exchanger [IHX], or RVACS). The sequences are formed from possible combinations of success and failure of the various systems. Each sequence ends either with a safe shutdown condition or one of 23 accident types. The accident types cover loss of the shutdown heat removal system after neutronic shutdown, four levels of severity of transient overpower without scram, two levels of severity of loss of flow without scram, two levels of severity of loss of heat sink without scram, and combinations of the above.

3) Core Response Event Trees and Core Damage Categories.

For each accident type, a core response event tree is developed to identify the possible core scenarios until neutronic shutdown is accomplished. The event trees include reactivity feedback mechanisms for intact fuel (Doppler, thermal expansion), molten fuel and clad motion (with and without eutectic formation), and sodium voiding. The mechanisms may be adequate to cause neutronic shutdown with no further damage.

On the other extreme, they may enhance energetic events before shutdown. Each scenario formed from possible combinations of reactivity feedback and energetic events leads to one of 12 core damage categories. The categories cover the spectrum of possible fractions and types of fission products released from the core, damage to the vessel seal as a result of accident energetics, location and coolability of fuel debris formed, if any, coolant enthalpy at the time of neutronic shutdown, and the states of the shutdown heat removal system.

4) Containment Response Event Trees and Radionuclide Release Categories

For each core damage category, a containment response event tree is developed to identify possible radioactive material transport scenarios until a stable configuration is reached. The event trees include events relevant to long term coolability, timing of vessel failure, delayed energetics resulting from potential recriticality, and radioactive material release timing and paths. Each scenario results either in complete retention of radionuclides within the reactor vessel indefinitely, or in one of nine categories of radionuclide release to the environment. The release categories are characterized by the fractions of different groups of radionuclides released as a function of time, from the incipience of the accident until a stable end state is reached.

5) Institutional Response and Consequence Types

For each radionuclide release category, the likely institutional responses in terms of the timing for evacuation and evacuation effectiveness are determined. Four types of consequences are then evaluated for each category of radionuclide release; latent and early fatalities given the above institutional responses, and latent and early fatalities assuming complete institutional failure.

The risk model combines the above elements probabilistically with proper accounting for dependencies between the events and event sequences. With the use of proper probability and consequence values, the model produces the following risk measures for one year of reactor operation: 1) probability of early fatality to an individual within a one mile radius from site, and 2) probability of latent fatality to an individual within a 10 mile radius from site. The procedures used for quantifying the probabilities and consequences are summarized below.

A3.3 Quantification Procedures

A summary of the procedures and data sources used for quantification is presented in Table A3.2.1. A brief discussion is presented below. The detailed procedures, data sources, and data values used are contained in Section A4.

1) Initiating Events

   The list of initiating events for FY85 PRISM design has been updated to reflect changes in the design. The expected frequencies of normal shutdown for refueling and forced shutdown have been updated to reflect new operation ground rules. The expected rate of reactivity faults and component failures have been updated by incorporating recently-developed reliability analyses. Data sources for the probability estimates include Nuclear Plant Reliability Data System (NPRDS), Clinch River Plant Risk Assessment (CRPRA), and GESSAR site seismic frequency curves.

2) System Event Trees

   The conditional probabilities of failure of the systems in these trees, given each initiating event, have been estimated using fault trees, reliability block diagrams, the FRANCALCI computer code, and dependency analysis. The estimates are based on appropriate component failure modes, testing, and repair. Dependency analyses used include analysis of the functional dependence of a component on its interfacing components and environment, fragility analysis of components under seismic events, and use of Beta factors which express the conditional probability of multiple component failures given that one component has been found in a failed state. Data sources used in the above analysis include NPRDS, CRPRA, the Reactor Safety Study (WASH-1400), generic fragility data, and engineering judgment.

3) Core Response Event Trees

The conditional probabilities of events in these trees, given each accident type, have been analyzed by judgment based on ANL analysis using the SASSYS computer code, GE analysis using the ARIES code, and bounding analysis for reactivity feedback assessments. Fuel clad eutectic formation and fission product release during normal operation have been based on test results reported by ANL. The types and amounts of fission products released in-vessel under accident conditions have been estimated by adjusting the release fractions used in the Reactor Safety Study to account for the low temperatures of eutectic formation and fuel melting in the PRISM metal core.

4) Containment Response Event Trees

The conditional probabilities of events in these trees, given each core damage category, have been estimated based on ANL assessment of fuel debris coolability and late energetics due to recriticality, on the reliability of the shutdown heat removal system to continue operation until accident conditions are removed, and on bounding calculations for the reactor vessel creep rupture under accident conditions. The cumulative fractions of the core radioactive material inventory released as a function of time has been estimated by HEDL for each accident scenario using a thermal analysis computer code. The thermal model used accounts for decay heat, sodium concrete reaction, and sodium fire in air when these conditions are present. Rates of energy generation, leakage, and attenuation of radioactive materials were assigned by judgment based on test results and analysis of similar accident situations.

5) Consequences

Early and late facilities, given each release category, have been calculated by Sandia using the MACCS computer code. Cases with and without evacuation were run. Sensitivity analyses to evaluate the importance of fuel/concrete reactions and accident mitigation were also investigated. All calculations used NUREG-1150 assumptions, e.g., shielding factors and relocation criteria. Fifty-four radioisotopes were used for the analysis. The inventory of each isotope was estimated by the ORIGEN computer code for the FY86 PRISM reference metal core. The population distribution and meteorological data for the GESSAR site were used.

6) Risk Estimation

The RISKSP computer code was used to estimate the risk measures defined in the NRC safety goal policy statement, cumulative probability distributions of early and latent fatalities, and probabilities of the 23 accident types, 12 core damage categories, and nine release categories. The code used the event trees of the risk model and the probabilities described in items 1 through 5 above. Uncertainties in the input probability estimates were assigned by judgment. The code propagated the uncertainties in input data using a Monte Carlo sampling procedure which appropriately accounts for statistical dependencies.

TABLE A3.2-1

QUANTIFICATION PROCEDURES AND DATA SOURCES

Risk Model Element	Procedure	Data Source
Initiating Events	- Fault tree analysis - Availability analysis	- Nuclear Plant Reliability Data System (NPRDS) - CRPRA - GESSAR site seismic frequency
System Event Trees	- Fault tree analysis - Reliability block diagrams - FRANCALC-1 - Mathematical availability models - Fragility analysis - Dependency analysis	- Reactor Safety Study (WASH-1400) - NPRDS - CRPRA - LLNL generic fragility data
Core Response Event Trees	- SASSYS - ARIES - Parametric evaluations of accident energetics - Bounding analysis of reactivity worth and rate	- EBR-II metal fuel tests - WASH-1400 - Metal fuel data handbook - Appendix E of this PSID
Containment Response Event Trees	- Thermal analysis - Bounding calculations	- Nuclear Systems Materials Handbook
Consequences	- MACCS - ORIGIN	- NUREG-1150 - GESSAR site population distribution and meteorological data
Risk	- RISKSP	- Above elements

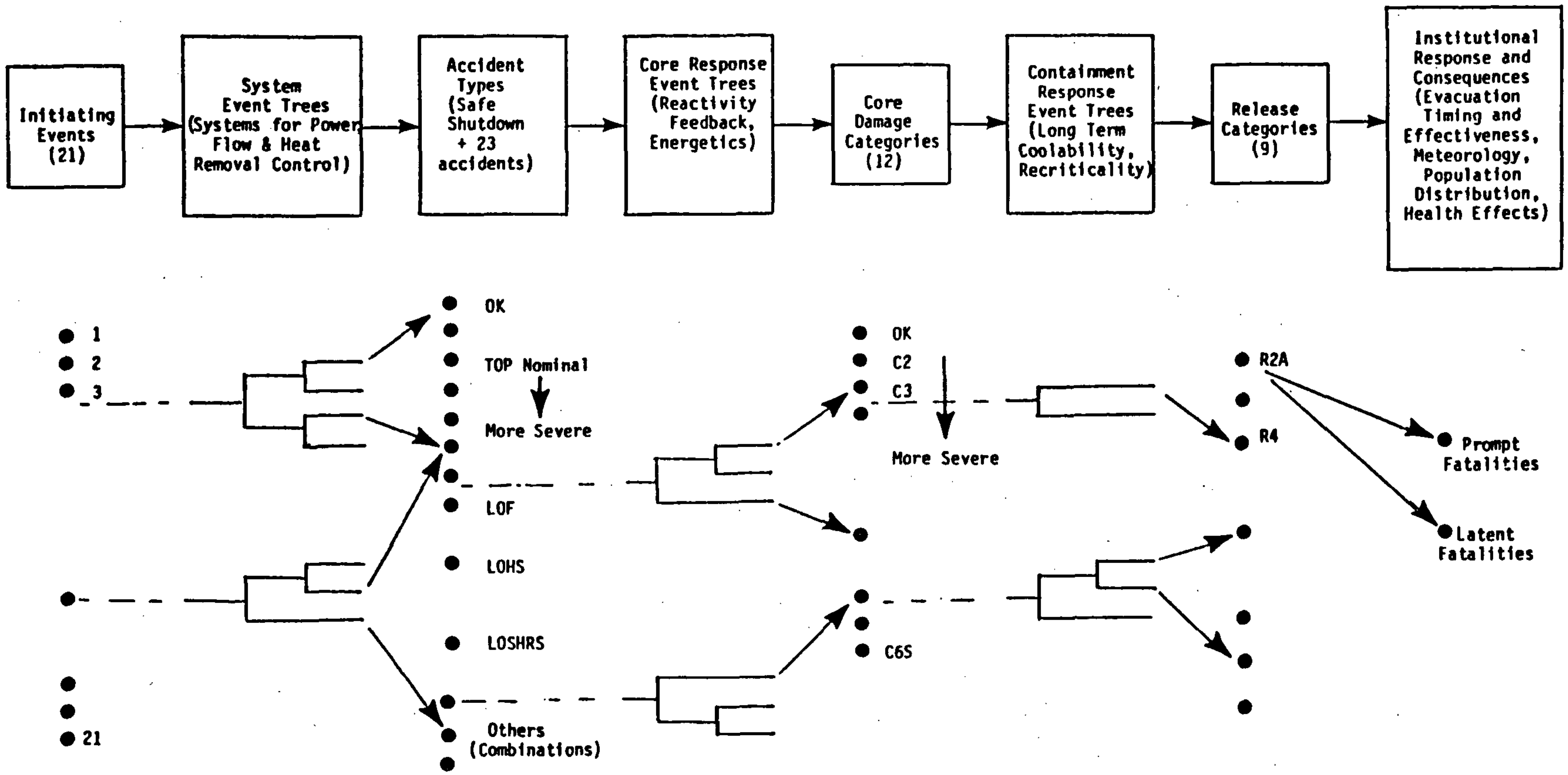

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Figure A3.2–1 PRISM RISK MODEL STRUCTURE

A 4.0

A4.0 RISK ANALYSIS

In this section, detailed risk analysis of the PRISM module is presented. Starting with the initial conditions that the plant is operating at full power, Section A4.1 defines twenty-one mutually exclusive and collectively exhaustive initiating events. By definition, all initiating events require the module to shutdown. Given an initiating event, plant systems are expected to conduct the transition from full power to decay heat levels in a safe manner and to maintain the module in a safe shutdown condition until the reason for shutdown no longer exists. Section A4.1 provides estimates of the expected frequency of occurrence of each initiating event and the required outage duration to remove the cause of shutdown for each.

Section A4.2 displays the possible responses of the module systems to each initiating event. Systems of interest include those designed to control the module power, coolant flow, and heat removal. The possible success and failure modes of these systems may lead to safe shutdown, continued safe operation, or one of twenty-three accident types. Each accident type presents an abnormal power or flow transient pattern which threatens the core integrity. Section A4.2 defines the systems of interest, their failure modes and criteria, probability of failure, and the probability of each accident type.

Section A4.3 describes the possible core response scenarios to each accident type. The scenarios are displayed in terms of reactivity feedback mechanisms which may become active in the course of the core response to each accident type. The possible scenarios could lead to minor or no clad damage or to one of eleven other core damage categories. The core damage categories are described in terms of: 1) the fraction of fission products released and where released, 2) fraction, location and form of molten fuel, if any, 3) damage to the reactor vessel or vessel seals, if any, and 4) the primary sodium enthalpy at the end of the transient. The section defines the reactivity feedback mechanisms, fission product release mechanisms, core damage categories, and the conditional probability of each given each of the accident types defined in Section A4.2.

Section A4.4 displays the possible containment responses to each core damage category until a stable end state is reached. Long term core coolability and recriticality considerations are assessed to evaluate the possibility of radioactive material release to the environment. The section defines nine possible stable end states. One of these states is continuous safe shutdown with no further consequences. The remaining eight end states contain the possible spectrum of radioactive material release patterns. The section provides the probability of each release category and the event sequences which may lead to each given each of the core damage categories defined in Section A4.3.

Section A4.5 contains an assessment of the public health consequences resulting from each release category. The results are evaluated under different evaluation assumptions. The procedures and data used in the analyses are also discussed.

A4.1 Initiating Events

A4.1.1 Introduction

As indicated above, the initial condition assumed for this PRA is that the plant is operating at full power. Given this initial condition, an initiating event for a module is defined as an event for which the module would or should be shutdown. Shutdown will continue for a period of time until the cause of shutdown is removed. The cause of shutdown could be an abnormal condition such as an uncontrolled withdrawal of a control rod, or normal shutdown for refueling. The objectives of this section are to:

1) define an exhaustive list of initiating events appropriate for PRISM,

2) estimate the expected frequency of each initiating event, and

3) estimate the mission time of the shutdown heat removal system given each initiating event. This time is defined as the mean time to remove the cause of shutdown and return the affected module to its initial state of operation (MTTR).

Review of initiating events of past PRA applications and the PRISM design resulted in the list of Table A4.1.1. The list can be thought of as composed of three groups:

(1) reactivity insertions other than seismic,

(2) external events (primarily earthquakes), and

(3) heat removal faults.

The definitions and basis for probability and MTTR assessments of these events are presented in the following sections.

A4.1.2 Initiating Event Definitions, Frequencies and MTTR

The first three initiating events include all reactivity insertion events other than those caused by another initiating event (such as an earthquake). For the purposes of subsequent core and vessel response analysis in the PRA, this broad category of events has been divided into three separate initiating events. The basis for this subdivision is the severity of the transient that would result given failure to shut down.

<u>INITIATING EVENT 1:</u> Reactivity Insertion Within Design Capability ($0.07 to $0.18)

This event is defined as a reactivity insertion significant enough that the plant should be shut down, but within the capability of the design to tolerate without fuel damage despite failure to scram. These events include withdrawal of up to three control rods. In principle, this event should be specified by a domain in the 2 dimensional space of reactivity insertion rate and total magnitude. For PRISM, it is judged that there is no credible way to obtain large, high ramp rate events, so all reactivity insertion events considered here occur at a low rate.

Transient analysis for a single control rod withdrawal ($0.06) without scram in the PRISM metal core gives an initial increase in power to about 110% nominal. The power then drops to 103% nominal and remains at this level. The peak power level of 110% is below the RPS setpoint for scram but will initiate a manual fast runback. Therefore PRISM will easily override reactivity insertions of $0.06 or less without the need to scram even if the fast runback is not initiated. On the other hand preliminary calculations indicate that the Doppler feedback alone can balance a reactivity insertion of up to $0.18 without fuel melting. Consequently, the range of this event has been defined as between $0.07 and $0.18 which is equivalent to the withdrawal of up to 3 control rods. Such an event will cause the RPS to trip the reactor when the flux reaches its set point of 112%.

The frequency of occurrence of this event should be dominated by control rod withdrawal. Other potential causes are cover gas entrainment, large fission gas bubble release from pins, and foreign material. Gas entrainment is designed against. Simultaneous failure of large numbers of pins is extremely unlikely. The CRBRP-1 PRA (Ref.A4.2-1) lists the most likely cause of foreign material as primary pump lube oil leak at 10^{-5}/yr. Since PRISM does not use lube oil this cannot occur. The frequency of single rod withdrawal has been estimated by a preliminary functional fault tree. The value obtained was 10^{-4}/yr. The dominant contributing events are erroneous setpoints for PCS control parameters and failure of the PCS decision logic circuit. Note that the occurrence of this event does not require failure of the RPS to detect and respond to the above failure, since these events are considered in the system response event trees.

The mean time to restore normal operation after such an event is estimated to be 600 hours due to the need to determine the cause of the event.

<u>INITIATING EVENT 2</u>: Reactivity Insertions Capable of Fuel Damage ($0.18 to $0.36)

This event is defined as reactivity insertions which would result in some fuel damage if the reactor shutdown system does not function, despite normal reactivity feedback. Based on transient analysis of the reactor response to a hypothetical withdrawal of 6 control rods ($0.36) without scram, a few fuel pins may fail before inherent negative reactivity feedback mechanisms effectively reduce the reactor power. Consequently this event covers the spectrum of reactivity insertions above those of IE1, up to the withdrawal of 6 control rods. The event will cause reactor trip by the RPS when the flux increases to 112% of nominal.

The frequency of this event may not differ greatly from that in Initiating Event 1, since occurrence of rod withdrawal was estimated to result mainly from PCS logic failure or erroneous setpoints for control parameters rather than independent failures in rod mechanisms. Although these failures should be rarer for a large withdrawal than for a lesser one; however, conservatively, the same event frequency value is used, namely 10^{-4}/year. Similar to IE1, a MTTR of 600 hours has been assigned this event.

INITIATING EVENT 3: Extreme Reactivity Insertions (greater than $0.36)

These events are defined as reactivity insertions beyond the nominal reactivity worth for withdrawal of the six control rods. Such events would require withdrawal of all the control rods and another fault such as an enrichment error. Similar to IE1 and IE2, this event will cause a reactor trip by the RPS when the flux increases to 112% of nominal. The frequency is estimated conservatively as 1 x 10^{-6}/year by the same reasoning as given above for Initiating Event 2. Due to the serious nature of this event, a MTTR of six months (4380) is assigned.

INITIATING EVENT 4: Earthquake (0.3g to 0.375g)

The PRISM reactor is expected to override an OBE (.15g) and continue operation. The PCS fast power runback will shutdown the reactor for earthquakes up to SSE (.15g to .3g). Consequently, this initiating event is defined as earthquakes for which the plant would be shut down by RPS action (i.e., greater than SSE) but small enough that major systems should function. The frequency of this range of events is 1 x 10^{-4}/year. Since the PRISM module is seismically isolated and the IHTS and BOP should be able to tolerate this magnitude without damage, an MTTR of 120 hours is assigned to allow time for a damage survey.

INITIATING EVENT 5: Earthquake (0.375g to 0.825g)

This event is defined as earthquakes clearly within the capability of the seismic isolation system. The frequency of this range is 1.9 x 10^{-5}/year. Damage to the BOP is expected, so an MTTR of 4380 hours is assigned.

INITIATING EVENT 6: Earthquake (greater than 0.825g)

This event is defined as an earthquake which might conceivably exceed the capability of the seismic isolation system. The frequency of this range of events is 7.1 x 10^{-7}/year. An MTTR of six months (4380 hours) is assigned. For such severe earthquakes, the actual mean-time-to restore normal operations may be much longer than six months. The value six months is used to account for the effect of the grace period on the effective SHRS mission duration. The SHRS grace period is the length of time from loss of shutdown heat removal until temperatures reach the point at which fuel failure occurs. This delay is caused by the large heat capacities of coolant and structures. Restoration of SHRS during this grace period would prevent fuel damage; hence, SHRS would not have failed in its safety mission to prevent fuel failure.

Initially, the grace period for PRISM is about 30 hours, but after six months it is at least two weeks. Thus, if the SHRS system were to fail after the plant had been shut down for six months, at least two weeks would be available for repair. Moreover, the system temperature would be low enough for fuel removal if necessary. Hence, the maximum effective SHRS mission is about six months.

<u>INITIATING EVENT 7:</u> Vessel Fracture

This event is defined as a complete circumferential vessel rupture. Such catastrophic failure may occur due to the presence of a large initial flaw in a circumferential weld, which grows during service, due to thermal cycling. When the critical size is reached the vessel fractures. The PRISM reactor vessel design has substantially lower stresses and vulnerability to failure than typical LMFBR vessels. In particular, the vessel has the advantage of 1) factory manufacture and inspection, and 2) simple geometry with no penetrations, nozzles, or other stress raisers. As a result, the critical crack size required for fracture is exceptionally large. Consideration of this fact and other factors in probabilistic fracture mechanics leads to the conclusion that occurrence of non-seismically induced fracture of the PRISM vessel is incredible.

Physically meaningful failure rates as low as 10^{-11}/yr have been calculated for other structures, hence a value of 10^{-13}/yr is used here for PRISM as representative of the incredibility of this event. A value of MTTR=4380 hours (six months) was assigned for this event.

Should such an incredible event occur, the impact stresses on the containment vessel have been estimated to be well within its structural capability. Therefore, the containment vessel will support the reactor vessel and any primary sodium leaking into the containment vessel. The reactor vessel drop will cause a reactivity insertion ramp due to the core/control rod relative motion. The RPS will trip the reactor on high flux detection. Decay heat removal will be effectively removed by RVACS from the primary sodium in the containment vessel.

<u>INITIATING EVENT 8:</u> Local Core Coolant Blockage

The CRBRP Risk Assessment report (CRBRP-1 Appendix III) identifies three possible causes of local blockages:

(1) Failure of a filter in the on-line sodium clean-up system;

(2) Leakage of mechanical sodium pump lubrication oil;

(3) Undetected inadvertent introduction of foreign material during refueling.

The first two events are not applicable to the PRISM design, since there is no on-line filter nor pump lube oil. The frequency of the third event was estimated as 1.8×10^{-6} per year. Similar to IE7, the value of MTTR for this event was assigned as 4380 hours.

Should the above blockage lead to coolant voiding of one or more subassembly, the resulting reactivity addition will cause a reactor trip on high flux by the RPS. For less severe blockages which may lead to clad failure, or moderate flux increase, the PCS or the operator will bring the reactor to orderly shutdown on failed fuel detection or mismatch of power, flow, and control rod position. For the purpose of this assessment, it is conservatively assumed that the blockage is severe enough to require RPS action to trip the reactor.

INITIATING EVENT 9: Reactor Vessel Leak

The frequency of this event is estimated at 10^{-6}/year. A value of six months (4380 hours) was assigned for the MTTR of this event.

The drop of sodium level in the reactor vessel will initiate a scram by the RPS. The leaking primary coolant will fill the containment vessel to a level above the IHX inlet, thus allowing decay heat removal via the balance of plant, ACS and RVACS.

INITIATING EVENT 10: Loss of One Primary Pump

The failure rate for PRISM EM pumps is 2.4×10^{-6} failures/hour each. Failure rate of the electric power to individual pumps is estimated at 2.6×10^{-6}/hr. Thus the loss of flow from one pump has a frequency of 5×10^{-6}/hr. There are four pumps; hence the frequency is:

$$f = 4(5 \times 10^{-6}/\text{hr})(8000 \text{ hrs of operation/year})$$
$$= 0.16/\text{year}$$

The mean time to recover has been assumed the same as a refueling outage, i.e., 600 hrs. This estimate is conservative since half of the failures are due to loss of electric power to the pump and will not need as much time to recover.

The loss of flow from one pump results in a low discharge pressure which initiates a scram and trip of the other three pumps by the RPS. Decay heat is removed via the BOP, ACS, and RVACS.

INITIATING EVENT 11: Loss of Substantial Primary Coolant Flow

This event is defined as loss of electric power to two primary pumps simultaneously.

The electric power distribution system uses two busses to feed power to the four EM pumps in a reactor. Each bus feeds two separate pumps. One bus also feeds power to the IHTS mechanical pump, while the other bus feeds the SG recirculation pump. Therefore, loss of power from one of the two busses will lead to loss of power to two pumps and loss of power to either the IHTS pump or the SG recirculation pump. The failure rate of either one of the two busses, but not both, has been estimated as 6×10^{-6}/hr or ~5×10^{-2}/yr. Since the event does not include failure of any reactor components, a mean time to recover of eight hours has been assumed.

The plant response to this initiating event is similar to that for IE10.

INITIATING EVENT 12: Loss of Operating Power Heat Removal

This event is dominated by failures of the main feedwater control valve. IEEE Standard 500 recommends 9.72×10^{-6}/hour as a failure rate. This is 0.08 failures per year. The repair time of 86 hours is an average from PWR experience.

INITIATING EVENT 13: Loss of Shutdown Heat Removal via BOP

This event is defined as a failure in the BOP such that not even decay heat can be removed via the BOP. Realistically, there are many off-normal modes of operation by which decay heat could be removed despite failure of the normal components performing this function. However, credit will only be taken here for use of the normal feedwater train and condenser system. Thus, the event "Loss of Shutdown Heat Removal via BOP" occurs if all three feedwater trains or both condensate trains are disabled at the same time.

The failure rate for each feedwater train is 10^{-4}/hour with a repair time of 48 hours. Each condensate train also has this same failure rate and repair time. Thus, each train has an unavailability of (10^{-4}/hours)(48 hours) = .0048. The three feedwater train system unavailability is $(.0048)^3 = 1.1 \times 10^{-7}$ and the two train condensate system unavailability is $(.0048)^2 = 2.3 \times 10^{-5}$. The repair time for the condensate system is given by $(1/48 \text{ hrs} + 1/48 \text{ hrs})^{-1} = 24$ hours. Similarly, for the feedwater system, it is 16 hours.

The frequency of system failure is just the unavailability divided by the system repair time; thus, for feedwater: $1.1 \times 10^{-7}/16$ hrs $= 6.9 \times 10^{-9}$/hr and for condensate: $2.3 \times 10^{-5}/24$ hrs $= 9.6 \times 10^{-7}$/hr. The frequency of either system failing (the initiating event frequency) is the sum 9.7×10^{-7}/hr or 0.008 per year. The mean repair time for this event is 24 hours.

<u>INITIATING EVENT 14:</u> Loss of Shutdown Heat Removal via IHTS

This event is defined as a failure which prevents removal of decay heat through both the Auxiliary (steam generator) Cooling System (ACS) or through the normal process of providing water to the steam generator and taking heat out through the BOP. The dominant failure mode for such events is a leak in the IHTS, thus requiring draining of this system for repair and preventing a sodium fire.

Frequencies of sodium piping leaks were estimated in the CRBRP report, GEFR-00554, as 1.3×10^{-6}/hour and for pump housing as 1.1×10^{-7}/hour. Thus, the total frequency of this initiator is the sum $(1.4 \times 10^{-6}$/hr)(8760 hrs/yr) ~0.01/year. The repair time for such events is comparable to a refueling outage due to the need for sodium drain and refill; hence, 600 hours is used.

<u>INITIATING EVENT 15:</u> IHTS Pump Failure

The failure rate of 5.5×10^{-6}/hour (.05/year) is used based on GEFR-00554. For the same reason as for Initiating Event 14, above, the repair time is 600 hours.

<u>INITIATING EVENT 16</u>: Station Blackout

This event is defined as loss of the capability to provide electric power sufficient to remove the operating power heat load. This means loss of all off-site and on-site electric power sources capable of running the BOP, IHTS and primary pumps. The failure rates for off-site power and on-site power from one power block are 10^{-5}/hour, and 10^{-4}/hour, respectively. The repair times are 1/2 hour for off-site power and 1000 hours for on-site power. It is assumed here that power can be supplied to the primary pumps from either off-site power or from either of two power blocks on-site. Thus the frequency of loss of all power is calculated as follows:

	(f_i/hr) Frequency	(t_i/hrs) Repair time	$q_i = f_i t_i$ Unavailability
Off-site	10^{-5}	0.5	5×10^{-6}
On-site block 2	10^{-4}	1000	0.1
On-site block 2	10^{-4}	1000	0.1

Loss of all three power sources:

Unavailability $Q = q_1 q_2 q_3 = 5\times10^{-8}$

Residence time $T = (1/t_1+1/t_2+1/t_3)^{-1} = 0.5$ hour

Frequency $F = Q/T = 10^{-7}$/hr $= 8\times10^{-4}$/year

However, the scenario for such an event is that first one on-site block becomes unavailable, then, while it is under repair, the second block fails. The residence time for both blocks in a failed state, each with a 1000 hour repair time, is $(1/1000 \text{ hr} + 1/1000 \text{ hr})^{-1} = 500$ hours. Then, during this 500 hours, off-site power loss occurs. However, this scenario will not occur because there will be a safety related technical specification that the plant not operate for more than some period (say, 36 hours) with both on-site power sources down. Hence, the above calculation of frequency must be reduced by a factor of 36/1000. The resulting frequency of loss of all power to a module while the module is operating is 3×10^{-5}/year.

A repair time of 1200 hours is conservatively used for this event to allow for inspecting the module following such a transient.

<u>INITIATING EVENT 17:</u> Very Large Na-H_2O Reaction

This event is a very large sodium water reaction in the steam generator. This would disable heat removal via IHTS and would challenge various other systems. A frequency of 6 x 10^{-8}/year is used, based on the estimate for steam generator tube leaks. Smaller leaks would be considered a normal loss of heat removal via IHTS (Initiating Event 14).

The mean time to repair such a catastrophic event would be the maximum time of 6 months (4380 hours). Since PRISM has been designed to tolerate a full sodium-water reaction without dump, this event is not expected to affect the module directly.

<u>INITIATING EVENT 18:</u> Spurious Scram and Transients Inadequately Handled by PCS

This event covers spurious scrams caused by RPS circuitry faults, and transients which should have been controlled by the PCS fast runback system but were not as a result of PCS failure.

As discussed under initiating event 20 for forced shutdown, PRISM is designed to accommodate 5.5 events per year which may need shutdown by PCS fast power runback. Conservatively assuming that 10% of these events are inadequately handled by PCS, the RPS will be challenged by .55 such events per year. The mean time to recover from such events is conservatively assumed to be the same as the refueling time of 600 hours.

Past LWR experience has been one or two spurious scrams per year. However, many of these occur in the first two years of plant operation. The PRISM RPS system design permits for better signal validation and interpretation so that such events would be reduced to an insignificant frequency. A value of 0.04/year is used. Again, due to the better control information system of PRISM, determination of the source of the spurious scram should be rapid, so a recovery time between twenty-four and forty-eight hours is expected.

From the above discussion, initiating event 18 is conservatively assumed to occur at the frequency of 0.6/yr and to require 600 hours to recover.

INITIATING EVENT 19: Normal Shutdown

This is the planned refueling outage; hence, a frequency of 0.6/year and the typical outage time of 600 hours are assumed. This is based on the intended PRISM refueling cycle of twenty months.

INITIATING EVENT 20: Forced Shutdown

The PRISM duty cycle includes 331 fast power runback events for the plant life of 60 years, i.e. an average of ~5.5 events/year. The PRISM availability goal is 85%, which includes both planned (refueling) and forced outages. Assuming conservatively that all the unavailability is due to forced outages, leads to an average outage time of <240 hours per outage event. Therefore, a frequency of 5.5/yr and MTTR of 240 hours is conservatively assigned to forced outages.

INITIATING EVENT 21: RVACS Blockage

This event is a blockage of RVACS air flow sufficient to threaten successful shutdown heat removal, if needed. Earthquake induced blockages are included within Initiating Events 4 through 6, rather than here. None of the internally induced failures of PRISM systems thus far evaluated have shown a capability of disabling RVACS. Spontaneous structural failures are rare but possible. Thus, the only causes of RVACS blockages sufficient to require shutdown are catastrophic external events and structural failure. These have been evaluated as having a frequency of about 10^{-8}/year. The repair time for such an event is judged to be comparable to that for repair of typical large components, 86 hours. Removal of blocking material sufficient to restore the 5% of unblocked flow which is needed for successful shutdown heat removal is considered repair.

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Table A4.1-1

INITIATING EVENTS FREQUENCY AND MEAN TIME TO RECOVER

	Initiating Event (IE)	f(1)	t_m(2)
1	Reactivity Insertion 0.07\$-0.18\$	1.0E-4	600
2	Reactivity Insertion 0.18\$-0.36\$	1.0E-4	600
3	Reactivity Insertion >0.36\$	1.0E-6	4380
4	Earthquake 0.3g to 0.375g	1.0E-4	120
5	Earthquake 0.375g to 0.825g	1.9E-5	4380
6	Earthquake >0.825g	7.1E-7	4380
7	Vessel Fracture	1.0E-13	4380
8	Local Core Coolant Blockage	1.8E-6	4380
9	Reactor Vessel Leak	1.0E-6	4380
10	Loss of One Primary Pump	1.6E-1	600
11	Loss of Substantial Primary Coolant Flow	5.0E-2	8
12	Loss of Operating Power Heat Removal	8.0E-2	86
13	Loss of Shutdown Heat Removal via BOP	8.2E-3	24
14	Loss of Shutdown Heat Removal via IHTS	1.0E-2	600
15	IHTS Pump Failure	5.0E-2	600
16	Station Blackout	3.0E-5	1200
17	Large Na-H_2O Reaction	6.0E-8	4380
18	Spurious Scram and Transients Inadequately Handled by PCS	0.6	600
19	Normal Shutdown	0.6	600
20	Forced Shutdown	5.5	240
21	RVACS Blockage	1.0E-8	86
	TOTAL:	6.398	

1. f = initiating event frequency on a per year basis; values are given in exponential form where XE-Y = $X10^{-Y}$.
2. t_m = Shutdown heat removal mission time in hours = expected (or mean) time required to restore to normal power operation.

A4.2 System Event Sequences

A4.2.1 Introduction

As discussed in Section A4.1, an initiating event requires the module to be shut down until the cause of shutdown is removed. The systems responsible to realize shutdown are those for control of the module power, coolant flow, and heat removal. Possible responses of these systems may lead to safe shutdown and restart of the module operation as expected, or may lead to an abnormal situation which will henceforth be called an accident.

To systemically identify all possible accident types, a system response event tree has been developed for each initiating event. The system event trees are a part of the risk model and display the following important parameters:

a. Logical combinations of system responses which form accident sequences.

b. Dependencies between responses of the various systems.

c. Relation between accident sequences and the end state of either safe shutdown or one of twenty-three accident types.

d. Probabilities of various system responses and accident sequences.

The developed system event trees are shown in Figures A4.2-1 through A4.2-21. The figures show three distinct patterns for the event trees. The first pattern covers initiating events 1 through 18; except for the initiating event of earthquakes greater than 0.825g (initiating event 6). For this pattern, shutdown is initiated by RPS action. The second pattern is used for the large earthquake initiating event. In this case, the event tree explicitly includes response of the seismic isolators. The third pattern covers initiating events 19 through 21, where shutdown is initiated by PCS fast power runback.

Each of the system response event trees for initiating events 1 though 18 contains exactly seven events. Except for initiating event 6 (earthquake greater than 0.825g), each of the trees contains responses of the following systems:

(1) Reactor Protection System (RPS). This system senses the need to shut down and initiates the proper signals for power, flow, and heat removal control.

(2) Reactor Shutdown System (RSS). This system includes the control rods, control rod drive motors, and magnetic latches.

(3) Inherent Reactivity Feedback Features. These include the control rods, their drivelines and their guide tubes, the core restraint system, load pads of the core assemblies, and the grid plate.

(4) Primary Pumps. This includes the primary pumps and their power supply.

(5) Pump Coastdown System.

(6) Operating Power Heat Removal System (via Balance of Plant [BOP]).

(7) Shutdown Heat Removal via IHX or RVACS.

Failure criteria of the above systems are defined in the following sections. The sections also contain the probability models and data used for estimating the conditional probability of failure of each system.

As noted above, the system response event tree for the largest earthquakes (Figure A4.2-6) differs from the ones above in that it contains the event "Seismic Isolation Function." This event is included only in the one tree because the possibility of failure of this system for other events has been determined to be unrealistic. Failure of this function has been calculated to result in large structural deformations. These deformations have been assumed to put the control rods out of the core thus resulting in a large transient overpower and loss of flow. Moreover, gross structural failure of the vessels may occur and core meltdown is not unlikely. The probability of 0.00135 (shown in Figure A4.16) for seismic isolation failure was obtained by estimating the probability that the actual vertical acceleration of the isolation pads (during an earthquake of the specified peak ground acceleration) exceeds their capacity, which is 1.0g.

System response event trees for initiating events 19 through 21 are shown in Figures A4.2-19 through A4.2-21. Since these events present orderly PCS or manual shutdown, the only event of interest is the shutdown heat removal system capability to remove decay heat until ascent to full power operation.

The system event trees in Figures A4.2-1 through A4.2-21 display three types of dependencies, either explicitly or implicitly.

(1) Dependence on the Initiating Event. This dependence is accounted for in the definition of system success criteria (e.g., RPS sensors and setpoints which will result in scram, number of control rods which have to be inserted, duration for which SHRS must remove decay heat, degradation or loss of SHRS subsystems which must be assumed as a result of the initiating event).

(2) Dependence Between System Responses. This dependence has influenced the order in which the system responses are displayed in the event trees. An example of this dependence is the failure of the RSS to insert its control rods if the RPS fails to send a scram signal. Another example of system dependence is the successful shutdown by the RSS which renders the response of the inherent feedback system as irrelevant. These types of functional dependencies are represented in the event trees by different conditional probability estimates which depend on the preceding sequence of events, or by "straight through" or "don't care" lines which do not branch into success and failure branches for the dependent system response.

(3) Dependencies Between Subsystems of a System. These dependencies are factored in the system reliability models described in the

following sections. The probability estimates shown on the event trees of Figures A4.2-1 through A4.2-21 reflect these dependencies.

The system event trees shown in Figure A4.2-1 through A4.2-21 are to be interpreted as follows. At each node where the tree branches, the top branch means that the event presented at the top heading of the tree has occurred. The lower branch means that the event did not occur. Sequences formed from the various events lead either to safe shutdown and restart of operation of the affected module (S1) or to one of twenty-three accident types. Each accident type is presented in the event tree by a letter symbol, (e.g., S,P,F,H,G) which refers to a generic accident group followed by a number (e.g., 1,2,3,4) which refers to a level of severity of the accident type relative to its generic group. The level of severity increases with this number. For example, P3 is a more severe transient overpower than P1. The generic accident groups are:

(1) Protected (i.e., reactor is shut down by RSS) loss of the shutdown heat removal system (LOSHR) -- represented in the event tree by the accidents S_3 and S_5 (S_1 stands for successful shutdown and shutdown heat removal).

(2) Unprotected (i.e., reactor is not shut down because of RPS or RSS failure) transient overpower (TOP) -- represented in the event trees by the letter P.

(3) Unprotected loss of flow (LOF) -- represented in the event trees by the letter F.

(4) Unprotected loss of heat sink (ULOHS) -- represented in the event trees by the letter H.

(5) Unprotected combined TOP/LOF or TOP/ULOHS -- represented in the event trees by the letter G.

Table A4.2-1 contains the specific definitions of the accident types.

A4.2.2 Reactor Protection System Reliability

This section presents the procedures, data, and results of estimating the conditional failure probability, given an initiating event, of the Reactor Protection System (RPS). The presentation includes discussion of the success criteria, failure probability models, and results which were developed for this assessment.

A4.2.2.1 System Functions and Success Criteria

Given one of the initiating events defined in the previous section, the affected PRISM module is expected to:

(1) detect the occurrence of the initiating event;

(2) determine that the module is to be shut down;

(3) signal to the power, flow, and heat removal control systems to actuate shutdown; and

(4) bring the affected module to a safe shutdown and retain it in this condition until the cause of shutdown is removed.

The first three functions are performed by the Plant Control System (PCS) and RPS. Specifically, the PCS functions are to:

(1) continuously monitor the process parameters in the nuclear steam supply system (NSSS), the turbine/generator set (T/G), and balance of plant (BOP);

(2) alert the operator by appropriate alarms, fault reports, and margin-to-safety limit calculations so that the operator can take proper action for investment protection;

(3) signal to the reactor shutdown system (RSS) and flow control actuators to bring a module, a power block, or the plant to an orderly, safe, and optimal shutdown configuration; and

(4) request the RPS to enter a shutdown/maintenance mode if necessary so that appropriate protection actions are taken by the RPS before maintenance and repair are initiated.

The RPS functions are to:

(1) continuously monitor the process parameters in the reactor (neutron flux, cold pool and core outlet temperatures, pump discharge pressure, and primary sodium level);

(2) send a trip signal to the control rod release mechanisms and drive-in motors to assure insertion of the control rods;

(3) initiate coastdown of the primary EM pumps;

(4) assure head isolation valves are closed; and

(5) provide a trip signal to the PCS for flow adjustment in the intermediate loop, steam generator and T/G, and proper adjustments in the remainder of the plant.

The RPS design and operation to reliably accomplish these functions are summarized below. This is followed by a definition of the system success criteria.

The PRISM RPS has four identical, parallel logic trains or divisions. Each logic train consists of a sensor, analog input/amplifier/digital converter, digital logic unit, and trip actuator. Five parameters are used for reactor trips. Each logic train has one sensor input for each parameter. A multiplexer at the analog input to each logic train permits the selection of the desired parameter to be observed.

The RPS is housed in electronic equipment racks located in vaults adjacent to the head access area. There are four instrumentation vaults, each contains a division of the RPS. The RPS panel in each vault is physically separated from the Plant Control System and other non-safety related electronics. Only safety-related electronics and support equipment are located within the RPS instrument vaults. The electronic equipment and cabling design will minimize the risk of fire and/or toxic fume generation. Each division is provided with its own safety related, battery backed, uninterruptible power source. The only communication connections between the four channels are made by optically isolated cables. Thus, there are no common elements, functions, or electrical interconnections which could lead to an overall system failure.

Figure A4.2-22 shows a block diagram of an RPS division. The input data processor addresses a specific sensor, conditions the sensor's signal output, samples it, then converts the signal to a digital data word. In parallel with the analog signal processing, the sensor's analog voltage is checked for expected level as a verification of the sensor's correct operation. An auto calibration feature is also included to increase the reliability and confidence in the sensor data. This entails injecting a known test signal into the sensor circuit. The output signal from the sensor is compared against the test signal and the sensor's normal state information.

After converting the sensor signal into a digital word by an RPS division, the digital sensor information and its verification status are distributed to all divisions. This data exchange takes place via a redundant, optically isolated network connecting the four logic trains (one in each vault).

All digital logic units are able to process (simultaneously and in parallel) the identical data from all four divisional sensors observing a single parameter. The logic in each vault is able to perform the required verification and validation functions and make an independent two out of three (with one spare) decision as to the need for a reactor trip.

Should the readings or calculations processed by the CPU exceed a given RPS trip setpoint, each division outputs a trip signal. The signal is output as an optical input to a set of solid state trip breakers. The trip signal results in an interruption of the current flow through optical light sources in the breakers for each RPS division (fail-safe).

Optically coupled trip breakers at the output of each divisional logic train are hardwired in a two out of four logic. This final output logic assures that only one division may be taken out of service for any reason at any time without causing a reactor trip. The trip breakers are designed such that they must be energized to prevent a trip. Thus, a loss of power to two or more RPS divisions will assure that the reactor is automatically shut down (fail safe design). Each RPS division has 12 trip breakers for the control rod latch mechanisms (two per control rod) and 12 trip breakers for the control rod drive-in motors (two per control rod). Figures A4.2-24 and A4.2-25 show scram breaker logic for the latch and drive-in mechanisms.

If the electrical current for two or more sets of divisionalized breakers is interrupted (2 out of 4 logic), all control rods will be released to shut down the reactor.

The reactor trip is obtained by two diverse mechanisms: (1) de-energizing the magnetic latches which hold the control rods to their drive assemblies, and (2) energizing drive-in motors to insert the control rods.

Several levels of diagnostics are performed by the RPS automatically at differing intervals. These levels include: individual component calibration, checking of subsystem calibration/wellness, overall system performance, signal verification and validation, data exchange validation, and trip validation.

The four RPS divisions work together as a fault tolerant system, that is, any failure that occurs within each division is detected and confined. Reconfiguration occurs automatically to bypass a problem area. The system is capable of being repaired while operating. One entire division may be removed for service at any time without system degradation. The inputs are fully fault tolerant, that is, if a failure occurs within an input section, the failure is isolated and the system is reconfigured around the failure. Each of the four central processing logic units is capable of error detection, containment, and reconfiguration. Each optically coupled circuit breaker is provided with a test feature (an extra set of output contacts) such that the complete division may be automatically tested (from sensor input through to and including the trip breakers) at any time without the release of a control rod or initiating a reactor scram.

Based on the above considerations, the RPS will successfully perform its function if two or more divisions trip the scram breakers to insert all control rods.

A4.2.2.2 Fault Tree Analysis

As defined above, the success of RPS is to trip the scram breakers in two or more of the RPS four divisions. Failure of one RPS division to trip its scram breakers leaves the corresponding scram breakers in an energized (no shutdown) state. From the sequence of responses to each initiating event described in Section 4.2.2.1, the failure of the RPS division results from one of the following events.

(1) Failure to sense abnormal conditions which require shutdown. Abnormal conditions include unacceptable process parameter values sensed by the RPS sensors.
(2) Failure to decide on reactor shutdown due to error of shutdown criteria, even if abnormal conditions are sensed. Errors in reactor parameters setpoints could lead to this failure.
(3) Failure to transmit and process sensed abnormal conditions so that a decision to shut down may be taken. This failure covers the data handling and transmission system wiring and electronics for data acquisition, signal processing and voting, and comparators. It also covers failure of hardware and software for RPS reconfiguration around faults and for RPS diagnostics.

(4) Failure to de-energize the scram breakers even if abnormal conditions are accurately sensed, transmitted and processed, and appropriate shutdown criteria are met. This failure has been conservatively defined as failure to trip at least one-latch scram breaker and one-motor drive-in scram breaker.

The fault tolerant capability of the PRISM RPS means that the occurrence of one of the above failures in an RPS division will not completely disable that division. The affected division will reconfigure to use information from the unfailed divisions. In effect, the reconfiguration capability transforms the redundancy at the division level to redundancy at the subdivision levels of sensors and remainder of the RPS channels. This is illustrated in the block diagram of Figure 4.2-26.

The RPS success criterion defined earlier means that the system will fail if three or more divisions out of the four divisions fail to perform their function. With fault tolerance effectively providing redundancy at the subdivision levels as shown in Figure 4.2-26, the system will have the following failure modes: (1) failure of at least three out of four sensors, (2) failure of at least three out of four channels to trip the scram breaker sets, and (3) error in the setpoints of the sensed parameter for all four divisions.

Note that all four division setpoints will have to be in error since the PRISM RPS does not allow power operation unless all setpoints match each other and a fifth file maintained independently by the PCS. (The RPS also would not allow operation if its setpoints are inconsistent with the corresponding setpoints of the PCS.)

The fault tree of Figure A4.2-27 displays the logical relationship between the above failure modes and the event of "RPS failure." An 'OR' gate is used to combine the three failure events above since the occurrence of any of them leads to the top event.

Detailing the RPS fault tree beyond the level shown in Figure A4.2-27 has been avoided to preserve the simplicity of the logical relationships. However, each of the shown events is related to more refined equipment and operation strategy for testing and repair in the following section. Mathematical expressions establishing the relationships are developed in that section.

A4.2.2.3 Fault Tree Quantification

The failure events shown in the fault tree of Figure 4.2-27, refer to functional failures of components. For example, if a component is out for repair and cannot be recalled to perform its function when an initiating event occurs, the component is considered to be in a failed state. Therefore, estimating the probability of the above events involves estimation of component unavailability, whether the component unavailability is due to actual failure, or outage for testing or repair. We will use the following notations for probability estimation:

$P(X|Y)$ = conditional probability of Event X given that Event Y has occurred

q_d = time-independent component unavailability given a demand for the component to function

λ = expected component failure rate, failures/hour

T = period between testing for periodically tested components, hours

t = expected outage duration needed for maintenance and repair of a component, hours

We will also use the following testing strategy definitions:

a) <u>Continuously Monitored Components</u> - For components tested at short time intervals relative to the time to scram (~0.2 sec.).

b) <u>Periodically Tested Components</u> - For components tested at regular intervals such as every month, every refueling, etc.

c) <u>Staggered Testing</u> - For similar components where testing of any two components is done at regular intervals. For example, a staggered testing of four components in a quadruple system may be conducted as follows: Component 1 tested at time 0, Component 2 tested at time = 1 week, Component 3 tested at time = 2 weeks, Component 4 tested at time = 3 weeks; then the cycle is repeated with Component 1 tested at time = 4 weeks, etc.

Figure A4.2-27 shows that RPS failure results from sensor failure (Event A in Figure A4.2-27), failure to trip the scram breakers (Event B) and setpoint errors (Event C). The different nature of these events and the different testing and repair strategies used have led to different probability models for these events. In the following discussion, each event is analyzed separately to estimate the probability of independent and dependent failures.

<u>Event A (Sensor Failure)</u>

Table A4.2-2 shows the sensors used by the RPS and the process parameters (directly measured or calculated) which are used by the RPS for reactor shutdown.

Transient analysis using the ARIES-P code indicates that the RPS parameter whose setpoint is first reached depends on the initiating event. In particular, the analysis shows that:

(1) The flux trip point will be the first to reach for a reactivity insertion event.
(2) The flux/flow trip point will be the first to reach for a flow coastdown event
(3) The core inlet temperature or outlet temperature will be the first to reach for a loss-of-heat sink event.

The RPS sensors have quadruple redundancies. Since the RPS is designed to shut down the reactor if two out of four sensors of any type read abnormal conditions, RPS sensor failures which result in no request to shut down must satisfy the following conditions:

(1) Sensor fails in a mode which erroneously shows a safe reading, e.g., an underestimate of flux or temperature or overestimate of flow. Note that a sensor failure in the general sense does not necessarily lead to failure of the RPS to shutdown.

(2) Three out of the four sensors must fail in an unsafe mode for a parameter detection to fail (e.g., flux detection given a reactivity insertion).

(3) RPS monitoring of the state-of-the-sensor-health must fail to detect the above failures or fail to request a shutdown. (NOTE: The RPS sensor output must be within a given Go-NoGo range to be accepted as a verified sensor signal.)

(4) The above failures exist prior to, coincident with, or as a result of the initiating event.

To estimate the conditional probability of RPS sensor failure given an initiating event, the following assumptions are used.

(1) The sensors are continuously monitored via signal verification capabilities of the RPS.

(2) The sensors of each RPS division are automatically tested every four hours. The four divisions are sequentially tested on an hourly basis. The testing is performed using the procedure of injecting a known signal through the sensing circuit which was described in the previous section.

(3) The probability of a sensor malfunction to escape detection by continuous monitoring between two periodic tests is 0.1. Periodic testing is assumed to be perfect, i.e., no sensor malfunction escapes detection.

(4) Dependent failure of sensors constitutes 1% of the sensor failure modes (i.e., a beta factor of 0.01). Dependent failure of continuous monitoring is 50% of continuous monitoring failures (i.e., a beta factor of 0.5).

(5) The sensors for each initiating event are as shown in Table A4.2-4.

(6) The RPS will scram if two sensors of the same type are found in a fail state. The reactor will continue operation until refueling if one or no sensors fails. The time between refueling is assumed to be T = 20 months or 14,400 hours.

The above assumptions lead to the following independent and dependent failure probabilities for a sensor type (e.g., flux sensor).

$$P_{indep} = (4\ \lambda\ T/2)\ [25\ (\lambda \times 0.1)^2/4] \quad \text{(Equation 4.2-1)}$$
$$= 1800\ \lambda^3 \quad \text{(Equation 4.2-2)}$$

$$P_{dep} = [\lambda x(4/2)x0.1] \times \beta_S \times \beta_{mc} \quad \text{(Equation 4.2-3)}$$
$$= \lambda \times 0.2 \times 0.01 \times 0.5$$
$$= 10^{-3}\lambda \quad \text{(Equation 4.2-4)}$$

In Equation 4.2-1, the first term in brackets presents the average unavailability of one sensor. The sensor is modeled as an unrepairable component over the period T. The second term presents the failure probability of two out of the three remaining sensors. Each sensor is modeled as tested periodically every four hours, with sensors of different divisions tested sequentially in a staggered fashion as explained earlier. The squared term in Equation 4.2-1 represents the rate of undetected failures.

In Equation 4.2-3, the first term presents the probability of a sensor failure without detection before periodic testing. The second and third terms present the probabilities that all other sensors and continuous monitoring fail concurrently due to a common cause.

Based on the NPRDS historic failure data reports (Ref. A4.2-2), the following failure rates for sensors were estimated:

Fission chamber flux detectors:	2.15×10^{-6}/hr
Compensated ion chamber flux detectors	2.18×10^{-6}/hr
Temperature detectors:	8.6×10^{-7}/hr
Pressure detectors:	5×10^{-6}/hr

To simplify the calculation for various initiating events, a generic sensor failure rate value of 10^{-6}/hr was used for all sensor types, i.e.,

$$\lambda = 10^{-6}/\text{hr} \qquad \text{(Equation 4.2-5)}$$

The above failure rate is assumed applicable to all initiating events except seismic events. Using this failure rate in Equation 4.2-2 and 4.2-4 leads to the following failure probability of one sensor type:

$$P_{indep} \sim 2 \times 10^{-15} \qquad \text{(Equation 4.2-6)}$$

$$P_{dep} = 10^{-9} \qquad \text{(Equation 4.2-7)}$$

As seen from Table A4.2-4, initiating events leading to reactivity addition must be detected by the flux sensors, vessel leaks must be detected by primary sodium level sensors, loss-of-flow initiators must be detected by both flux and pressure sensors, and the remaining initiating events must be detected by one of two sensor types, e.g., either core inlet temperature sensors or core outlet temperature sensors. This leads to the following probability estimates.

For initiating Events 1, 2, 3, 8, and 9:

$$P\,(A|IE) = 10^{-9}$$

For initiating Events 10, 11, 16, and 18:

$$P\,(A|IE) = 2 \times 10^{-9}$$

For initiating Events 7, 12, 13, 14, 15, and 17

$$P\,(A|IE) \leq 10^{-11}$$

(Equation 4.2-8)

Sensor failure probability for the seismic initiating events (IE 4, 5, 6) are discussed at the end of this section. Note that initiating Events 19, 20, and 21 do not require RPS action. Therefore, failure probabilities of RPS given these events are not estimated.

<u>Event B (Failure of at Least 3 Out of 4 Channels to Trip All Scram Breakers)</u>

This event is defined as failure of three out of the four RPS divisions due to failures other than sensor failure. Each division may encounter such a failure if one of the following two events occur:

1) Failure of an RPS logic unit, or
2) Failure of the scram breakers to trip.

These events are discussed separately below, then combined to estimate the probability of Event B given each initiating event.

An RPS logic unit processes sensor data, distributes the equivalent digital information to other RPS divisions, compares the measured value of a trip parameter to its setpoint, and sends a signal to trip the scram circuit breakers if necessary. A failure of an RPS logic unit for our purpose here means failure in the above functions which lead to failure to send a scram signal when one is needed.

Each RPS channel contains an input signal conditioning electronics (preamp, multiplexer, amplifier, filter, analog/digital converter), processor (CPU, memory, clock, I/O equipment, communication for receiving/transmitting information), electronic voters and comparators, and data busses. The exact number and design configuration of these components is not yet available. Assuming the worst configuration for these components (connected in series), that a channel has 20 such components, and using a generic failure rate for electronic devices of 10^{-6}/hour which is consistent with WASH-1400 and NRPDS data, the following failure rate estimate is obtained:

$$\lambda = 2x10^{-5}/\text{hour}$$

The above value of λ is consistent with the WASH-1400 estimate of general instrumentation (calibration shift) failure of $3x10^{-5}$/hour.

As stated earlier, each RPS vault has 12 circuit breakers for the latch release mechanism (two for each rod), and 12 circuit breakers for the motor drive-in mechanism (also two for each rod). Failure of the circuit breakers to trip will be conservatively defined as failure to trip breakers for any single control rod. (i.e., the corresponding success mode is tripping of circuit breakers of all rods).

The Reactor Safety Study (WASH-1400) estimates scram breaker time-independent unavailability, qd, as 10^{-4}. It estimates a failure rate of 10^{-7}/hr for failing to open a normally closed relay, including failure in wiring.

To estimate the conditional probability of Event B given an initiating event, the following assumptions are used:

1) The RPS logic units are continuously monitored using analogue and digital signal verification of the RPS.

2) The logic unit and scram breakers of each RPS division are periodically tested every four hours. The four divisions are sequentially tested on an hourly basis. The testing is performed using the procedure of injecting a known signal through the sensing circuit and up to the trip breakers which was described in the previous section.

3) Staggered testing is perfect, i.e., faults are detected at 100% efficiency. The probability of an RPS logic unit failure to escape detection by continuous monitoring between periodic tests is 0.01.

4) Dependent failure of RPS divisions constitute 0.1% of the division failure modes (i.e., a beta factor of 10^{-3}). Dependent failure of continuous monitoring of the logic unit has a beta factor of 0.5.

5) The RPS will scram if two logic units are found in a failed state. The reactor will continue operation until repair of a logic unit is completed in case one logic unit fails. The mean time to repair is assumed to be four hours.

6) A beta value of 10^{-2} is assumed for scram breakers in the same RPS vault (latch and motor drive in breakers of the same rod). For scram breaker dependence across the RPS vaults, a beta factor of 10^{-3} is assumed.

Using the above assumptions, Event B will occur if one of the following conditions exist:

1) One channel is out for repair (t = 4 hours), two or more channels fail and escape detection by continuous monitoring, or
2) Three or more channels fail and escape detection by continuous monitoring.

Consider first the case of independent failures. The unavailability of a channel due to repair is given by:

$$U_R = \binom{4}{1} \left(\lambda_{LU}t + [(q_d + \lambda_{SB}t)\binom{12}{1}]^2\right) \quad \text{(Eq 4.2-9)}$$

$$= 4(2x10^{-5}x4+[(10^{-4}+10^{-7}x4)12]^2)$$

$$= 3.3 \times 10^{-4} \quad \text{(Eq. 4.2-10)}$$

The first term in Equation 4.2-9 represents the number of different channels which could be out for repair. The second term presents the unavailability of a logic unit. The third term presents the unavailability of two scram breakers, one from the twelve breakers for latch mechanisms and one from the twelve breakers for the drive-in motors. This third term assumes independent failure of the two scram breakers.

The assumptions of continuous monitoring and staggered testing for Event B are similar to those for the sensors of Event A. However, to simplify the expression for the probability of independent failure, use has been made of the fact that non-staggered periodic testing leads to a higher failure probability than that of a staggered testing. Therefore:

$$Pin_{indep} \leq U_R \binom{3}{1} (\lambda_{LU}T/2 + [(q_d+\lambda_{SB}T/2)\binom{12}{1}]^2)x0.01^{\ 2} \quad \text{(Eq. 4.2-11)}$$

where T = test period for a channel
= 4 hours

In Equation 4.2-11, the bracketed terms correspond to independent failure of two channels which are periodically tested at the same time (i.e., not in a staggered fashion as it is the case for the PRISM RPS). This gives an upper bound for the failure probability.

Substituting the proper values in Equation 4.2-11, we get

$$Pin_{dep} \leq 1.7 \times 10^{-16} \quad \text{(Eq. 4.2-12)}$$

Dependent failure probability is estimated using an equation similar to Equation 4.2-3 with proper account for inner channel breaker dependent failures. Thus,

$$P_{dep} = [\lambda_{LU} \times 4/2 \times 0.01 + (q_d + \lambda_{SB}\, 4/2) \times 0.01 \times \beta_{SB}]\ \beta_c\ \beta_{mc} \quad \text{(Eq. 4.2-13)}$$

In Equation 4.2-13, the first term in brackets refers to undetected logic unit failure in one channel. The second term refers to undetected dependent failure of two scram breakers in the same channel. The last two terms, β_c and β_{mc}, refer to the beta factors for dependencies across the channels, and dependencies of continuous monitoring failures.

Substituting the proper values in Equation 4.2-13, we get

$$P_{dep} = 2 \times 10^{-10} \quad \text{(Eq. 4.2-14)}$$

<u>Event C (Setpoint Errors)</u>

The RPS has at least four trip setpoints given an initiating event. Examples of these setpoints are power, power/flow ratio, and rate of change of some process parameter for each RPS division. It is assumed here that the RPS will shut down the reactor if two or more trip variables deviate beyond their respective setpoints. Erroneous input of setpoints by the operator may allow the corresponding process variable to deviate beyond safe limits given an initiating event.

As discussed earlier, the RPS will not operate unless setpoints for a particular parameter are the same for all divisions and match PCS setpoints. Therefore, if n is the number of trip parameters which will be first reached for an initiating event, then the number of input errors in the setpoints is given by

$$\begin{aligned} N &= 4 \text{ (RPS divisions) } \times n \text{ (trip parameters)} \\ &\quad + n \text{ (PCS file trip parameters)} \\ &= 5n \\ &= 5 \text{ for errors in a single trip parameter, e.g., flux} \\ &= 10 \text{ for errors in two trip parameters} \end{aligned}$$

To estimate the failure probability due to independent failure, P_{indep}, let:

- p = human or input device error leading to the above type of erroneous set point per set point;

then the probability of independent set point errors which lead to RPS failure given an IE, Event C, is given by:

$$P_{indep}(C|IE) = p^{5n} \qquad \text{(Eq. 4.2-15)}$$

Using the WASH-1400 human error probability of 10^{-3} per action, the above probability will have the following values for the number of setpoints of interest:

$$P_{indep}(C|IE) = 10^{-15} \quad \text{for n=1 parameters} \qquad \text{(Eq. 4.2-16)}$$

$$= 10^{-30} \quad \text{for n=2 parameters} \qquad \text{(Eq. 4.2-17)}$$

To estimate the probability of dependent setpoint errors, the following requirements for changing the RPS setpoints in PRISM have been considered.

1) Setpoints can only be changed when the reactor is shut down.

2) Setpoints must be changed independently in each RPS vault. As before, a probability of setpoint error in the RPS is assumed to be 10^{-3}.

3) Setpoint changes must follow restrictive administrative procedures. As before, a beta factor of 10^{-3} is assumed for dependent failures across channels.

4) Setpoints for all parameters are periodically checked by RPS. The setpoints must be consistent and exactly the same for similar parameters in all RPS channels to continue module power operation. Detection failure probability of 10^{-1} is assumed.

5) Setpoints must match a PCS file independently maintained by the reactor operator from the operator's control console. A probability of PCS file setpoint error of 10^{-3} is assumed.

Using the above probability values, the dependent set point error probability is given by:

$$P_{dep}\ (C|IE) = 10^{-10} \qquad \text{(Eq.4.2-18)}$$

Using the above probability estimates, the RPS failure probability is given by:

$$P_{indep}(\text{RPS Failure}|IE) \leq 2\text{x}10^{-15} + 1.7\text{x}10^{-16} + 10^{-15}$$

$$\leq 3\text{x}10^{-15} \qquad \text{(Eq. 4.2-19)}$$

$$P_{dep}\ (\text{RPS Failure}|IE) = 1.3 \times 10^{-9} \text{ for TOP}$$
$$= 2.3 \times 10^{-9} \text{ for LOF}$$
$$= 3 \times 10^{-10} \text{ for LOHS}$$

The above estimate strictly applies for initiating events where failure rates are comparable to those used in the above development. In particular, the estimates do not apply for earthquakes $\geq$ SSE as discussed below.

A preliminary assessment of components failure under SSE loading has been made using generic fragility data. It was found that relay failure probability given an SSE is in the order of 10^{-3}. This leads to the following estimate of event B probability:

$$P(B|SSE) \sim (12 \times 10^{-3})^2 \quad 3 \qquad \text{(Eq. 4.2-20)}$$
$$\sim 10^{-11}$$

Based on the obtained fragility analysis, it was judged that other events will contribute negligibly to the RPS failure. Since the above estimate is less than the dependent failure estimate for reactivity insertion, that estimate will be used, hence:

$$P(\text{RPS Failure}|SSE) \sim 1.3 \times 10^{-9} \qquad \text{(Eq. 4.2-21)}$$

Fragility analysis for seismic events beyond SSE has not been conducted but the probability of RPS failures given such events is expected to be higher than that of Equation 20.

A4.2.2.4 Results

Conditional probabilities of RPS failure given an initiating event have been estimated using the fault tree analysis and quantification described above. The results are shown in Table A4.2-5.

The results show dependence of the RPS on the initiating events. Dependency modeling in this analysis has been confined to three types:

(i) dependence on the seismic loading for SSE,

(ii) dependence on the process parameters and instrument types which respond to the initiating event.

(iii) common cause dependence using judgmental beta factor values.

The results of Table A4.2-5 confirm the high reliability expected from the RPS due to the redundancy, diversity, and testing strategy incorporated. The contributions of dependent and independent failures to sensor failure, RPS channel failure and setpoint error are displayed in Figure A4.2-28. As shown in the figure, RPS failure is caused almost exclusively by dependent failure. Sensor failures are the main contributors to RPS failure given TOP and LOF initiators. Notice that the sensor failure probaiblity has been based on the first process parameters to exceed its trip point without credit for subsequent trips by other process parameters. Consequently, the RPS failure probabilities for TOP and LOF initiators are conservative.

A4.2.3 Reactor Shutdown System Reliability:

The PRISM Shutdown System (RSS) consists of six Control Rod Units (CRU) which are used for reactivity control purposes. The absorber bundles of these control rod units are partially inserted into the core during normal reactor full power operation. Movement of the absorber bundle in and out of the core during normal reactor operation is controlled by the Plant Control System (PCS). In response to a scram demand by the Reactor Protection System (RPS), all the available CRU's will be inserted into the core by either a quick release mechanism or a scram drive-in motor.

Fault tree analysis is used to estimate the failure probability of the RSS given a scram initiating event. Fault trees, in general, describe ways in which the system failure can occur, and then is used to determine the probability of system failure. Given a particular undesired failure event, the fault tree identifies various combinations of failure events that lead to the undesired failure event under consideration.

The reactor shutdown system has the capability to shutdown the reactor and maintain it subcritical at cold shutdown condition of 450°F average temperature, with adequate margin for uncertainty in reactivity factors. Two different success criteria are used in response to all initiating events:

1) Reactivity Insertion Events:

Insertion of one out of five control rods (absorber bundle) is adequate for successful reactor shutdown (one control rod which is assumed to be withdrawn from the core in this type of event is considered to be unavailable).

2) Other Initiating Events

One out of six control rods is required for successful shutdown.

Each control rod is capable of inserting the absorber bundle in two bascially different and effectively diverse ways; these are:

1) A fast acting quick release mode whereby the absorber bundle is decoupled and thereby freed to be inserted into the core by force of gravity. This is accomplished by a conventional scram mechanism which is initiated by cutting off electrical power to an electromagnet that holds the latch in which the handle of the absorber bundle is held in the up (closed) position.

2) A slower acting mode whereby the absorber bundle is lowered into the core by means of electrical drive motor. A power scram electrical drive motor, switched on by the RPS, drives in the absorber bundle. The motor is capable of overcoming any credible resistance developed between the absorber bundle and the stationary channel.

The only credible common cause failure (multiple concurrent and dependent failures) which could prevent simultaneous insertion of all six control rods is postulated to occur if the core support platform should grossly bow (buckle). Major bowing could result from suddent (and uneven) temperature changes but gross bowing would have to be seismically induced. The common cause failure is represented by a single block attached to the top event by an 'OR' gate logic symbol. To evaluate the common cause failure probability a beta factor of 0.05 is conservatively assumed for the reactivity insertion initiating events and initiating events involving core blockage or EM pump failure. The median capacity of core support platform fragility is assumed at 2.1g to evaluate seismically induced common mode failure. A beta factor of 0.001 is assumed for all other initiators.

The fault trees in Figures A4.2-29 through A4.2-33 are constructed for the RSS reliability evaluation for three different types of accident initiators. The fault tree for single control rod is depicted in Figure A4.2-34. The failure probability data in Table A4.2-6 was used for fault tree quantification. The conditional probabilities of RSS failure given each initiating event are presented in Table A4.2-7.

A4.2.4 Pump Trip Event Probability

In accordance with the system event trees of this section, two failure modes of pump trip are of interest:

(1) failure to trip the pumps when a trip signal from the RPS is received, and

(2) failure <u>not to</u> trip the pumps when <u>no</u> signal from the RPS is received.

The first failure mode (failure to trip the pumps when a trip signal is received) is of particular interest if the reactor is shut down but the shutdown heat removal system is unavailable. In this case, the primary sodium temperature will increase by decay heat and primary pump power addition. Analysis of this case shows that the sodium heatup is extremely slow (taking ~24 hours to reach sodium boiling). For such grace period, manual pump trip by the operator is almost certain to occur before significant primary sodium heatup. Based on probability considerations of manual trip failure and of other accident sequences which may lead to similar primary sodium heatup, the first failure mode, in the case of slow transients, was judged as insignificant.

The second failure mode (failure not to trip the pumps when no signal from the RPS is received) leads to a loss-of-flow accident if the pump trip is accompanied by failure to scram. At the time this assessment was made, no design existed for the pump trip circuit or logic. Consequently, a trip circuit identical with the scram latch release logic (Figure A4.2-24) was assumed for this analysis. Only the automatic trip circuit was considered in the analysis. Using Figure A4.2-24, the following failure criteria were obtained:

1) At least 3 out of 4 relays must fail closed for failure to trip the pump when a trip signal is received, and

2) At least 2 out of 4 relays must fail open for failure not to trip the pumps when no trip signal is received.

Using the above criteria, the following results were obtained for all initiating events except seismic events, station blackout event, or the initiating event of loss of substantial coolant flow due to pumps failure.

Probability of failure to trip the pumps given a trip signal and the initiating event = 1.2×10^{-9}.

Probability of pump trip given no trip signal and the initiating event = 4×10^{-6}.

For initiating events involving seismic events, judgment based on the analysis of Section A4.2.2.3 was used. Table A4.2-8 contains the obtained conditional failure probabilities given each initiating event.

A4.2.5 Primary Pump Coastdown System Reliability

In the event the primary pumps are tripped (i.e., interruption of power to the pumps), a controlled coastdown of the pumps is required to prevent reactor core temperatures from exceeding acceptable limits following a scram failure. The primary pumps are electromagnetic (EM) and require power during a coastdown for about two minutes. A synchronous machine (motor-generator) with a solid state controller is used to supply power and to control the EM pump during coastdown. One coastdown power supply system is provided for each EM pump. A complete coastdown system (which includes the EM pump) consists of five basic elements as follows:

1) EM Pump
2) Synchronous Machine

3) Regulator
4) Circuit Breaker
5) Housing (Structure)

Pump coastdown failure is defined here as the failure of at least three out of four EM pumps. The fault tree diagrams shown in Figures A4.2-35 through A4.2-38 were constructed for the failure probability evaluation.

The only credible common cause failure which fails at least 3 out of 4 power supplies simultaneously is postulated to be a large earthquake. A beta factor of .005 is used for the three simultaneous coastdown system failures other than the seismically initiated system failure. Fragility analysis was used to evaluate the seismically induced component and system failure probabilities. The failure probability and fragility data are listed in Table A4.2-9. The conditional probabilities of primary coastdown failure given each initiating event are shown in Table A4.2-10.

A4.2.6 Inherent Reactivity Feedback System Reliability

The PRISM reactor has the distinctive inherent safety characteristics of limiting the rate and extent of power increase when the fuel temperature or the primary sodium temperature increases. This is accomplished by:

1) negative fuel temperature reactivity coefficient (Doppler, axial fuel expansion), and

2) negative coolant temperature coefficient (control rod drive line expansion, core radial expansion).

The magnitude and effectiveness of these coefficients depend on the initiating event, as-built fuel and control rod assemblies, and irradiation history. They also depend on failures which limit the magnitude of the negative coolant temperature reactivity coeffieint of Item 2 above. Only these failures are of interest in the system event tree analysis discussed in this section. These failures define boundary conditions for the analysis of the core response event trees in th4e next section, where other factors affecting the madnitude and effectiveness of the inherent reactivity feedback are considered probabilistically.

As defined in this section, the inherent reactivity feedback system includes those parts of the core which provide a net negative reactivity feedback when the primary sodium heats up. The change in reactivity results from three distinctive sources:

1) Control rod motion relative to the core. This may be induced by CR extension tube expansion or reactor vessel expansion.

2) Fuel subassembly bowing or dilation of the subassembly load pads which have the effect of reducing the effective fuel density in the core region.

3) Core support grid plate expansions which also reduces the effective fuel density in the core region.

To allow these feedbacks to occur, the control rods must be able to move in their guide tubes, the fuel subassemblies must be able to move against the core restraint system and the grid plate must be able to expand. In this work, failure of all control rods to move, or structural failures which prevent the fuel subassemblies from moving or extending in the right geometry, or both, constitutes the failure of the inherent reactivity feedback system.

From the system event trees presented in this section, two conditions of the control rods have to be considered:

1) one condition for which the control rods are stuck, and

2) the other condition when the control rods can move but the inherent feedback fails due to structural failure.

The conditional probability of the inherent feedback system given stuck rods has been estimated using the fault trees of Section A4.2.3.

Study of these trees showed that the conditional probability of stuck control rods account for about 10% of the failure probability to insert the control rods. Therefore, the following conditional probability has been used for all initiating events except seismic events and the vessel failure event:

Probability of inherent safety system failure
given RSS has failed = .1

The probability of structural failures has been assigned by judgment as follows: given that the CR's can move, the probability has been assigned as 10^{-6}, for initiating events not including siesmic events greater than SSE or the vessel failure event. For the vessel failure event or an earthquake >0.825g, the probability of losing the inherent feedback system has been assigned the value 1 if the RSS has failed, and the value 0.03 if the CR's can move freely. For the intermediate seismic event (0.375g to 0.825g), the corresponding probabilities were 0.5 if RSS fails and 1.2×10^{-5} if the CR's can move freely.

Table A4.2-11 contains the conditional probability of the inherent reactivity feedback failure given each initiating event.

A4.2.7 Operating Power Heat Removal System Failure

The operating power heat removal system includes the IHTS, steam generator, T/G set and the balance of plant required to remove the heat from a module operating at power. Some of the initiating events defined in this PRA mean that the system is unavailable; e.g., the initiating events of loss of IHTS pump, loss of heat removal via balance of plant. For other initiating events, mechanical components of the system are available for the short periods of interest in the system event tree analysis. The system control however may fail in two ways: 1) failure to initiate system isolation when a control signal is received, or 2) failure not to initiate system isolation when no signal is received.

The above failure modes are similar to those considered for pump trip in Section A4.2.4. Using the same approach in that section with appropriate control system failure rates the following results were obtained for all initiating events except seismic events and events including failure of the operating heat removal system.

Probability of failure to actuate system isolation given a trip signal and the initiating event = 1.2×10^{-6}

Probability of failure not to actuate system isolation given no trip signal and the initiating event = 4×10^{-4}.

For initiating events involving seismic events, judgment based on the analysis of Section A4.2.2.3 was used. Table A4.2-12 contains the obtained conditional failure probabilities given each initiating event.

A4.2.8 Shutdown Heat Removal System Reliability

Figure A4.2-39 presents the generic reliability model which was used for assessing the reliability of this system. As indicated on the top system level reliability block diagram (Figure A4.2-39-a), the residual heat removal process can be visualized as taking place in two stages. In the first stage the decay heat load is transferred from the reactor core to the primary coolant by circulating primary coolant through the fuel assemblies. As shown in Figures A4.2-39-b and -c, the first stage of the operation is entirely dependent upon the structural integrity of the primary reactor structures (Block 110) and integrity of the primary coolant boundary (Block 120)

In the second stage of the process, the residual heat load is removed from the primary coolant and transferred indirectly to the ultimate heat sink (atmospheric air) by one of three paths, which, in order of usage preference, are as follows:

<u>Path 1</u>:

Path 1 is used for normal reactor shutdown operations. To bring the reactor temperatures from full power down to hot standby (540°F), steam is routed through the turbine (Block 141) and the main condenser

(Block 144). Bringing the reactor down to 400°F for refueling requires steam to be bypassed to the main condenser through the turbine bypass valving (142 series blocks).

Path 2:

Path 2 is used in the event normal condenser cooling (i.e. Block 144) is not available. In this mode, the auxiliary cooling system (Block 130) removes heat from the shell side surface of the steam generator (included in Block 136) by natural circulation air flow.

Path 3:

Path 3 is formed by the reactor vessel air cooling system (RVACS) which removes heat directly from the reactor vessel. RVACS Block 160 relies solely upon natural air draft which is continually monitored for air flow rate and temperature. Decay heat load removal is assured in this mode if RVACS operates at ten percent or more of its rated capacity.

For purposes of this assessment it is assumed that pony motor driven operation of the mechanical (centrifugal) Intermediate Heat Transport System (IHTS) pump (Block 137) is required for operation of the systems represented by both paths 1 and 2.

Path 1 is dependent upon active operation of the systems which from the turbine-generator island (referred to as the Balance of Plant (BOP) in this section, and identified as Block 140). But the PRISM modular approach to plant design brings with it certain operational advantages that tend to enhance BOP reliability when one or more reactors are shutdown and in the decay heat removal mode.

For example, shutdown of one reactor following full power operation of the power block results in a reduced load to the BOP. The BOP can respond to this reduction by cutting back from a three feedwater pump system (Block 143) operation to two, and from what amounts to essentially a two main condensate system (Block 144) train which are each sized at two-thirds system capacity to a single train operation. The systems/equipment taken out of service represent a reserve capacity of on-line equipment which becomes immediately available to replace any equipment that fails during the decay heat removal period. Furthermore, the margin of reserve increases with each successive reactor shutdown. This reserve capacity which exists during the shutdown heat removal operation is reflected in the reliability block diagram covering the BOP (Figure A4.2-39-e) by showing the individual trains involved in a redundant arrangement.

Prior to occurrence of the initiating event the overwhelming majority of equipment is continuously monitored for operability status. Either the equipment is required to operate to support the normal power operation (e.g., IHTS, BOP) or the system is operating and being monitored for performance whether it is needed or not (e.g., RVACS). All other equipment (e.g., louvered damper actuators used with the ACS and the IHTS pump pony motor) are amenable to being checked for operability status during normal power operation without detrimental effect to itself or normal power operation. For this reason, it is assumed that the availability of all equipment is 1.0 at the beginning of the decay heat removal mission. The exceptions to these are, of course, those instances where failure of an equipment itself constitutes the shutdown initiating event. These exceptions are identified in Table A4.2-13.

Table A4.2-14 lists each system element represented in the reliability block diagrams of Figure A4.2-38 and gives the associated time dependent failure rate and repair rates for each. These data were assigned by judgment based on operating experience or conservative estimating techniques such as those used on the CRBRP and LSPB projects.

Table A4.2-13 presents initiator dependent failure probability values. Also shown in column b of this table are the residual heat removal operation mission times applicable to the situation associated with each initiating event.

The FRANCALC-1 computer code was used to compute the dependent failure probabilities presented in Table A4.2-15. No grace period was assumed. The times to failure and repair events were assumed to be exponentially distributed. Values are shown for the following three cases.

Case 1:

Conditional probability that decay heat removal via IHTS and BOP fails given the initiating event.

Case 2:

Conditional probability that decay heat removal via RVACS fails given Case 1.

Case 3:

Conditional probability that all shutdown heat removal paths have failed given the initiating event.

The results shown in Table 4.2-15 show significant dependence on the initiating events. Two factors are responsible for this dependence:

1) The duration needed for shutdown heat removal which varies between a minimum of eight hours for IE 11 to a maximum of 4380 hours for seven other initiating events.

2) The impact of the initiating event on the systems needed for shutdown heat removal.

REFERENCES - SECTION A4.2

A4.2-1 "Reactor Safety Study, "WASH-1400 (NUREG-75/014), USNRC, October 1975.

A4.2-2 "Nuclear Plant Reliability Data System - 1980 Annual Reports of Cumulative System and Component Reliability, "NUREG/CR-2232, USNRC, September 1981.

TABLE A4.2-1

DEFINITIONS OF ACCIDENT TYPES

Accident Type	Definition
S3	LOSHR with reactor shut down and no initial core damage
S5	LOSHR with reactor shut down but with initial partial core damage or blockage (Fermi I type accident), or with added heat due to initial transient.
P1	TOP with reactivity addition of $0.07 to $0.18.
P2	TOP with either (1) reactivity addition of $0.18 to $0.36 or (2) smaller reactivity addition with loss of inherent reactivity feed back.
P3	TOP with either (1) reactivity addition >$0.36 or (2) reactivity addition of $0.18 to $0.36 with loss of inherent reactivity feedback.
P4	TOP with both reactivity addition >$0.36 and loss of inherent reactivity feedback.
P1S,...,P4S	Same as P1,...,P4, respectively, except that the accident is also accompanied by LOSHR.
F1	LOF due to pump trip with failure to scram but with successful flow coastdown and inherent reactivity feedback.
F3	Same as F1, except with failure of flow coastdown or failure of inherent reactivity feedback, or both.
F3S	Same as F3, except that the accident is also accompanied by LOSHR.
H2	ULOHS resulting from loss of heat removal capability with failure to scram either (1) at nominal power with loss of inherent feedback due to stuck CR's, or (2) at an elevated power of up to 125% nominal.

TABLE A4.2-1
(Continued)

DEFINITIONS OF ACCIDENT TYPES

Accident Type	Definition
H3	ULOHS due to loss of heat removal capability with failure to scram either (1) at up to 125% with loss of inherent feedback or (2) at power >125%.
H1S	ULOHS with failure to scram at nominal power with successful inherent reactivity feedback but with LOSHR.
H2S, H3S	Same as H2 and H3, except that the accidents are also accompanied by LOSHR.
G3	A combined P2/F3 or P3/F1.
G4	A combined P4/F1 or P3/F3.
G1S	A combined P2/F1 or P1/F1 with LOSHR.
G3S, G4S	Same as G3 and G4, except that the accidents are also accompanied by LOSHR.

TABLE A4.2-2

RPS PARAMETER LIST

Parameter-Sensor	RPS Trip Supported
Flux - Wide Range[1]	Wide range absolute flux Rate of change in flux Flux/pump discharge pressure (Power to flow ratio)
Flow - Pump discharge pressure	Primary coolant flow Flux/pump discharge pressure (Power to flow ratio)
Temperature -	
Core outlet temperature	Core outlet absolute temperature
Cold pool temperature	Cold pool absolute temperature Loss of IHTS (Rise in cold pool temperature)
Pressure -	
Pump discharge pressure	Flux/pump discharge pressure (Power to flow ratio)
Level -	
Primary coolant level	Level (Rate of change of level and absolute level)
Electrical Power -	
Instrument power supply voltage	Loss of instrument power

1 - Flux is a measure of power

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TABLE A4.2-4 - FIRST RPS TRIPPED PARAMETER GIVEN AN INITIATING EVENT

		First Setpoint Reached			
Event Number	Event Name	Flux	Flux/ Flow	Primary Na Level	Cold Pool or Core Outlet Temp.
1	Reactivity Insertion 0.07$ to 0.18$	√			
2	Reactivity Insertion 0.18$ to 0.36$	√			
3	Reactivity Insertion >0.36$	√			
4	Earthquake 0.3g to 0.375g	√			
5	Earthquake 0.375g to 0.825g	√			
6	Earthquake >0.825g	√			
7	Vessel Fracture	√		√	
8	Local Core Coolant Blockage	√			
9	Reactor Vessel Leak			√	
10	Loss of One Primary Pump		√		
11	Loss of Substantial Primary Flow		√		
12	Loss of Operating Power Heat Removal				√
13	Loss of Shutdown Heat Removal via BOP				√
14	Loss of Shutdown Heat Removal via IHTS				√
15	IHTS Pump Failure				√
16	Station Blackout		√		
17	Na-H_2O Reaction IHX Failure				√
18	Spurious Scram and Transients inadequately handled by PCS*		√		
19	Normal Shutdown - NA**				
20	Forced Shutdown - NA				
21	RVACS Blockage - NA				

* : First setpoint reached depends on cause. Conservatively use Flux/Flow.

**: Not Applicable: Shutdown by PCS or Manually .

TABLE A4.2-5

CONDITIONAL (PER DEMAND) FAILURE PROBABILITY
OF RPS GIVEN INITIATING EVENT

Initiating Event No.	Conditional Probability
1	1.3 E-9
2	1.3 E-9
3	1.3 E-9
4	1.3 E-9
5	1.0 E-6
6	1.0 E-3
7	3.0 E-10
8	1.3 E-9
9	1.3 E-9
10	2.3 E-9
11	2.3 E-9
12	3.0 E-10
13	3.0 E-10
14	3.0 E-10
15	3.0 E-10
16	2.3 E-9
17	3.0 E-10
18	2.3 E-9
19	NA
20	NA
21	NA

NA : Not Applicable. Shutdown by PCS or Manually.

TABLE 4.2-6

FAILURE RATE AND TEST INTERVAL DATA FOR RSS

Component	Failure Mode	Time Dependent: λt (failure/hr	Time Dependent: Test Interv (T) (hours)	Demand-Dependent: λd (failure/demand	Total Failure Probability (per demand)
EM Latch Holder Assembly	Binding of Armature	$1x10^{-9}$	17,520	-	$8.76x10^{-6}$
Tension Tube to Driveline Interface	Binding	$1x10^{-7}$	17,520	1x10-3	$1.88x10^{-3}$
Latch Assembly	Seizure	$1x10^{-6}$	17,520	1x10-2	$1.88x10^{-2}$
Absorber Bundle	Binding	$1x10^{-6}$	17,520	$1x10^{-3}$	$1.88x10^{-3}$
Scram Motor	Fail to Operate	$1x10^{-8}$	24	$1x10^{-5}$	$1.01x10^{-5}$
Electrical Power Supply	Open	$1.1x10^{-5}$	24	-	$1.32x10^{-4}$
Gear Assembly	Fail to Operate	$1x10^{-7}$	24	$1x10^{-5}$	$1x12x10^{-4}$
Carriage Assembly	Fail to Operate	$1x10^{-8}$	17,520	$1x10^{-5}$	$9.76x10^{-5}$
Control Rod Structure	Structural Failure	$1x10^{-8}$	24	-	$1.2x10^{-7}$

TABLE A4.2-7

CONDITIONAL (PER DEMAND) FAILURE PROBABILITY
RSS

Initiating Event No.	Conditional Scram Failure Probability
1	$2.89x10^{-7}$
2	$2.89x10^{-7}$
3	$2.89x10^{-7}$
4	$3.47x10^{-8}$
5	$1.23x10^{-5}$
6	$3.0x10^{-2}$
7	1.0
8	$2.89x10^{-7}$
9	$5.78x10^{-9}$
10	$2.89x10^{-7}$
11	$2.89x10^{-7}$
12	$5.78x10^{-9}$
13	$5.78x10^{-9}$
14	$5.78x10^{-9}$
15	$5.78x10^{-9}$
16	$5.78x10^{-9}$
17	$5.78x10^{-9}$
18	$5.78x10^{-9}$
19	$5.78x10^{-9}$
20	$5.78x10^{-9}$
21	$5.78x10^{-9}$

TABLE A4.2-8

CONDITIONAL (PER DEMAND) PROBABILITY OF NO PUMP TRIP

Initiating Event No.	Given Signal	Probability	Initiating Event No.	Given Signal	Probability
1	Yes	1.2 E-9	12	Yes	1.2 E-9
	No	1-4 E-6		No	1-4 E-6
2	Yes	1.2 E-9	13	Yes	1.2 E-9
	No	1-4 E-6		No	1-4 E.6
3	Yes	1.2 E-9	14	Yes	1.2 E-9
	No	1-4 E-6		No	1-4 E-6
4	Yes	1.2 E-9	15	Yes	1.2 E-9
	No	1-4 E-6		No	1-4 E-6
5	Yes	6.0 E-9	16	Yes	0
	No	1-4 E-6		No	0
6	Yes	0	17	Yes	1.2 E-9
	No	0		No	1-4 E-6
7	Yes	0	18	Yes	1.2 E-9
	No	0		No	1-4 E.6
8	Yes	1.2 E-9	18	Yes	1.2 E-9
	No	1-4 E-6		No	1-4 E-6
9	Yes	1.2 E-9	19	Yes	1.2 E-9
	No	1-4 E-6		No	1-4 E.6
10	Yes	1.2 E-9	20	Yes	1.2 E-9
	No	1-4 E-6		No	1-4 E-6
11	Yes	0	21	Yes	1.2 E-9
	No	0		No	1-4 E-6

1-4 E-X = $1\text{-}4 \times 10^{-X}$

TABLE A4.2-9

DATA USED FOR PUMP COASTDOWN RELIABILITY EVALUATION

Component	Equipment Fragility: Median Capacity	Equipment Fragility: Standard Deviation	Failure Probability (per demand): Nonseismic Initiators	Failure Probability (per demand): Seismic Initiator .3g	Seismic Initiator .6g	Seismic Initiator 1.2g
EM Pump	8.9 g	0.65	2.2×10^{-7}	2.2×10^{-7}	2.2×10^{-7}	2.2×10^{-7}
Synchronous Machine	12.1 g	0.65	5×10^{-7}	5×10^{-7}	5×10^{-7}	5×10^{-7}
Circuit Breaker	4.1 g	0.65	5×10^{-8}	5×10^{-8}	5×10^{-8}	5×10^{-8}
Regulator	2.72g	0.65	1×10^{-7}	1×10^{-7}	1×10^{-7}	1×10^{-7}
Housing (Structure)	1.1 g	0.45	~0	~0	1.55×10^{-3}	6.61×10^{-1}

TABLE A4.2-10

CONDITIONAL FAILURE PROBABILITY
PUMP COASTDOWN SYSTEM

Initiating Event Number	Pump Coastdown Conditional Failure Probabiliity
1	$4.35x10^{-9}$
2	"
3	"
4	$4.35x10^{-9}$
5	$1.94x10^{-8}$
6	$5.83x10^{-1}$
7	$4.35x10^{-9}$
8	$4.35x10^{-9}$
9	$4.35x10^{-9}$
10	$4.35x10^{-9}$
11	$4.35x10^{-9}$
12	$4.35x10^{-9}$
13	$4.35x10^{-9}$
14	$4.35x10^{-9}$
15	$4.35x10^{-9}$
16	$4.35x10^{-9}$
17	$4.35x10^{-9}$
18	$4.35x10^{-9}$
19	$4.35x10^{-9}$
20	$4.35x10^{-9}$
21	$4.35x10^{-9}$

TABLE A4.2-11

CONDITIONAL FAILURE PROBABILITY OF INHERENT FEEDBACK

Initiating Event No	RSS Faiure	Conditional Failure Probability	Initiating Event No.	RSS Failure	Conditional Failure Probability
1	Yes	1. E-1	12	Yes	1. E-1
	No	1. E-6		No	1. E-6
2	Yes	1. E-1	13	Yes	1. E-1
	No	1. E-6		No	1. E-6
3	Yes	1. E-1	14	Yes	1. E-1
	No	1. E-6		No	1. E-6
4	Yes	1. E-1	15	Yes	1. E-1
	No	1. E-6		No	1. E-6
5	Yes	5. E-1	16	Yes	1. E-1
	No	1.2 E-5		No	1. E-6
6	Yes	1. E-0	17	Yes	1. E-1
	No	3. E-2		No	1. E-6
7	Yes	1. E-0	18	Yes	1. E-1
	No	3. E-2		No	1. E-6
8	Yes	1. E-1	19	Yes	1. E-1
	No	1. E-6		No	1. E-6
9	Yes	1. E-1	20	Yes	1. E-1
	No	1. E-6		No	1. E-6
10	Yes	1. E-1	21	Yes	1. E-1
	No	1. E-6		No	1. E-6
11	Yes	1. E-1			
	No	1. E-6			

TABLE A4.2-12

CONDITIONAL PROBABILITY OF NO OPERATING HEAT REMOVAL

Initiating Event No.	Trip Signal	Conditional Failure Probability	Initiating Event No.	Trip Signal	Conditional Failure Probability
1	Yes	1-1.2 E-6	12	Yes	1
	No	4 E-4		No	1
2	Yes	1-1.2 E-6	13	Yes	1
	No	4 E-4		No	1
3	Yes	1-1.2 E-6	14	Yes	1
	No	4 E-4		No	1
4	Yes	1-1.2 E-6	15	Yes	1
	No	4 E-4		No	1
5	Yes	.5	16	Yes	1
	No	.5		No	1
6	Yes	1	17	Yes	1
	No	1		No	1
7	Yes	1-1.2 E-6	18	Yes	1-1.2 E-6
	No	4 E-4		No	4 E-4
8	Yes	1-1.2 E-6	19	Yes	1-1.2 E-6
	No	4 E-4		No	4 E-4
9	Yes	1-1.2 E-6	20	Yes	1-1.2 E-6
	No	4 E-4		No	4 E-4
10	Yes	1-1.2 E-6	21	Yes	1-1.2 E-6
	No	4 E-4		No	4 E-4
11	Yes	1-1.2 E.6			
	No	4 E-4			

TABLE A4.2-13

SHRS INITIATOR DEPENDENT EQUIPMENT FAILURE PROBABILITIES/RATES

Initiating Event (IE)	t_m(2)	Probability Failure Given IE					
		110	120	130	140	150	160
a	b	c	d	e	f	g	h
1. Reactivity Insert 0.07-0.18$	600	-	-	-	-	-	-
2. Reactivity Insert 0.18-0.36$	600	-	-	-	-	-	-
3. Reactivity Insert >0.36$	4380	-	-	-	-	-	-
4. Earthquake 0.3 to 0.375g	120	-	-	0.01	0.01	-	-
5. Earthquake 0.375 to 0.825g	4380	-	-	0.10	0.10	0.001	[10^2]
6. Earthquake >0.825g	4380	-	-	1.0	1.0	0.01	[10^4]
7. Vessel Fracture	4380	-	(#)	-	-	-	-
8. Local Core Coolant Blockage	4380	-	-	-	-	-	-
9. Reactor Vessel Leak	4380	-	(#)	-	-	-	-
10. Loss of One Primary Pump	600	-	-	-	-	-	-
11. Loss of Substantial Prim Flow	8	-	-	-	-	-	-
12. Loss of Oper Pwr Heat Removal	86	-	-	-	-	-	-
13. Loss of S/D Heat Removal via BOP	24	-	-	-	(1.0)	-	-
14. Loss of S/D Heat Rem via IHTS	600	-	-	(1.0)	-	-	-
15. IHTS Pump Failure	600	-	-	-	-	-	-
16. Station Blackout	1200	-	-	-	(#)	-	-
17. NaH_2O Reaction IHX Failure	4380	-	-	(1.0)	-	-	-
18. Spurious Scram and Transient Inadequately Handled by PCS	600	-	-	-	-	-	-
19. Normal Shutdown	600	-	-	-	-	-	-
20. Forced Shutdown	240	-	-	-	-	-	-
21. RVACS Blockage	86	-	-	-	-	-	(1.0)

Notes:

t_m = mission time in hours = expected (or mean) time required to restore to normal power operation

() = values shown in parentheses are assigned by definition of the associated initiating event

= signifies that some portion of this system is assumed to be failed (block 121 for IE 7 and IE 9 and block 141 for EI 16. See Table A4.2-14 for definition of blocks.

[] = degradation factor used to multiply the probability of failure due to random causes

TABLE A4.2-14

SHRS TIME DEPENDENT FAILURE AND REPAIR RATES

Block I. D.	Major Subsystem/Equipment/Feature	Failure Rate ($x10^{-6}$f/hr)	Repair Rate ($x10^{-3}$r/hr)
110	Primary Coolant Flow Path (Fig. A4.2-39b)		
111	Reactor Foundation (Silo & Superstructure)	*	
112	Reactor Module Support Structure	*	
113	Reactor Vessel & Cont. Vessel (as support Struct.)	*	
114	Reactor Vessel Head Structure	*	
115	Reactor Vessel Internals	*	
120	Primary Coolant Boundary (Fig A4.2-39c)		
121	Reactor Vessel (leak integrity)	0.0001	0.06
122	Containment Vessel (leak integrity)	0.0001	0.06
130	Secondary Coolant Boundary (Fig A4.2-39d)		
131	Intermediate Heat Exchanger (IHX)	0.1	2.5
132	IHTS Piping (leak integrity)	1.0	1.7
133	IHTS Pump Housing (leak integrity)	1.0	1.7
134	Ancillary IHTS Svc Supp System (leak integ.)	1.0	250
135	IHTS Structural Support System	0.0001	0.45
136	Steam Generator (leak integrity)	0.02	1.25
137	IHTS Pump (Pony motor driven pump operation)	5.0	1.7
140	Balance of Plant (BOP) (Fig A4.2-39e)		
141	Turbine-Generator Set (system)	10.0	4.0
142	Turbine Bypass Valve	10.0	40.0
143	Main Feedwater System Train	100.00	20.0
144	Main Condensate System Train	100.00	20.0
150	Auxiliary Cooling System (ACS) (Fig A4.2-39f)		
151	Steam Generator Shroud Structure	0.00001	42.0
152	Louvered Damper System	0.1	250
153	Power Driven Damper Actuator	0.01	250
154	Manual Damper Actuator	0.01	250
160	Reactor Ves. Air Cool. Sys. (RVACS) (Fig A4.2-39g)		
161	Air Vent System	0.000001	42.0
162	Material Surface Emissivity Characteristics	0.000001	0.06
170	Off-Site Electrical Power Supply (Fig A4.2-39h)		
171	Preferred Off-site	10.0	2000.0
172	Reserved Off-site	10.0	2000.0
180	On-Site Electrical Power Supply (Fig A4.2-39h)		
181	Associated Power Block	100.0	1.0
182	Sister Power Block	100.0	1.0

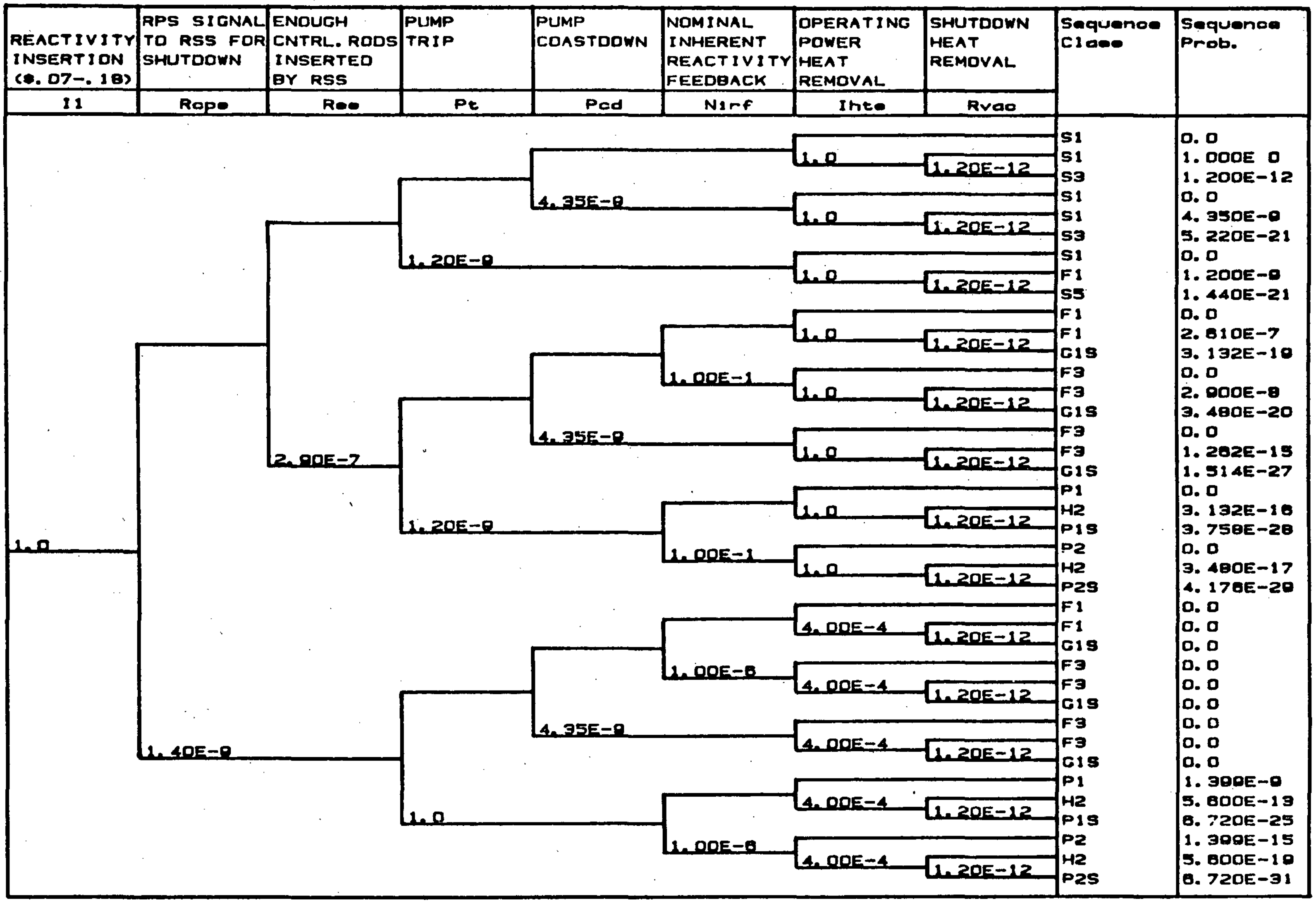
* = Negligibly small failure rate except possibly under extreme earthquake initiating event.

TABLE A4.2-15

CONDITIONAL FAILURE PROBABILITY OF SHRS

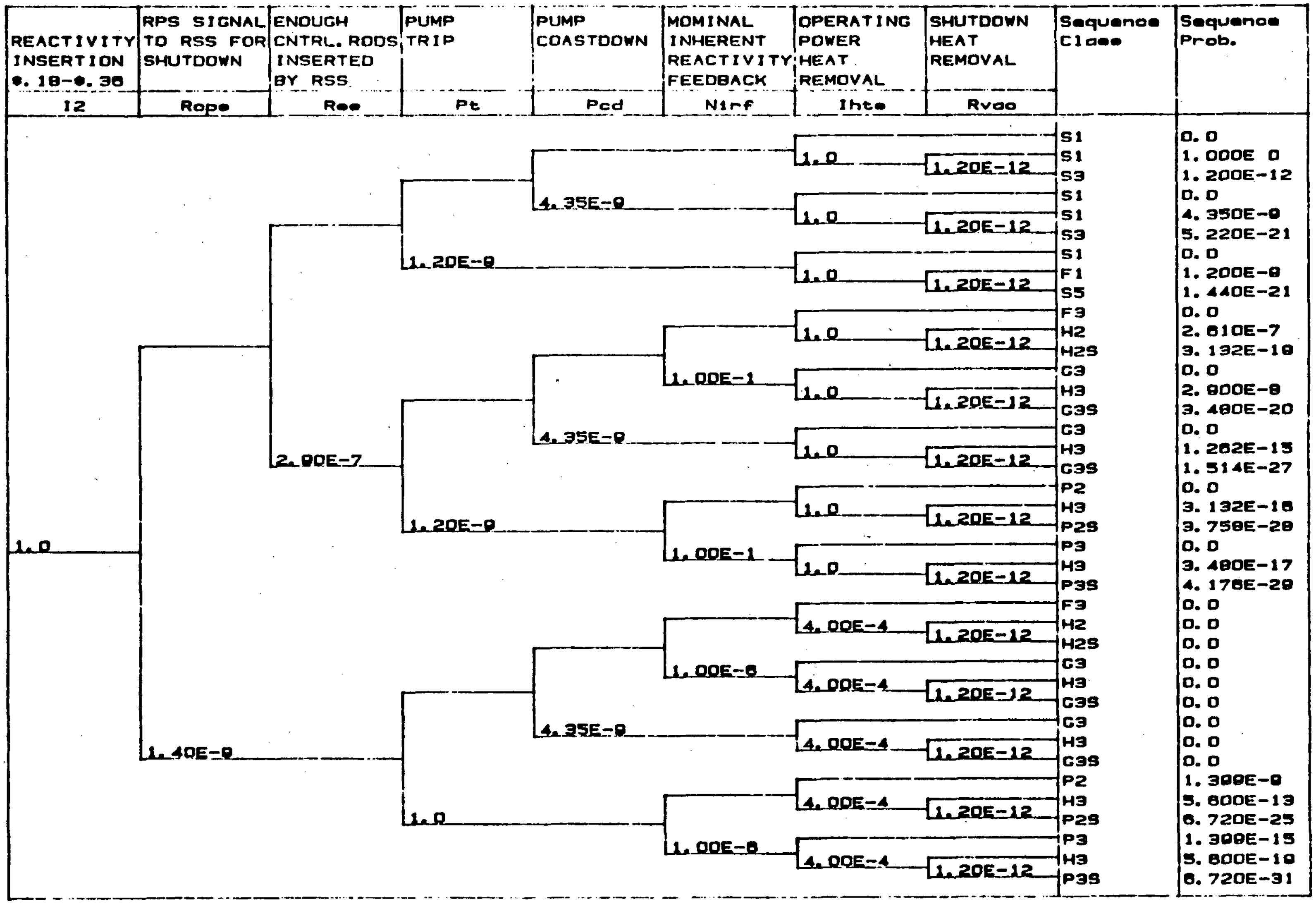
Initiating Event (IE)	Via IHTS & BOP Given IE	Via RVACS Given Failure Via IHTS,BOP and IE	SHRS Failures
1. Reactivity Insertion 0.07 to 0.18$	2.0E-3	6. E-10	1.2 E-12
2 Reactivity Insertion 0.18 to 0.36$	2.0E-3	6. E-10	1.2 E-12
3. Reactivity Insertion $\geq$.36$	1.5E-2	4.4 E-9	6.6 E-11
4. Earthquake 0.3 to 0.375g	6.0E-3	1.2 E-10	7.2 E-13
5. Earthquake 0.375 to 0.825g	1.0E-1	4.4 E-7	4.4 E-8
6. Earthquake >0.825g	1.0E+0	4.4 E-5*	4.4 E-5*
7. Vessel Fracture	1.0E+0	4.4 E-7	4.4 E-7
8. Local Core Coolant Blockage	1.4E-2	4.4 E-9	6.2 E-11
9. Reactor Vessel Leak	1.4E-2	4.4 E-7	6.2 E-9
10. Loss of One Primary Pump	2.0E-3	6.0 E-10	1.2 E-12
11. Loss of Substantial Prim Flow	1.9E-5	1.6 E-11	3.0 E-16
12. Loss of Oper Pwr Heat Removal	2.8E-4	8.6 E-11	2.4 E-14
13. Loss of S/D Heat Removal via BOP	7.9E-5	4.8 E-11	3.8 E-15
14. Loss of S/D Heat Rem via IHTS	1.0E+0	6.0 E-10	6.0 E-10
15. IHTS Pump Failure	2.0E-3	6.0 E-10	1.2 E-12
16. Station Blackout	2.9E-3	1.2 E-9	3.5 E-12
17. NaH_2O Reaction IHX Failure	1.0E+0	4.4 E-9	4.4 E-9
18. Spurious Scram and Transient Inadequately Handled by PC	2.0E-3	6.0 E-10	1.2 E-12
19. Normal Shutdown	2.0E-3	6.0 E-10	1.2 E-12
20. Forced Shutdown	7.9E-4	2.4 E-10	2.0 E-13
21. RVACS Blockage	2.8E-4	1.0 E+0	2.8 E-4

* These valves apply when the seismic isolators function successfully. In case of isolator failure, these values should be replaced by 1.0.



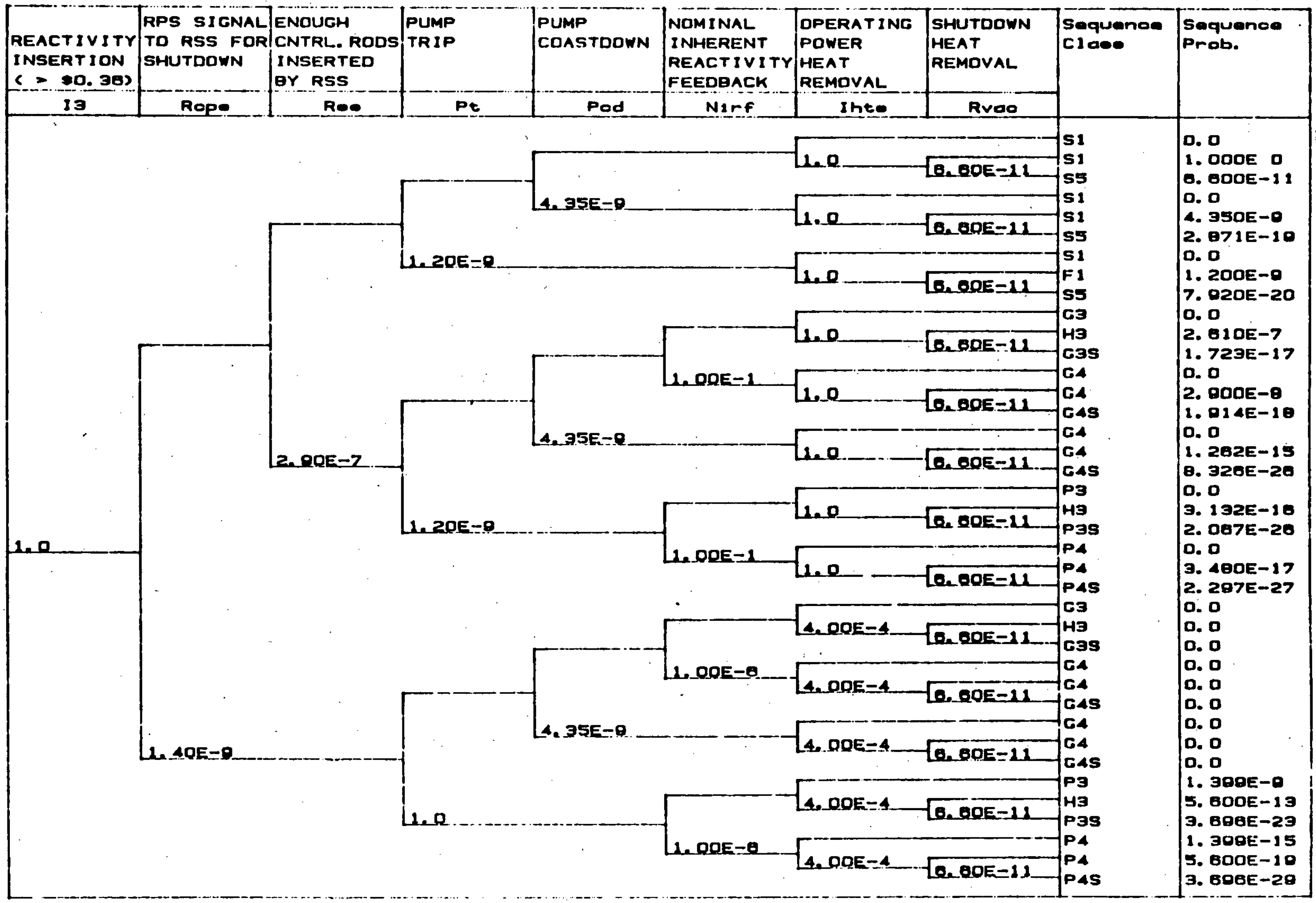
B: I1.TRE10-05-1987 SYS.R.TREE IE1: Reactivity Inser. (%.07-%.18)

Figure A4.2-1



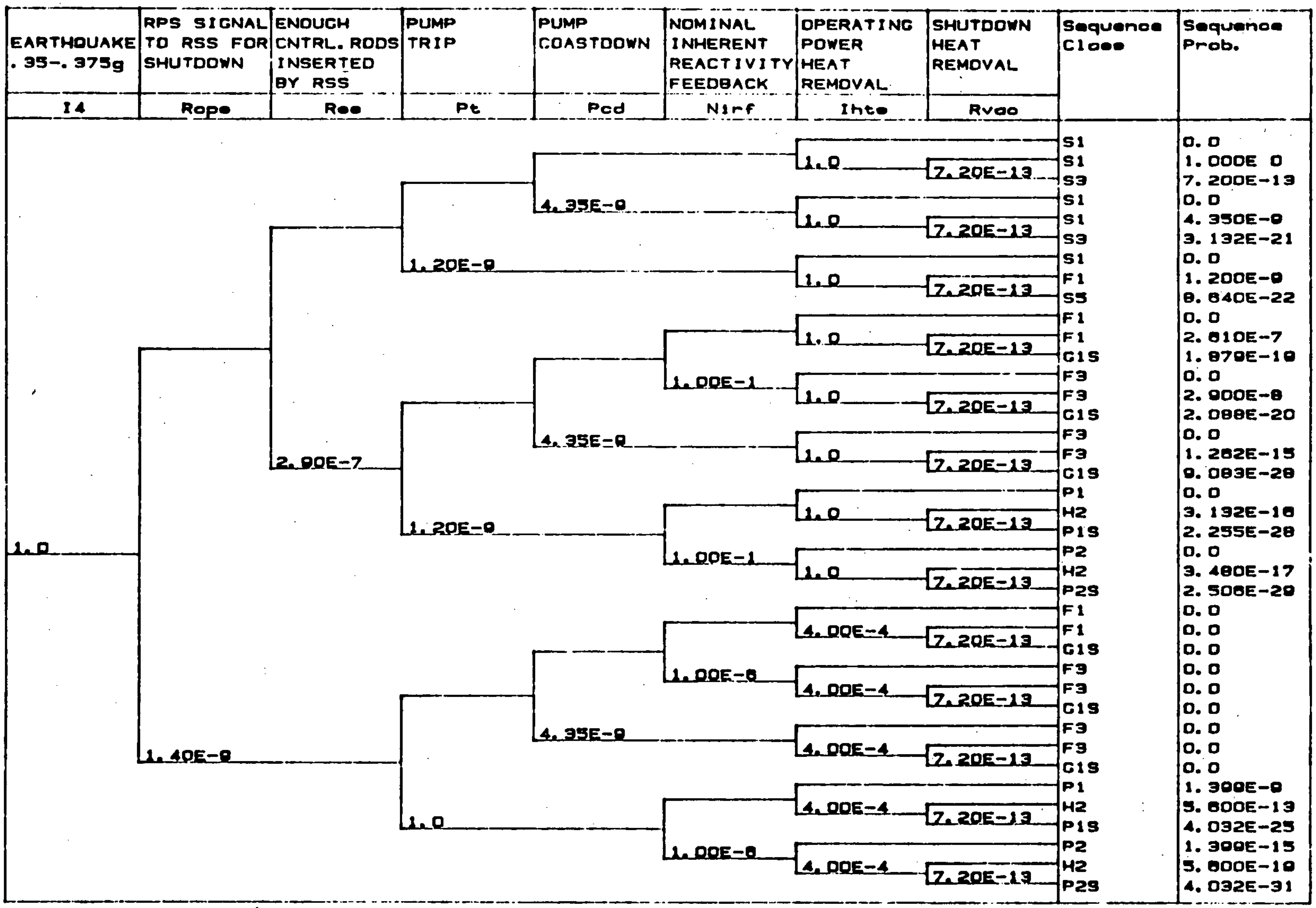
B: I2. TRE10-05-1987 SYS. R. TREE IE2: Reactivity Inser. ($.18-$0.36)

Figure A4.2-2



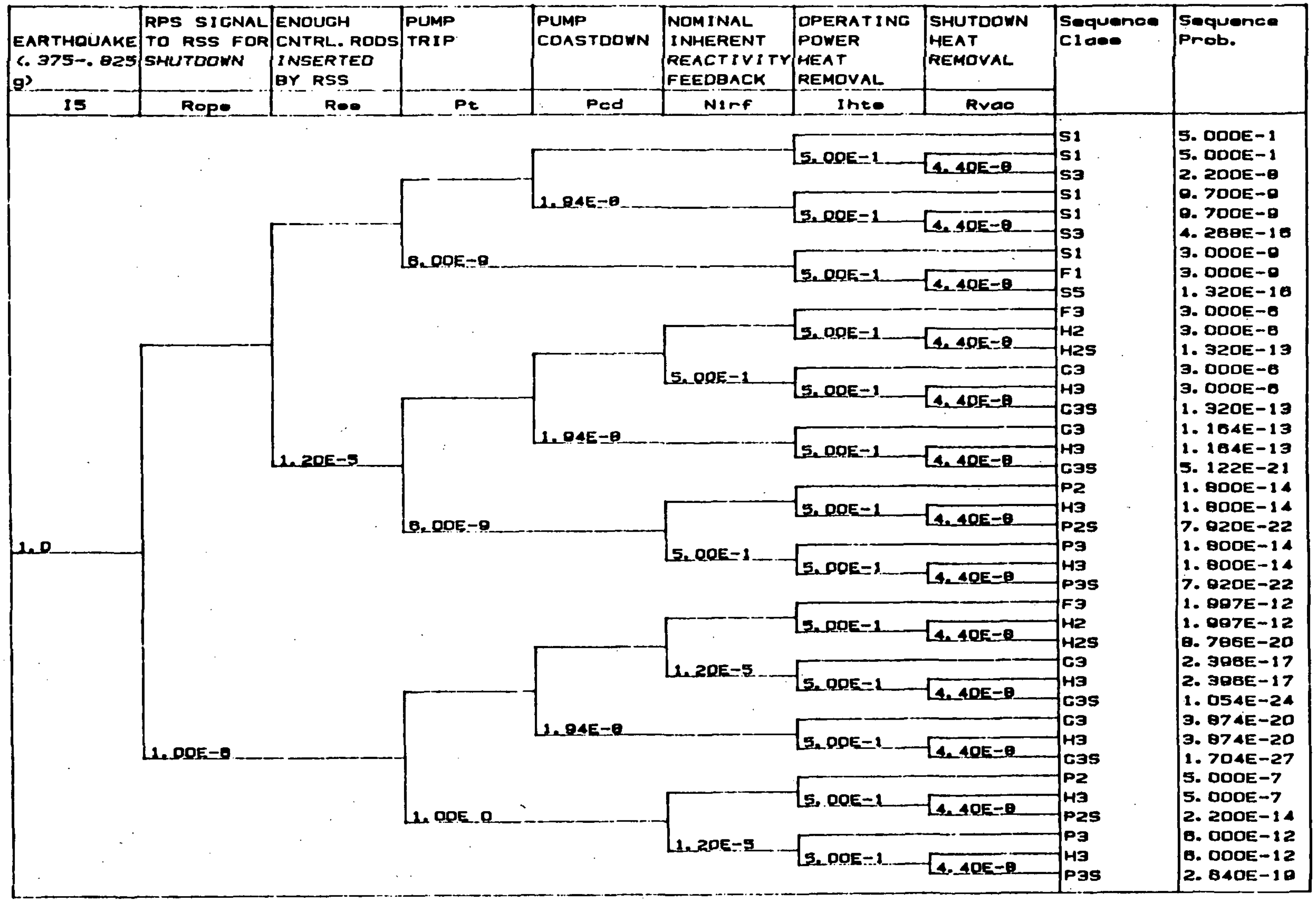
B: I3.TRE10-05-1987 SYS. R. TREE IE3: Reactivity Insr. (>$0.36)

Figure A4.2-3



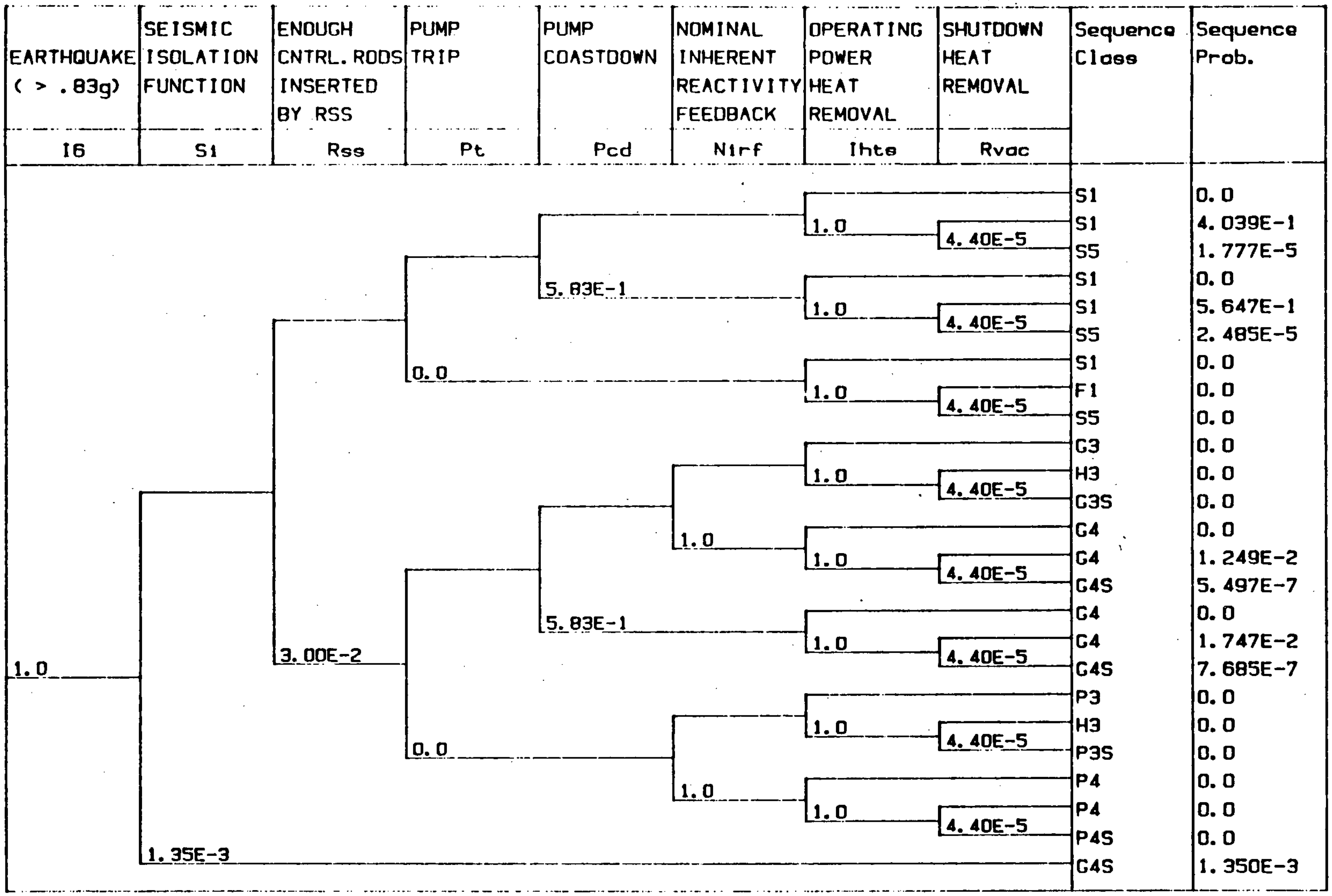
B: I4.TRE10-05-1987 SYS. R. TREE IE4: Earthquake (.3 to .375g)

Figure A4.2-4



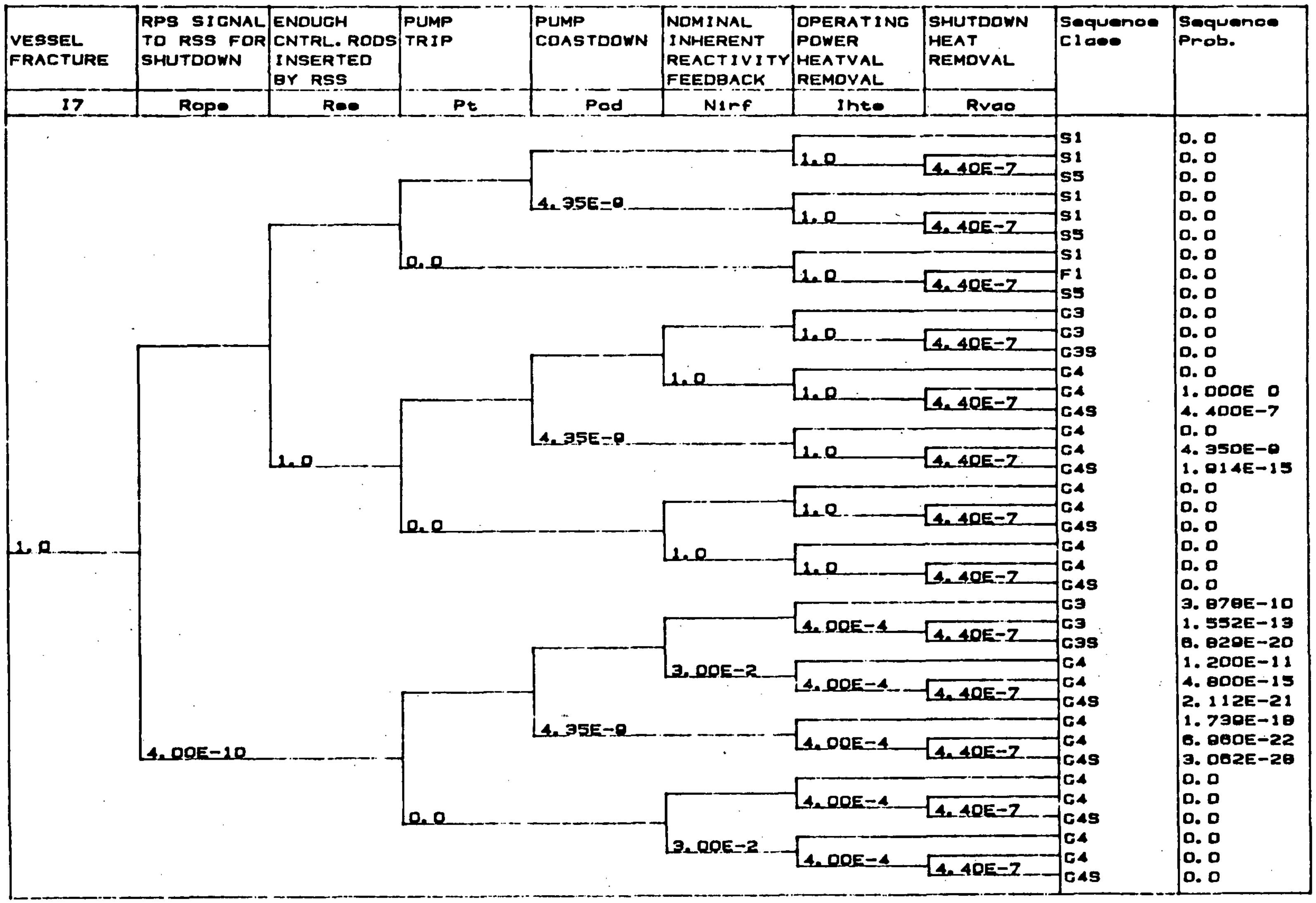
B. IS.TRE10-02-1987 SYS. R. TREE IES. Earthquake(.375 to .875g)

Figure A4.2-5



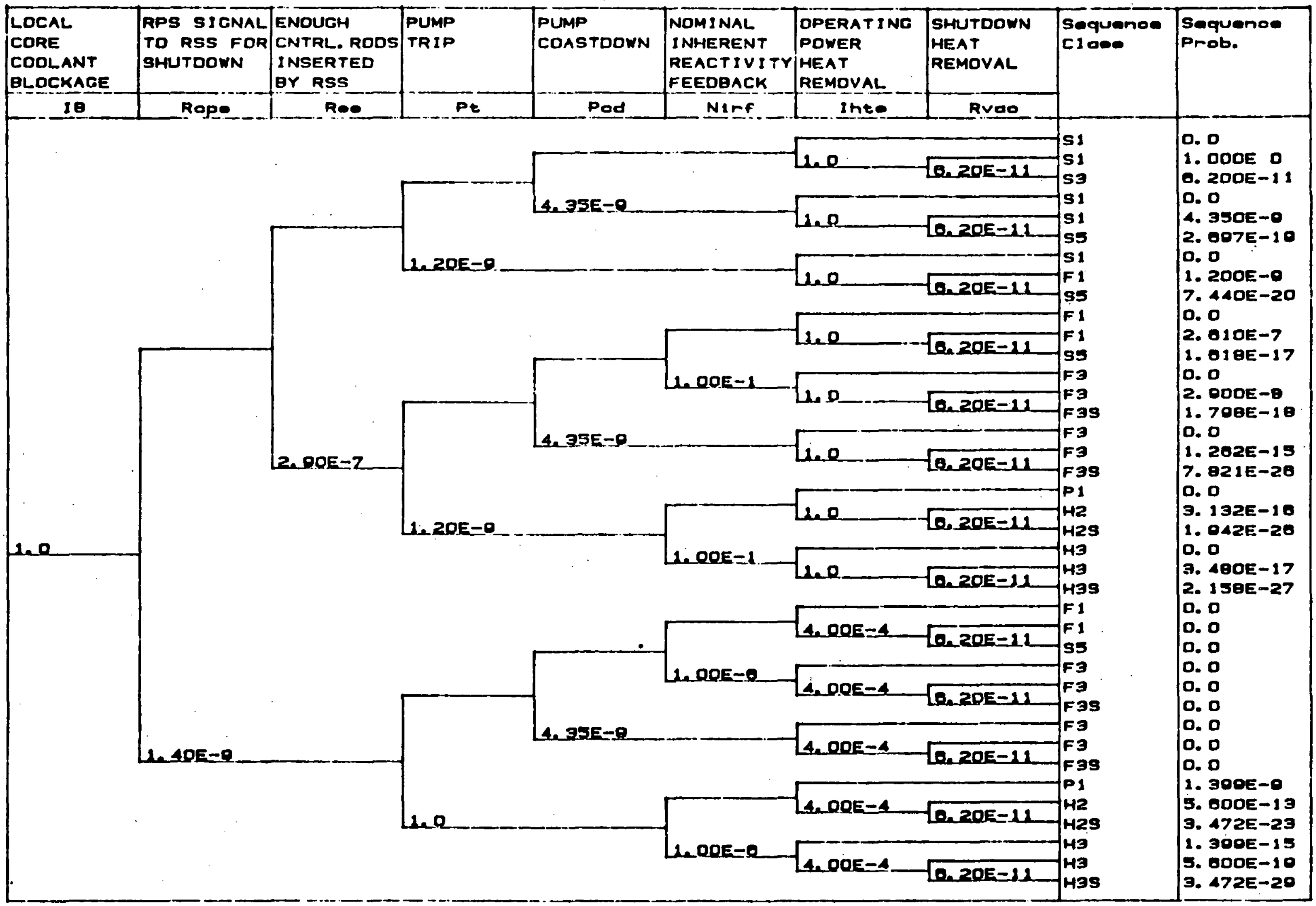
B: I6.TRE10-01-1987 SYS.R.TREE IE6: Earthquake (Over .825g)

Figure A4.2-6



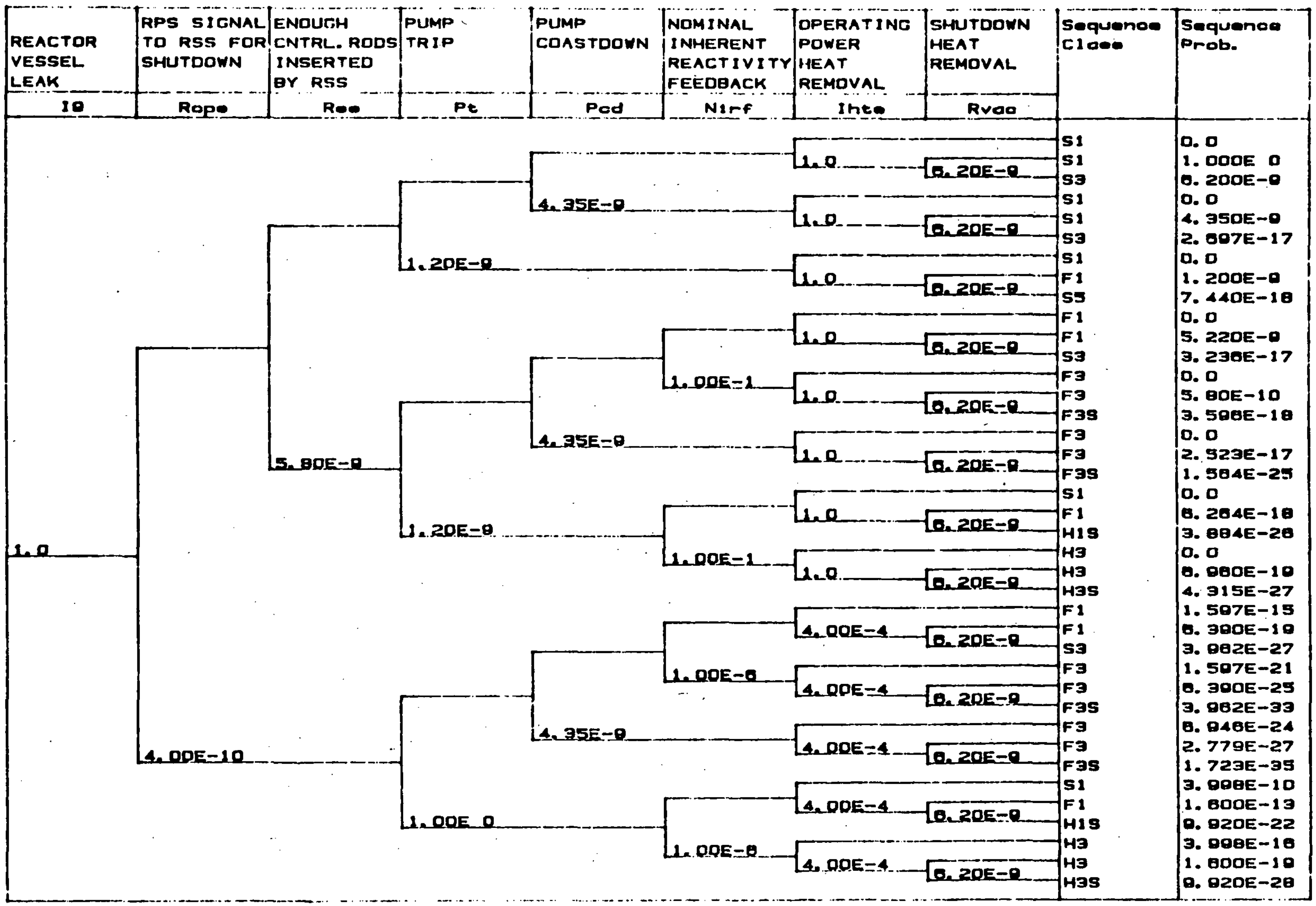
B. I7. TRE10-05-1987 SYS. R. TREE IE7. Vessel Fracture

Figure A4.2-7



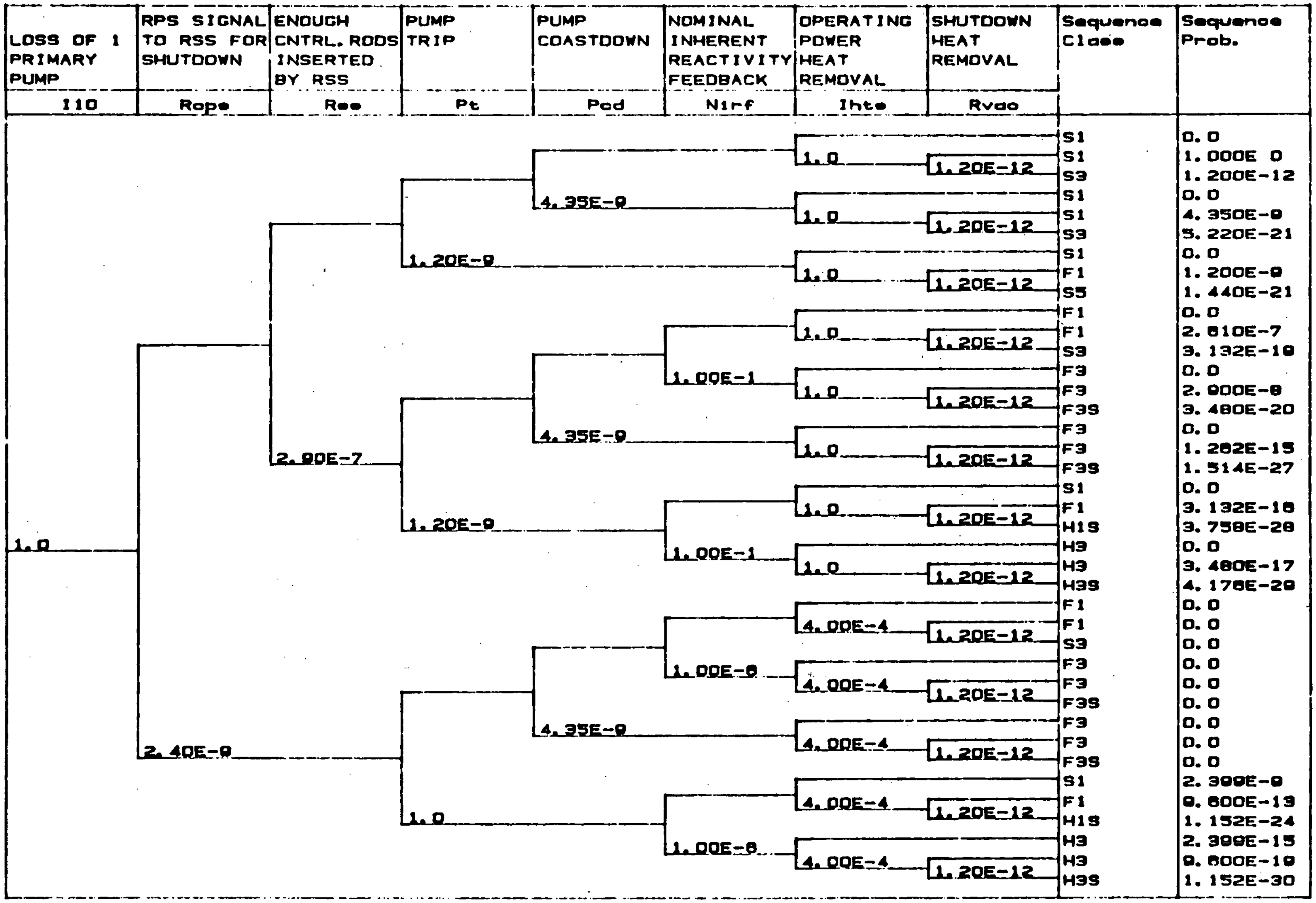
B. IB. TRE10-05-1987 SYS. R. TREE IEB. Local Core Coolant Blockage

Figure A4.2-8



B. I9. TRE10-05-1987 SYS. R. TREE IE9. Reactor Vessel Leak

Figure A4.2-9



B: I10. TRE10-05-1987 SYS. R. TREE IE10: Loss of One Primary Pump

Figure A4.2-10

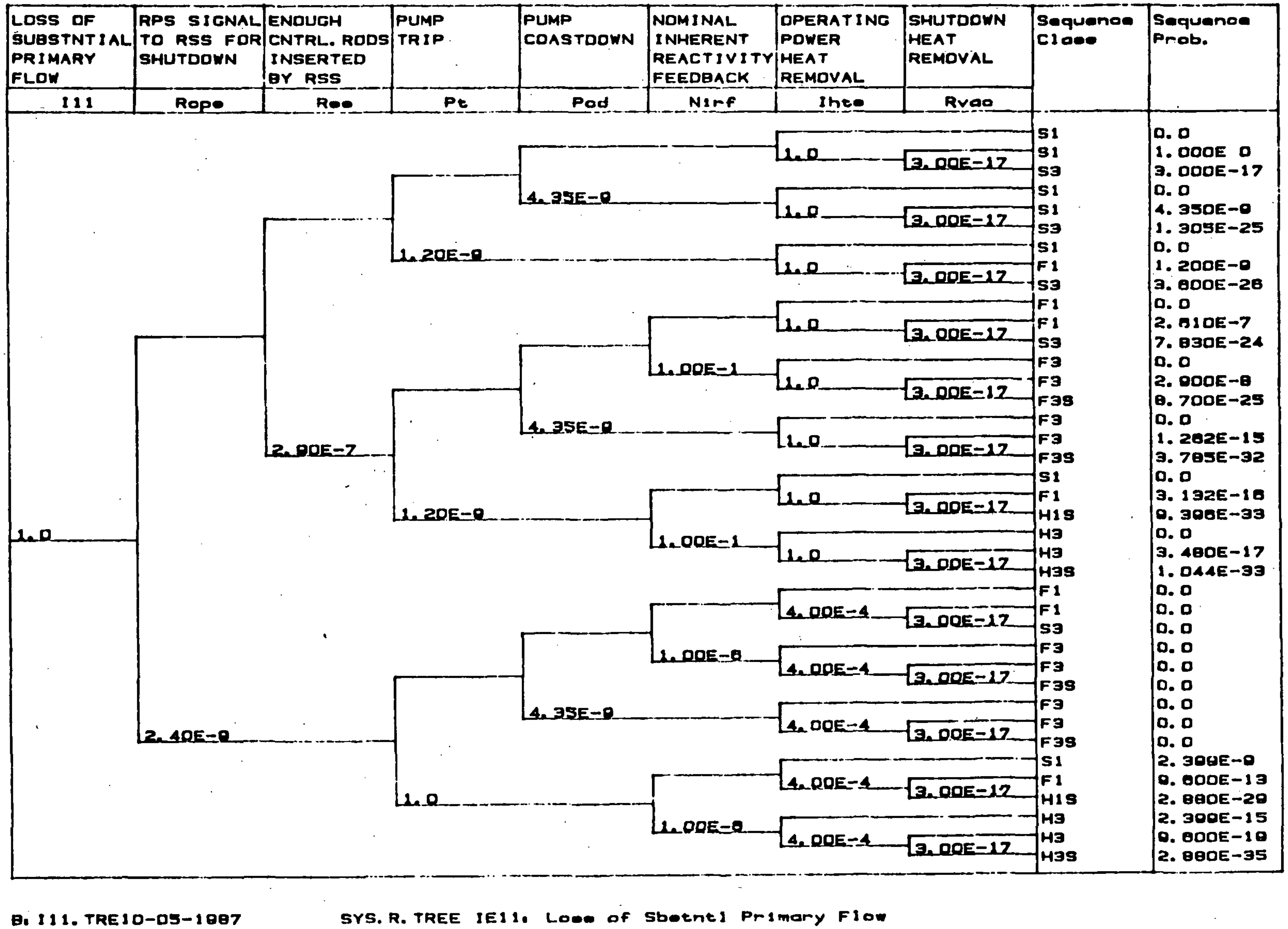
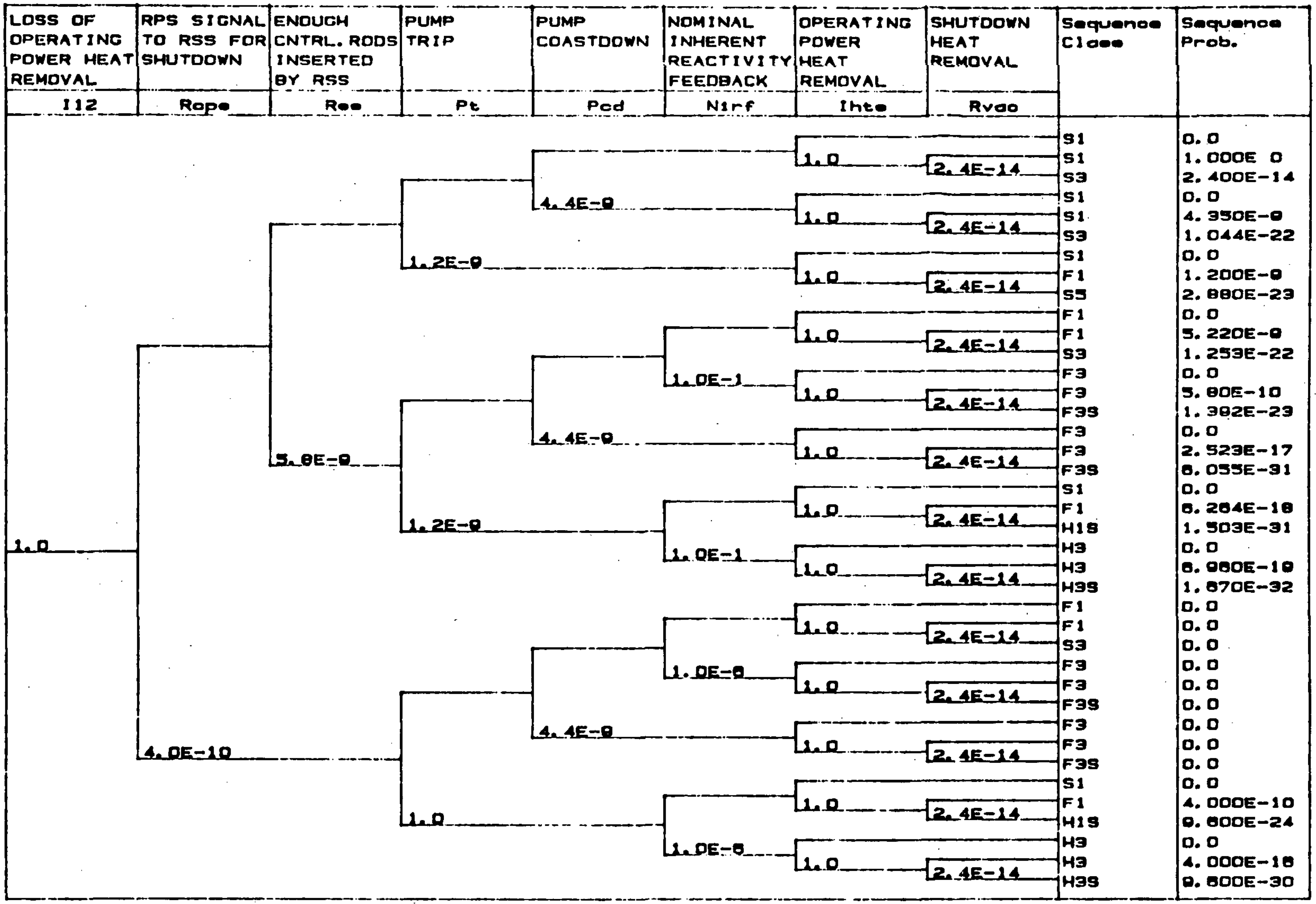
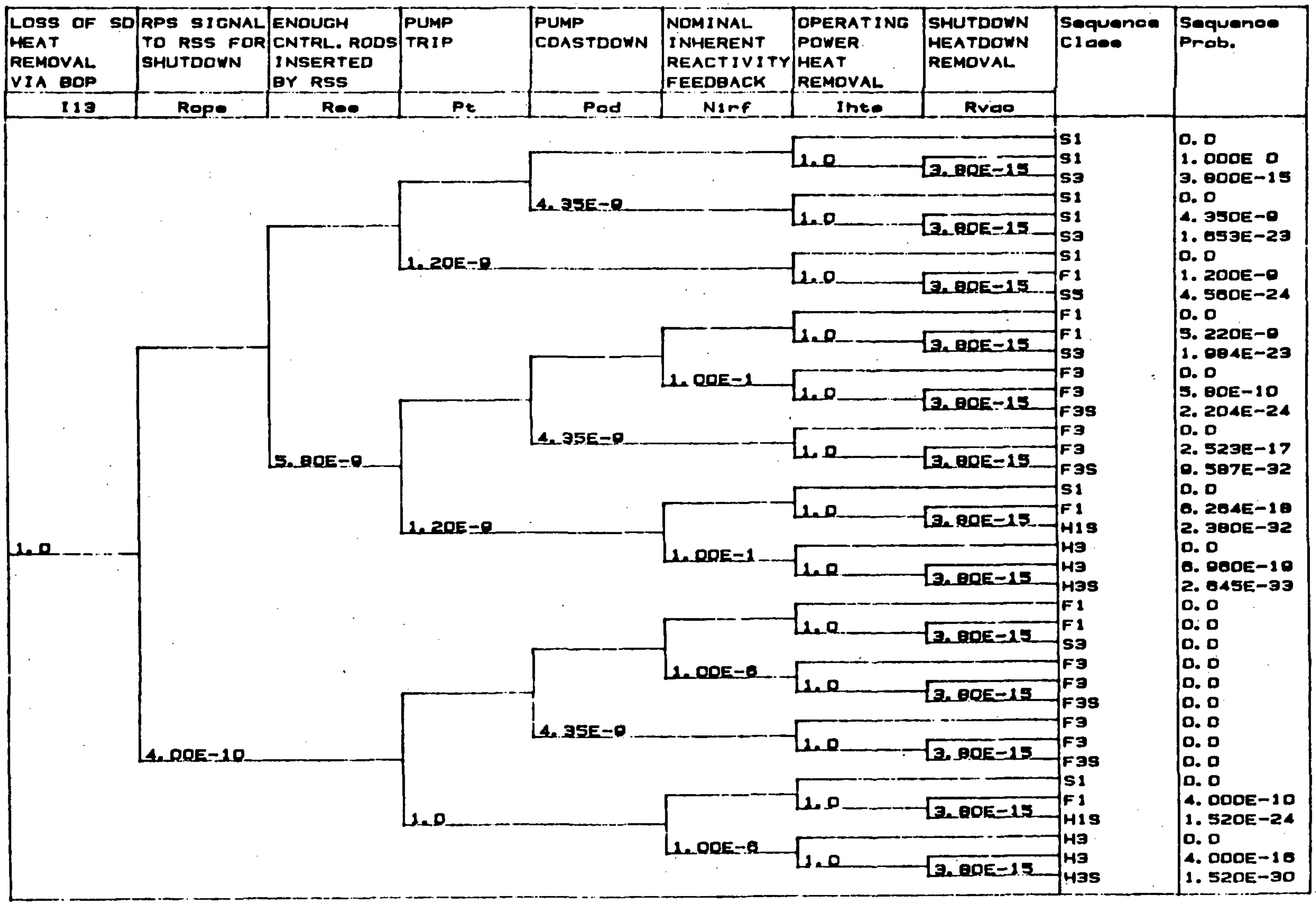


Figure A4.2-11



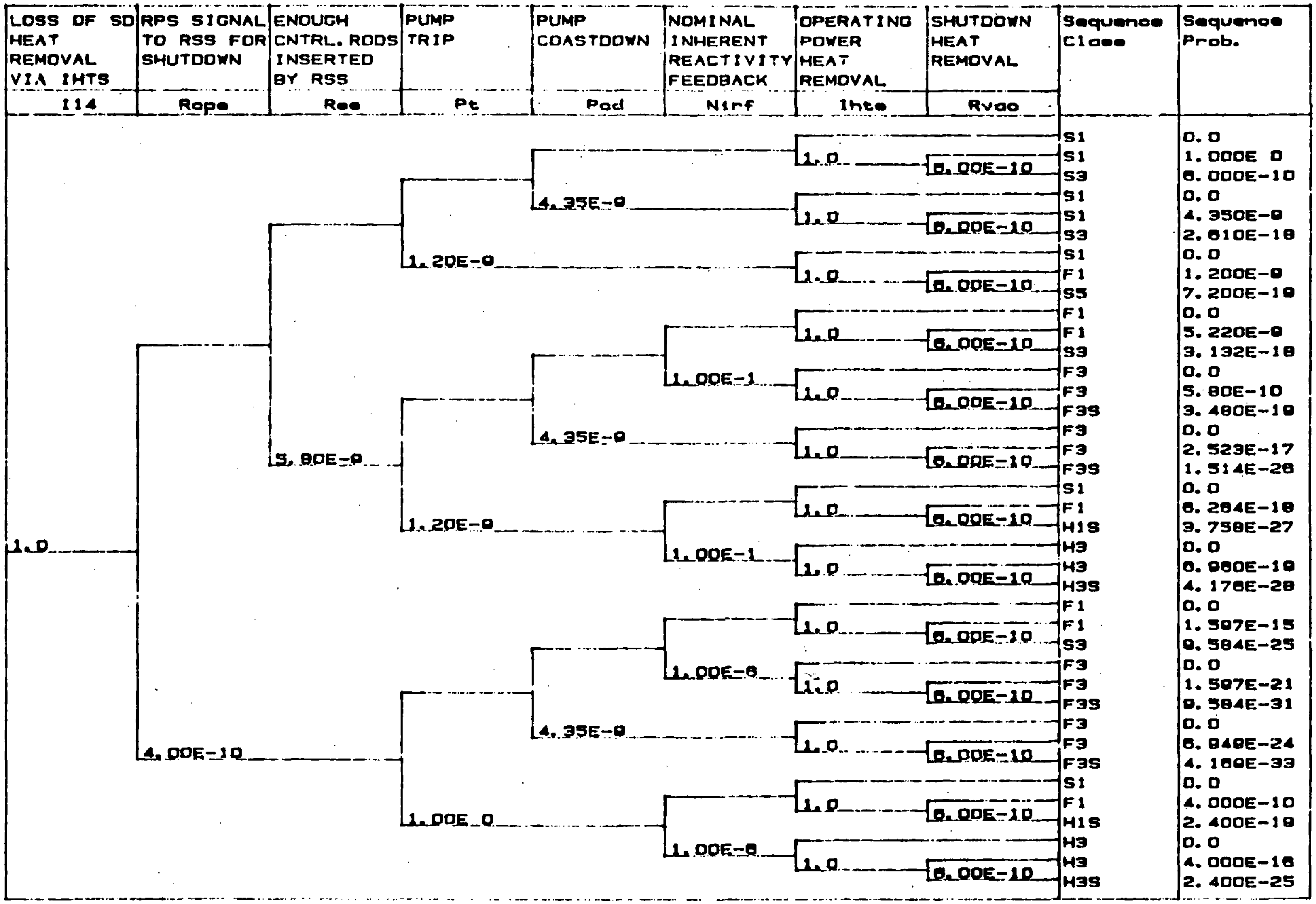
B. I12. TRE10-05-1987 SYS. R. TREE IE12. Loss of Oper. Pwr Heat Removl

Figure A4.2-12



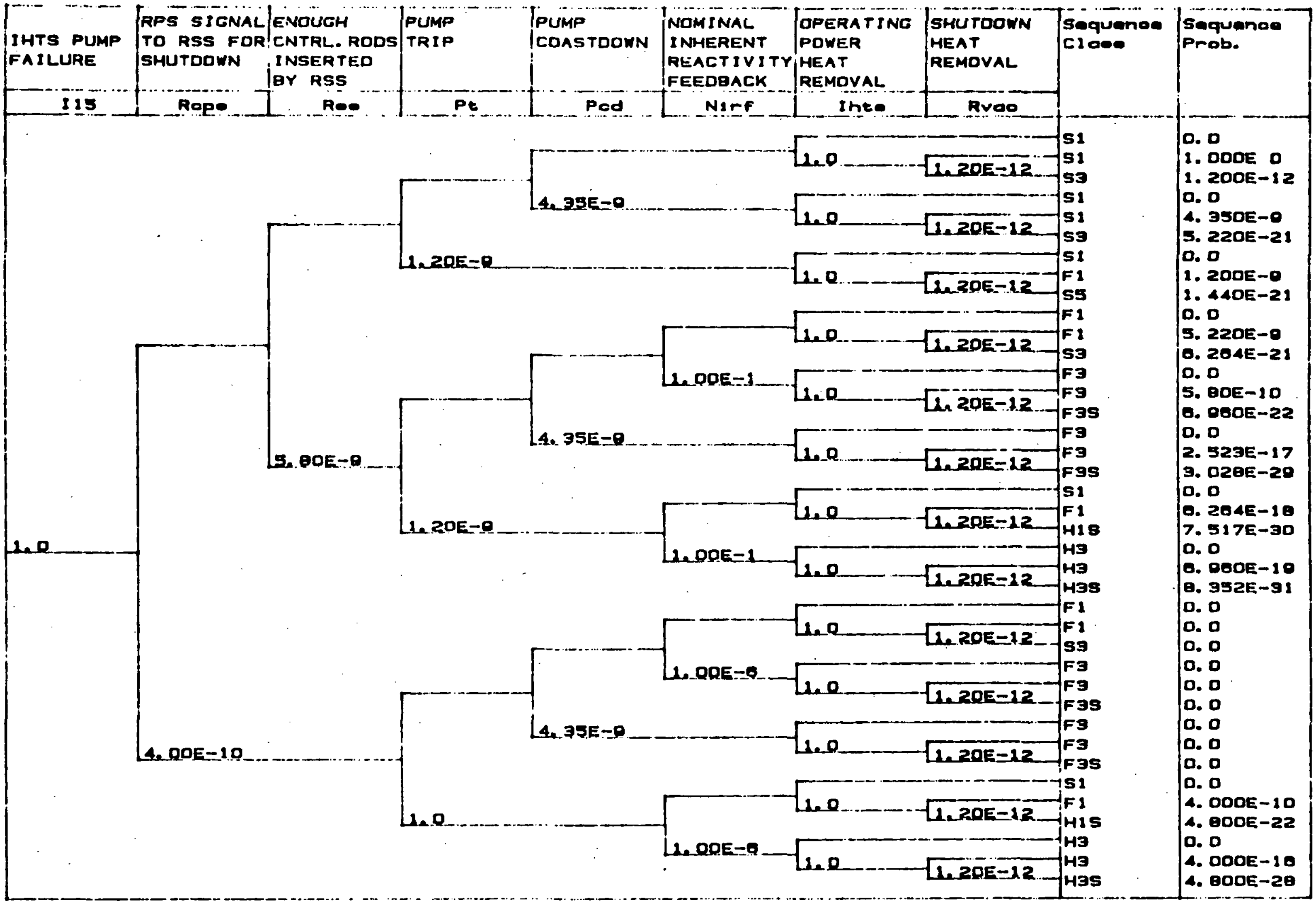
B. I13.TRE10-05-1987 SYS. R. TREE IE13: Loss of S/D Ht. Rem. via BOP

Figure A4.2-13



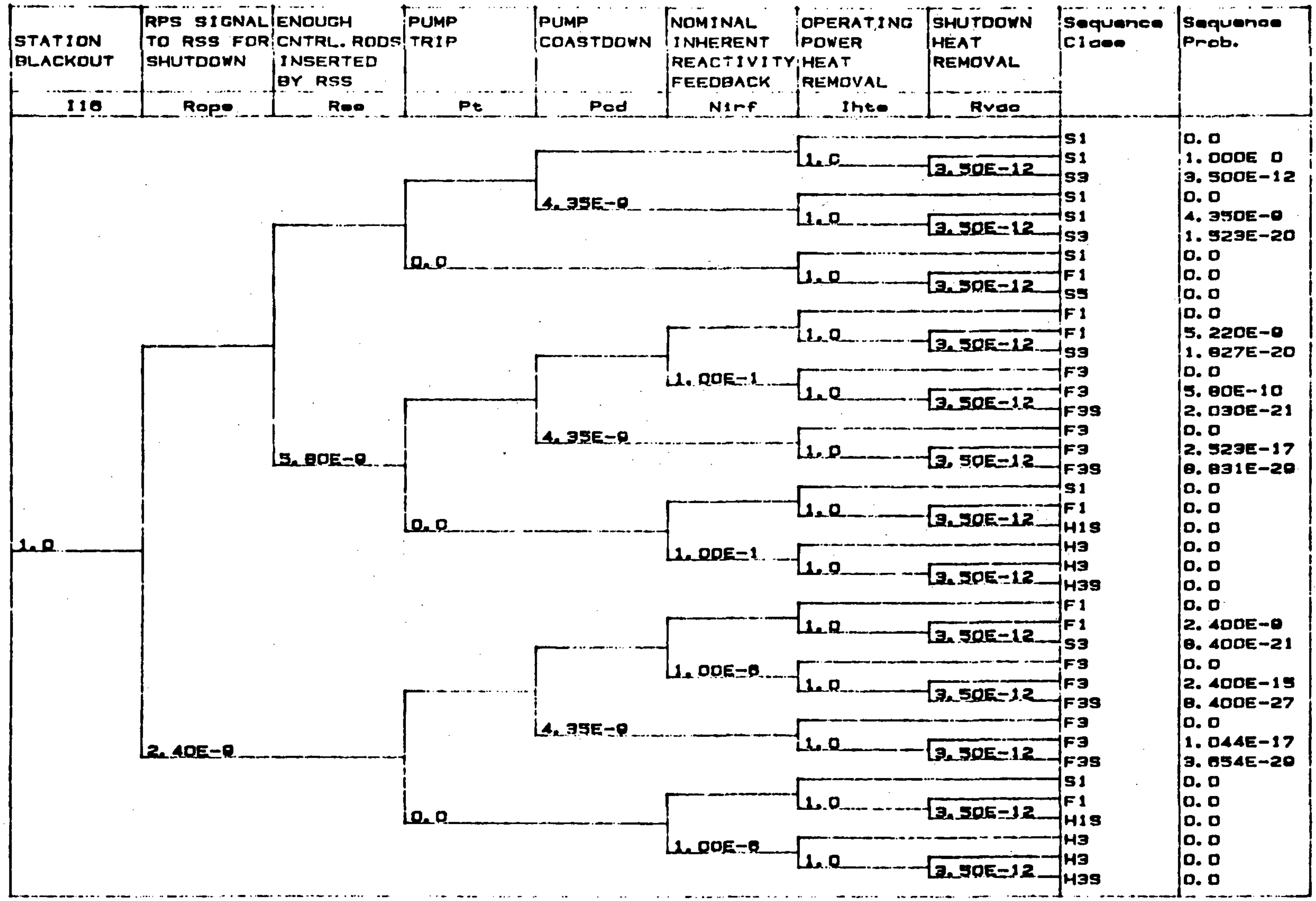
B. I14.TRE10-05-1987 SYS.R.TREE IE14. Loss of S/D Ht.Rem. via IHTS

Figure A4.2-14



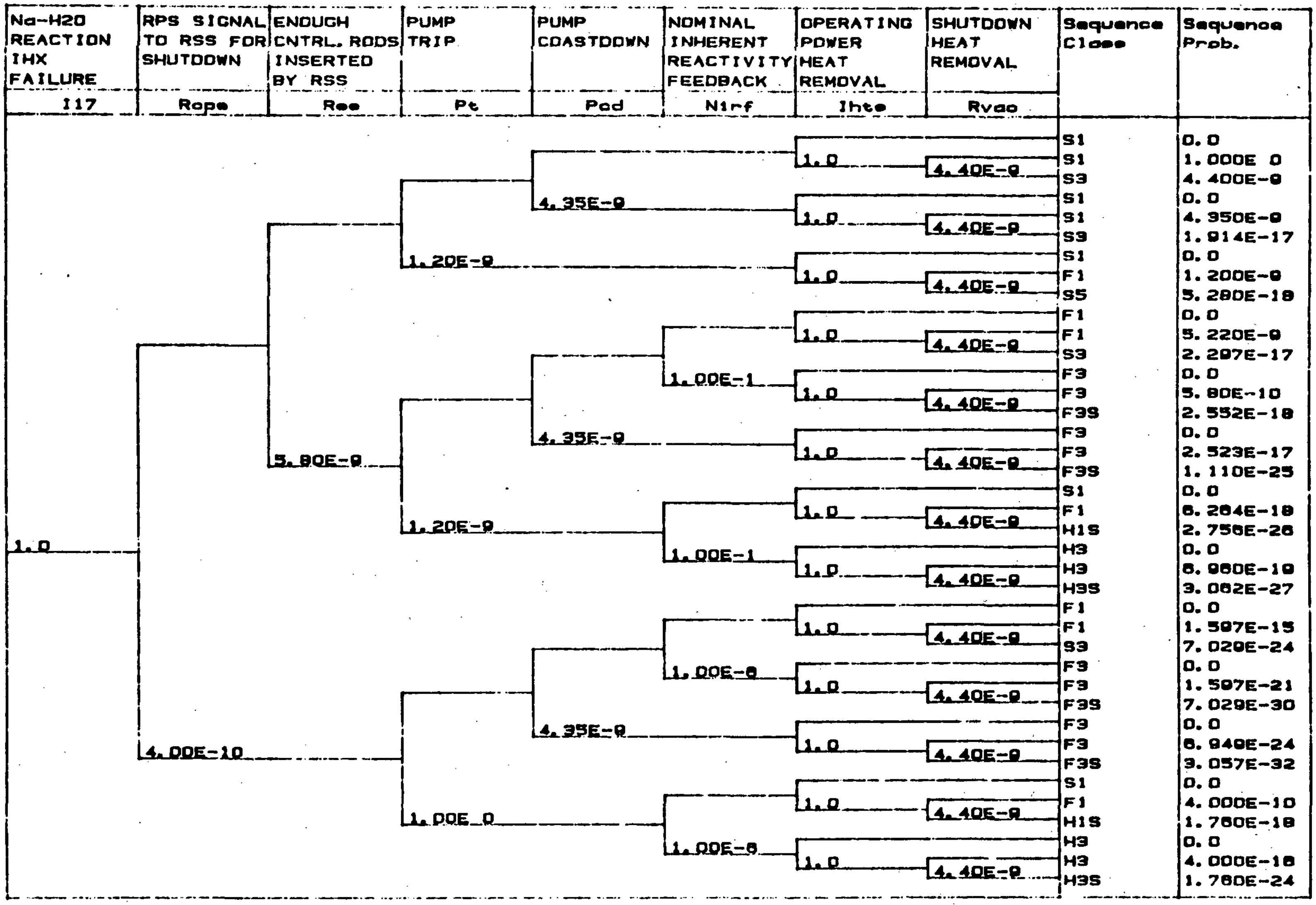
B: I15.TRE10-05-1987 SYS.R.TREE IE15: IHTS Pump Failure

Figure A4.2-15



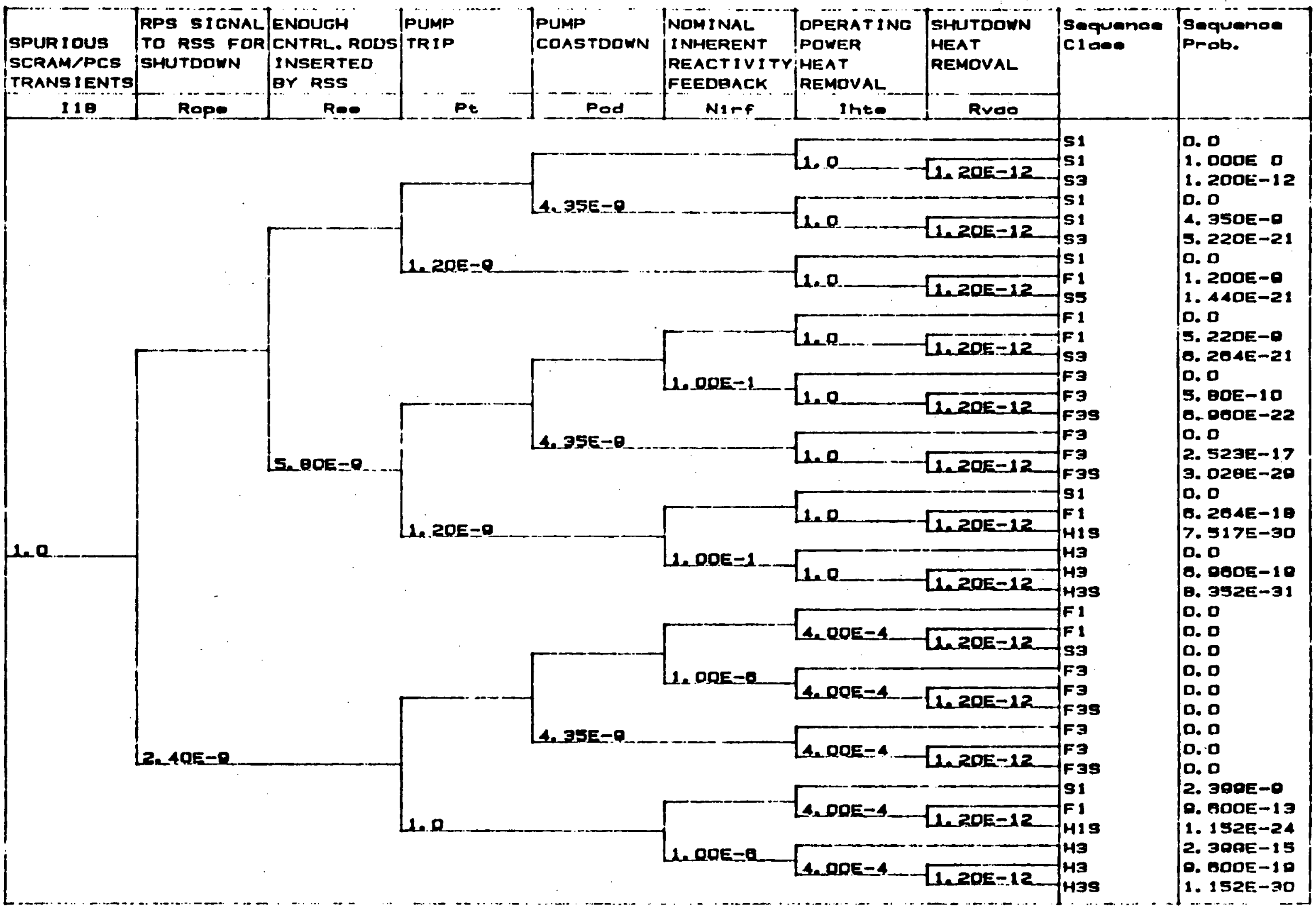
B. I18. TRE10-05-1987   SYS. R. TREE IE18. Station Blackout

Figure A4.2-16



B: I17.TRE10-05-1987 SYS. R. TREE IE17: Large Na-H2O Reactn. IHX Fail

Figure A4.2-17



8. I18. TRE10-05-1987 SYS. R. TREE IE18. Spurious Scram/ Transients

Figure A4.2-18

NORMAL SHUTDOWN	SHUTDOWN HEAT REMOVAL	Sequence Class	Sequence Prob.
I19	SHR		
1.0		S1	1.000E 0
	1.20E-12	S3	1.200E-12

B: I19.TRE10-03-1987 SYS.R.TREE IE19: NORMAL SHUTDOWN

Figure A4.2-19

FORCED SHUTDOWN	SHUTDOWN HEAT REMOVAL	Sequence Class	Sequence Prob.
I20	SHR		
1.0		S1	1.000E 0
	2.00E-13	S3	2.000E-13

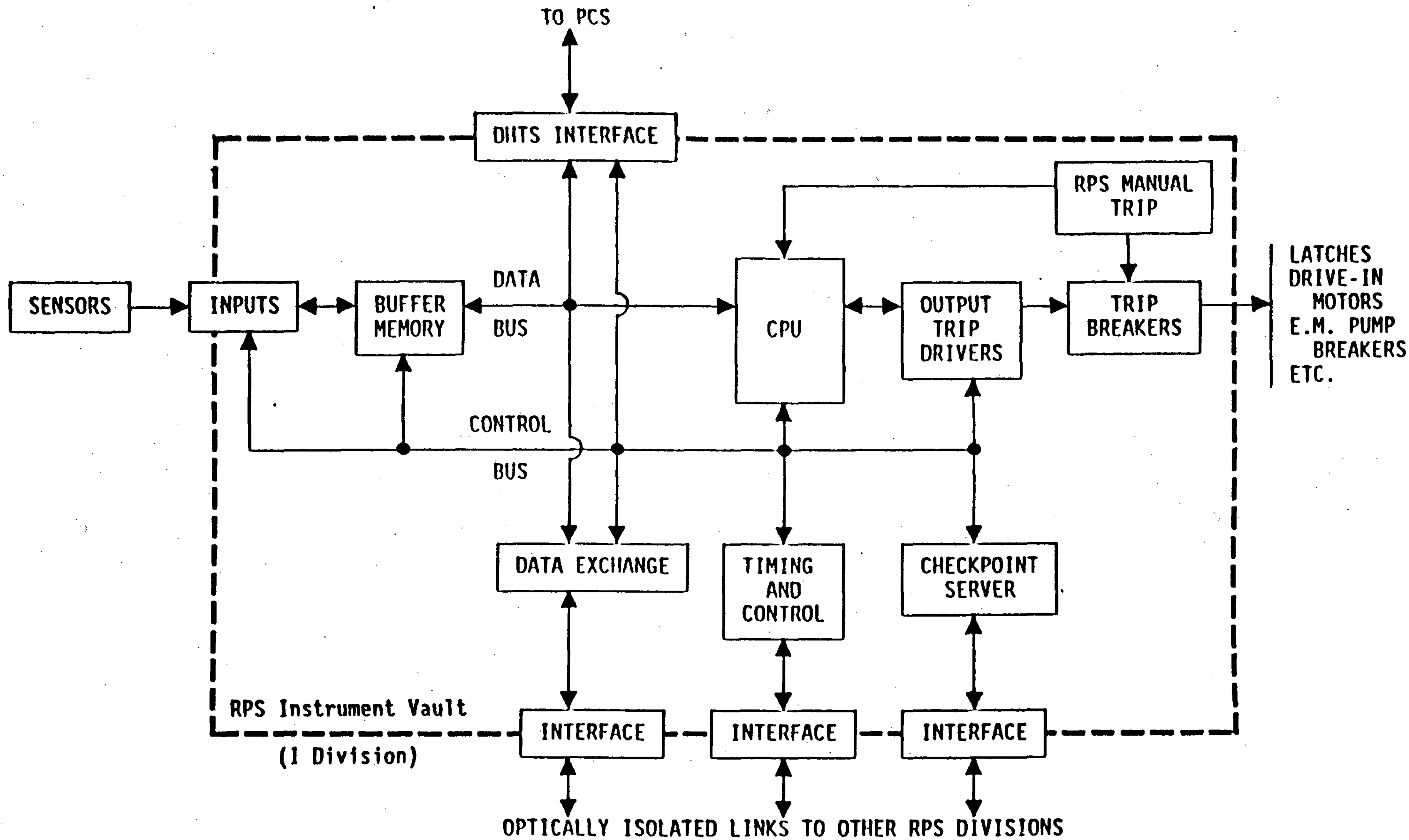
B: I20.TRE10-03-1987 SYS.R.TREE IE20: FORCED SHUTDOWN

Figure A4.2-20

RVACS BLOCKAGE I21	SHUTDOWN HEAT REMOVAL SHR	Sequence Class	Sequence Prob.
1.0		S1	9.997E-1
	2.80E-4	S3	2.800E-4

B: I21.TRE10-03-1987 SYS.R.TREE IE21: RVACS BLOCKAGE

Figure A4.2-21



Note: Only "TO PCS" is not safety related.

(Typical of 4 Divisions)

Figure A4.2-22 - RPS BLOCK DIAGRAM

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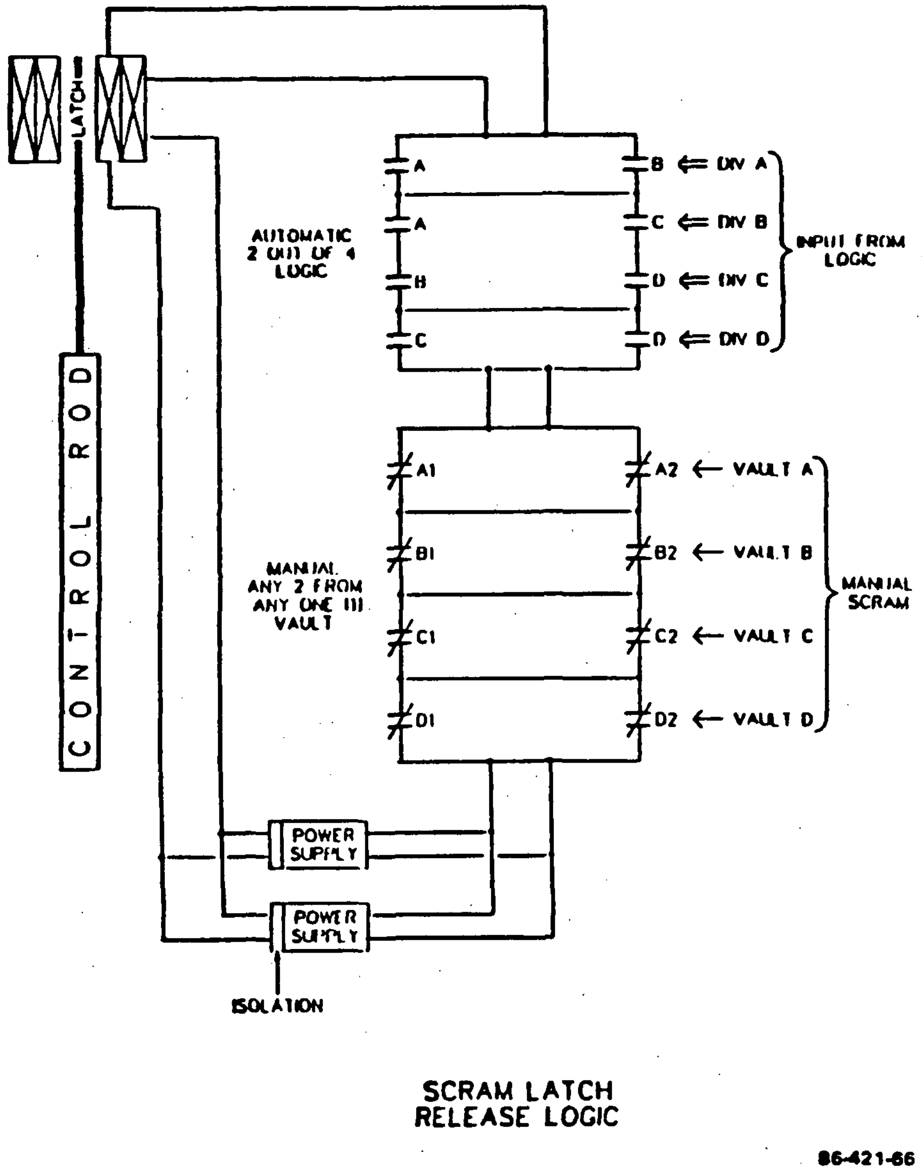


Figure A4.2-24 **CONTROL ROD SCRAM LATCH RELEASE SWITCHING LOGIC**

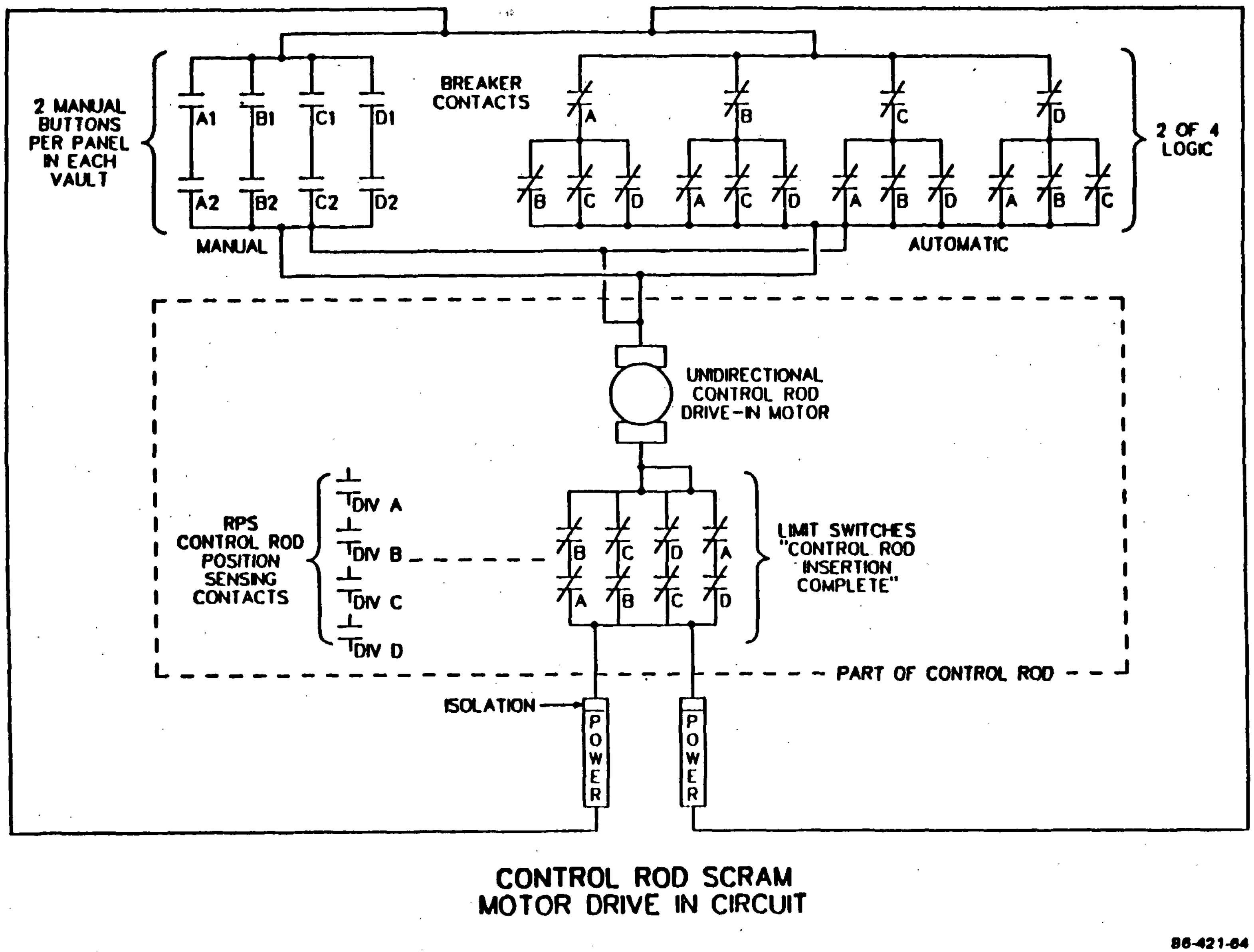


Figure A4.2-25 **CONTROL ROD DRIVE-IN SWITCHING CIRCUIT**

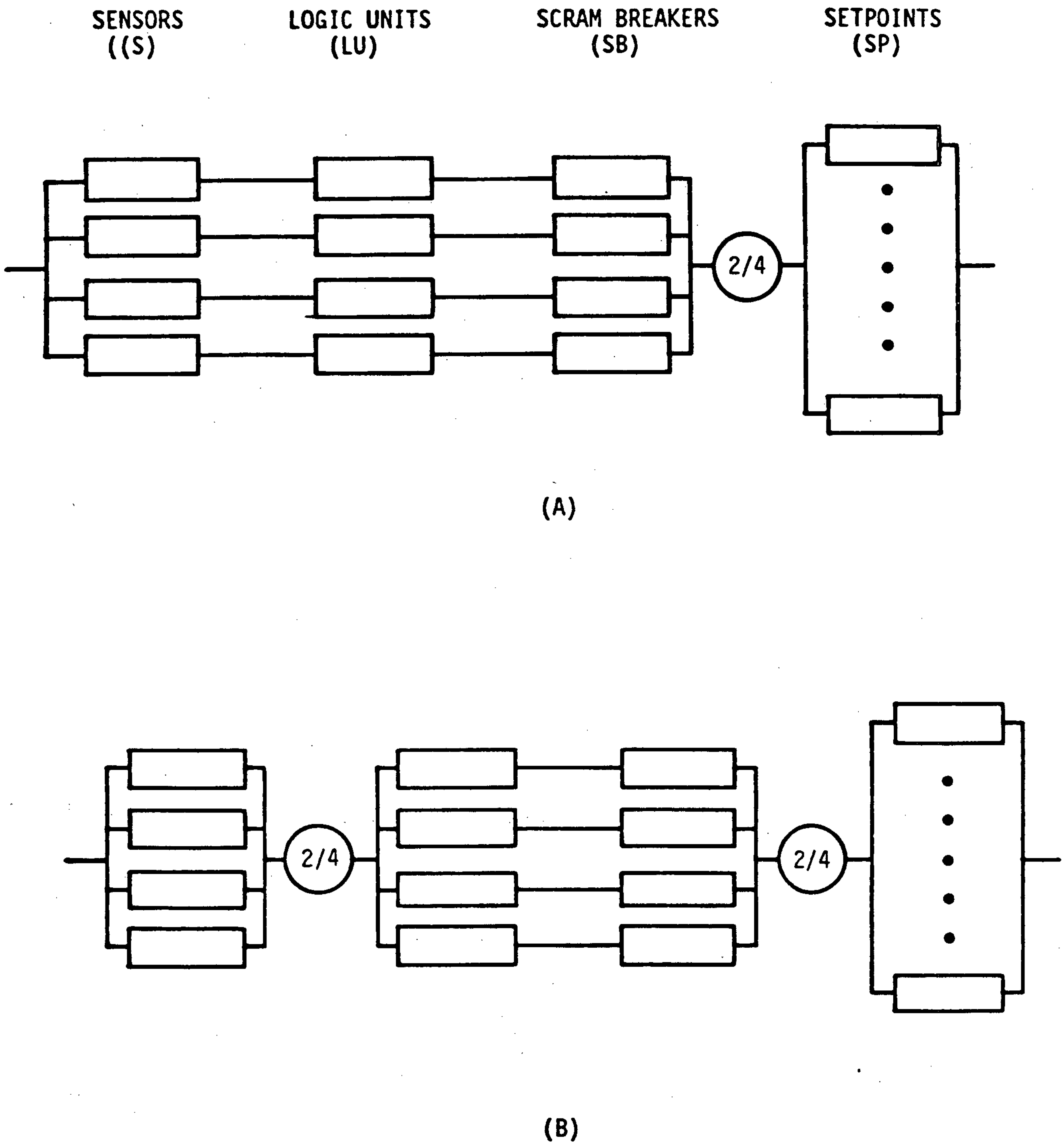


Figure A4.2-26 - RPS RELIABILITY BLOCK DIAGRAMS SHOWING THE EFFECT OF RECONFIGURATION

(A) RPS WITHOUT RECONFIGURATION CAPABILITY

(B) PRISM RPS WITH ITS RECONFIGURATION CAPABILITY

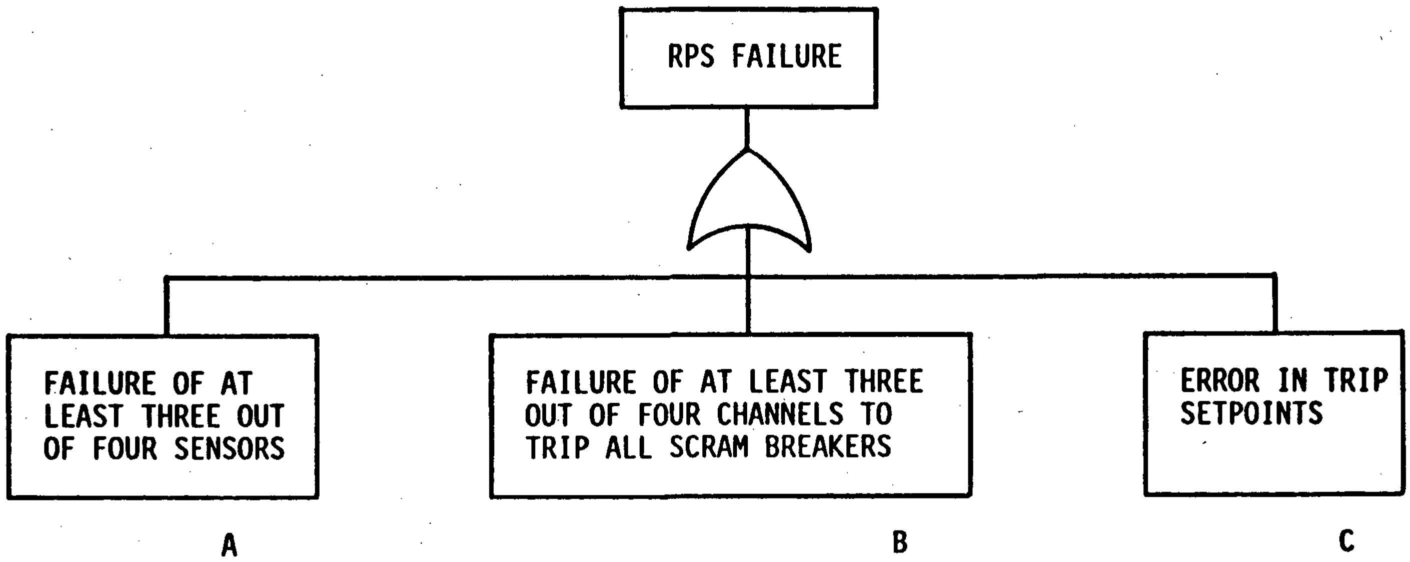
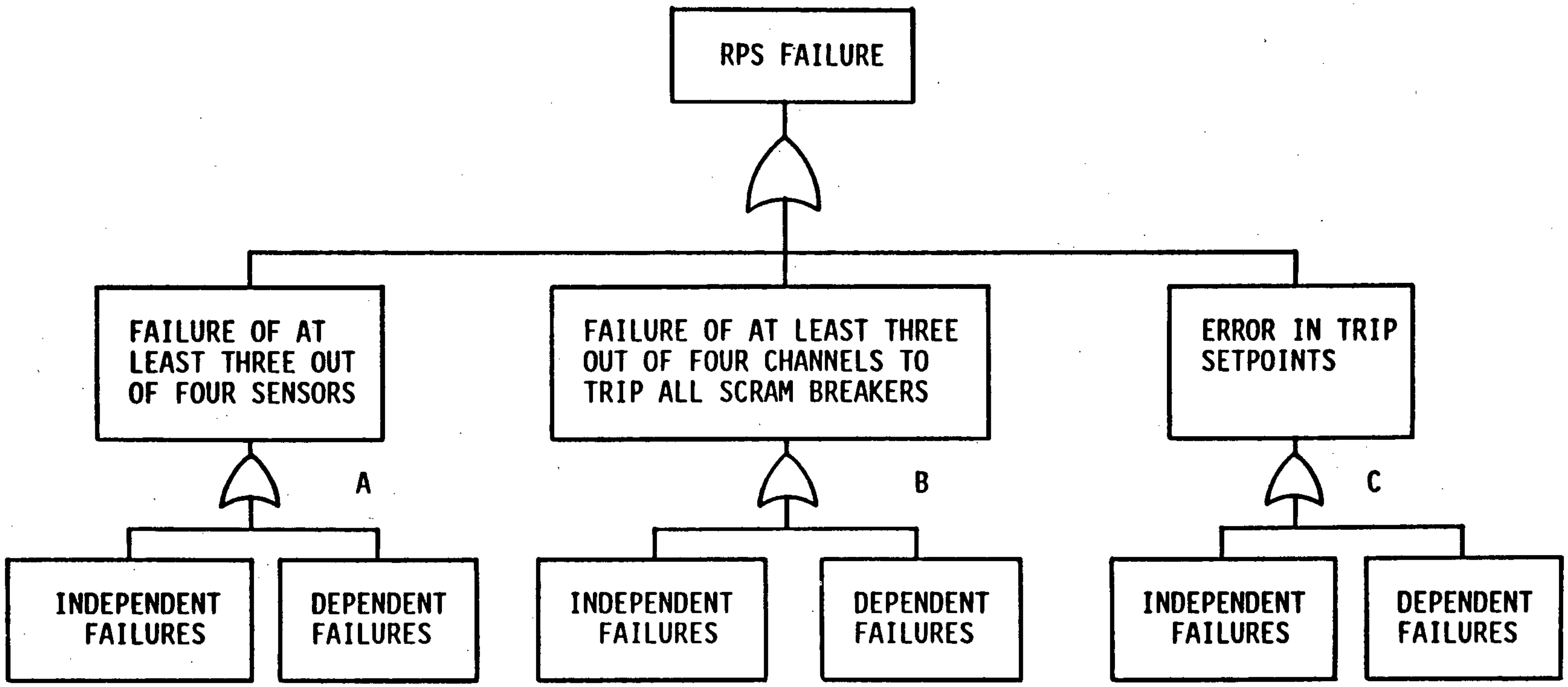


Figure A4.2-27 - FAULT TREE FOR RPS FAILURE



INITIATING EVENT	A: Independent	A: Dependent	B: Independent	B: Dependent	C: Independent	C: Dependent
TOP[1]	*[4]	10^{-9}[5]	*	2×10^{-10}	*	10^{-10}
LOF[2]	*	2×10^{-9}	*	2×10^{-10}	*	10^{-10}
LOHS[3]	*	*	*	2×10^{-10}	*	10^{-10}

(1) TOP: Transient over power initiating events
(2) LOF: Loss of flow initiating events
(3) LOHS: Loss of heat sink initiating events
(4) *: Negligible failure probability per demand
(5) Number indicates failure probability per demand

Figure A4.2-28 - CONTRIBUTORS TO RPS FAILURE

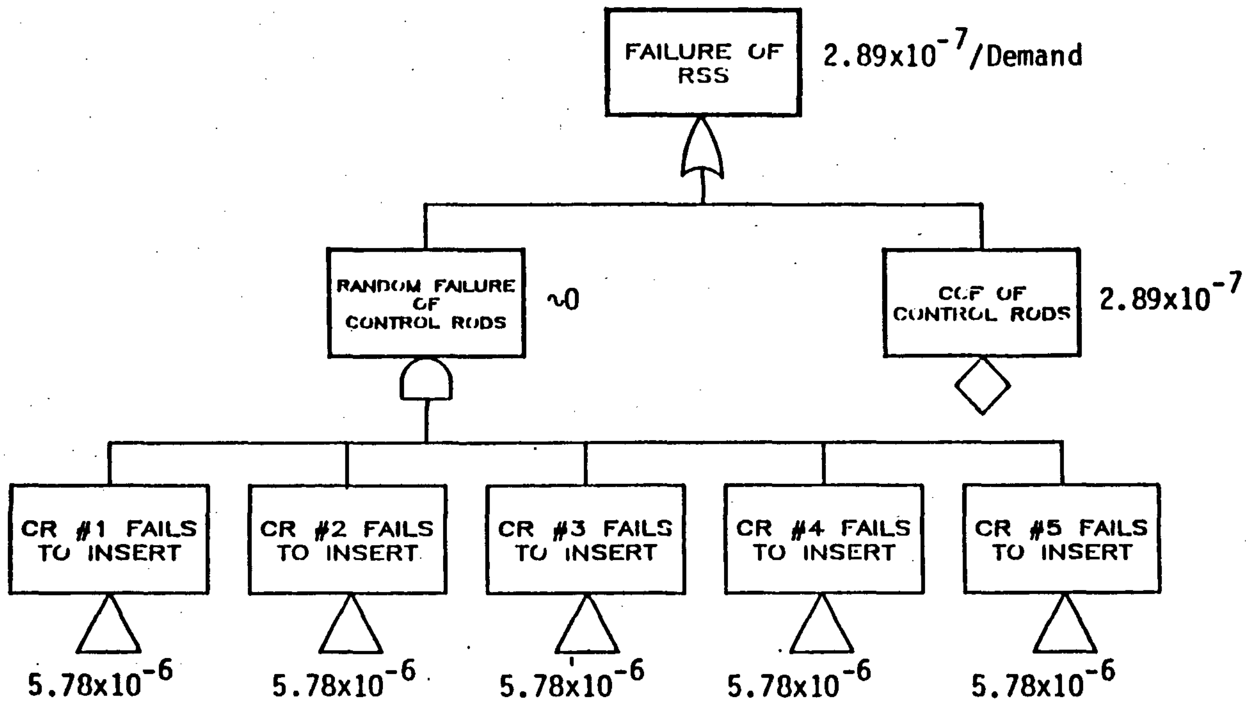


Figure A4.2-29 FAULT TREE FOR RSS FAILURE GIVEN A REACTIVITY INSERTION

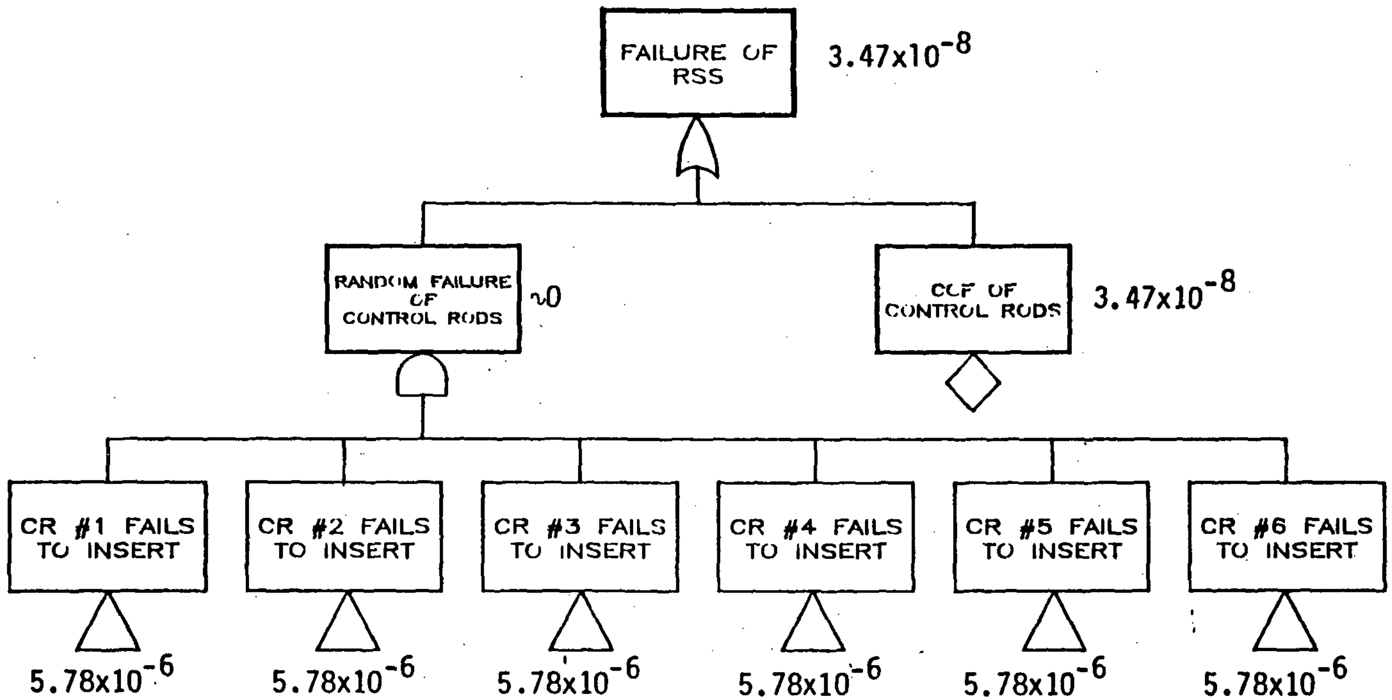


Figure A4.2-30 FAULT TREE FOR RSS FAILURE FOR SEISMICALLY INDUCED SHUTDOWN, INITIATING EVENT 4

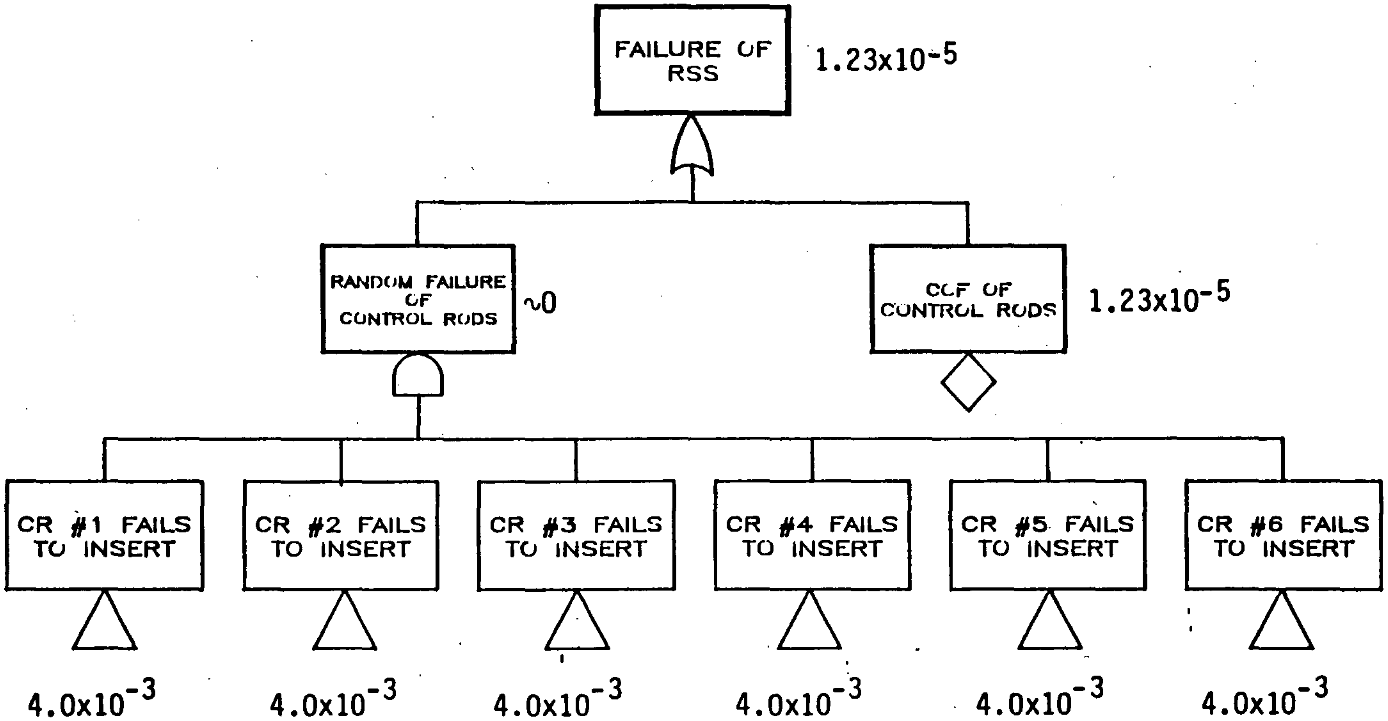


Figure A4.2-31 - FAULT TREE FOR RSS FAILURE FOR SEISMICALLY INDUCED SHUTDOWN, INITIATING EVENT 5

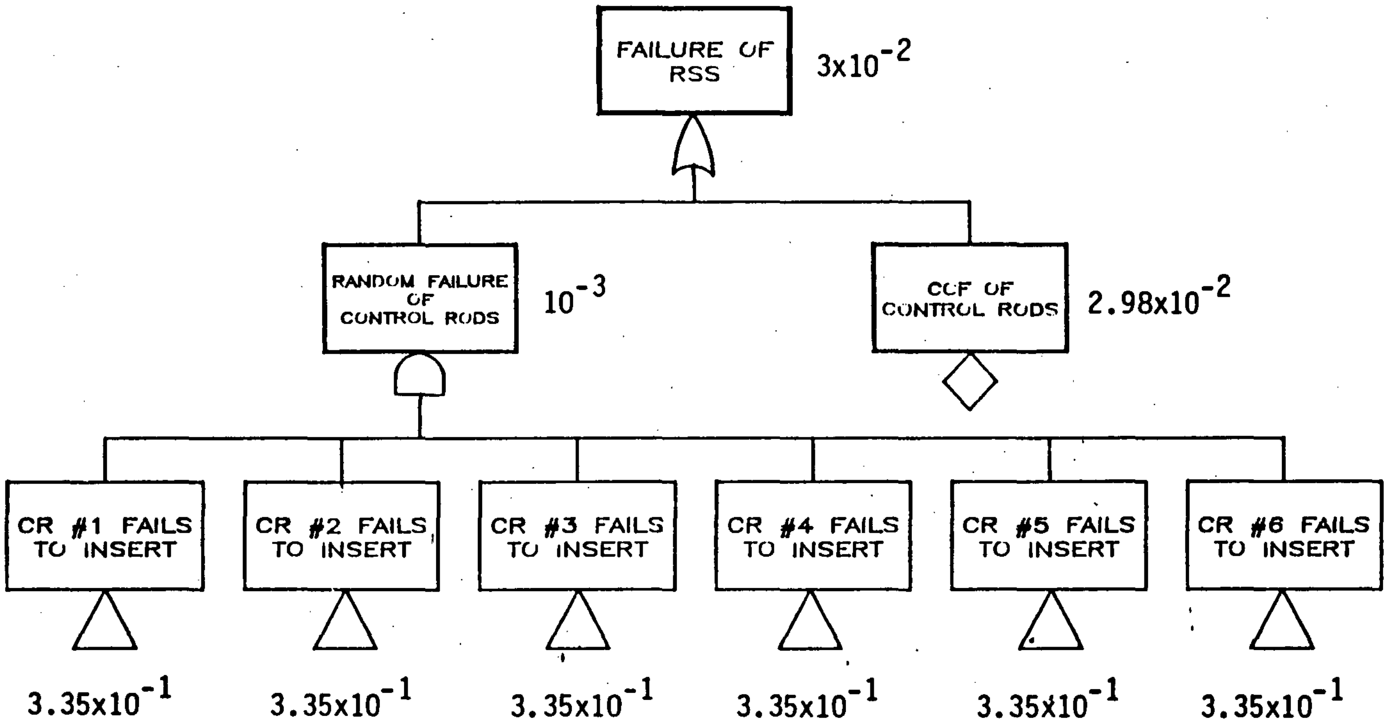


Figure A4.2-32 FAULT TREE FOR RSS FAILURE FOR SEISMICALLY INDUCED SHUTDOWN, INITIATING EVENT 6

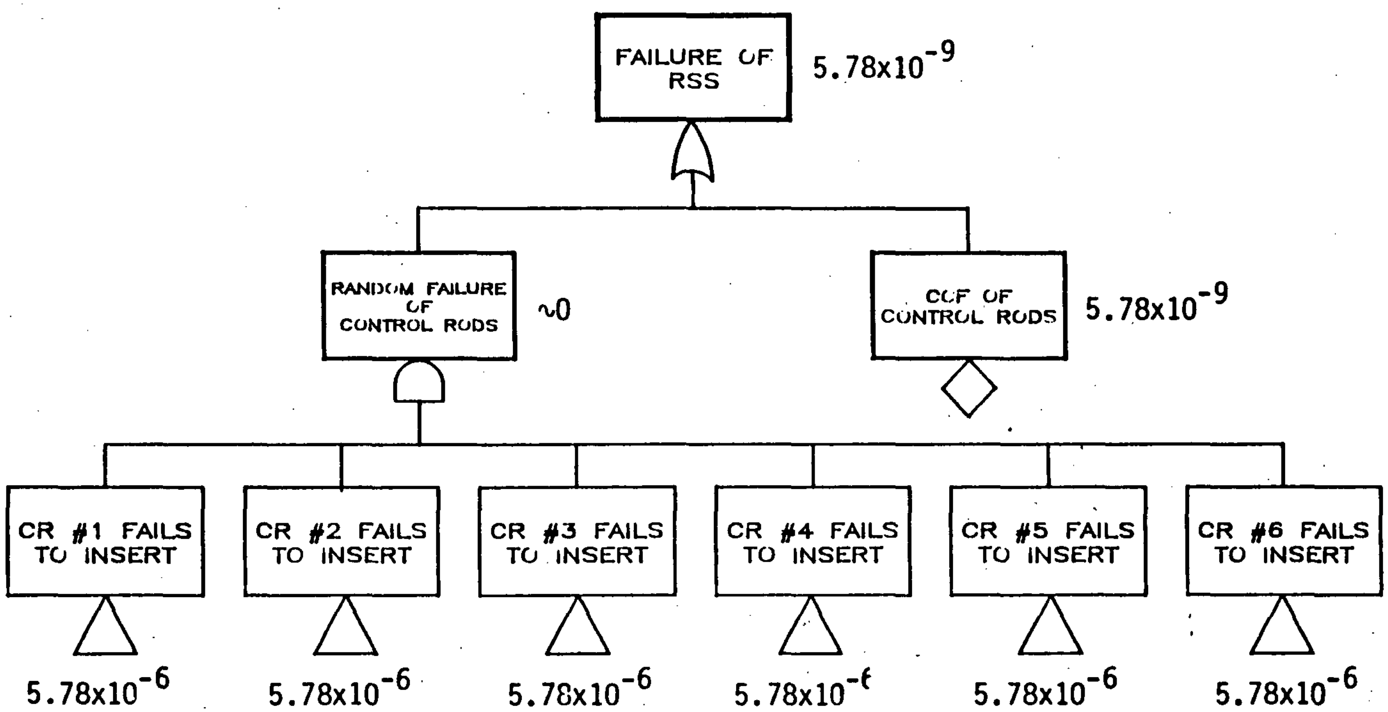


Figure A4.2-33 FAULT TREE FOR RSS FOR INITIATORS WHICH REQUIRE ONE CR FOR SHUTDOWN

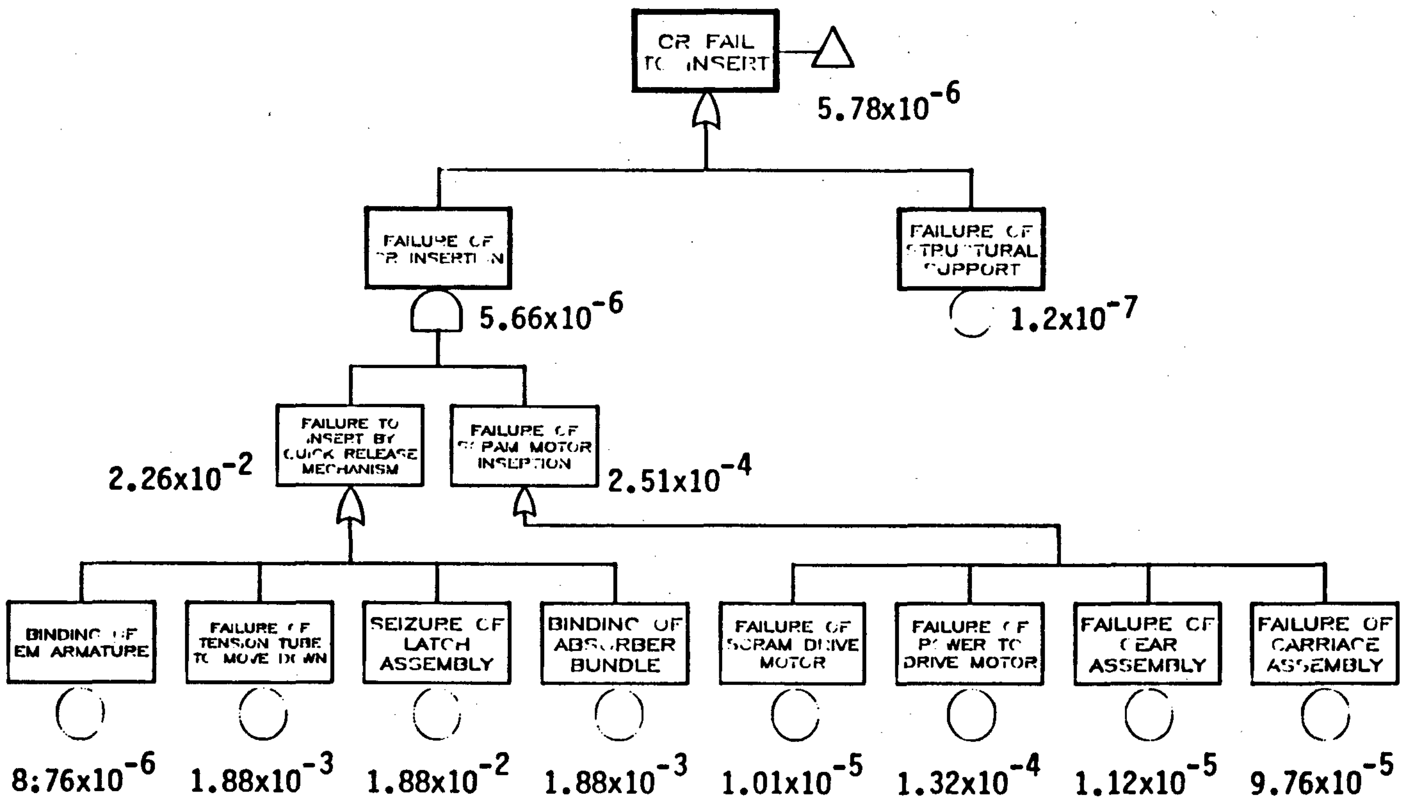


Figure A4.2-34 FAULT TREE FOR A SINGLE CR INSERTION

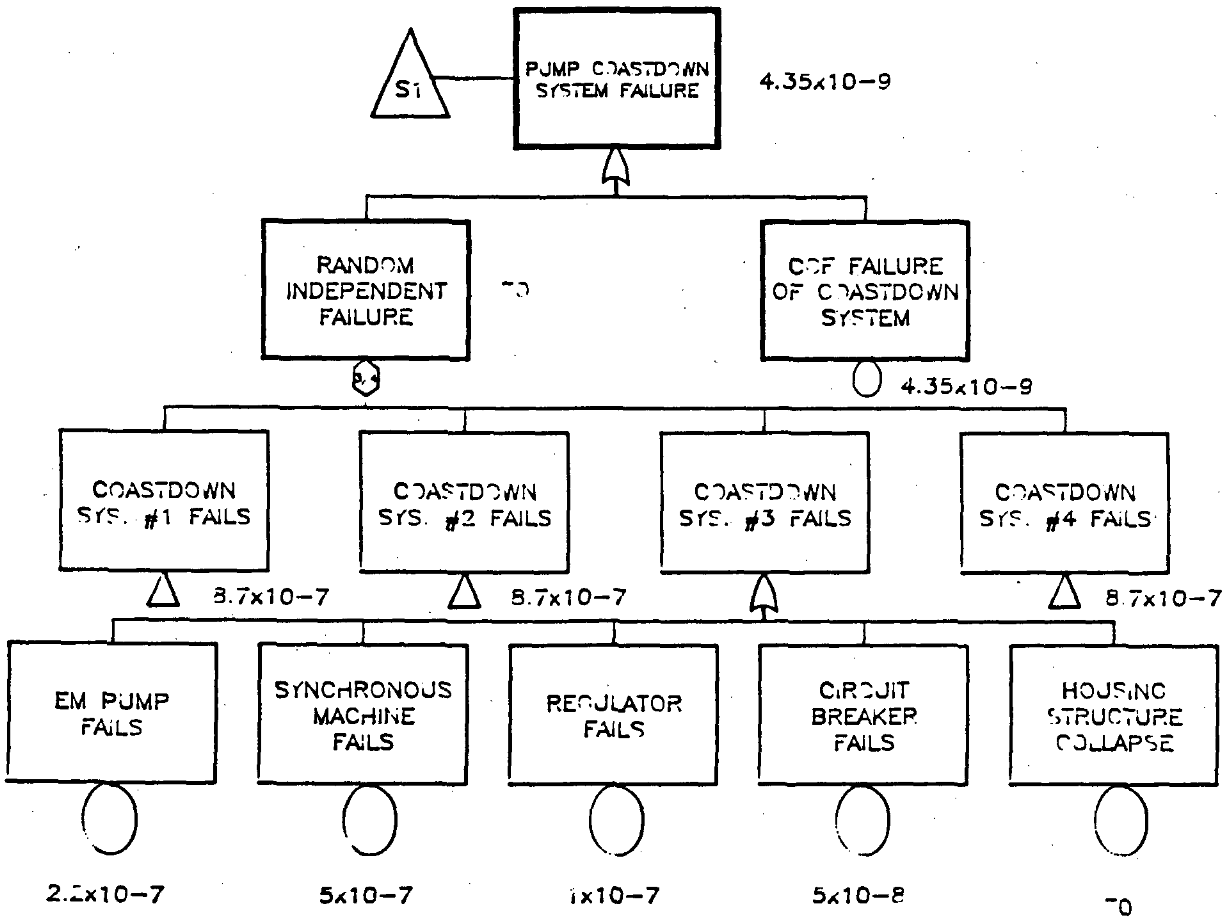


Figure A4.2-35 PUMP COASTDOWN SYSTEM FAULT TREE FOR NON-SEISMIC ACCIDENT INITIATORS

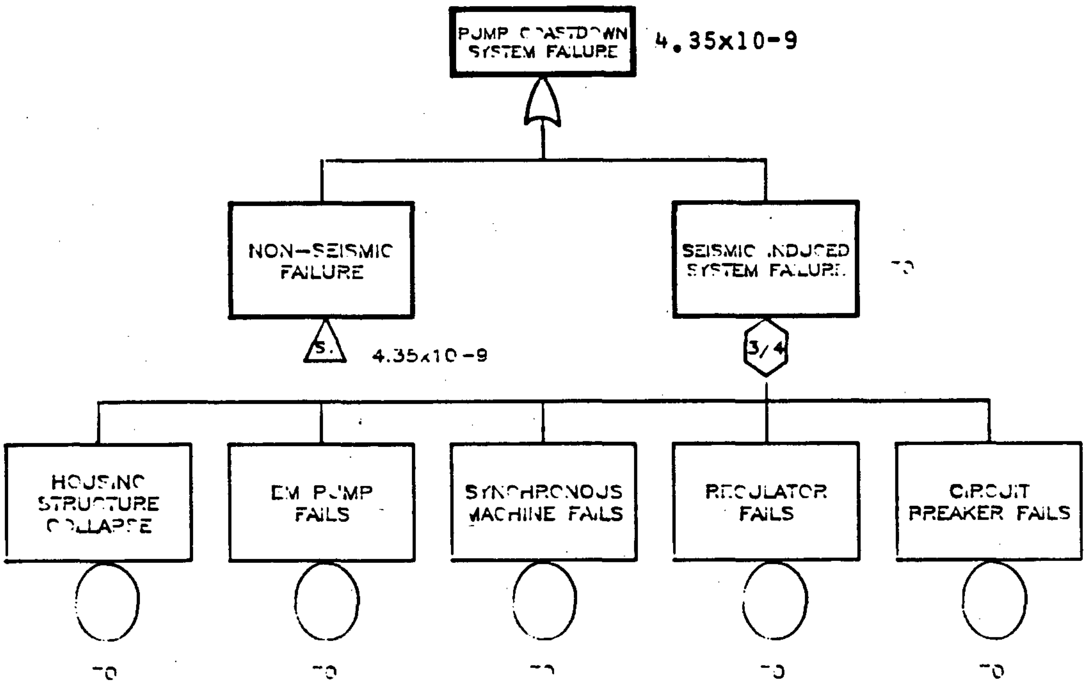


Figure A4.2-36 - PUMP COASTDOWN SYSTEM FAULT TREE FOR SEISMIC INITIATOR - INITIATING EVENT 4

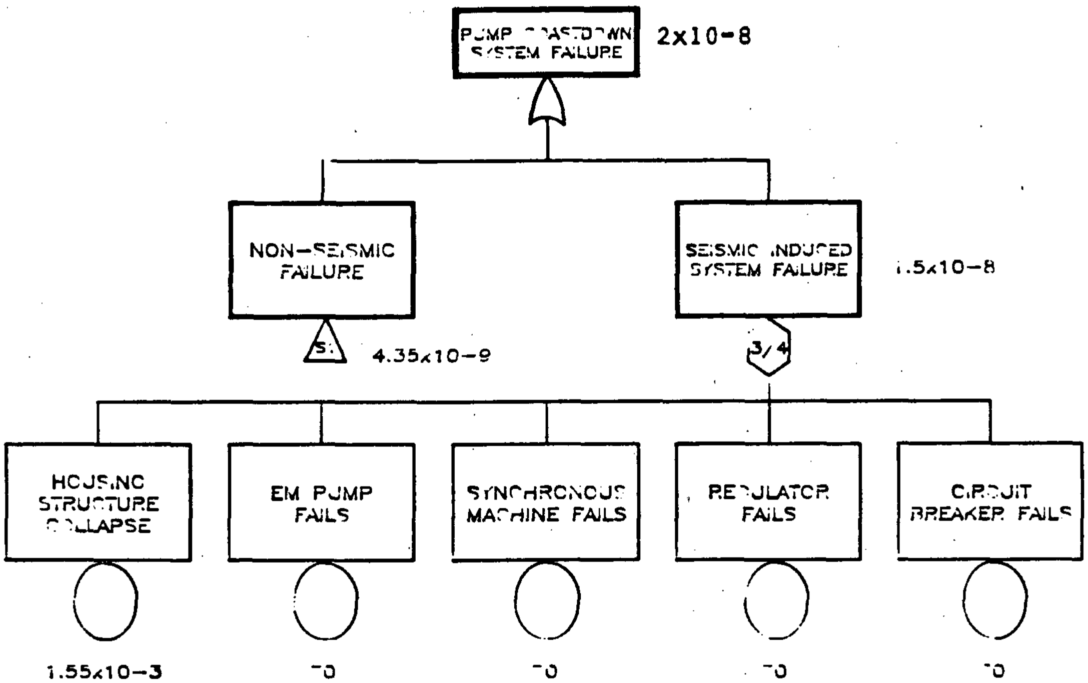


Figure A4.2-37 - PUMP COASTDOWN SYSTEM FAULT TREE FOR SEISMIC INITIATOR - INITIATING EVENT 5

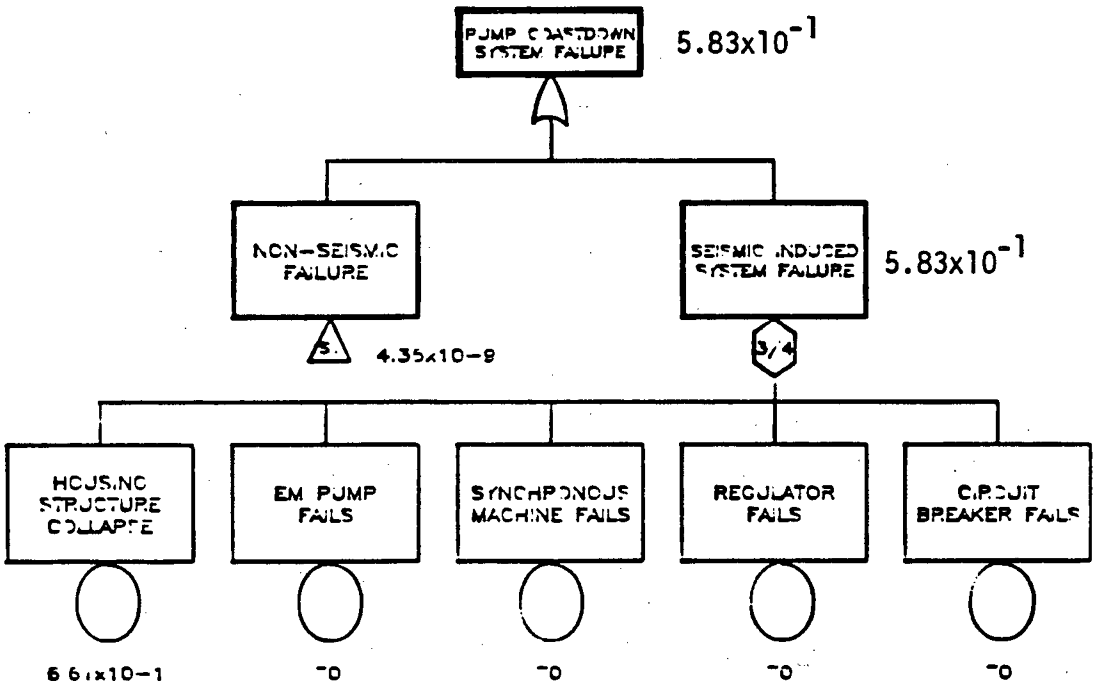
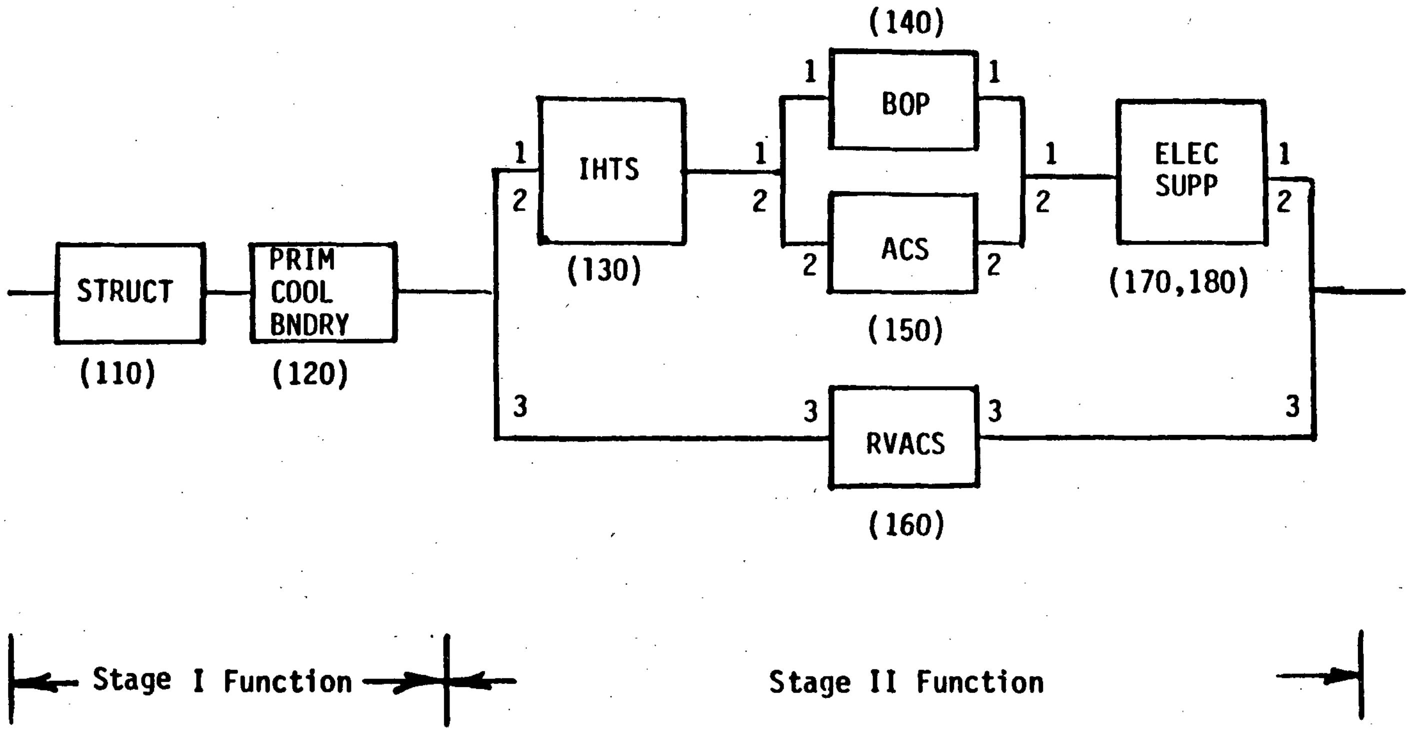
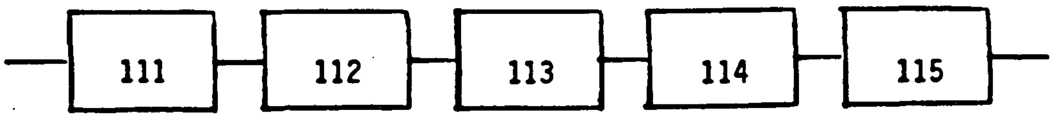


Figure A4.2-38 - PUMP COASTDOWN SYSTEM FAULT TREE FOR SEISMIC INITIATOR - INITIATING EVENT 6

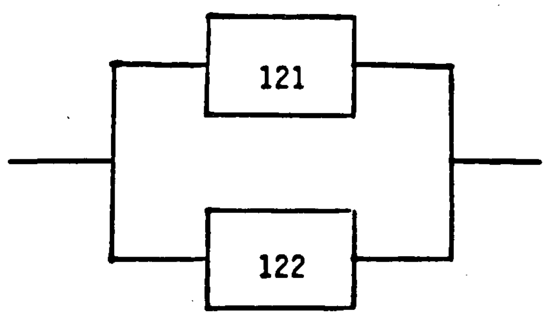


a) System Level

Figure A4.2-39 PRISM SHRS RELIABILITY BLOCK DIAGRAM

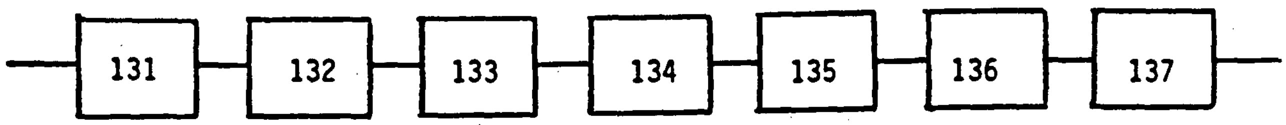


b) Primary Coolant Flow Path (110)

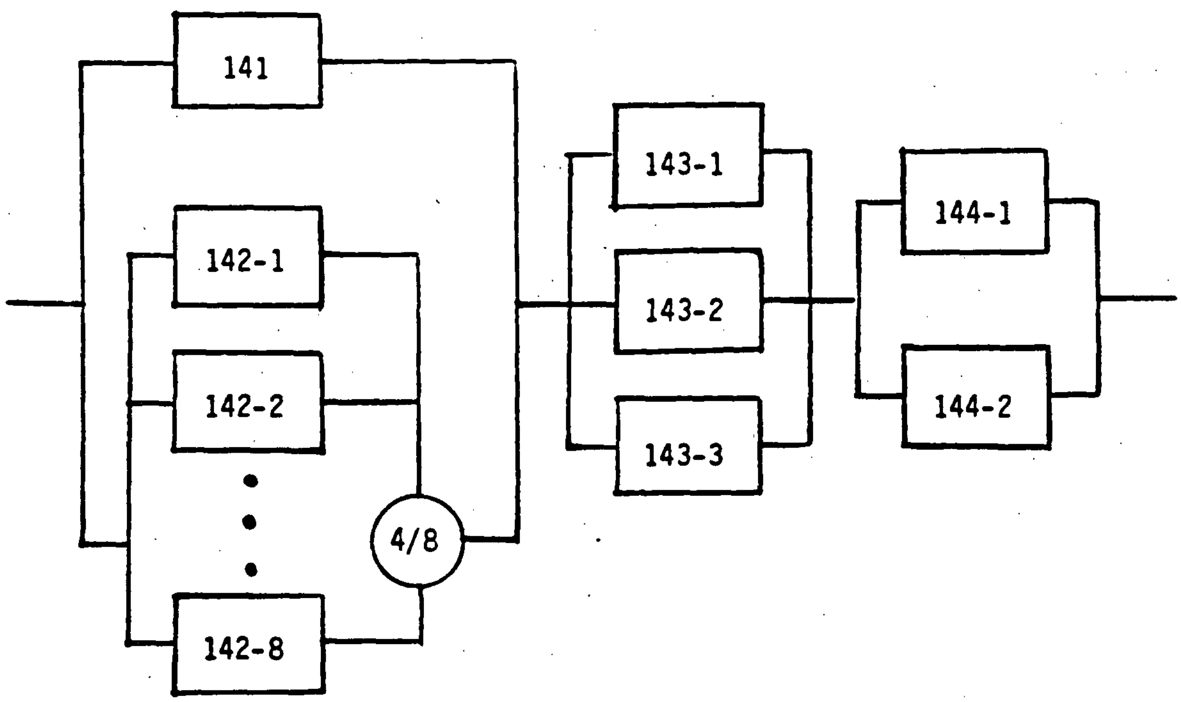


c) Primary Coolant Boundary (120)

Figure A4.2-39 PRISM SHRS RELIABILITY BLOCK DIAGRAM (cont)

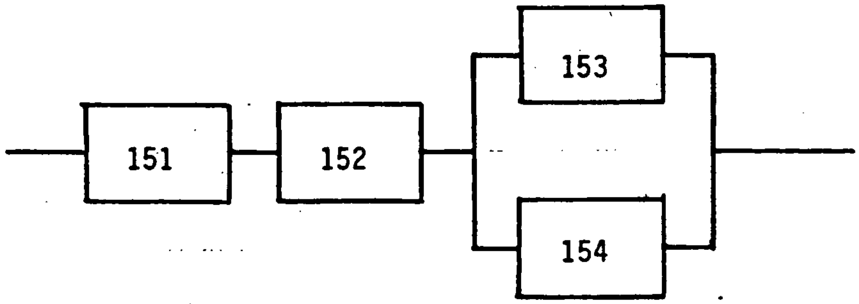


d) IHTS (130)

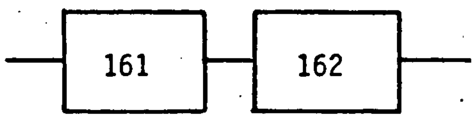


e) BOP (140)

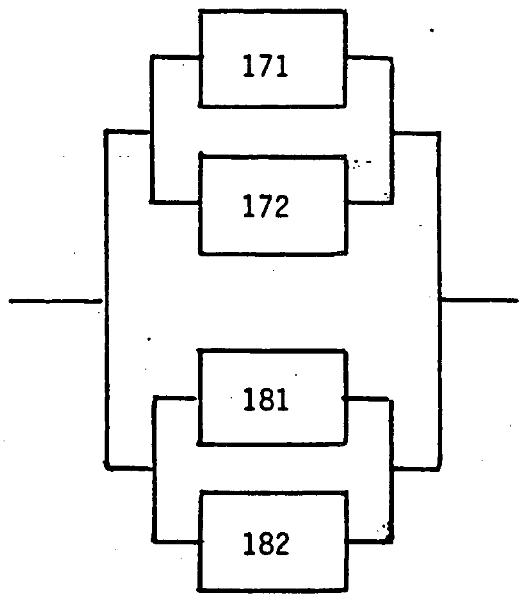
Figure A4.2-39 PRISM SHRS RELIABILITY BLOCK DIAGRAM (cont)



f) ACS (150)



g) RVACS (160)



h) Electrical Power Supply System (170,180)

Figure A4.2-39 PRISM SHRS RELIABILITY BLOCK DIAGRAM (cont)

A4.3 Core Response Event Trees

A4.3.1 Introduction

Given one of the accident types defined in the previous section, the reactor core will go through a transient phase until neutronic shutdown is accomplished, either by natural processes or by human intervention. Core response event trees define the possible scenarios and end states of this transient phase. The end states are differentiated by the following parameters:

1. Configuration and physical form of the radioactive material source. The source includes the fuel, fission products and radioactive sodium.

2. Leak paths which may be opened up during the transient phase and which may lead to the release of sodium or radioactive material from the reactor vessel.

3. Primary coolant enthalpy and state of the shutdown heat removal system.

The above parameters depend on the following factors:

1. Irradiation history of the reactor core before the accident occurs.

2. Mechanisms of radioactive material release from the fuel and cladding which may be active in the course of an accident.

3. Mechanisms of energy generation and distribution which may be active in the course of an accident

A general discussion of these factors is presented below. This is followed by the event tree models which define the scenarios and end states for each accident type.

A4.3.2 Irradiation History

Irradiation history of the reactor core determines two significant risk-related parameters:

1. The radioactive material inventory, and

2. The mode of release of the radioactive materials from the fuel and the cladding.

As described below, the PRISM reactor is characterized by a nonuniform irradiation pattern across its core. This nonuniformity of irradiation may result in the time phasing of accidents over a relatively long period of time with corresponding reduction in the accident severity.

The goal exposure of the PRISM driver fuel (42 assemblies per module) is set at 147.3 Mwd/Kg (~14.7 at %). The selected fuel management scheme requires this exposure to be achieved in three operating cycles of 20 months each. At the end of an equilibrium cycle (EOEC), one third of the driver fuel assemblies are irradiated at this goal exposure, one third irradiated to 2/3 goal exposure, and the remaining driver fuel assemblies are irradiated to 1/3 goal exposure. On refueling, the one third of the driver assemblies which has been irradiated to goal exposure is moved to in-vessel storage racks where it decays for the duration of one cycle. Assemblies with fresh fuel replace the ones moved to the storage racks. Thus, at the beginning of equilibrium cycle (BOEC), the core has three batches of driver fuel, (of 14 fuel assemblies each), which are irradiated to 0, 1/3,and 2/3 of the goal exposure.

The core internal and radial blankets are irradiated to much lower levels. The peak exposure of the internal blanket (25 assemblies) and radial blanket (36 assemblies)is 55 Mwd/Kg.

A4.3.3 Modes of Radioactive Material Release

The PRISM Reactor uses a U-Pu-Zr metal alloy fuel (70% U, 20% Pu, 10% Zr) and an HT9 cladding. During normal operation, the fission gas and volatile fission products produced by fuel burnup may escape from the fuel and collect in the fission gas plenum. In the course of an accident, fuel may interact with the clad to form a low melting-point eutectic. Under more severe conditions, fuel may melt, vaporize, or oxidize if exposed to an oxidizing substance. The following discussion identifies the factors which affect the fraction of fission products released for each mode of release. Release fractions for each mode of release are then estimated for use in subsequent analysis. The estimates are assigned by judgement based on the irradiation pattern of the PRISM fuel and radioactive material release information in the following sources: the results of metal fuel tests reported in Ref. A4.3-1 and the analysis of oxide fuel release reported in Ref. A4.3-2.

A4.3.3.1 Gas Plenum Release

Test results of fission product release from metal fuel during normal operations are reviewed in Ref. A4.3-1.

No results were reported for U-Pu-Zr alloy fuels. However, the reported cases show a strong dependence of the release fraction on the fuel smear density and burnup. For the EBR-II fuel used in the Mark II core, which has the same smear density of 75% as the PRISM driver fuel, the fraction of fission gas released to the fission gas plenum increases with burnup as shown in Figure A4.3-1. The figure shows that the release fraction is less than 50% for burnups less than ~2.5 at %, and increase to 87% for 15 at % burnup. The figure also shows that the fission gas retained in the fuel maintains a constant pressure over burnups greater than about 5%, which means that all fission gas generated above this burnup is released at the same rate of generation.

Ref. A4.3-1 reports that the liquid fission products, primarily cesium, were distributed between the fuel and bond sodium. Specifically, recent gamma scanning of several Mark-II elements has shown that the amount of cesium released to the bond sodium in the plenum region increased with burnup, and that up to 40% had been transported above the fuel at 5.5 at % burnup.

The tests reported in Ref. A4.3-1 show that the solid fission products remained in the fuel matrix either in solid solution or as intermetallic compound precipitates.

Compared to the release from the U-Pu mixed oxide fuel used in Ref. A4.3-2, the above results indicate that significantly larger fractions of gaseous and volatile fission products are released to the fission gas plenium from the metal fuel (only 3% of gases and 15% of Cs are released from the oxide fuel of Ref. A4.3-2). On the other hand, significantly smaller fractions of solids are released to the fission gas plenum from the metal fuel. (1 to 10% are released from the oxide fuel of Ref. A4.3-2). The first observation is attributed to the interconnected porosities which form in the metal fuel and allow unimpeded release of gases and volatiles (Ref. A4.3-1). The second observation is attributed to the lower operating temperature of the metal fuel, which impedes the mobility of solid fission products.

Based on the above observations, and the irradiation pattern of the PRISM metal core, fractions of the fission product released to the gas plenum during operation were estimated for different fission product groups as shown in Table A4.3-1.

The fission products released to the fission gas plenum and bond sodium during normal operation will escape the fuel pin in case of a clad failure. The escape fraction depends on the location and size of clad failure and on the fission products enthalpy. These parameters depend in turn on the accident scenario which leads to clad failure. To simplify the assessment in this work, the dependence of the escape fraction on the accident scenario was not considered, and the average escape fraction values estimated in Ref. A4.3-2 were used. The escape fractions and fractions of fission product released on clad failure are shown in Table A4.3-2

A4.3.3.2 Eutectic Release:

The PRISM metal fuel and cladding have a low melting-point eutectic (~725°C) composed of one part fuel to 4 parts cladding. The rate of eutectic formation increases exponentially from about 725°C to 1080°C as shown in Figure A4.3-2. The eutectic formation is impeded by Zr which tends to migrate to the innermost and outermost regions of the metal fuel (Ref. A4.3-1). Consequently, cladding attack by eutectic formation is not expected to consume all the cladding. In this study, up to 25% of the cladding is assumed to be involved in eutectic formation for temperatures greater than 725°C. Based on the PRISM fuel pin dimensions and the fuel-clad fractions for eutectic formation, it is estimated that 2% of the fuel will combine with 25% of the clad to form the eutectic alloy.

The low eutectic melting point means that only noble gases (Xe, Kr), halogens (Iodine with a boiling point of 183°C, Br with a boiling point of 59°C), and alkali metals (Cs with boiling point of 685°C) have a chance of being released from the fuel. Other fission products have boiling points above the eutectic melting point by at least 250°C. Consequently, the release fractions of these fission products from the formed eutectic liquid has been assumed as 0. A release fraction of 100% of the noble gases, halogens, and alkali metals in the eutectic liquid has been assumed for this study. The results are summarized in Table A4.3-3.

A4.3.3.3 Meltdown Release

The melting point of the metal fuel (~1150°C) is much less than that of oxide fuel (~2300°C). Consequently, a smaller fraction of fission products with boiling points greater than 1150°C is expected to be released from molten metal fuel than is released from molten oxide fuel. Consequently, the mean fractions estimated in Ref. A4.3-2 for release of these fission products from oxide fuel were reduced by a factor of 10 for application in this study. For more volatile fission products, a release fraction of 1.0 has been used. The obtained release fractions are shown in Table A4.3-4.

A4.3.3.4 Vaporization Release

For extreme accidents where a portion of the fuel is vaporized, a release fraction of 1 is used for all fission products in the vaporized fuel. Except for fission gases, the fission products released in this mode are assumed to form aerosol sized particles upon their release.

A4.3.4 Energy Generation and Distribution

The nuclear energy generated in an accident depends on the reactivity feedback which becomes active in the course of the accident. For the PRISM reactor, the reactivity feedback mechanisms respond in three distinctive time frameworks:

1. Prompt feedback - which responds almost instantaneously to changes in the fuel temperature. The Doppler effect and fuel axial expansion (if unimpeded by fuel/clad friction forces) provide this feedback.

2. Delayed feedback - which responds to changes in primary coolant temperature in relatively longer time framework (from a second to a few minutes). Thermal expansion of the control rods, core radial expansion, and vessel expansion provide this feedback.

3. Long term feedback - which includes relatively slow feedback mechanisms such as burnup or reactivity changes induced by human actions.

Accident analyses so far conducted for the PRISM metal core (References A4.3-3 and A4.3-4) and for feasibility studies of metal-fuel cores (References A4.3-5 thru A4.3-7) have concentrated on relatively slow transients where delayed reactivity feedback becomes effective. Results of these analyses show benign consequences with system temperatures below the threshold values for clad/fuel eutectic point, sodium boiling, and fuel melting. Since the occurence of these phenomena is necessary for any measurable public consequences, accident conditions which may lead to their occurrence and the consequences of such occurrence are discussed below.

A4.3.4.1 Clad Failure by Clad/Fuel Eutectic Formation

Clad failure by eutectic formation and the subsequent release of the formed fuel/clad alloy could affect the accident progression in three ways: First, the resulting fuel motion could have negative or positive reactivity feedback depending on the direction of the fuel motion and the location of release. Second, solidification of the formed alloy in cold regions may cause flow blockages. Third, the release of the high-pressure fission gas from the gas plenum may impede the primary coolant flow by causing flow chugging or flow stagnation.

The average reactivity worth of a PRISM fuel assembly is estimated at about $5 for both the fuel and structural material in the subassembly. Structural material reactivity worth is estimated at about -.3$. The corresponding reactivity worth of fuel only is $5.3 per subassembly. For an eutectic formed from 2% fuel and 25% clad (see previous section), the reactivity worth is .02 x 530_ -.25 x 30_ = 10.6-7.5 ~3_ per assembly. If 25% of all driver fuel clad (42 assemblies) fail by eutectic formation, the reactivity worth of the formed alloy is $1.26. Therefore, if an acident results in such a massive failure and the formed alloy is swept out of the core, this results in a negative reactivity of -1.26$. This is judged as an upper bound of the negative reactivity due to eutectic alloy sweepout.

Under full pump flow conditions, the eutectic alloy formed is expected to be driven upward where it may stick to upper core structures like a lubricant, may settle in low-flow areas as the primary flow changes direction before it enters the IHX, or may flow with the primary coolant to the lower coolant (cold) plenum where it may freeze and block subassembly inlet orifices. The latter scenario is judged as unlikely.

Under reduced flow conditions, e.g., following pump coastdown, the eutectic alloy may flow downward under gravity. As it reaches the lower coolant region, the lower coolant temperature may cause it to freeze and block the coolant passages. This blockage scenario is judged as more likely than the one with full pump flow.

The fuel pin irradiated to goal exposure is expected to produce about 700 cm^3 of fission gas at STP. The pressure in the fission gas plenum in such a pin could reach about 90 atmospheres. Release of this gas following clad failure by eutectic formation would create flow resistance and may result in flow reversal. In particular, under low flow and high power conditions such as experienced in loss of flow accident, this could lead to rapid sodium coolant break-up and voiding. From the above observations, it is concluded that failure of the cladding by eutectic formation is expected to speed up the primary coolant heat-up and voiding in cases where the power to flow ratio is above normal such as in severe loss of flow accidents

A4.3.4.2 Sodium Voiding

Voiding the PRISM reactor core from sodium will add a reactivity of $5.13. It is estimated that it will take about 2.6 full-power-seconds (FP secs) to vaporize the sodium in the core region. Therefore, if the primary coolant is brought to a standstill at full power, it will take 2.6 secs to void the core. The average reactivity ramp corresponding to this scenario is about 2$/sec. The significance of this ramp rate derives from the following considerations.

The Doppler effect (prompt reactivity feedback) for the PRISM reactor is estimated to add a negative reactivity of 36¢ if the fuel temperature is raised from its normal operation to its melting point. The time constant of metal fuel cores is estimated at $\sim$.3 sec. Therefore, reactivity ramp rates greater than $\sim$0.36/0.3 sec = 1.2 $/sec are expected to be fast enough to cause fuel melting and possible fuel vaporization before delayed reactivity feedback mechanisms have time to be effective. This means that sodium voiding could lead to core disassembly.

A4.3.4.3 Accident Energetics

As indicated earlier, accident analyses conducted so far for the PRISM metal core cover only slow transients where delayed feedback mechanisms become effective. More severe accidents involving, for example, coherent sodium voiding resulting from failure of pump coastdown, could lead to superprompt critical core (net reactivity greater than $1) and subsequent energetic disassembly. To assess the consequences of such accidents on radioactive material release from the core and on the structural integrity of the vessel, use has been made of the parametric evaluations in Reference A4.3-8 and reported accident analyses for the FFTF in References A4.3-9 and A4.3-10.

Reference A4.3-8 conducted parametric evaluations to study the effect of the Doppler coefficient, power flattening, and the equation of state, among other factors, on the explosive energy resulting from reactivity additions. Review of these evaluations indicated that the PRISM core and the FFTF should have comparable energetics under severe transients leading to core disassembly. Consequently, the FFTF accident analyses reported in Reference A4.3-9 and A4.3-10 were used as a basis for assessing the core response for PRISM accidents involving sodium voiding.

A4.3.5 Event Tree Models

Reactivity feedback considerations have dictated the use of two types of core response event trees. The first type applies to loss of flow (LOF) and loss of heat sink (LOHS) accident types. For these accidents, the primary coolant is the first to respond by heat-up. If inherent negative feedback resulting from this heatup is adequate for shutdown without core damage, the analysis reduces to questions of decay heat removal and long term coolability. If inherent feedback mechanisms are inadequate for safe shutdown, then the accident may progress through the phases of eutectic formation and penetration of the cladding, sodium voiding, meltdown, and energetic accidents.

The second type of event trees applies to transient overpower (TOP) and combined transient overpower/loss-of-flow accidents. For these accidents, the reactor does not shutdown unless control rods are inserted in the core. With adequate inherent reactivity feedback, the reactor power will stabilize at an elevated level. This gives room for human intervention to shutdown the reactor or for the long term reactivity feedback mechanisms such as burnup as a last resort. If the reactor is shutdown without damage by any of these mechanisms, then the situation becomes similar to safe shutdown following the LOF and LOHS accidents with questions only regarding decay heat removal and long term coolability requiring investigation. If safe shutdown is not accomplished, then accident progression to the phases of eutectic formation, sodium voiding, fuel melting and sweepout, and energetics are investigated.

The above considerations lead to a spectrum of accident scenarios with a spectrum of core damage levels, radionuclide releases, and structural impact. The possible spectrum of outcomes is divided into twelve core damage categories. Six of these apply to situations where shutdown heat removal is available. The other six categories apply to similar scenarios when the shutdown heat removal system is unavailable. The damage categories are defined in the following section. This is followed by definitions of the events in the core response event trees and the probabilities assigned for these events.

A4.3.6 Core Damage Categories

Core damage categories are defined in terms of the following four parameters:

1. Fission products: Fraction released and where released

2. Fuel: Fraction released, physical form if outside cladding, where released

3. Vessel or vessel seals damage, if any

4. Primary coolant temperature

The categories are labled C1, C2,...C6 for end states with the shutdown heat removal system (SHRS) available and C1S through C6S for corresponding cases with SHRS unavailable. The damage severity increases as one moves from C1 to C6 (or C1S to C6S).

A4.3.6.1 Core Damage Categories C1 and C1S

In these categories, the reactor is shutdown with , at most, 25% of the clad failing by creep rupture, overpressure, swelling, overcooling, thermal stresses or cycling. No eutectic alloy or fuel penetrates through the clad. Twenty-five percent of fission gas and volatiles of the fission gas plenum is released. The categories have the following parameters:

1. Fission products released:

Fission Product	Fraction Released	Location
Xe, Kr	15%	Cover Gas
I, Br	2%	Primary Na
Cs, Rb	4%	Primary Na
Others	0	-----

2. Fuel released: None

3. Vessel or seal damage: None

4. Primary sodium temperature: Normal operation.

A4.3.6.2 Core Damage Categories C2 and C2S

In these categories, the reactor is shutdown with at most 2% fuel released in clad/fuel eutectic form. All fission products in the gas plenum are released. Radionuclides in the 2% of the fuel released in eutectic form are also released. The categories have the following parameters:

1. Fission product released:

Fission Product	Fraction Released	Location
Xe, Kr	65%	Cover Gas
I, Br	8%	Primary Na
Cs, Rb	19%	Primary Na
Other	2%	Plated out on on upper core structures

2. Fuel released, 2% in eutectic form in primary Na coolant

3. Vessel or seal damage: None

4. Primary Na temperature 750°C

A4.3.6.3 Core Damage Categories C3 or C3S

These categories are similar to C2 except that the reactor operates at an elevated power (100-125% nominal power) as a result of a TOP before shutdown is achieved. The categories have the following parameters:

1. Fission product released:

Fission Product	Fraction Released	Location
Xe, Kr	75%	Cover gas
I, Br	10%	Primary Na
Cs, Rb	20%	Primary Na
Other	2%	Plated out on upper core structures

2. Fuel release: 2% eutectic form in primary Na coolant

3. Vessel or seal damage: None

4. Primary Na temperature: 800°C

A4.3.6.4 Core Damage Categories C4 and C4S

In these categories, the reactor is shutdown with 15% fuel molten, of which 1/3 (5%) is squirted out of the fuel pins to form coolable debris or partial coolable blockages (a Fermi I-type accident). The remaining molten fuel stays within the fuel pin. The molten fuel release is preceded by a eutectic melt release involving 2% of the fuel. The categories have the following parameters:

1. Fission products released:

Fission Product	Fraction Released	Location
Xe, Kr	70%	Cover gas
I, Br	20%	Primary Na
Cs, Rb	30%	Primary Na
Te	2%	Primary Na
Sr	.2%	Primary Na
Others	.05%	Primary Na

2. Fuel release: 5% in the form of coolable debris or blockages
   2% in the form of eutectic alloy in the primary coolant

3. Vessel or seal damage: None

4. Primary Na temperature: 800°C

A4.3.6.5 Core Damage Categories C5 and C5S

In these categories, the reactor is shutdown with 50% of the fuel molten, of which half (25%) is squirted out of the fuel pins, fragmented into debris, and swept out of the core region to resettle on the horizontal baffles and in the IHX. Distribution of these locations is uncertain due to lack of mechanistic analysis. The possibility of local concentrations on the IHX lower walls and melt-through to the lower vessel head cannot be excluded at this time. The debris collecting in the lower vessel head may also concentrate around the core support skirt area leading to a possibility of vessel melt-through. These categories have the following parameters:

1. Fission products released:

Fission Production	Fraction Released	Location
Xe, Kr	85%	Cover gas
I, Br	50%	Primary Na
Cs, Rb	60%	Primary Na
Te	5%	Primary Na
Sr	1%	Primary Na
Other	1%	Primary Na

2. Fuel release: 24% - Debris on thermal baffle, lower vessel head and IHX
   1% - Aerosol-size particles in primary Na

3. Vessel and seal damage: No immediate damage. Chance (~1%) of vessel melt-through due to noncoolable local debris concentration.

4. Primary Na temperature: 850°C

A4.3.6.6 Core Damage Categories C6 and C6S

In these categories the reactor is shutdown with 100% of the fuel molten, 10% of which is initially in a vapor form and subsequently forms aerosol-size particles. Vessel head seal is damaged as a result of the accompanying energetics leading to the release of 5% of the fuel in aerosol form and 1000 Kg of Na to the head access area. The categories have the following parameters:

1. Fission products released:

Fission Product	Fraction Released	Location
Xe, Kr	100%	Head access area
I, Br	80%	Primary Na
Cr, Rb	90%	Primary Na
Te	25%	Primary Na
Sr	10%	Primary Na
Other	10%	Primary Na

2. Fuel release: 25% - in debris form on thermal baffle and upper vessel structures
   65% - in debris form on core support plate
   5% - in aerosol form in head access area
   5% - in aerosol form in primary Na

3. Vessel and seal damage: Seal damage leading to release of 5% fuel (aerosol) and 1000 Kg of primary Na to head access area

4. Primary Na temperature: 850°C

A4.3.7 Core Response Events Definitions and Probabilities

As discussed earlier, the loss of flow (LOF) and loss of heat sink (LOHS) accidents are presented by the same event tree structure. The transient overpower (TOP) and combined Transient Overpower/Loss of Flow (TOP/LOF) accidents are presented by the same event tree structure which is different from that of the LOF and LOHS.

The LOF and LOHS event trees are presented in Figures A4.3-3 through A4.3-6 for the four accident types LOF: F1, LOF; F3, LOHS; H2; and H3, LOHS. Each tree is described in terms of five events:

1. Clad unfailed by fuel/clad eutectic;

2. No Na boiling or voiding;

3. Flow unimpeded by blockage or fission gas release;

4. Energy released insignificant;

5. Energy released undamaging to vessel boundary.

Definition of these events and the basis for the probabilities assigned in Figures A4.3-3 through A4.3-6 are described below.

A4.3.7.1 Clad Unfailed by Fuel/Clad Eutectic

This event means that the reactor reaches its stable end state of neutronic shutdown without fuel/clad eutectic formation. For the nominal loss of flow accident F1, analysis in Reference A4.3-4 has shwon that the primary coolant temperature reaches a maximum temperature of 675°C in less than 50 secs. The temperature then drops to a constant of ~650°C. The fuel centerline temperature drops to a value very close to the coolant temperature after about 100 secs. From these results, it is concluded that the fuel/clad temperature is expected to stabilize at 660°C for this accident. At this temperature, the possibility of forming a fuel/clad eutectic alloy is remote as discussed earlier. Analysis in Ref. A4.3-4 has shown a larger margin to eutectic formation for the nominal loss of heat sink accident.

Reference A4.3-11 has estimated the standard deviation in SASSYS coolant temperature calculation for metal fuel cores at ~30°C. Therefore, at the 2.33σ level (Probability = 0.99), the stabilized fuel/clad temperature for F1 is expected to be less than 660 + 2.33 x 30 ~730°C. At this temperature, clad/fuel eutectic formation is too slow to be of concern. The fuel/clad temp for H2 at the 2.33σ level is expected to be less than that for F1.

Based on the above observations, fuel/clad eutectic formation has been assigned a conditional probability of 0.01 given F1 or H2 as shown in Figures A4.3-3 and A4.3-5.

The LOF accident F3 covers a spectrum of LOFs which are more severe than F1 due to faster-than-nominal (pump coastdown) loss of in-core flow or loss of the inherent negative reactivity feedback effectiveness. The LOHS accident H3 covers also a spectrum of LOHS accidents where the negative inherent reactivity feedback is ineffective. In both accidents, heatup of the primary sodium coolant to cause rapid fuel/clad eutetic penetration of the clad is almost certain. Consequently, the conditional probability of failure by fuel/clad eutectic formation has been assigned the value of 1 as shown in Figures A4.3-4 for F3 at A4.3-6 for H3.

A4.3.7.2 No Na Boiling or Voiding

This event means that the reactor reaches its stable end state without bulk sodium boiling or voiding. For the nominal LOF accident (F1), Reference A4.3-12 estimates that the margin to sodium boiling is about eight times the standard deviation associated with SASSYS results, i.e., $P(T_{Sat} \leq T_{F1})$ ~.67 x 10^{-15} for a normally distributed T_{F1}. Consequently, the probability of sodium boiling or voiding has been assigned the value of 0. in Figure A4.3-3. The nominal LOHS accident (H2) (Fig. A4.3-5) has also been assigned the probability value of zero for the sodium boiling or voiding based on the discussion of the previous section.

As discussed earlier, accidents F3 and H3 cover a spectrum of loss of flow and reactivity feedback effectiveness. For moderate deviations from the nominal flow and reactivity feedback of accidents F1 and H2, primary sodium temperatures may not reach the saturation point. An account of system event sequences leading to such a range of moderate deviations and an analysis to quantify the flow decay patterns and extent of reactivity feedback effectiveness which constitute the bounds of these deviations are required for objective probability assessment of the event of no sodium boiling for these accidents. Due to the lack of such analyses at this time, a conditional probability of .5 has been assigned subjectively to this event given F3 or H3 as shown in Figures A4.3-4 and A4.3-6.

A4.3.7.3 Flow Unimpeded by Blockages or Fission Gas Release

This event means that the reactor stabilizes by shutdown without local coolant voiding which may add reactivity and without local blockages (Fermi I-type accident). The event is meaningful only if no bulk sodium boiling or voiding occurs. For the nominal LOF (F1) and LOHS (H1), the probability of this event is judged as negligible.

As discussed earlier, the fuel/clad eutectic alloy penetrating the clad in an LOF may move downward, freeze, and form blockages in the cold sodium region. It was also noted earlier that the LOF accident F3, covers a spectrum of flow decay patterns and reactivity feedback loss. Therefore, only a portion of the sequences assigned to F3 is expected to result in significant core involvement in eutectic formation. Moreover, only a portion of those cases resulting in significant core involvement are expected to cause complete local blockage or flow starvation. Consequently, a conditional probability value of .01 for impeding the flow by blockages or fission gas release has been assigned by judgement for F3 as shown in Figure A4.3-4. Due to the higher power level encountered in LOHS, a higher probability value of 0.1 has been assigned to H3 (Fig. A4.3-6).

A4.3.7.4 Energy Release Insignificant

This event means that the reactor is shutdown with much of the core fuel (85% or more) intact. No core disassembly or structural damage is expected. The event may contain partial core meltdown (15% or less) due to partial flow starvation or due to addition of reactivity ramp rates in the order of a few cents per sec (less than 10¢/sec) for a total reactivity addition of less than $1.

A typical scenario leading to the above event is the incoherent voiding of up to 10 driver fuel subassemblies. As the voiding becomes more coherent (e.g., ten assemblies void at the same time leading to a ramp rate of ~50¢/sec, or more assemblies are voided, the probability of significant energy release increases. These considerations lead to the conditional probability assignment shown in Figures A4.3-4 and A4.3-6 for the LOF accident F3 and the LOHS accident H3. The shown probabilities have been assigned by subjective judgement. Notice that the conditional probability of significant energy release is higher when whole-core sodium voiding or boiling occurs. Notice also that the conditional probability of significant energy release given partial core boiling or voiding is higher for the LOHS accident (H3) than that for the LOF accident (F3). This reflects our judgement that voiding in the former case will be more coherent due to the higher power level and sodium temperature expected in an LOHS accident.

A4.3.7.5 Energy Released Undamaging to Vessel

This event means that the reactor is neutronically shutdown but with more damage than the earlier event (15% to 50% of the core molten). No energetic core disassembly is expected but up to 25% of the core is squirted out of the fuel pins to form debris which may have a slight chance (1%) of forming locally uncoolable concentrations which may melt through the vessel. The event results from partial flow starvation or net reactivity ramp rates from 10¢/sec to 60¢/sec, up to a total reactivity addition of ~$2.

A typical scenario leading to the above event is the coherent voiding of up to 20 driver subassemblies. If coherent voiding involves more subassemblies, energetic core disassembly becomes more likely. Based on the sodium void worth, the maximum voiding rate for the PRISM core, and analyses of the FFTF (Refs. A4.3-9 and A4.3-10), the maximum damage expected from core disassembly is failure of the PRISM seals between the vessel head and the rotating plug. This worst-scenario case is judged to lead to a whole core meltdown with 10% of the fuel vaporized. Based on the analysis in Reference A4.3-10, it was concluded that failure of the vessel seals will lead to release of 5% of the fuel in aerosol form and 1000 Kg of primary sodium to the head access area.

The necessary condition of the coherent voiding of 20 or more driver assemblies for energetic core disassembly is reflected in the probability assigned for this event for the LOF and LOHS accidents F3 and H3 shown in

Figures A4.3-4 and A4.3-6. The probabilities shown have been based on subjective judgement.

The core response scenarios shown in Figures A4.3-3 through A4.3-6 were assigned to core damage categories C1 through C6 in a straightforward manner, as a result of the above definition of core response events, and the definitions of core damage categories presented earlier. Notice the close relationship between the definition of the core response events and the core damage categories.

The event trees for the TOP and combined TOP/LOF accident types are shown in Figures A4.3-7 through A4.3-12. Each tree has six events. Three of the events are the same as those used for the previous event trees (flow unimpeded by blockage or fission gas release, energy released insignificant, and energy released undamaging to vessel). Consequently, these events are not discussed further here for brevity. Notice, however, that the probabilities shown for these events in Figures A4.3-7 through A4.3-12 have been assigned by judgement based on available analyses, and on consideration of the reactivity added externally or resulting from sodium voiding, and fuel motion. Notice also that from definition of the accident type the initial reactivity addition increases as one moves from P1 through P4 and from G3 to G4. This explains the general trend of increasing probability of core damage as one moves from P1 through P4 and G3 to G4.

Definition of the remaining three events (shutdown before clad failure by fuel/clad eutectic, shutdown by fuel/clad sweepout, and shutdown before significant damage) and the basis for their probability assignments are discussed below.

A4.3.7.6 Shutdown Before Clad Failure by Fuel/Clad Eutectic

As discussed earlier, the PRISM inherent reactivity feedback mechanisms are not expected to shutdown the reactor for transient overpower accidents without scram. For the nominal TOP of a control rod withdrawal without scram, Ref. A4.3-4 shows that the power level stabilizes at about

120% of the full power. This power level is expected to be higher for more severe TOPs.

Despite the fact that the power stabilizes at some elevated level which may be within the capability of the BOP for some time, the resulting situation is undesirable since it may lead to subsequent failures by creep rupture of the cladding, fuel/clad eutectic formation, or creep rupture of BOP or vessel components. For this reason, it is expected that shutdown will be tried by human intervention to bring the reactor under control and repair the cause of the accident. This leads to the first event in the TOP and TOP/LOF event trees; namely, shutdown before clad failure by fuel/clad eutectic melting.

Following a TOP, the reactor is expected to be shut down by human intervention. The probabilities of this event for different TOP and TOP/LOF types have been assigned by judgement on possible human error under the given stress situation and the grace period allowed before failures begin to occur. Therefore, conditional probability values of .01, .05, .5, and .99 were assigned for failure of shutdown by human intervention given the TOPs P1, P2, P3, and P4 respectively. Notice that the severity of the TOP increases as one moves from P1 through P4. Higher probabilities of failure were assigned for the TOP/LOF accidents (G3 and G4) than their corresponding TOP accidents (P_3 and P_4) since it was judged that more stressful situations and less grace period may be encountered with these accidents.

A4.3.7.7 Shutdown by Fuel/Clad Sweepout

This event means that the reactor becomes subcritical as a result of the fuel/clad eutectic sweepout outside the core region. As indicated in Section A4.3.4, this could add a negative reactivity of -$1.26.

For the nominal TOP accident P1, which is bounded by a reactivity insertion of 35 cents, the sweepout of fuel/clad eutectic alloy outside the core region is almost certain to shut the reactor down at the reactor normal operating temperature. The negative reactivity of -$1.26 is not enough however to make the reactor subcritical at the refueling temperature (temperature and power defect of the metal core concept is ~\$1.7). For more severe TOP accidents, there is less margin to shutdown by the eutectic alloy sweepout. Therefore, the conditional probability of failure to shutdown by this mechanism increases as one moves from P1 through P4 as shown in Figures A4.3-7 through A4.3-10. The same trend is shown for the TOP/LOF accidents in Figures A4.3-11 and A4.3-12. The probabilities shown on the figures for this event have been assigned by judgment based on the uncertainty in the reactivity worth of the eutectic alloy formed, the timing of its formation across the core, and the sweepout pattern.

A4.3.7.8 Shutdown Before Significant Damage

Given that the reactor shutdown system (RSS) did not insert enough control rods for shutdown, that early human intervention to shutdown failed, that the fuel/clad eutectic alloy sweepout failed, the reactor may still operate at an elevated power as long as such power can be removed and no further damage occurs. The burnup reactivity swing for the 20 months of one-cycle operation is ~\$1.0. With another \$1.26 from fuel/clad eutectic sweepout, running the reactor through its cycle will add a negative reactivity of -\$2.26. This more than offsets the worst reactivity addition of \$1.75 of this study. However, a preliminary investigation of the BOP capability to remove 100% to 115% power over a given period decreases exponentially with the mission time as shown in Figure A4.3-13. The figure shows a probability of failure of 10^{-3} for operations beyond about two days and a probability of failure of .44 for operation beyond a year.

Another preliminary investigation of the BOP capability to remove 150% power showed that the BOP will fail to remove the power beyond 5 minutes. The above findings were used to evaluate the risk (as seen by the decision maker confronting the accident) of waiting for burnup to turn the accident around against other options for human intervention to shutdown within 2 days, a year, or within 5 minutes if the power level is at 150% of the nominal power. This evaluation resulted in the subjective probability values shown in Figures A4.3-7 through A4.3-12. Notice that the probability of failing to shutdown before further core damage increases with the accident severity from P1 to P4 and G3 to G4.

The assignment of the core event sequences to core damage categories in Figures A4.3-7 through A4.3-12 is straightforward due to the direct correspondence between the events defined in the trees and the core damage categories defined earlier.

The event trees shown in Figures A4.3-14 through A4.3-26 involve failure of the shutdown heat removal system. The trees reflect the fact that the loss of shutdown heat removal accidents S3 and S5 lead to core damage categories C1S and C2S respectively by definition. The remaining trees are identical in their structures, probabilities, and core damage category assignments to the corresponding ones in Figures A4.3-3 through A4.3-12, except for the inclusion of loss of the SHRS as a part of the accident type and core damage category definition.

REFERENCES - SECTION A4.3

A4.3-1 "Performance of Metallic Fuels and Blankets in Liquid-Metal Fast Breeder Reactors," L. C. Walters, B. R. Seidel, J. H. Kittel, Nuclear Technology, Vol. 65, May 1984, pp. 179.

A4.3-2 "Reactor Safety Study," WASH-1400 (NUREG-75/014), USNRC, October 1975.

A4.3-3 "Analysis of Unprotected Loss-of-Flow and Transient Overpower Accidents in PRISM with Oxide and Metallic Fuel Designs," E. E. Morris, R. A. Wigeland and J. E. Cahalan, ANL, ANL-PRISM-4, January 1986.

A4.3-4 APPENDIX E of this PSID.

A4.3-5 "Preliminary Safety Assessment of the Integral Fast Reactor Concept," J. E. Cahalan et al., ANL, ANL-IFR-3, January 1985.

A4.3-6 "Unprotected Loss-of-Flow Accidents. Performance of Metal Fuel," S. F. Su and R. H. Serj, ANL, ANL-IFR-5, February 1985.

A4.3-7 "Quasi-Static Transient Analysis," R. H. Serj, ANL, ANL-IFR-26, October 1985.

A4.3-8 "Fast Reactor Meltdown Accidents Using Bethe-Tait Analysis," R. A. Meyer and B. Woolfe, Advances in Nuclear Science and Technology, Vol. 4, 1968, p. 197.

A4.3-9 "An Analysis of the Unprorated Transient Overpower Accident in the FTR," A. E. Walter et al., HEDL-TME-75-50, HEDL, June 1975.

A4.3-10 "Risk Analysis of Application to FFTF-Second Cycle," K. G. Feller and D. E. Hurd, GE, GEFR-00029, January 1977.

A4.3-11 "Addendum 1 to ANL-PRISM-5, 1986 (Risk Evaluation of Severe Accident Progression in PRISM): Revised Inherent Shutdown Probability Estimates for the PRISM Metal Core LOF and TOP Scenarios and Their Impact on Risk," C. J. Mueller, ANL, personal correspondence, August 1986.

TABLE A4.3-1

FRACTIONS OF FISSION PRODUCTS RELEASED TO THE FISSION GAS PLENUM

Fission Product	Chemical Group	Release Fraction From Fuel to Gas Plenum
Xe, Kr	Noble Gases	.72
I, Br	Halogens	.2
Cs, Rb	Alkali Metals	.5
Sr, Ba	Alkaline Earths	10^{-3}
Te	Tellurium	10^{-2}
Other	Others	Negligible

TABLE A4.3-2

FISSION PRODUCT ESCAPE FRACTIONS AND RELEASE FRACTIONS ON CLAD FAILURE

Fission Product	Chemical Group	Escape Fraction	Release Fraction On Clad Failure
Xe, Kr	Noble gases	1	.72
I, Br	Halogens	1/3	.07
Cs, Rb	Alkali Metals	1/3	.17
Sr, Ba	Alkaline Earths	10^{-4}	~0
Te	Tellurium	10^{-3}	~0
Others		0	0

TABLE A4.3-3

RELEASE FRACTION FROM EUTECTIC ALLOY

Fission Product	Release Fractions (of Fission Products Carried By the Eutectic Fuel to Surrounding Medium)
Xe, Kr	1.0
I, Br	1.0
Cs, Rb	1.0
Others	None Released From Eutectic Alloy

TABLE A4.3-4

MELTDOWN RELEASE FRACTIONS

Fission Products	Release Fraction (of Fission Products Carried By the Molten Portion of the Fuel)
Xe, Kr	1.0
I, Br	1.0
Cs, Rb	1.0
Te	0.85
Sr, Ba	0.01
Noble Metals	0.003
All Others	0.0003

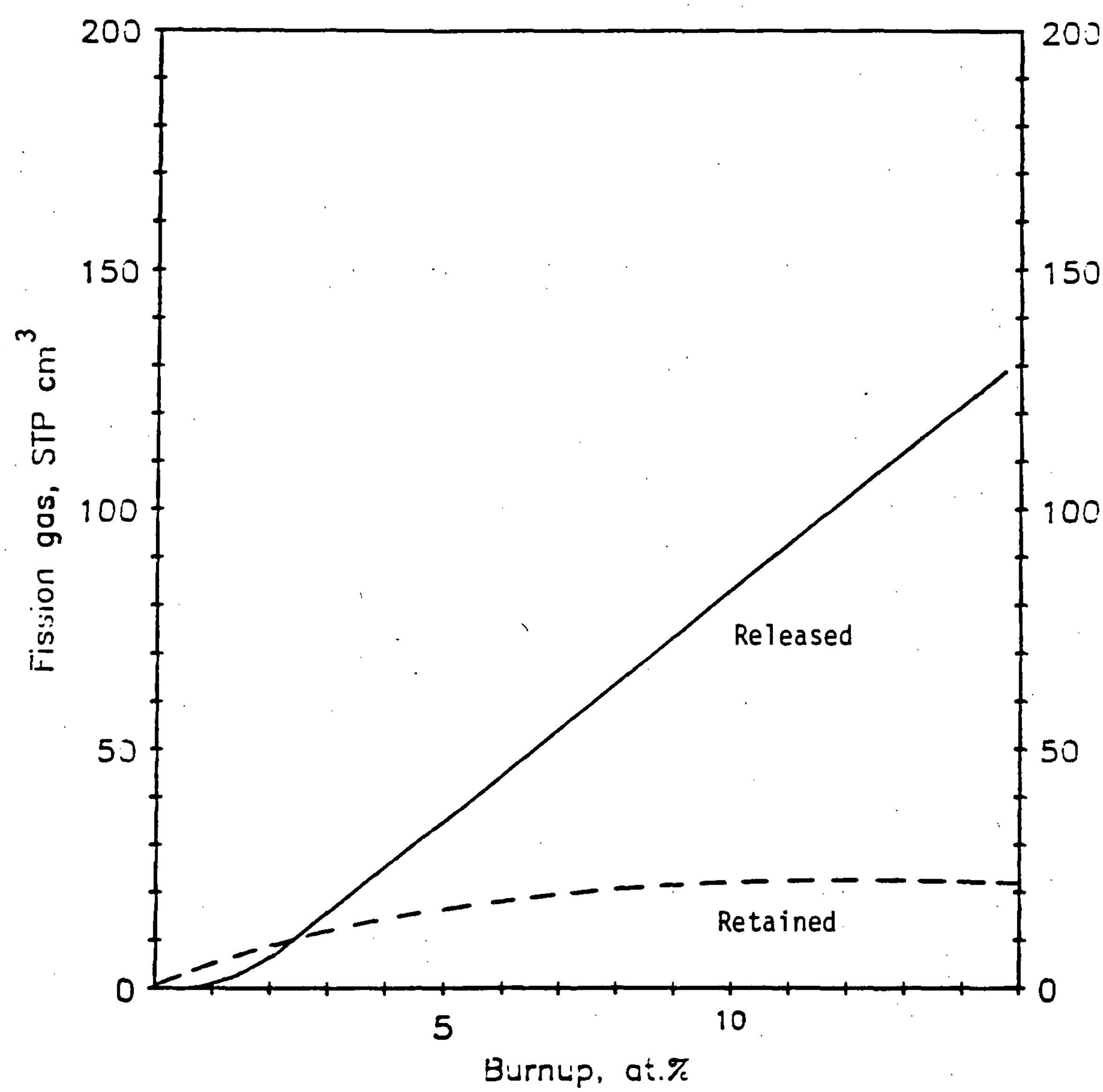


Figure A4.3-1 - RELEASED AND RETAINED FISSION GAS IN MARK-II FUEL ELEMENTS

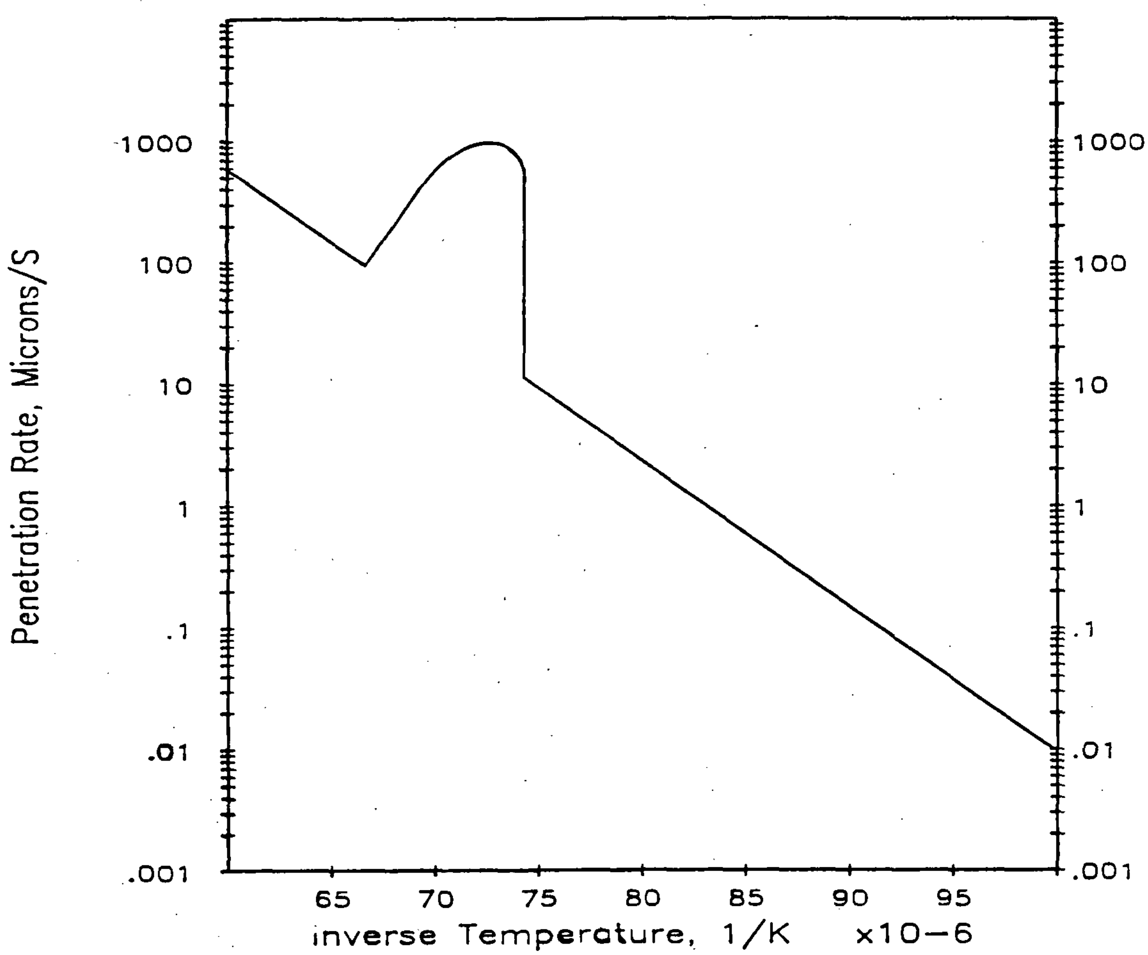


Figure A4.3-2 RATE OF EUTECTIC PENETRATION OF CLADDING

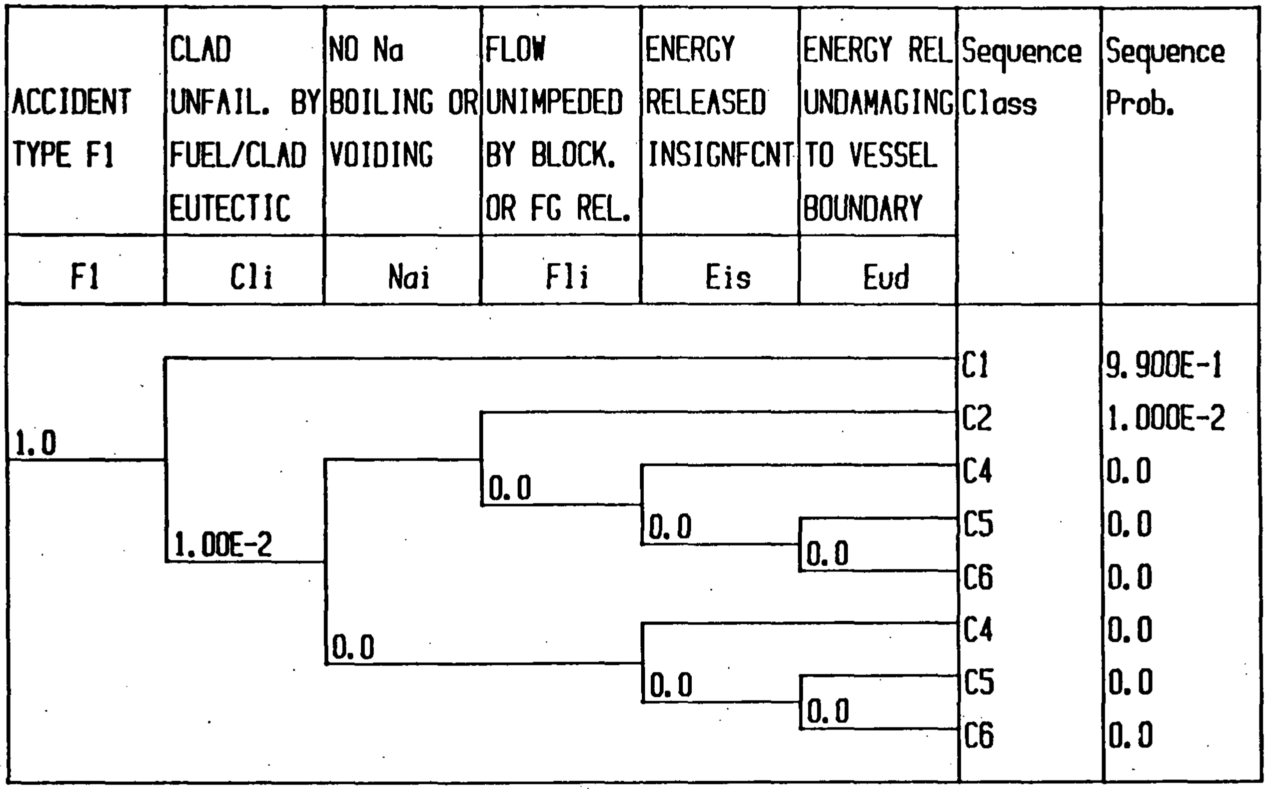


Figure A4.3-3 - PRISM: Core Response Event Tree - LOF (F1)

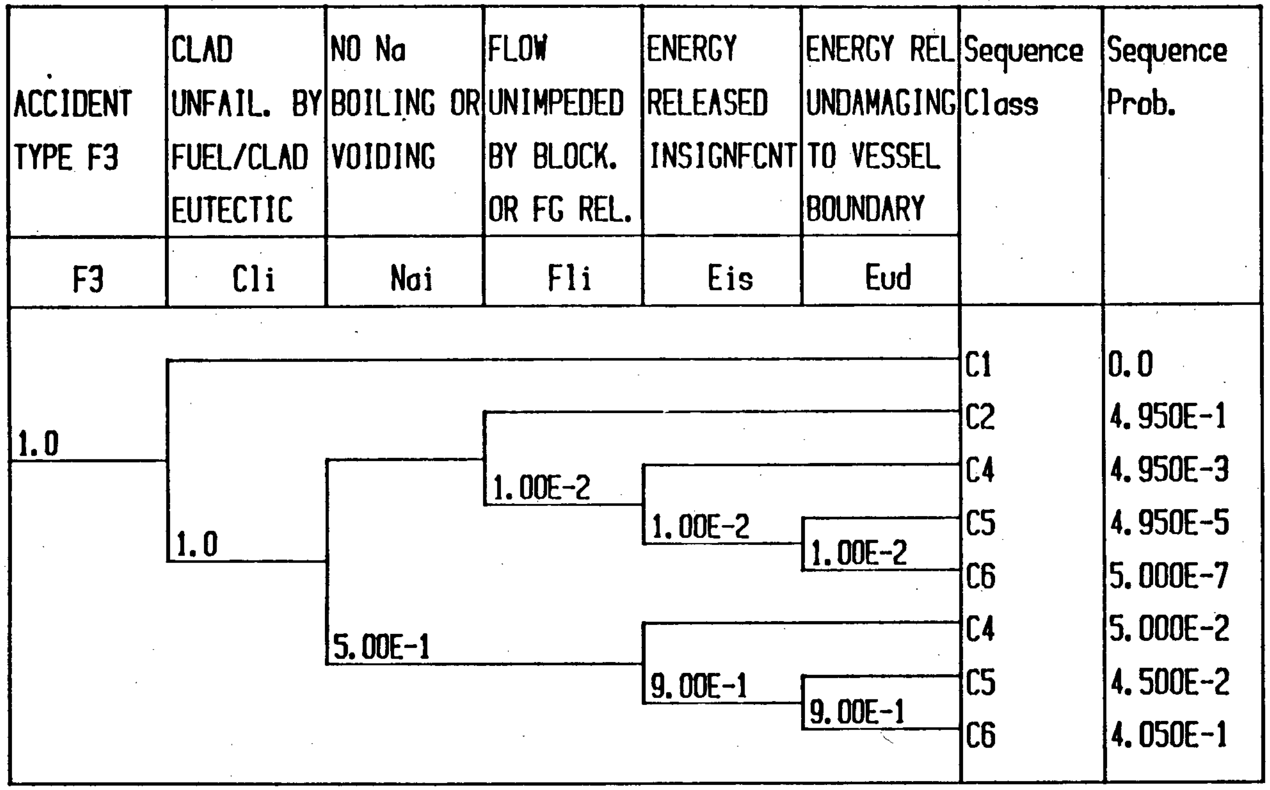


Figure A4.3-4 - PRISM: Core Response Event Tree - LOF (F3)

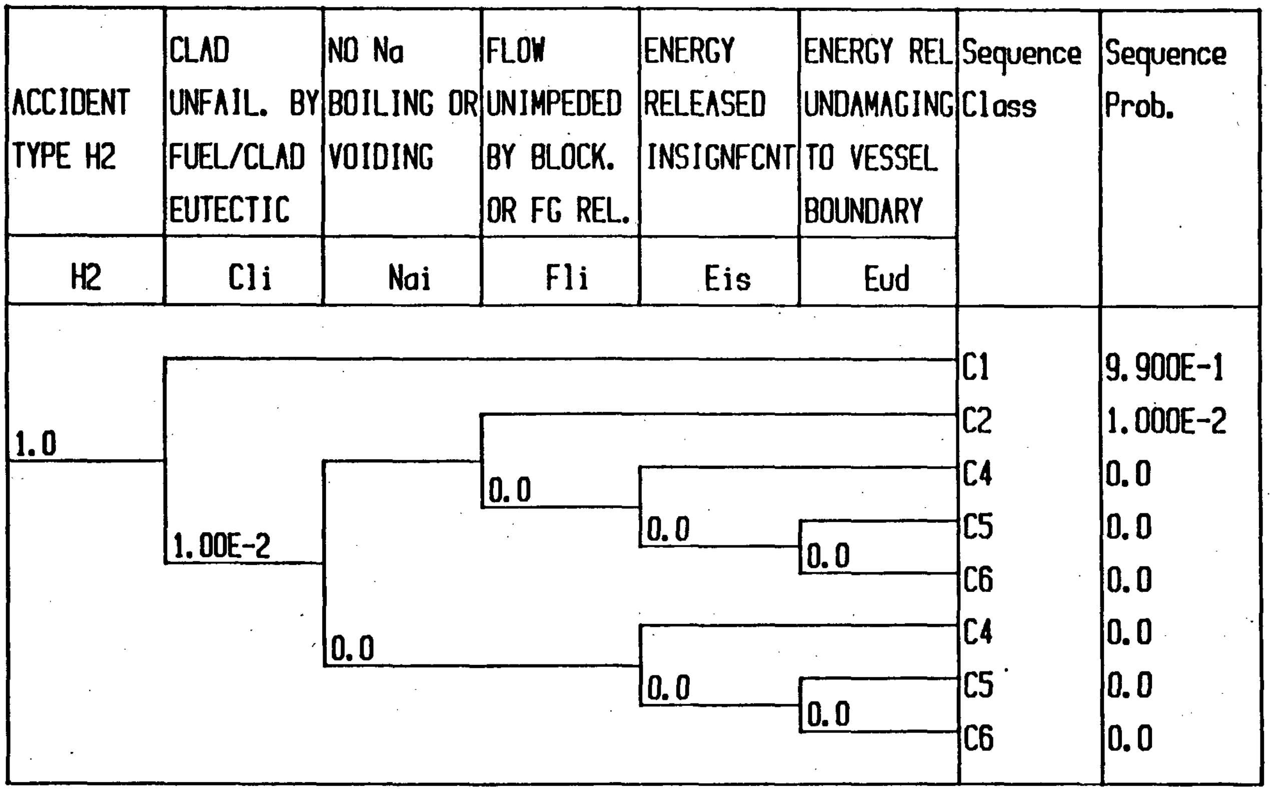


Figure A4.3-5 - PRISM: Core Response Event Tree - ULOHS (H2)

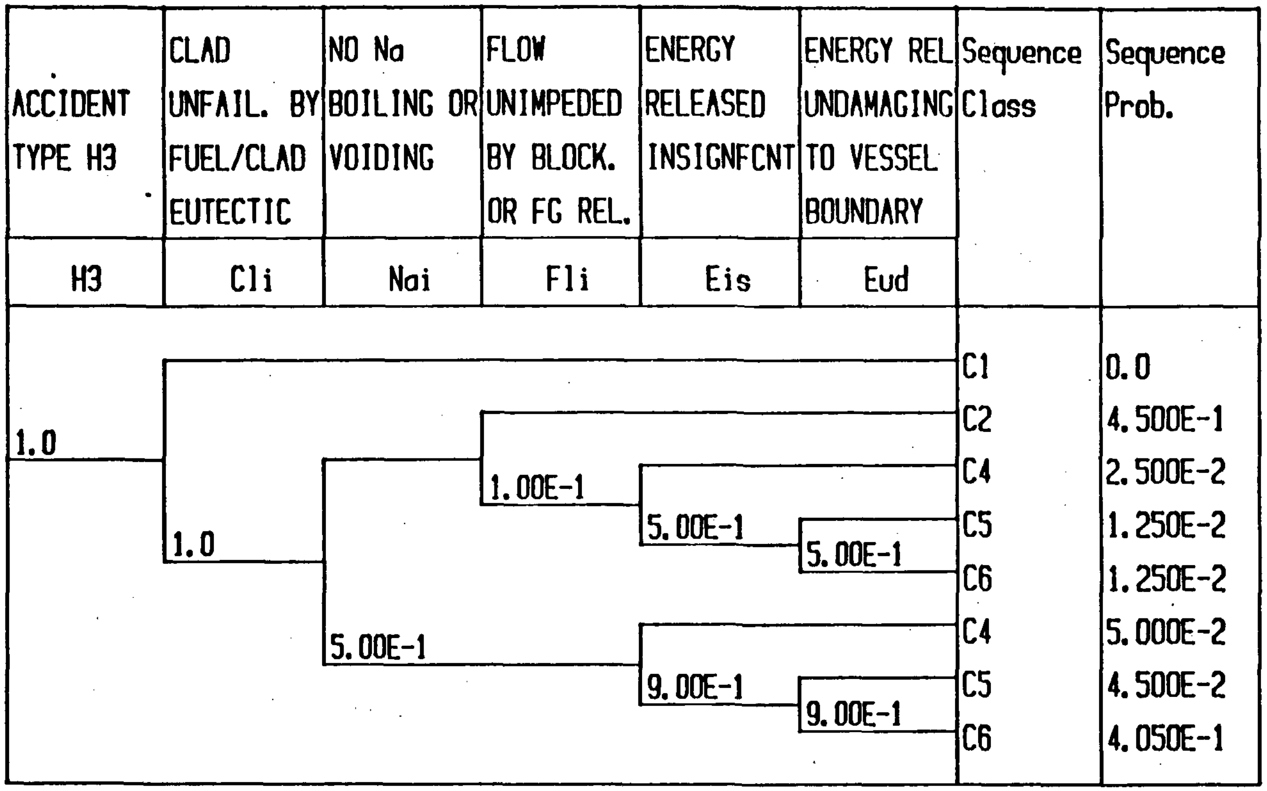


Figure A4.3-6 - PRISM: Core Response Event Tree - ULOHS (H3)

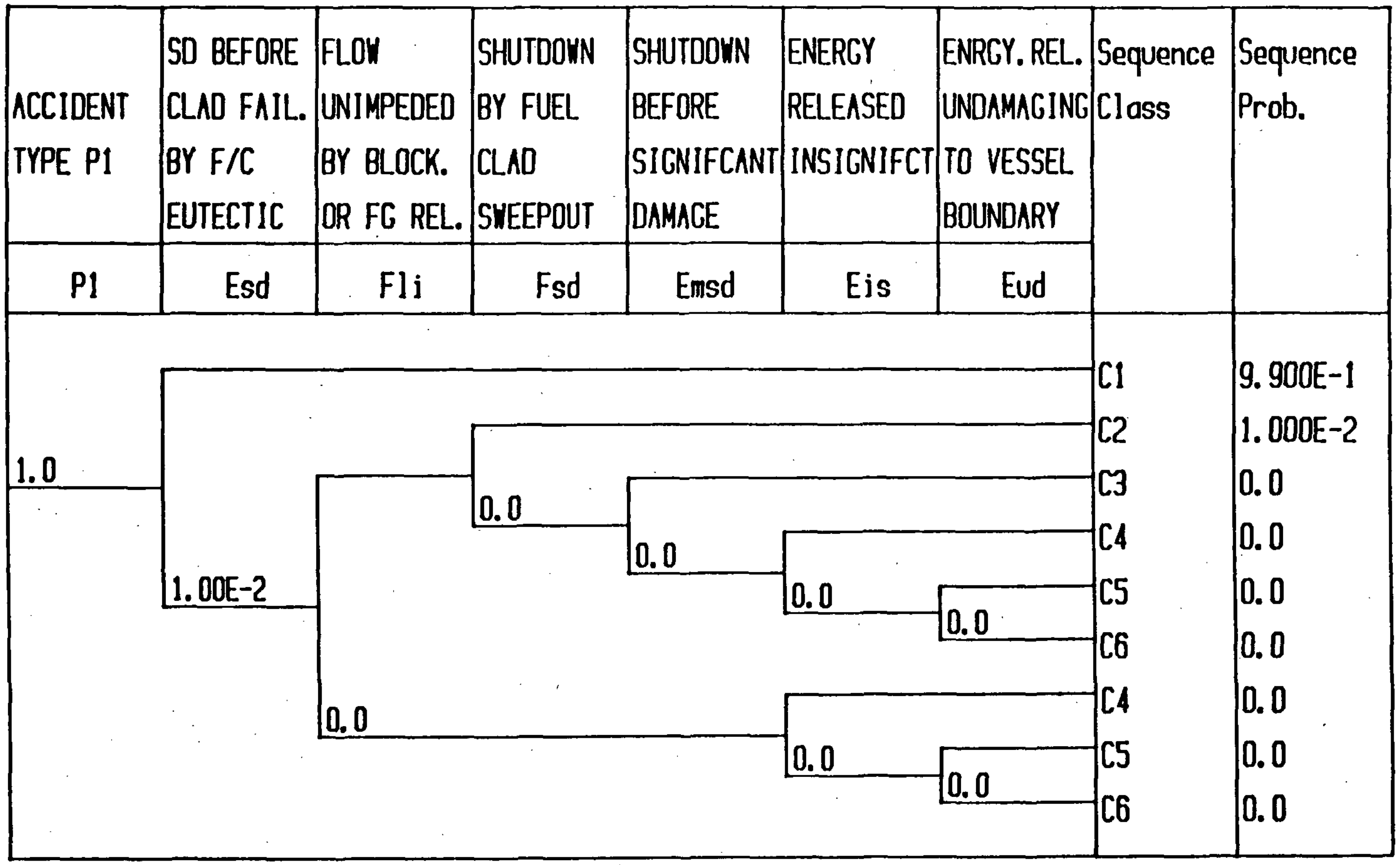


Figure A4.3-7 - PRISM: Core Response Event Tree - TOP (P1)

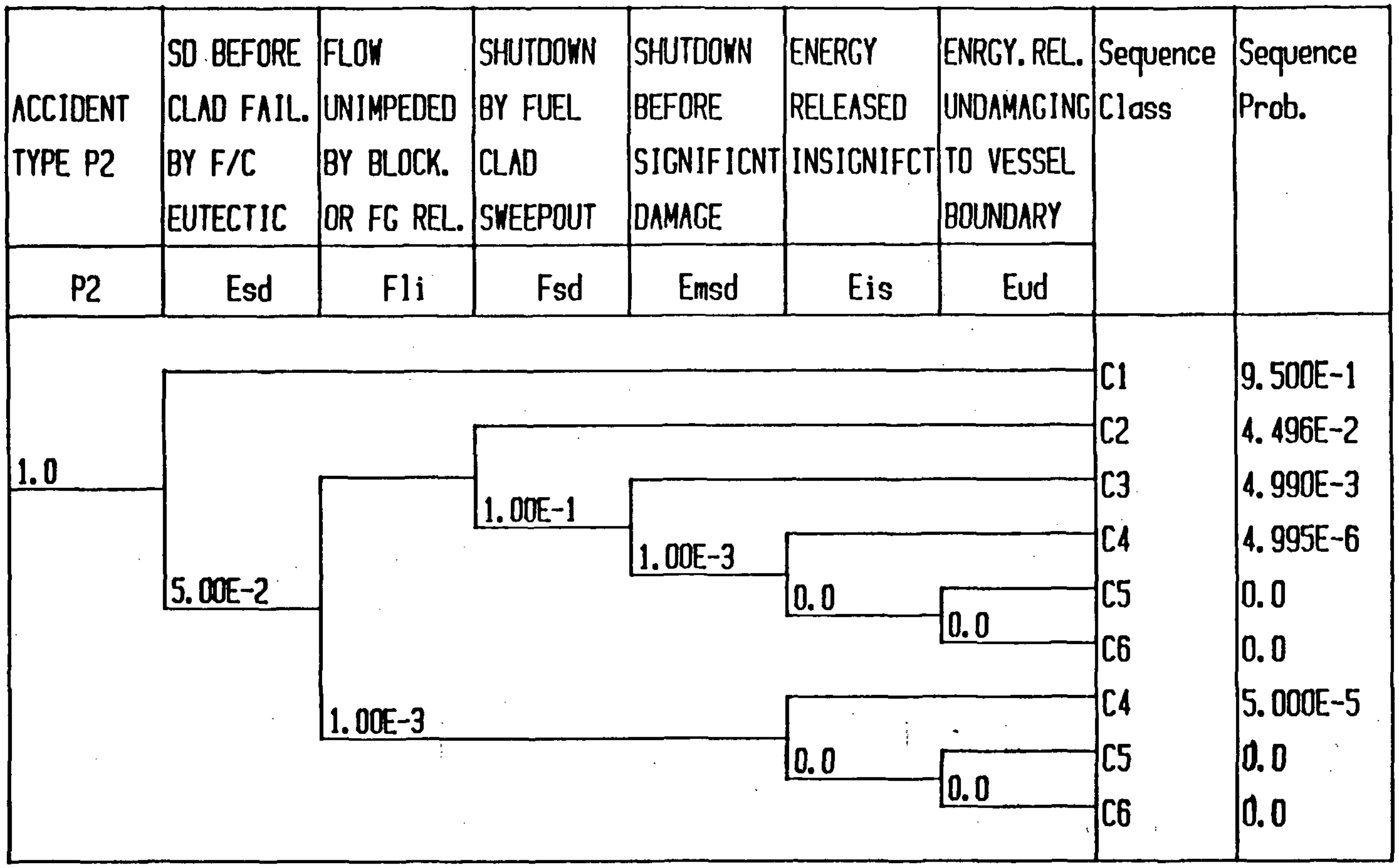


Figure A4.3-8 - PRISM: Core Response Event Tree - TOP (P2)

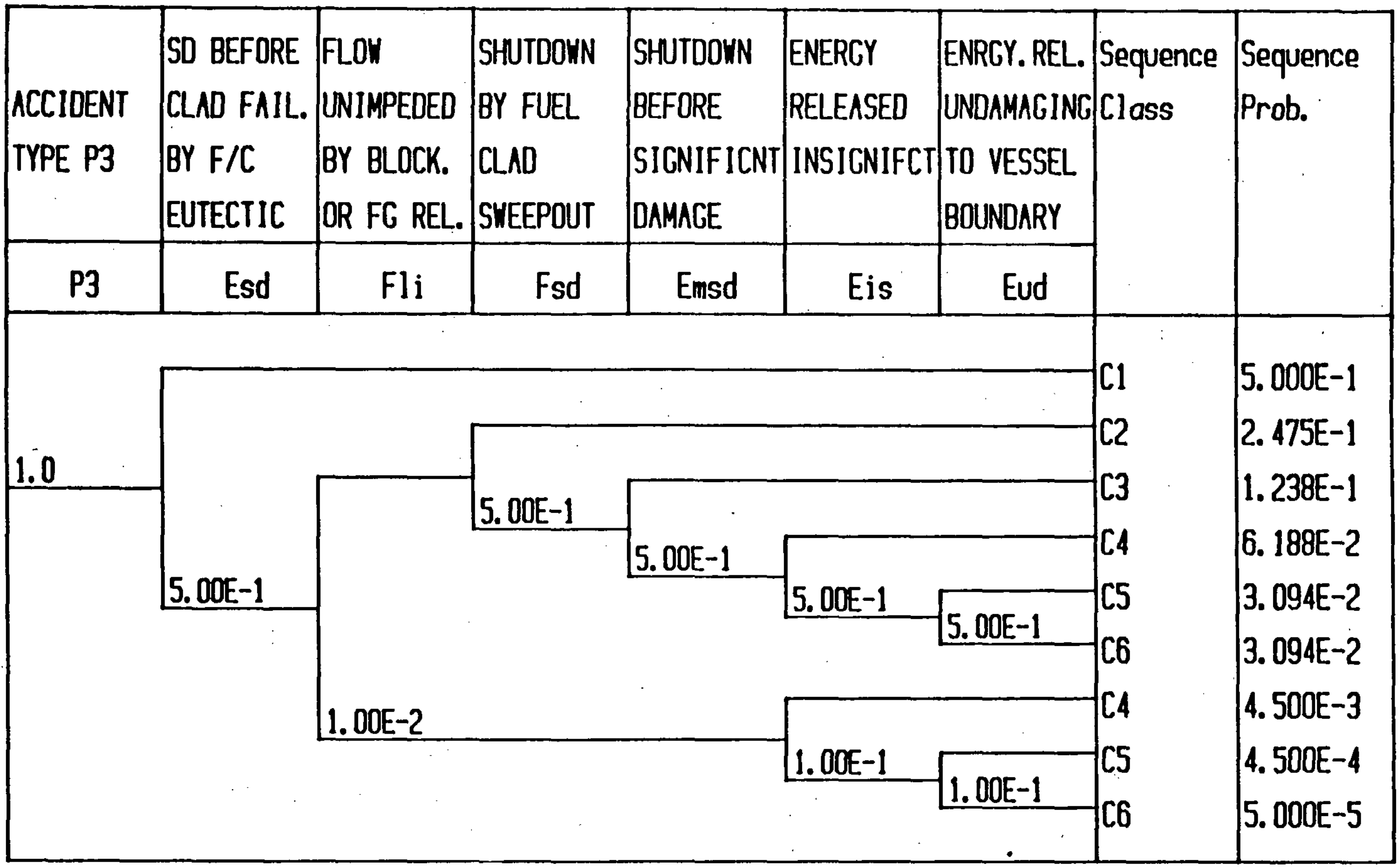


Figure A4.3-9 - PRISM: Core Response Event Tree - TOP (P3)

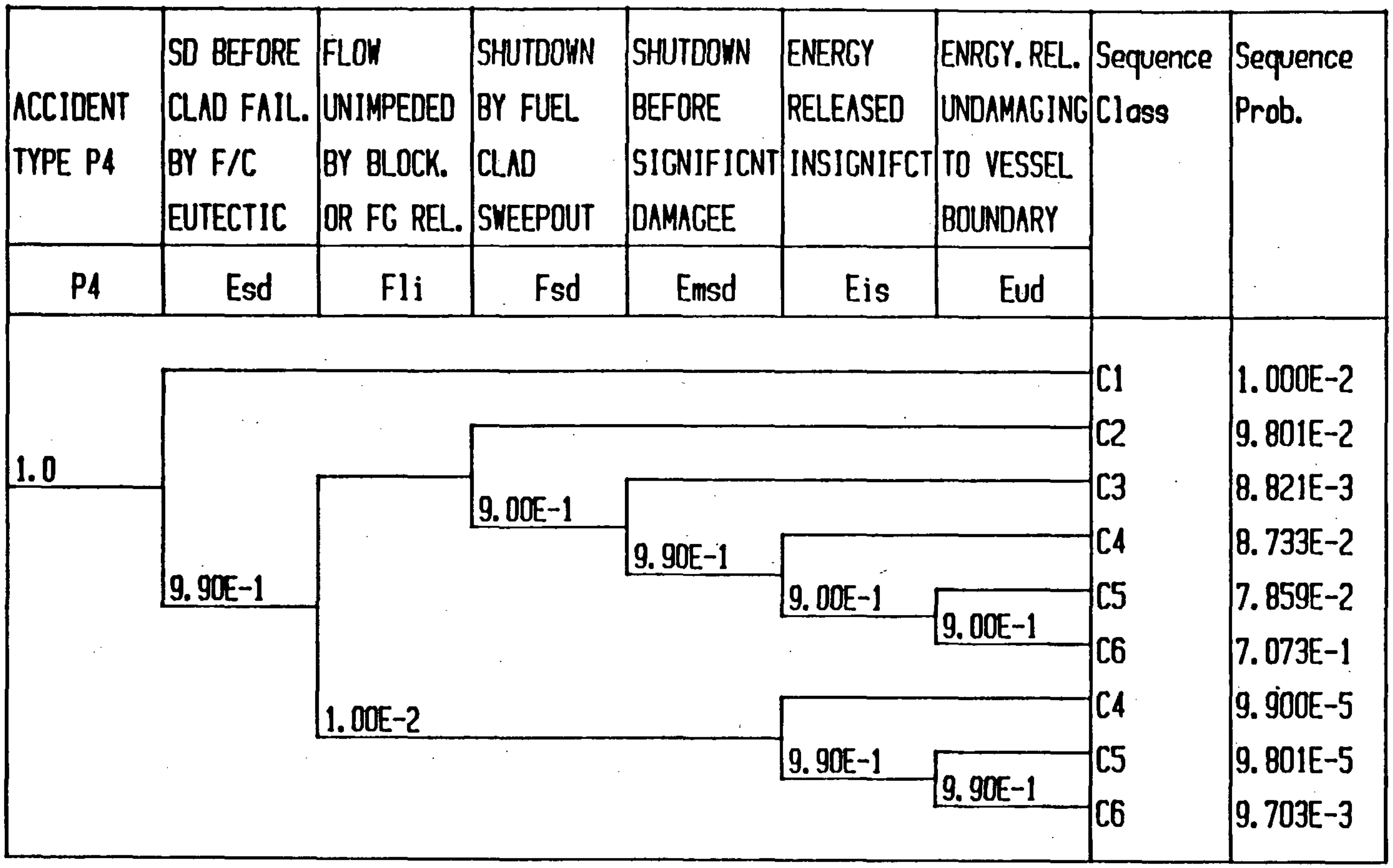


Figure A4.3-10 - PRISM: Core Response Event Tree - TOP (P4)

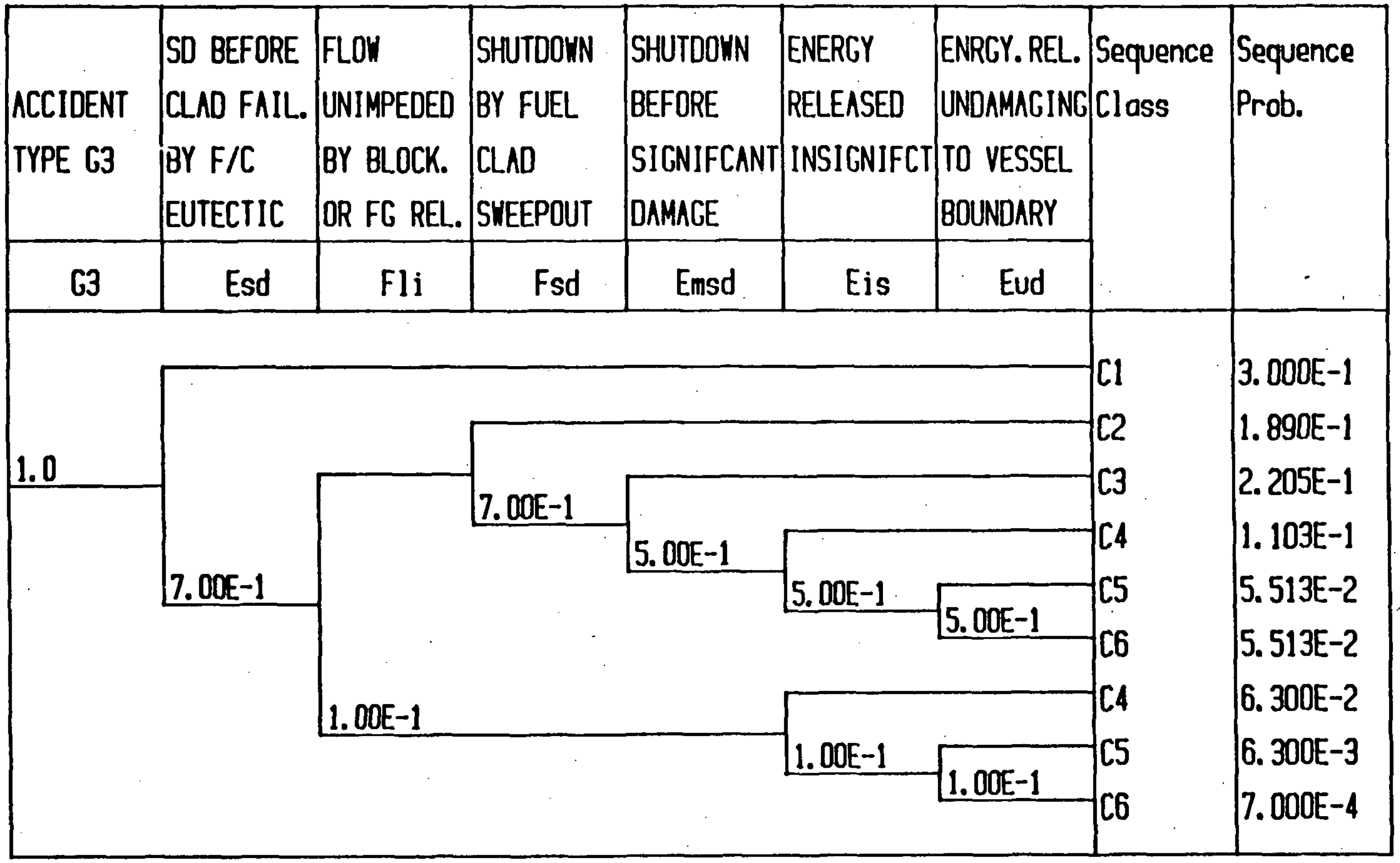


Figure A4.3-11 - PRISM: Core Response Event Tree - TOPLOF (G3)

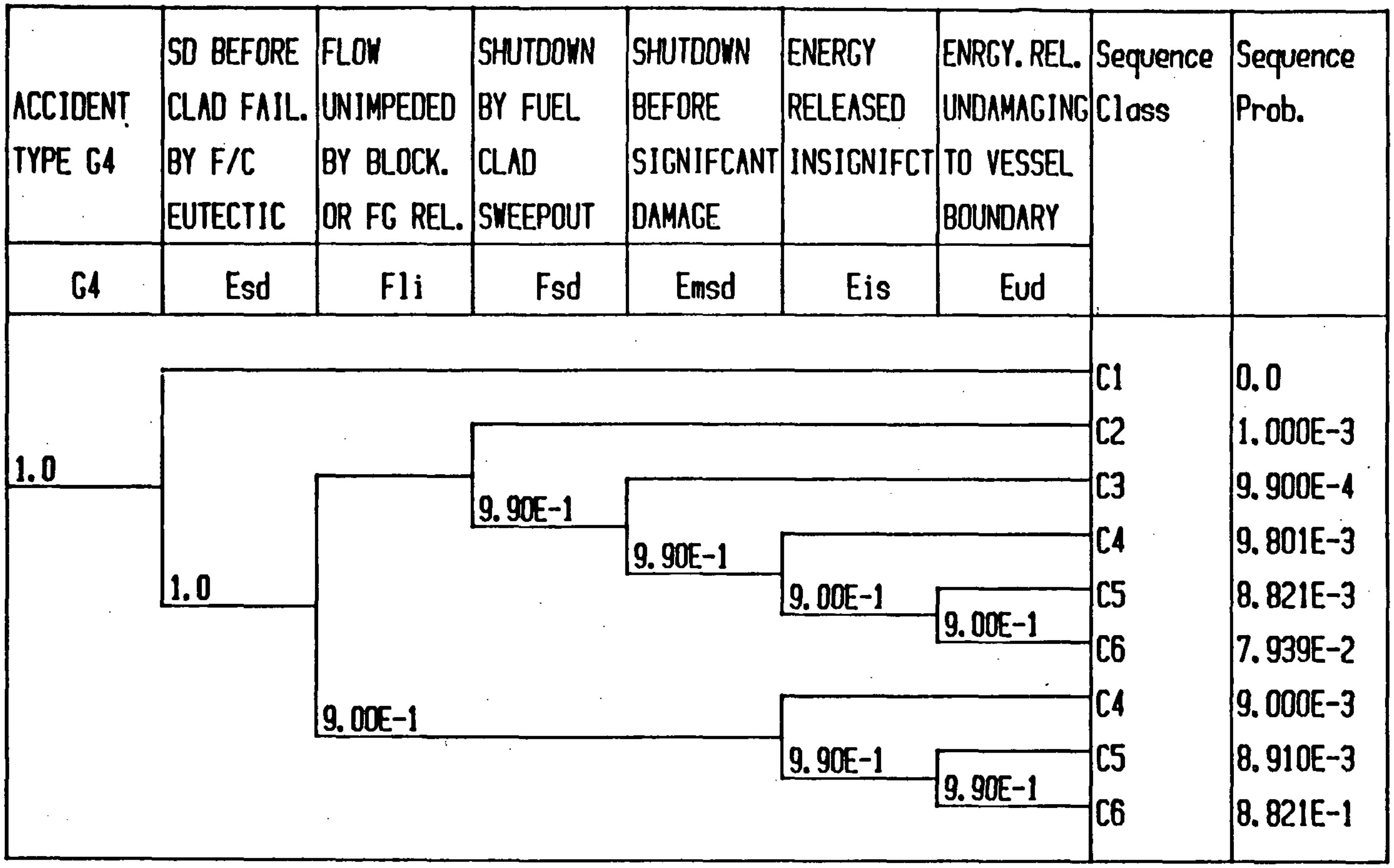


Figure A4.3-12 - PRISM: Core Response Event Tree - TOPLOF (G4)

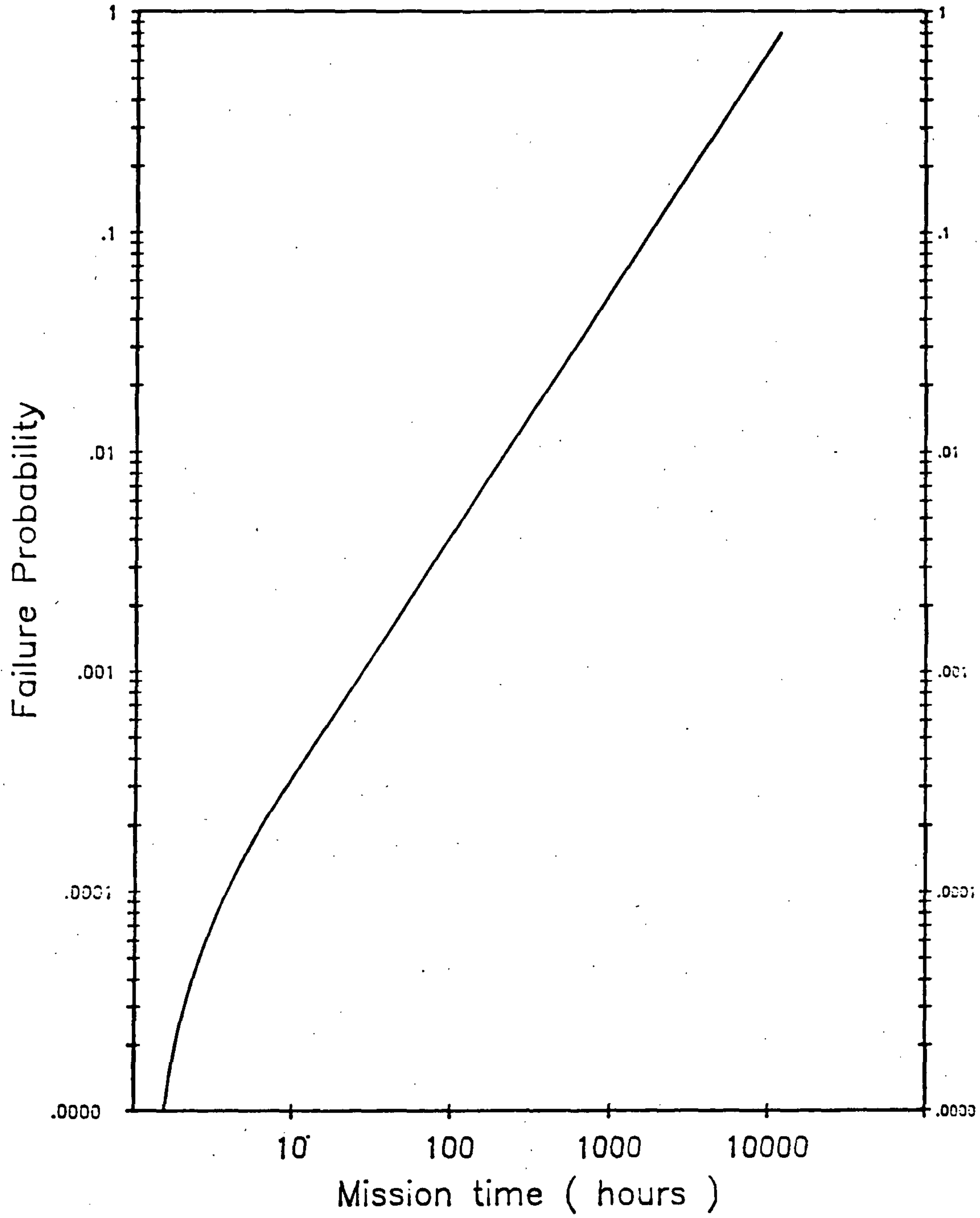


Figure A4.3-13 BOP FAILURE PROBABILITY FOR 100% TO 115% MODULE POWER OUTPUT

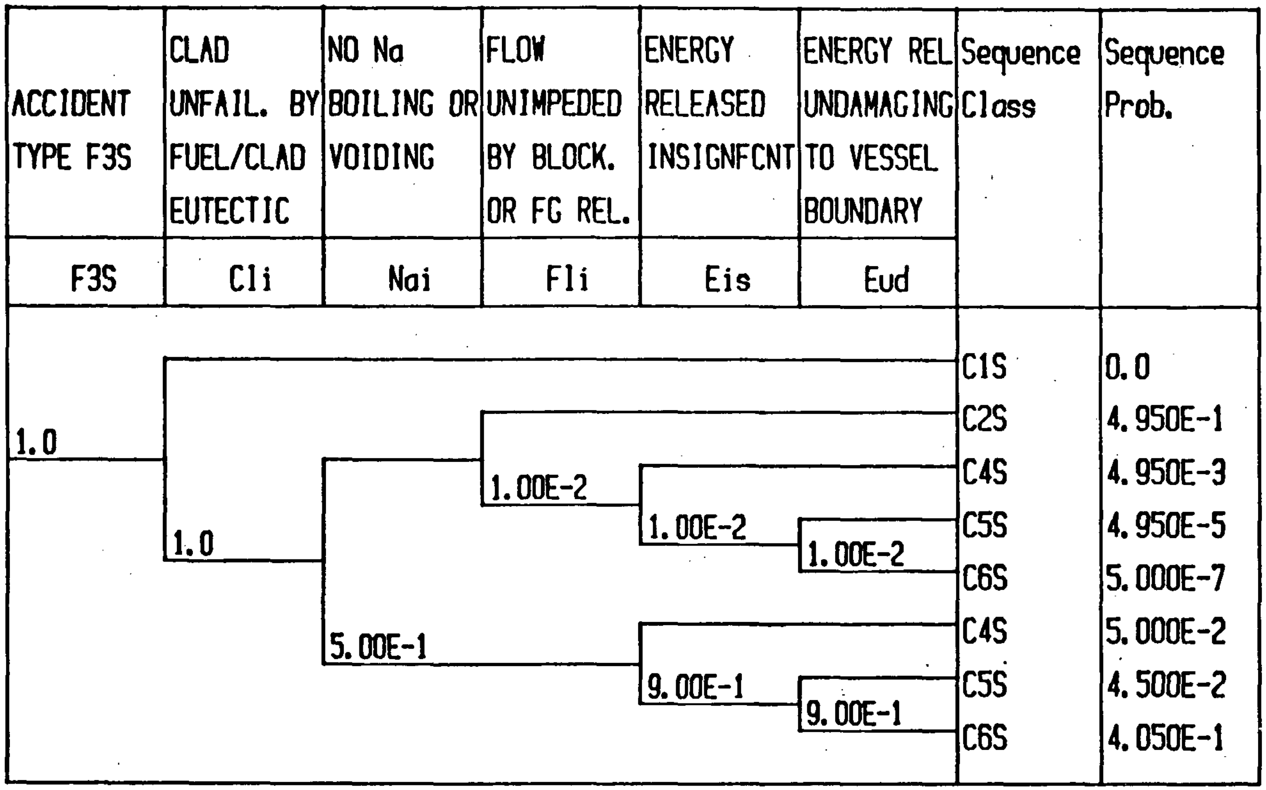


Figure A4.3-14 - PRISM: Core Response Event Tree - LOF (F3S)

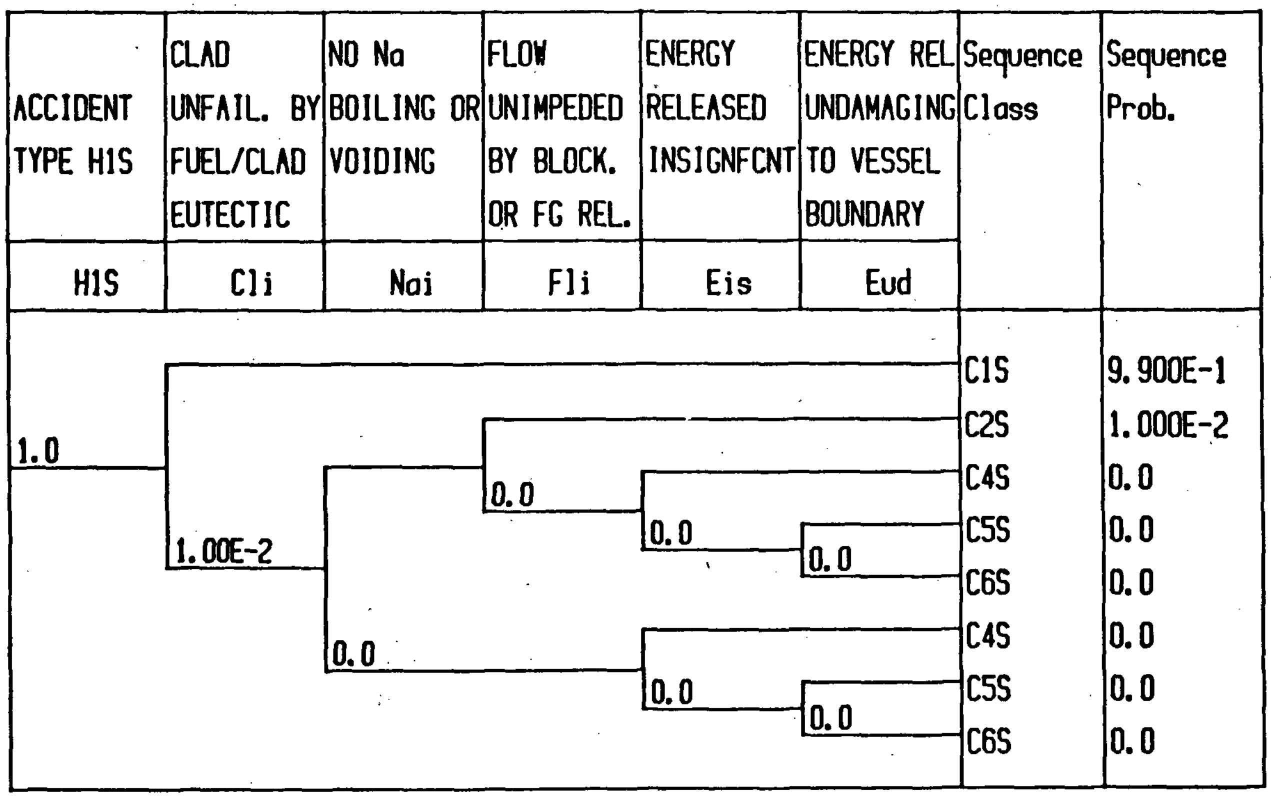


Figure A4.3-15 - PRISM: Core Response Event Tree - ULOHS (H1S)

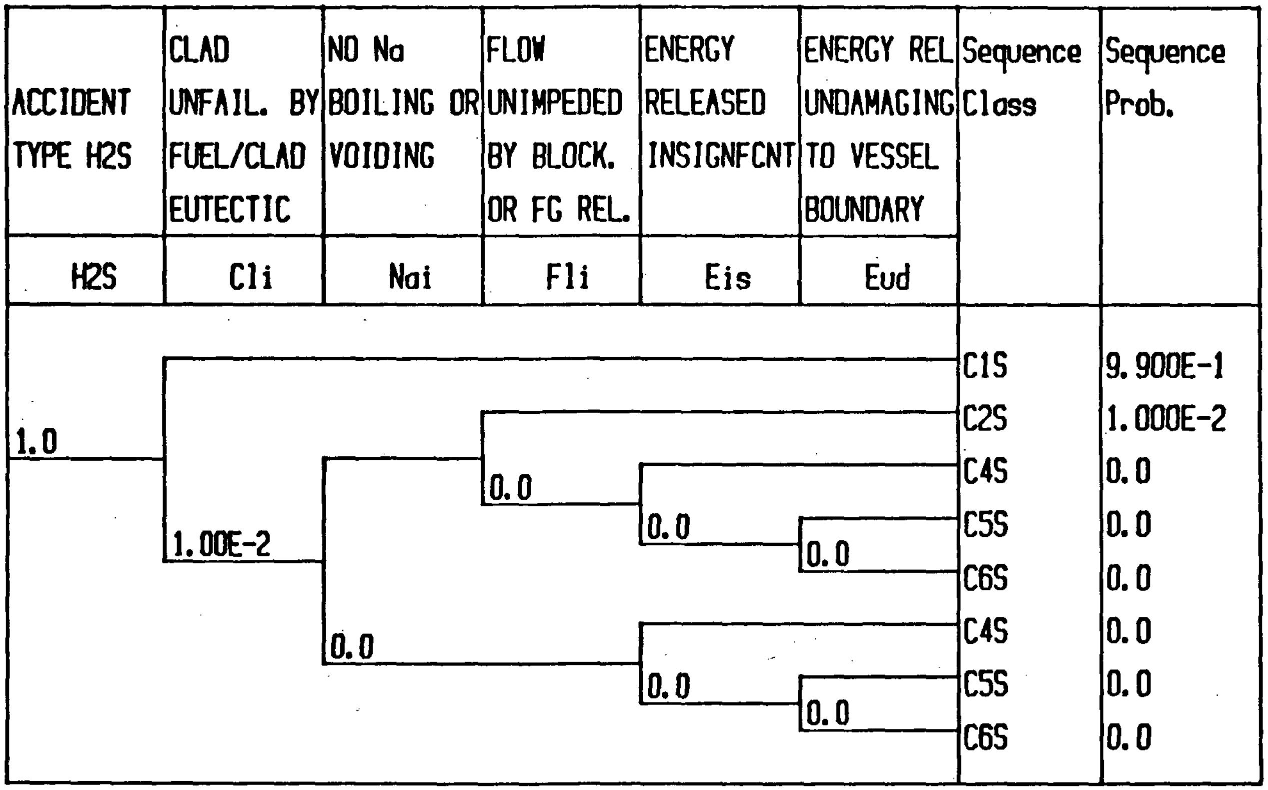


Figure A4.3-16 - PRISM: Core Response Event Tree - ULOHS (H2S)

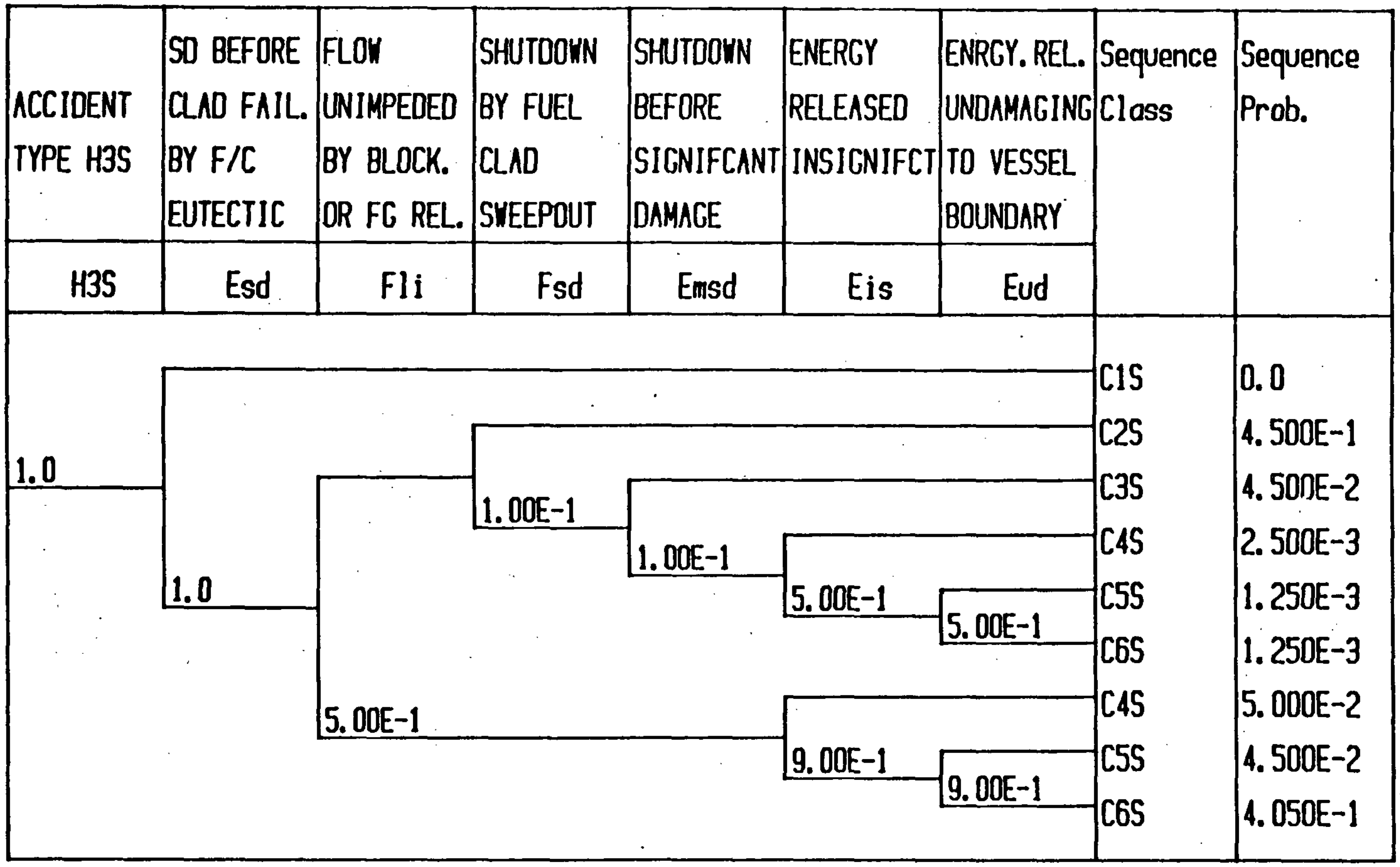


Figure A4.3-17 - PRISM: Core Response Event Tree - ULOHS (H3S)

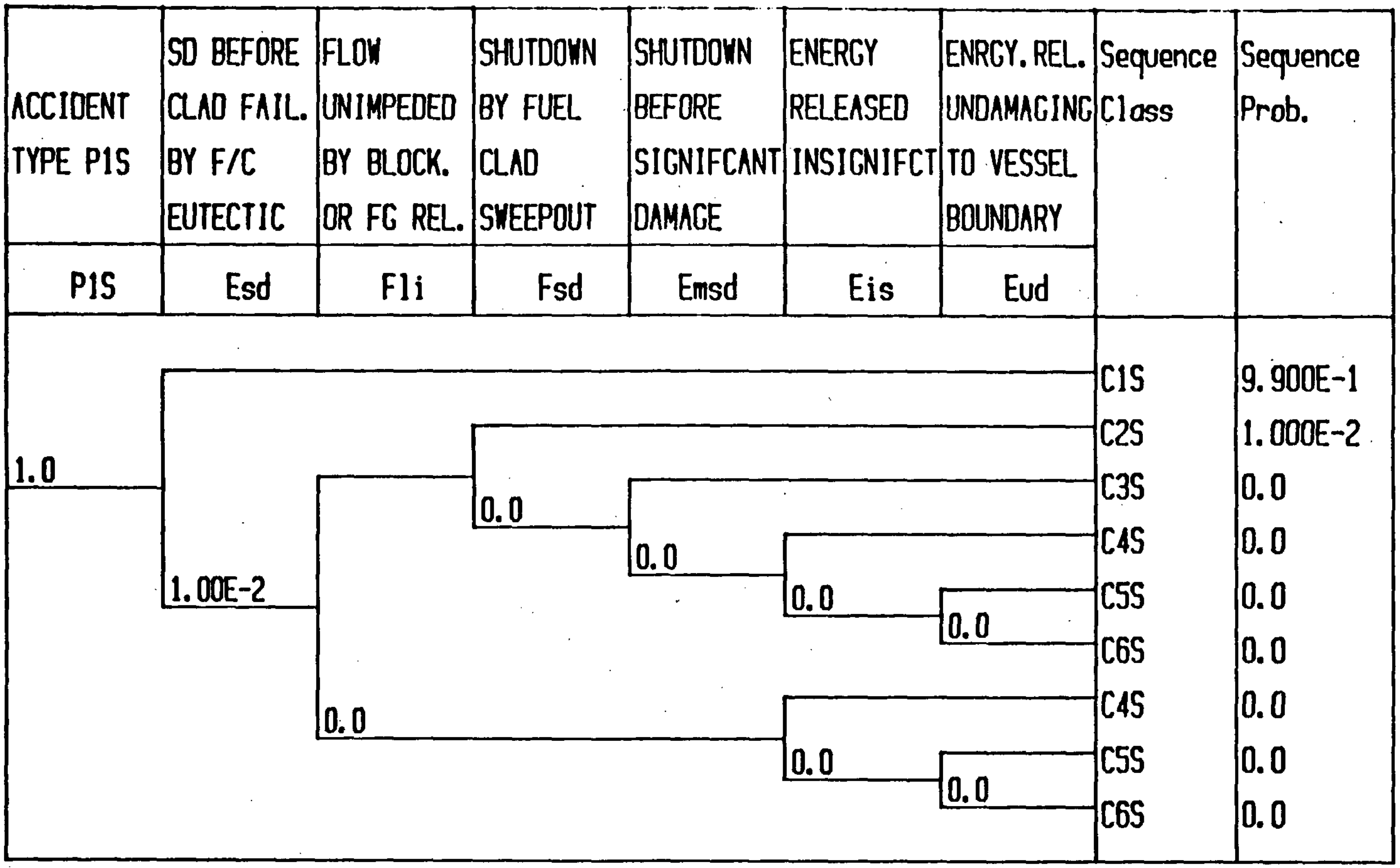


Figure A4.3-18 - PRISM: Core Response Event Tree - TOP (P1S)

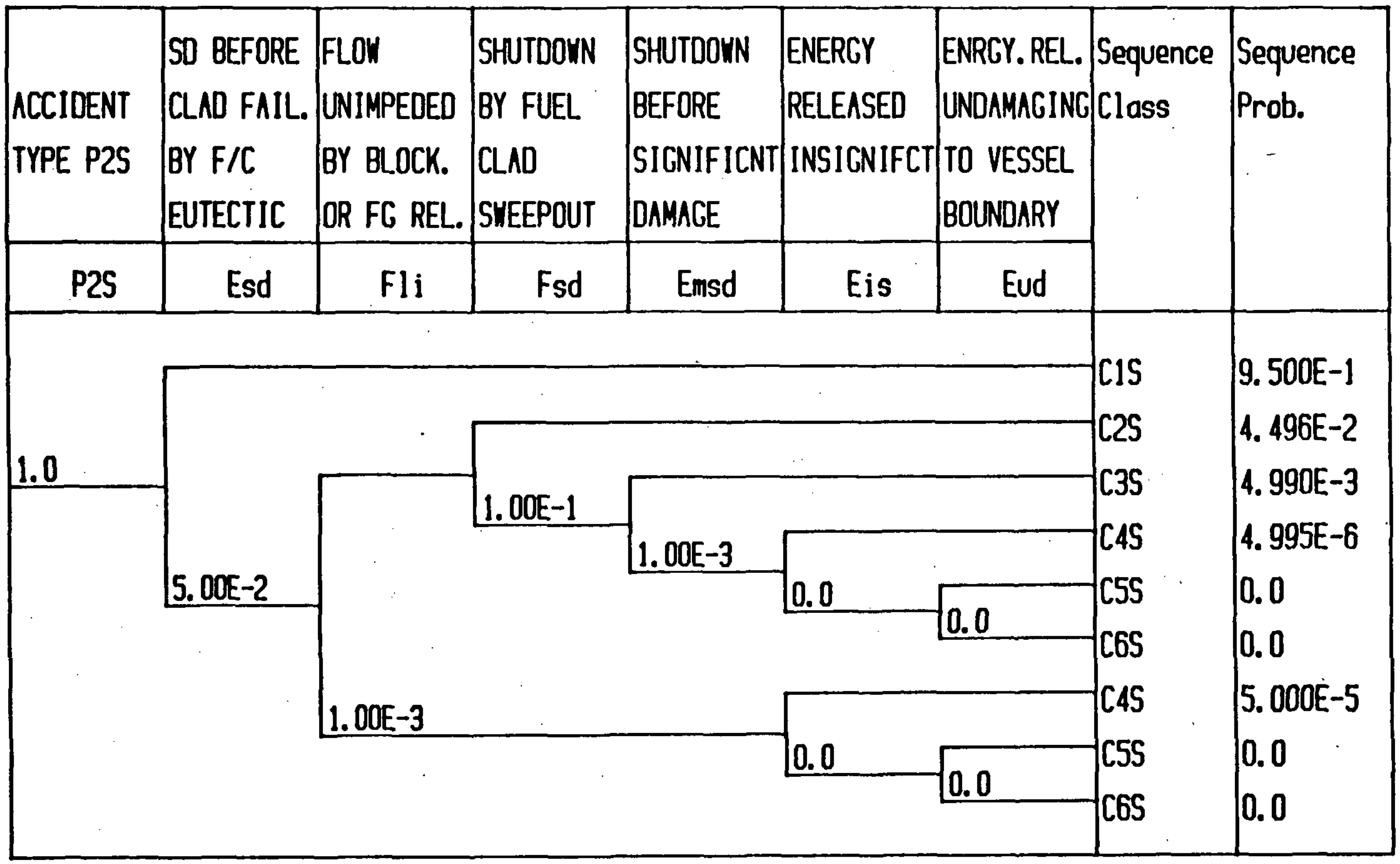


Figure A4.3-19 - PRISM: Core Response Event Tree - TOP (P2S)

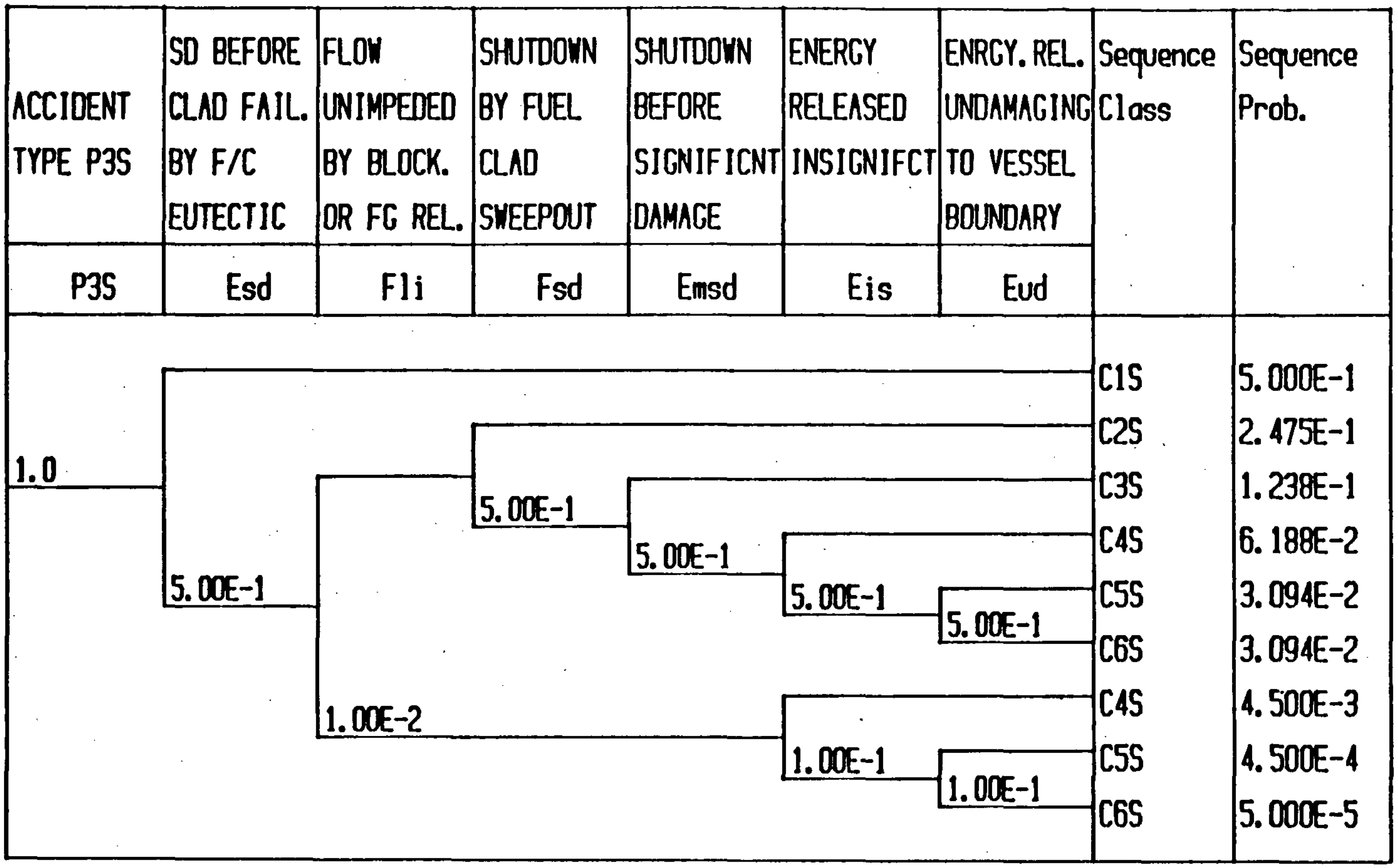


Figure A4.3-20 - PRISM: Core Response Event Tree - TOP (P3S)

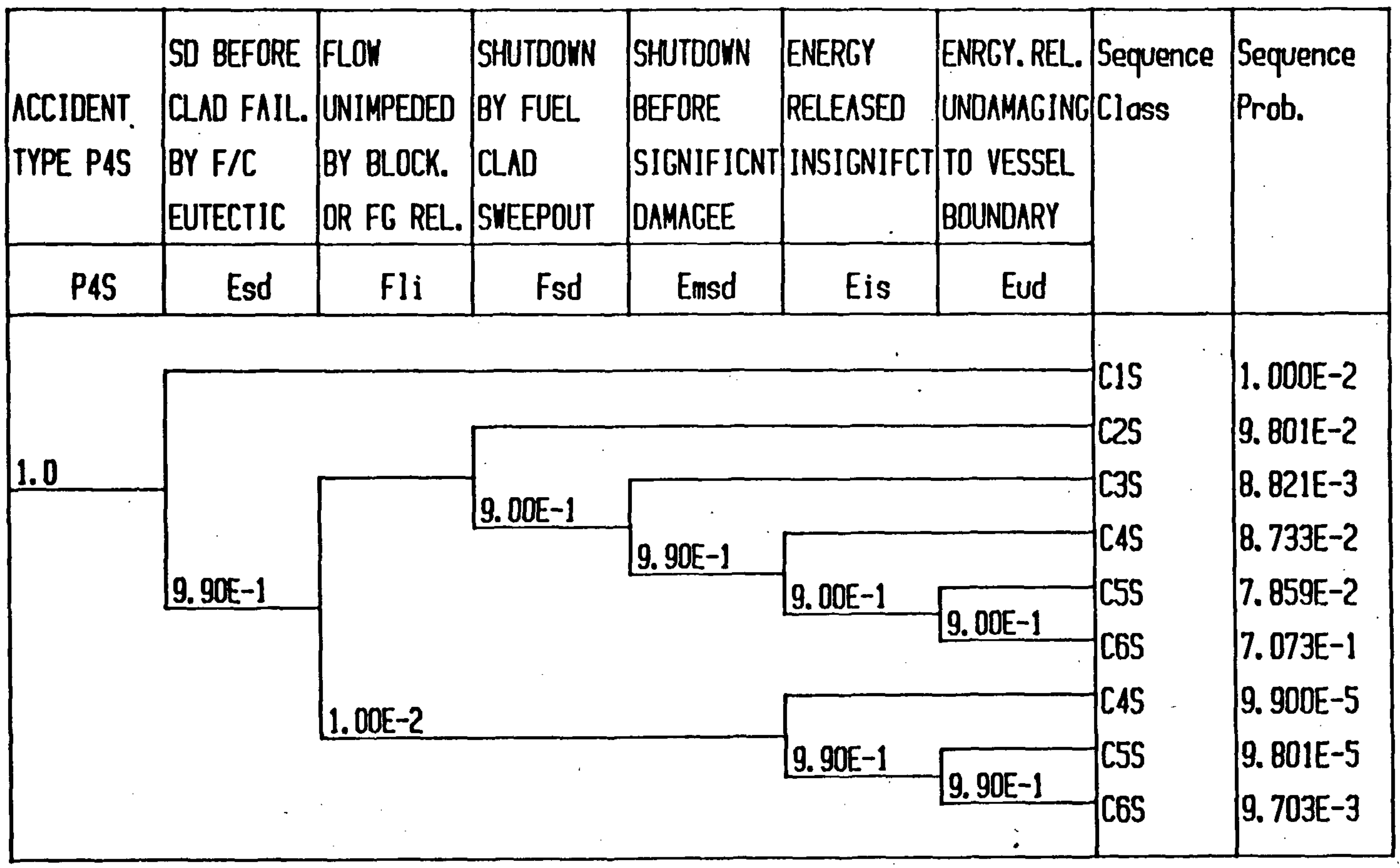


Figure A4.3-21 - PRISM: Core Response Event Tree - TOP (P4S)

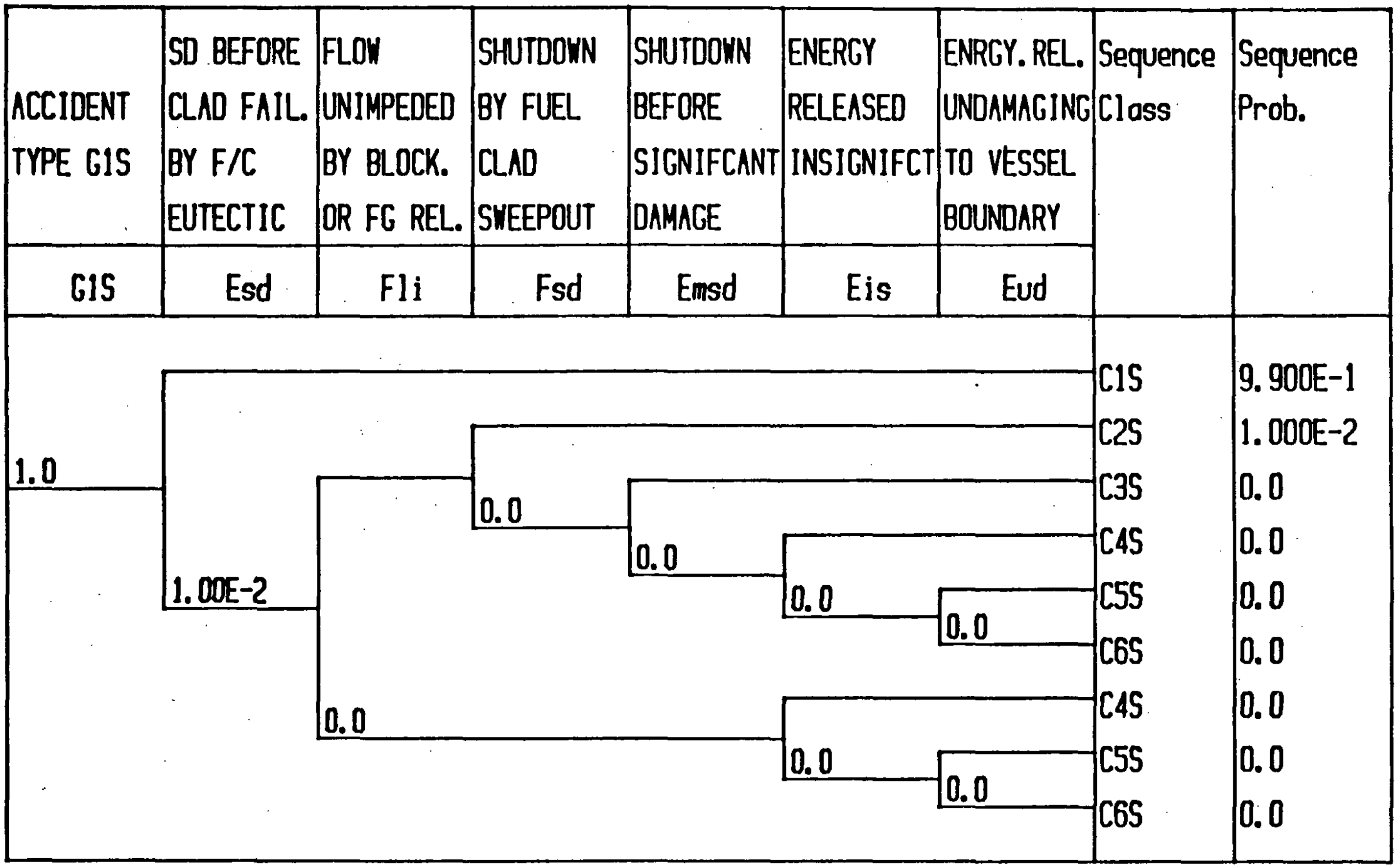


Figure A4.3-22 - PRISM: Core Response Event Tree - TOPLOF(G1S)

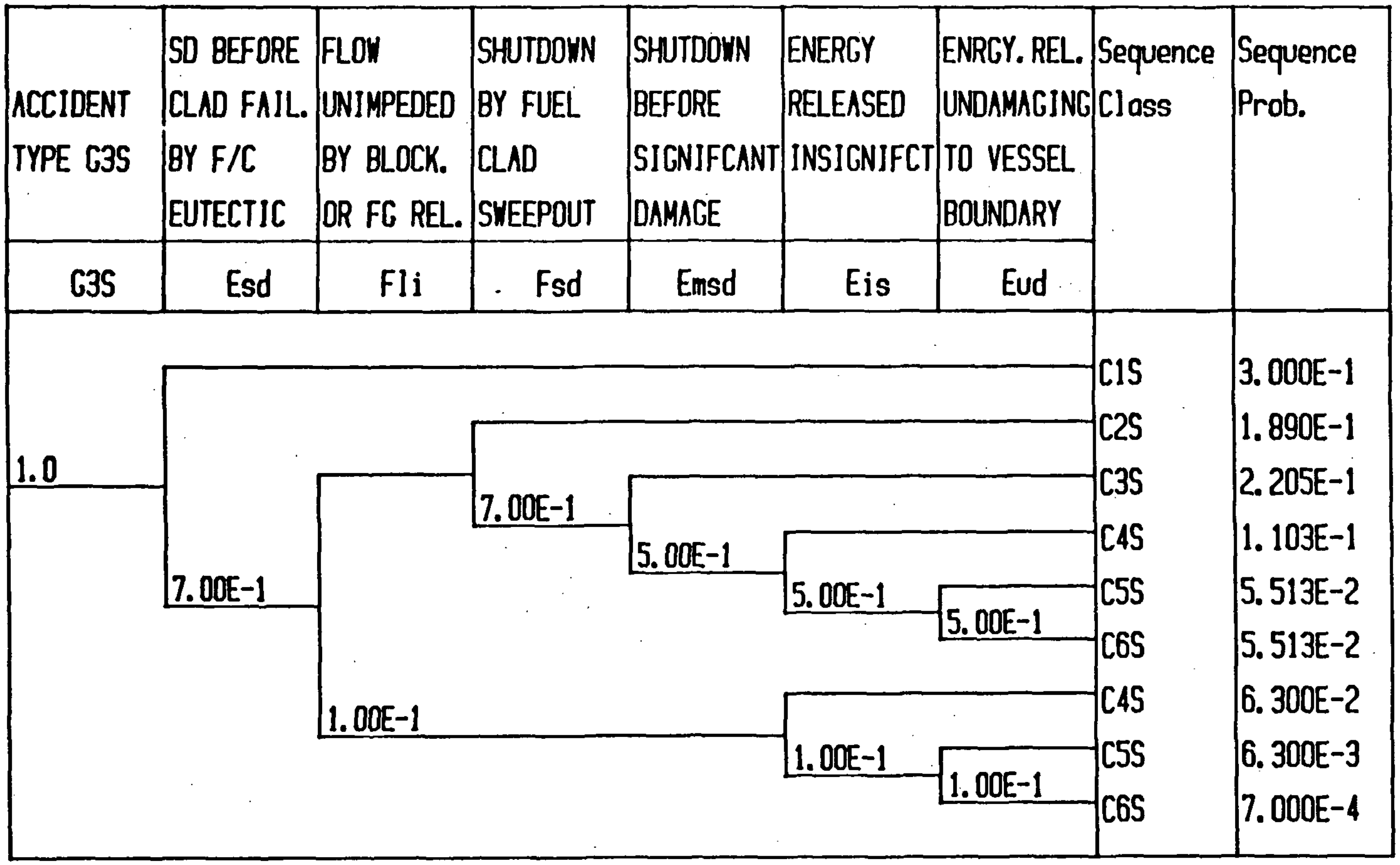


Figure A4.3-23 - PRISM: Core Response Event Tree - TOPLOF(G3S)

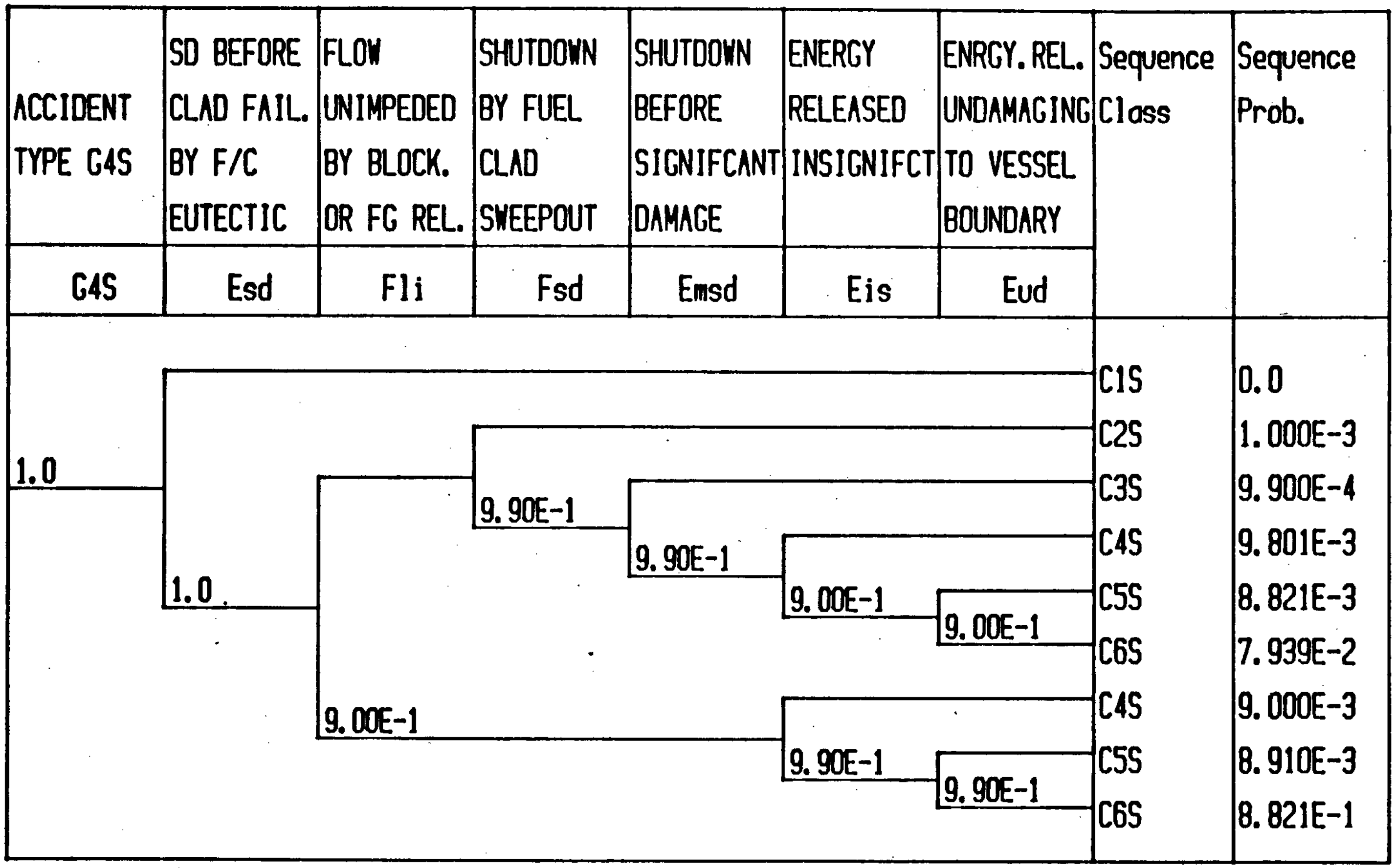


Figure A4.3-24 - PRISM: Core Response Event Tree - TOPLOF(G4S)

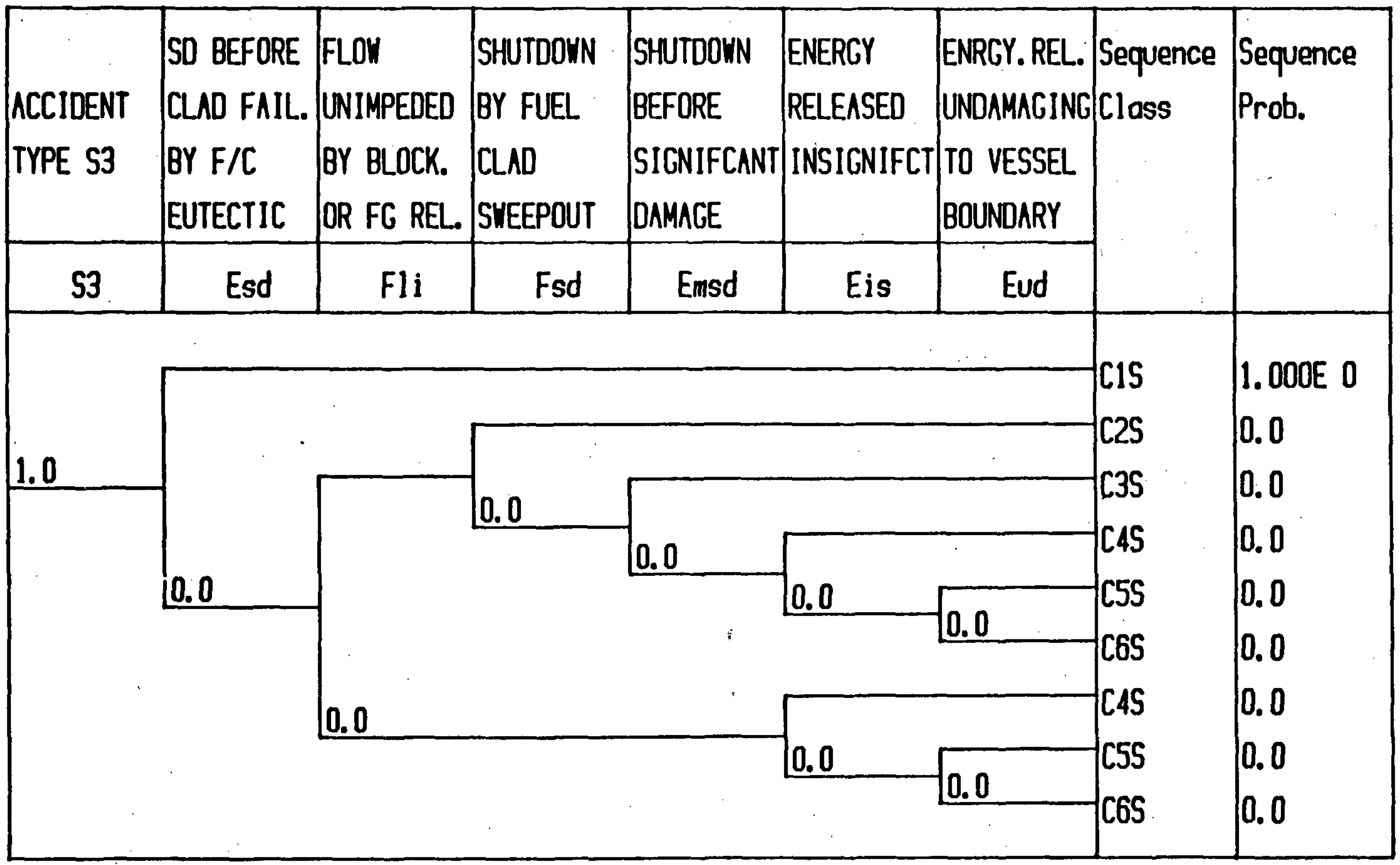


Figure A4.3-25 - PRISM: Core Response Event Tree - LOSHR (S3)

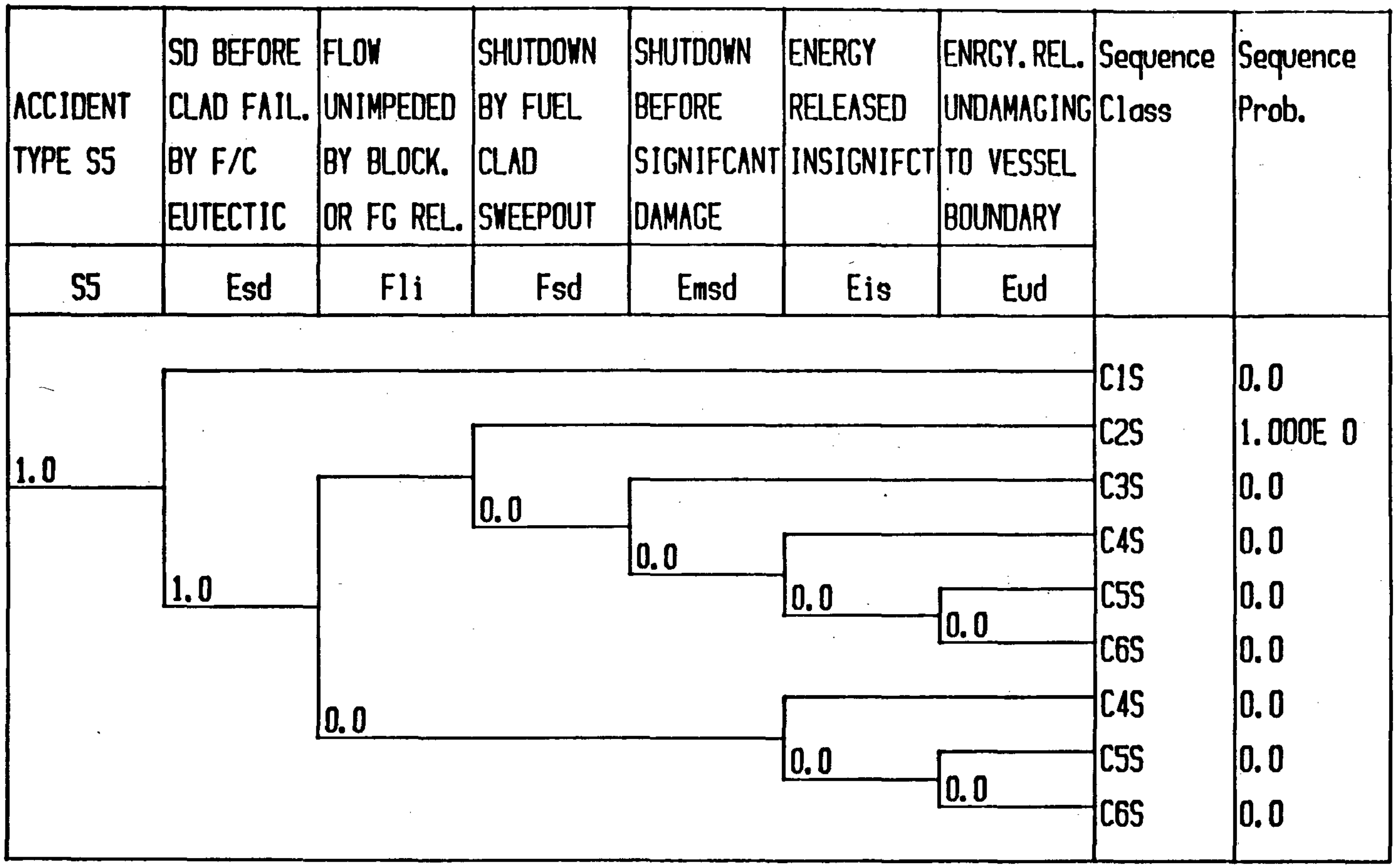


Figure A4.3-26- PRISM: Core Response Event Tree - LOSHR (S5)

A4.4 Containment Response Event Trees

A4.4.1. Introduction

The purpose of the Containment Response Event Trees is to determine, given each of the 12 Core Damage Categories described previously, the probability of reaching each of the possible Containment Release Categories (including the category "OK" i.e. no release). The 12 Core Damage Categories represent a mutually exclusive, collectively exhaustive set of states of the reactor core at the end of the early phase of accident response. This early phase is termed here "core response" while the later phase is referred to as "containment response". The Containment Release Categories likewise represent a set of mutually exclusive collectively exhaustive states of the system. These categories differ from one another in the magnitude and timing of radioisotopes released, and include one category to cover the remaining possibility that there is no release.

There are two major tasks which make up the containment response analysis phase of this risk assessment. The first task is the development of an event tree for each of the 12 Core Damage Categories. The second task is to perform a computer calculation simulating a typical event sequence leading to each of the Containment Release Categories, which are the outcomes of these trees. The result of these computer calculations is a quantitative description of the nature of the release. Specifically, the cumulative fraction of the core inventory of each of five isotope groups released as a function of time is determined. The first task was performed by General Electric Company, the second by Hanford Engineering Development Laboratory. These quantitative descriptions of the release categories can then be used in the next phase of the risk assessment to calculate the expected public consequences given each release.

Subsections A4.4.2 through A4.4.6 discuss the first task; namely, event tree development, while subsection A4.4.7 discusses the second task of release category characterization. Figures A4.4-1 through A4.4-12 show the containment Response Event Tree for Core Damage Category C1. Subsections A4.4.2 through A4.4.5 discuss respectively the four events in the similar Containment Response Event Trees for all categories. These events address uncertainties in the areas of debris coolability, early vessel thermal failure, core uncovery due to sodium boil-off, and energetic re-criticality at the time of late core collapse. Subsection A4.4.6 discusses the definition and assignment of Containment Release Categories to each of the outcomes of the event trees.

A.4.4.2 Event: Early Debris Coolable

Several of the 12 Core Damage Categories are events in which there has been partial core meltdown so that there is some amount of core debris re-distributed onto in-vessel structures at the end of this early phase. The failure of this event constitutes early release of the core from the primary system.

For Core Damage Categories C1, C2, C3, C1S, C2S, and C3S there is little or no debris so the probability of failure is zero. For categories C6 and C6S there has been a 100% core meltdown so little credit is taken for the possibility of coolability and a failure probability of 0.90 is assigned. For categories C5 and C5S there has been a 25% core melt event. Events of this nature were discussed in a Risk Evaluation of Severe Accident Progression in PRISM (Reference A4.4-1) by Argonne National Laboratory. A probability of failure of 0.01 was assigned in Reference A4.4-1. Categories C4 and C4S have 5% early core debris. There appears to be little or no chance that such a small quantity of debris could cause vessel failure. A failure probability of 10^{-4} has been assigned so as not to arbitrarily rule out this possibility.

A4.4.3 Event: No Early Vessel Thermal Failure

This event is considered in the event trees when the event Early Debris Coolable takes the success branch. It is included because, even though the debris geometry is coolable, a loss of shutdown heat removed would result in an increase in sodium temperature up to the boiling point. Depending on the stress levels in various structures such high temperature could lead to failures that would result in an early melt-through.

The probability of this event is the product of the probability of loss of Shutdown Heat Removal System (SHRS) given an accident resulting in the core damage state, and the probability of early thermal vessel failure given loss of SHRS. Based on the failure rate of 4.4×10^{-7} per year and SHRS mission time of 1/2 year for core damage events, a probability of loss of SHRS of $(4.4 \times 10^{-7}/\text{yr})(1/2\ \text{yr}) = 2.2 \times 10^{-7}$ is obtained.

In order to judge the likelihood of creep-induced structural failure of the vessel, a calculation was performed using the creep rate equation in the Nuclear Systems Materials Handbook. The stress in the top of the reactor vessel is 2700 psi. The temperature which the sodium might reach in a pressurized vessel before boiling would appear to be in the range 1600 to 1700 degrees Fahrenheit. At 1600 degrees the equation gives a creep rate of 1% in 400 hours; at 1700 degrees it is 1% in 40 hours. Since the normal scenario of sodium boil-off following loss of SHRS leads to core uncovery and meltdown after 99 hours, the term "Early" for the event "Early Vessel Thermal Failure" means substantially earlier than 99 hours. Since several percent creep would be necessary to result in loss of integrity of the vessels, this event appears unlikely before 99 hours. Thus a conditional probability of 0.10, given loss of SHRS, is conservatively assigned for Early Vessel Thermal Failure due to the substantial uncertainties in the validity of the creep rate equation and in the temperature.

Thus the product of the above two probabilities gives a probability for Early Vessel Thermal Failure of $(2.2 \times 10^{-7})(0.10) = 2.2 \times 10^{-8}$.

A4.4.4 Event: No Core Uncovery

If there is no early vessel thermal failure but SHRS has been lost, the primary system will heat up to the sodium boiling temperature and sodium boil-off will occur. When the core is uncovered meltdown and melt-through will occur. If SHRS could be restored before core uncovery, melt-through would be prevented. However, the mean repair time for SHRS repair obtained by Reference A4.4-2 is two years, due to the catastrophic nature of the failure involved. Hence there is no chance of preventing core uncovery once SHRS has been lost. Thus the probability of SHRS failure given a core damage event, namely 2.2×10^{-7}, is also the probability of core uncovery.

A4.4.5 Event: No Late Energetic Expulsion

When any of the three preceding events in the Core Response Event Trees takes the failure branch there will be a large-scale core meltdown. The possibility of energetic re-criticality at this time of core collapse has been discussed in Reference A4.4-21 and was assigned a probability of 0.01, which has been used here.

A4.4.6 Assignment of Containment Response Event Sequences to Containment Release Categories

The magnitude and timing of the releases which would occur depend on the nature of the core damage category and on the particular combination of containment response events which then occur. Argonne National Laboratory identified in Reference A4.4-1 a number of containment release categories sufficient to cover all the possible combinations of events. In performing this PRA, GE and HEDL reviewed those categories and decided that they provided a set of qualitatively different releases adequate for this PRA. The only modification was to eliminate some very small releases for which no physically possible scenario could be found for PRISM. These categories will be defined below.

The most severe release scenario would be an early core disruptive phase resulting in an energetic expulsion from the vessel with 100% core melt and melt-through of the vessels. The early meltdown releases fission products to the sodium, while the energetic transient also raises the sodium temperature, thus reducing the time required to heat up to the boiling point, at which time release of fission products along with the boiling sodium occurs. This results in a large release at an earlier time than other events. This release category is identified as R4A and occurs, by definition, given Core Damage Categories C6 or C6S, when the event Early Debris Not Coolable occurs. When the Early Debris Coolable branch is taken for C6 or C6S the resulting Containment Release Category is identified as R3.

If the early core disruptive accident results in only a partial core melt with no energetic expulsion, as in Core Damage Categories C5 and C5S, then early debris coolability failure results in a melt-through release category identified as R2A. If debris is coolable and cooling is maintained, then no release results from this accident and so a success state "OK" can be reached.

The six remaining Containment Release Categories result from events where the cause of the release is loss of shutdown heat removal leading to sodium boil-off, meltdown at core uncovery, and melt-through of the vessels. These events result when either of the events, No Early Vessel Thermal Failure or No Core Uncovery, take the failure branch. Three of the remaining categories are designated by the prefix R6 and three by R8, the difference being that the R6 categories represent the case of No Late Energetic Expulsion, while the R8 categories do have such an expulsion. The three R6 categories, R6A, R6U, and R6S, differ as to the nature of any core disruptive accident which may have occurred initially. R6A represents the cases C1 and C1S where there has been no early transient, so the sodium is initially at normal temperature. R6U covers Core Damage Categories C2, C3, C4, C2S, C3S, C4S where there has been an early transient resulting in elevated sodium temperatures but with no large scale core damage. R6S represents the cases C5 and C5S where there has been an early transient plus large scale core melting (25%).

The three R8 categories are designated R8A, R8U, and R8S where the suffixes A, U, S have the same significance as for R6.

In most cases the assignment of one of the above nine categories or of the success state "OK" to an outcome branch of the Containment Response Event Trees is obvious from the definition of the category. Exceptions will be described below. Table A4.4-1 summarizes the nine Containment Release Categories.

In all the Containment Response Event Trees the assignment of categories to the outcomes following from Early Vessel Thermal Failure and from Core Uncovery have been the same in the same tree. In reality the Early Vessel Thermal Failure would modify somewhat the timing of the release, but the difference was judged to have a minor effect on subsequent consequences so no distinction was made.

In the trees for Core Damage Categories C4, C4S, C5, and C5S the outcome, Early Debris Not Coolable With Late Energetic Expulsion, has been taken to mean debris bed re-criticality and has been assigned to R4A.

A.4.4.7 Calculation of Radioisotope Release for the Containment Release Categories

For each of the Containment Release Categories in Table A4.4-1 it is necessary to determine the amount and timing of the release of radioactive isotopes from the plant. This release description will then be wed in subsequent calculations described in Section A4.5 to obtain the public consequences of each release. In Subsection A4.4.7.1 information common to the treatment of all the release calculations will be given. Subsection A4.4.7.2 contains a brief description of the sequence of events for each specific release category. Subsection A4.4.7.3 discusses the results of the calculations. The methods, models, and calculations described below were performed by Hanford Engineering Development Laboratory (HEDL) in direct support of this PRISM PRA.

A4.4.7.1 General Methods and Assumptions

In order to determine the timing of various events in the transport of the radioisotopes out of the system, it is necessary to simulate the dynamic thermal response of the system when subjected to each of the nine release category scenarios. To do this a thermal model of PRISM was developed and implemented as a set of equations and data using general purpose differential equation simulation software. The contents of this model will be discussed under two general topics: (1) the thermal model, and (2) radioisotope transport assumptions.

The geometric model of PRISM embodied in the thermal calculations is an axi-symmetric model in which lateral heat transport only is considered. This should be a relatively minor conservatism since the area of the head and vessel bottom is small compared to the lateral surface area. This r-z geometry is divided into lumped parameter regions such as the core, the above core sodium, the below core sodium, the shield and structures in vessel, the collector cylinder, the concrete silo, the Head Access Area (HAA) volume, and the HAA structures. Heat capacities of each region are taken into account in dynamic heat transfer equations. Energy sources accounted for include decay heat in the fuel, sodium fires, and sodium-concrete (water) reactions. Natural circulation in the vessel is estimated in order to account for in-vessel temperature distributions. The energy input from the fuel batch stored above the core was not accounted for; however, this should be a relatively minor addition.

For events having expulsion of sodium from the vessel, it is assumed that this expulsion is upward into the Head Access Area (HAA), and that a sodium fire occurs. For events with melt-through of the vessels, the sodium leaks out to fill the RVACS annulus, thus disabling RVACS. A quiescent sodium pool fire then occurs at the surface at the annulus. This fire is terminated when one foot of oxides has accumulated on the surface. Since the silo has no metal liner a sodium-concrete reaction is assumed to take place. Once the sodium heats up to the boiling point the sodium vapor is assumed to be driven out to the RVACS exhaust ports where it burns in direct contact with the external atmosphere producing direct aerosol release.

The above thermal model of PRISM was validated by performing a calculation for comparison to those performed by GE for RVACS transients. Very close agreement was achieved, thus indicating the accuracy of the model and data used.

In treating the release of radioisotopes, five groups of isotopes were defined to be accounted for separately: (1) noble gases, (2) volatiles, (3) halogens, (4) fuel, and (5) solids. These isotope groups are composed of the following principal elements: Noble Gases, Krypton and Xenon;

Volatiles, Cesium and Tellurium; Halogens, Iodine; Fuel, Pu and other actinides; Solids, Strontium and Barium.

At the time that these calculations were being developed the reference PRISM core used oxide fuel, hence all calculations of the release timing were based on oxide rather than metal fuel. The potential significance of differences with metal fuel was scoped by developing three metal fuel release descriptions by extrapolation from the oxide results. These releases were then used to calculate public consequences. These sensitivity calculations will be discussed in Section A4.5. One of the principal differences would appear to be the early release of most of the noble gases and volatiles from metal fuel. This is due to two facts: (1) most of these volatiles migrate out of metal fuel into the plenum during normal operations, and (2) clad failure due to fuel-clad metal eutectic penetration is expected to occur before sodium boiling, thus releasing the plenum gases. The other major effect of metal fuel is that such fuel is capable of reacting to form more stable compounds such as oxides, thus adding an additional source of energy. Furthermore, the reaction of the metal core with concrete-water provides a possible mechanism by which the solids and fuel might be converted to aerosol-sized particles suspended in the sodium and hence available for release.

In the release calculations discussed here the Noble Gases are assumed to be released to the cover gas region at the time of fuel failure. When a leak path develops in the head, these gases leak out into the HAA and on into the atmosphere. In all scenarios the leak rate assumed from the HAA is 100 volume % per day.

In the early energetic expulsion categories, R3, and R4A, 10% of the core is assumed expelled into the HAA. This is a conservative estimate. For the late energetic expulsions, R8, 5% of all isotope groups is expelled.

For all release categories, 100% of the volatiles and halogens are assumed released from the fuel, when it melts down, and these isotopes are dissolved in the sodium. These isotope groups are then released in proportion to the sodium as it burns or boils off into the atmosphere. Also released into the sodium at meltdown is approximately 0.0014 of the fuel and solids in the form of aerosol-sized particles which are then carried along with the sodium.

Attenuation by aerosol agglomeration and settling occurs in the HAA for releases following this path. The releases following the RVACS exhaust path are given no credit for attenuation by plateout or fallout because during boil-off the sodium vapor is assumed to be transported as such to the exhaust ports before being burned to oxide aerosol.

A4.4.7.2 Specific Accident Sequence Descriptions

Tables A4.4-2 through A4.4-10 contain descriptions of the sequences of events relevant to the calculation of the releases resulting from each of the nine Containment Release Categories. Time zero in these descriptions is the time at which neutronic shutdown occurs.

A4.4.7.3 Resultant Releases

Curves A,D, and E on Figures A4.4-13 through A4.4-21 show the cumulative fraction of the total core inventory of each of the isotope groups released as a function of time for the nine Containment Release Categories. Curve A shows the Nobles Gases, D the Volatiles and Halogens, and E the Solids and Fuel.

The exact timing of these releases is primarily significant in determining acute fatalities: most of the releases are not significant until substantially after five hours. Five hours is sufficient to begin an effective evacuation of the public; hence, timing subsequent to five hours is of less importance. Only for release category R4A is there significant release early. Thus, despite approximations used in the above calculations, the conclusion that there should be few or no acute fatalities given evacuation would not be strongly affected.

REFERENCES - SECTION A4.4

A4.4-1 "A Risk Evaluation of Severe Accident Progression in PRISM," C. Mueller et al., ANL, ANL-PRISM-5, January 1986.

TABLE A4.4-1

CONTAINMENT RELEASE CATEGORIES

1.	R2A	25% early core melt transient, early debris not coolable
2.	R3	100% early core melt transient with energetic expulsion, debris coolable, no melt-through
3.	R4A	100% early core melt transient with energetic expulsion, early debris not coolable
4.	R6A	No early transient, loss of SHRS and core uncovery, no late energetic expulsion
5.	R6U	Early transient, minor core damage, otherwise same as R6A
6.	R6S	Early transient, 25% core melt, otherwise sameas R6A
7.	R8A	No early transient, loss of SHRS and core uncovery,late energetic expulsion
8.	R8U	Same as R6U but with late energetic expulsion
9.	R8S	Same as R6S but with late energetic expulsion

TABLE A4.4-2

ACCIDENT SEQUENCE DESCRIPTION

R2A - 25% CORE MELT, MELT-THROUGH

Time Hours	Event
0	25 % Core Melt Guard Vessel Failure Sodium Fire Sodium Water Reaction
12.	Core Sodium Boiling
14.7	Bulk Sodium Boiling Purging Noble Gases *(through failed head seal)* Surge in Sodium Burning
36.0	Boiling Rate Dropping Water Release Diminishing
54.6	Coolable Damaged Core (75%) Uncovers & Melts Remainder of Source Term Mobilizes
111.4	Sodium Depleted

TABLE A4.4-3

ACCIDENT SEQUENCE DESCRIPTION

R3 - ENERGETIC CDA - DEBRIS COOLABLE

Time Hours	Event
0	100% Core Melt Coolable Core Debris Bed 10% of Core Expelled Initially into Confinement
16	Maximum System Temperature (1112°F)
16-200	Cooldown

TABLE A4.4-4

ACCIDENT SEQUENCE DESCRIPTION

R4A - ENERGETIC CDA - DEBRIS NOT COOLABLE

Time Hours	Event
0	100% Core Melt 10% of Core Expelled Initially in Confinement Guard Vessel Failure Sodium Fire Sodium Water Reaction
6.7	Bulk Boiling in Lower Vessel Surge in Sodium Burning Sodium Condenses and Refluxes
26	Overall Bulk Boiling of Sodium Purging of Noble Gases (through failed head)
114	Sodium Depleted

TABLE A4.4-5

ACCIDENT SEQUENCE DESCRIPTION

R6A - SHUTDOWN, LOSS OF HEAT REMOVAL

Time Hours	Event
0	Core Intact Adiabatic Heatup From Fission Products Head Venting by Warpage or Seal Failure
25.2	Bulk Boiling in Upper Vessel
99	Core Uncovered Core Melting Guard Vessel Melt-Through Driven Sodium Burning Begins
120	Boiling Ceases in Upper Vessel Sodium Burning Becomes Normal
124	Sodium Depleted

TABLE A4.4-6

ACCIDENT SEQUENCE DESCRIPTION

R6U - TOP OR LOF WITH LOSS OF SHUTDOWN HEAT REMOVAL

Time Hours	Event
0	Core Intact Adiabatic Heatup From Fission Head Venting by Warpage or Seal Failure
4.9	Bulk Boiling in Upper Vessel
68.5	Core Uncovered Core Melting Guard Vessel Melt-Through Driven Sodium Burning Begins
86.8	Sodium Depleted

TABLE A4.4-7

ACCIDENT SEQUENCE DESCRIPTION

R6S - TOP OR LOF WITH CORE MELT
LOSS OF SHUTDOWN HEAT REMOVAL

Time Hours	Event
0	30% Initial Core Melt Head Venting by Warpage or Seal Failure
5.6	Bulk Boiling in Upper Vessel
7.0	Bulk Boiling in Lower Vessel
64	Core Uncovered Bulk Core Melting Guard Vessel Melt-Through Driven Sodium Burning Begins
86.9	Sodium Depleted

TABLE A4.4-8

ACCIDENT SEQUENCE DESCRIPTION

R8A - SHUTDOWN, LOSS OF SHUTDOWN HEAT REMOVAL
5% ENERGETIC EXPULSION AT CORE COLLAPSE

Time Hours	Event
0	Core Intact
	Adiabatic Heatup From Fission Products
	Head Venting by Warpage or Seal Failure
25.2	Bulk Boiling in Upper Vessel
99	Core Uncovered
	Core Melting
	Bulk Boiling in Lower Vessel
	Guard Vessel Melt-Through
	Driven Sodium Burning Begins
	5% Energetic Expulsion
120	Boiling Ceases in Upper Vessel
124	Sodium Depleted

TABLE A4.4-9

ACCIDENT SEQUENCE DESCRIPTION

R8U - TOP OR LOF WITH LOSS OF SHUTDOWN HEAT REMOVAL
5% ENERGETIC EXPULSION AT CORE COLLAPSE

Time Hours	Event
0	Core Intact Adiabatic Heatup from Fission Products Head Venting by Warpage or Seal Failure
4.9	Bulk Boiling in Upper Vessel
64	Core Uncovered Core Melting Bulk Boiling in Lower Vessel Guard Vessel Melt-Through Driven Sodium Burning Begins 5% Energetic Expulsion
86.8	Sodium Depleted

TABLE A4.4-10

ACCIDENT SEQUENCE DESCRIPTION

R8S - TOP OR LOF WITH CORE MELT, LOSS OF SHUTDOWN
HEAT REMOVAL, 5% ENERGETIC EXPULSION AT CORE COLLAPSE

Time Hours	Event
0	30% Initial Core Melt Head Venting by Warpage or Seal Failure
5.6	Bulk Boiling in Upper Vessel
7.0	Bulk Boiling in Lower Vessel
64	Core Uncovered Bulk Core Melting Guard Vessel Melt-Through Driven Sodium Burning Begins Sodium Burning Becomes Normal 5% Energetic Expulsion
86.9	Sodium Depleted

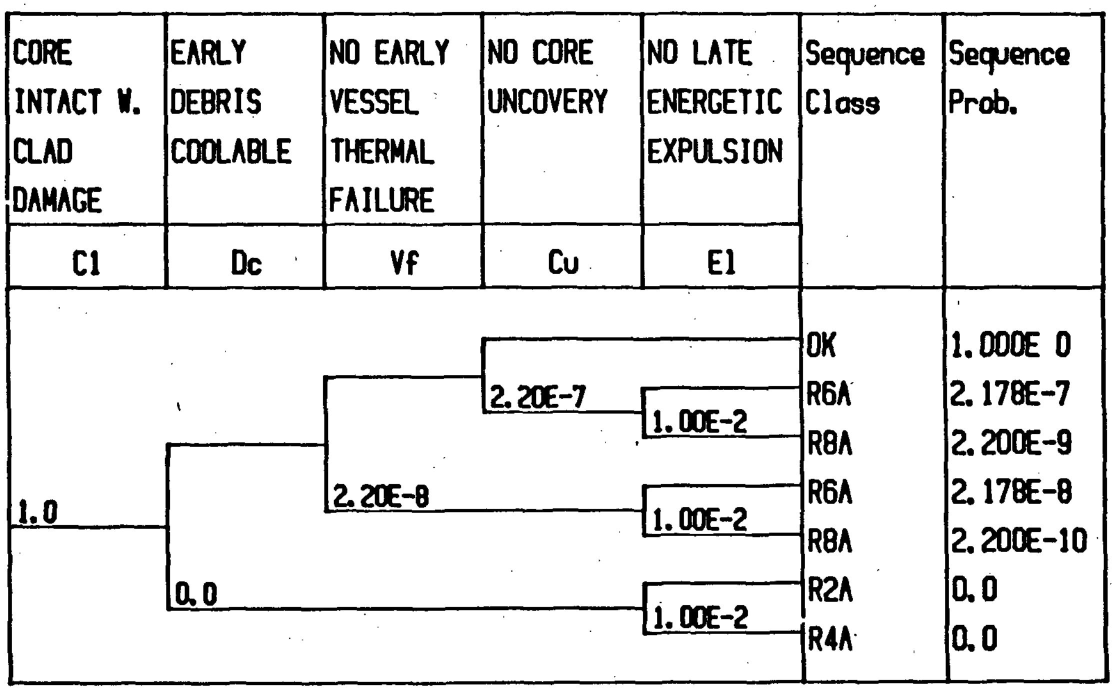


Figure A4.4-1 - CONT.R.TREE C1: Core Intact Clad Damage

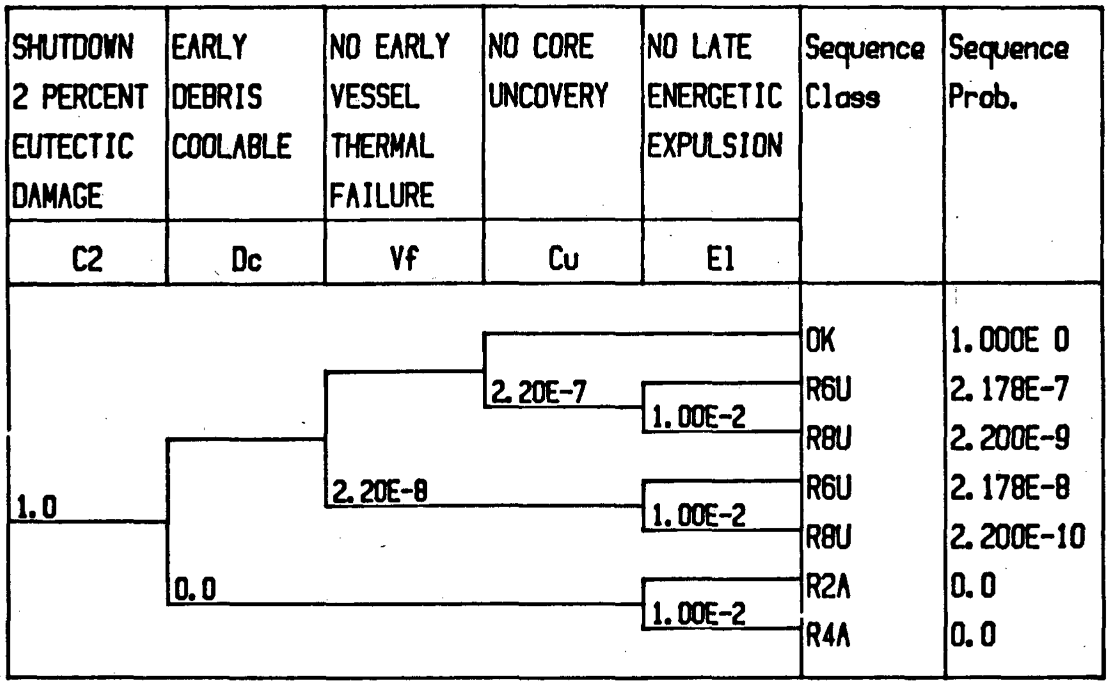


Figure A4.4-2 - CONT.R.TREE C2: Shutdown, 2% Core Damage

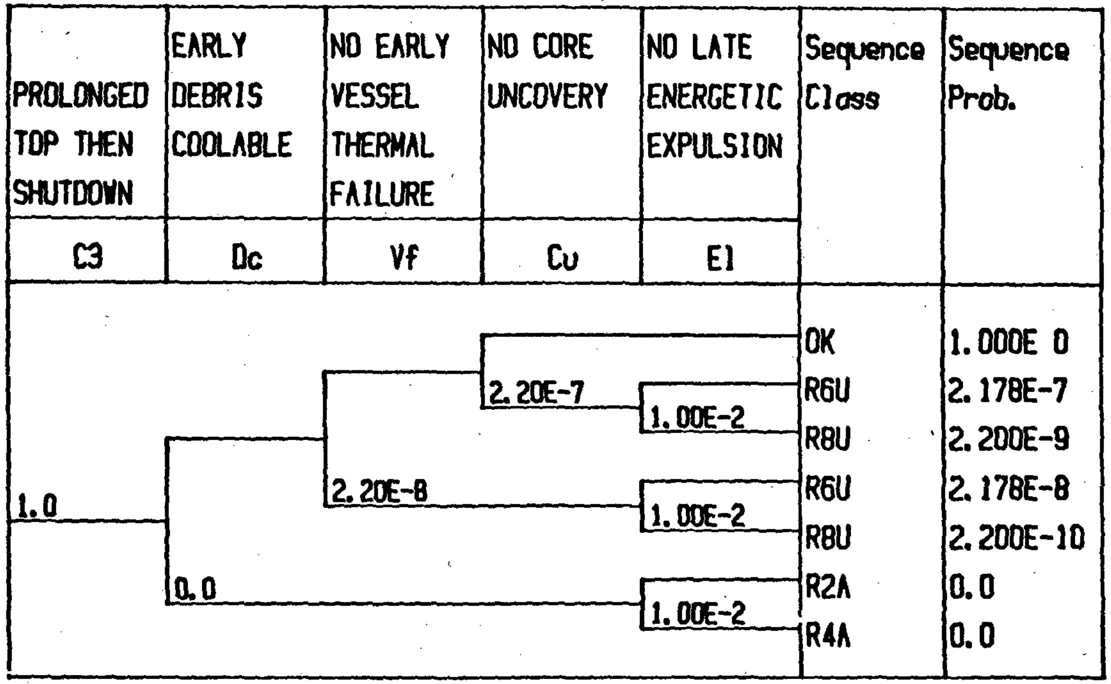


Figure A4.4-3 - CONT.R.TREE C3: Prolonged TOP then S/D

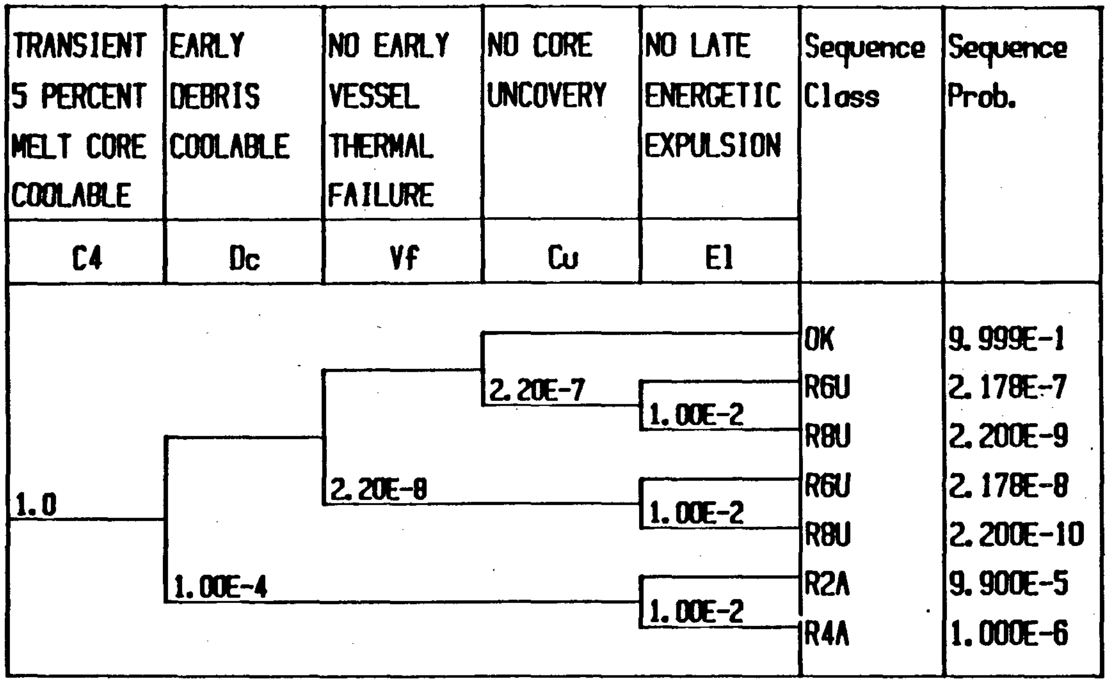


Figure A4.4-4-CONT.R.TREE C4: 5% Early Core Melt

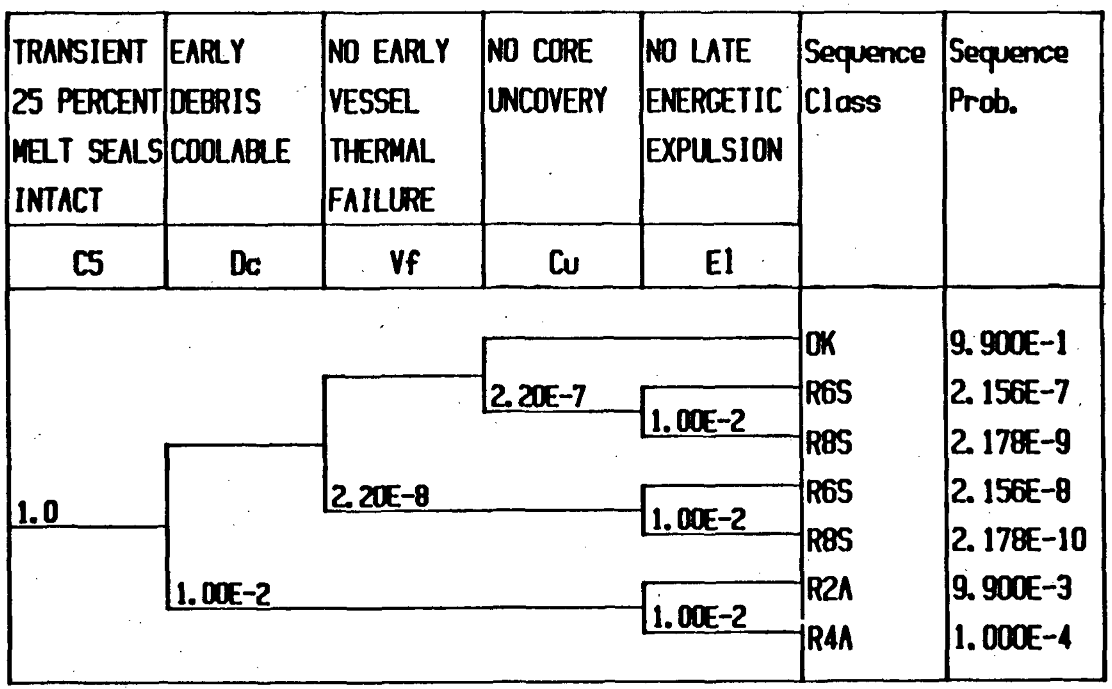


Figure A4.4-5 -CONT.R.TREE C5: 25% Early Core Melt

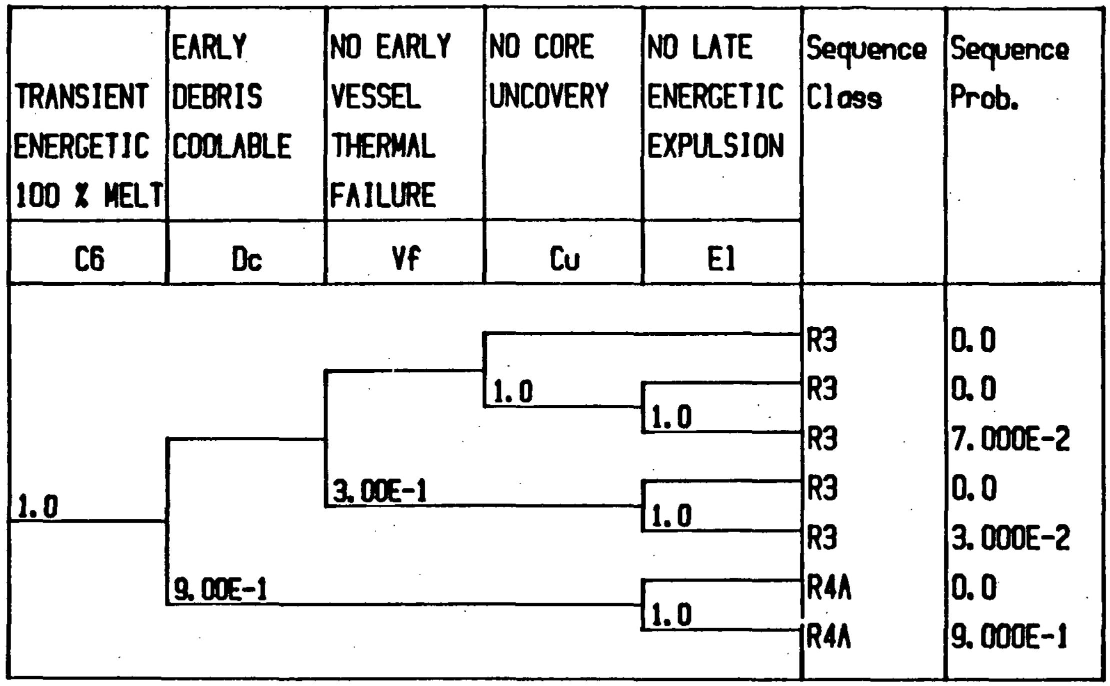


Figure A4.4-6 - CONT.R.TREE C6: 100% Early Core Melt

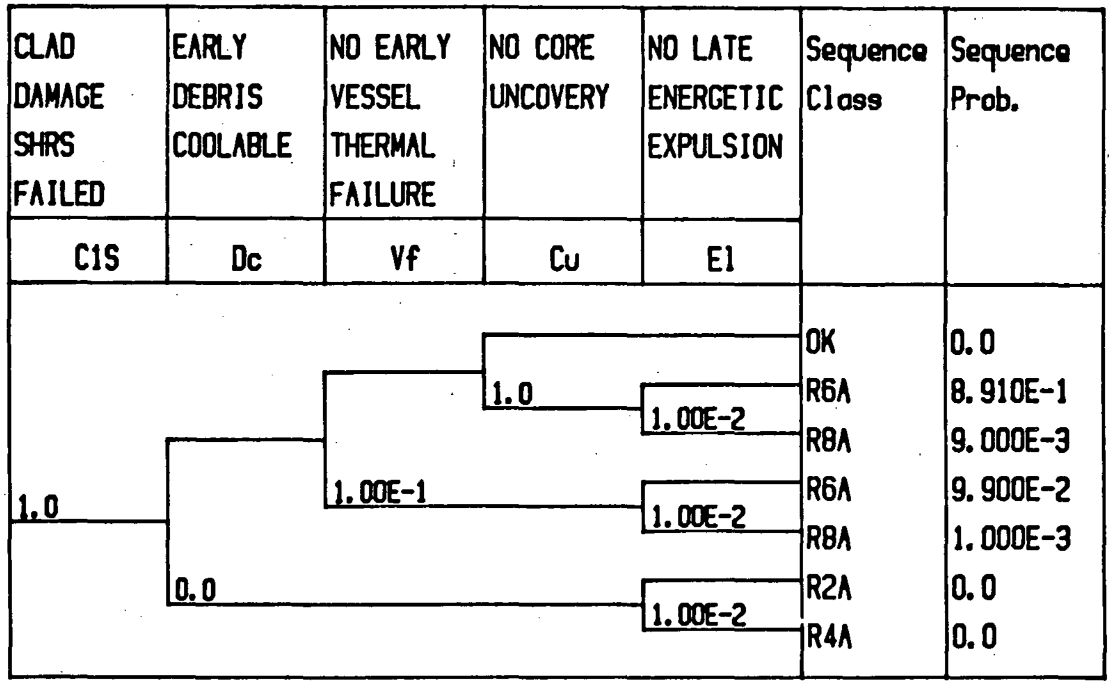


Figure A4.4-7 - CONT.R.TREE C1S: C1 With SHRS Failure

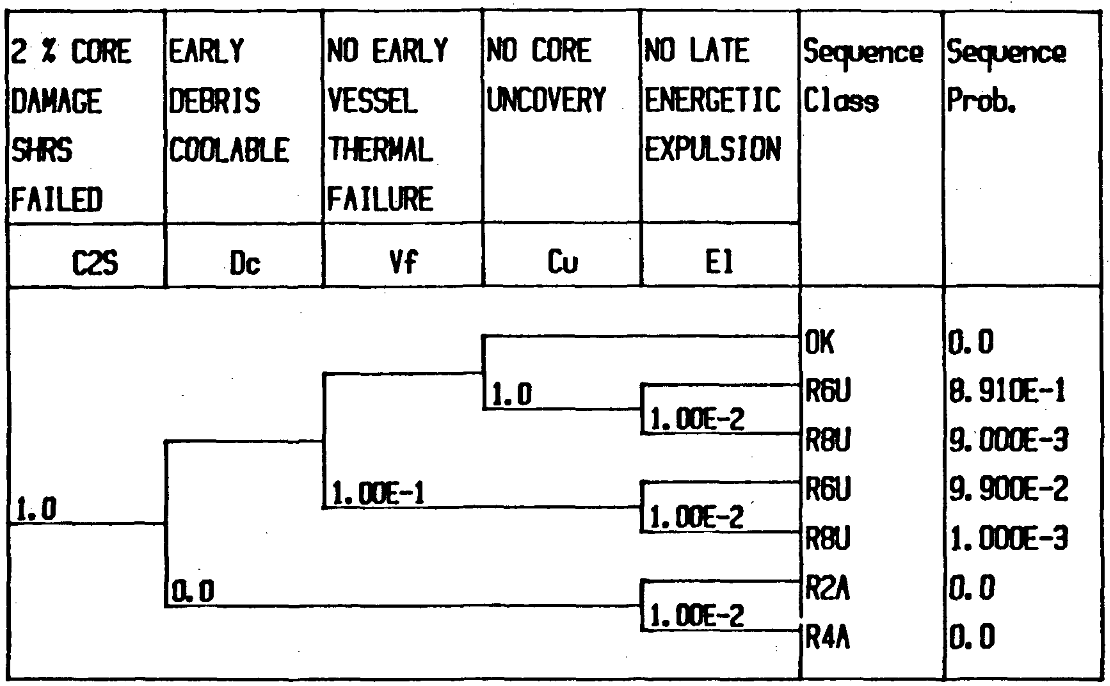


Figure A4.4-8 - CONT.R.TREE C2S: C2 With SHRS Failure

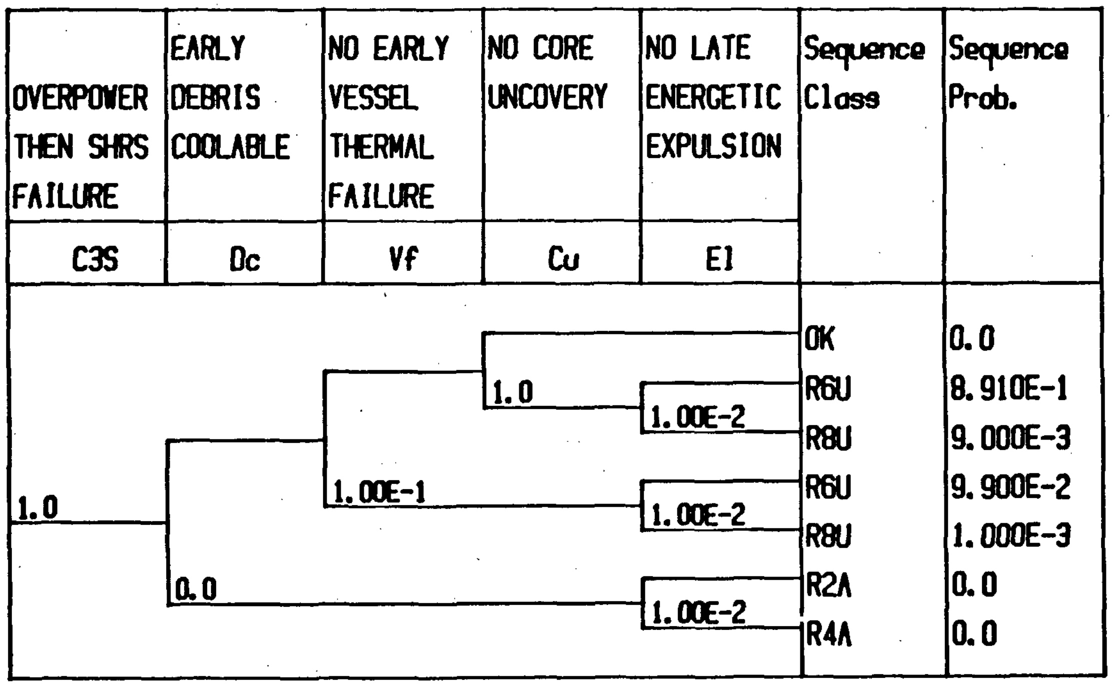


Figure A4.4-9 - CONT.R.TREE C3S: C3 With SHRS Failure

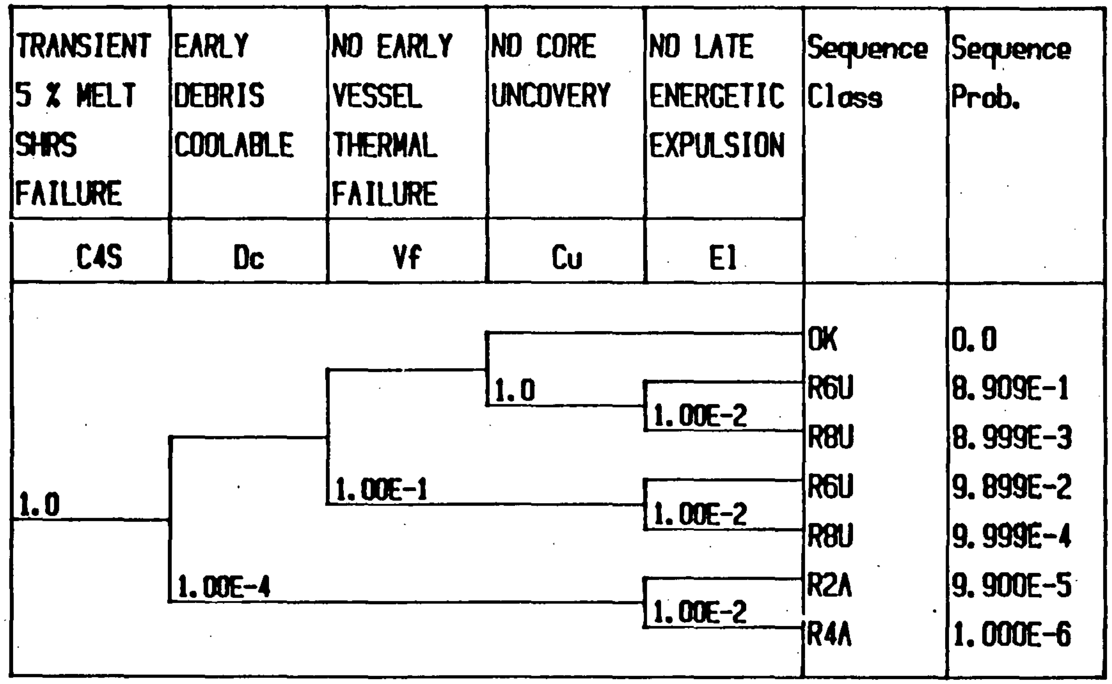


Figure A4.4-10- CONT.R.TREE C4S: C4 With SHRS Failure

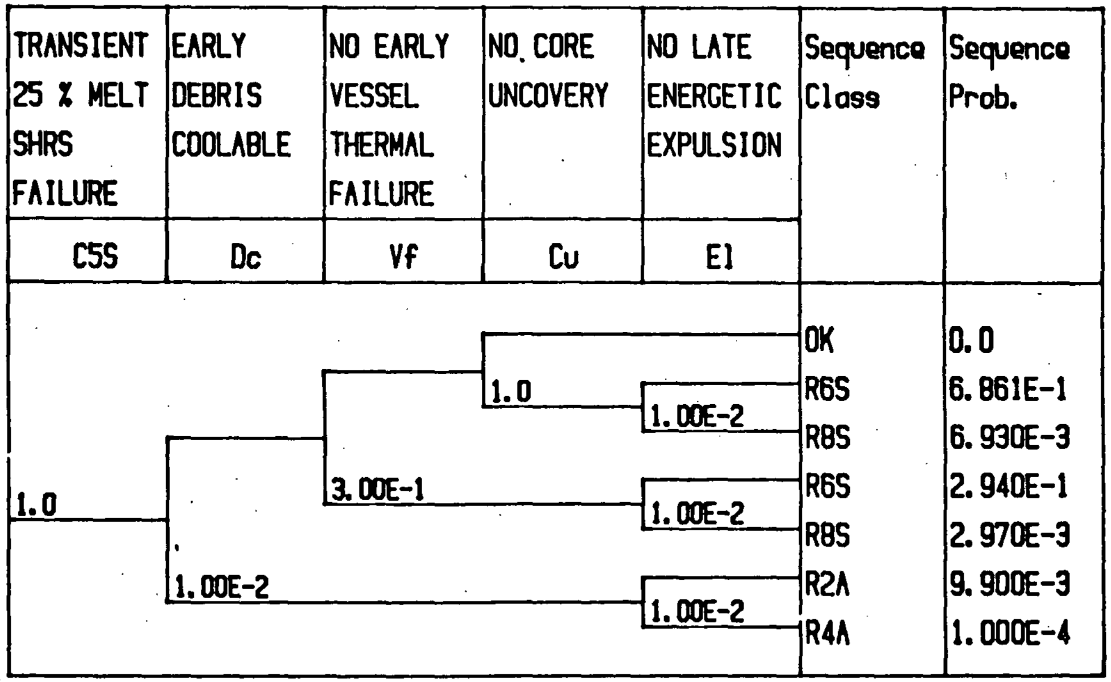


Figure A4.4-11 - CONT. R. TREE C5S: C5 With SHRS Failure

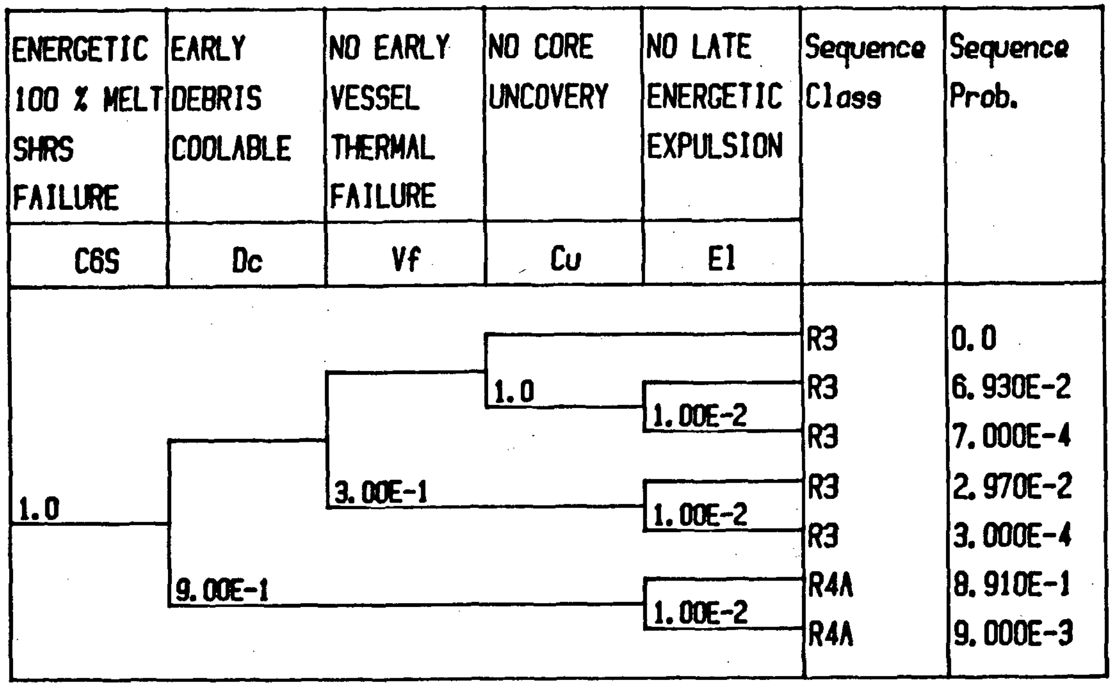


Figure A4.4-12 - CONT.R.TREE C6S: C6 With SHRS Failure

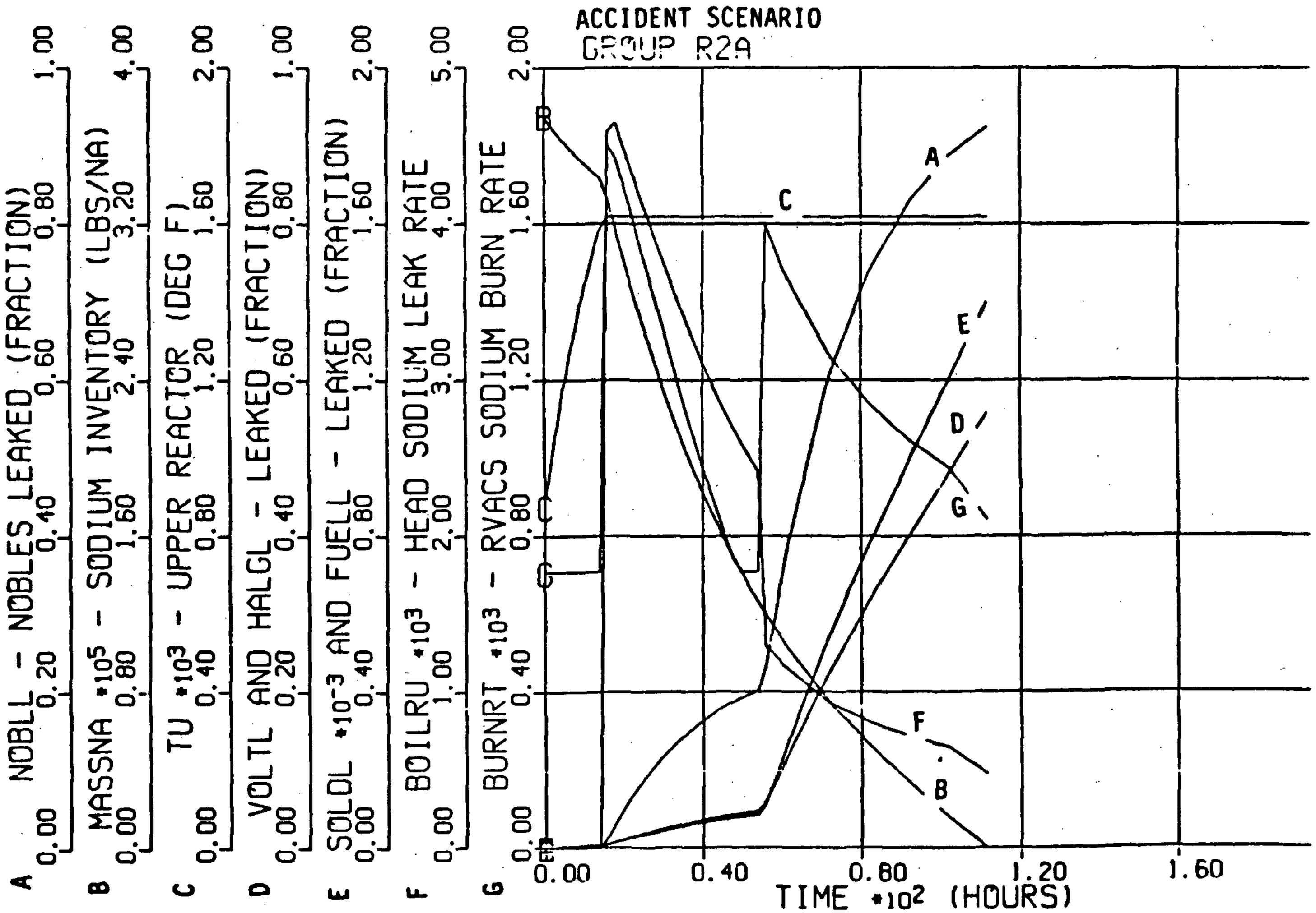


Figure A4.4-13 Accident Scenario for Containment Release - Category R2A

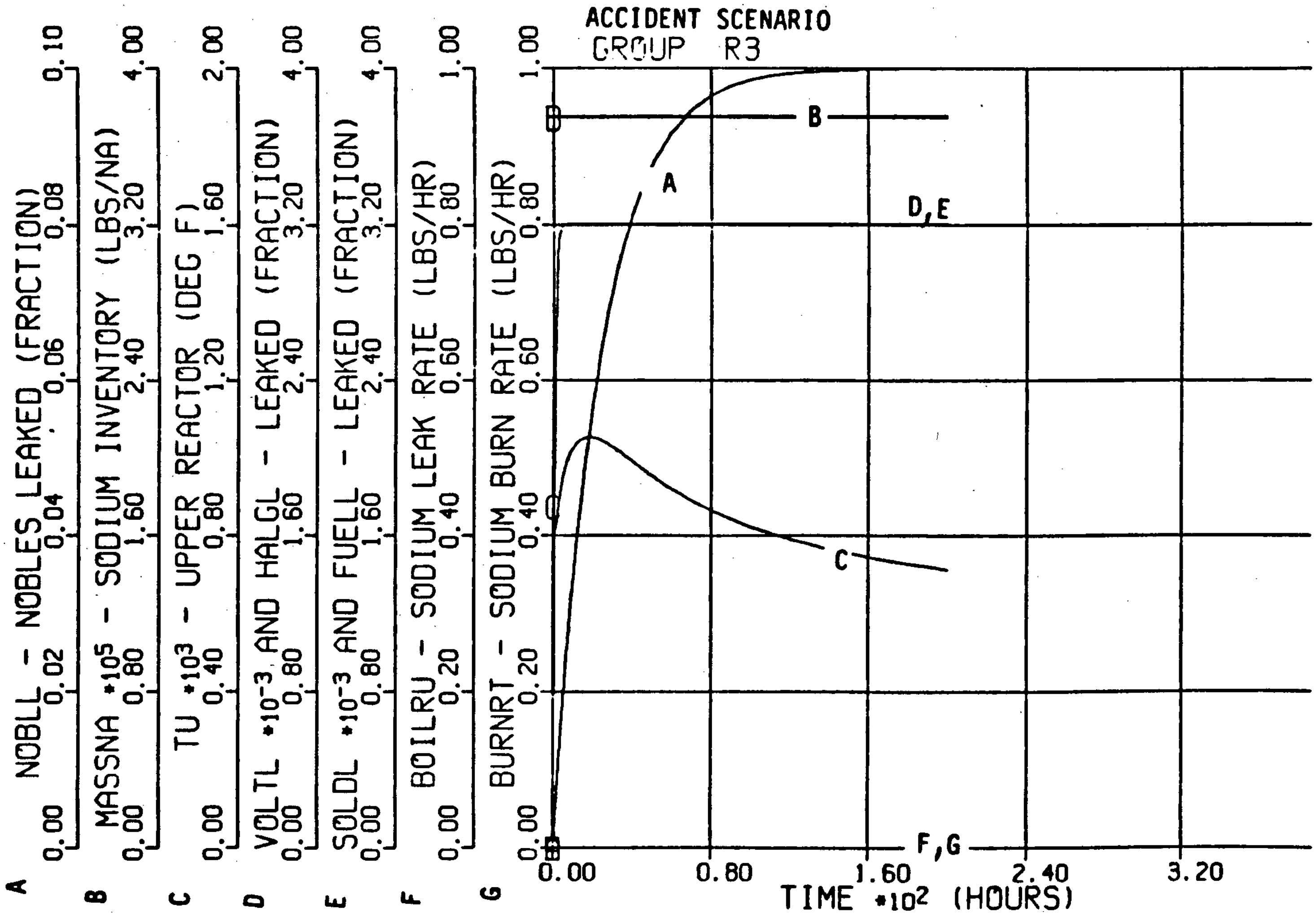


Figure A4.4-14 Accident Scenario for Containment Release - Category R3

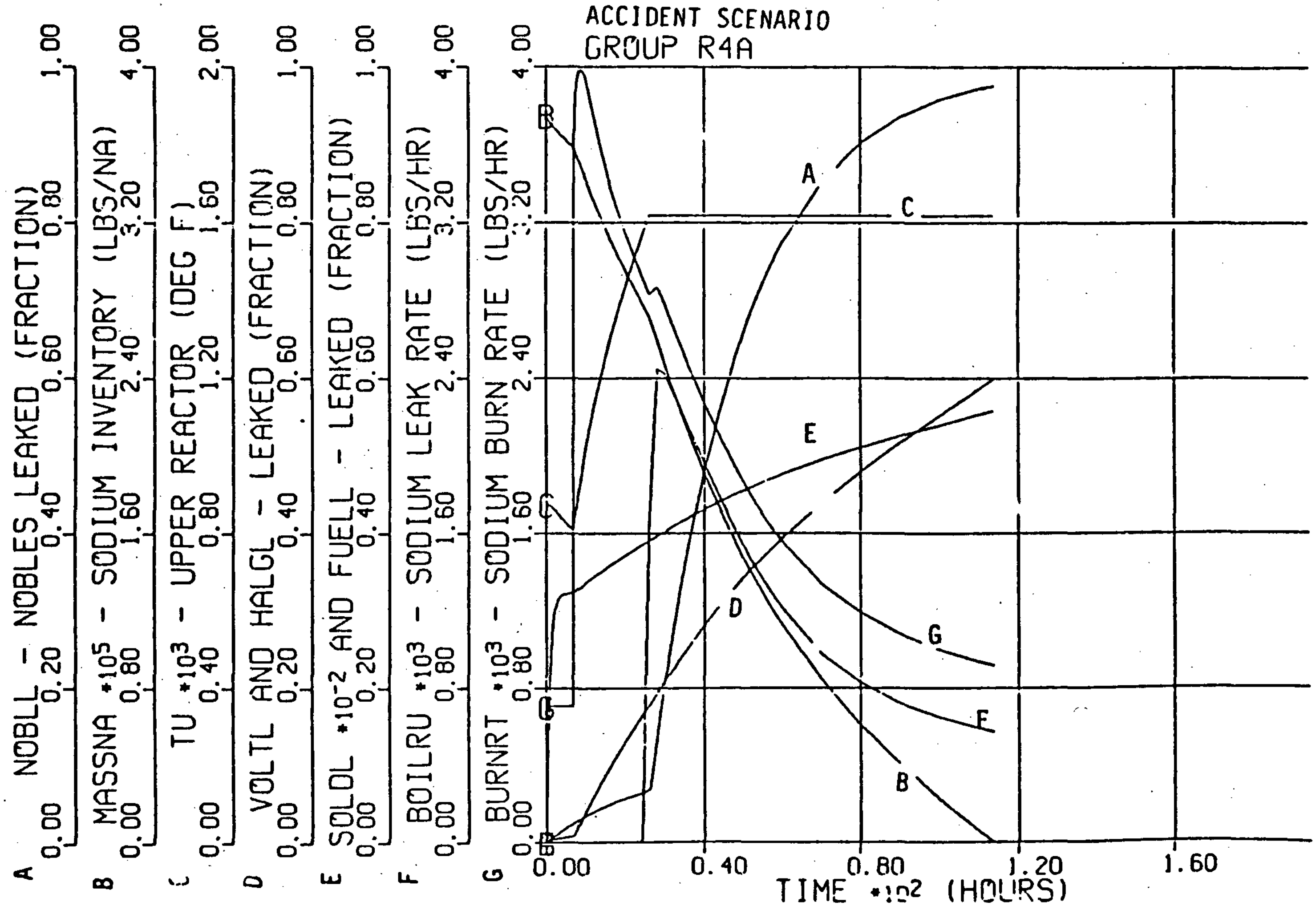


Figure A4.4-15 Accident Scenario for Containment Release - Category R4A

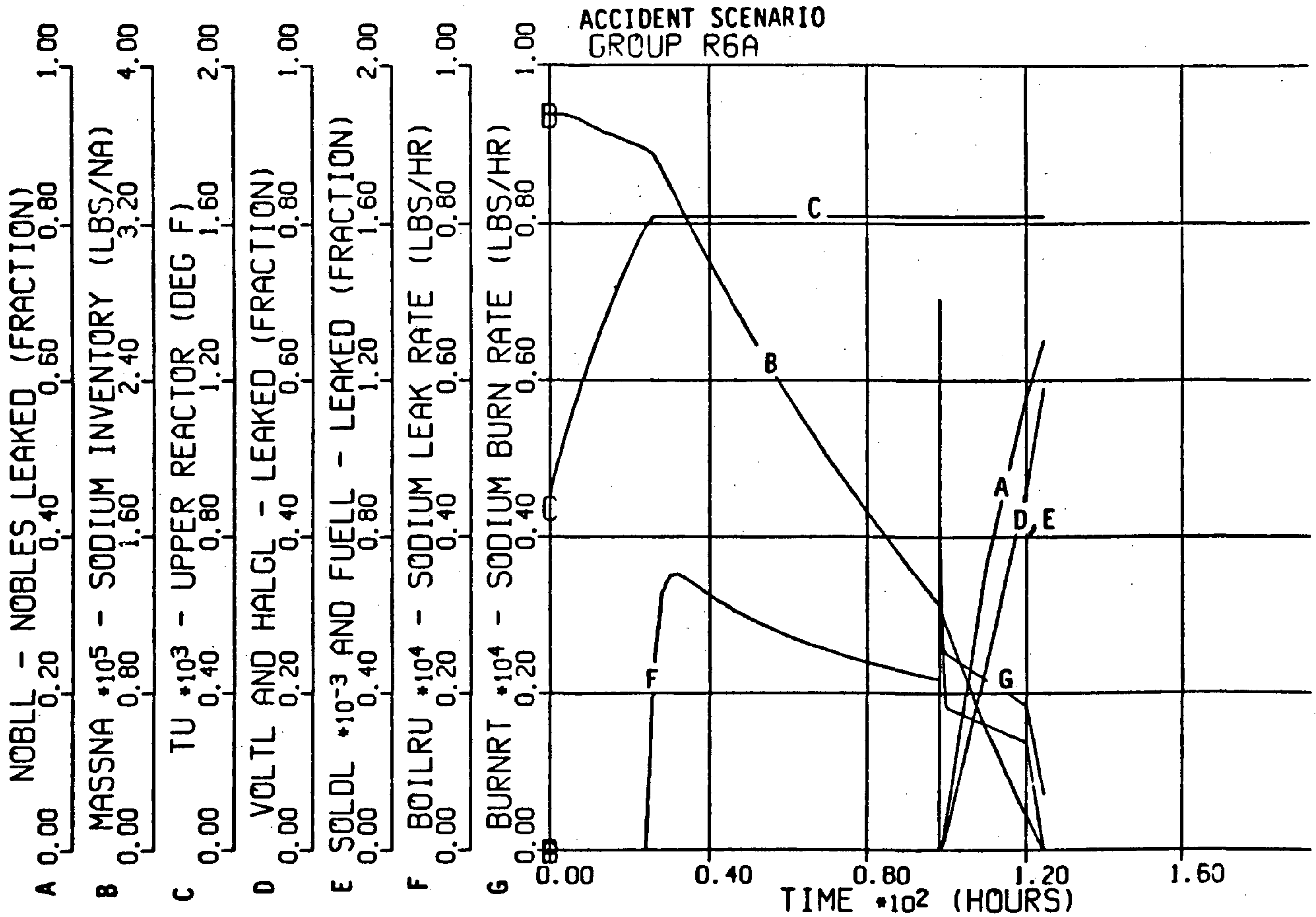


Figure A4.4-16 Accident Scenario for Containment Release - Category R6A

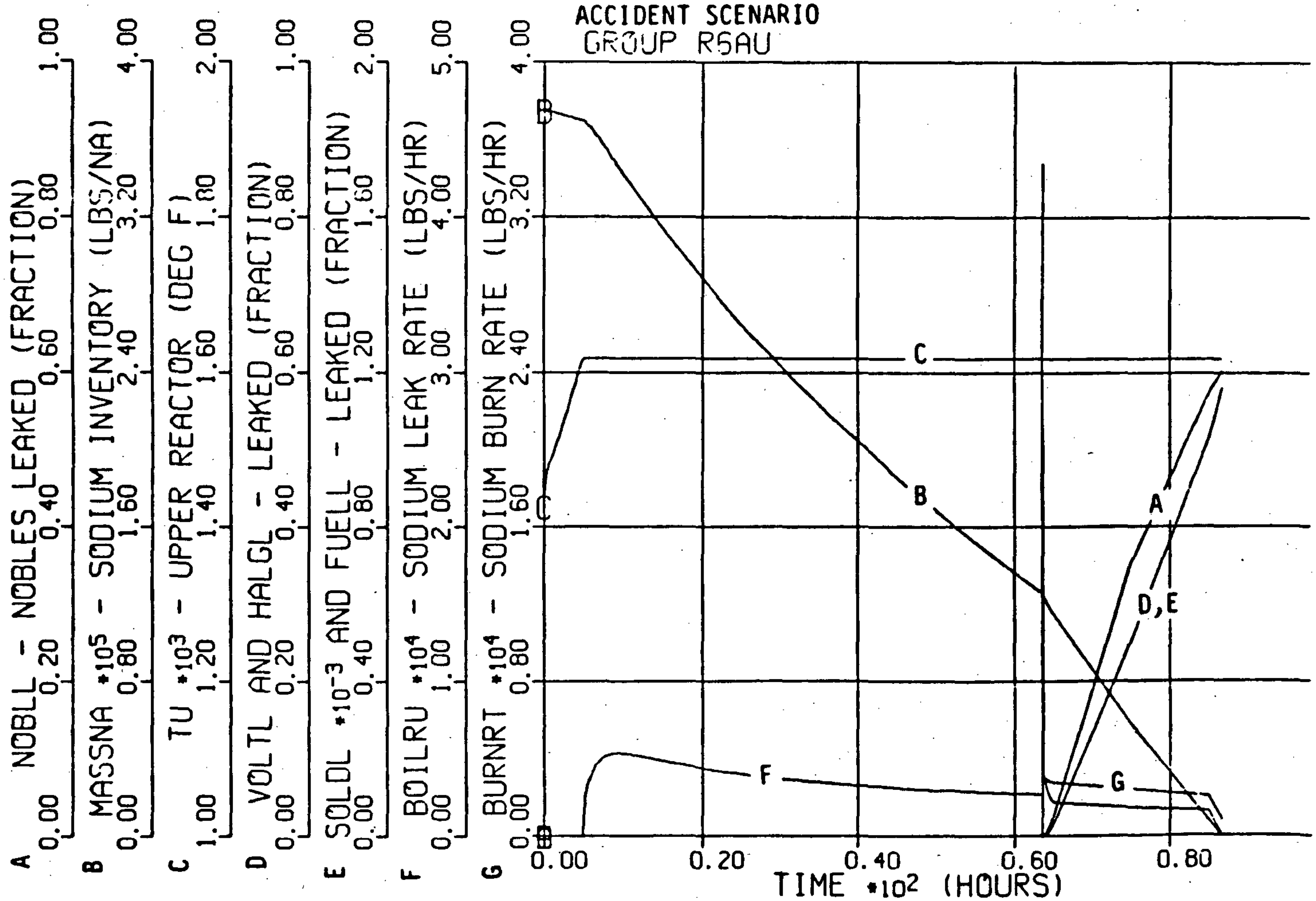


Figure A4.4-17 Accident Scenario for Containment Release - Category R6U

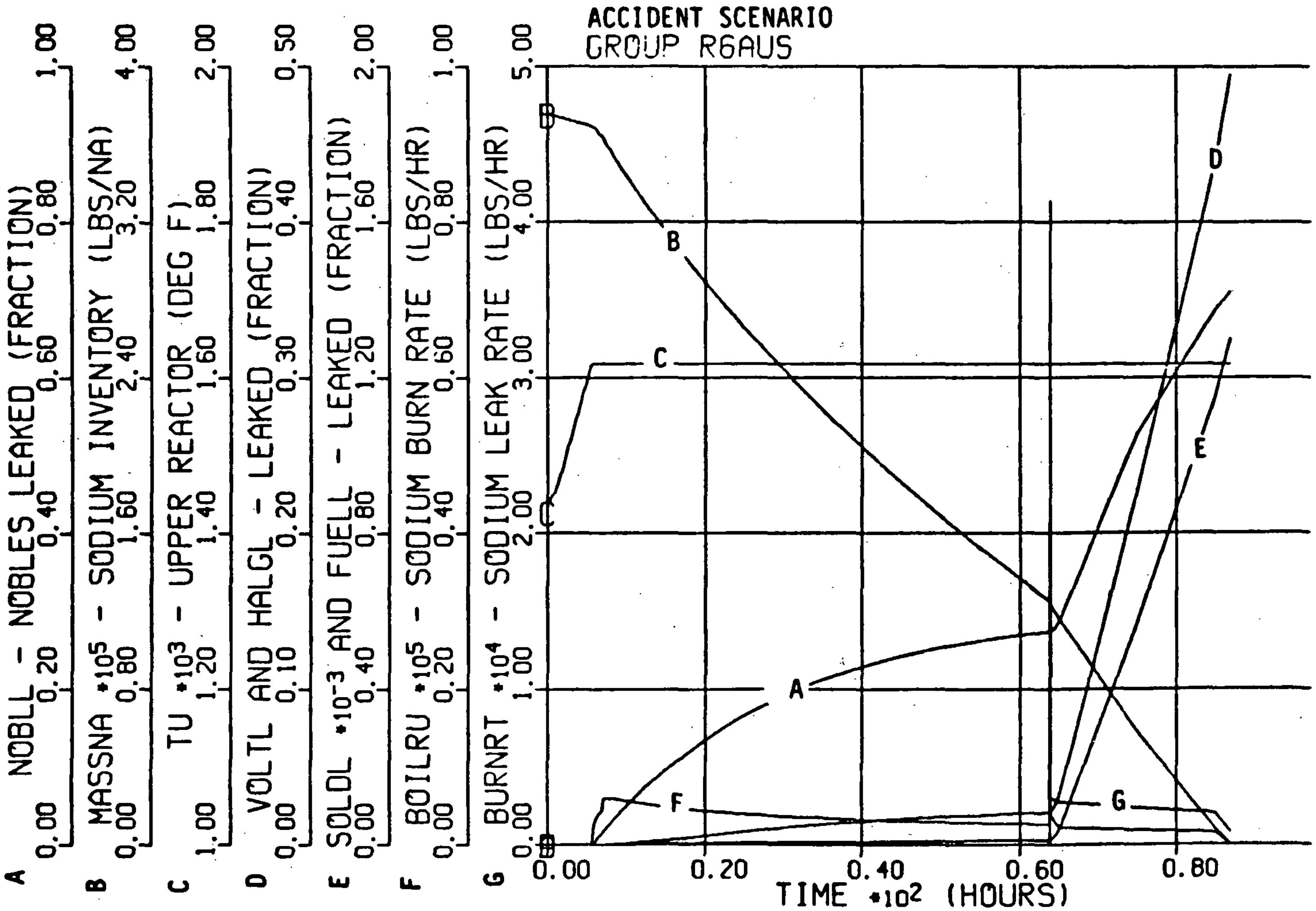


Figure A4.4-18 Accident Scenario for Containment Release - Category R6S

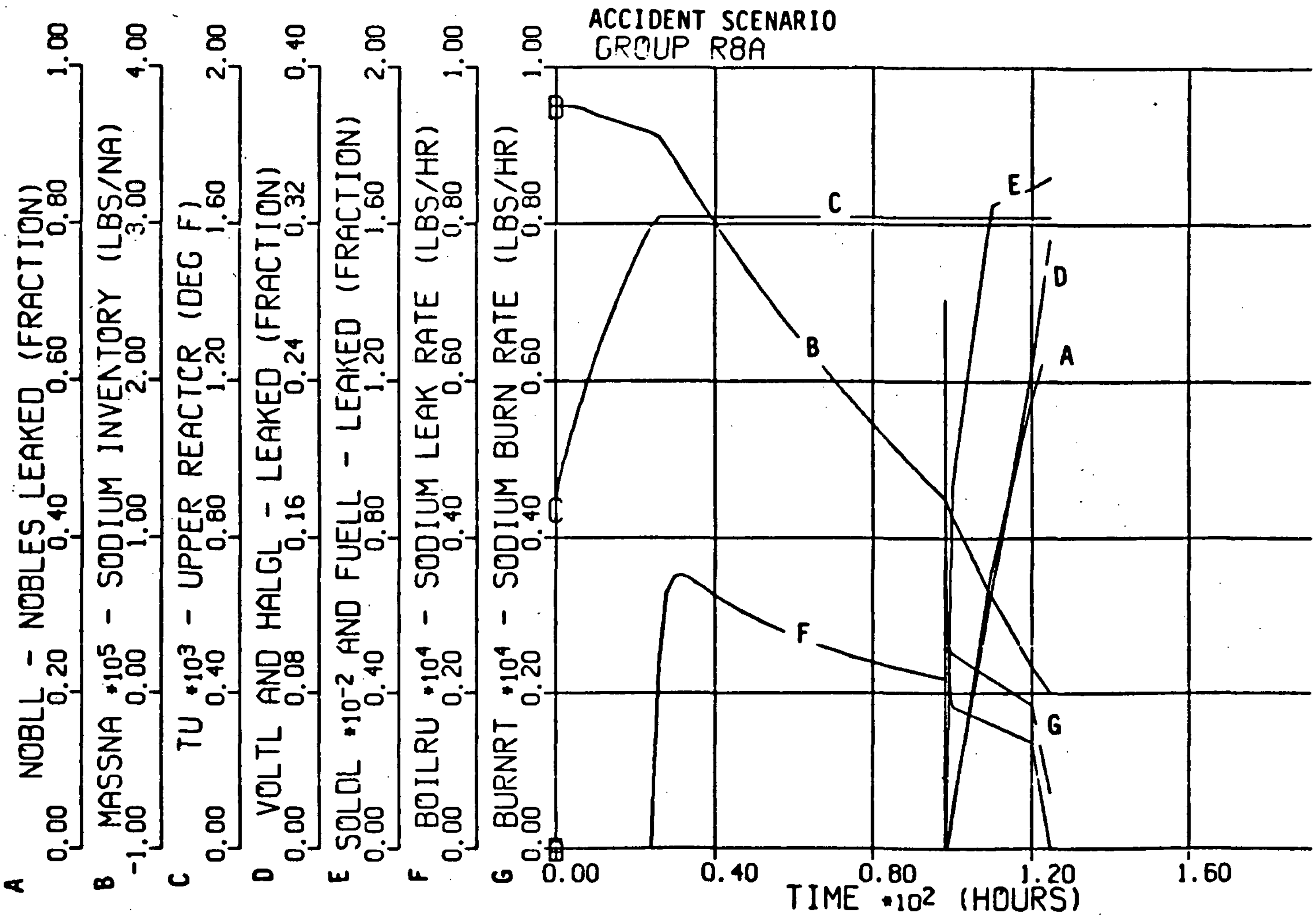


Figure A4.4-19 Accident Scenario for Containment Release - Category R8A

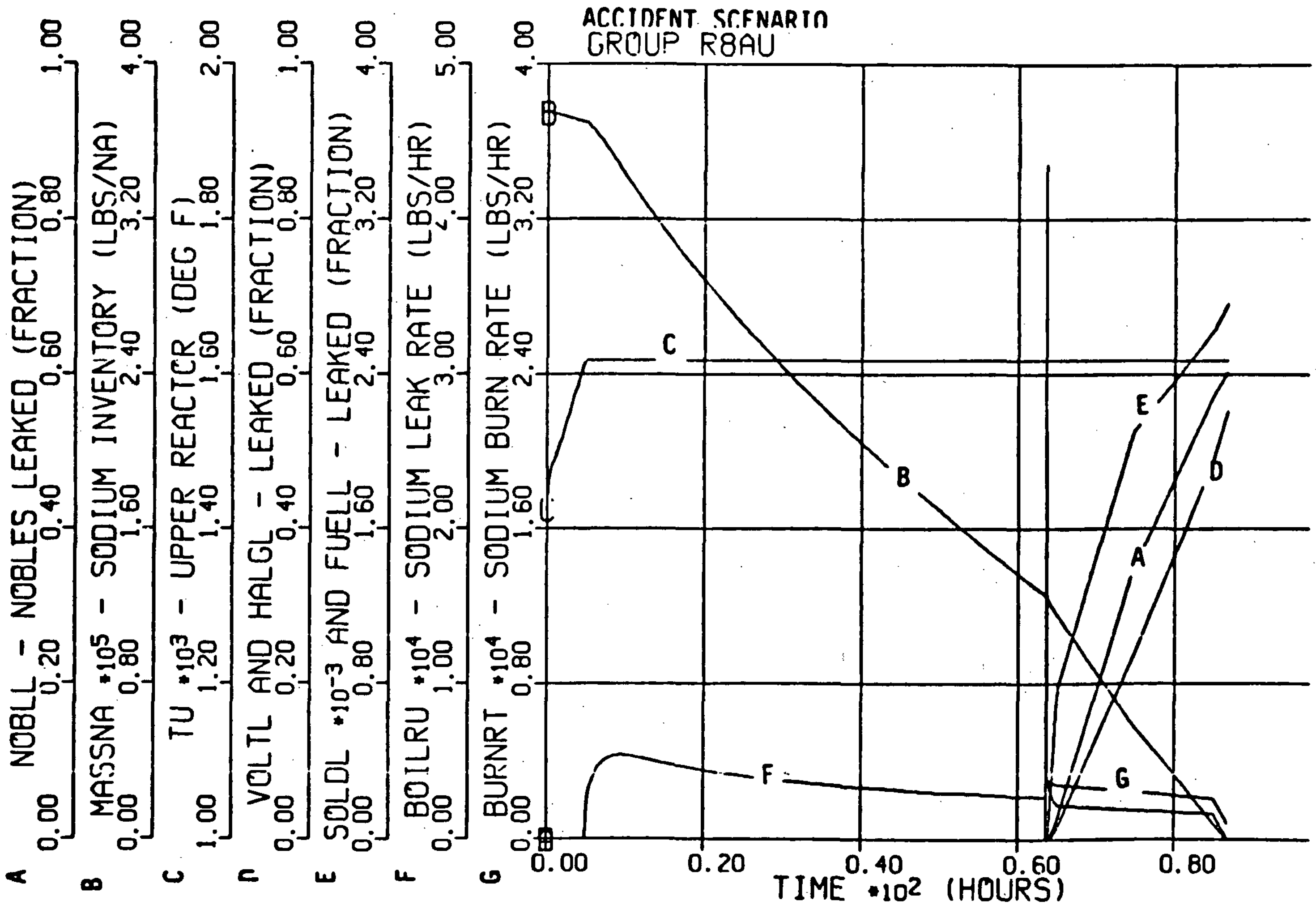


Figure A4.4-20 Accident Scenario for Containment Release - Category R8U

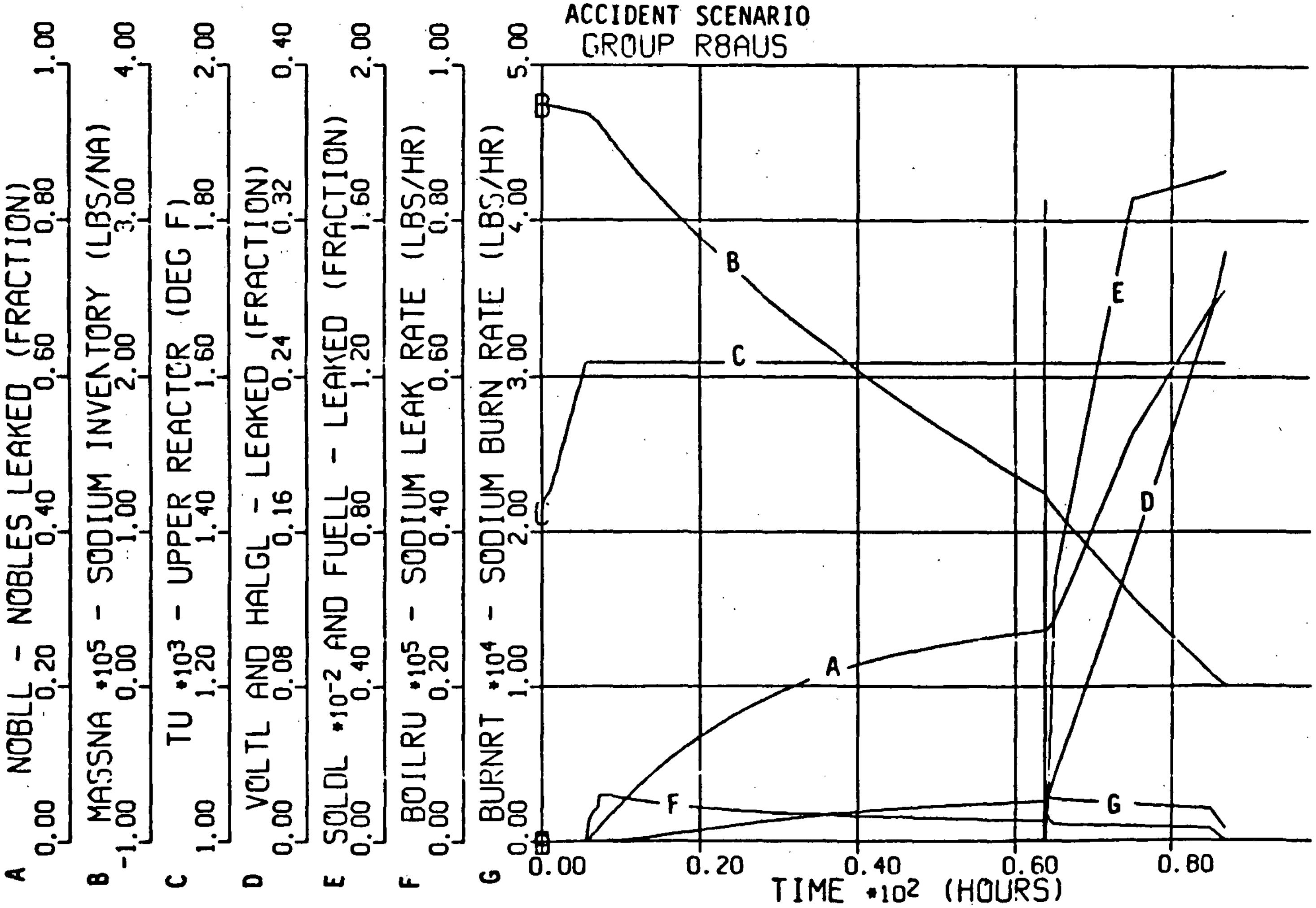


Figure A4.4-21 Accident Scenario for Containment Release - Category R8S

A4.5 Evaluation of Consequences

A4.5.1 Introduction

For each of the quantitative release descriptions obtained in Section A4.4 for the nine Containment Release Categories, several types of public risk measures were calculated using the MACCS code by Sandia National Laboratory. These risk measures include number of prompt fatalities, number of latent cancer fatalities, probability of prompt fatality to persons within one mile of the plant, and probability of latent cancer fatality to persons within 10 miles of the plant. Input data and assumptions needed to perform these calculations are described in Subsection A4.5.2. The calculational model and methods are discussed briefly in Subsection A4.5.3. The results are given in Subsection A4.5.4.

A4.5.2 Input Data and Assumptions for Consequence Calculations

<u>Source Terms</u>

The magnitude and timing of the release of each isotope group were those described in Section A4.4. These releases are described in terms of fraction of total core inventory. The absolute core inventory of each of a list of 54 isotopes was also provided as input. This inventory was calculated by use of the ORIGEN computer code for the FY86 PRISM reference metal core.

In addition to the nine releases calculated as described in Section A4.4 four additional releases were evaluated. These releases were chosen to test the sensitivity of the results to the fact that the releases were determined under the assumption of oxide fuel, and to test the effectiveness of terminating the releases by emergency action as at Chernobyl. Thus, in addition to releases R2A through R8S, public consequences were calculated for R4AM, R4AME, R6AM, R6AME. The suffix M indicating metal fuel and E indicating a release terminated early due to emergency action. The release descriptions for the metal fuel cases R4AM, R6AM were obtained by extrapolation from the corresponding oxide releases R4A and R6A as follows: first, the overall timing of the release was accelerated by a factor of 1.5 due to the addition of energy from the oxidation of the core in a fuel-concrete-water reaction. In addition, the total amount of fuel and solids released was increased to 15% under the assumption that the fuel-concrete reaction might produce a large fraction of aerosol-sized particles capable of being carried off with the sodium.

The emergency action cases R4AME and R6AME were obtained by truncating the R4AM release at 17 hours and the R6AM release at 50 hours.

Societal Response to Release

The first institutional response required for calculation of the consequences of release is the time at which the authorities order an evacuation. This point in time would vary depending on the sequence of events, i.e. on the Containment Release Category and is highly uncertain. For release category R2A a warning time of 1/2 hour is used. In R2A, R6S, and R8S a 25 % core meltdown has occurred at time zero; however, 15 min to 1 hr is required to melt through the vessels. Hence, the authorities may delay issuing the evacuation order, so 1/2 hour is used. In the case of R3 and R4A there has been an immediate energetic expulsion at time zero; thus there is no reason to doubt the occurrence of an accident, so a warning time of 0.3 hour is used. For all other release categories there is a gradual heatup and boil-off of sodium due to the loss of heat removal. This process takes 3 to 4 days, hence there is plenty of time for authorities to reach a proper understanding of the situation and to act in time.

A rule of thumb was used that the evacuation would be ordered when the scenario had progressed 0.6 of the time to core melt. This is 60 hours in the case of R6A and R8A, and 41 hours in the case of R6U and R8U. The next issue is the response of the public to the evacuation order. This is highly uncertain. However, a fit to actual experience with evacuation due to hazardous substance releases was obtained by using 3 population subgroups: (1) 30% delay 1 hour (2) 40% delay 3 hours, and (3) 30% delay 5 hours, then evacuate at 10 mph radially away from the plant. To simplify the calculations, the 5 hour delay was conservatively used for the whole population. In addition, two complete sets of calculations were performed, one using this 5-hour delay evacuation, and one in which none of the population evacuates. This was done to test the sensitivity of the results to evacuation assumptions.

Wherever possible, all calculations used NUREG-1150 assumptions (e.g., shielding factors and relocation criteria) for all site independent input. Thus, in all calculations non-evacuees projected to receive a groundshine dose of 25 Rem to bone marrow during the first seven days after plume passage were relocated at one day after plume passage; and evacuees projected to receive a groundshine dose to bone marrow of less than 25 Rem during the first seven days after plume passage (including any dose received before or during evacuation) were returned to their residences at one day after plume passage. At seven days after plume passage, any people still living at their residences were relocated, if they were projected to receive a groundshine dose to bone marrow or an inhalation dose to the lung (due to long-term resuspension) of 25 Rem in 30 years. Further, people who had evacuated or relocated according to any of the preceding criteria, were returned to their residences as soon as some combination of decontamination and temporary interdiction (to allow decay) decreased projected groundshine doses to bone marrow or resuspension lung doses to no more than 25 Rem in 30 years.

Site Data

The public consequences were calculated for the WASH-1400 Site 6 Eastern U.S. Coastal Site, which was also used for the GESSAR II risk assessment. The population distribution for this site is moderately above the average for actual U.S. reactor sites. The meteorological data used was that for New York City, which is known to have virtually the same weather as site 6.

A4.5.3 Calculation Methods

All of the consequence calculations were performed using the MACCS code by Sandia National Laboratory, Albuquerque. MACCS is a much improved version, soon to be released, of the U.S. Nuclear Regulatory Commission standard radiological consequence code CRAC2. Like CRAC2, MACCS propagates a Gaussian plume of radioisotopes over a system of radial coordinates having a population distribution specified. A statistical sampling technique is used to account for the effect of all possible weather conditions. The improvements in MACCS are: (1) an improved architecture facilitating uncertainty studies, (2) a multi-plume dispersion model that includes a multi-step crosswind concentration profile, (3) a dry deposition model that depends on particle size, (4) an improved wet deposition model that does not over-predict ground concentrations produced by rainout, and (5) expanded sets of environmental transport, dosimetry, health effect, and economic cost models.

In performing these calculations, one modification was made to MACCS; namely, the usual 10 hour limit on plume durations was removed in order to better approximate the spreading of long duration plumes. However, the total duration of release summed over all plumes was limited to about 50 hours in order not to throw away early dose as would result if the last plume had not left the computational grid before the seven day end of the early (emergency) phase was reached.

Due to the late changes in the NRC definition of the consequence, "probability of latent cancer fatality within 10 miles", this consequence measure had to be calculated by an approximate formula. The formula is:

$$\text{Probability of Latent Cancer Fatality within 10 miles} = \frac{\text{Total latent cancer fatalities}}{\text{Total Red Marrow Population Dose}} \times \frac{\text{Red Marrow Pop. dose w/in 10 miles}}{\text{Population within 10 miles}}$$

A4.5.4 Results of Consequence Calculations

The estimated public consequences given that a release occurs are presented in Table A4.5-1 for the case where evacuation is permitted to occur, and Table A4.5-2 for the case where no evacuation occurs. Table A4.5-1 shows that with evacuation permitted, none of the categories causes prompt fatalities (i.e. fatalities within one year). Latent fatality risk measures are essentially equal for all release categories except R3 energetic core disassembly with coolable debris, which is distinctly lower. This is expected since all release categories except R3 eventually release a similar total quantity of radioisotopes.

With the effects of metal fuel factored in, categories R4AM and R4AME do produce a fractional number (.02) of prompt fatalities. What this means is that, even for this most extreme case, some people receive a dose that is just above the minimum threshold for prompt fatality.

Table A4.5-2 shows that even preventing any evacuation typically produces only fractional prompt fatalities. The "no evacuation" case is clearly highly conservative since for most of the release categories the accident scenario is such that 3 to 5 days are actually available to make decisions and implement an evacuation.

The NRC risk measures for individual and societal risk for comparison to safety goals are calculated by multiplying the values in columns C and D of Table A4.5-1 by the probability of occurrence of the corresponding release category, then summing the results.

TABLE A4.5-1

CONSEQUENCES GIVEN EACH CONTAINMENT RELEASE CATEGORY
WITH EVACUATION

TYPE OF CONSEQUENCE

	A.	B.	C.	D.
Containment Release Category	Prompt Fatalities Given the Release	Latent Fatalities Given the Release	Probability of Prompt Fatalities <1 mile Given the Release	Probability of Latent Fatality <10 miles Given the Release
1. R2A	0	941	0	0.224E-3
2. R3	0	59	0	0.313E-4
3. R4A	0	1390	0	0.277E-3
4. R6A	0	1230	0	0.179E-3
5. R6U	0	1220	0	0.195E-3
6. R6S	0	1120	0	0.162E-3
7. R8A	0	1130	0	0.180E-3
8. R8U	0	1210	0	0.196E-3
9. R8S	0	1120	0	0.191E-3
10. R4AM	2.35E-2	1290	4.5E-5	0.188E-3
11. R4AME	2.17E-2	582	4.19E-5	0.85E-4
12. R6AM	0	924	0	0.134E-3
13. R6AME	0	16	0	2.3E-6

TABLE A4.5-2

CONSEQUENCES GIVEN EACH CONTAINMENT RELEASE CATEGORY
WITHOUT EVACUATION

TYPE OF CONSEQUENCE

Containment Release Category	A. Prompt Fatalities Given the Release	B. Latent Fatalities Given the Release	C. Probability of Prompt Fatalities <1 mile Given the Release	D. Probability of Latent Fatality <10 miles Given the Release
1. R2A	1.10	1000	1.98E-3	0.327E-3
2. R3	0.0014	86	2.67E-6	0.103E-3
3. R4A	7.17	1520	1.22E-2	0.552E-3
4. R6A	0.0050	1270	9.72E-6	0.244E-3
5. R6U	0.0038	1270	7.4E-5	0.276E-3
6. R6S	0.0041	1150	7.92E-6	0.273E-3
7. R8A	0.0149	1190	2.84E-5	0.298E-3
8. R8U	0.0588	1270	1.14E-4	0.318E-3
9. R8S	0.0093	1070	1.79E-5	0.280E-3
10. R4AM	124	3320	0.114	1.21E-3
11. R4AME	12	1180	0.016	4.29E-4
12. R6AM	53	2940	0.056	2.07E-3
13. R6AME	0	18	0	6.5E-6

APPENDIX B
TMI - RELATED REQUIREMENTS AND SAFETY ISSUES

APPENDIX B

TMI RELATED REQUIREMENTS AND SAFETY ISSUES

Appendix B

TMI Related Requirements and Safety Issues

NRC's policy on severe accident issues for future reactor designs (Reference B-1) requires (1) demonstration of compliance with the procedural requirements and criteria of the current Commission regulations, including the Three Mile Island requirements as reflected in 10 CFR 50.34f, and (2) demonstrations of technical resolution of all applicable Unresolved Safety Issues (USI) and the medium - and high priority Generic Safety Issues (GSI).

The applicability of the requirements and safety issues to the Liquid Metal Reactors (LMR) has been addressed in the "Liquid Metal Reactor Generic Safety Issues Safety Report" (Reference B-2). Reference B-2 also provides the LMR response to the safety issues. These responses are applicable to the PRISM design, however, there are instances where the resolution is design specific and thus, could not be fully resolved in Reference B-2. Reference B-2 identifies the Action Plan Item/Issues using the numbering scheme in the prioritization listing of NUREG-0933 (Reference B-3); this same identification system and the description of the issues have been retained in the following discussion.

Item A-47: Safety Implication of Control Systems

Description - Control system failures or malfunctions may accentuate the adverse consequences of accidents or transients. These failures or malfunctions may occur independently or as a result of an accident or transient and in addition to any control system failures that may have initiated the event. Although it is believed that control system failures are not likely to result in loss of safety functions that could lead to serious events, or result in conditions that safety systems are not able to cope with, in-depth studies have not been performed to support this belief.

PRISM Approach - The PRISM plant control system (PCS) is physically and functionally independent from the safety protection systems, structures, and components. The PCS equipment is physically separated from safety protection equipment, and the PCS does not have any direct physical interfaces with safety equipment. The PCS indirectly interfaces with safety systems via the Data Handling and Transmission System (DHTS) to monitor safety system status. The DHTS has physical interfaces with safety systems but these interfaces are not required for performing safety protection functions. All failure modes of the PCS and DHTS leave intact the safety features of the plant without compromising their reliability or their capability to meet safety requirements.

The Reactor Protection System (RPS) is designed to prevent the PCS from transmitting commands that interfere or countermand a safety command. During normal operation the PCS can freely operate the reactor within specified limits which are monitored by the RPS. Once a reactor trip signal is given, either by the operator or the RPS, the RPS rejects all commands from the PCS except for a request to enter the shutdown/ maintenance mode of operation.

Item C-5: Decay Heat Update

Description - Best estimates of decay heat data and associated uncertainties must be updated using the related work of research groups. This update can be incorporated in future revisions of the current regulations referring to ECCS performance.

PRISM Approach - The PRISM decay heat calculations incorporate the latest available nuclear data, and will be updated as newer data become available. The greatest uncertainties in the decay heat calculations are in the fission yields and the neutron cross-sections. For the PRISM metal fuel core, these are calculated using the ENDF/B-V nuclear data library [Reference: "ENDF/B Summary Documentation, 3rd Edition (ENDF/B-V)", edited by R. Kinsey, Brookhaven National Library Report BNL-NCS-17541 (ENDF-201), July, 1979]. This data base was generated under the direction of the Cross Section Evaluation Working Group sponsored by the U.S. Department of Energy.

Item B-56: Diesel Reliability

Description - Loss of onsite power events necessitate reliance on the onsite emergency diesel generators for successful removal of the decay heat. Emergency onsite diesel generators at operating plants have demonstrated an average starting reliability of about 0.94 per demand. The NRC's goal for new plants is a diesel generator starting reliability 0.99 per demand.

PRISM Approach - PRISM does not use diesel generators for emergency power, so this issue does not directly apply. No standby power is needed in PRISM to remove decay heat. The Reactor Vessel Auxiliary Cooling System (RVACS) uses naturally circulating outside air to dissipate all of the reactor's decay heat. RVACS performs its function without any human or mechanical action. Primary sodium flow through the reactor core is maintained by natural circulation. Upon loss-of-power to the reactor, the reactor is scrammed by release of the control rods. The control rod is normally held in place by an electromagnetic latch, which releases upon loss-of-power. A power-flow mismatch is avoided immediately after shutdown by the flow coastdown of the EM pumps; the coastdown of the EM pump is achieved by the inertia of a synchronous motor that will generate electicity for the EM pump as it coasts down.

Although standby AC power is not needed to cool the reactor, standby on-site power subsystems provide uninterruptable backup AC power for selected plant loads necessary to maintain an orderly shutdown and avoid equipment damage. Standby Class 1E DC power is provided by batteries for up to 12 hours. Two non-class 1E gas turbine powered generators provide emergency power for selected non-safety related loads for investment protection. The gas turbine powered generators are not required for decay heat removal and would only be needed to provide power for monitoring the plant status if the loss of off-site power continued beyond the 12 hour capacity of the batteries.

Issue 51: Proposed Requirements for Improving the Reliability of Open Cycle Service Water Systems

Description - Operating experience of open cycle service water systems (SWSs) has shown fouling by aquatic bivalves at approximately 45% of all sites. Some of the reported fouling events had serious impact on the reliability of the SWS. This system is the ultimate heat sink that, during an accident or transient, cools the reactor building component cooling water heat exchangers. In turn, these exchangers cool the RHR heat exchangers as well as safety-related pumps and are cooling coils. Fouling of the safety-related SWS either by mud, silt, corrosion products, or aquatic bivalves has led to plant shutdowns, reduced power operation for repairs and modifications, and degraded modes of operation. Improvements of surveillance and preventive maintenance programs at sites where bivalves are known to exist could significantly improve SWS reliability. The following related issues have been combined with the issue of whether or not the NRC Staff should develop requirements for improving the reliability of open cycle water systems: Issue 32, "Flow Blockage in Essential Equipment Caused by Corbicula", and Issue 52, "SSW Flow Blockage by Blue Mussels".

PRISM Approach - This issue is not applicable to the PRISM design. PRISM does not have a safety related service water system. PRISM uses the Reactor Vessel Auxiliary Cooling System (RVACS) to remove the reactor's decay heat; this system uses naturally circulated outside air as the ulimate heat sink.

Issue 79: Unanalyzed Reactor Vessel Thermal Stress During Natural Convection Cooldown

Description - This issue addresses a concern of potential generic safety significance relating to an unanalyzed reactor vessel thermal stress that could occur during natural convention cooling of PWR reactors.

PRISM Approach - The PRISM reactor vessel has been analyzed for the thermal stresses resulting from the most severe transient, which is the natural circulation cooldown resulting from the scram following the loss of all secondary cooling. Following scram and flow coastdown, the primary sodium flow through the reactor core is maintained by natural circulation. The decay heat generated in the core is removed by the primary sodium and transferred to the reactor vessel. The heat is in turn transferred to the containment vessel and then to an air flow stream which naturally circulates as it is heated.

Item HF01.1: Staffing and Qualification

Description - This item will address the following:

(a) The NRC will determine the minimum appropriate shift crew staffing composition. This determination will be made from developed personnel projection and allocation models and from evaluations of job and task analysis and probabilistic risk assessment data. Current staffing practice of both foreign and domestic utilities were surveyed to evaluate current practices, regulations and current staffing levels, considering such variables as plant size, control room arrangement and configuration, and plant layout. A rule for inclusion in 10 CFR Part 50, 50.54(m) (2) was prepared regarding licensed operator staffing. A review of SRP Section 13.1.3 which includes staffing will be developed.

(b) The need for engineering expertise on shift will be decided. This descision will be based in part upon the functions and duties required using the results of job/task analysis and evaluation of the current shift technical advisor experience. Consideration will also be given on how best to incorporate this expertise into the plant crew compliment. A proposed rule for 10 CFR 50 has been prepared and a final policy statement on the inclusion of engineering expertise on shift has been developed.

PRISM Approach - The PRISM concept has been developed with a goal to achieve an inherently safe plant. This inherent safety was accomplished using multiple small reactors. Extensive automation has been incorporated in the design to enable a minimum operating staff to supervise control of the plant and its multiple reactor modules. A high degree of redundancy has also been incorporated in the control system to reduce the need for the operator to assume control of the plant or portion of the plant as a result of a single malfuntion.

These control systems and plant design features place different requirements on the plant staff than are placed on the current generation of LWR's. Since the minimum staffing levels for current generation LWR's have been included in 10 CFR 50, it is anticipated that a rule change will be necessary to reflect the reduced staffing requirements expected for the PRISM concept.

Item HF01.2: NPP Personnel Qualifications Requirements

Description - This item will address the following:

(a) The minimum training, education and experience requirements for shift operating crews will be determined from a review of job and tasks analysis data. The relationship between education, training, and experience will be assessed and the trade-offs among these related factors determined. A rule for 10 CFR 50 will be prepared on minimum crew qualifications.

(b) A study will be done of the feasibility and value of licensing or certifying nuclear power plant personnel other than operators. A rule on degree requirements for the operating staff will be prepared.

Prism approach - As with the PRISM approach for Item HF01.1, the personnel qualifications for PRISM may differ significantly from those required for the current generation of LWR's. The personnel qualifications must be addressed along with the staffing requirements in the anticipated rule change actions for PRISM.

REFERENCES

B-1. Federal Register, Vol. 50, No. 153, p. 32138, August 8, 1985.

B-2. "Liquid Metal Reactor Generic Safety Issues Safety Report," EPRI/DOE - CoMO - 001, May, 1986.

B-3. Emrit, R. and W. Minners, "A Prioritization of Generic Safety Issues," NUREG - 0933, revised June, 1985.

APPENDIX C
DESIGN CONSIDERATIONS REDUCING SABOTAGE RISK

APPENDIX C

DESIGN CONSIDERATIONS REDUCING SABOTAGE RISK

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APPENDIX C

DESIGN CONSIDERATIONS REDUCING SABOTAGE RISK

C.1 Introduction

This appendix provides a description of the PRISM design features that reduce the risk from postulated acts of sabotage. Features are described which both inhibit sabotage and prevent the level of damage which could lead to a release of radioactivity in excess of the site suitability source term specified for 10CFR100 evaluations.

The basic required plant safety functions for either transient or sabotage initiated events are those to shut down the reactor, maintain core cooling, remove decay heat, and control radionuclide release.

Reducing sabotage risk has been a long-standing and continuing objective for energy-producing as well as other industries. Radiological sabotage, as defined in 10CFR73.2, is the major concern addressed by this appendix. Radiological sabotage is differentiated from war-related or subversive-type sabotage principally by the degree of damage that may be inflicted, i.e., rocket or armored vehicle attacks or damage from high level explosives are sabotage initiators beyond the scope of this appendix.

This appendix considers acts of sabotage which may be postulated to be the work of a single individual or a group of individuals and may be committed by a person on the power plant staff (insider) and/or by other individuals (outsider). Both the insider and outsiders are assumed to be well trained, capable and determined to accomplish their goal within limitations of the design-basis threat as defined in 10CFR73(a)(1).

A plant vulnerability assessment was performed in which radiological sabotage and the theft of Strategic Nuclear Materials were assessed. Adversary actions which might cause serious radiological consequences were identified. Vulnerable points within the plant were also identified, and the design was revised to eliminate or reduce the vulnerabilities. Where vulnerabilities could not be totally eliminated, security measures were introduced to protect the systems involved.

C.2 PRISM Sabotage Inhibitors

The PRISM design incorporates a multiple approach to minimize the risks from radiation sabotage. The inherent design margins limit the public risk from beyond-design-basis events. The design of the plant has integrated consideration of nuclear plant safety and physical security to address the threat of sabotage and minimize interferences with reactor safety, operations, and maintenance. The physical security system provides detection, assessment and delay against the outsider threats of sabotage and restricts movement and access within the plant, particularly access to the nuclear island and fuel facilities. PRISM also uses advanced control and protection systems design with a high degree of automation, redundancy, and self-diagnosis; the fault tolerant design is capable of recognizing and adapting to failure of its own (and other) elements while continuing to maintain its designed level of system performance.

C.2.1 Plant Protection Features

The conceptual design of the plant protection features is described in a Safeguards and Security report to be submitted under separate cover. The Safeguards and Security report is withheld from public disclosure pursuant to Section 2.790(d) 10CFR2, Rules of Practice. The PRISM Safeguards and Security report addresses the following:

a. Physical barriers
b. Access requirements
c. Detection aids
d. Communication requirements
e. Response requirements

C.3 Summary and Conclusions

This appendix and the separately submitted Safeguards and Security report provide a description of the PRISM design features that inhibit postulated acts of sabotage. The four-level security system, plus the inherent safety characteristics of the PRISM design, provide the means for inhibiting and mitigating postulated acts of sabotage. A vulnerability analysis has verified the effectiveness of these means against the design basis threat. The PRISM design assures the public health and safety against postulated acts of radiological sabotage.

APPENDIX D
PRISM DUTY CYCLE EVENT DESCRIPTIONS

APPENDIX D

PRISM DUTY CYCLE EVENT DESCRIPTIONS

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D 1.0

D1.0 Introduction

This appendix specifies the steady state and operational transients which will be considered in evaluating and analyzing the structural design of the systems and components of the nuclear steam supply system (NSSS) for the PRISM plant.

The duty cycle events are based on consideration of the duty cycles for CRBRP, FFTF, PLBR, LSPB and operating PWR plants. The descriptions of the events are based on the planned operational strategy for the nine module PRISM plant and previous analysis conducted for FFTF, CRBRP, PLBR, LSPB or for the system support of DOE's large component development program. The selected events are representative of conditions which are considered to occur during plant operation and which are sufficiently severe or frequent to be of possible significance to the cyclic behavior of plant components. The events described herein are based on best estimate assumptions; they are meant primarily for use in component stress analysis and do not necessarily represent actual plant operation. The event frequencies are selected on a conservative basis, guided by consideration of the operational objectives. The transient analysis of these events, when used as a base for conservative component structural design, will provide confidence that the component is appropriate for its application over the design life of the plant.

The plant consists of nine NSSS modules connected to three turbine-generators each with its own BOP. The duty cycle is specified for a one-loop NSSS module used in this nine NSSS module plant. The steam generating system will have one evaporator, one recirculation pump, and one steam drum in each module. One, two or three NSSS modules are capable of supplying steam to a particular turbine and BOP. Full power turbine operation requires the three associated reactor modules to be available.

PRISM duty cycle events are described in terms of ASME Section III loading categories, namely Levels A, B, C and D service limits. As such, they are directly applicable to ASME Section III systems and components. For other design basis criteria such as ASME Section VIII and ANSI B31.1, the selection of appropriate design conditions and load combinations are based on these duty cycles. Alternative criteria developed for application of Level D type loading conditions to ASME Section VIII and ANSI B31.1 is the responsibility of the designer.

D 2.0

D2.0 Plant Operation

Although it is anticipated that the plant will operate as a base-loaded plant, it will be capable of part-load operations during its sixty (60) year design life. Each power block will be capable of loading and unloading from 25% to 100% of rated power. The design equivalent availability factor is 85%.

The systems and components shall be designed for a service life of 60 years comprised of the following operating conditions:

Condition	Hours	Years
100% Power Operation	467,230	53.3
70% Power Operation	2,630	0.3
25% Power Operation	3,500	0.4
Hot Standby, 550°F	8,770	1.0
Refueling & Maintenance, 400°F	43,830	5.0
Total Life	525,960	60.0

D 3.0

D3.0 Level A Service Limits (Normal Conditions)

D3.1 Definitions

According to the ASME Code*, Level A Service Limits are all loadings

to which the components may be subjected in the performance of their specified service function. These limits were formerly referred to as Normal Conditions which were "any condition in the course of system start-up, operation in the design power range, hot standby, refueling and system shutdown, other than Upset, Emergency, Faulted or Testing Conditions."

D3.2 Events

<u>A-1a - Dry System Heat-up, Sodium Fill, Heat-up to Refueling Temperature</u>

For design purposes, the heat-up of the entire sodium system, exclusive of the steam generators, of an NSSS module will be treated as a temperature increase at the outer surface of the sodium boundary from ambient, 70°F, to 450°F at a constant rate of 5°F/hr. After a soak at 450°F surface temperature to preheat the internals to a nominal 400°F, the surface will be allowed to cool to 400°F.

Prior to the heat-up cycle, there will be three cycles of sodium side pressure reduction to nearly full vacuum and back filling with either helium in the primary heat transport system (PHTS) or argon in the intermediate heat transport system (IHTS) to one atmosphere. After the heat-up cycle, there will be one pressure cycle from ambient to maximum attainable vacuum with slow back filling to one atmosphere using either

*ASME Boiler and Pressure Vessel Code, Section III, "Rules for Construction of Nuclear Power Plant Components," 1983 Edition, Summer 1985 Addenda, American Society of Mechanical Engineers, New York, NY.

argon or helium. The water side of the steam generators will be filled with nitrogen at atmospheric pressure prior to heat-up. All piping and components containing sodium will be heated by electrical heaters mounted external to the piping or component, as applicable. The steam generator system (SGS) will be heated from the water side using steam from an auxiliary boiler. Following the heat-up, the PHTS and IHTS are filled with 400°F sodium.

It is specified for design purposes that this event will occur a total of 1 time for the entire module and 12 times for the IHTS.

<u>A-1b - Cooldown from Refueling, Sodium Drain, Dry System Cooldown</u>

The sodium systems are drained and back filled with argon to one atmosphere prior to cooldown below 400°F. The cooldown will be considered as a decrease from 550°F to 70°F. For design purposes, the cooldown rate will be 10°F per hour.

The number of cooldowns will be equal to the number of heat-ups given for Event A-1a.

<u>A-2a - Start-up from Refueling Temperature to Hot Standby Conditions</u>

The plant start-up event from refueling temperature is a heat-up transient between the normal refueling temperature of 400°F and the temperature conditions that exist at hot standby (550°F). For design purposes, the primary sodium temperature shall increase at a maximum average rate of 50°F/hr between 400°F and 550°F. This heat-up rate will be achieved by utilizing the sodium pumps at 100% flow, the SGS recirculation pumps, and reactor power. This change in temperature shall be accomplished by using the plant control system at a power ramp rate slow enough to assure not exceeding the temperature rate.

For design purposes, it is specified that this event shall occur 60 times during the 60-year design life.

A-2b - Automatic or Manual Startup from Hot Standby Conditions to 25% Power

The plant start-up event from hot standby conditions is an automatic controlled heat-up transient from the reactor outlet hot standby temperature of 550°F to the temperature and flow conditions which exist at 25% thermal power. This event is the action to either return the plant to operation following a reactor trip or from refueling. The 25% power level is the lower end of the normal power operating range.

The reactor power is gradually increased at a rate such that the primary and intermediate sodium temperatures increase at a maximum average rate of 3°F per minute, and with the sodium pumps operating at 100% flow. During this heat-up the evaporator outlet temperature will increase to saturation conditions. Steam from the drum will be used for warming up the main steam line and the turbine bypass line if the turbine is not operating in conjunction with other NSSS modules. If the turbine is operating, the reactor power will be increased to 25% in preparation for the switch to power range automatic control.

The frequency of this event over the 60 year design life is 573 times.

A-2c - Automatic Turbine-Generator Warm-up, Roll and Loading to 8% Load

Following start-up of the NSSS module to 25% power with the generation and bypass of steam, the operator can initiate an automatic turbine-generator warm-up, roll and loading to 8% load. The main steam line to the turbine will be warm because of the previous steaming. The turbine will begin to roll and electrical load will be placed on the turbine-generator following automatic synchronization. The turbine-generator load will be increased to 25% of the normal electric power output per module.

The frequency of this event over the 60 year design lifetime is 573 times.

A-3 - Normal Shutdown

A-3a - Shutdown from Hot Standby Conditions to Refueling Temperature

The temperature reduction event to refueling temperature from hot standby conditions is a cooldown transient from hot standby conditions to the normal reactor refueling temperature of 400°F. This event is essentially Event A-2a reversed in sequence. The IHTS sodium pumps will be run at pony motor speed during the cooldown between operating temperatures and 400°F. Decay heat removal shall be through the SGS, RVACS or the ACS in combination. For design purposes, it is expected that this event occurs 60 times during the 60-year design life.

A-3b - Shutdown from 25% Power to Hot Standby Conditions

The plant shutdown event to hot standby conditions is a cooldown transient from 25% power using the normal control system to the hot standby isothermal temperature of 550°F. This event is essentially Event A-2b reversed in sequence with the reactor taken subcritical when the primary hot leg temperature has been reduced to slightly above 550°F. Decay heat removal will be through either the SGS or the ACS, and the RVACS in combination. The IHTS pump pony motor will be used during decay heat removal.

For design purposes, it is expected that this event occurs 60 times during the 60 year design life.

A-4 - Weekly Loading and Unloading

The plant weekly loading and unloading events are represented by a power change between 25% load and 100% load at an average rate of 1% of rated power per minute. The power change can be made up of incremental changes of up to 10% of rated power at 2% of rated power per minute. Load changes in this region are accomplished by varying reactor power while holding sodium flow rates, turbine throttle inlet pressure and temperature constant.

For design purposes, this event is assured to occur 2880 times unloading and 2880 times loading during the 60 year design life.

A-5 Daily Loading and Unloading

The plant daily loading and unloading events are represented by a power change between 50% and 100% load at an average rate of 0.417% of rated power per minute. The power change can be made up of incremental changes of up to 10% of rated power at 2% of rated power per minute. Load changes in this region are accomplished by varying reactor power while holding sodium flow rates, turbine throttle inlet pressure and temperature constant.

For design purposes, this event is assumed to occur 14400 times unloading and 14400 times loading during the 60 year design life.

A-6 - Steady-State Temperature Fluctuations

This event consists of the sodium temperature variations produced by power fluctuations within the plant control system dead bands. This fluctuation is taken to be ± 12°F for the primary and ± 12°F for the intermediate systems and is based on expected dead band fluctuations.

Since the system is not expected to exhibit major temperature variations within the control dead band, the frequency for this event is considered to be conservative.

For design purposes for the 60 year design life, it is specified that this event occurs 1.0×10^7 times.

A-7 - Steady-State Flow Induced Vibrations

This event consists of the vibrations in the system produced, for example, by TBD fluctuations in sodium pressure due to the interaction between the vanes in the impellers and the turning and diffusion vanes in the pumps.

For design purposes during the 60 year plant design life, this event is specified to occur TBD times.

A-8 - Module Out of Service

The power block may be operated at a reduced power level (67% or less) with one or two NSSS modules out of service for extended periods of time. This will be accomplished by a method which remains to be determined. The inactive NSSS module or modules are assumed to be at a temperature of 400 or 550°F.

The power block is assumed to experience three months of two NSSS module operations each year. For each NSSS module, one month of inactive and two months of active operation during the two NSSS module operating period is assumed.

A-9 - Step Load Increase or Decrease of 10% of Rated Power

A ±10% "step" change (defined to be a fast ramp at 60% of rated power per minute) in load demand is an assumed maximum load transient due to disturbances in the electrical network into which the plant output is tied. The control system is designed to maintain plant operating conditions without reactor trip following a ±10% step change (fast ramp) in plant load demand in the range between 25% and 100% full load. In effect, during load change conditions, the control system maintains steam pressure at the turbine throttle by automatic control operations on reactor temperature and power.

For design purposes during the 60 year design life, it is specified that this event occurs 1500 times for the step increase and 1500 times for the step decrease.

A-10 - Turbine Steam Inlet Valve Testing

A functional test of the turbine steam inlet valves will be performed on a weekly schedule while the unit is carrying load. The purpose of this test is to ensure proper operation of the main steam throttle valves, governor valves, reheat stop valves, and interceptor valves. These vital control devices might remain motionless throughout long periods of steady-state operation and develop otherwise undetected failures.

For design purposes, it is expected that this event occurs 2655 times during the 60 year design life. It should be noted that since this testing would be performed in conjunction with the unloading to 25% power, the frequency for unloading to 25% should not be increased.

A-11 - Fast Ramp Load Changes of 25% of Rated Power

The fast ramp load change of 25% of rated power occurs within the plant power range of 25% to 100% power. It consists of a power ramp of 10% of rated power at 10% of rated power per minute followed by an additional power ramp of 15% of rated power in the same direction at 5% of rated power per minute. During the load changes, the plant control system maintains steam pressure at the turbine throttle by automatic control operations on reactor temperature and power.

For design purposes, this event is specified to occur 30 times increasing and 30 times decreasing during the 60 plant design life.

D4.0 Level B Service Limits (Upset Conditions)

Unless otherwise stated, events that result in a module, power block or plant trip are assumed to start at 100% power operation and terminate at hot standby conditions.

D4.1 Definitions

According to the ASME Code, Level B Service Limits are all specified loadings which the component or its support must withstand without damage requiring repair. These were previously considered Upset Conditions which were defined as "any deviations from Normal Conditions anticipated to occur often enough that design should include a capability to withstand the conditions without operational impairment. The Upset Conditions include those transients which result from any single operator error or control malfunction, transients caused by a fault in a system component requiring its isolation from the system and transients due to loss of load or power. Upset Conditions include any abnormal incidents not resulting in a forced outage and also forced outages for which the corrective action does not include any repair of mechanical damage. The estimated duration of an Upset Condition shall be included in the Design Specifications."

D4.2 Events

<u>B-1 - Reactor Trip</u>

This transient includes anticipated trips due to malfunctions (including rapid reactivity transients) which produces a reactor protection system (RPS) trip (caused by a PCS investment protection trip level being exceeded), as well as spurious trips covering those situations in which a PCS trip level is not actually exceeded but a trip occurs due to a fault in the control system or the plant instrumentation. This transient also includes manual activation by a plant operator. The reactor trip involves only one of the operating NSSS modules. The other operating NSSS modules and associated turbine-generator continue to operate.

B-1a - Reactor Trip from Full Power with Maximum Decay Heat

This transient involves a trip of a single reactor module at time = 0.0. The control rods are released at time = 0.2 seconds, with full insertion in 2 seconds. The RPS then initiates the tripping of the primary and intermediate sodium pumps at t = 0.5 seconds. The sodium pumps coastdown to pony motor flows over a period of approximately two minutes. The turbine continues to operate at reduced load from the remaining operating modules. The normal feedwater system continues to operate. The feedwater flow to the steam drum is maintained by using the smaller start-up control valve from the normal feedwater source.

The initial decay heat level for this transient is the decay heat which is associated with a core in operation for a significant time with allowances for uncertainties.

For design purposes, this event is specified to occur 150 times during the 60 year design life.

B-1b - Reactor Trip from Full Power with Minimum Decay Heat

The same operational sequence of Event B-1a is assumed for this transient. The initial decay heat level for this transient is the decay heat which is associated with a core in operation for a short duration with allowances for uncertainties. This event results in smaller temperature differentials at the end of the transient.

For design purposes, this event is specified to occur 75 times during the 60 year design life.

B-1c - Reactor Trip from Partial Power with Minimum Decay Heat

The same operational sequence is assumed for this transient as for a reactor trip from full power. Initial power level is 40% and the initial decay heat level is the decay heat associated with 40% power operation with allowances for uncertainties.

For design purposes, this event is specified to occur 75 times during the 60 year design life.

B-2 - Uncontrolled Control Rod Movement

This is a general category of events which result from control system malfunctions. The Event B-2 category includes three events: an uncontrolled rod insertion from full _T initial conditions and two rod withdrawal cases. These events are identified in the duty cycle to provide assurance of their consideration in the overall transient analysis task and as a basis for the determination of plant protection system requirements.

B-2a - Uncontrolled Rod Insertion

A single rod is inserted at a rate which causes a TBD% per second reduction in thermal power due to an assumed malfunction of the controller on that rod. (This event is not to be confused with a rod drop, which is an unlatching of the rod resulting in a free fall of the control rod.) The sodium flow rates are constant during this event. The steam flow decreases because of lower energy input and the feedwater flow is controlled to follow steam flow. The turbine flow control valve maintains the turbine pressure. It is assumed that this event occurs when full system _T's are present. The thermal power level at the beginning of the transient is 100%. A PCS requested trip of the affected module is assumed to be initiated by the flux-delayed flux function.

For design purposes, this event is specified to occur 15 times during the 60 year design life.

B-2b - Uncontrolled Rod Withdrawal from Startup with Automatic Trip

The initial conditions for this event are hot standby with the initial decay heat for the transient is the decay heat which is associated with a core in operation for a short duration with allowances for uncertainties. Primary and intermediate pumps are operating at 100% flow. Uncontrolled withdrawal of one control rod at 2_/sec. then occurs. During the withdrawal, all sodium flows remain at initial values. Affected NSSS module reactor trip is initiated by the flux-delayed flux function.

For design purposes, this event is specified to occur 3 times during the 60 year plant design life.

B-2c - Reactor Loading at Maximum Rod Withdrawal Rate

From initial plant conditions of 25% reactor thermal power, 100% sodium flow, and ∇25% electrical output, the plant supervisory controller requires the plant to increase in load. During the rod withdrawal, a mechanical malfunction in one reactor module results in maximum mechanical rod withdrawal speed. Affected NSSS module reactor power increases from 25% at a rate determined by the reactivity rate, TBD _/sec.

The turbine increases output by picking up the additional steam flow. Feedwater flow will be a function of steam flow. A PCS requested trip of the affected module is assumed to be initiated by the flux-delayed flux function.

For design purposes, this event occurs 3 times during the 60 year plant design life.

B-3 - Complete or Partial Loss of One Primary Pump Flow

There are two events in this category: partial loss of flow in one primary pump and the total loss of power to one primary pump.

B-3a - Partial Loss of Primary Pump Flow

The primary flow in one pump is assumed to decrease from 100% to a TBD level due to a ramp down in pump voltage. The voltage in the unaffected primary sodium pumps remains at their initial values. No action is taken to terminate the event for 10 minutes.

This transient provides an envelope to encompass control malfunction and operator errors causing mismatches in the primary pump flows. The transient will result in an increased reactor outlet temperature and a redistribution of temperatures within the IHXs in the affected module.

For design purposes, this event is specified to occur 3 times per pump during the 60 year plant design life.

B-3b - Loss of Power to One Primary Pump

The primary pump voltage in one module is assumed to decay to zero. The other primary pumps are assumed to remain at full voltage. The intermediate pump flow remains at initial values until reactor/pump trip. Affected reactor module trip is initiated when the ratio of primary to intermediate pump flow is less than approximately 0.85. Following the reactor trip, the remainder of the pumps and the steam/water side respond as for normal reactor trip, Event B-1a.

This transient provides an envelope to encompass those events that would cause the pump to be tripped or those which result from control failures more severe than those in Event B-3a or from significant operator errors in controlling primary flow.

For design purposes, this event is specified to occur 3 times per pump during the 60 year plant design life.

B-4 - Complete or Partial Loss of Intermediate Pump

There are two events in this category: a partial loss of intermediate flow in one NSSS module, and the coastdown of the intermediate pump to pony motor speed.

B-4a - Partial Loss of Intermediate Pump

The intermediate flow in one NSSS module is assumed to drop from 100%

flow to 85% flow (a level immediately above the primary to intermediate pump flow mismatch nominal trip setting). All other flows remain at their initial values. No action is taken to terminate the event for 10 minutes. The event is characterized by an increase in intermediate cold leg and primary hot leg temperature.

For design purposes, this event is specified to occur 3 times per pump during the 60 year plant design life.

B-4b - Loss of Power to Intermediate Pump

The intermediate pump in one NSSS module is assumed to coast down to pony motor speed. The primary pump flows remain at their initial values until the reactor/pump trip. A reactor trip is initiated when the primary - intermediate flow mismatch reaches the nominal setting. Following the trip, the remainder of the pumps and the steam/water side are treated as for the normal reactor trip, Event B-1a.

For design purposes, this event is specified to occur 3 times per pump during the 60 year plant design life.

B-5 - Reduction or Loss of Feedwater Flow

B-5a - Trip of One Feedwater Pump

The steam plant is assumed to include 3 - 33 1/3% capacity (of full flow) motor driven feedwater pumps with runout capability to 40% of full flow. Upon loss of one operating feed pump, the other pumps will run out on their head-flow curves. It is intended that this runout capability with reduced load demand will result in no reactor trip.

For design purposes, this event is specified to occur 45 times during the 60 year plant life.

B-5b - Loss of Normal Feedwater Flow To All Steam Generators Supplying One Turbine

This transient includes three cases: (a) loss of one feedwater pump with failure of its outlet check valve: (b) loss of feed pump suction, for all pumps and (c) closure of all the feedwater control valves or feedwater isolation valves. A reactor trip for all affected modules will be initiated on the intermediate IHX inlet temperature trip function after the steam generator dryout. Following plant trip decay heat will be removed by the RVACS heat removal system plus the ACS using the IHTS pony motors.

For design purposes, this event is specified to occur 8 times during the 60 year plant design life.

B-6 - Intermediate Pump Pony Motor Failure

Following a normal plant trip, Event B-1a, the pony motor on the intermediate pump fails to operate. The affected pump coasts down and stops. The transient is characterized by a temperature increase in the cold leg of the primary circuit of the affected NSSS module, and a long term mismatch in primary to intermediate system flow in the affected NSSS module since the intermediate system flow in the affected NSSS module will be provided by the natural circulation driving head only. Following plant trip decay heat will be removed by the RVACS and through natural circulation in the IHTS/SGS.

For design purposes, this event is specified to occur 3 times per module during the 60 year plant design life.

B-7 - Inadvertent Water-Side Isolation and Blowdown of the Steam Generator

The event is assumed to be initiated by one of the following: (a) inadvertent operator action, (b) inadvertent activation caused by sodium/ water reaction system instrumentation or equipment failure, or (c) operator response to a false water to sodium leak indication. This transient results in the water-side isolation and dumping of the steam generator in an individual module. The event terminates at refueling conditions for the affected module.

This event is initiated by a signal which is assumed to instantaneously close the normally open isolation valves in the feedwater inlet line and the steam outlet line from the drum. Simultaneously, the dump valves in the water-side inlet line of the affected evaporator and the power relief valves on the steam drum are assumed to open. The steam/water side pressure decreases until the power relief and water dump valves shut. Affected NSSS module reactor trip occurs due to a high intermediate IHX sodium inlet temperature. Decay heat removal is maintained through the RVACS and the ACS using the IHTS pony motors.

For design purposes, this event is specified to occur 3 times per module in the 60 year plant design life.

B-8 Loss of Feedwater to One Module

This event is assumed to result from inadvertent closure of the main feedwater control valve in one module. The event is characterized by an increase in water inlet temperature to the evaporator as the temperature of the recirculation water increases toward the saturation temperature. The recirculation flow continues. A reactor trip is initiated on the steam-feedwater flow mismatch control function or on low water level in the steam drum. Following the reactor trip feedwater flow is supplied through the smaller start-up valve which is assumed to function properly. Therefore, decay heat can be removed by operation of the steam generator, ACS and RVACS in parallel.

For design purposes, this event is specified to occur 6 times per module during the 60-year design life.

B-9 - Feedwater Control Valve Failed Open

This event assumes that the feedwater control valve for one steam generator fails in the full open position with the plant at the 25% power (chosen because it envelopes failures at 100% and represents the lower limit of the normal load range). The affected NSSS module reactor will be tripped based on a high steam drum water level control function or a low steam/feedwater flow ratio control function.

For design purposes, this event is specified to occur 6 times per module during the 60 year plant design life.

B-10 - Turbine Trips

B-10a - Turbine Trip Without Immediate Reactor Trip

This event assumes that a turbine is tripped from full power (turbine stop valves close instantaneously). A 60% maximum steam dump bypass system is provided to the condenser which is capable of 60% bypass. The remainder is provided by atmospheric dump capability. Normal steam drum pressure is maintained. Steam flow to all closed feedwater heaters is terminated with closure of the turbine throttle valves. The affected reactor modules are not tripped coincident with the turbine trip. The reactor modules will take a step power demand reduction of 10% followed by a 2% per minute ramp to bring the module power within the steam bypass and condenser capacities. Then the module will operate at constant power until the cause for the turbine trip is determined. Based on the cause of the turbine trip, the reactors will either be shutdown or the plant will be returned to 100% power when the turbine is returned to service.

For design purposes, this event is specified to occur 105 times during the 60 year plant design life.

B-10b - Turbine Trip with Reactor Trip (Loss of Main Condenser or Similar Problem)

Turbine trip is assumed to occur in conjunction with a loss of the main condenser, and thus the turbine dump (bypass) is unavailable. This causes the main steam flow to decrease to zero initially. The steam system pressures then increase and the evaporator and steam drum power-operated relief valves open, returning the steam flow to about 100%. Affected reactor modules will trip coincident with turbine trip based on loss of condenser vacuum. The sodium pumps coast down and the steam flow and pressure is reduced. The transient in sodium temperature is similar to the reactor trip from full power, Event B-1a.

For design purposes, this event is specified to occur 14 times during the 60 year plant design life.

B-11 - Loss of All Off-Site Power

The loss of all off-site power is assumed to occur. It is assumed that no plant trip occurs, with continued affected power block operation. Affected NSSS modules take a step power demand reduction of 10% of rated power followed by a 2% per minute ramp to bring the NSSS module within the steam bypass and condenser capacities. Then the NSSS module will operate at constant power until the cause is determined and corrected. The main turbine auxiliary circuit remains operational supplying the unit load. After resynchronization to the grid the turbine is reloaded to 100% power in 20 minutes.

For design purposes, this event is specified to occur 8 times during the 60 plant design life.

B-12 - Turbine Bypass Valve Openings

B-12a - Inadvertent Opening of One Turbine Bypass Valve

From a given power operation, it is assumed that one turbine bypass valve (rated at 7.5% of full loop steam flow) is fully opened. The bypassing of the steam results in decreased steam flow to the turbine generators. The excess steam flow results in a decrease of steam pressure at the main steam header. The turbine throttle valve closes slightly to maintain turbine inlet pressure. Electrical power generation will be reduced as a result of the reduced turbine inlet steam flow. The reactor operating conditions will be returned to the previous steady state conditions in a relatively short period of time.

For design purposes, this event is specified to occur 3 times during the 60 year plant design life.

B-12b - Turbine Bypass Valve Fails Open Following Reactor Trip

This transient is included as it is representative of the transients that can rapidly blow-down and cool the SGS. Following reactor and subsequent turbine trip, the steam bypass system is used to maintain correct steam pressures and flows. Failure of a valve in this system in the open direction causes excessive steam flow with decreasing steam generator pressures and temperatures. The feedwater system will supply adequate water. It is assumed that after TBD minutes operator action results in closure of the bypass valve.

For design purposes, this event is specified to occur 8 times during the 60 year plant design life.

B-13 - Inadvertent Opening of Steam Generator Outlet Safety/Power Relief Valves

A steam relief valve opening at a steam generator outlet is assumed to occur and remains in a stuck-open position. The response is somewhat like Event B-7.

For design purposes, this event is specified to occur 4 times per evaporator module during the 60 year plant design life.

B-14 - Plant Shutdown in Response to Small Water-to-Sodium Leak Indication

This transient results in a NSSS module shutdown and affected steam generator depressurization for those water-to-sodium leak indications where immediate isolation and drainage is considered necessary to prevent wastage and a possible large leak which would actuate the IHTS rupture disks. For duty cycle damage calculation conservatism a reactor trip is assumed, followed by steam generator depressurization. This is then followed by intermediate system drain and cooldown.

A reactor shutdown is initiated automatically or by operator actions following input from the steam generator leak detection system. Following the shutdown, the leaking steam generator is manually isolated and the waterside dumped. The system response is similar to Event B-7.

For design purposes, this event is assumed to occur once per steam generator during the 60 year design life.

B-15 - Loss of Power to Recirculation Pump

The recirculation pump in one loop is assumed to coast down. The lower flow to the evaporator results in superheated steam at the evaporator outlet. The affected NSSS module reactor and sodium pumps will be tripped. The normal decay heat removal path is available via the SGS and condenser.

For design purposes, this event is specified to occur 8 times per module during the 60 year plant design life.

D 5.0

D5.0 Level C Service Limits (Emergency Conditions)

All emergency events that result in a reactor trip shall be considered to result in a transient followed by a cooldown to refueling conditions.

D5.1 Definitions

According to the ASME Code, Level C Service Limits are all loadings which permit large deformations in areas of structural discontinuity. The occurrence of stress to Level C Limits may necessitate the removal of the component from service for inspection or repair of damage to the component or supports. This was formerly referred to as Emergency Conditions which were defined as "Those deviations from Normal Conditions which require shutdown for correction of the conditions or repair of damage in the system. The conditions have a low probability of occurrence but are included to provide assurance that no gross loss of structural integrity will result as a concomitant effect of any damage developed in the system.

D5.2 Frequency

Since the individual emergency events are not statistically expected to occur during the life of the plant, the number of event occurrences specified as a design basis are based on conservative judgement. Therefore, it is recommended that each plant component be designed to accommodate 4 cycles of the most severe emergency event. If consecutive occurrences of any two like or unlike emergency events produce a more severe effect than the four isolated occurrences of the most severe individual event, then the design must also accommodate this more severe sequence.

D5.3 Events

C-1 - Primary Pump Electrical Failure

The event involves an instantaneous loss of voltage for one primary pump while the system is operating at 100% power. Primary system sodium flow in the affected pump decreases rapidly to zero and then reverses as the unaffected pumps run out on their head/flow curves. A reactor trip of the affected NSSS module will be initiated by the primary pump power supply function or a drop in core inlet pressure. Sodium flow in the intermediate circuit decays as in a reactor trip from full power, modified by changes in natural circulation head and momentum effects. The event causes high core coolant outlet temperatures for a few seconds. The transient for IHTS and BOP components is essentially the same as a reactor trip, Event B-1.

C-2 - Intermediate Pump Mechanical Failure

The impeller of the intermediate system pump is assumed to stop, causing the flow in that circuit to decrease rapidly. The failure is assumed to prevent IHTS pony motor operation. A reactor trip of the affected module is initiated by a high IHX primary outlet temperature indication. The normal trip transient sequence is followed thereafter. The event is similar to a reactor trip transient but with slightly higher primary system temperatures, since the intermediate flow is limited to that produced by natural circulation.

C-3 - Rupture Disk Failure in SGS Sodium-Water Reaction Protection System

Flow of intermediate sodium or cover gas through the failed rupture disks will initiate trip of the affected reactor module based on pressure sensors downstream of the double rupture disk and activation of the steam generator water-side blowdown system as in Event B-7. The pressure sensors will also signal the intermediate pump to coast down and stop. The affected steam generator will be automatically isolated and blown down. Pressure and temperature builds up in the sodium-water reaction products relief system (SWRPRS) downstream of the rupture disks. Decay heat removal is by RVACS.

C-4 - Inadvertent Water-Steam Isolation & Dump of a Steam Generator Module With Failure of the Inlet or Outlet Isolation Valve to Close

These transients assume the same conditions as Event B-7 except that one of the steam generator isolation valves fails to close. The water or steam flow may continue to enter the steam generator of the affected NSSS module.

C-4a - Feedwater Inlet Isolation Valve Failure

Failure of a feedwater inlet isolation valve to close will result in reduced feedwater flow to the steam generators of the other two NSSS modules. In addition, the feedwater entering the affected steam generator will result in continuing steam and water blowing through the water dump valves and the outlet power relief valve. The turbine steam flow will be reduced by approximately one-third. It is assumed that the feedwater pumps can maintain the steam dump flows and the plant will initially continue operation. It is assumed the affected module will be tripped on a steam-feedwater flow mismatch control function or on water level in the steam drum control function.

C-4b - Steam Generator Steam Outlet Isolation Valve Failure

If the steam generator steam outlet isolation valve fails to close, the transient is essentially the same as Event B-7 since there is a check valve downstream of the outlet isolation valve. The check valve stops backflow from the steam header.

C-5 - Water-Side Isolation of a Steam Generator With Failure of the Dump Valves to Open

This transient assumes the same conditions as Event B-7 except the water and steam dump valves at the steam generator fail to open. The steam and feedwater flows will stop and the input heat will raise the pressure until the outlet power relief valves open, drying the unit at pressure (instead of a dump and dryout at low pressure). Decay heat removal is through the RVACS and ACS using the IHTS pony motor.

C-6 - Uncontrolled Control Rod Movements

These two events result from multiple control system malfunctions and failures of related plant instrument channels.

C-6a - Uncontrolled Rod Withdrawal from 100% Power

An uncontrolled withdrawal of one control rod causes the reactor power to increase from 100% based on the reactivity rate imposed. The power increase is terminated just below plant trip settings. A manual reactor trip occurs after TBD minutes. Sodium flows are maintained at initial values until the trip occurs. Initial decay heat level is the nominal level. The transient results in temperatures similar to a normal trip, but from high initial values.

C-6b - Uncontrolled Rod Withdrawal from Startup to Trip Point with Delayed Manual Trip

The initial conditions for this event are hot standby with nominal, full power decay heat (i.e. just restarting after trip). Primary and intermediate pumps are operating at rated flow. Uncontrolled withdrawal of one control rod at maximum speed then occurs. The power increase is terminated at a point just below plant trip setting. The transient is terminated after TBD minutes by a manual trip. Flows are maintained at initial values until after the manual trip.

C-7 - Recirculation Pump Mechanical Failure

The impeller of the recirculation pump is assumed to stop causing the water flow to the evaporator to decrease rapidly. The sequence of events is similar to that described in Event B-15 with the temperature changes occurring at faster rates.

C-8 - Design Basis Leak

The IHTS shall accommodate the Design Basis Leak equivalent to three (3) double ended guillotine steam generator tube failures occurring at 1 second intervals. See Event D-4 for the steam generator event description.

D 6.0

D6.0 Level D Service Limits (Faulted Conditions)

D6.1 Definitions

According to the ASME Code, Level D Service Limit events permit gross general deformations with some consequent loss of dimensional stability and damage requiring repair, which may require removal of the component from service. This was formerly referred to as Faulted Conditions, which were defined as "those conditions or combination of conditions associated with extremely low probability postulated events whose consequences are such that the integrity and operability of the system may be impaired to the extent that considerations of public health and safety are involved. Such considerations require compliance with safety criteria as may be specified by jurisdictional authorities."

D6.2 Frequency

These events are postulated to occur one time for each of the seven D events in the 60-year plant design life.

D6.3 Events

D-1 - Feedwater Line Ruptures

Components and systems shall be designed so that in the event of any of the following faulted events, sodium and SGS water/steam boundaries shall maintain their structural integrity (with the exception of the initiating failure).

D-1a - Feedwater Line Rupture Between Storage Drum and Inlet Check Valve

This event assumes a rupture of a feedwater line between the inlet check valve and the storage drum. The result of the event is blowdown of the steam generator, the steam drum in the affected NSSS module, interruption of steam flow from the affected NSSS module (due to steam outlet check valve closure), and consequential transients within the module. The steam generator in the affected NSSS module dries out and is not available for normal decay heat removal. Decay heat is removed through RVACS and ACS using the IHTS pony motor. A stoppage of steam flow in the affected NSSS module occurs shortly after the rupture, due to reduced pressure in the steam drum. A NSSS module reactor trip occurs based on the steam-to-feedwater flow mismatch control function or on a water level in the steam drum control function. A low pressure signal from the steam drum protection subsystem results in closure of the main and auxiliary feedwater isolation valves in the affected NSSS module. This avoids excessive loss of plant feedwater. The transient is similar to Event B-7.

D-1b - Feedwater Line Rupture in Main Incoming Header

The main reduction in feedwater line pressure will cause the feedwater line check valves to close simultaneously at the inlet to all three steam drums. A reactor trip for all 3 NSSS modules will be initiated on steam-to-feedwater flow mismatch control function. The results of this event for each NSSS module are similar to those for Event B-5b, loss of feedwater flow to all steam generators supplying one turbine, and thus this event is evaluated as part of Event B-5b.

D-2 - Steam Line Ruptures

These events postulate ruptures of the piping in the steam lines. The events are also postulated to insure that the steam generators and supports are capable of withstanding the reaction forces from the rupture without propagating failures to the units themselves.

Components and systems shall be designed so that in the event of any of the following faulted events, sodium and steam generator system water/ steam boundaries shall maintain their structural integrity (with the exception of the initiating failure.)

D-2a - Single Module Steam Line Rupture

A steam line rupture is postulated between the steam drum outlet and the steam header inlet isolation and check valves. An immediate loss of 1/3 of the plant steam flow occurs. A NSSS module reactor trip will occur due to steam-to-feedwater flow mismatch control function. Depressurization of the NSSS module steam piping will occur, and the steam header inlet check valve will close. The turbine steam flow will be reduced by approximately one-third.

A low pressure signal from the steam drum protection subsystem closes the feedwater isolation valve. A dryout of the affected steam generator will occur.

D-2b - Main Steam Line Rupture

A steam line rupture is postulated to occur between the manifold where the three NSSS module steam lines join together and the main steam line isolation valve. The turbine admission valve will rapidly close and the turbine will trip. Since the pressures have dropped, the turbine bypass will not open. Feedwater flow will increase rapidly through the units but will initially be unable to equal the blowdown steam flowrate. The steam generator outlet isolation valve in each NSSS module will be closed because of low steam drum pressure. Affected power block reactor trips will occur on steam- to-feedwater flow mismatch control function or on a water level in the steam drum control function. Steam system pressure in the three NSSS modules will stabilize at the steam generator outlet vent valve setting. Decay heat removal in the three NSSS modules will be maintained in the short term by the SGS venting of steam and in the long term by the RVACS and ACS.

D-3 - Recirculation Line Breaks

Components and systems shall be designed so that in the case of this faulted event, sodium and steam generator system water/steam boundaries shall maintain their structural integrity (with the exception of the initiating failure).

The reactor is tripped on steam-to-feedwater flow mismatch control function or on a water level in the steam separator drum control function. The recirculation stops or reverses. Steam flow to the main steam header from the affected NSSS module also stops. The steam generator sodium temperature initially decreases, but then increases toward the hot leg value. The steam generator blows down to atmospheric pressure. The transient is similar to Event B-7.

D-4 - Design Basis Steam Generator Sodium-Water Reaction

This event consists of a rupture equivalent to three (3) double-ended guillotine steam generator tube failures occurring at 1 second intervals during a reactor trip from full power with a minimum decay heat, which results in rupture disk actuation, automatic isolation and blowdown of the affected steam generator, stopping intermediate flow and may result in manual activation of the sodium drain system. The IHTS experiences a pressure transient resulting from the reaction. Pressure and temperature builds up in the sodium-water relief system downstream of the rupture disk.

This event is considered a Level D (faulted) event for the affected steam generator. See Event C-8 for the IHTS event description.

Affected module decay heat removal is maintained through the RVACS.

D-5 - IHTS Line Rupture

This is the design basis for the RVACS. The line rupture results in a sodium fire with aerosol that plates out on the RVACS heat transfer surfaces. A reactor scram is initiated on primary-intermediate flow deviation or IHX primary outlet temperature. The resulting transient is a slow heat-up to a higher than normal temperature ∇1300°F due to the reduced RVACS performance.

D-6 - Reactor Vessel Leak to Containment Vessel

This is the design basis event for the containment vessel. A small leak develops in the reactor vessel. A reactor scram is initiated upon detection of sodium in the annulus between the reactor and containment vessel walls. The containment vessel continues to fill with sodium with an attendant reduction in the reactor vessel sodium level. Decay heat removal is maintained by the normal heat removal path via the SGS. The volume in between the reactor and containment vessels is sized to assure the inlets to the IHXs are not uncovered.

D-7 - Rupture in High Pressure Primary Circuit

This event assumes a rupture in any of the components connecting the primary pump discharge to the core assembly inlet. Failure results in a reduction in flow delivered to the reactor core and a surge in vessel sodium level. Two types of events are postulated:

1. Guillotine failure of a primary piping leg. This leads to a reduction in flow delivered to the core inlet plenum and, in addition, backflow through the effected leg.

2. Failure resulting in leakage from the inlet plenum structure resulting in shunting of a portion of the core coolant to either the hot plenum or the cold plenum depending upon the location of the failure.

In both cases a reactor scram occurs based on the change in primary pump discharge pressure and/or the core bulk outlet temperature. Decay heat is removed by the normal heat removal path via the SGS.

D-8 Extreme Steam Generator Sodium-Water Reaction

This event consists of a continuation of Event D-4, in which, after ruptured disk activation, failure of the systems providing automatic isolation and blowdown occurs. Additional steam generator tubes fail with continued water ingress until the IHTS pressure builds up to and is maintained in equilibrium with the steam side pressure.

This event is considered a Level D (faulted) event for the affected steam generator and IHTS.

Decay heat removal of the affected NSSS module is maintained through RVACS.

D 7.0

D7.0 Seismic Events

All structures, systems, and components important to safety shall be capable of withstanding the effects of the operating basis earthquake (OBE) without loss of capability to remain functional and to withstand the effects of the safe shutdown earthquake (SSE) without loss of capability to perform their safety functions.

D8.0 Summary

A listing of the duty cycle events and their frequencies is contained in Table D-1.

TABLE D-1

DUTY CYCLE EVENTS

EVENTS	TRANSIENTS	PRISM MODULE/NSSS DESIGN FREQUENCY
Level A Service		
A-1a	Dry System Heat-up Sodium Fill, Heat-up to Refueling Temperature	1 primary 12 intermed.
A-1b	Cooldown from Refueling Temperature Sodium Drain Dry System Cooldown	1 primary 12 intermed.
A-2a	Startup from Refueling Temperature	60
A-2b	Startup from Hot Standby Conditions	573
A-2c	Turbine Generator Warm-up, Roll and Loading to 25%	573
A-3a	Shutdown to Refueling Temperature	60
A-3b	Shutdown to Hot Standby Conditions	60
A-4	Weekly Loading and Unloading	2880 up & down
A-5	Daily Loading and Unloading	14400 up & down
A-6	Steady State Temperature Fluctuations	1.0×10^7
A-7	Steady State Flow Induced Vibrations	TBD
A-8	Module Out of Service	120 Active 60 Inactive
A-9	Stepload Increase or Decrease of 10% of Rated Power	1500 up & down
A-10	Turbine Steam Inlet Valve Testing	2655
A-11	Fast Ramp Load Changes of 25% of Rated Power	30 up & down

TABLE D-1 (Cont)

DUTY CYCLE EVENTS

EVENTS	TRANSIENTS	PRISM MODULE/NSSS DESIGN FREQUENCY
Level B Service - UPSET		
B-1a	Reactor Trip from Full Power with Maximum Decay Heat	150
B-1b	Reactor Trip from Full Power with Minimum Decay Heat	75
B-1c	Reactor Trip from Partial Power with Minimum Decay Heat	75
B-2a	Uncontrolled Rod Insertion	15
B-2b	Uncontrolled Rod Withdrawal from Startup with Automatic Trip	3
B-2c	Reactor Loadings at Maximum Rod Withdrawal Rate	3
B-3a	Partial Loss of Primary Pump Flow	3 per pump
B-3b	Loss of Power to One Primary Pump	3 per pump
B-4a	Partial Loss of Intermediate Pump	3
B-4b	Loss of Power to Intermediate Pump	3
B-5a	Trip of One Feedwater Pump	45
B-5b	Loss of Feedwater Flow to All Generators Supplying One Turbine	8
B-6	Intermediate Pump Pony Motor Failure	3
B-7	Inadvertent Waterside Isolation and Blowdown of the Steam Generator of One Module	3
B-8	Loss of Feedwater to One Module	6
B-9	Feedwater Control Valve Failed Open	6

TABLE D-1 (Cont)

DUTY CYCLE EVENTS

EVENTS	TRANSIENTS	PRISM MODULE/NSSS DESIGN FREQUENCY
Level B Service - UPSET (Continued)		
B-10a	Turbine Trip Without Immediate Reactor Trip	105
B-10b	Turbine Trip with Reactor Trip (Loss of Main Condenser or Similar Problem)	14
B-11	Loss of All Off-Site Power	8
B-12a	Inadvertent Opening of One Turbine Bypass Valve	3
B-12b	Turbine Bypass Valve Fails Open Following Reactor Trip	8
B-13	Inadvertent Opening of a Steam Generator Outlet Safety/Power Relief Valve	4
B-14	Plant Shutdown in Response to Small Water to Sodium Leak Indication	1
B-15	Loss of Power to Recirculation Pump	8

TABLE D-1 (Cont)

DUTY CYCLE EVENTS

EVENTS	TRANSIENTS	PRISM MODULE/NSSS DESIGN FREQUENCY*
LEVEL C Service - EMERGENCY		
C-1	Primary Pumps Electrical Failure	
C-2	Intermediate Pump Mechanical Failure	
C-3	Rupture Disk Failure in SGS Sodium-Water Reaction Protection System	
C-4a	Isolation and Dump of Steam Generator Module-Feedwater Inlet Isolation Valve Failure	
C-4b	Isolation and Dump of a Steam Generator Module - Outlet Steam Line Isolation Valve Failure	
C-5	Water Side Isolation of a Steam Generator with Failure of the Dump Valves to open	
C-6a	Uncontrolled Rod Withdrawal from 100% Power	
C-6b	Uncontrolled Rod Withdrawal from Startup to Trip Point with Delayed Manual Trip	
C-7	Recirculation Pump Mechanical Failure	
C-8	Design Basis Leak	

* Design frequency is 4 cycles of worst event for each component (isolated events). Combination of 2 consecutive like or unlike events must also be accommodated.

TABLE D-1 (Cont)

DUTY CYCLE EVENTS

EVENTS	TRANSIENTS	PRISM MODULE/NSSS DESIGN FREQUENCY
Level D Service - FAULTED		
D-1a	Feedwater Line Rupture Between Storage Drum and Inlet Check Valve	1
D-1b	Feedwater Line Rupture in Main Incoming Header	1
D-2a	Single Module Steam Line Rupture	1
D-2b	Main Steam Line Rupture	1
D-3	Recirculation Line Break	1
D-4	Design Basis Steam Generator Sodium-Water Reaction	1
D-5	IHTS Line Rupture Plus Sodium Fire	1
D-6	Reactor Vessel Leak to Containment Vessel	1
D-7	Rupture in High Pressure Primary Circuit	1
D-8	Extreme Steam Generator Sodium-Water Reaction	1
Seismic Events		
	Operating Base Earthquake (Upset)	5
	Safe Shutdown Earthquake (Faulted)	1

APPENDIX E
ANALYSIS OF SELECTED BEYOND DESIGN BASIS EVENTS

APPENDIX E

ANALYSIS OF SELECTED BEYOND DESIGN BASIS EVENTS

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E 1.0

E.1 Introduction

This section presents inherent safety performance evaluations which include overall system effects, as appropriate, through the intermediate heat transport (IHTS) and balance of plant (BOP) systems. The PRISM core and primary heat transport system is designed to assure benign performance during a selected set of events without either reactor control or protection system intervention (referred to as "unprotected"). Based on their very low probability of occurrence these events are designated as beyond the design basis (BDBE) events and are included in the design process to assure public safety. The events considered are:

- Unprotected loss of primary flow and loss of IHTS cooling (ULOF)
- Unprotected loss of IHTS cooling (ULOHS)
- Unprotected control rod withdrawal (UTOP)

Each of the BDBE's is evaluated on a nominal basis in the following subparagraphs. The results are then summarized and compared to the set of acceptance criteria (next paragraph) in Section E.5. It is concluded that the metal core PRISM design will successfully meet all of the inherency criteria with margin.

The criteria used to judge the adequacy of the module inherent performance are based on providing for public safety by assuring the integrity of the fuel rods and the primary system structures. The criteria consider the duration of two key periods during the accident transients. For some transients there is a brief interval shortly after the start of the event during which the highest temperatures occur. For the brief highest temperature period of the transient, the most likely cladding midwall failure mechanism is expected to be stress-rupture due to weakening of the HT9 cladding at high temperature. For this situation, the cladding midwall temperature limit is 1450°F. For the longer period of the transient at lower temperatures, the most likely cladding failure mechanism is the formation of a low-melting point eutectic between the cladding and the metal fuel. The fuel-cladding interface temperature limit for this situation is 1290°F.

The containment function of the vessel structures and boundaries is protected by limiting their temperatures to be less than 1300°F. This BDBE criteria is equivalent to the reactor structures design basis Level D (faulted) condition.

Besides thermal damage protection, the criteria also precludes dynamic loads on the vessel by ensuring that margins are maintained relative to fuel melting (2000F) and sodium boiling (1800F). These physical phenomena are to be avoided since they are considered necessary initial events in the development of any severe dynamic loadings.

Table E.1-1 presents the criteria and the quantitative values selected for the reference metal core PRISM design.

Calculations have been completed for three beyond design basis events over a time interval of two thousand seconds, covering the initial system responses. Additionally a longer time frame evaluation that shows the transition to RVACS operation (∿thirteen hours) has been completed for the loss of flow scenario.

The three unprotected events evaluated are the loss of primary flow (LOF), reactivity insertion (TOP), and loss of heat sink (LOHS). Both the beginning and end of an equilibrium cycle core condition (BOC or EOC) system response were evaluated. However, the details of both analyses are not discussed in this appendix as one configuration is usually more limiting. The initial plant condition was assumed to be full power operation (with equilibrium decay heat levels) where the maximum transient effects are anticipated due to the assumed events.

The reactor protection system and reactor controller subsystem actions have been ignored for the loss of flow and loss of heat sink inherency analyses. As appropriate, control responses in the IHTS & BOP are represented for the TOP scenario where the BOP is integral to the system behavior.

Nominal design, physics and materials data were assumed in the following analyses except for the application of plus two-sigma uncertainties on the decay heat values. The key ARIES code features and reactivity feedback models are discussed in Chapter 15 except for the following which are unique to the inherency analyses. The thermal-mechanical feedback due to core duct bowing and load pad contact was separated from the gridplate expansion and correlated to the ANL NUBOW3D results shown in Figure E.1-1. The control rod driveline expansion was also accounted for in these analyses. The reference design is without special CRDL thermal extenders and has a stagnant sodium region inside of the shroud over an approximate 20 foot length. Heat transfer and fluid dynamics in the vessel cold wall region annulus are also accounted for relative to the CRDL expansion. The effect of the core grid plate expansion is included via a simple algorithm (-0.434 ¢/mill of the core average radius change) coupled to the primary loop thermal-hydraulic calculation; with a 200-second heat transfer time constant. At very low core flows (e.g., 1/2%) the radial temperature gradient causing duct bowing is assumed to collapse. The model sets the bowing reactivity to zero and switches the load pad contact effects back into the radial expansion (gridplate) model. The full PHTS, IHTS and BOP systems, along with RVACS heat losses are also considered, in addition to the reactor core.

Additional assumptions which are unique to the individual events are discussed in the separate analysis sections which follow.

Table E.1-1

KEY TRANSIENT PERFORMANCE CRITERIA FOR INHERENT SAFETY EVALUATIONS

Safety Goal	Performance Goal	Criteria
Contain Radio-active Materials	Maintain Cladding Integrity	Nominal peak fuel centerline temperature less than solidus temperature. (2000°F)
		Long-term nominal peak fuel surface temperature less than fuel - cladding eutectic temp. (1290°F)
		Short-term nominal peak mid-wall cladding temperature less than thermal creep strength limit. (1450°F)
	Maintain Primary Boundary Integrity	Primary boundary structural temperature less than 1300°F.
	Preclude Dynamic Loading	Nominal peak subchannel sodium temperature less than local sodium saturation temperature. (1800°F)
		Nominal peak fuel centerline temperature less than solidus temperature. (2000°F)

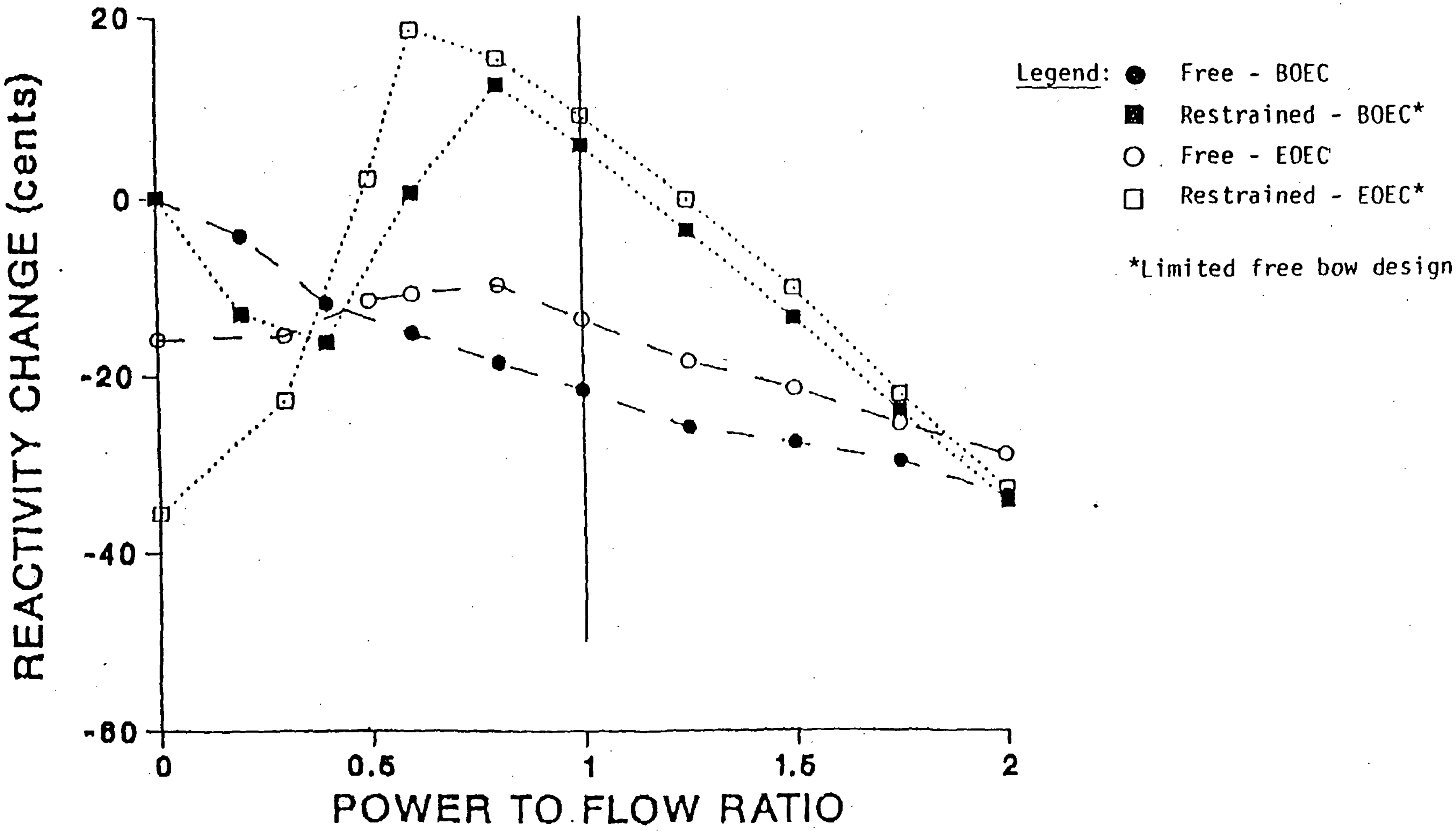

Figure E.1-1 - CALCULATED REACTIVITY BEHAVIOR BY NUBOW-3D FOR PRISM METAL CORE

E 2.0

E.2 Loss of Flow BDBE Performance

The unprotected loss of primary flow transient was examined at both the BOC and EOC condition. The primary control rods are inserted 8.2 and 2.0 inches at BOC and EOC, respectively. Note that the rods have a 36" stroke centered about the core midplane; hence the 2" insertion at EOC is actually a full out position for the rods. The IHTS flow was arbitrarily assumed to be instantaneously zero concurrent with the primary pump trip at time zero. Hence, long-term heat rejection is only by RVACS. The BOC configuration is the most challenging due to the positive reactivity that is the result of relative motion between the core and control rods from reactor vessel thermal expansion.

The trip of the primary pumps causes a rapid flow coastdown and equally rapid coolant temperature increases. These temperatures result in a strong negative bowing feedback which offsets the sodium density effect and rapidly reduces the core power. After about five minutes the core gridplate heating and expansion becomes the dominant negative reactivity. Likewise, reactor vessel heating and extension, which pulls the core away from the top supported control rods, becomes the dominant positive reactivity. After one-half hour (1800s) the net reactivity is approximately zero, natural circulation flow exists (about 2.5%) and the primary system temperature is slowly increasing due to the difference, about 12 MWt, between core decay power and RVACS rejection. See Figures E.2-1 through E.2-5 for the detailed response during the first 2000s of this event.

A longer term analysis (13 hours) examined the transition to long term heat removal by RVACS. The result shows that the inherency limits are met and that the reactor vessel average temperature remains well below 1300F. The sharp drop in core outlet temperature near 1.5 hours (5400s) is due to the reactor outlet plenum sodium overflow into the upper vessel voided annulus region. This lowers the overall flow circuit pressure drop by bypassing the IHX and results in higher natural circulation flows. This in turn has a beneficial (-25 cent) effect on core bowing feedback; see Figures E.2-6 through E.2-9.

At thirteen hours the reactor vessel average and peak temperatures are 1117°F and 1150°F, respectively, with RVACS rejecting 2.7 MWt. This exceeds the core power which is 2.0 MWt (0.5%). The reactor average coolant temperature is 1182F with the core subcritical by 31 cents. Beyond this time the vessel and primary system temperatures should decline based on the indicated surplus of heat rejection capacity.

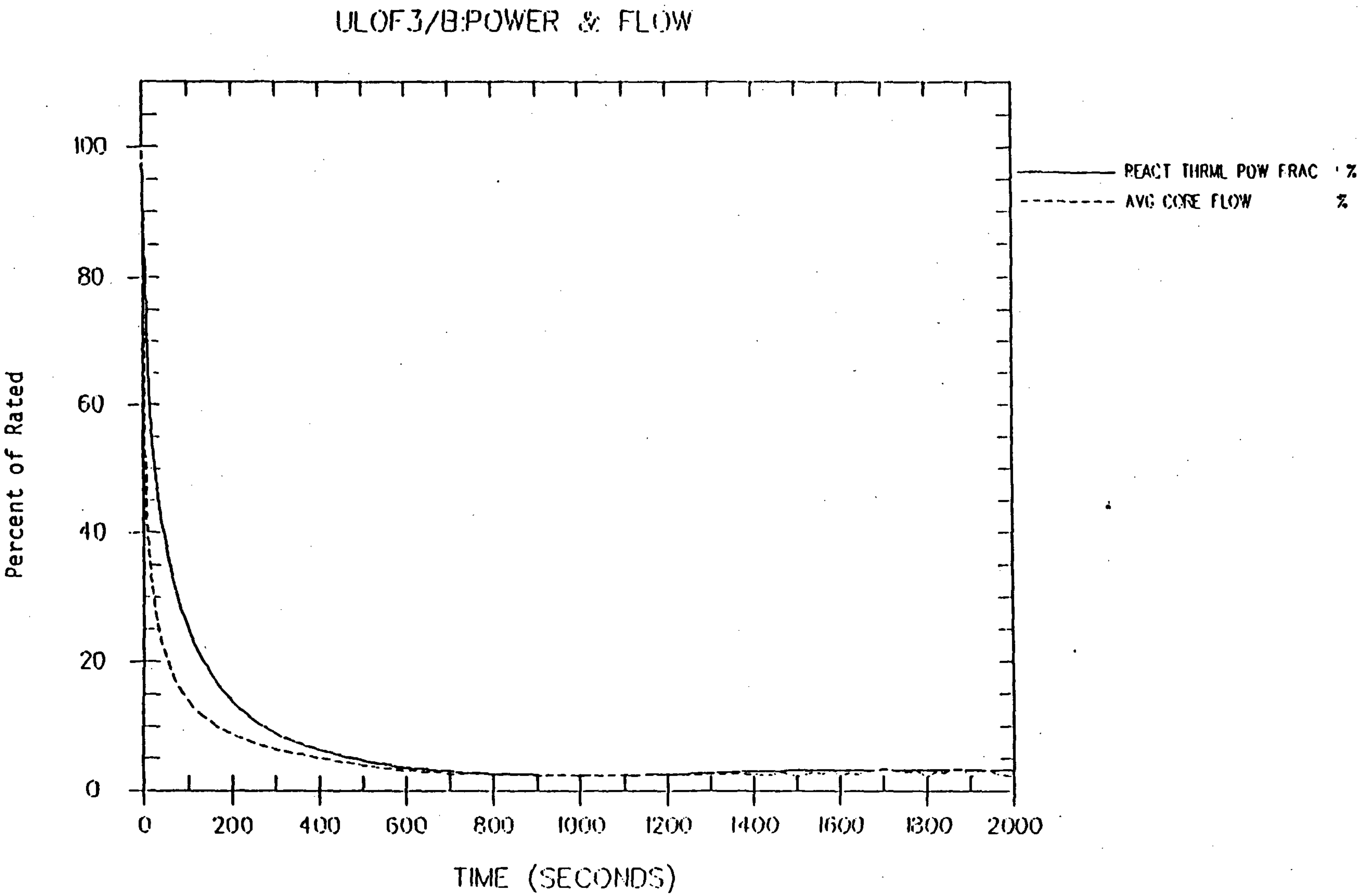

Figure E.2-1 BOEC CORE POWER AND FLOW DURING UNPROTECTED LOSS OF FLOW AND LOSS OF IHTS COOLING

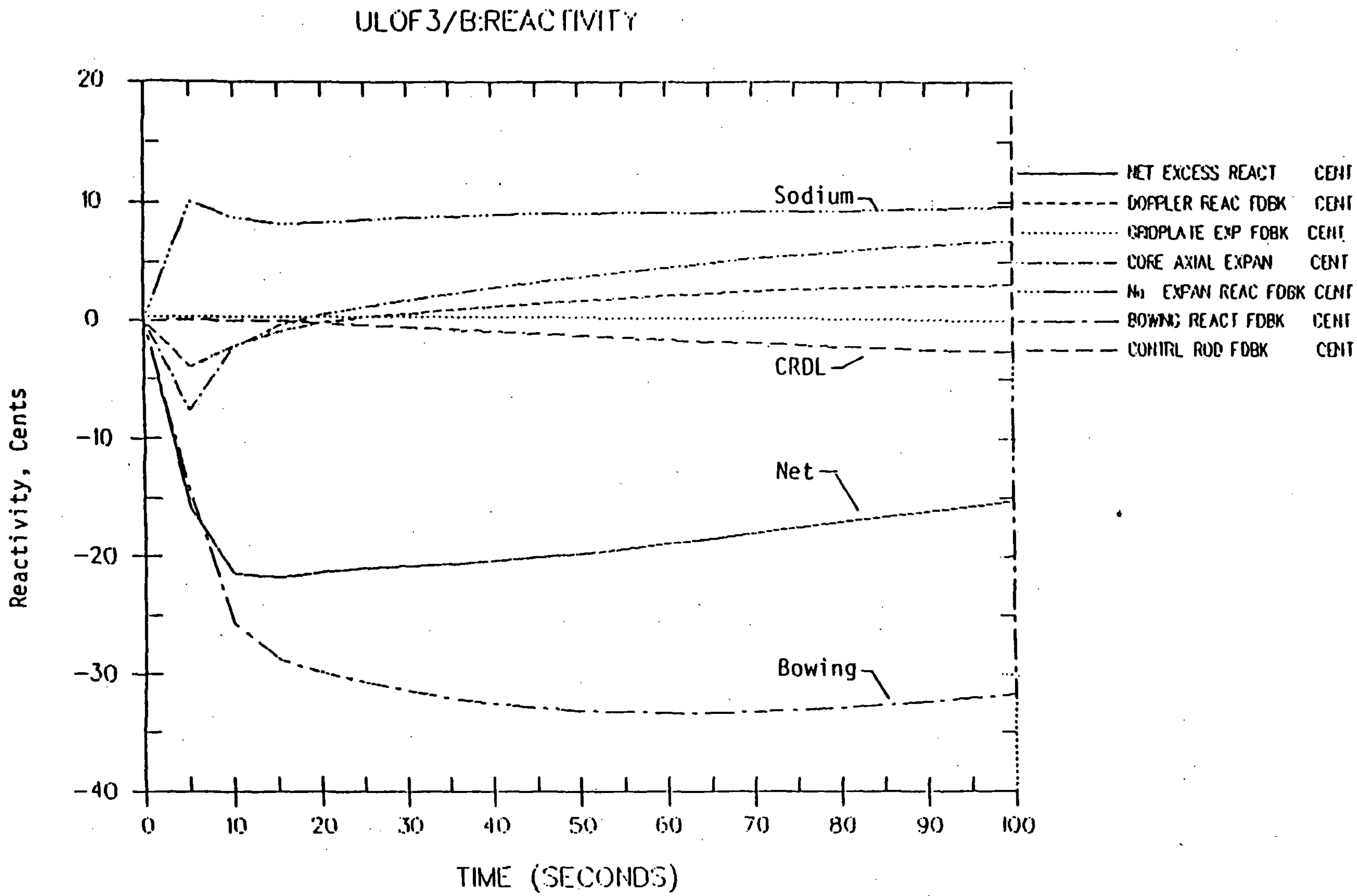

Figure E.2-2 BOEC INITIAL REACTIVITY COMPONENTS FOR UNPROTECTED LOSS OF FLOW AND LOSS OF IHTS COOLING

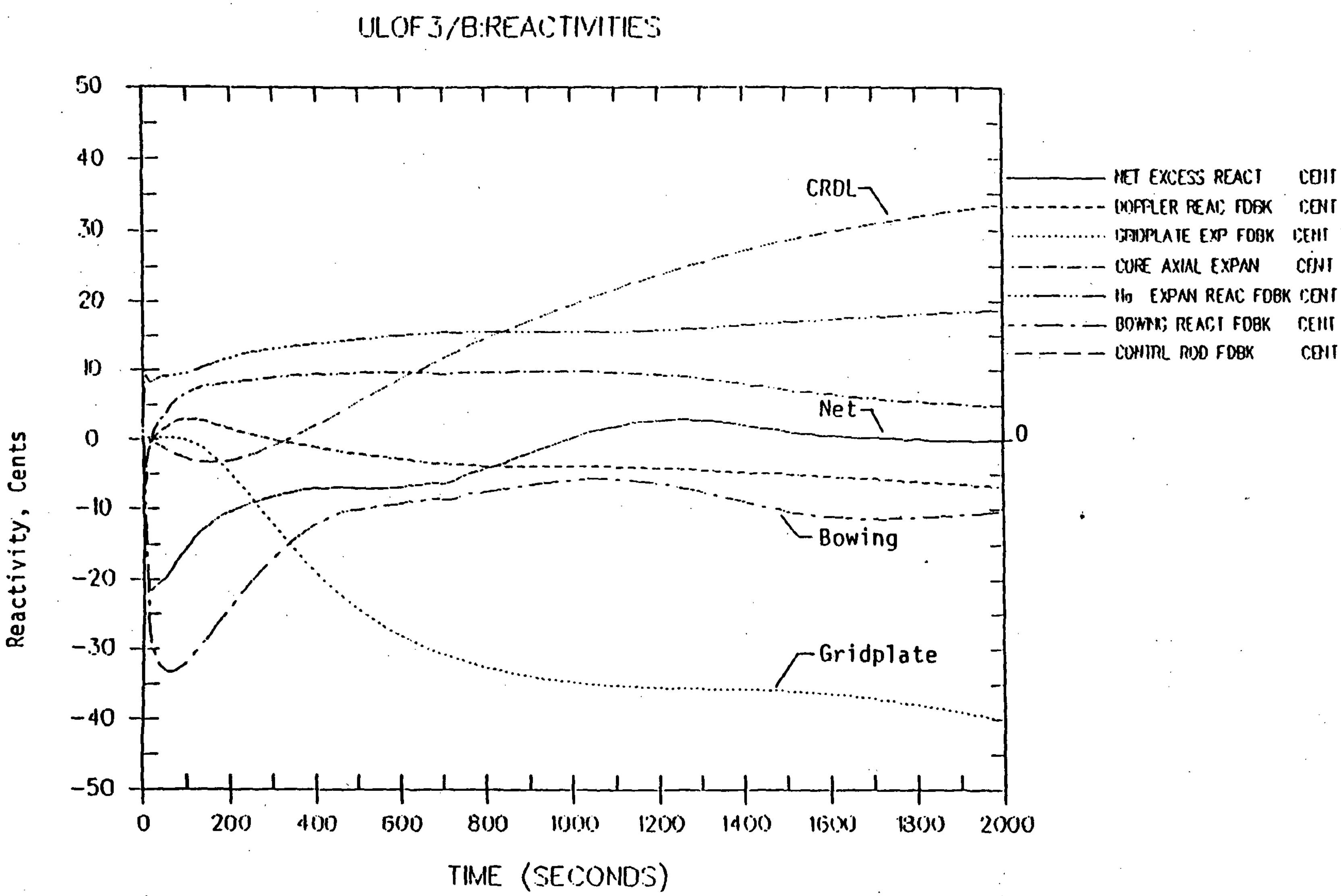

Figure E.2-3 BOEC REACTIVITY COMPONENTS DURING UNPROTECTED LOSS OF FLOW AND LOSS OF IHTS COOLING

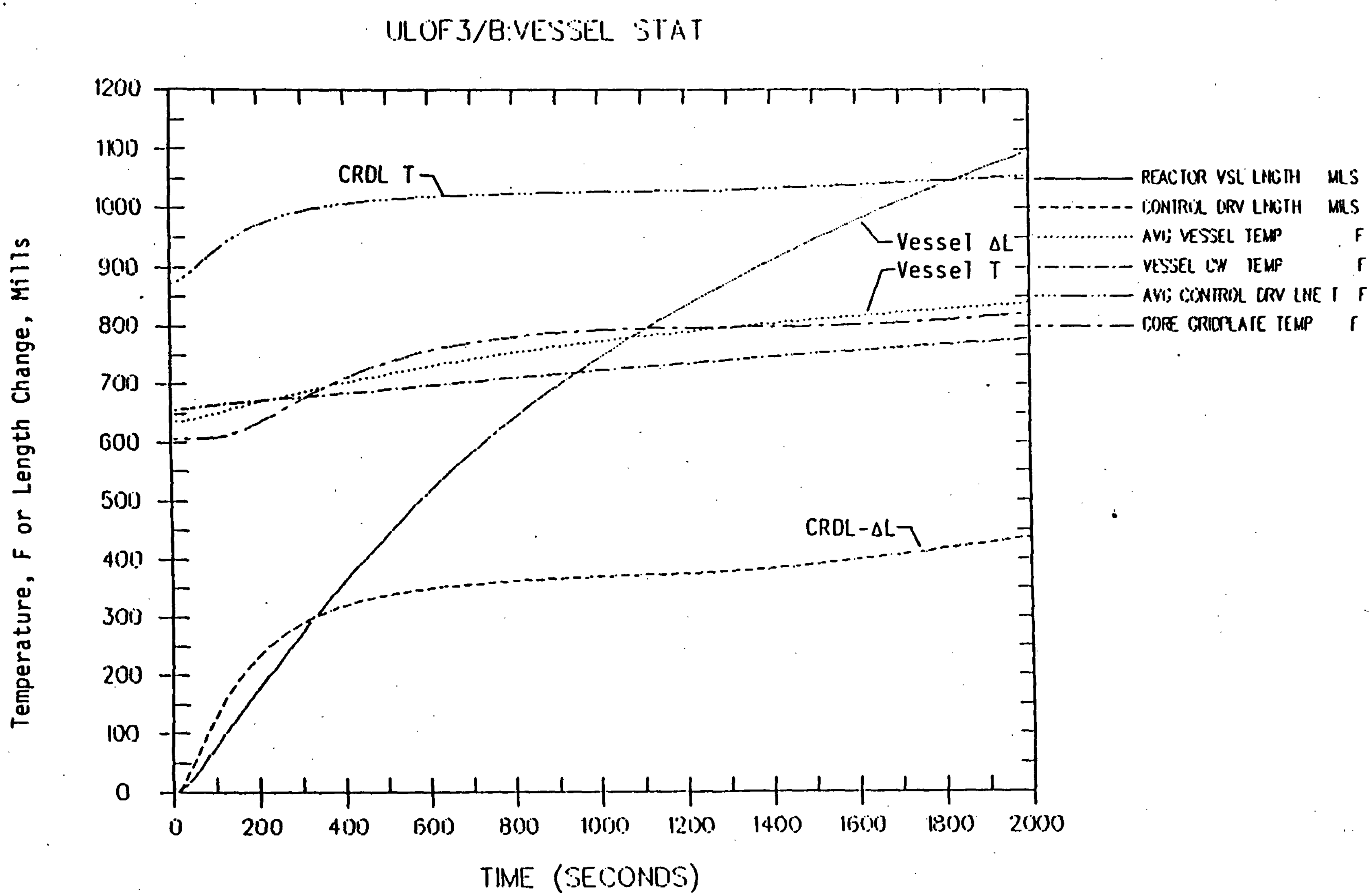

Figure E.2-4 BOEC VESSEL CONDITIONS CONTROLLING CRDL EXTENSION (ULOF/LOHS)

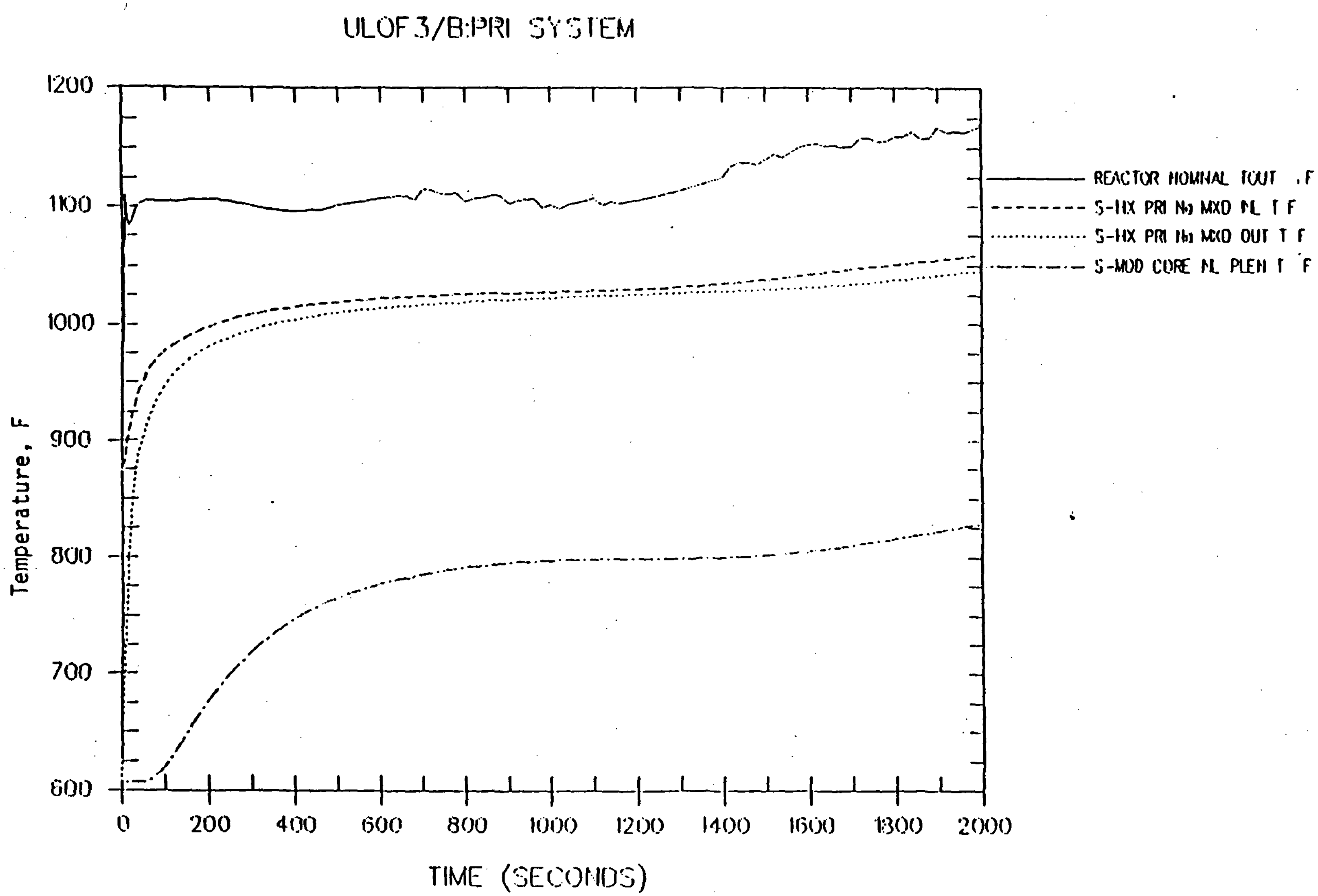

Figure E.2-5 BOEC PRIMARY COOLANT TEMPERATURES DURING UNPROTECTED LOSS OF FLOW AND LOSS OF IHTS COOLING

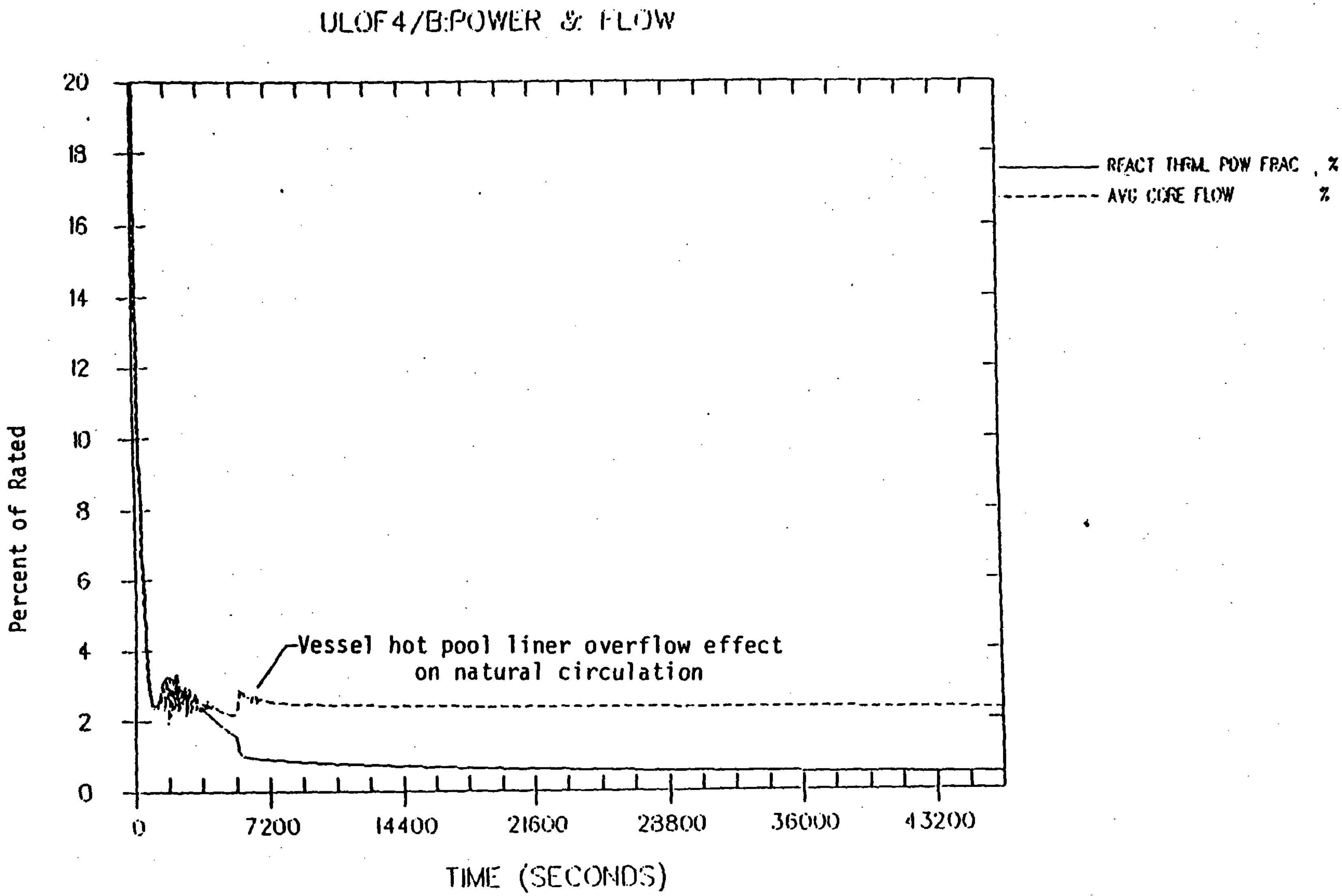

Figure E.2-6 BOEC CORE POWER AND FLOW DURING UNPROTECTED LOSS OF FLOW AND LOSS OF IHTS COOLING

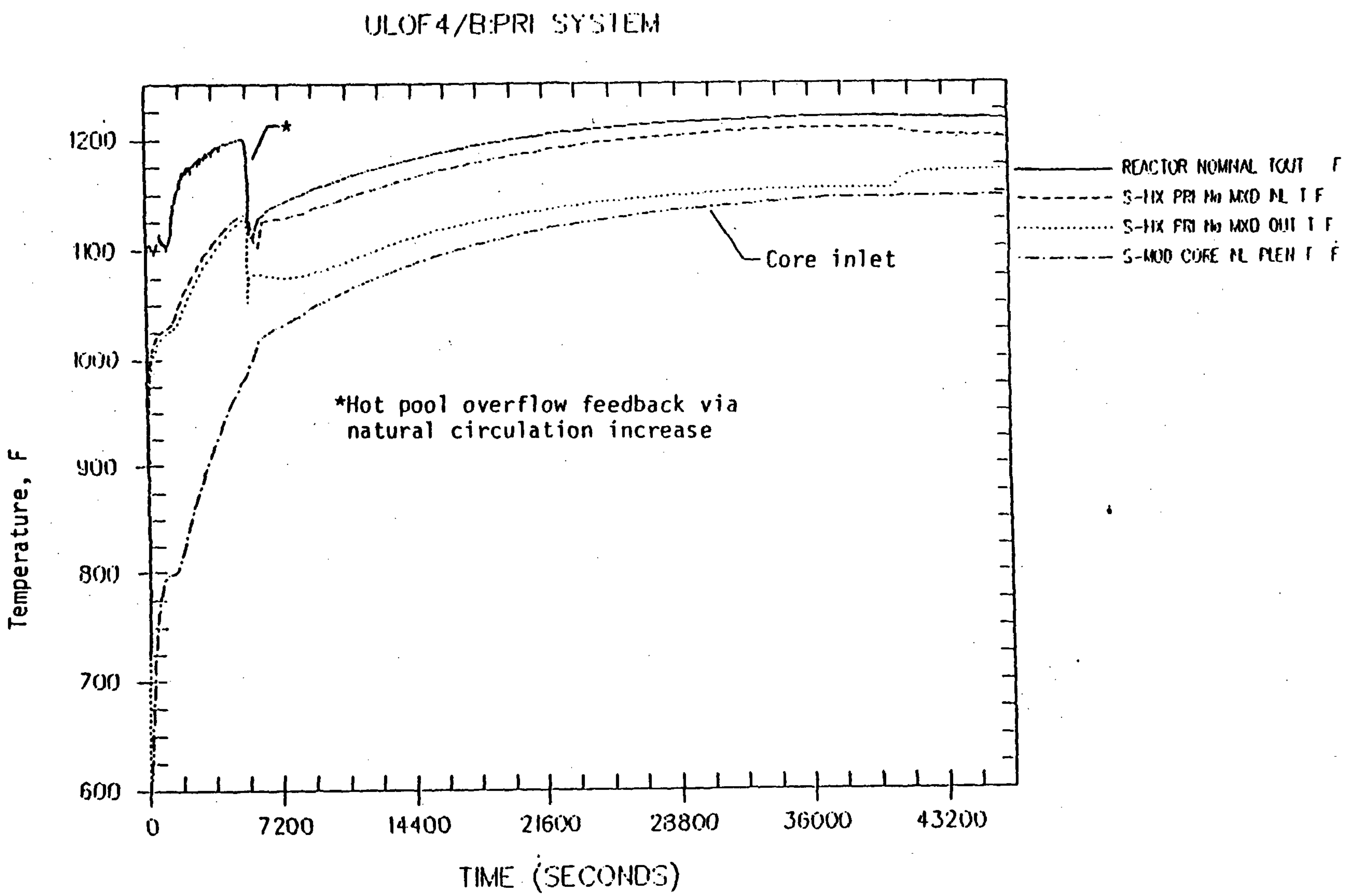

Figure E.2-7 BOEC PRIMARY SYSTEM RESPONSE TO UNPROTECTED LOSS OF FLOW AND LOSS OF IHTS COOLING

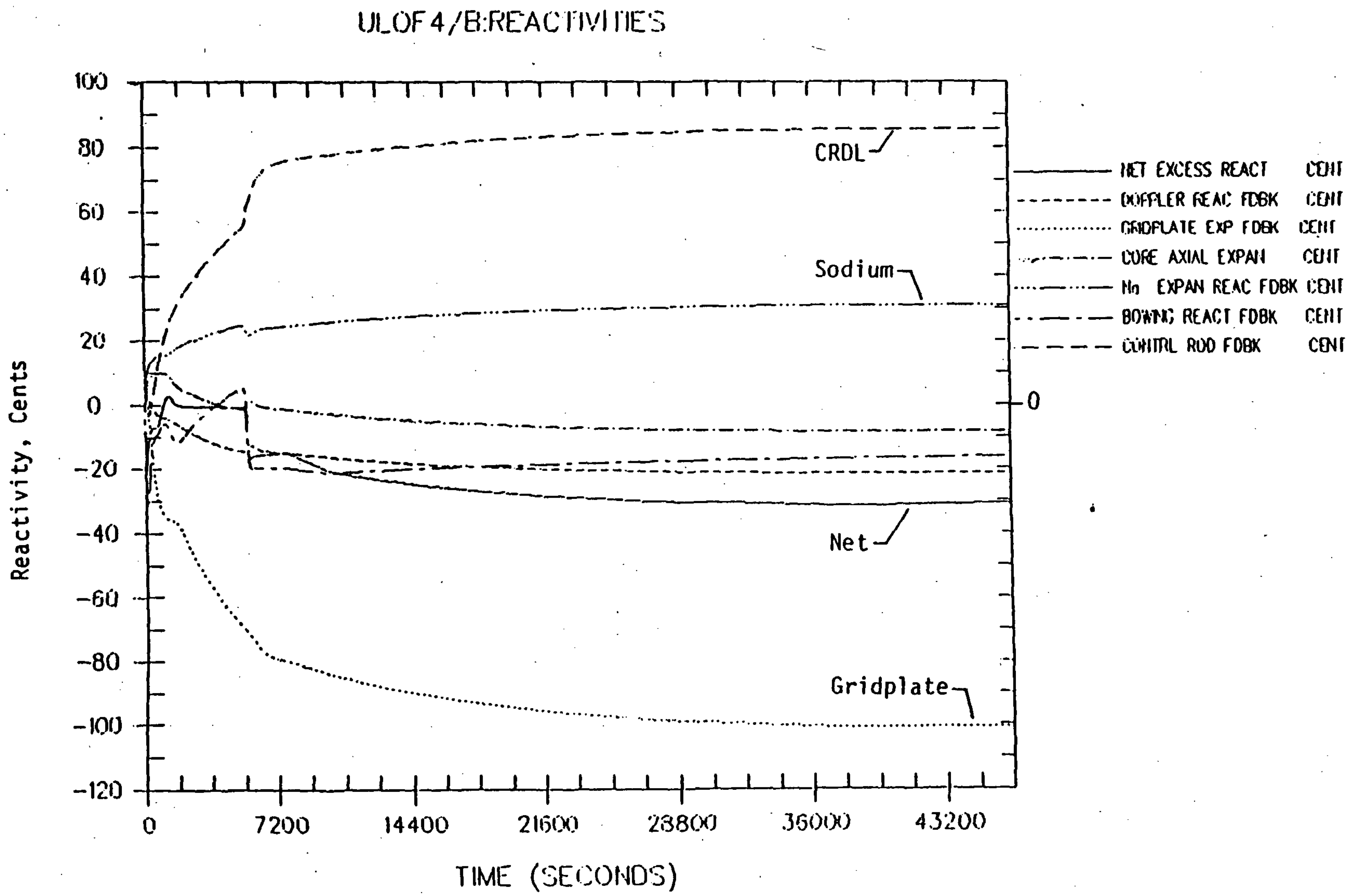

Figure E.2-8 BOEC REACTIVITY RESPONSE TO UNPROTECTED LOSS OF FLOW AND LOSS OF IHTS COOLING

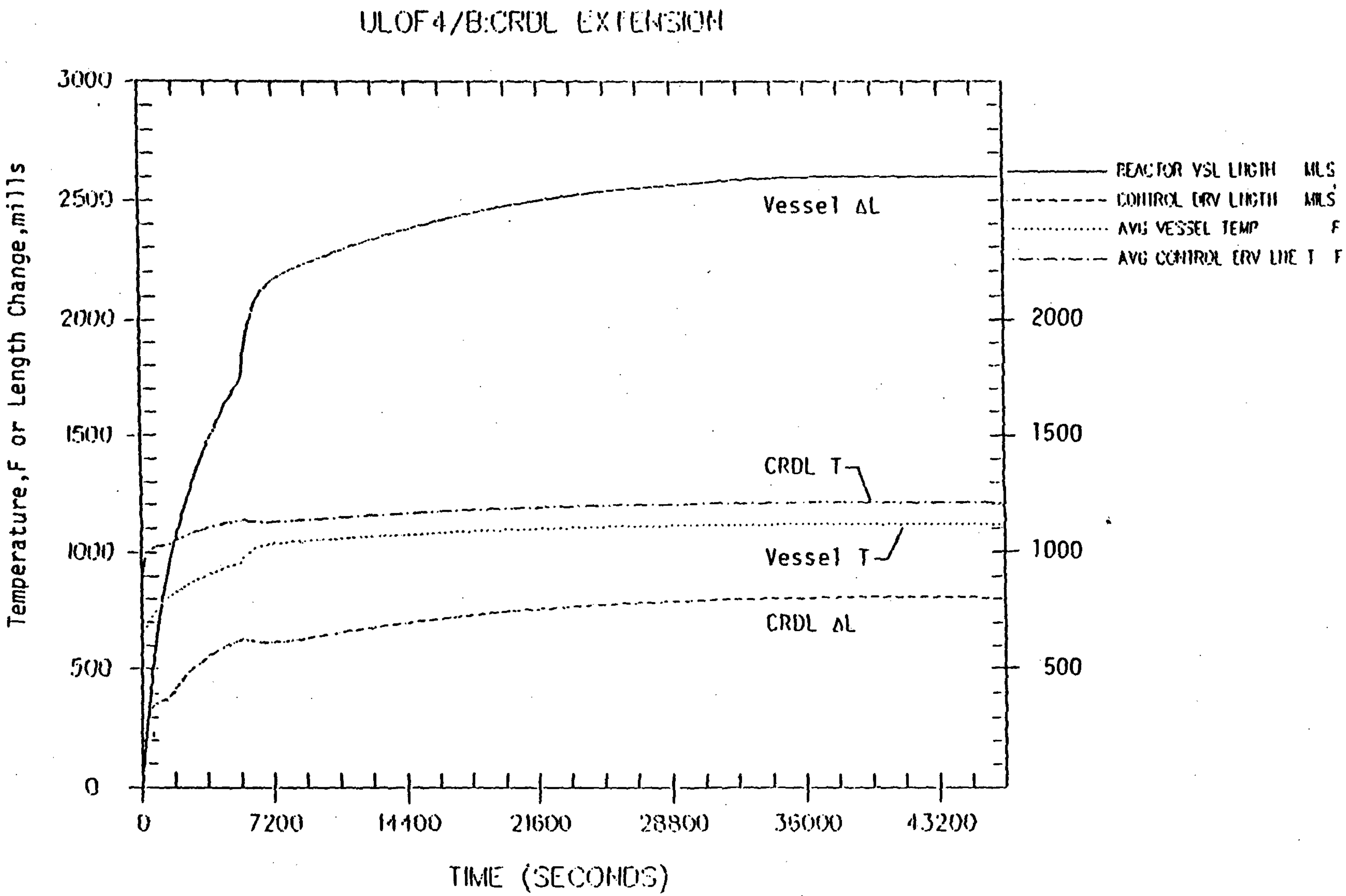

Figure E.2-9 BOEC COMPONENTS OF CRDL EXTENSION (ULOF/LOHS)

E 3.0

E.3 Reactivity Insertion BDBE Performance

For this evaluation the BOC core configuration was selected primarily due to the somewhat smaller Doppler and bowing magnitude (fuel will heat up) and the higher fuel specific power. To be conservative, the fuel column axial expansion is assumed to be governed by the cladding temperature, i.e., fuel expands radially and locks to the stronger cladding which restrains the fuels ability to expand. The reactivity insertion characteristic was based on the highest single control rod worth, (at any time in life) being withdrawn at nominal speed. A magnitude of 35 cents was inserted at a rate of 2 cents per second.

Based on the above assumptions the 35¢ reactivity insertion at full power conditions was evaluated for a ten minute period assuming complete failure of the reactor control and protection systems. The expected response of the fuel and coolant to the increased power is to heat up and, as such, no strong positive reactivity feedbacks exist. The major mitigating mechanisms are the Doppler, bowing and fuel axial expansion. The power peaks at 1.55P (i.e., 155% of nominal) at the termination of the 35¢ insertion and then reaches a new, stable state at 1.26P. The core outlet temperature rises 111°F, (including an inlet temperature rise of 38°F) which is sufficient to balance the reactivity insertion and, with a 26% rise in IHX ΔT, dissipate the additional energy into the IHTS and BOP. The core and primary system responses are shown in Figures E.3-1 through E.3-5. All primary system temperatures are well within the inherency criteria for this event.

The BOP, at 600 seconds, has not been able to readjust for the change in the core conditions. The plant controls were represented as a turbine leading mode with a setpoint of 100% of rated conditions. Hence, the BOP system is trying to maintain the turbine condition by supplying additional feedwater and venting the excess steam at a drum pressure setpoint of 1100 psi. However, the feedpumps cannot deliver the desired higher flow against the established high drum pressure, even with the feedwater valve full open. Feedwater flow actually dips slightly below rated conditions and cannot match the 26% increase in power delivered to the steam generator.

As a consequence the steam drum dries out which triggers a loss of the recirculation system (current ARIES logic) which changes the character of the TOP transient to a loss of heat sink event from elevated conditions.[1]

An extended analysis was performed at the EOEC condition where drum dryout occurs at 400 sec, resulting in the assumed loss of the recirculation system and isolation of the feedwater and drum. With full primary and IHTS flow the system temperatures rapidly increase following the loss of the steam generator heat sink. Coolant temperatures peak at 1147°F (IHTS) and 1206°F (core outlet) before reducing to near 1100°F. Dominant feedbacks are the rapid gridplate expansion (-91¢) and later bowing (at low power-to-flow ratio) which, in combination with Doppler make the reactor subcritical by -89 cents. The transient progression is shown in Figures E.3-6 through E.3-10. At 2000 sec (∿one-half hour) the reactor system is quasi-stable, but, slowly heating due to the imbalance in power generation over heat rejection by RVACS and the Auxiliary Cooling System (ACS). For long term acceptable conditions to exist, primary the pumps must be shut off as they supply more energy than the combined RVACS and ACS can reject. Clearly, since the event is one-half hour old at this time and heating slowly, there is sufficient time for operator action to correct the failure.

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[1] Under the assumption that this event occurs simultaneously in the three modules serving the one turbine. If the event were confined to a single reactor, the plant supervisory controller may throttle back the other two reactors and reach a sustainable power block balance between thermal power and feedwater/turbine capability.

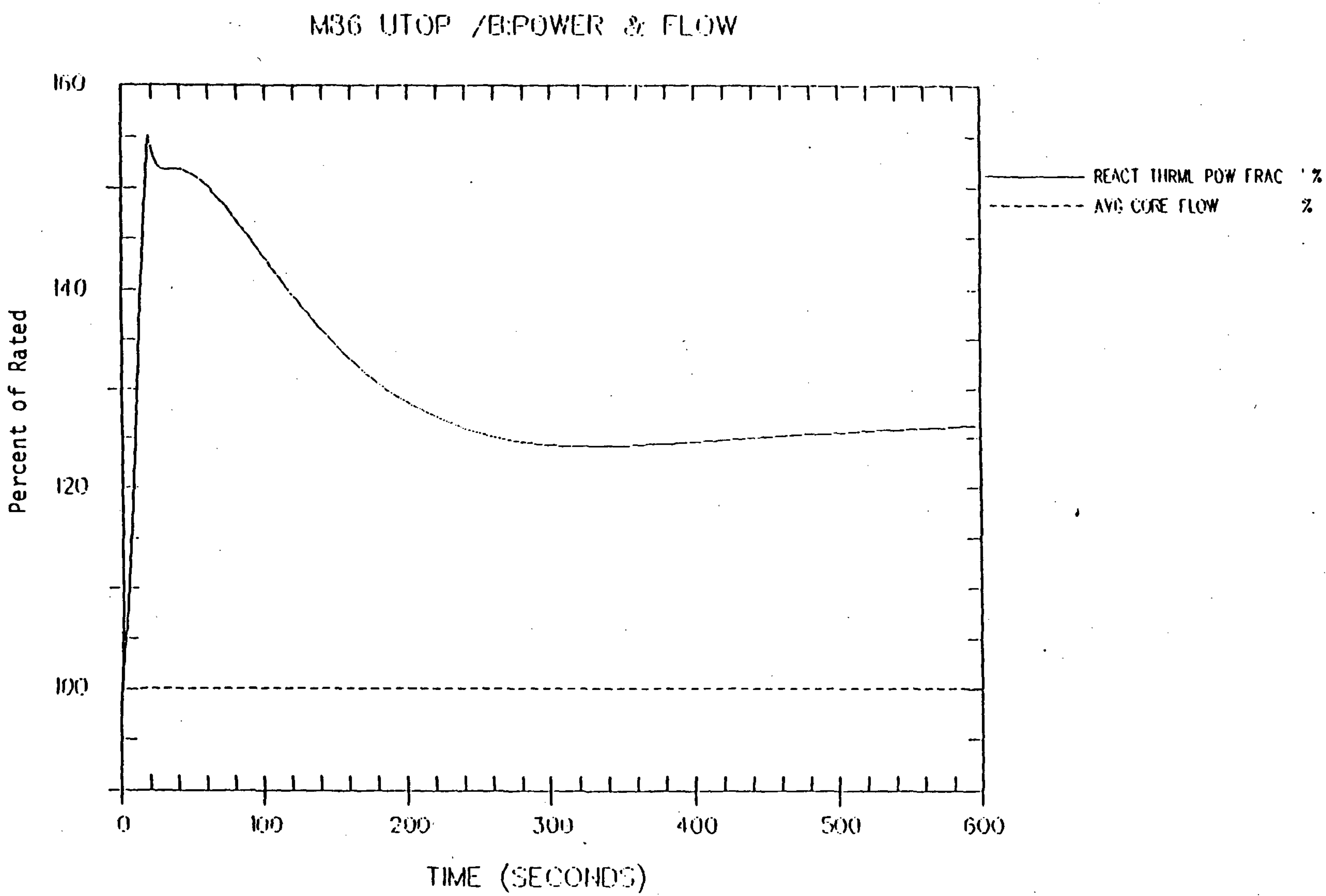

Figure E.3-1 BOEC CORE POWER AND FLOW RESPONSE TO UNPROTECTED CONTROL ROD WITHDRAWAL (35 CENTS)

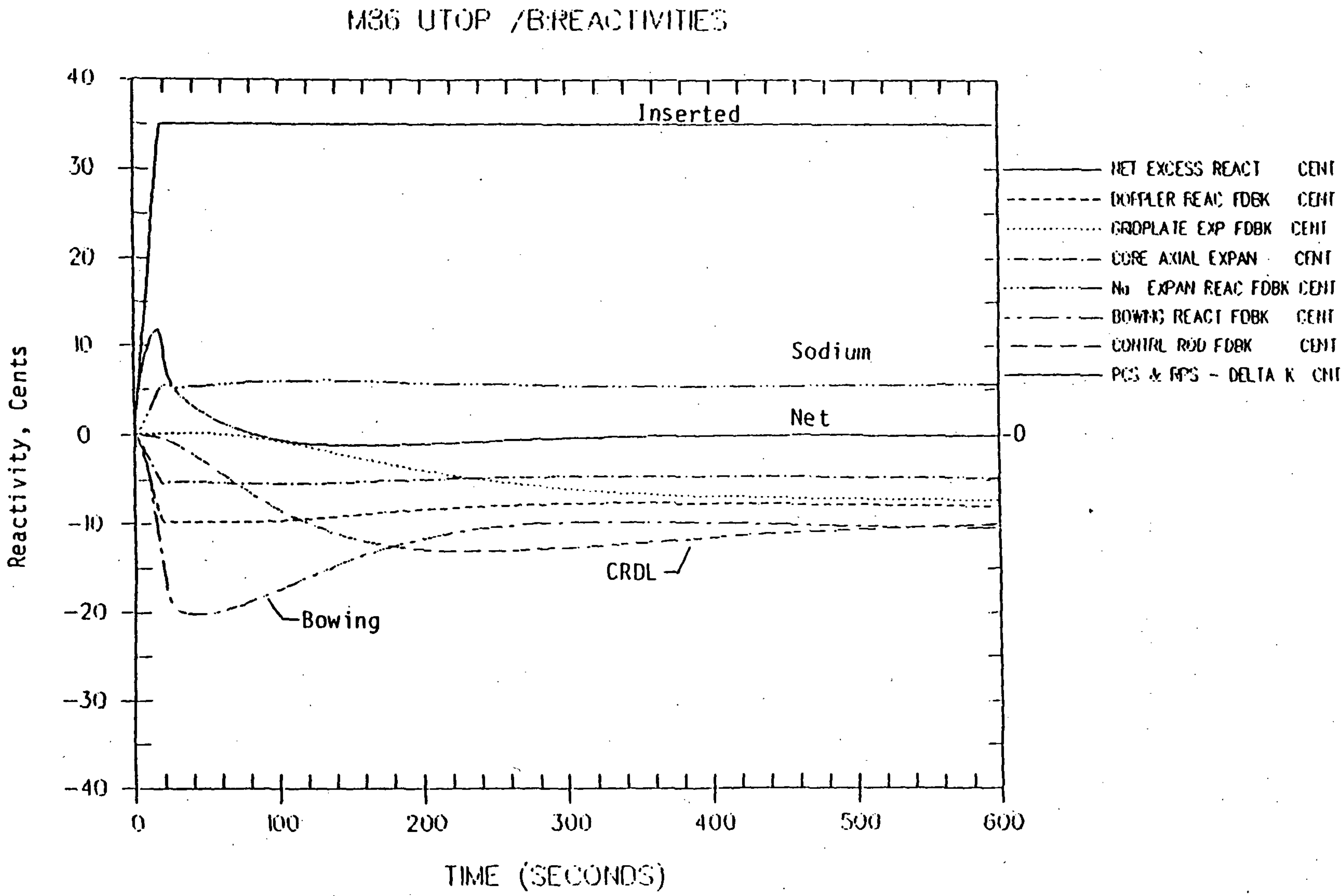

Figure E.3-2 BOEC REACTIVITY RESPONSE TO UNPROTECTED CONTROL ROD WITHDRAWAL

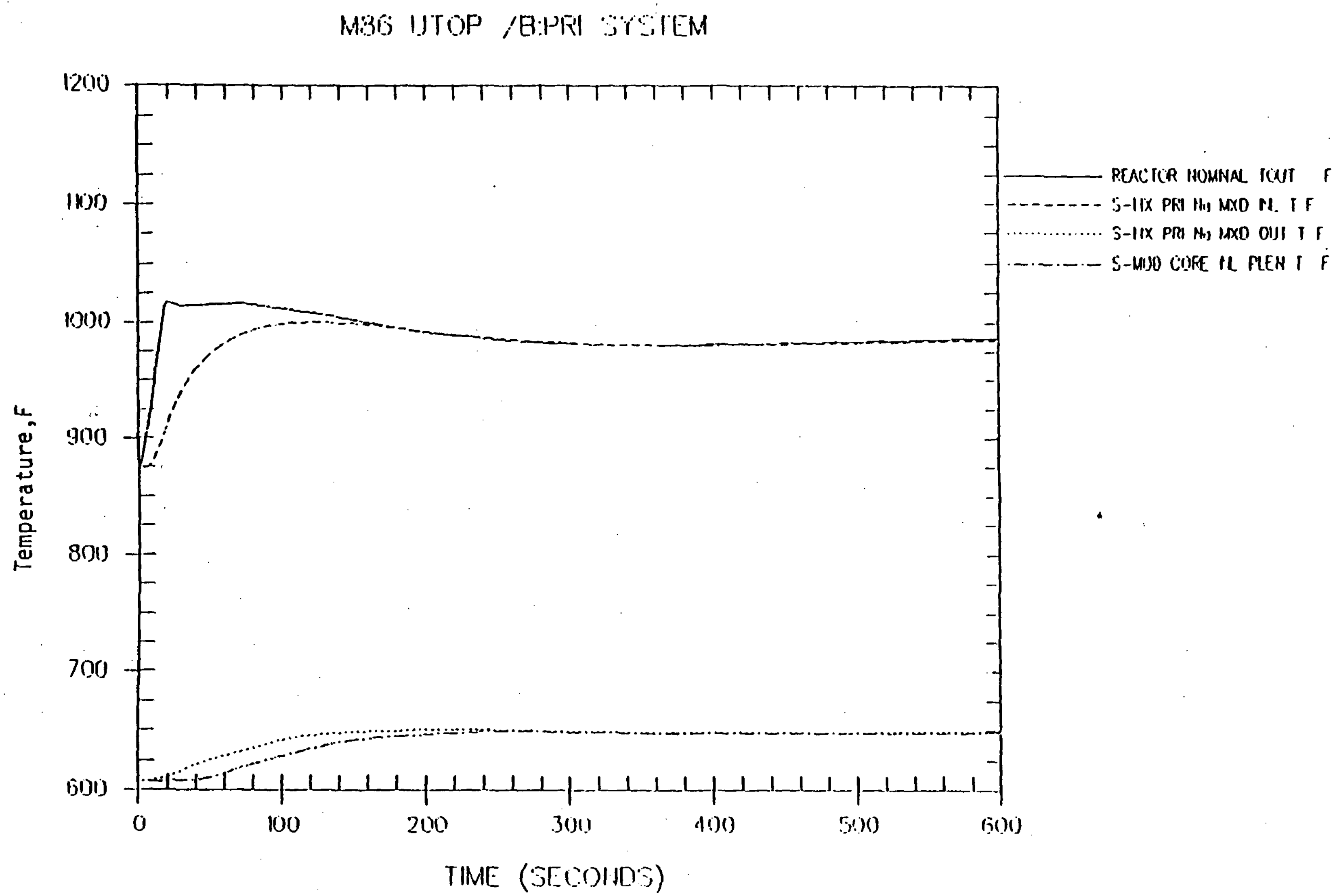

Figure E.3-3 BOEC PRIMARY COOLANT TEMPERATURE RESPONSE TO UNPROTECTED CONTROL ROD WITHDRAWAL (35 CENTS)

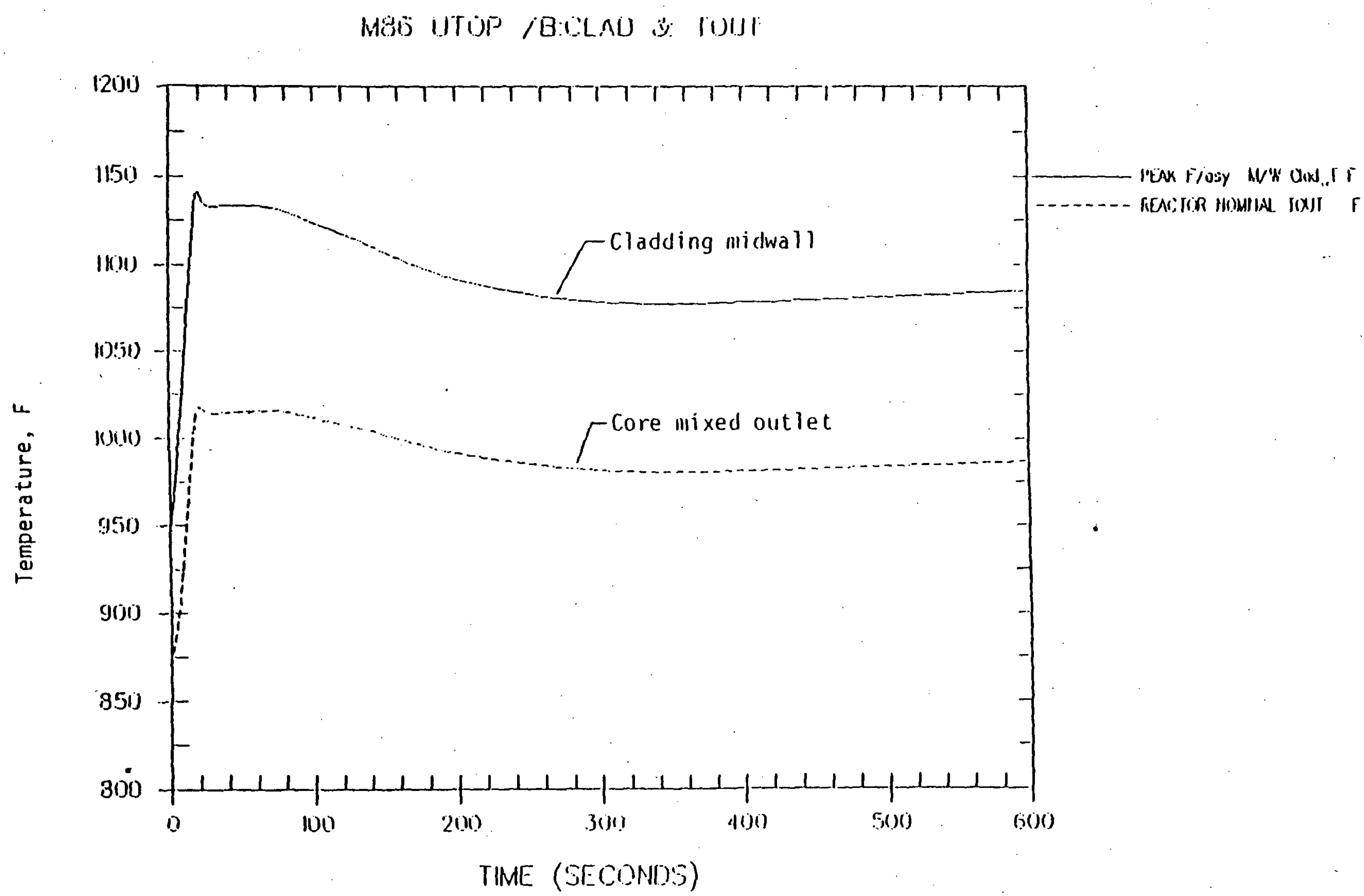

Figure E.3-4 BOEC PEAK FUEL CLADDING AND CORE OUTLET TEMPERATURE RESPONSE TO UNPROTECTED CONTROL ROD WITHDRAWAL (35 CENT)

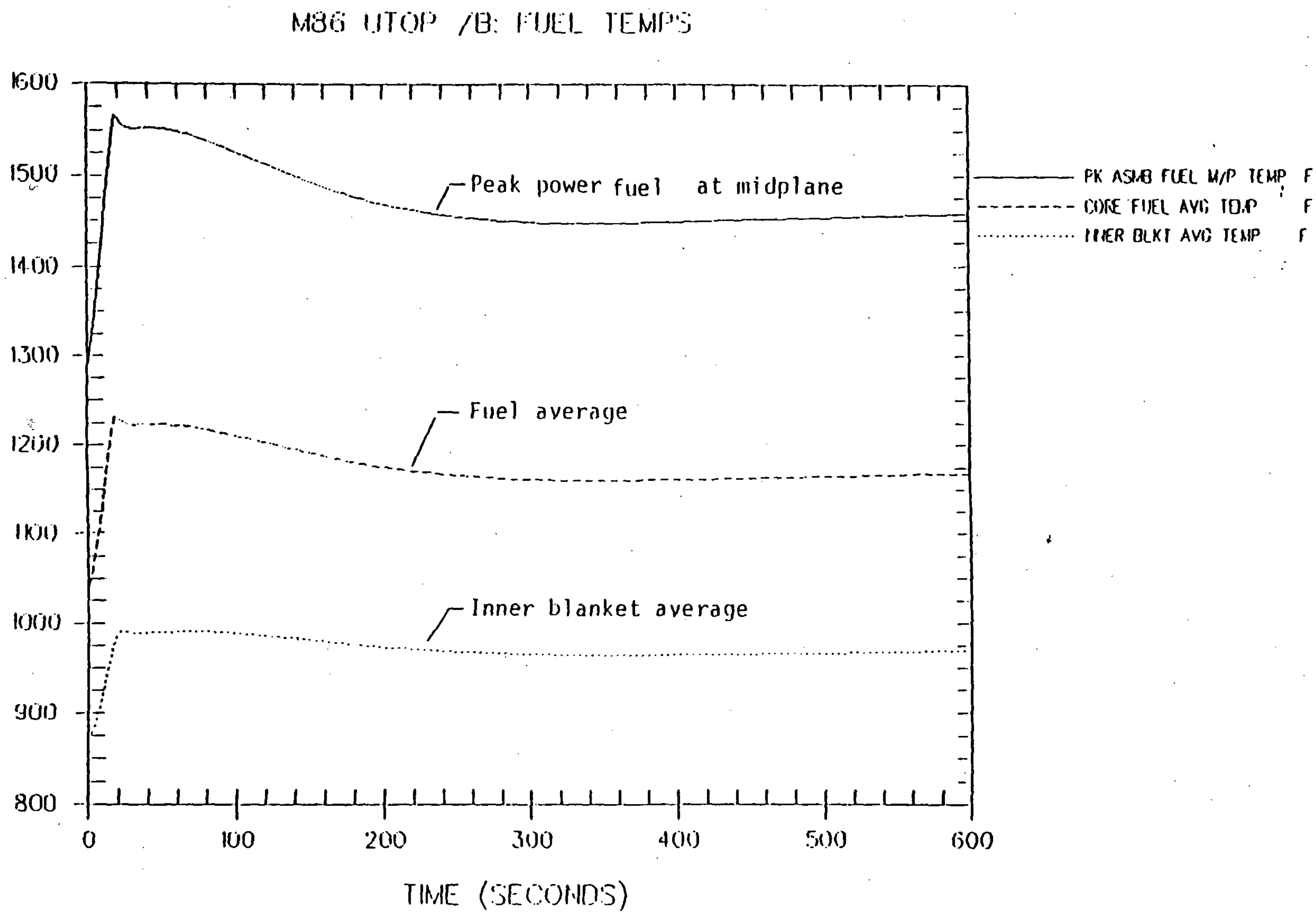

Figure E.3-5 BOEC FUEL TEMPERATURE RESPONSE TO UNPROTECTED CONTROL ROD WITHDRAWAL (35 CENT)

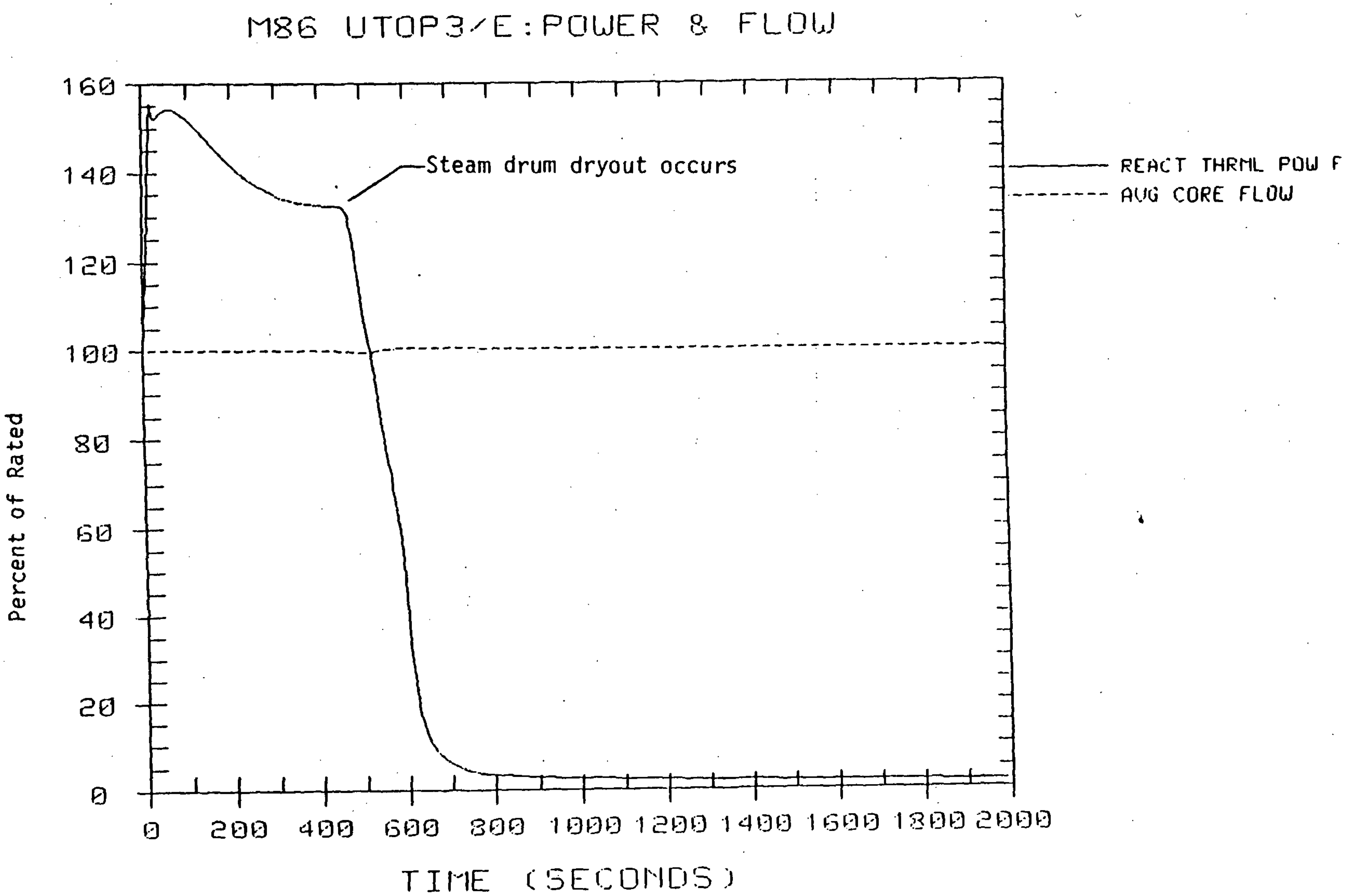

Figure E.3-6 EOEC CORE POWER AND FLOW RESPONSE TO UNPROTECTED CONTROL ROD WITHDRAWAL (35 CENTS)

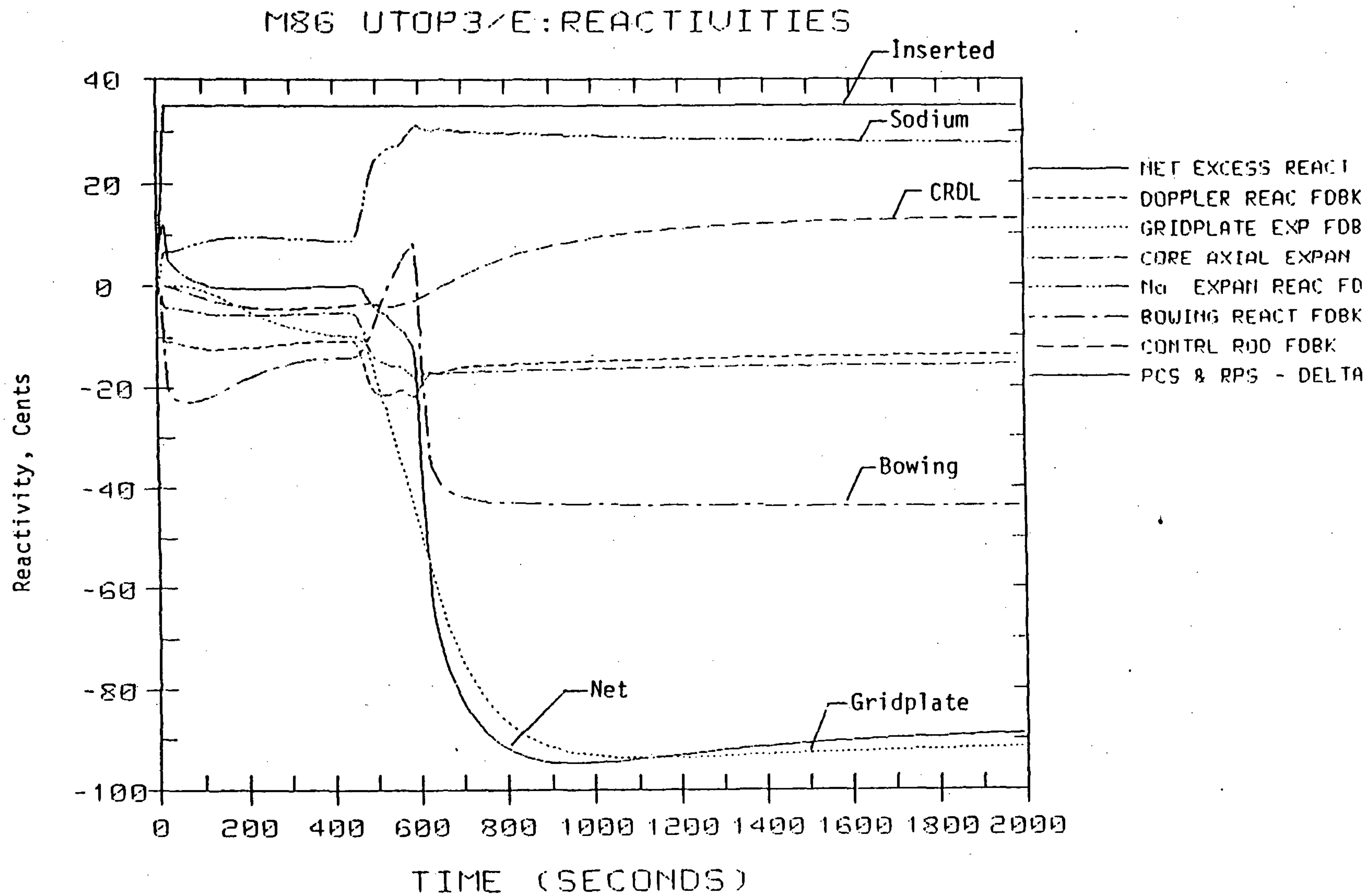

Figure E.3-7 EOEC REACTIVITY RESPONSE TO UNPROTECTED CONTROL ROD WITHDRAWAL (35 CENTS)

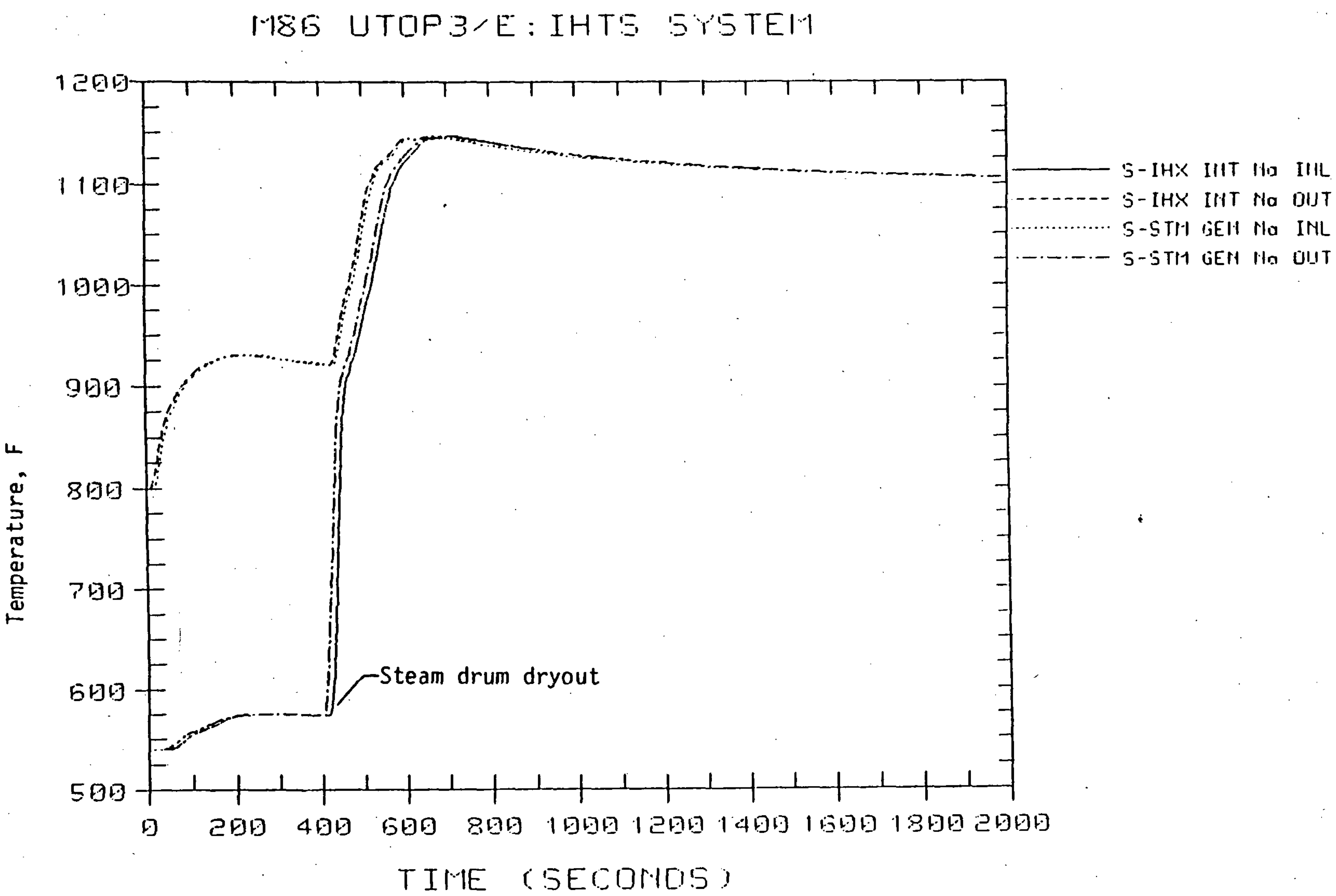

Figure E.3-8 EOEC IHTS TEMPERATURE RESPONSE TO UNPROTECTED CONTROL ROD WITHDRAWAL (35 CENTS)

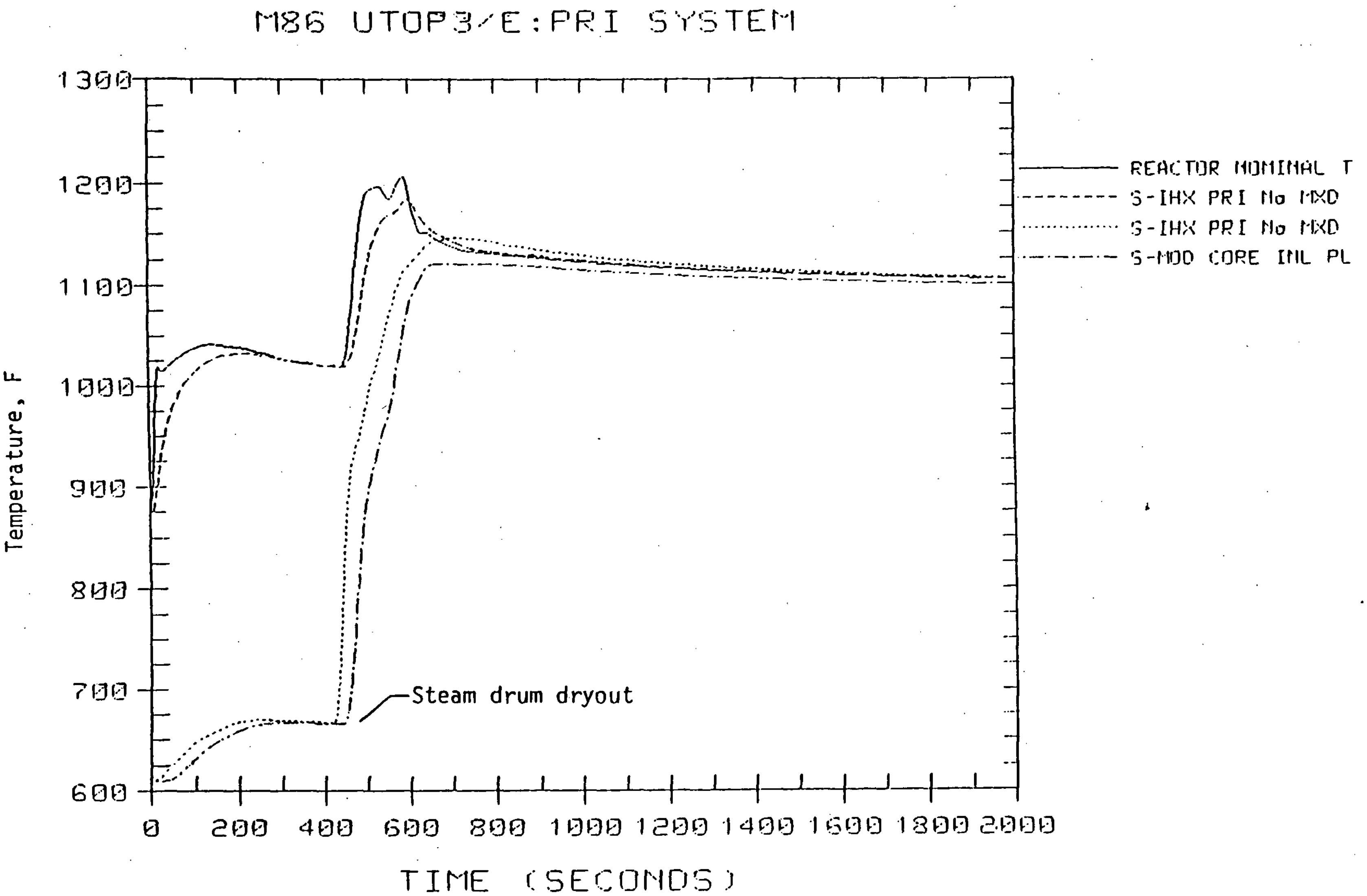

Figure E.3-9 EOEC PRIMARY COOLANT TEMPERATURE RESPONSE TO UNPROTECTED CONTROL ROD WITHDRAWAL (35 CENTS)

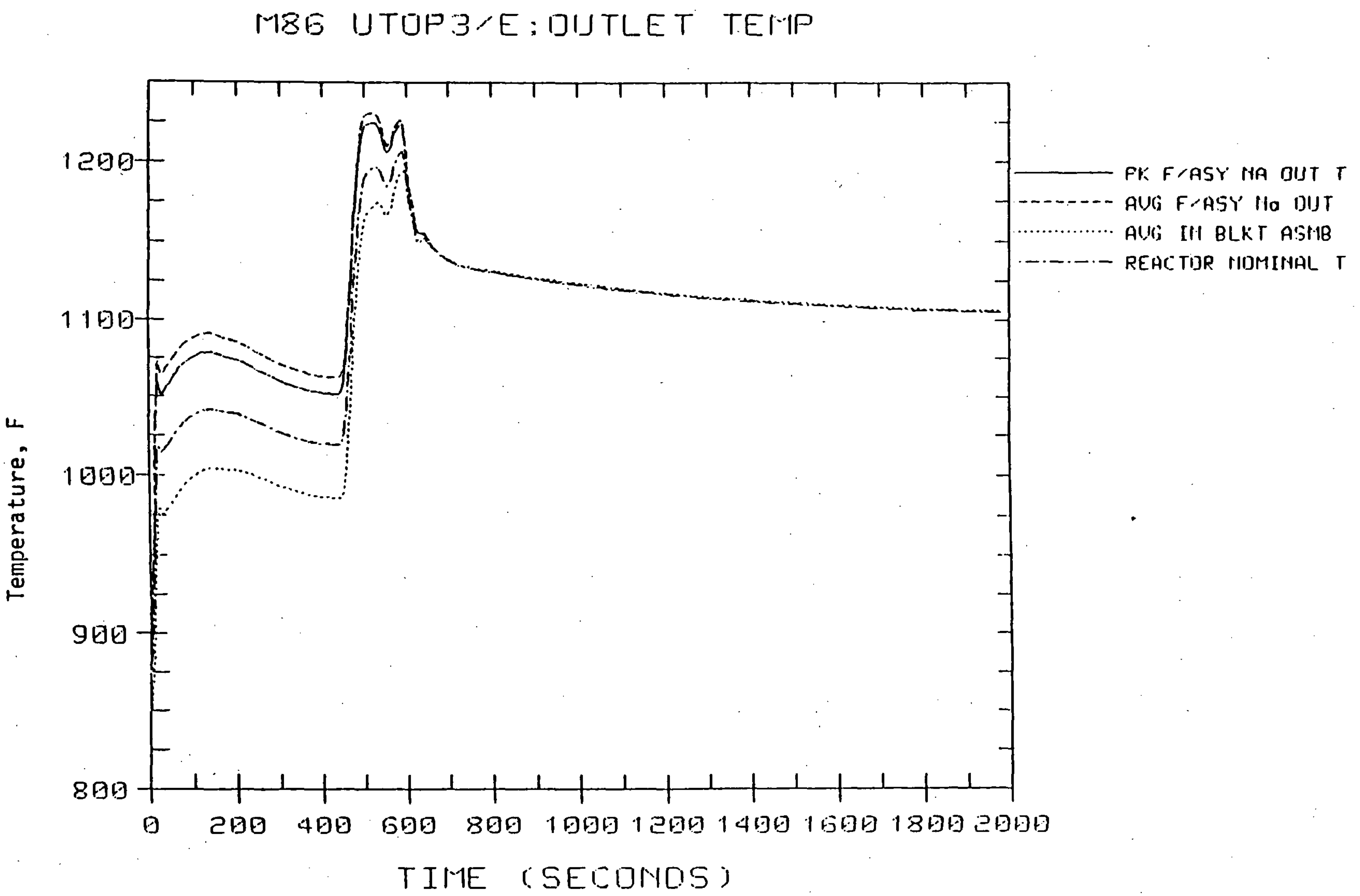

Figure E.3-10 EOEC CORE ASSEMBLY OUTLET TEMPERATURE RESPONSE TO UNPROTECTED CONTROL ROD WITHDRAWAL (35 CENTS)

E 4.0

E.4 Loss of Heat Sink BDBE Performance

This event was analyzed for the BOC core condition due to the positive reactivity contribution from the CRDL net extension. The IHTS flow is arbitrarily assumed to be zero at time zero with the primary pumps still at rated conditions.[2] With the loss of the IHTS, the primary temperature drop across the IHX rapidly collapses and, with full primary flow, the core inlet rapidly heats up to about 980°F thereby raising the overall core temperatures as well. Because of the rapid lower plenum heating the radial expansion of the gridplate becomes a dominant reactivity feedback mechanism (-68 cents). With full pump flow and an upper vessel pool temperature of 980°F, vessel liner overflow did not occur. Hence, rapid upper vessel heating does not occur to pull the control drives away from the core. The net reactivity remains negative throughout the transient leading to a rapid power decrease and isothermal coolant conditions (980°F) in the primary system, as shown in Figures E.4-1 through E.4-4.

The fuel assembly temperatures are all well within the inherency limits (Figure E.4-5 and E.4-6). The core is 33 cents subcritical at low decay power (1.5%) and, with full flow the core delta-T has collapsed. As in the previous TOP case the automatic control system and operator actions will now be of importance since the primary pumps, which contribute a major heat source relative to RVACS capabilities, must be tripped.

[2] Considered to be limiting LOHS event. Other events such as loss of feedwater would have a similar but delayed response.

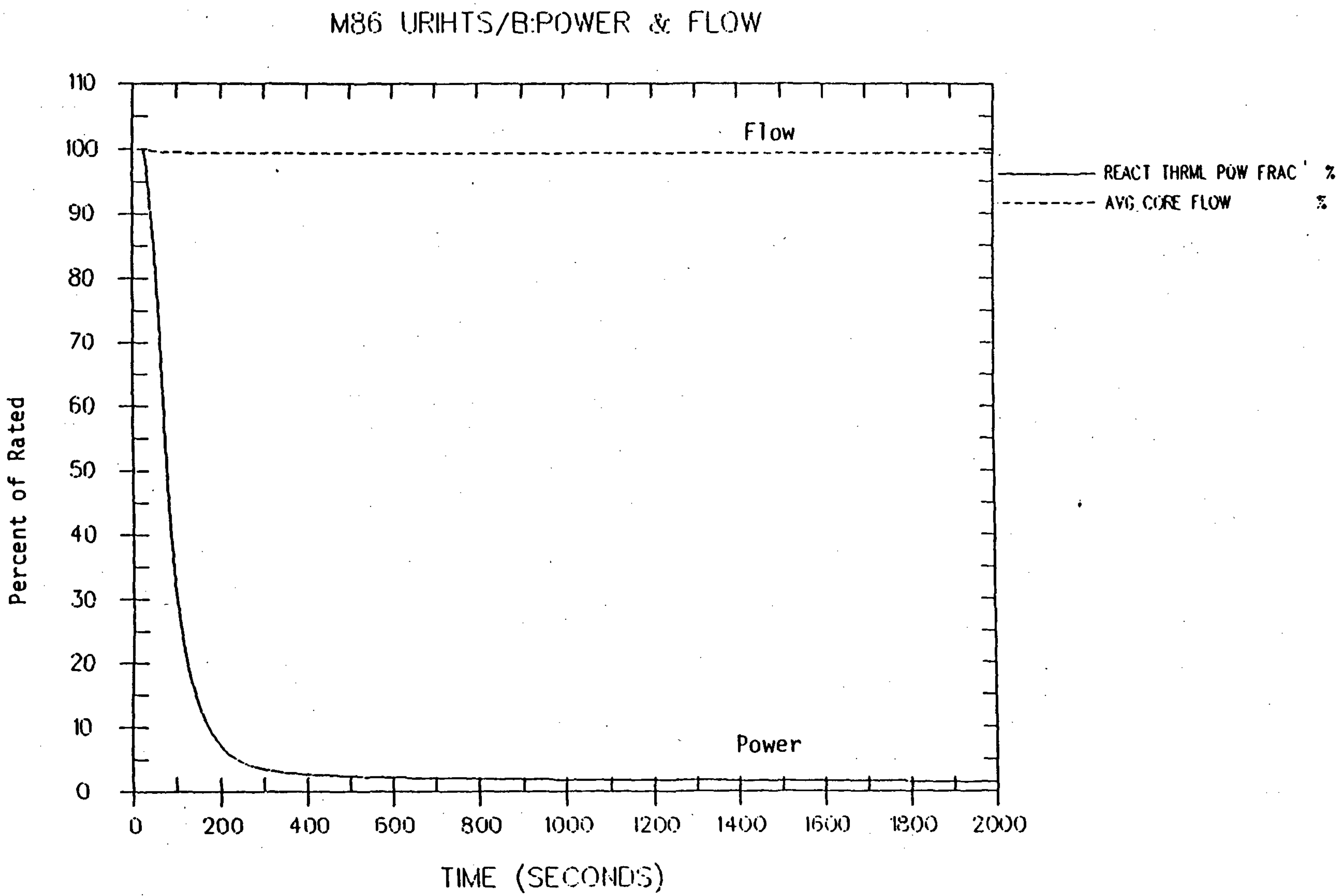

Figure E.4-1 BOEC CORE POWER AND FLOW RESPONSE TO UNPROTECTED LOSS OF IHTS COOLING

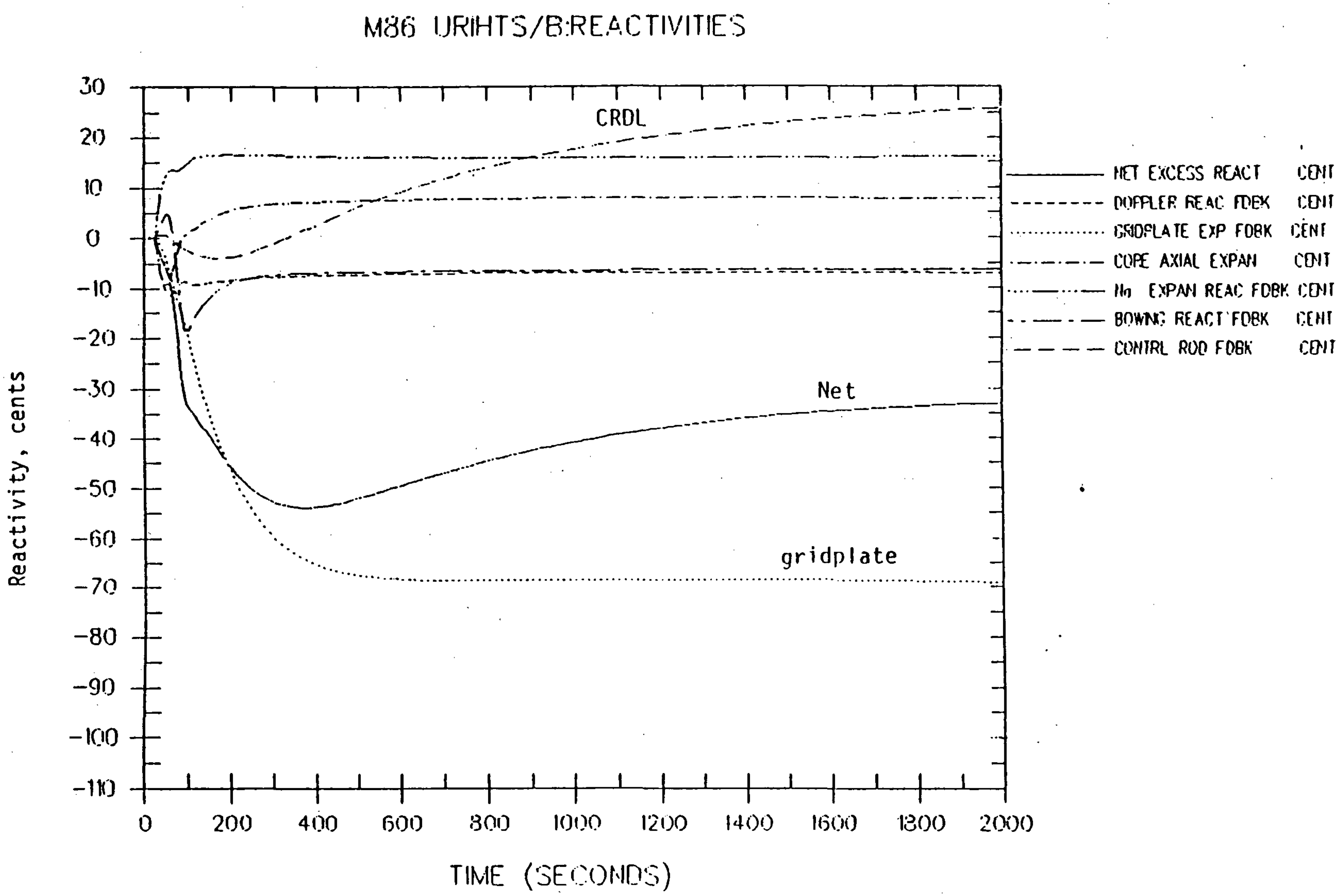

Figure E.4-2 BOEC REACTIVITY RESPONSE TO UNPROTECTED LOSS OF IHTS COOLING

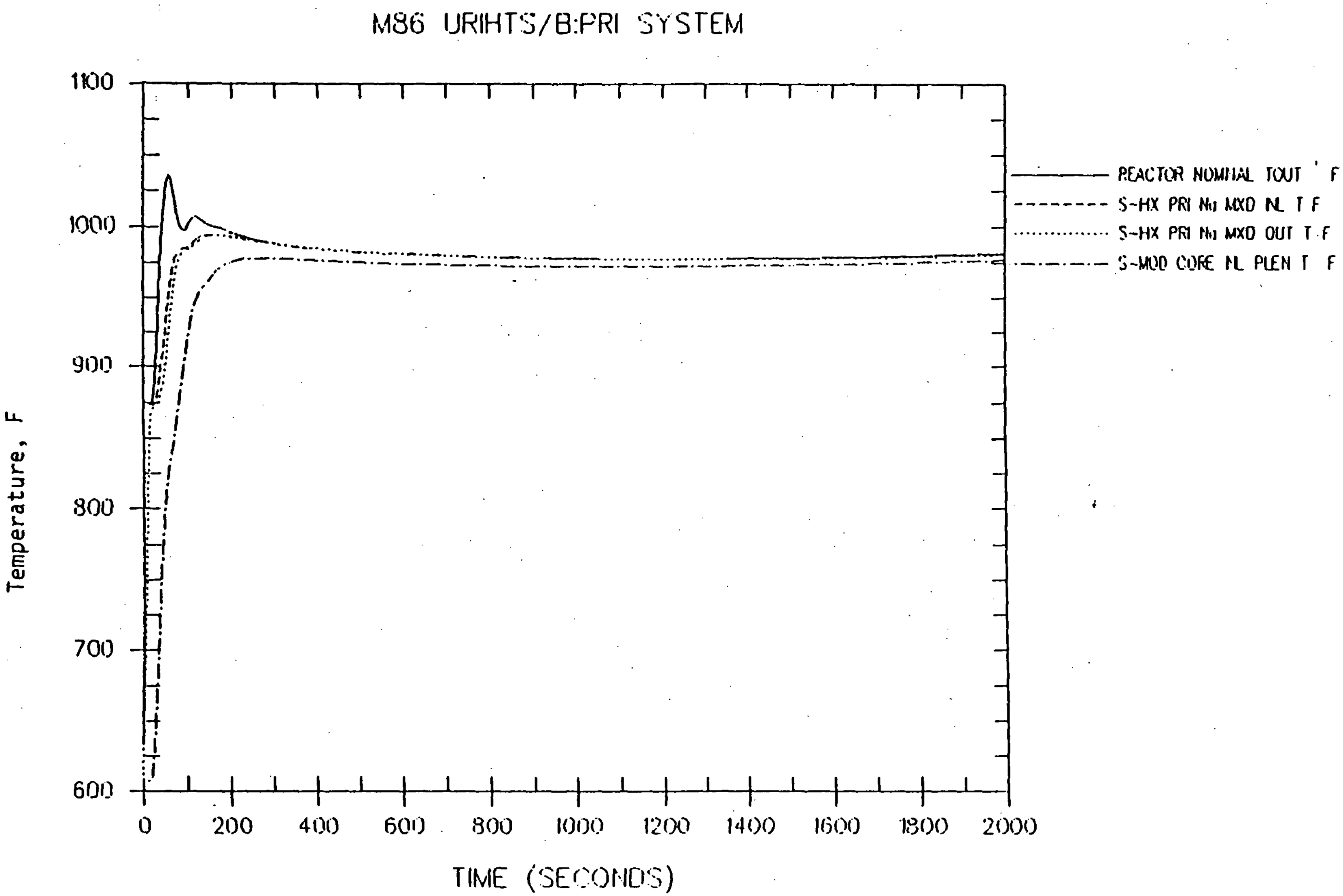

Figure E.4-3 BOEC PRIMARY COOLANT TEMPERATURE RESPONSE TO UNPROTECTED LOSS OF IHTS COOLING

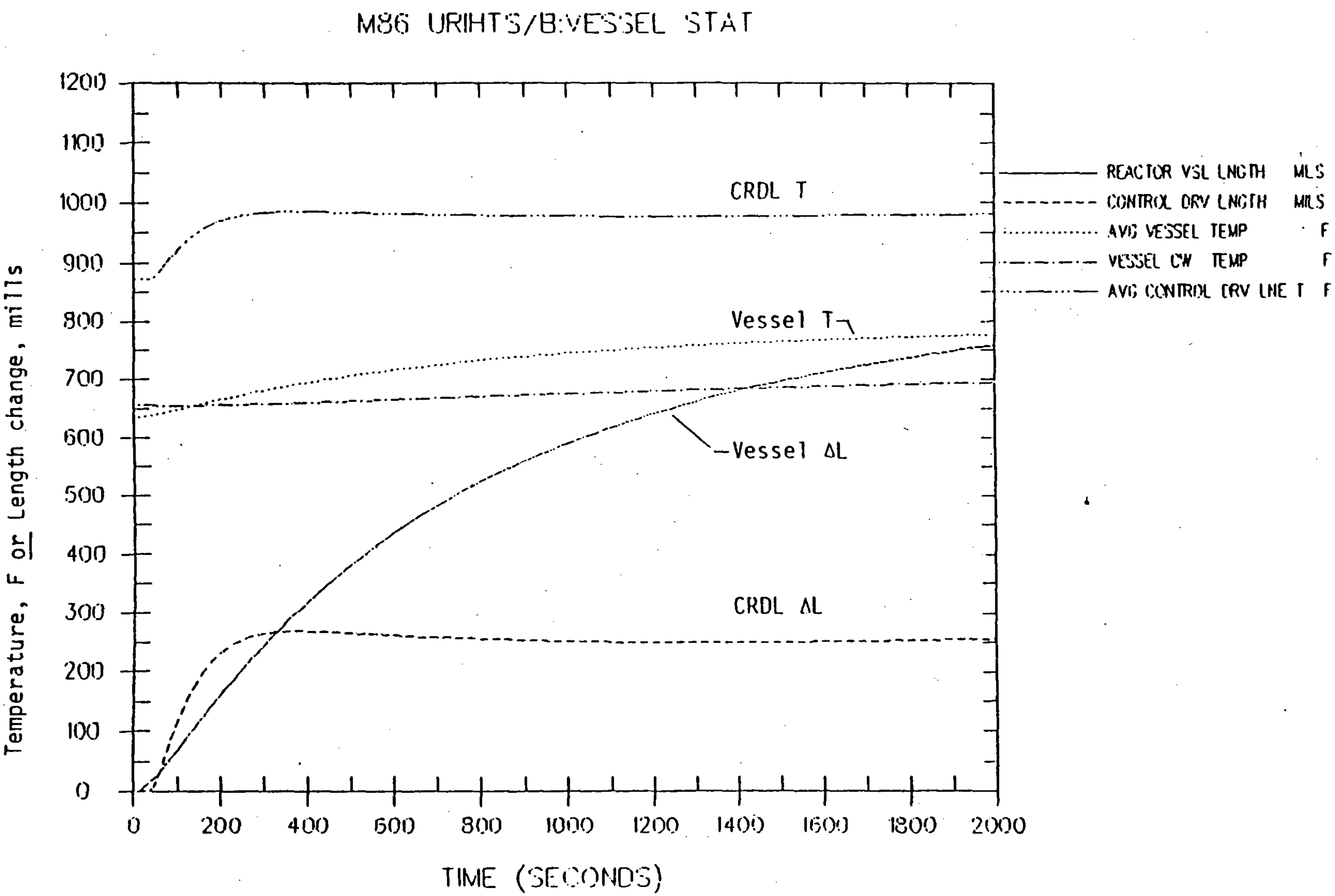

Figure E.4-4 BOEC VESSEL COMPONENTS WHICH DETERMINE CRDL EXTENSION DURING ULOHS

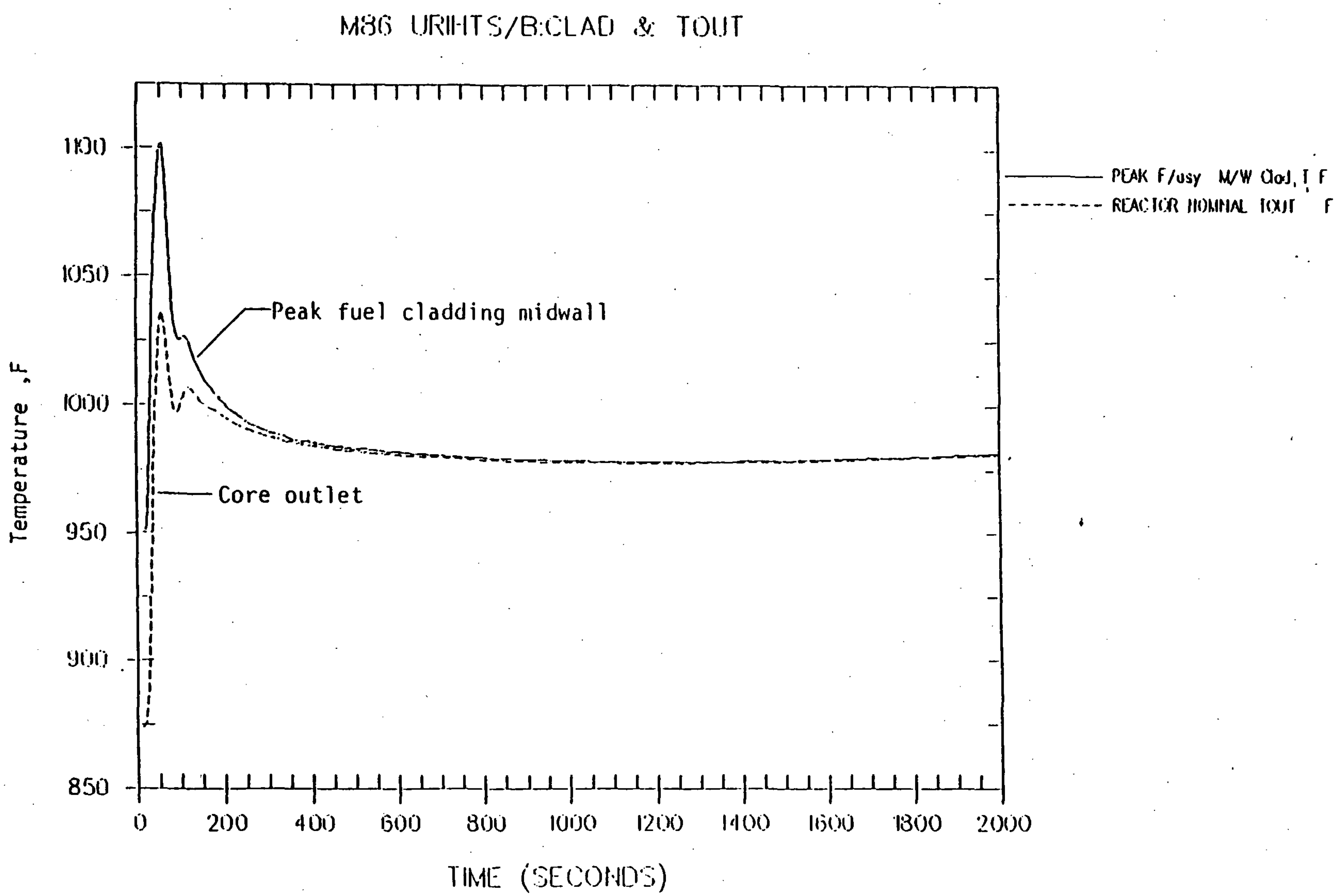

Figure E.4-5 BOEC PEAK CLADDING AND CORE OUTLET RESPONSE TO UNPROTECTED LOSS OF IHTS COOLING

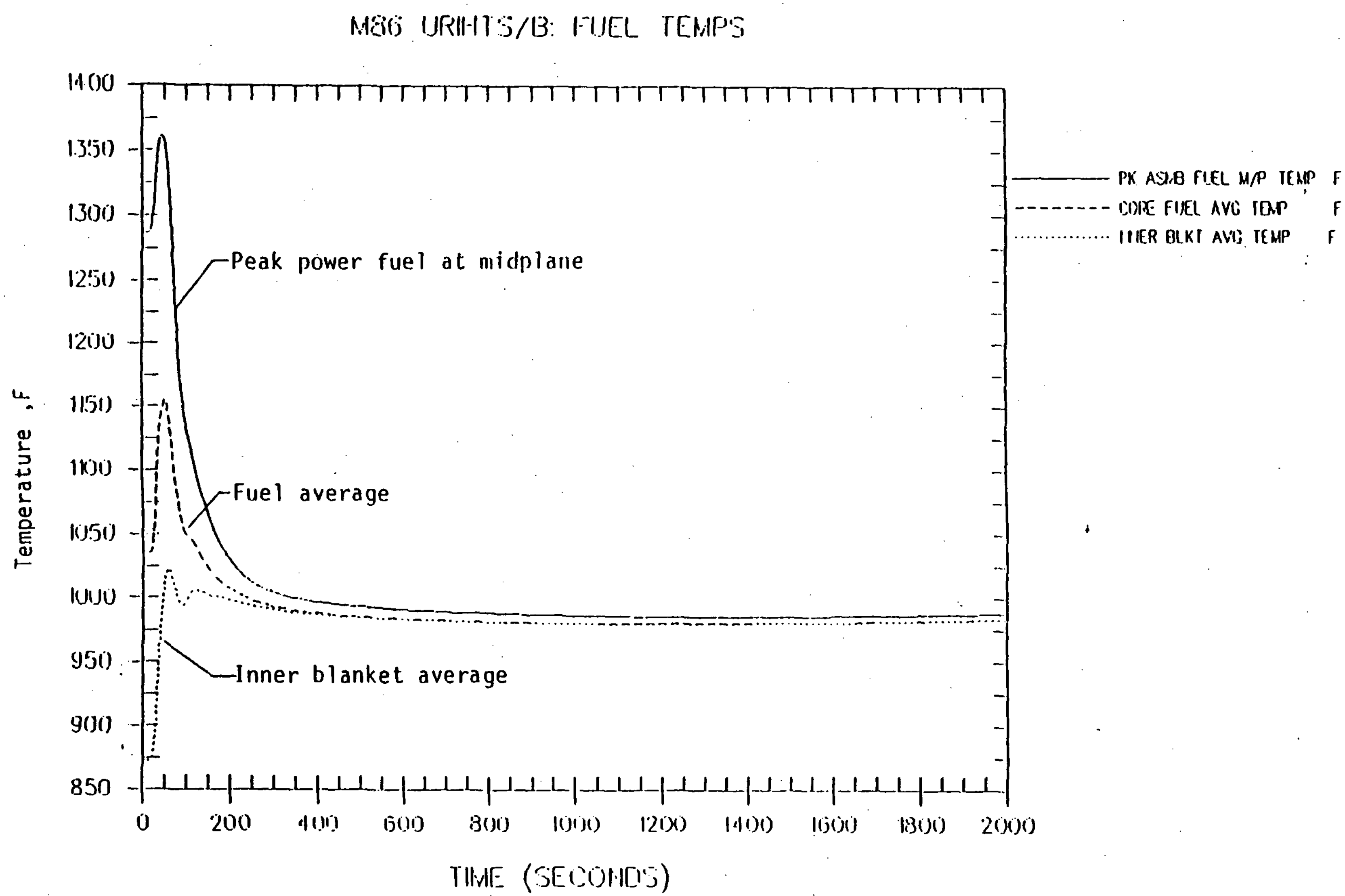

Figure E.4-6 BOEC FUEL TEMPERATURE RESPONSE TO UNPROTECTED LOSS OF IHTS COOLING

E 5.0

E.5 Summary of BDBE Response Evaluations

Analyses were performed on the PRISM metal core and plant systems to evaluate the overall inherent safety performance achieved. The calculations addressed the unprotected scenarios for loss-of-flow (LOF), loss-of-heat sink (LOHS) and single control rod withdrawal (TOP) all initiated from 100% rated plant conditions. The loss of cooling scenarios also assumed the instantaneous stoppage of the IHTS flow.

Figures E.5-1 through E.5-6 compare the response of the PRISM metal core and PHTS design during these generic BDBE's with the adopted inherency limits of:

- No boiling in peak fuel assembly (∿1800°F)
- Cladding midwall short-term limit of 1450°F
- Reactor structure long-term limit of 1300°F
- No fuel melting (∿2000°F)
- Fuel-cladding interface below onset of rapid eutectic formation (1290°F).

These nominal basis evaluations indicate that the metal core design has satisfied all of the criteria stated above, with margin. The fuel cladding interface temperature limit of 1290F is satisfied based on its being less than the long term peak fuel temperatures shown in Figure E.5-4.

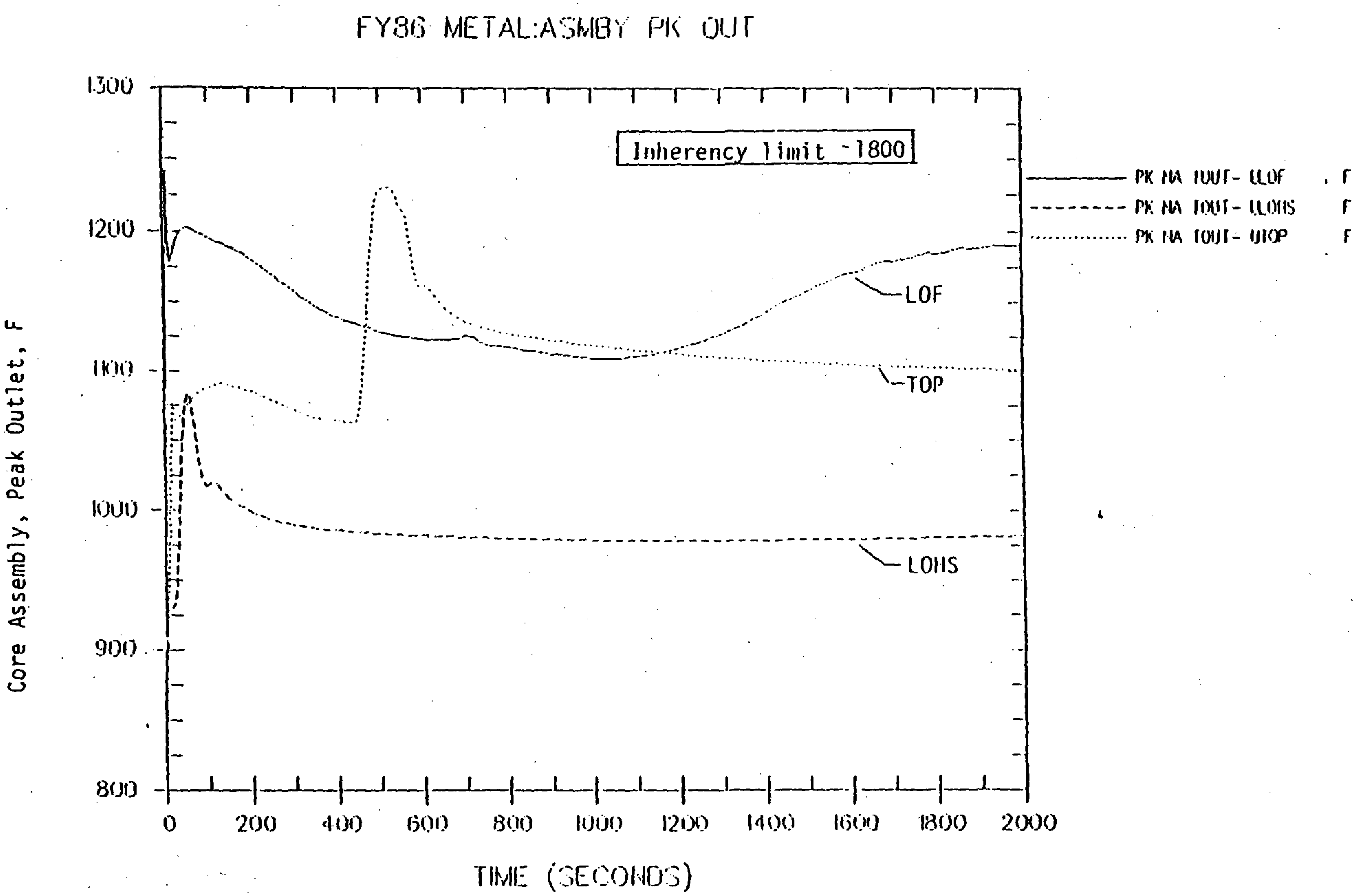

Figure E.5-1 EVALUATION OF INHERENCY LIMIT FOR NO FUEL ASSEMBLY BOILING FOR THREE BEYOND DESIGN BASIS EVENTS

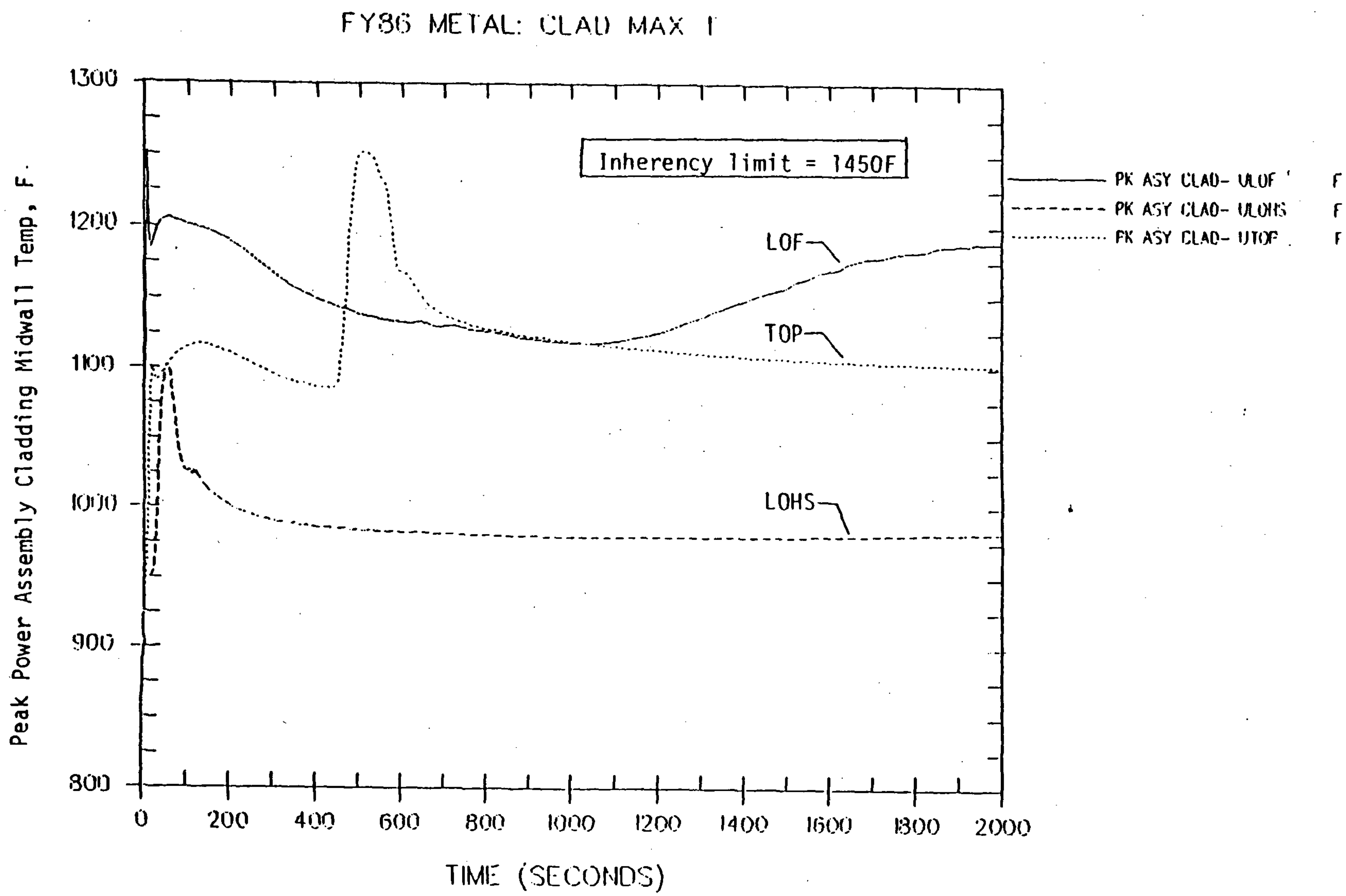

Figure E.5-2 EVALUATION OF INHERENCY LIMIT ON CLADDING TEMPERATURE FOR THREE UNPROTECTED BEYOND DESIGN BASIS EVENTS

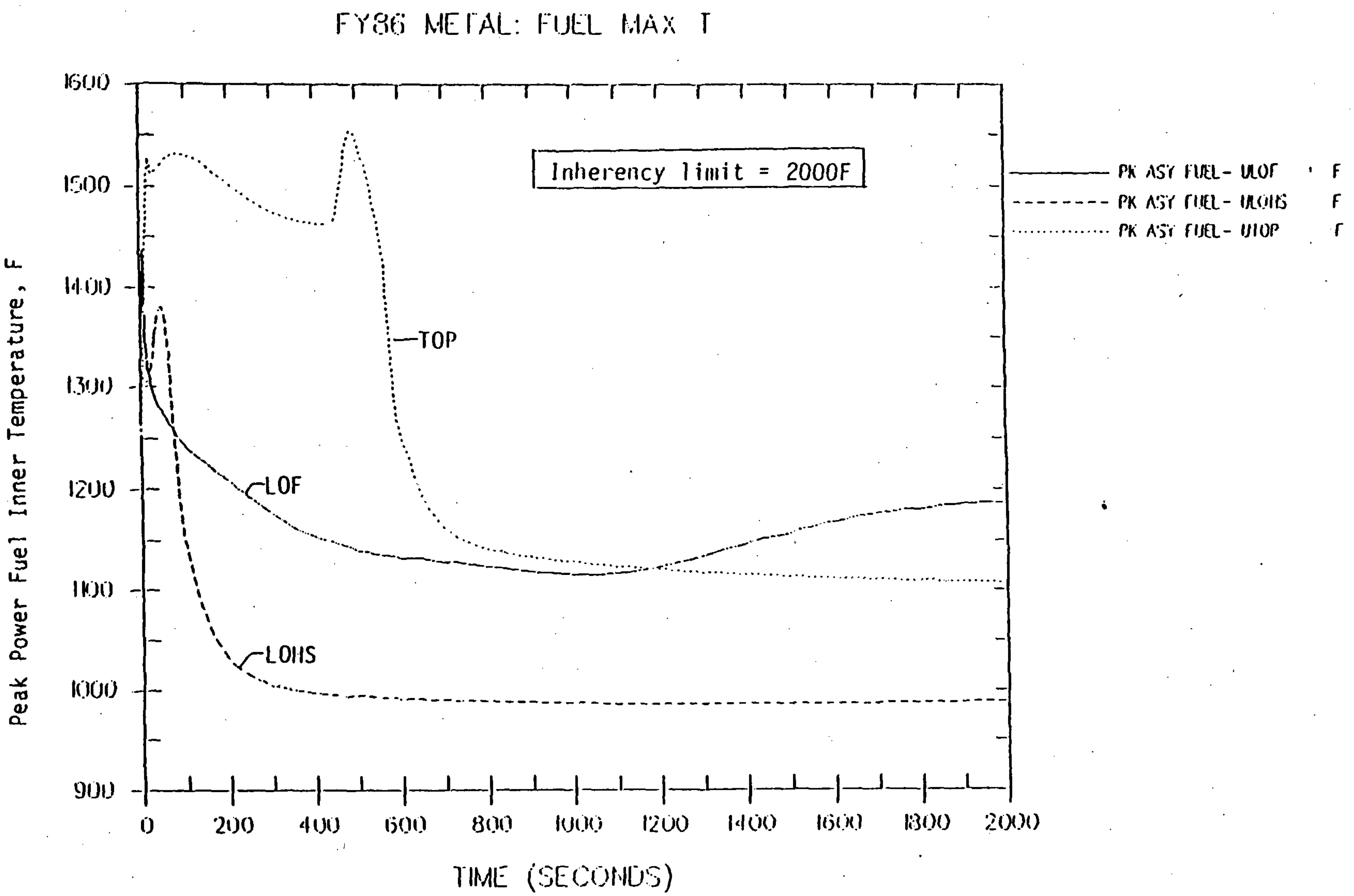

Figure E.5-3 EVALUATION OF INHERENCY LIMIT OF NO FUEL MELTING FOR THREE BEYOND DESIGN BASIS EVENTS

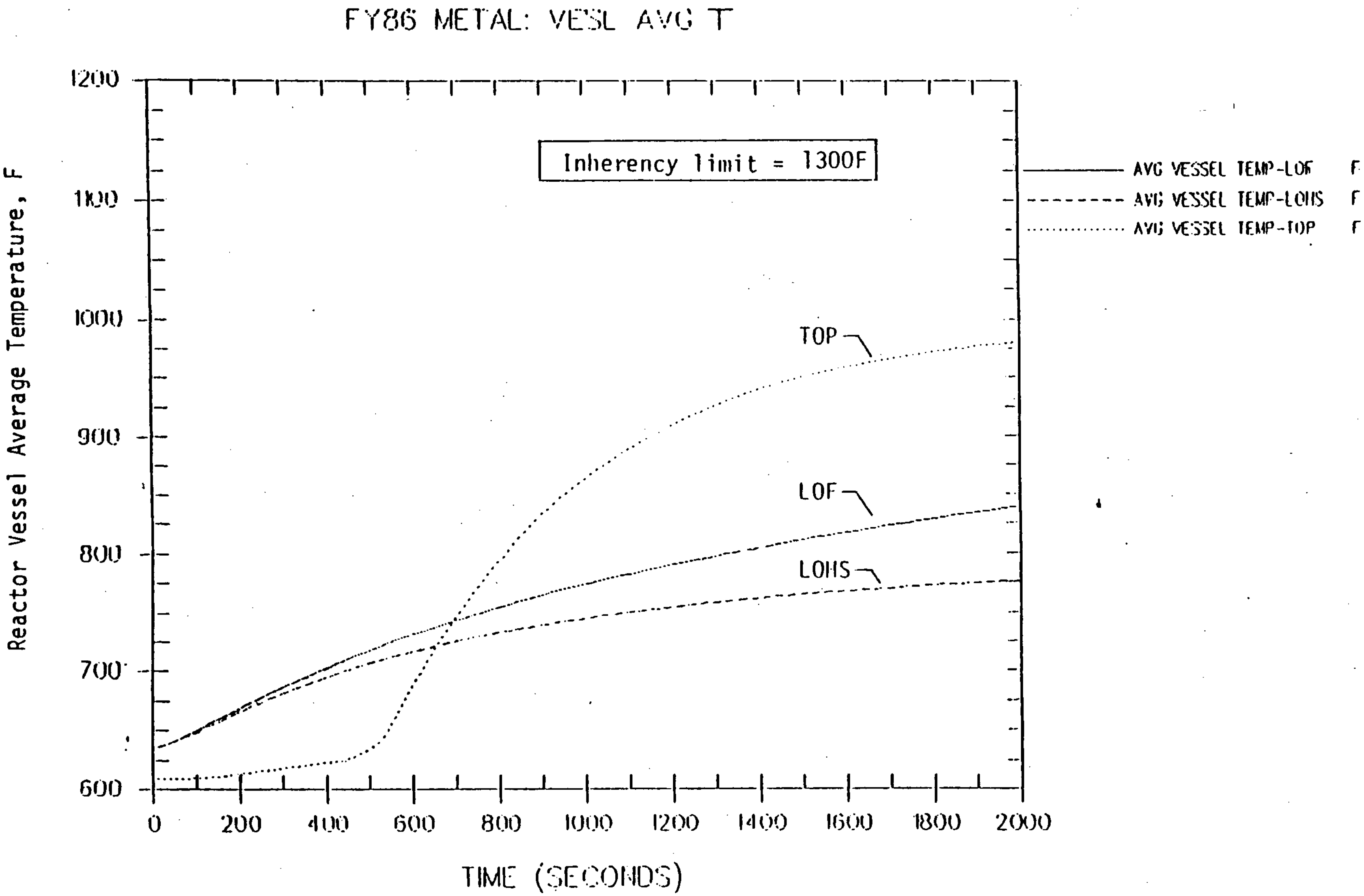

Figure E.5-4 EVALUATION OF INHERENCY LIMIT ON REACTOR STRUCTURES FOR THREE BEYOND DESIGN BASIS EVENTS

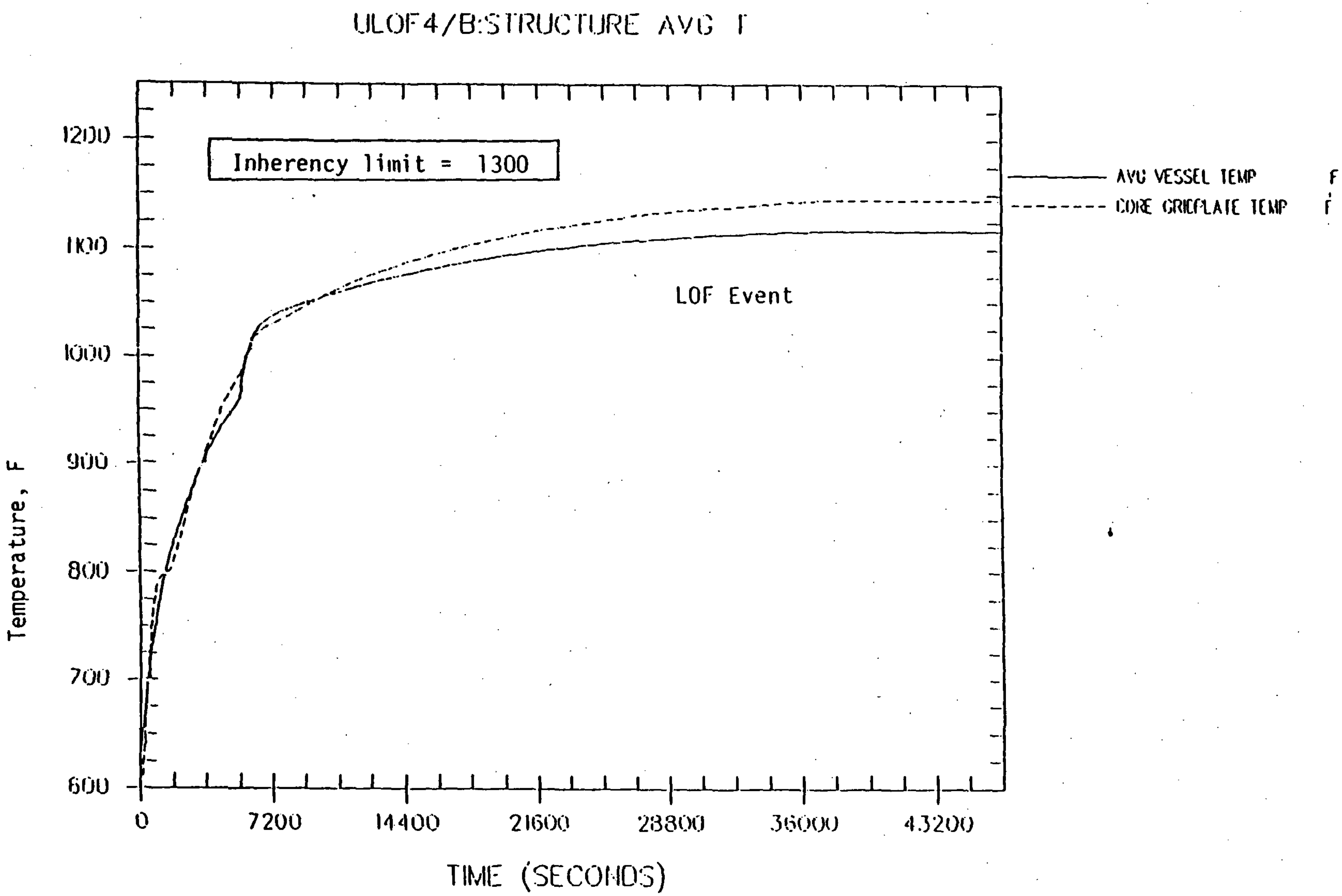

Figure E.5-5 COMPARISON OF STRUCTURE TEMPERATURE WITH INHERENCY LIMIT FOR LOSS OF FLOW AND LOSS OF IHTS COOLING BDBE