HTGR-86-011
Revision 1

DRAFT

HTGR

PROBABILISTIC RISK ASSESSMENT OF THE MODULAR HTGR PLANT

AUTHORS/CONTRACTORS

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1. SUMMARY

A preliminary probabilistic risk assessment (PRA) has been performed for the modular HTGR (MHTGR). This PRA is preliminary in the context that although it updates the PRA issued in January to include a wider spectrum of events for LBE selection, the final version will not be issued until September. There are two basic reasons why a September issue is necessary:

1. To bring the PRA into agreement with the MHTGR design described in the PSID (this preliminary PRA addresses the design as it was envisioned in late January).

2. To analyze in greater detail some of the dominant safety risk contributions recently identified.

The primary function of the assessment was to assure compliance with the NRC interim safety goals imposed by the top-level regulatory criteria (Ref. 1-1), and utility/user requirements regarding public evacuation or sheltering (from Ref. 1-2). In addition, the assessment provides a basis for designer feedback regarding reliability allocations and barrier retention requirements as well as providing a basis for the selection of licensing basis events (LBEs) and the safety classification of structures, systems, and components. The assessment demonstrates that both the NRC interim safety goals and utility/user imposed sheltering/evacuation requirements are satisfied. Moreover, it is not anticipated that design changes introduced since January will jeopardize compliance with the interim safety goals or utility/user requirements.

A major concern in performing a PRA is completeness. Specifically, this concern is whether there are events, not included in the event trees, that can appreciably increase the predicted plant safety risk envelope. One technique for assuaging this concern is to characterize plant safety with a Master Logic Diagram (MLD). By applying this technique to the MHTGR, a broad event spectrum was evaluated, and events potentially significant relative to plant safety were identified. The initiating events analyzed in this PRA were selected as a result of this safety characterization. Only some of the events considered important in the MLD will be evaluated this year, due to budget and schedule limitations. Evaluating the remaining events will be an important element in next year's PRA task.

Two criteria were utilized to select initiating events for evaluation in this analysis:

1. The initiating events were potentially dominant safety risk contributors, or

2. The initiating events were important for bridging.

The inclusion of initiating events of importance relative only to bridging (i.e., initiating events that contribute negligibly to the overall plant safety risk envelope) is necessary in order to demonstrate MHTGR safety in the PSID.

Seven initiating events are evaluated in the analysis:

1. Primary coolant leaks.
2. Steam generator leaks.
3. Loss of the Heat Transport System (HTS).
4. Loss of offsite power and inadvertent turbine trip.
5. Seismic activity.
6. Control rod bank withdrawal.
7. Anticipated transients requiring reactor scram.

Of these events, loss of the HTS, loss of offsite power and inadvertent turbine trip, and anticipated transients requiring reactor scram are included solely because of their importance to bridging.

From the seven initiating events selected, four accident families were found to result in an offsite release: primary coolant leaks, steam generator leaks, conduction cooldowns (which include all control rod bank withdrawal initiated events that result in offsite doses), and earthquakes. The salient risk assessment results for these accident families are summarized in Table 1-1. Primary coolant leaks involve the leakage of fission products through the primary coolant pressure boundary into the reactor building at a rate sufficient to cause a reactor trip. Members of the steam generator leak accident family are all initiated by a small or moderate steam generator leak and include at least one subsequent failure that results in fission product release from the reactor vessel to the reactor building or atmosphere. The conduction cooldown accident family is composed of primary coolant leak, steam generator leak, control rod bank withdrawal, and earthquake initiated accidents in which forced convection core cooling is lost. Events initiated by earthquakes where forced convection cooling is maintained involve primary coolant boundary failure and release of fission products from the reactor vessel into the reactor building.

As indicated in Table 1-1, the results of the risk assessment with respect to the NRC interim safety goals (from Ref. 1-1) predict no acute fatalities and 1×10^{-8} latent fatalities per plant year The NRC interim safety goals of 5×10^{-7} acute fatalities per plant year and 2×10^{-6} latent fatalities per plant year are thus readily satisfied. With respect to utility/user safety requirements (Ref. 1-2), all accident families have been shown to fall below the Protective Action Guide (PAG) dose limits at a frequency of 5×10^{-7} per plant year. In addition to meeting the goals individually, accumulation of all accident families indicated compliance with the utility/user safety goals of 5 rem to the thyroid and 1 rem to the whole body at the plant Exclusion Area Boundary (EAB) as well. Table 1-2 compares the expected consequences of important release categories belonging to each accident family given in Table 1-1 with the utility/user requirements for sheltering/evacuation derived from the PAGs. Release categories with mean frequencies above 0.025 (at least once in the lifetime of the plant) are compared to the 10CFR50 Appendix I annualized risk requirements in Table 1-3. In order to provide assurance that all appropriate release categories are included, those whose mean frequencies fall below 0.025 within a factor of 2

TABLE 1-1
COMPARISON OF THE MODULAR HTGR OVERALL RISK
TO NRC INTERIM SAFETY GOALS

Accident Family	Risk Aspect
Primary Coolant Leaks	Mean Risk = 0 Acute fatalities/plant year 8×10^{-9} Latent fatalities/plant year Mean Frequency = 0.1/plant year Mean Consequence = 6×10^{-4} rem, whole body 5×10^{-4} rem, thyroid
Steam Generator Leaks	Mean Risk = 0 Acute fatalities/plant year 4×10^{-11} Latent fatalities/plant year Mean Frequency = 2×10^{-4}/plant year Mean Consequence = 2×10^{-3} rem, whole body 5×10^{-2} rem, thyroid
Depressurized Conduction Cooldowns	Mean Risk = 0 Acute fatalities/plant year 2×10^{-9} Latent fatalities/plant year Mean Frequency = 4×10^{-4}/plant year Mean Consequence = 4×10^{-2} rem, whole body 3×10^{-2} rem, thyroid
Earthquakes	Mean Risk = 0 Acute fatalities/plant year 3×10^{-10} Latent fatalities/plant year Mean Frequency = 3×10^{-4}/plant year Mean Consequence = 9×10^{-3} rem, whole body 9×10^{-3} rem, thyroid

TABLE 1-1
(Continued)

Accident Family	Risk Aspect
Total	Total Mean Risk = 0 Acute fatalities/plant year 1×10^{-8} Latent fatalities/plant year <u>NRC Interim Safety Goals (from Ref. 1-1)</u> 5×10^{-7} Acute fatalities/plant year 2×10^{-6} Latent fatalities/plant year

TABLE 1-2
COMPARISON OF MODULAR HTGR EVENTS TO OFFSITE
SHELTERING/EVACUATION REQUIREMENTS

Release* Category	Mean Frequency (per plant year)	Mean Consequence (rem) Whole Body	Mean Consequence (rem) Thyroid	Utility/User Requirement (Ref. 1-2) (rem) Whole Body	Utility/User Requirement (Ref. 1-2) (rem) Thyroid
PC-3	6×10^{-6}	3×10^{-3}	7×10^{-3}	1	5
PC-4	5×10^{-3}	3×10^{-3}	4×10^{-3}	1	5
PC-5	1×10^{-5}	2×10^{-3}	5×10^{-2}	1	5
PC-6	2×10^{-2}	2×10^{-3}	2×10^{-3}	1	5
PC-10	0.1	2×10^{-4}	6×10^{-6}	1	5
CC_s-14	2×10^{-6}	2×10^{-3}	0.1	1	5
S/G-3	5×10^{-6}	1×10^{-2}	0.6	1	5
S/G-9	7×10^{-5}	2×10^{-3}	0.1	1	5
S/G-4	1×10^{-4}	5×10^{-4}	5×10^{-3}	1	5
CC_p-9	1×10^{-4}	5×10^{-2}	2×10^{-2}	1	5
CC_p-12	2×10^{-4}	2×10^{-4}	6×10^{-6}	1	5
EQ-1	3×10^{-4}	9×10^{-3}	1×10^{-2}	1	5
CC_e-2	7×10^{-5}	0.2	0.1	1	5
CC_e-3	9×10^{-7}	0.9	6×10^{-2}	1	5

*See Table 1-4 for release category descriptions.

TABLE 1-3
COMPARISON OF MODULAR HTGR EVENTS TO
APPENDIX I REQUIREMENTS

Release Category	Mean Frequency (per plant year)	Mean Consequence (mrem) Whole Body	Mean Consequence (mrem) Thyroid	Annualized Dose (mrem per plant year) Whole Body	Annualized Dose (mrem per plant year) Thyroid
	Normal Operation	0.05	0.15	0.05	0.15
PC-6	0.02	2	2	0.04	0.04
PC-10	0.1	0.2	0.006	0.02	6×10^{-4}
		Total Annual Dose		0.11	0.19
		Appendix I Limit		5.0	15.0

are also included in the annualized risk tabulation. It was assumed for the purposes of goal comparison that releases from normal operation on a yearly basis were 1% of the allowables as they have not as yet been evaluated (this evaluation is not part of the PRA). Descriptions of release categories presented in Tables 1-2 and 1-3 are given in Table 1-4.

The results of the risk assessment show that the primary coolant leak family of accidents dominates safety risk for the MHTGR at high frequencies. Primary coolant leak consequences, which result from the release of circulating activity and some liftoff of material plated out on primary circuit surfaces, are low relative to other accident families identified. The conduction cooldown family of accidents is the most consequential with respect to whole body gamma doses because of the increased release of noble gases during the core thermal transient. Steam generator leaks are on the average the largest contributors to thyroid doses because of the increased release of iodines from initially failed fuel hydrolysis.

In the primary coolant leak family of accidents, PC-6 dominates both thyroid and latent fatality risk (in terms of rem per plant year) and dominates these risks for all accident families identified in the assessment as well. Release category PC-5 dominates thyroid doses for primary coolant leaks because of continued fan operation in the reactor building. Whole body gamma doses for primary coolant leaks are dominated by PC-3 and PC-4 which encompass the larger leak areas identified.

With respect to steam generator leaks, S/G-3 (involving a moderate water ingress with primary relief valve failure) results in the largest whole body gamma and thyroid doses. S/G-3 is also the dominant thyroid dose contributor for all identified accident families with a mean frequency above 5×10^{-7} per plant year. Release category S/G-9 (similar to S/G-3 except relief valve failure does not occur) is the dominant source of both thyroid and latent fatality risk in the steam generator leak accident family.

The conduction cooldown family of accidents is dominated by release category CC_e-2 which is initiated by seismic activity (see Table 1-4). CC_e-2

TABLE 1-4
RELEASE CATEGORY DESCRIPTIONS

Release Category	Description
PC-3	Large primary coolant leak where 13 in.$^2 \leq A <$ 30 in.^{2} occurs. HTS or SCS core cooling is maintained. HPS pumpdown and reactor building are ineffective. Release is through the dampers.
PC-4	Moderate primary coolant leak where 1 in.$^2 \leq A <$ 13 in.^{2} occurs. HTS or SCS core cooling is maintained. HPS pumpdown and reactor building are ineffective. Release is through the dampers.
PC-5	Small primary coolant leak where 3×10^{-2} in.$^2 \leq A <$ 1 in.^{2} occurs. HTS or SCS core cooling is maintained. Reactor building fans fail to disengage, filters are not isolated. Release is through the dampers.
PC-6	Small primary coolant leak where 3×10^{-2} in.$^2 \leq A <$ 1 in.^{2} occurs. HTS or SCS core cooling is maintained. HPS pumpdown occurs. Reactor building functions properly. Release is through the dampers.
PC-10	Very small primary coolant leak where 3×10^{-5} in.^{2} $\leq A < 3 \times 10^{-2}$ in.^{2} occurs. HTS or SCS core cooling is maintained. HPS pumpdown occurs. Reactor building functions properly. Release is initially through the dampers. Subsequent release is by building leakage.
EQ-1	An earthquake occurs with an intensity greater than 0.4 g. Reactor trip of all four modules occurs. HTS cooling is unavailable. SCS cooling succeeds. Instrument line failure causes leakage in all four modules. Release is through the reactor building.
SG-3	Moderate steam generator tube leak occurs. Moisture monitor detection fails. Reactor trip occurs on high primary coolant pressure. Main circulator trip occurs. Operator intervention succeeds in isolating the steam generator. SCS core cooling succeeds. Primary relief train valve opens and fails to reclose. Release is through the dampers.

TABLE 1-4
(Continued)

Release Category	Description
SG-4	Small steam generator tube leak occurs. Moisture monitors detect ingress. Automatic steam generator isolation succeeds. Dump system valves do not successfully reclose. SCS core cooling is maintained. Release path is through the dump system tank relief valves.
SG-9	Moderate steam generator tube leak occurs. Moisture monitor detection fails. Reactor trip occurs on high primary coolant pressure. Main circulator trip occurs. Operator intervention succeeds in isolating the steam generator. SCS core cooling succeeds. Primary relief train valve opens and fails to reclose. Release is through the dampers.
CC_s-14	Moderate stem generator tube leak occurs. Moisture monitor detection fails. Reactor trip occurs on high primary coolant pressure. Main circulator trip occurs. Operator intervention succeeds in isolating the steam generator. SCS core cooling fails, RCCS cooling succeeds. Primary relief train valve opens and fails to reclose. Release is through the dampers.
CC_p-9	Moderate primary coolant leak where 3×10^{-2} in.$^2 \leq A <$ in.^{2} occurs. HTS and SCS core fail, RCCS cooling succeeds. Reactor building functions properly. Release is through the dampers.
CC_p-12	Very small primary coolant leak where 3×10^{-5} in.$^2 \leq A < 2 \times 10^{-3}$ in.^{2} occurs. HTS and SCS core cooling fail, RCCS cooling succeeds. HPS pumpdown occurs. Release is by leakage through the building.
CC_e-2	An earthquake occurs with an intensity greater than 0.4 g. Reactor trip of all four modules occurs. HTS and SCS cooling are unavailable. RCCS cooling succeeds. Instrument line failure causes leakage in all four modules. Release is through the reactor building.

TABLE 1-4
(Continued)

Release Category	Description
CC_e-3	An earthquake occurs with an intensity greater than 0.4 g. Reactor trip of all four modules fails. HTS and SCS cooling are unavailable. RCCS cooling fails. Instrument line failure causes leakage in all four modules. Release is through the reactor building.

is the principal contributor to both the thyroid and latent fatality risks, as well as being a contributor to the largest expected thyroid dose. A second thyroid dose contributor with a dose approximating that of CC_e-2 is release category CC_s-14 initiated by a moderate steam generator leak. Release category CC_e-3, also initiated by seismic activity, is the dominant contributor to whole body gamma doses for conduction cooldowns as well as all accident families identified in the risk assessment.

The family of accidents initiated by earthquakes, excepting those that result in conduction cooldowns, includes only one release category designated EQ-1. This release category is not dominant relative to the overall MHTGR safety risk envelope.

Release categories with mean frequencies per plant year below 5×10^{-7} have not been listed in Table 1-2. One category belonging to the conduction cooldown family of accidents should, however, be mentioned for completeness. Release category CC_p-2 contributes the largest expected thyroid dose of 0.8 rem for all accident families identified in the assessment. This category does not, however, contribute significantly to either thyroid or latent fatality risks.

This assessment examined a broad event spectrum in order to identify events potentially dominant with respect to plant safety. From this examination, seven initiating events were selected for detailed evaluation. Evaluation results indicate that the MHTGR satisfies the interim NRC safety goals (from Ref. 1-1) and the utility/user requirements (from Ref. 1-2). Although this assessment has not addressed the MHTGR design described in the PSID, incorporating the design changes into the PRA is unlikely to impact goal compliance or alter the selected LBEs.

1.1 References

1-1. "Top-Level Regulatory Criteria for the Standard HTGR," HTGR-85-002/0, (PC-000169/0), January 1985.

1-2. "Utility/User Requirements for the Modular High Temperature Gas-Cooled Reactor Plant," GCRA 86-002/Rev. 1 (PC-000217/1), March 1986.

2. INTRODUCTION AND OBJECTIVES

2.1 Programmatic Objectives

The objective of this risk assessment is to support the design of the plant by providing the results needed to assure goal compliance. As Fig. 2-1 illustrates, PRAs for plant safety furnish the basis needed to demonstrate compliance with the NRC interim safety goals as well as user goals (Ref. 2-1). Moreover, the PRA provides feedback to the designers in the form of the reliability allocations and barrier retention requirements needed to perform the functional analysis. In addition, PRA results facilitate the selection and evaluation of LBEs, plus the safety classification of plant systems, structures, and components (Ref. 2-2). With respect to LBE evaluation, PRA considerations are essential for distinguishing among anticipated operational occurrences, design basis events (DBEs), and emergency planning basis events.

2.2 Risk Assessment Objectives

The objectives of the safety risk assessment are:

1. Identify events potentially significant relative to plant safety.

2. Estimate the occurrence frequency of these events.

3. Assess the event consequences.

4. Quantify statistical uncertainties in the frequency and consequence estimates.

5. Evaluate the risk to the public.

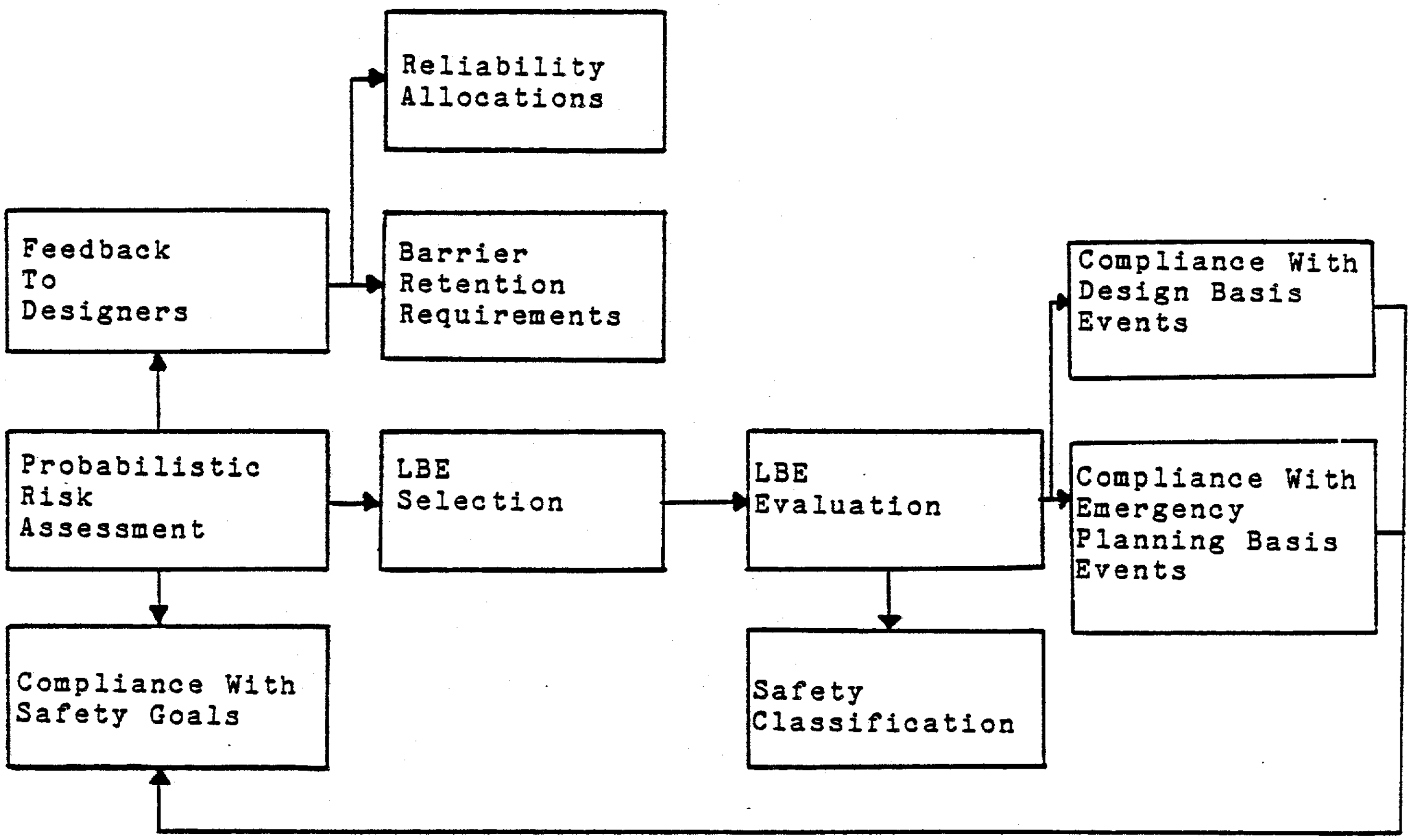


Figure 2-1. PRA Input to the Design Evaluation

2.3 Report Content

This report documents the analysis and results of a risk assessment for the MHTGR using PRA techniques. Section 3 describes the methodology needed for probabilistic safety risk assessment. Section 4 describes how PRA results are utilized to promote goal compliance. Section 5 gives a brief plant description, with emphasis on those systems that are important to safety risk. Events with a potentially significant safety impact are identified in Section 6. Section 7 presents the frequency assessment, while Section 8 discusses the dose consequences in terms of physical phenomena leading to fission product release. The results in terms of safety risk plots of dominant events and their contribution to the overall risk envelope are discussed in Section 9 with respect to the safety goals.

2.4 References

2-1. "Utility/User Requirements for the Modular High Temperature Gas-Cooled Reactor Plant," GCRA 86-002/Rev. 1, (PC-000217), March 1986.

2-2. Houghton, W. J., and L. L. Parme, "Application of Bridging Methods for Standard HTGR Licensing Bases," HTGR-86-017/Rev. 1, (908699/1), February 1986.

3. PROBABILISTIC RISK METHODOLOGY

The assessment method for probabilistic risk is shown in Fig. 3-1. The method is begun by selecting initiating events and is then continued by constructing event trees for accident sequences, analyzing the sequences of events to obtain the probabilities and to evaluate the consequences of the escape of radioactivity, and finally utilizing the results to support the plant design. References 3-1 through 3-3 outline the risk methodology used in the assessment in more detail.

Initiating events are selected that have the potential to lead to the release of radioactivity from the plant. Once the initiating events are defined, a systematic presentation of the progression of the accident sequences from initiation to termination is provided in an event tree for each initiating event. To anticipate and understand these sequences, systems analysis is needed to show the transient response, such as core temperatures, and to show the response of active systems such as the ability of cooling systems to remove the decay heat under the conditions specified in the accident sequence. Intersystem dependencies may also be important.

The probability of occurrence of each event along each of the accident sequences within the event tree is obtained from fault tree analysis. A fault tree is a logic diagram which gives the probability of an undesired state of a system (e.g., loss of cooling) when the various component failure modes, probabilities, and dependencies are known. The component failure probabilities come from data banks containing standardized reliability values and/or raw experience data. In the evaluation of fault trees, it is important to consider common mode failures which can lead to simultaneous failure of redundant components or systems. Uncertainty analysis allows the generation of mean values for probabilities of accident sequences.

The analysis of consequences and physical phenomena for the accident sequences is simplified by grouping the sequences into a smaller number of release categories such that the system responses of sequences within a given

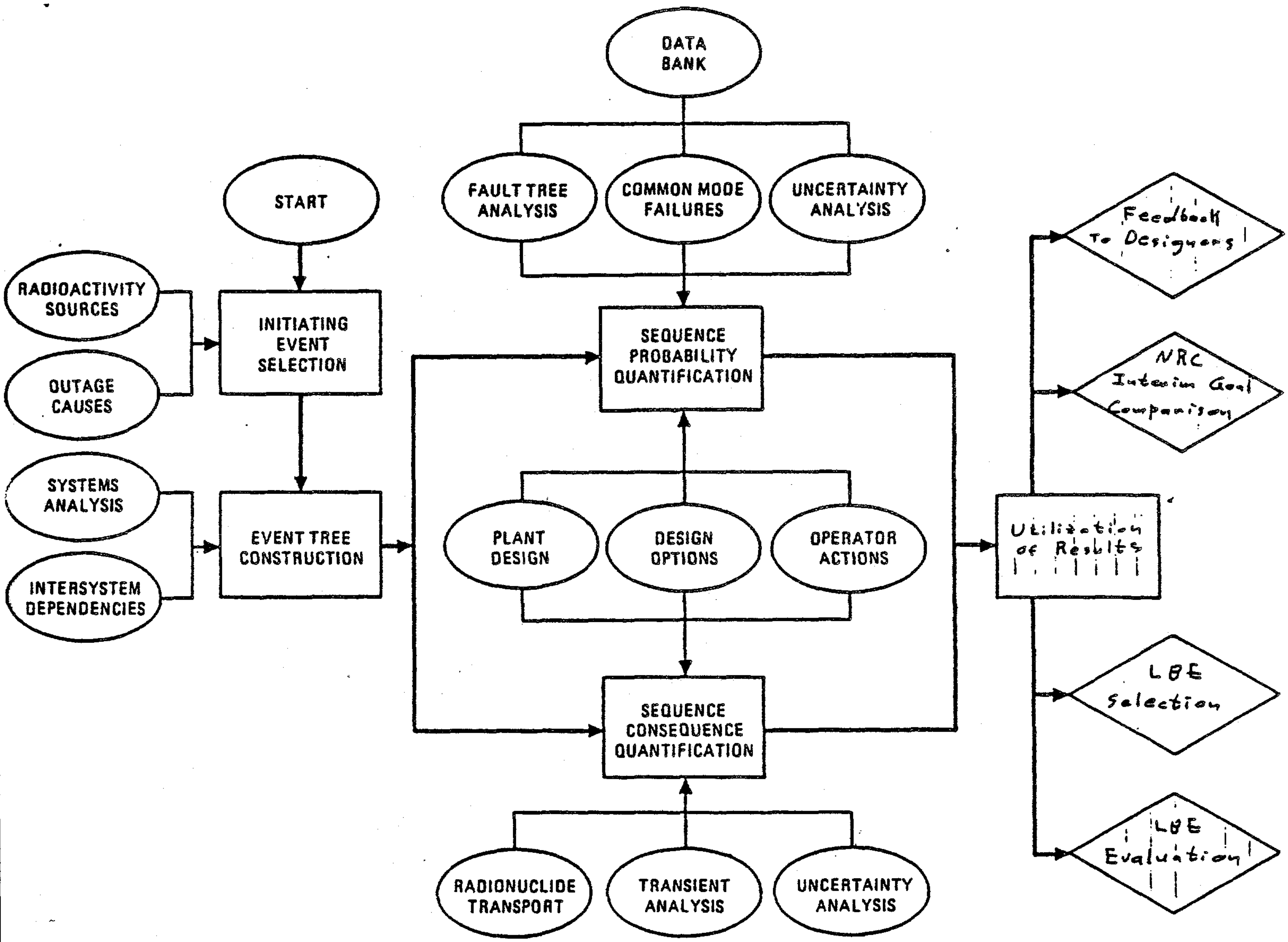


Figure 3-1. Probabilistic Risk Assessment Methodology and Uses

category are very similar and therefore result in about the same consequences. Given a release category, the transient analysis is done first to determine the condition of key components - such as the temperatures in the reactor core. Based on these results, the time-dependent radionuclide transport is calculated with the end result being radiation dose or health effects to the public and the uncertainty distributions for these accident consequences.

3.1 References

3-1. Bender, D. M., et al., "Safety Risk Assessment of the HTGR Steam Cycle Cogeneration Plant," GA-A17000, May 1983.

3-2. Fleming, K. N., et al., "HTGR Accident Initiation and Progression Analysis Status Report Phase II Assessment," GA-A15000, April 1978.

3-3. Project Staff, "PRA Procedures Guide," NUREG/CR-2300, January 1983.

4. UTILIZATION OF RESULTS

Safety protection for the HTGR comes from Goals 0, 3, and 4. These goals are described in detail in Ref. 4-1. Goal 0 supplies the interim NRC safety goals from NUREG-0880. These are individual risk goals for acute and latent fatalities. The risk assessment results include estimates of the mean acute and latent fatality risks, which are given in Section 9.

Goal 3 introduces regulatory requirements from 10CFR50 Appendix I and 10CFR100. Since 10CFR50 Appendix I requirements are imposed on Anticipated Operating Occurrences (Ref. 4-2), the mean risk (expressed as the average whole body and thyroid doses per plant year) for events with mean frequencies above 2.5×10^{-2} per plant year are also cited in Section 9.

Risk assessment results are not compared directly to 10CFR100 requirements. Instead, the evaluation in Sections 6 through 8, as well as the Section 9 results, provide input to the Bridging Methodology developed in Ref. 4-2. Part of the bridging process involves selecting DBEs which are compared to 10CFR100 requirements in the plant PSID.

Goal 4 requires that the dose PAGs in EPA-520/1-75-001 be satisfied at the emergency planning zone. User requirements (Ref. 4-3) further necessitate that the emergency planning zone not extend offsite. Hence, the PAGs for sheltering/evacuation must be satisfied at the EAB over a wide spectrum of events, therefore being required for Goal 3 as well as Goal 4 compliance. Events with mean frequencies $\geq 5 \times 10^{-7}$ per plant year are also compared to the sheltering/evacuation PAGs in Section 9. The comparison is performed on a complementary cumulative basis to assure that the PAGs are satisfied.

4.1 References

4-1. "Top-Level Regulatory Criteria for the Standard HTGR," HTGR-85-002/0 (PC-000169/0), January 1985.

4-2. Houghton, W. J., and L. L. Parme, "Application of Bridging Methods for Standard HTGR Licensing Bases," HTGR-86-017/1 (908699/1), February 1986.

4-3. "Utility/User Requirements for the Modular High Temperature Gas-Cooled Reactor Plant," GCRA 86-002/1 (PC-000217/1), March 1986.

5. PLANT DESCRIPTION

This safety risk assessment is based on the MHTGR Plant concept described in Ref. 5-1. Reference 5-1 pertains to the MHTGR design as of October 1985. This section highlights the major aspects of this design with emphasis on those features of particular relevance to the risk assessment.

The MHTGR Plant consists of four reactor modules and two turbine generator sets to achieve the nominal 558 MW(e) plant rating. Each reactor module is housed in a vertical cylindrical concrete enclosure which is fully embedded in the earth. The concrete enclosure is also designed to perform the function of a reactor building system.

The major components of the nuclear steam supply system (NSSS) are contained within the MHTGR Plant as shown in Fig. 5-1. The NSSS design parameters are listed in Table 5-1 (Ref. 5-2). Each module consists of separate vertical reactor and steam generator vessels connected by a horizontal coaxial cross duct. The annular reactor core, inner and outer graphite reflectors, associated supports, and restraining devices are installed in the reactor vessel. The active core consists of an annular array of fueled prismatic graphite blocks. Radially inside and outside of the active core are rings of replaceable hexagonal reflector columns. Reactivity control is accomplished by means of control rods and an independent reserve shutdown system (RSS). Six control rods are situated in the inner reflector and 24 in the outer reflector. Twelve RSS channels are located in the innermost row of fuel elements. Only the outer control rods are used during normal operation and hot shutdown in order to protect the inner rods from thermal damage. For cold shutdown conditions, the outer rods are used in conjunction with delayed insertion of either the inner rods or the RSS neutron-absorber pellets discharged from hoppers over the inner reflector. During moisture ingress events, the outer control rods and RSS are both inserted in order to provide long-term shutdown capability.

Cold helium at 258°C enters at the top of the core and leaves the bottom of the core at 687°C at nominal rated power. Refueling is done with the

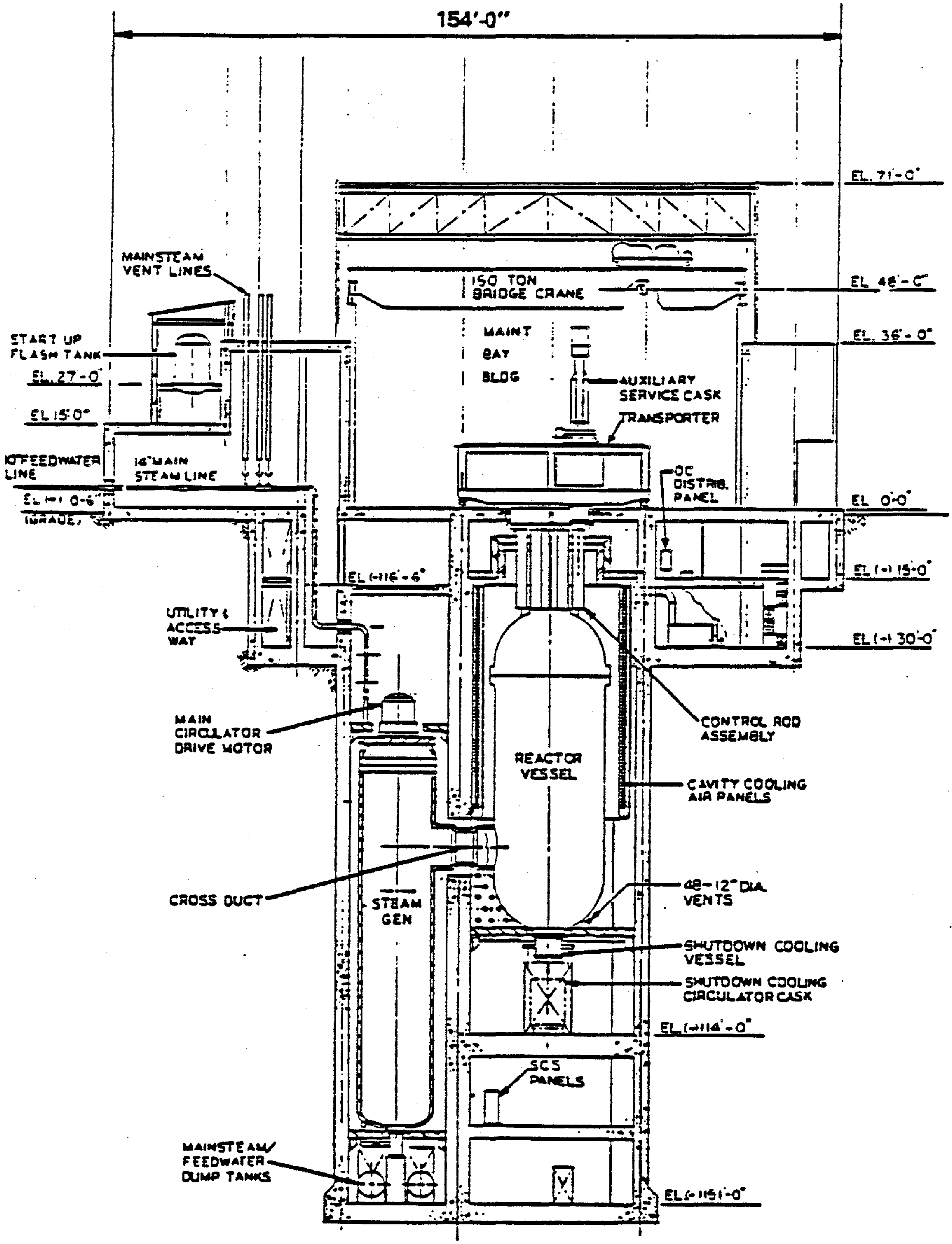


Figure 5-1. NSSS Arrangement Drawing

TABLE 5-1
NSSS DESIGN PARAMETERS

Item	Parameter
Reactor System	
Modules per station	4
Power per module, MW(t)/MW(e)	350/140 nominal
Coolant (helium) pressure at rated power	Helium at 6.38 MPa (925 psia) at circulator discharge
Cold helium temperature (at circulator discharge)	258°C (497°F)
Hot helium temperature (at core exit)	687°C (1268°F)
Feedwater temperature/pressure	193°C/20.68 MPa (380°F/3000 psia)
Steam temperature/pressure	541°C/17.3 MPa (1005°F/2515 psia)
Configuration description	Side-by-Side (SBS)
Vessel material	Carbon steel - Mn-Mo, SA 533 Gr B, Class 1
Reactor vessel overall height, w/CRDS and shutdown circulator	29.4 m (96.5 ft)
Reactor vessel outside diameter	6.8 m (22.4 ft)
Number of Components Per Module	
Steam generators	1
Circulators	1 main, submerged electric motor-driven 1 shutdown cooling, electric motor-driven
Shutdown heat exchangers	1
Control rods	30 (6 inner, 24 outer reflector rods)
Reserve shutdown channels	12 (inner row of core fuel elements)
Start-up system (flash tank)	1
Core and Fuel Cycle	
Fuel element configuration	Prismatic hex-block, 20.78 cm sides × 79.3 cm height
Fissile material	UCO
Power density	5.91 W/cm^3
Power peak/average axial ratio	1.4:1
Average enrichment	19.9% U-235
Fertile material	ThO_2

reactor shut down and depressurized. The vessel-head parts for the control system components also serve for core refueling. Approximately one-half of the fuel elements are replaced every one and one-half years with new fuel. This is based on a three-year fuel residence time.

Some of the major plant systems are the HTS, shutdown cooling system (SCS), reactor cavity cooling system (RCCS), reactor service system, and the plant protection and instrumentation system (PPIS).

For decay heat removal under pressurized or depressurized conditions, forced circulation using the HTS is the first option. If either the main circulator, steam generator, or balance of plant (BOP) systems is not operational, forced circulation using the SCS provides the next option for either pressurized or depressurized cooling. If both the HTS and the SCS are unavailable, decay heat removal is performed through conduction and radiation to the reactor vessel walls and RCCS panels.

The HTS consists of one main coolant loop, including a steam generator in series with a helium circulator and a helium shutoff valve assembly. The primary function of the HTS is to transfer nuclear heat generated in the core, during normal plant operating and shutdown conditions, to the steam generator.

The SCS provides a redundant means of reactor decay heat removal when the reactor is shut down. The SCS consists of a cooling loop with a heat exchanger in series with a helium circulator and associated shutoff valve assembly on the helium side. The shutdown cooling water system (SCWS) is shown in Fig. 5-2. The SCWS is a closed cooling loop that serves all four reactor modules. This system operates at a pressure lower than the normal primary system operating pressure. The current design consists of two 100% capacity shell-and-tube heat exchangers that reject heat to the service water system, two 100% capacity pumps, a 25% capacity pump, and the associated piping, valves, etc. In rating the pumps and heat exchangers, 100% denotes the capability to remove decay heat from all four modules simultaneously.

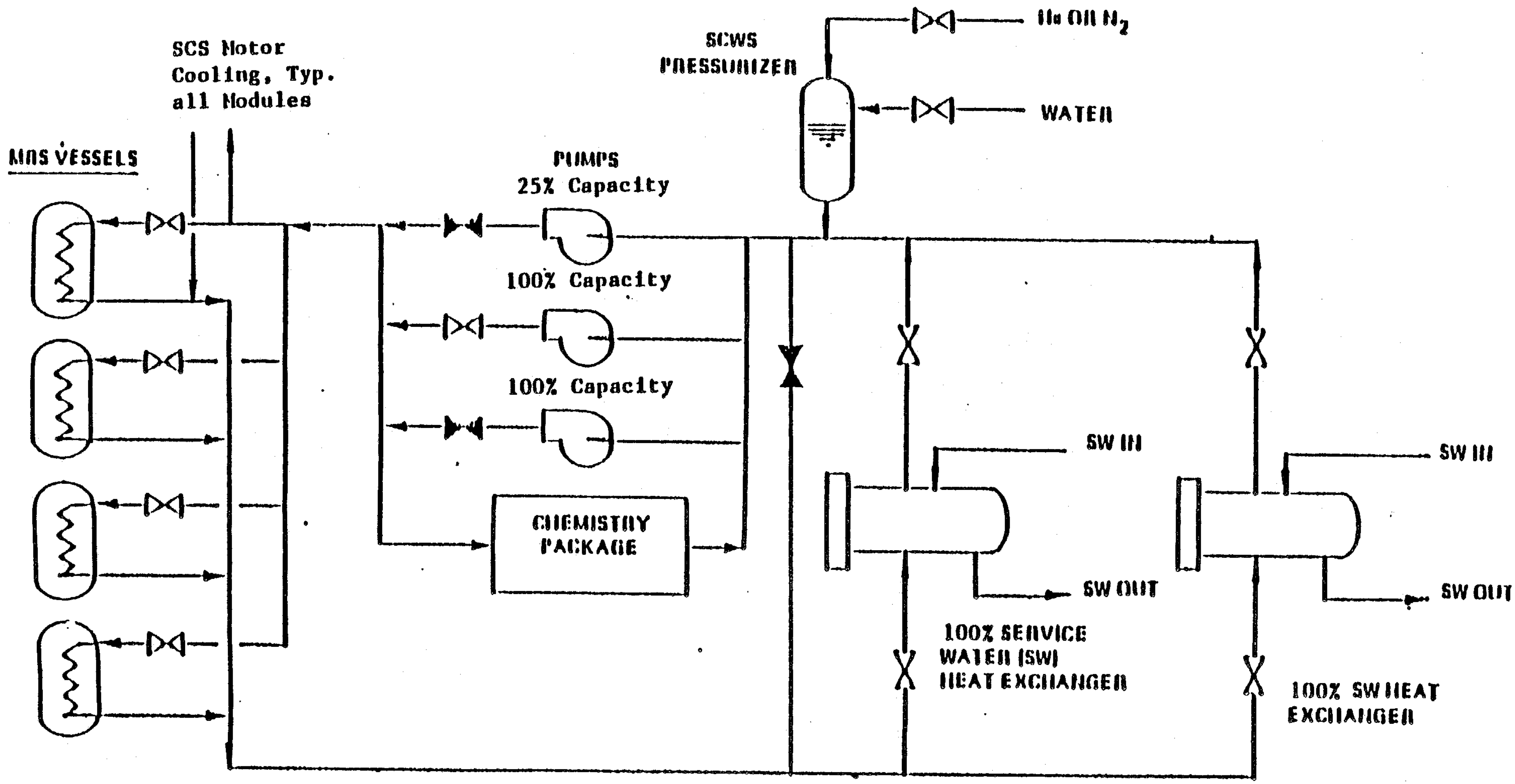


Fig. 5-2. Shutdown Cooling Water System

The RCCS is shown in Fig. 5-3. During normal power producing operations the system is required to remove heat from the reactor cavity in order to limit the temperature of the reactor vessel and internals to acceptable levels and to protect the concrete structures from overheating. When the reactor is shut down, decay heat is normally removed from the vessel via the steam generators to the main condenser, or via the SCS. However, in the event these paths are not available, decay heat is removed via conduction and radiation to the RCCS cooling panels and via the RCCS to the atmosphere.

Since the RCCS must remove both normal and decay heat loads, the system must function continuously while the reactor is at power or generating significant decay heat. The RCCS (Ref. 3) is a completely passive, air-cooled system which provides a high degree of reliability. The system removes heat from the reactor cavity by the natural convection of outside air through the cooling panels located in the reactor cavity. The cooling panels are divided into four quadrants, each quadrant having an annular inlet air duct and cylindrical outlet duct routed inside the inlet passage. This arrangement protects the structural concrete from the hot outlet air. The outlet duct is insulated to minimize heating of the inlet air. Gratings and screens are provided on the inlet passages to prevent blockage by foreign objects. Each reactor module has its own completely independent RCCS.

The reactor service system includes the helium purification subsystem (HPS). The HPS processes helium from the primary coolant loop to remove particulates, chemical impurities, and radioactivity. The system is also designed to transfer helium from the reactor vessel to the helium storage system in a controlled fashion, removing the radioactive impurities and as much of the chemical impurities as possible. This feature is used to depressurize the reactor vessel, approximately 30 h being required for complete pumpdown to storage.

The energy conversion system is illustrated in Fig. 5-4. It is composed of two identical systems, each receiving steam from two reactors. Steam from two modules is joined in a common header and is directed to the high-pressure

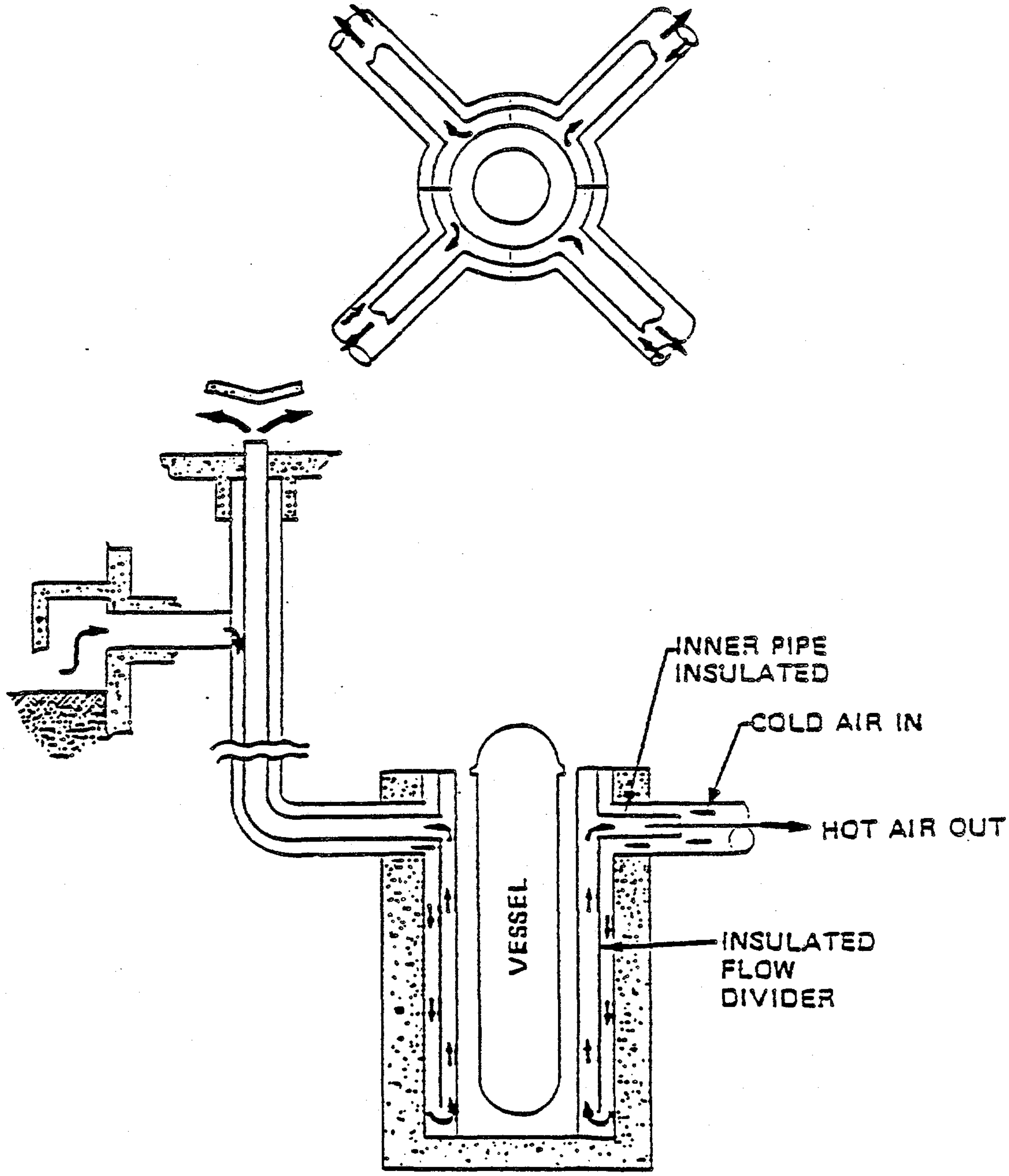


Figure 5-3. Reactor Cavity Cooling System

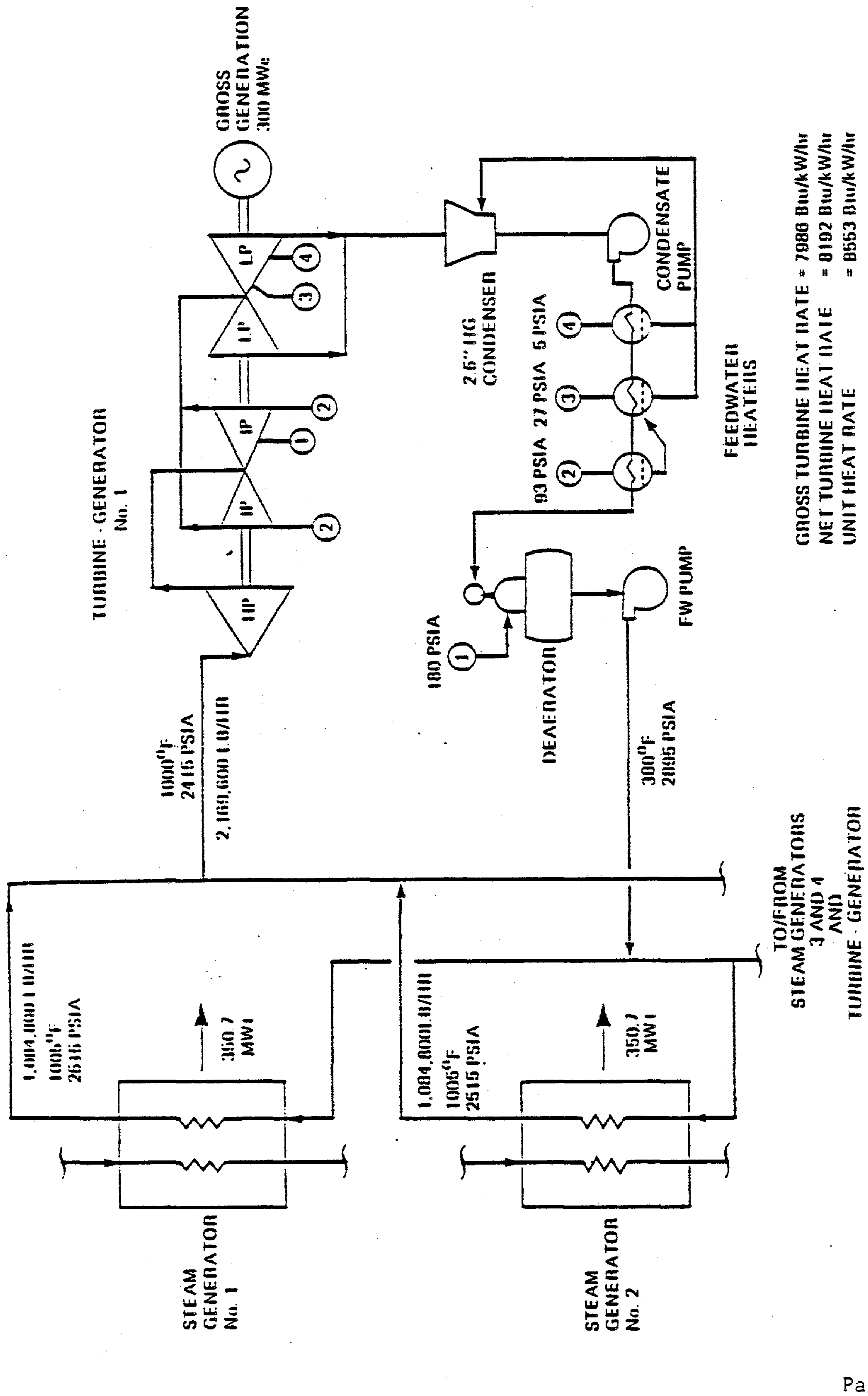


Fig. 5-4. Energy Conversion System

turbine. The high-pressure turbine exhaust is directed to the intermediate and then low-pressure turbines. The turbine cycle has three extraction/feedwater heating stages with a deaerating heater. The cycle has no reheat.

The nuclear island structures that house the reactor, the HPS, and all other nuclear systems, such as fuel handling, are equipped with normal heating and ventilating equipment. In addition, some of these systems are also located within the reactor building boundary, an envelope formed by low leakage structures, doorways, etc. The heating, ventilation, and air conditioning (HVAC) system maintains negative building pressures to ensure that air flows from less contaminated spaces to potentially more contaminated spaces, and that all air leaving the building is monitored and, if necessary, filtered. Small amounts of radioactivity that may be released into the reactor building are processed through high-efficiency particulate air (HEPA) filters and charcoal beds capable of removing 99% of all airborne particulates and 95% of all airborne halogens (iodine). However, the reactor building filtration system cannot withstand the pressure loads or process the gas/vapor flows which would result from excessive feedwater, main steam, or primary coolant boundary leakage. To allow for this possibility, the reactor building is equipped with dampers which will open when pressurized from the inside and reclose after the pressure transient. The reactor building response to an increase in the building pressure is to automatically shut the building fans off and isolate the filtration system to avoid damage to the filters which are engaged only during normal plant operational modes. When atmospheric pressure is reached and the pressure relieving dampers have closed, post-accident radioactive releases can only escape the building by leakage.

The safety protection, investment protection, and special nuclear area instrumentation subsystems constitute the PPIS. The safety protection subsystem governs reactor trip with the outer control rods and RSS. Reactor trip with the inner control rods, HTS shutdown, steam generator isolation and dump, primary coolant pumpdown, and SCS initiation are controlled by the investment protection subsystem. Although many functions performed by the investment protection subsystem are important relative to the plant safety risk envelope, the investment protection subsystem (unlike the safety protection subsystem) is not safety-related. The special nuclear area instrumentation subsystem include the reactor vessel pressure relief block valve interlock, PPIS information displays, post-accident monitoring instrumentation, and core performance instrumentation.

The reactor site has an EAB of 425 m resulting in a minimum of 140 acres for the site.

5.1 References

5-1. "Preliminary Concept Description Report 4 × 350 MW(t) HTGR Plant Side-by-Side Steel Vessel Prismatic Core Concept," HTGR-85-142/1 (908575/1), October 1985.

5-2. "Conceptual Design Data for 4 × 350 MW(t) Modular HTGR Plant," HTGR-86-023/0 (907807/6), January 1986.

5-3. "Reactor Cavity Cooling System 4 × 250 MW(t) HTGR Plant Side-by-Side Steel Vessel Concept," HTGR-85-136/0 (908570/0), September 1985.

6. MASTER LOGIC DIAGRAM

A major consideration in performing a PRA is completeness. Specifically, whether there are events, not included in the event trees, that can appreciably increase the predicted plant safety risk envelope. One technique for addressing this aspect is to characterize plant safety with an MLD. An MLD is a summary fault tree prepared to identify and group accident scenarios. The MLD helps guide the selection of dominant accidents and ensures that a wide spectrum of potentially important events is considered.

An MLD is an inductive logic tool. Its top-level event, for safety characterization, is defined as an uncontrolled radiological release (Fig. 6-1). The use of levels in the MLD is an ordering technique to help identify event combinations which may result in an uncontrolled release. By postulating a number of accident sequences for each logic path from the bottom to the top of the MLD, a broad spectrum of events is considered.

Selection of events that dominate the safety risk envelope is accomplished by considering the frequency and consequence of each MLD event. If both the frequency and consequence of one event are judged to be appreciably higher than the frequency and consequence of a second event, the first event will contribute negligibly to the comparison with goals and to the safety risk envelope and is not recommended for further analysis. Moreover, events that are judged to have an upper margin frequency below 5×10^{-7} per plant year are not recommended for further analysis because they do not impact the Goal 3 safety requirements. By applying these screening criteria systematically to the MLD, potentially dominant events are identified for inclusion in the detailed PRA.

Estimating MLD event frequencies and consequences can only be performed in a scoping manner (dominant events are evaluated in detail afterward). Since these scoping estimates are predicated upon prior experience, some events are identified which cannot be evaluated confidently (e.g., due to some unique plant design characteristic or a lack of adequate design definition).

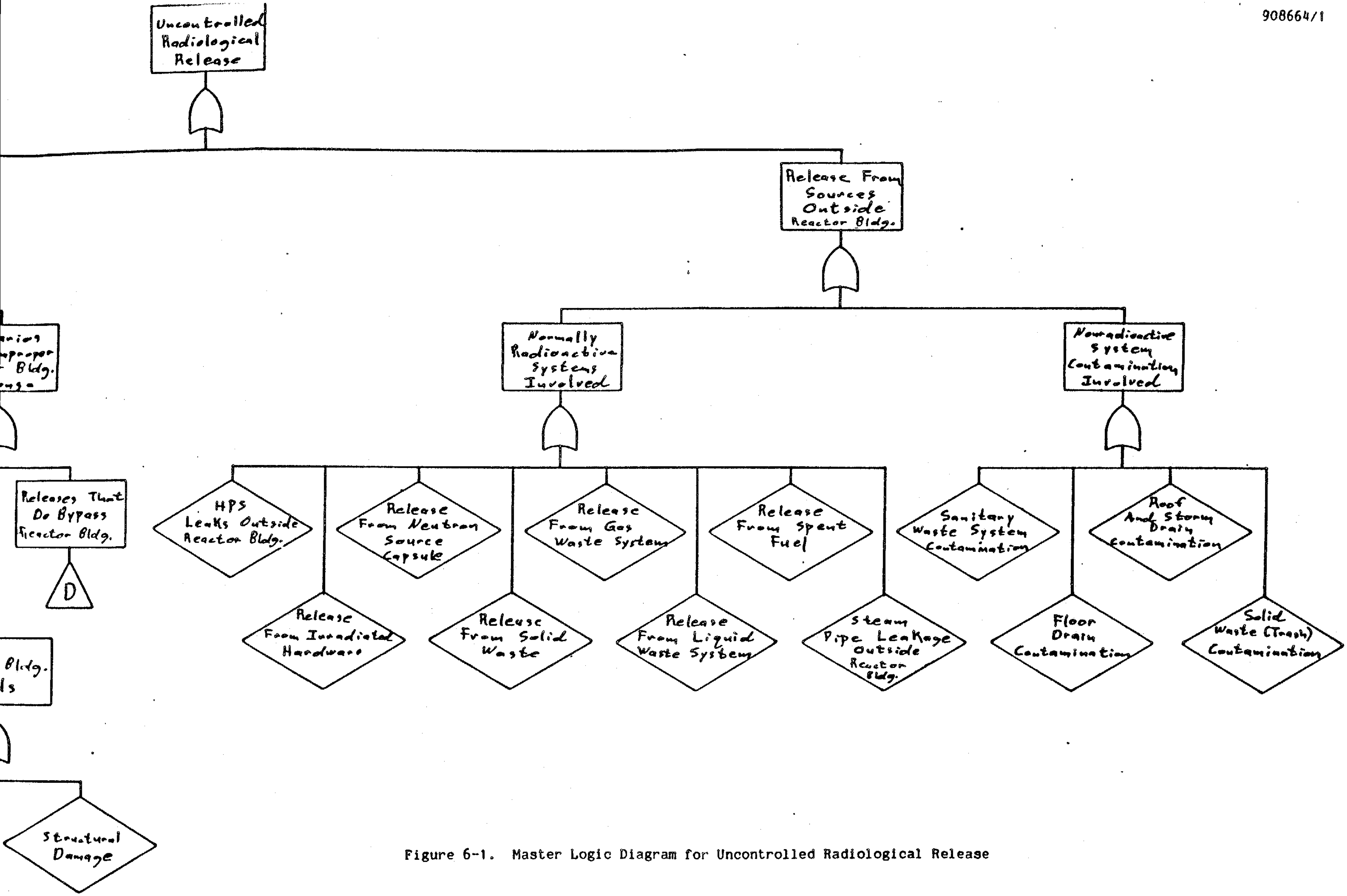
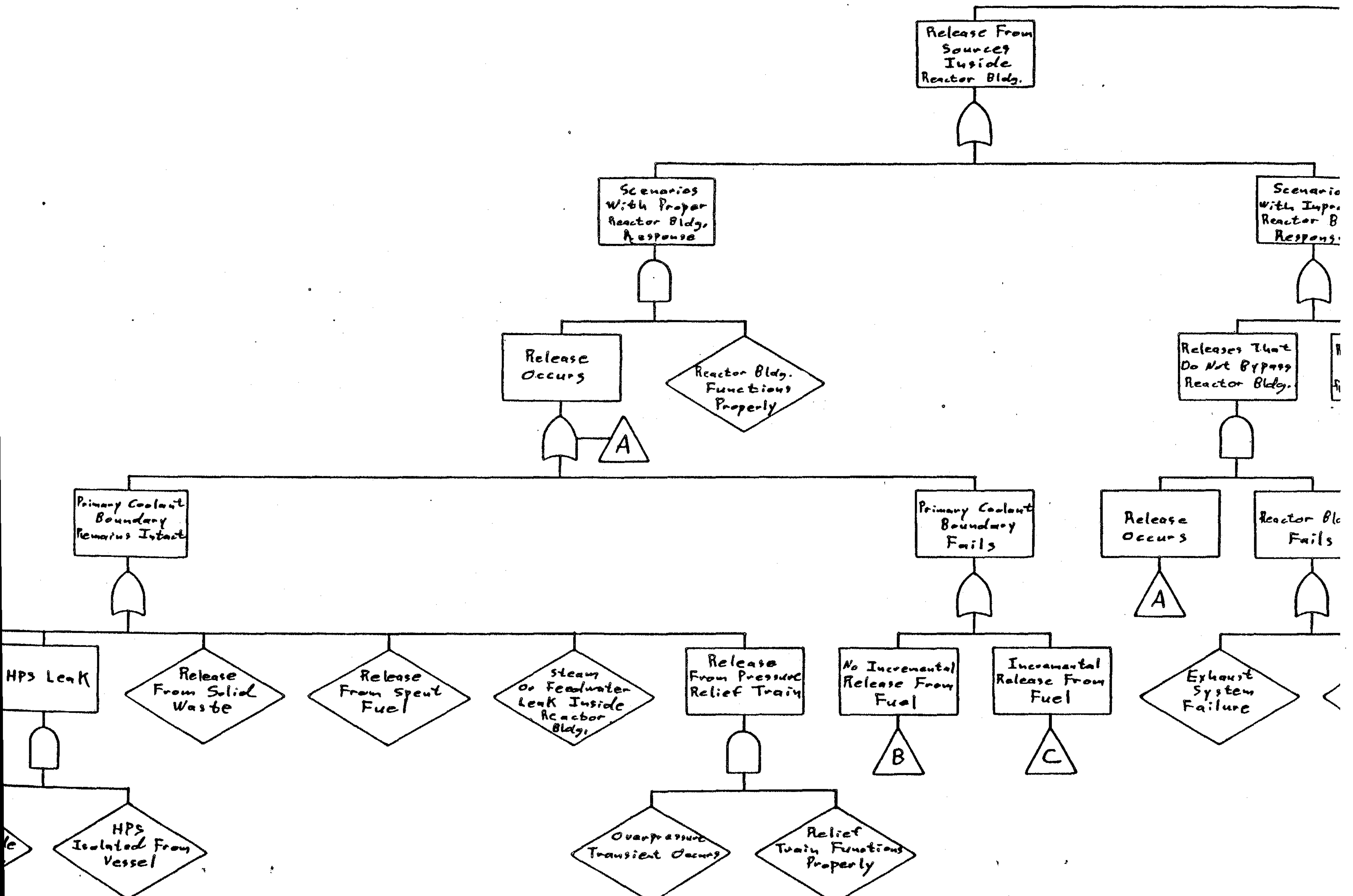
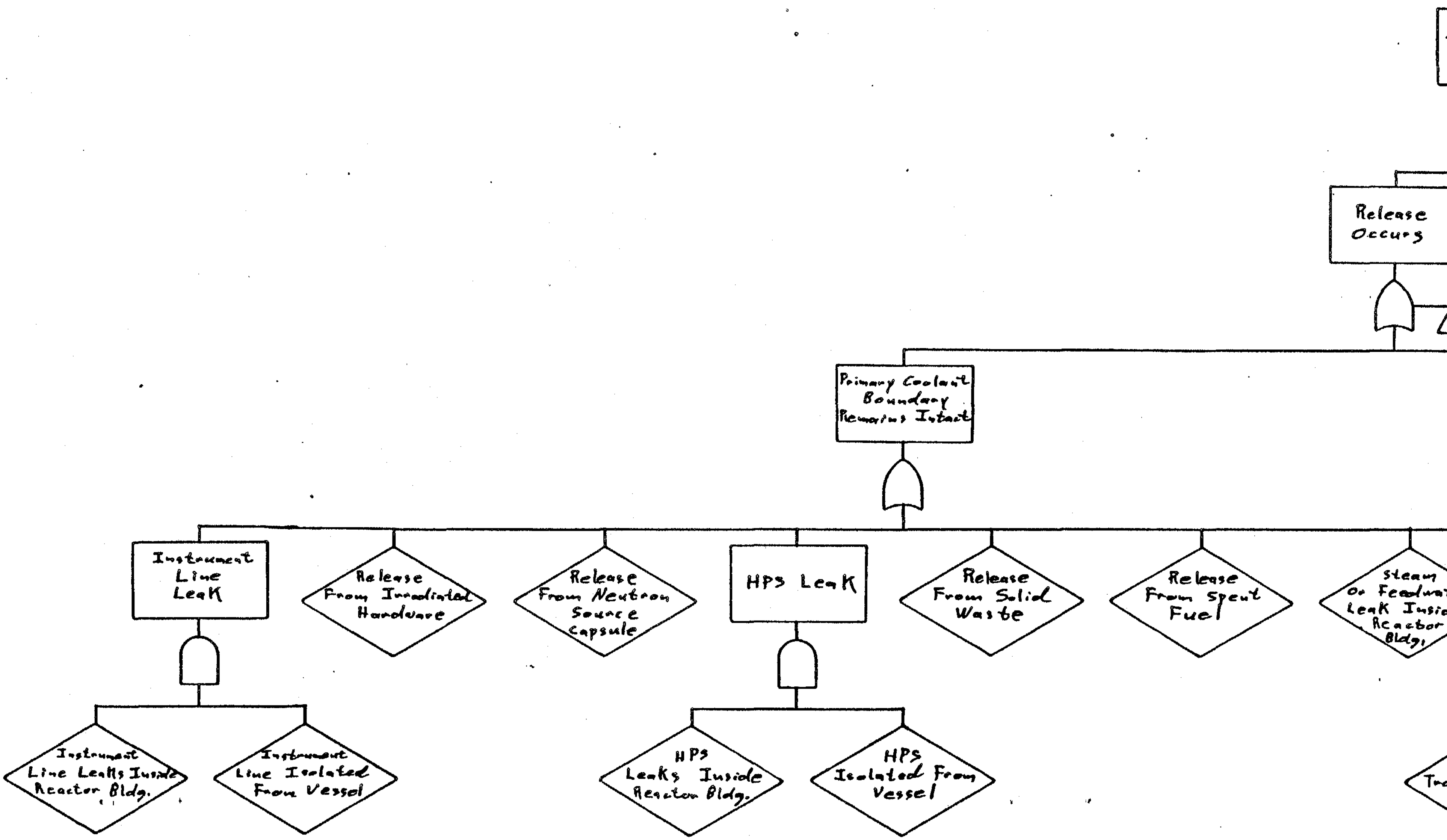


Figure 6-1. Master Logic Diagram for Uncontrolled Radiological Release



Release From Sources Inside Reactor Bldg.
Scenarios With Proper Reactor Bldg. Response
Release Occurs
A
Reactor Bldg. Functions Properly
Primary Coolant Boundary Remains Intact
Primary Coolant Boundary Fails
HPS Leak
Release From Solid Waste
Release From Spent Fuel
Steam Or Feedwater Leak Inside Reactor Bldg.
Release From Pressure Relief Train
No Incremental Release From Fuel
B
Incremental Release From Fuel
C
HPS Isolated From Vessel
Overpressure Transient Occurs
Relief Train Functions Properly
Releases That Do Not Bypass Reactor Bldg.
Release Occurs
A
Exhaust System Failure



Release Occurs
Primary Coolant Boundary Remains Intact
Instrument Line Leak
Release From Irradiated Hardware
Release From Neutron Source Capsule
HPS Leak
Release From Solid Waste
Release From Spent Fuel
Instrument Line Leaks Inside Reactor Bldg.
Instrument Line Isolated From Vessel
HPS Leaks Inside Reactor Bldg.
HPS Isolated From Vessel

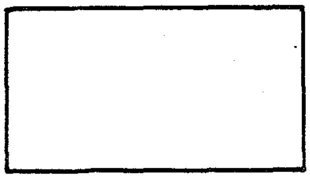
Events of this type are also recommended for further analysis in order to fully understand plant safety. In these cases, the future analysis may simply disclose that an event has a negligible safety impact. The 4 × 350 MW(t) MHTGR is presently in the conceptual design phase. Therefore, in constructing and evaluating the MLD, it is sometimes necessary to use engineering judgment regarding plant design details and the response of systems to certain transients. As the design evolves, the validity of these suppositions must be ascertained in order to assure PRA completeness.

Figure 6-1 is the MLD for uncontrolled radiological releases from the MHTGR. The symbols employed in constructing the MLD are defined in Table 6-1 (from Ref. 6-1).

Events in Fig. 6-1 are organized according to the radioactive source location. At the second level in the MLD, uncontrolled radiological releases are dichotomized into those arising from sources inside the reactor building and from sources located outside of the reactor building. These events are addressed in Sections 6.1 and 6.2, respectively. Figure 6-1 shows a limited substructure relating to releases from sources located outside the reactor building due to the current sparsity of design and operational detail.

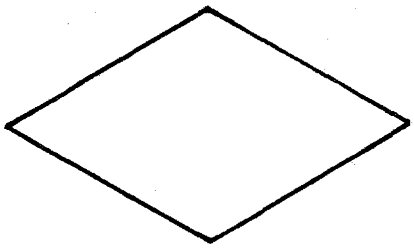
Releases from sources inside the reactor building are further divided into scenarios with and without a proper reactor building response to the initiating event (see Sections 6.1.1 and 6.1.2). For each reactor building response scenario, the MLD additionally addresses the behavior of the two major radiological barriers: the fuel and the primary coolant boundary. The MLD segment for releases from sources inside the reactor building exhibits the general trend that in moving from the lower left to the right in Fig. 6-1, the number of major barriers that fail increases. Thus, a spectrum of events ranging from uncontrolled releases with a proper reactor building response and an intact primary coolant boundary to events involving incremental fission product release from the fuel, loss of primary coolant integrity, and the reactor building being bypassed, is covered.

TABLE 6-1
MASTER LOGIC DIAGRAM SYMBOLS



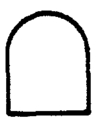
RECTANGLE

A general event or a gate output event resulting from the logical combination of contributory events acting through a logic gate.



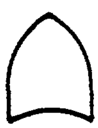
DIAMOND

An undeveloped terminal event not developed to its cause.



AND Gate

A logic gate that produces an output only when all input events occur.



OR Gate

A logic gate that produces an output when one or more of the input events occur.

Interbranch/Interpage TRANSFER

Transfers substructure from another branch or another page; has an identifying capital letter.

6.1 Releases from Sources Inside the Reactor Building

Any radiological release from a source located inside of the reactor building produces an offsite dose, since the reactor building does not contain pressure. Nevertheless, the reactor building can provide some dose attenuation. Hence, its response to an event influences the resultant risk.

6.1.1 Scenarios with Proper Reactor Building Response

The reactor building can mitigate offsite doses through three attenuation mechanisms:

1. Radioactive decay of nuclides during their residence time inside the reactor building.

2. Plateout on reactor building surfaces.

3. Gravitational settling of dustborne radionuclides.

Since the total release rate form the source to the reactor building and the reactor building egress rate are comparable, the effectiveness of these mechanisms is strongly dependent upon how rapidly the initial release occurs.

6.1.1.1 Releases with an Intact Primary Coolant Boundary (Proper Reactor Building Response). Figure 6-1 shows that releases with an intact primary coolant boundary that originate inside of the reactor building can be grouped into three classes.

1. Releases involving some primary coolant (i.e., releases from instrumentation or HPS line leaks as well as normal relief train operation).

2. Releases associated with maintenance or refueling (i.e., releases from irradiated hardware, neutron sources, solid waste, or spent fuel).

3. Releases initiated by steam or feedwater line leaks.

Although the doses produced by instrumentation and HPS line leaks are small (since the isolation systems limit the release), scoping calculations (Ref. 6-2) indicate that they have a relatively high occurrence frequency. Thus, they are potentially important contributors to the high-frequency, low-consequence portion of the safety risk envelope and require further analysis. Releases through the primary coolant relief train that are initiated by a pressurized conduction cooldown produce relatively high doses at frequencies on the order of 10^{-4} per plant year. These events are also potentially dominant risk contributors that require further analysis.

Uncontrolled releases that occur during maintenance and refueling have a relatively low frequency. Moreover, the offsite doses are quite small since the fission products are not in a volatile form. Therefore, these events contribute negligibly to the MHTGR safety risk envelope.

Scenarios involving steam or feedwater line leaks in conjunction with preserved primary coolant boundary integrity and a proper reactor building response are unimportant risk contributors. Data from Refs. 6-3 and 6-4 indicate that steam or feedwater leaks inside of the reactor building are less likely than primary coolant leaks. Also, the resultant doses, which are primarily due to the released tritium, are negligible. However, steam line breaks can be quite energetic, and scenarios involving steam line breaks that damage the primary coolant boundary or reactor building are addressed later in the MLD.

6.1.1.2 <u>Releases with Primary Coolant Boundary Failure (Proper Reactor Building Response)</u>. A primary coolant boundary failure introduces the possibility that the fuel body fission product inventory could be released. Thus, these events are dichotomized into scenarios with and without incremental releases from the fuel.

6.1.1.2.1 Scenarios Without an Incremental Release from the Fuel and Primary Coolant Boundary Failure (Proper Reactor Building Response). Figure 6-2 is the MLD subtree for events without incremental release from the fuel. In order to have a primary coolant boundary failure without an incremental release from the fuel, two concurrent conditions are necessary:

1. Primary coolant leakage must occur.

2. The fuel must be protected from excessive reactivity, high temperatures, and chemical attack (i.e., moisture or air ingress).

Subtree H in Fig. 6-2 shows that primary coolant leakage can be initiated by cooling tube leaks or by leaks in components comprising the primary coolant/reactor building atmosphere interface.

Events initiated by cooling tube leaks are presented in Fig. 6-3. Recalling that these events include a proper reactor building response and preclude incremental fuel release (Figs. 6-1 and 6-2), the three conditions needed for these events to occur are:

1. Secondary (i.e., water side) pressure must exceed the primary coolant pressure.

2. A cooling tube must leak.

3. There must be a failure to isolate the leak.

Physically, these events involve a cooling tube leak that overpressurizes the primary circuit and releases primary coolant through the relief train. Since none of the water can be transported to the reactor core (to preclude chemical attack), both the frequency and consequences of these events are low,

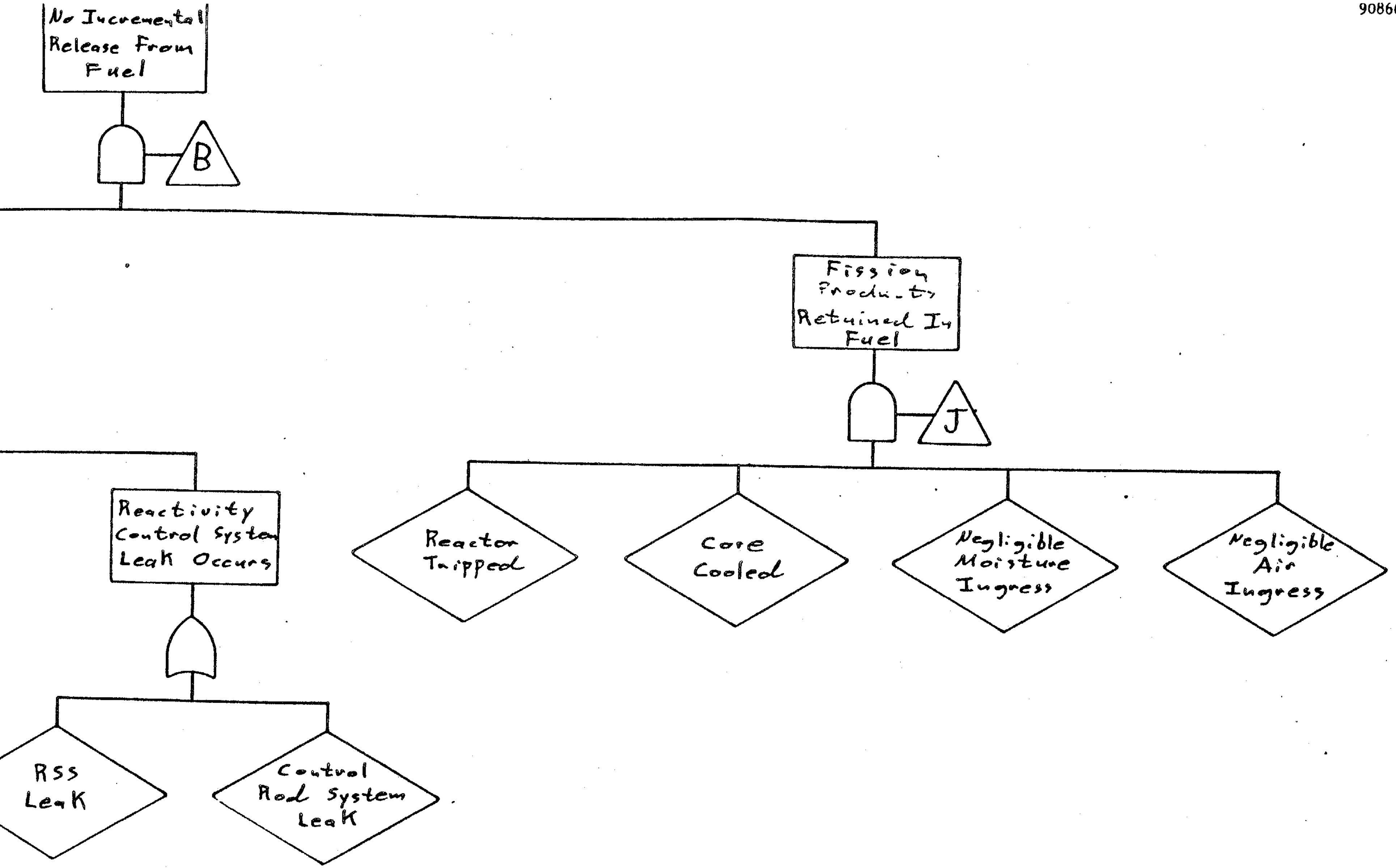
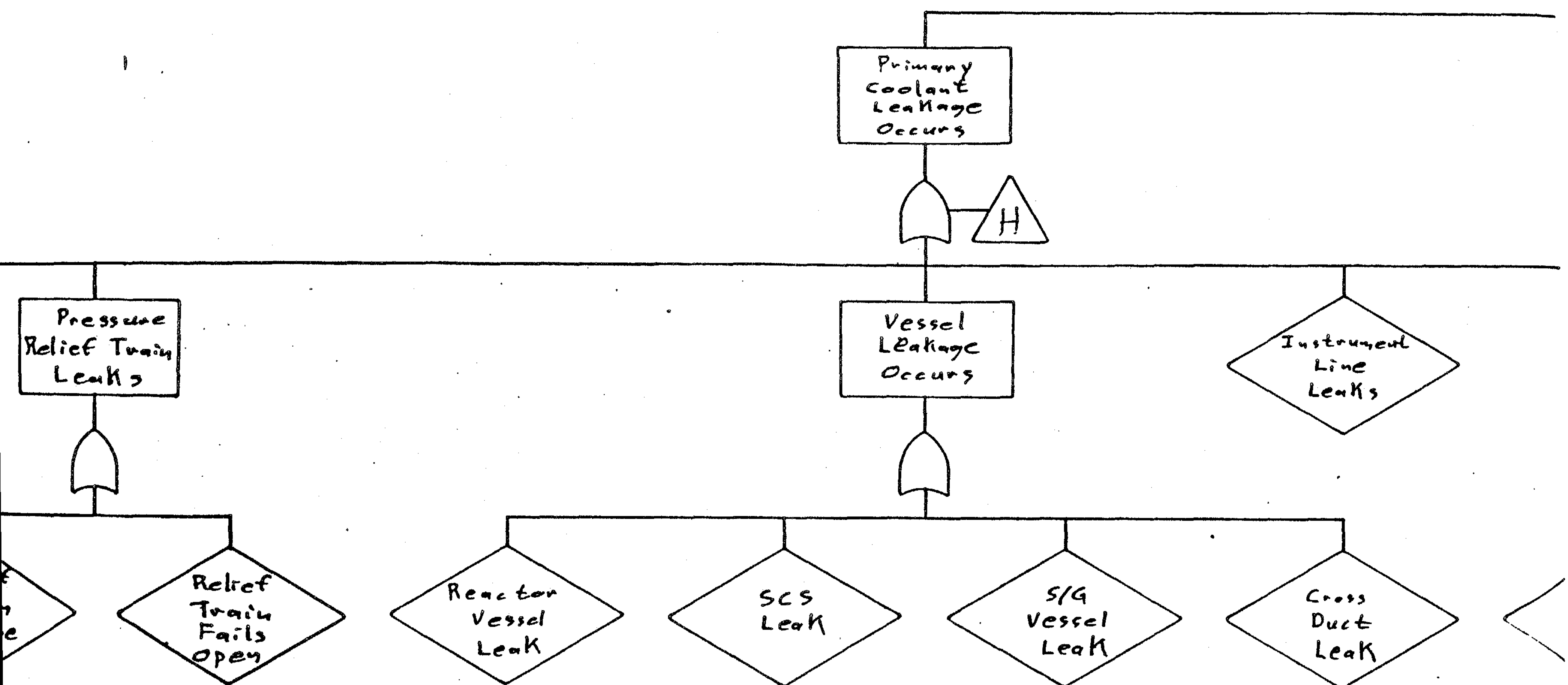
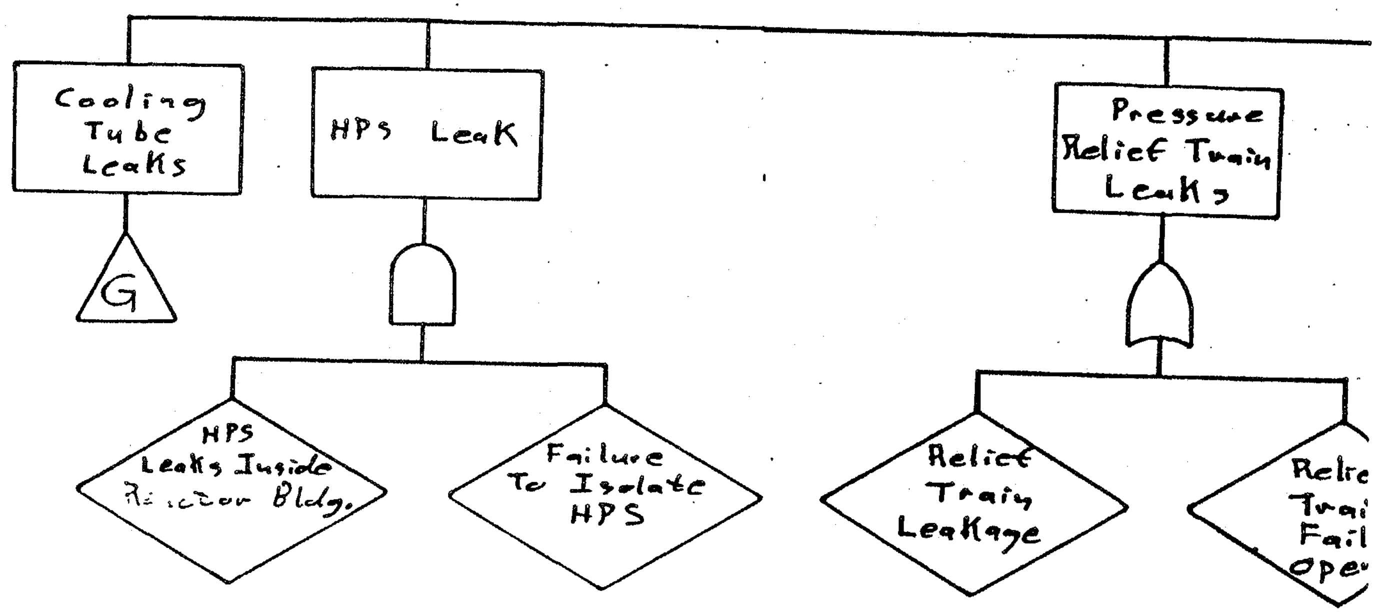


Figure 6-2. Subtree B for Events Without Incremental Release from the Fuel


Primary Coolant Leakage Occurs
H
Pressure Relief Train Leaks
Vessel Leakage Occurs
Instrument Line Leaks
Relief Train Fails Open
Reactor Vessel Leak
SCS Leak
S/G Vessel Leak
Cross Duct Leak



Cooling Tube Leaks
G
HPS Leak
Pressure Relief Train Leaks
HPS Leaks Inside Reactor Bldg.
Failure To Isolate HPS
Relief Train Leakage

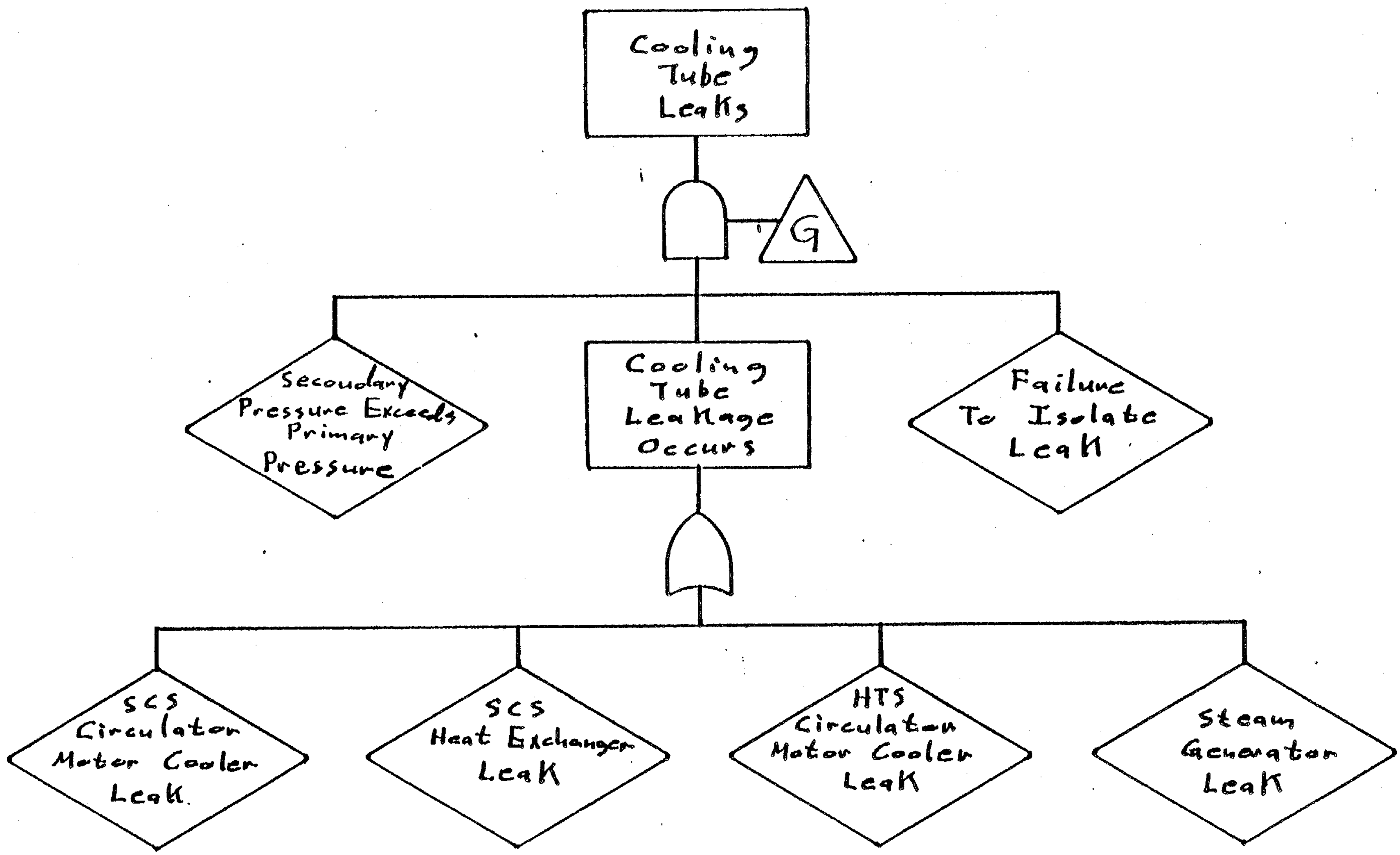


Fig. 6-3. Subtree G for Events Initiated by Cooling Tube Leaks with no Incremental Fuel Release

relative to moisture ingress events that will be addressed in Section 6.1.2.2. Hence, the safety risk envelope impact of Subtree G events is negligible.

Prior assessment (e.g., Ref. 6-5) demonstrates that primary coolant leak events from the remainder of Subtree B (i.e., leaks in components comprising the primary coolant/reactor building atmosphere interface) are important safety risk contributors that require additional analysis.

6.1.1.2.2 Scenarios with an Incremental Release from the Fuel and Primary Coolant Boundary Failure (Proper Reactor Building Response). Figure 6-4 includes the MLD subtree for events with an incremental release from the fuel. As described in Section 6.1.1.2.1, three mechanisms can result in an incremental radionuclide release from the fuel during an accident.

1. Thermal effects (i.e., conduction cooldown events).
2. Reactivity effects.
3. Chemical attack (i.e., oxidation or hydrolysis).

Referring to Table 6-1, the "OR" gate in Fig. 6-4 is inclusive. This means that in addition to incremental fuel releases resulting from individual mechanisms (e.g., thermal effects), combinations of release mechanisms must also be examined (e.g., thermal effects in conjunction with failed fuel particle hydrolysis).

Subtree E in Fig. 6-4 is the MLD segment for incremental fuel releases induced by thermal effects. Subtree E is explicitly restricted to scenarios with reactor trip (conduction cooldowns without reactor trip are addressed later under reactivity effects), but includes moisture ingress events (this incorporation arises from the linkage of Subtree E to Subtrees H and G, Figs. 6-2 and 6-3, respectively). Air ingress is not a concern in this part of the MLD because the proper reactor building response limits the amount of air available to enter the primary circuit. Subtree E also includes releases due to pressurized conduction cooldowns accompanied by failure to reduce the

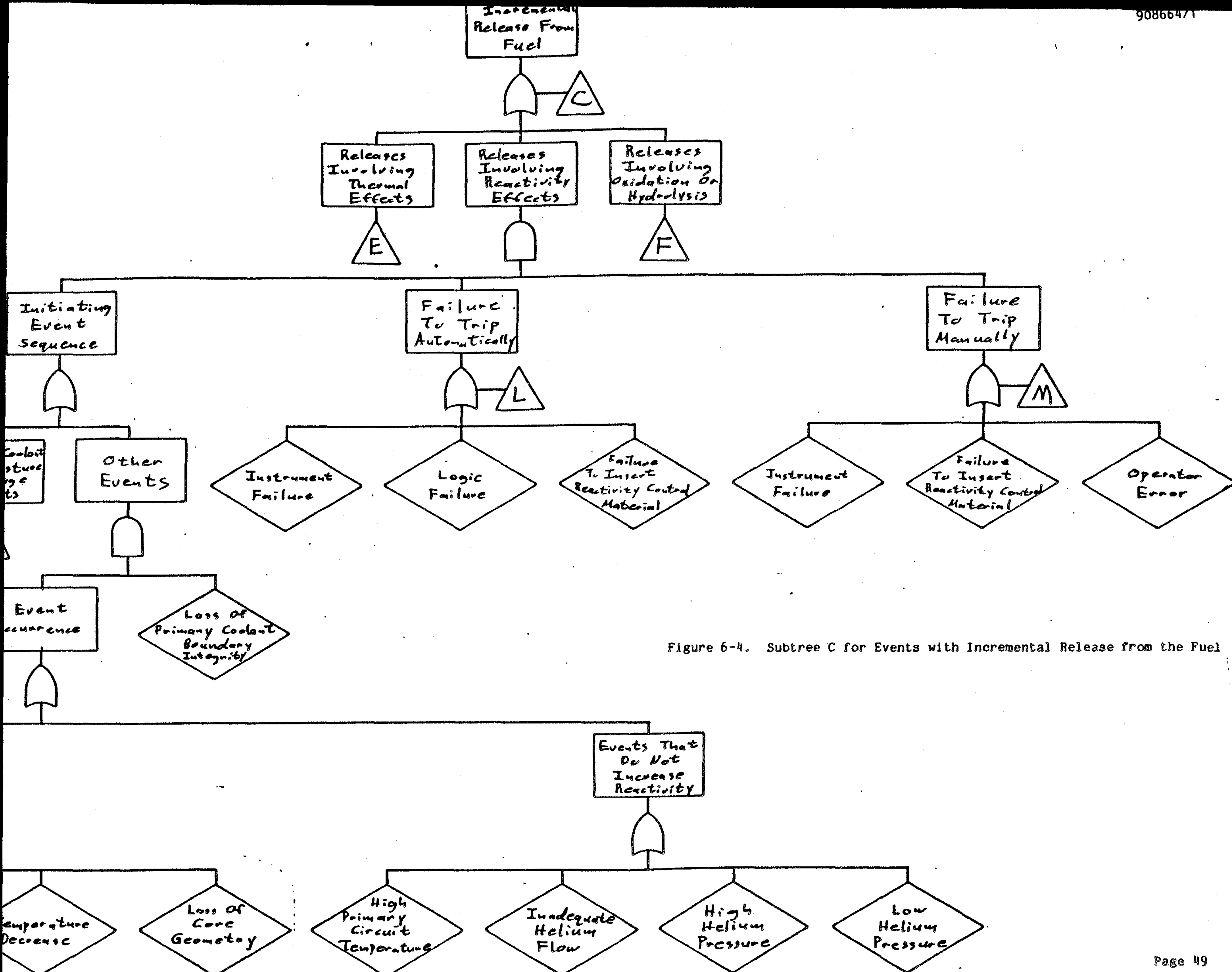
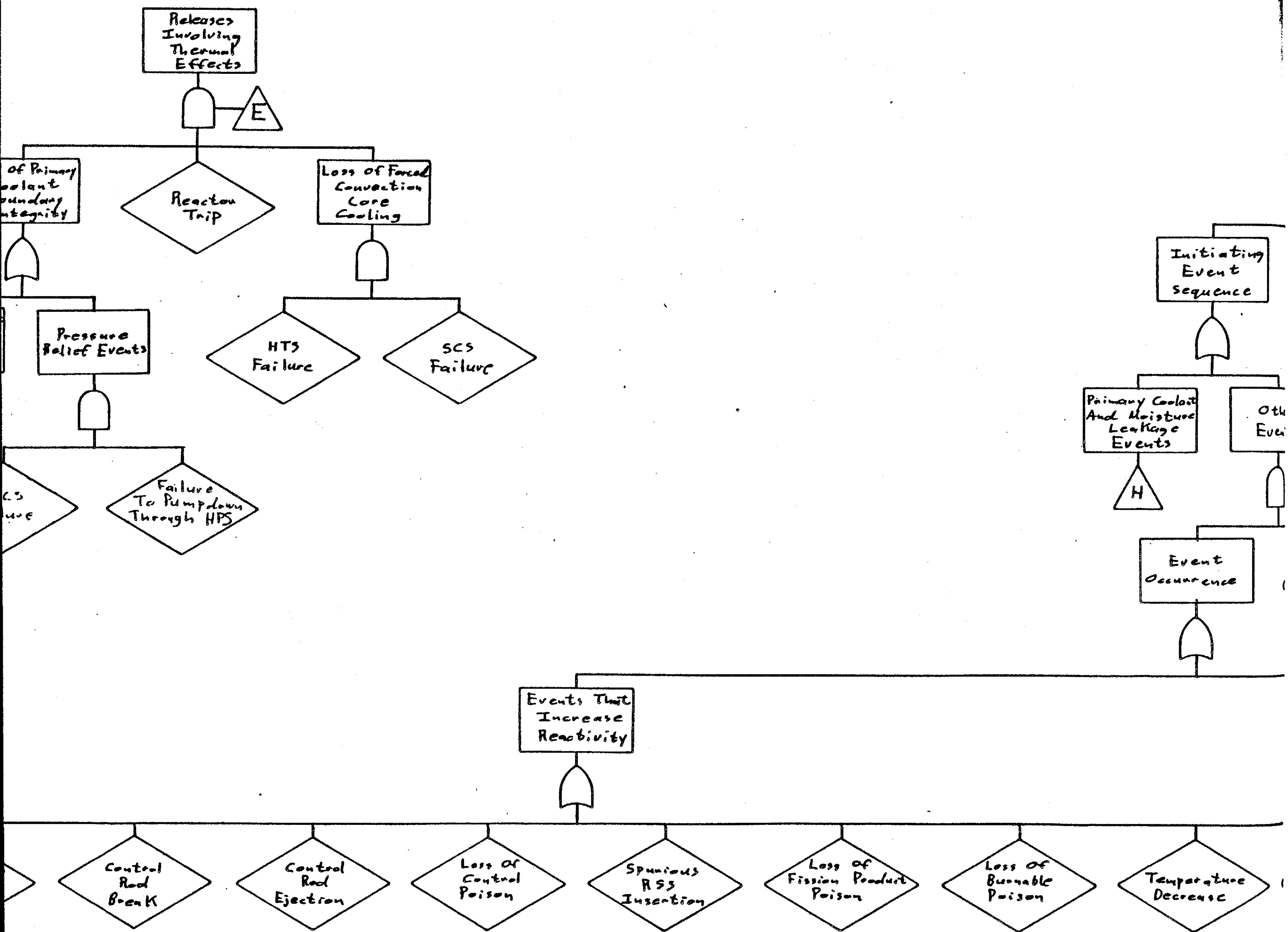
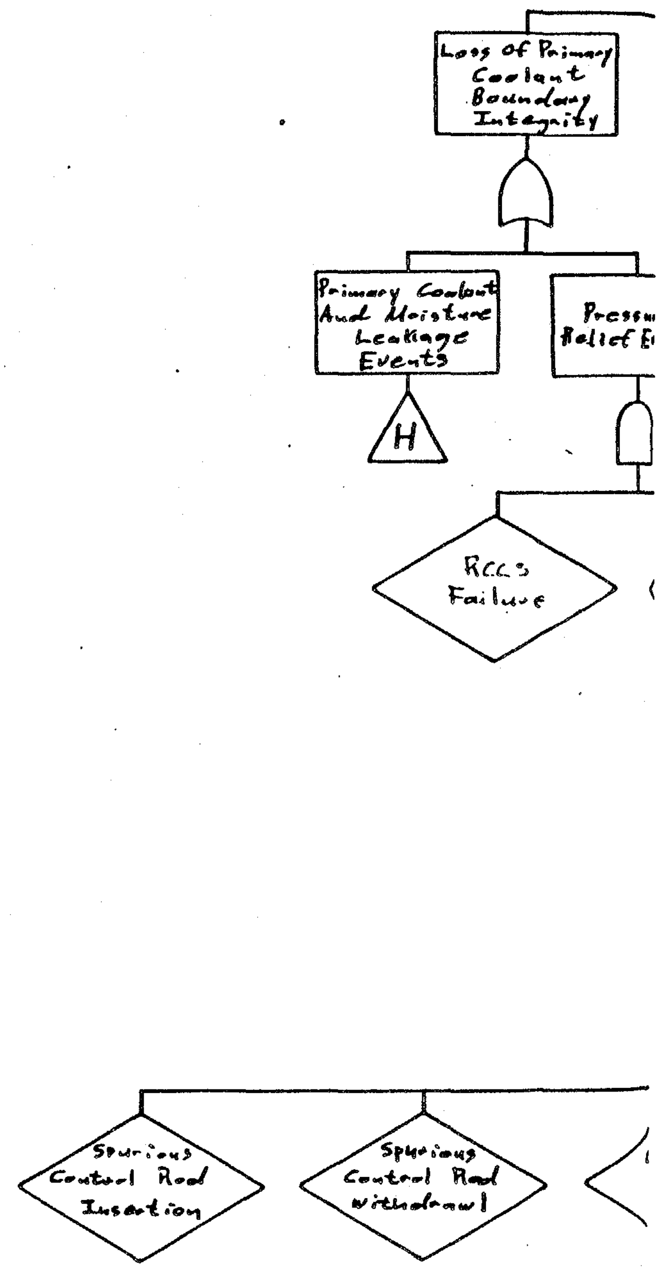


Figure 6-4. Subtree C for Events with Incremental Release from the Fuel


Releases Involving Thermal Effects
E
Reactor Trip
Loss of Forced Convection Core Cooling
Pressure Relief Events
HTS Failure
SCS Failure
Failure To Pumpdown Through HPS
Initiating Event Sequence
Primary Coolant And Moisture Leakage Events
H
Event Occurrence
Events That Increase Reactivity
Control Rod Break
Control Rod Ejection
Loss Of Control Poison
Spurious RSS Insertion
Loss of Fission Product Poison
Loss Of Burnable Poison
Temperature Decrease



Loss of Primary Coolant Boundary Integrity
Primary Coolant And Moisture Leakage Events
H
RCCS Failure
Spurious Control Rod Insertion
Spurious Control Rod Withdrawl

primary coolant pressure in a controlled manner if the RCCS also fails (this causes the primary circuit relief train to open). Analyses in Refs. 6-2 and 6-5 demonstrate that events involving thermally induced incremental fuel releases are important safety risk contributors.

Events involving reactivity-induced incremental fuel releases (Fig. 6-4) require an initiating event (in this case, an event that perturbs the initial state of the plant in a manner that requires negative reactivity insertion) in conjunction with a failure to trip. Failure to trip results if neither appropriate automatic nor manual action is taken. A failure to trip automatically can be caused by:

1. Instrument failure.
2. Control logic failure.
3. Mechanical failure that prevents inserting an adequate quantity of reactivity control material (i.e., the control rods and RSS).

Similarly, an (1) instrumentation failure, (2) mechanical failure that prevents inserting an adequate quantity of reactivity control material, and (3) operator error precludes a manual trip. In assessing the likelihood of a successful manual trip, consideration is given to the nature of the transient. For example, the probability of operator error during a fast reactivity addition event is virtually unity.

Two types of initiating events are identified:

1. Primary coolant leaks
2. Other initiating events

Primary coolant leaks were considered previously in Subtrees H and G (Figs. 6-2 and 6-3, respectively). Leaks in the primary coolant/reactor building atmosphere interface without a reactor trip by control rods or RSS occur at frequencies well below 5×10^{-7} per plant year over a broad spectrum of leak sizes (Ref. 6-5). Nevertheless, there is concern that very large leaks could produce shear forces high enough to causally preclude a reactor trip. Even though the frequency of such events is thought to be extremely low, additional analyses are needed to verify this contention. There is also a concern that under shutdown conditions, water droplet entrainment subsequent to a cooling tube leak could add enough reactivity to a module core that cold shutdown cannot be maintained by the control rods alone. Although the frequency of such an event is above 5×10^{-7} per plant year, further investigation is necessary in order to quantify the consequences.

The unique behavior of the annular core makes it difficult to ascertain which of the other initiating events in Subtree C substantially impact the safety risk envelope. Since further evaluations are clearly needed, the safety characterization of these other initiating events focused on identifying those that exert negligible influence on the safety risk envelope. Further analysis is not recommended for five initiating events.

1. Control rod break.
2. Loss of control poison.
3. Loss of burnable poison.
4. Temperature decrease.
5. Events that do not increase reactivity.

The concern with a control rod break is that the reactivity control system may overcompensate for the initial decrease in reactivity, resulting in a reactivity excursion. However, both the frequency and mean consequence of this event are below those for events initiated by a spurious control rod insertion.

Chemical attack could engender a loss of control and burnable poison. However, the quantity of air available to oxidize these poisons is limited, except in scenarios that include reactor building failure (see Section 6.1.2.1). Hydrolysis can also remove boron carbide from the core, but the frequency of such an event is judged to be small relative to 5×10^{-7} per plant year. (Chemical attack can also lead to a loss of fission product poison. The issue here, however, is radioactive decay--especially of xenon--which affects the long-term shutdown capability of the reactivity control system.)

Overcooling is the concern with temperature decrease events. Although overcooling can result in a light-water reactor (LWR) reactivity excursion, such events are benign in the HTGR. Peach Bottom operating experience demonstrates that the core power level responds almost instantaneously to increased cooling in a manner that maintains a constant core temperature profile. If no reactivity is added to the core subsequent to the increased cooling, xenon buildup will eventually lower the power level and core temperature. Hence, overcooling transients have no safety risk impact in an HTGR.

The principal concern with events that do not increase reactivity (i.e., high primary coolant temperature, inadequate helium flow, and high or low helium pressure) is an anticipated transient without scram (ATWS). Due to the large negative temperature coefficient of MHTGR fuel, the upper bound frequency of offsite doses resulting from ATWS events is also below 5×10^{-7} per plant year (Ref. 6-2).

Table 6-2 summarizes the safety characterization of releases involving reactivity effects (with proper reactor building response).

Chemical attack (i.e., oxidation or hydrolysis) is the third mechanism capable of causing an incremental radionuclide release from MHTGR fuel. Subtree F (Fig. 6-5) is the logic diagram for these events. Core cooling and reactor trip are explicitly included in Subtree F, since chemical attack without core cooling or a reactor trip was included in the safety characterization of thermal and reactivity effects. Since the air ingress events include a functional reactor building (Fig. 6-1), negligible fuel oxidation occurs. Therefore, these events have no safety risk impact. Even with core cooling, moisture ingress into the core hydrolyzes failed fuel particles. Nevertheless, prior assessments (Ref. 6-5) demonstrate that these events contribute insignificantly to the safety risk envelope, which is appreciably

TABLE 6-2
SAFETY CHARACTERIZATION OF RELEASES
INVOLVING REACTIVITY EFFECTS

Initiating Event	Is Further Analysis Recommended
Primary coolant leaks	Yes
Spurious control rod insertion	Yes
Spurious control rod withdrawal	Yes
Control rod break	No
Control rod ejection	Yes
Loss of control poison	No
Spurious RSS insertion	Yes
Loss of fission product poison	Yes
Loss of burnable poison	No
Temperature decrease	No
Loss of Core Geometry	Yes
Events that do not increase reactivity	No

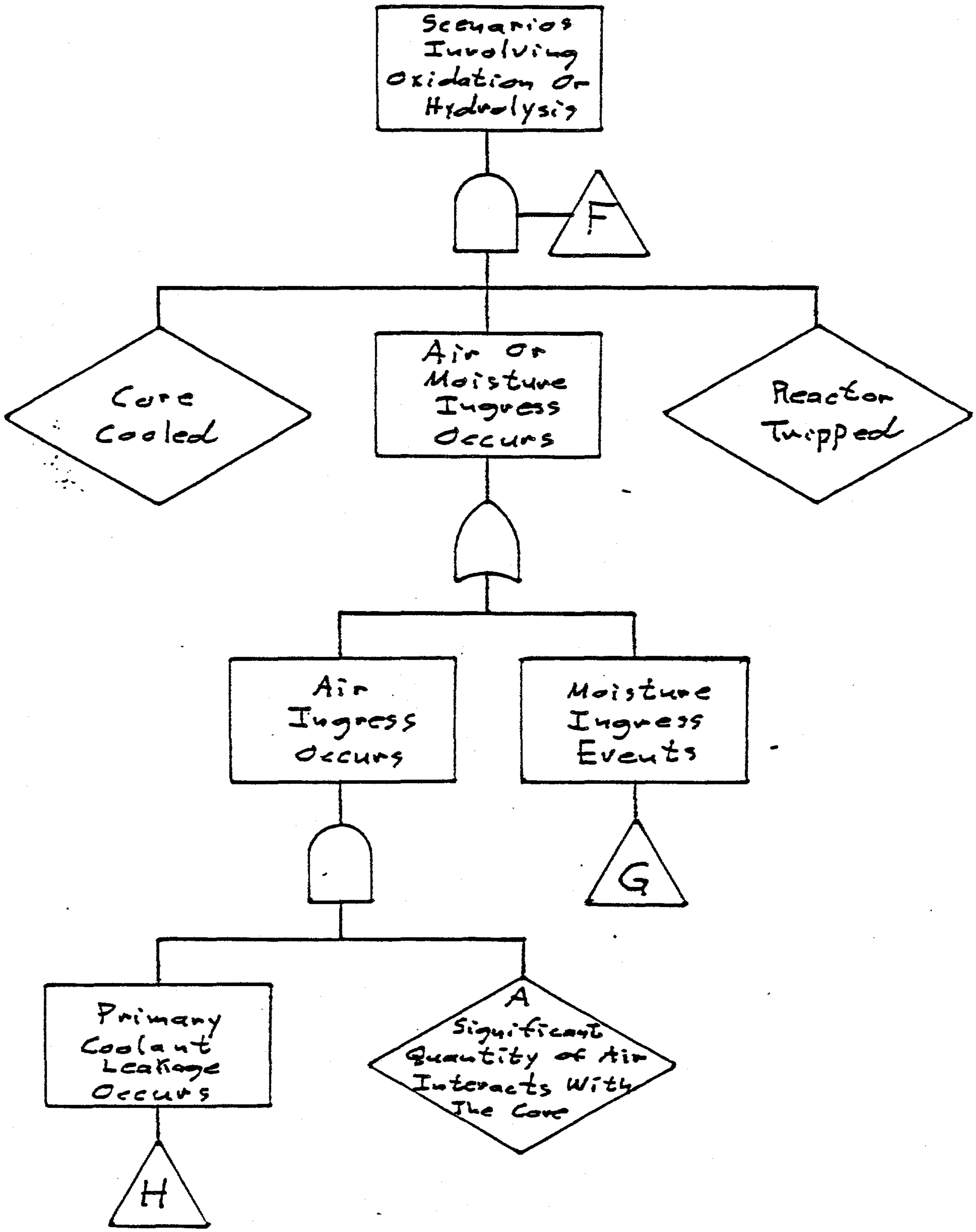


Figure 6-5. Subtree F for Scenarios Involving Chemical Attack

more sensitive to moisture ingress in conjunction with reactor building bypass (Section 6.1.2.2).

6.1.2 Scenarios with Improper Reactor Building Response

The effectiveness of the reactor building in mitigating offsite doses is dependent upon its egress rate. Failure to disengage or isolate the reactor building exhaust system (which normally forces air through the reactor building to control temperature and humidity), structural damage to the building or RCCS panels, and bypassing the reactor building altogether are the three reactor building failure modes (Fig. 6-1). In addition to increasing offsite doses, a high flow rate through the reactor building can result in a significant quantity of air ingress if the primary coolant boundary has a large hole in it.

6.1.2.1 Releases that Do Not Bypass the Reactor Building. An improper reactor building response can occur in conjunction with all of the releases assessed in Section 6.1.1. Thus, Subtree A (see Fig. 6-1) must be reexamined in order to characterize the safety risk implications of reactor building failure.

The first class of events in Subtree A involves releases with an intact primary coolant boundary. Thus, the radiological source term is relatively small. The probability that the reactor building exhaust system operates during an uncontrolled radiological release is also small, and the maximum impact of such a failure on offsite doses is a factor of 20 increase in the thyroid dose (if the release rate into the reactor building is rapid, exhaust system failure has a negligible consequence impact). Therefore, releases resulting from this first class of events in conjunction with an exhaust system failure are unimportant safety risk envelope contributors (Ref. 6-2). Although there is also a concern that the release could structurally damage the reactor building, only steam line breaks are deemed potentially energetic enough to accomplish this. Even so, the frequency of a steam line break that causes significant structural damage is estimated to be below 5×10^{-7} per plant year, since the building is being designed to withstand such an event. Consequently, releases with an intact primary coolant boundary and a reactor building failure have no identifiable safety risk significance.

The second class of events in Subtree A involves releases with primary coolant boundary failure but no incremental release from the fuel (see Subtrees B and G in Figs. 6-2 and 6-3, respectively). Releases initiated by cooling tube leaks (Fig. 6-3) have far less safety risk significance when accompanied by a reactor building failure than they do in events that bypass the reactor building (Section 6.1.2.2). Prior assessments (e.g., Ref. 6-5) indicate that reactor building failures have a negligible impact on the safety risk envelope relative to the other primary coolant leaks assessed in Section 6.1.1.2.1. Nevertheless, further investigation is warranted to confirm this conclusion as revised methods for quantifying primary coolant leak frequencies are developed and applied.

The third class of events in Subtree A involves primary coolant boundary failure with an incremental release from the fuel (Fig. 6-4). If the incremental release is caused by thermal effects (Subtree E in Fig. 6-4), prior analyses indicate that such events coupled with reactor building failure contribute negligibly to the safety risk envelope (Ref. 6-5). If the incremental fuel release is caused by reactivity effects (Subtree C), some events that include reactor building failure require additional examination to fully characterize their risk. Specifically, it is necessary to verify that the frequency of large primary coolant leaks that prevent reactor trip and cause structural damage to the reactor building is low. Events in which the leak is initiated by a control rod ejection, permits sufficient air ingress to oxidize the neutron poisons, or disrupts the core geometry, should also receive further consideration to assure that their risk is negligible. Even if the reactivity effects of air ingress are small, the safety risk impact arising from fuel particle oxidation must also be ascertained.

Chemical attack, by itself, can also cause incremental fuel releases. With both the primary coolant boundary and reactor building failed, air ingress can be a significant safety risk issue unless its frequency is demonstrated to be low. This is a concern even if core cooling is maintained, because the cooling rate in the current design is very slow under depressurized conditions. The dominant moisture ingress events, however, do not involve exhaust system failure nor reactor building structural damage.

6.1.2.2 Releases that Do Bypass the Reactor Building. In order to bypass the reactor building, the radiological source must be connected directly to the atmosphere by a conduit. Typically, such releases are initiated by component leakage and include a subsequent isolation failure. These events are developed in Subtree D (Fig. 6-6). As in previous MLD segments, Subtree D is dichotomized into events with and without an incremental release from the fuel.

6.1.2.2.1 Scenarios Without an Incremental Release from the Fuel (Reactor Building Bypassed). Events with no incremental release from the fuel consist of primary coolant leaks with:

1. Reactor trip
2. Core cooling
3. Negligible moisture ingress
4. Negligible air ingress

(See Subtree J in Fig. 6-2.) Steam generator leak scenarios (Subtree N in Fig. 6-6) have much lower consequences (since the amount of moisture ingress is negligible in order to avoid failed fuel particle hydrolysis) and a lower frequency than events involving fuel releases and reactor building bypass. An HPS leak without isolation may, however, be a dominant safety risk envelope event. In order to discern its relative importance, better design data regarding the HPS configuration in the maintenance bay building is needed.

The safety risk envelope contributions from other heat exchanger leaks (Subtree O in Fig. 6-6) and instrumentation leaks are bounded by conduction cooldowns initiated by small primary coolant leaks (Refs. 6-2 and 6-5).

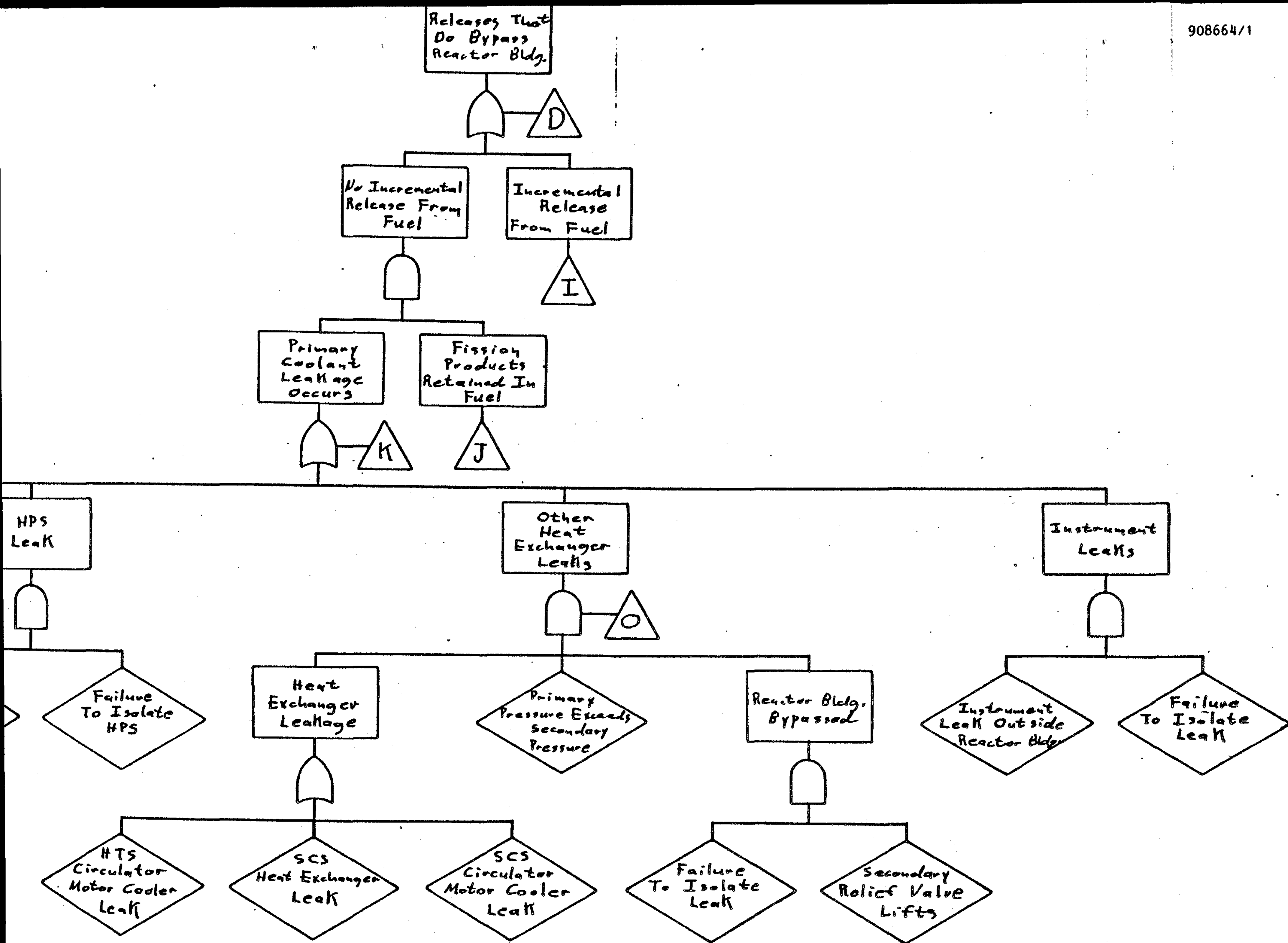
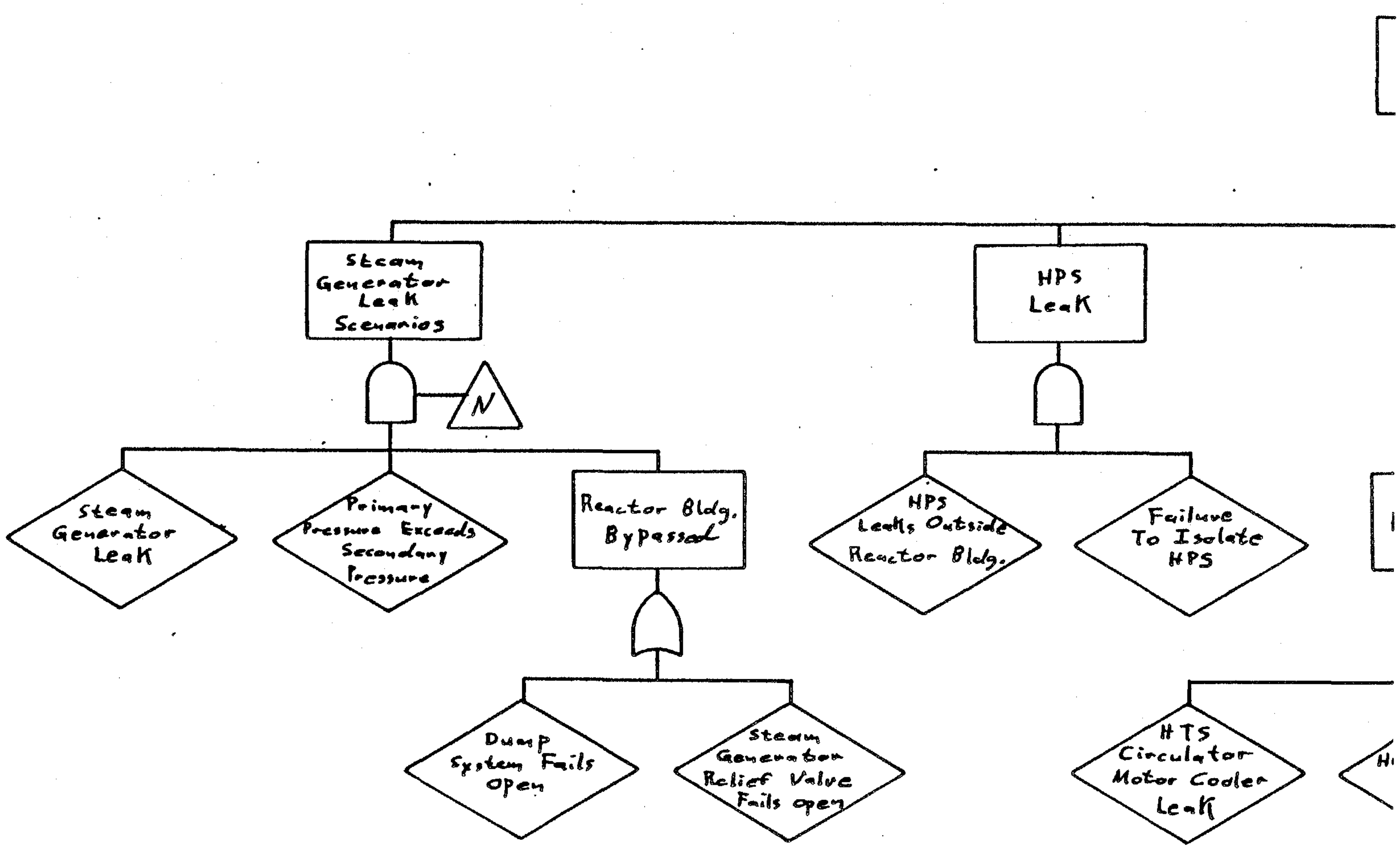


Figure 6-6. Subtree D for Releases that Bypass the Reactor Building



Steam Generator Leak Scenarios
N
HPS Leak
Steam Generator Leak
Primary Pressure Exceeds Secondary Pressure
Reactor Bldg. Bypassed
HPS Leaks Outside Reactor Bldg.
Failure To Isolate HPS
Dump System Fails Open
Steam Generator Relief Valve Fails open
HTS Circulator Motor Cooler Leak

6.1.2.2.2 Scenarios with an Incremental Release from the Fuel (Reactor Building Bypassed). Subtree I (Fig. 6-7) displays the events involving an incremental fuel release with the reactor building bypassed. All three fuel release mechanisms are represented. Releases involving thermal effects that bypass the reactor building are not considered important to safety risk. This is because at frequencies above 5×10^{-7} per plant year, higher doses can be delivered by depressurized conduction cooldowns to the reactor buildings at higher frequencies. The lower frequency of depressurized conduction cooldowns that bypass the reactor building is due to the relatively high probability that the leak can be isolated before all fission products are released (recall Subtree K in Fig. 6-6). Releases involving reactivity effects are even less likely and are predicted to occur at frequencies significantly below 5×10^{-7} per plant year (Ref. 6-2).

Releases involving hydrolysis, however, are important safety risk contributors. The salient steam generator leak scenario (Subtree N) involves primary circuit depressurization through the steam generator dump system (Ref. 6-5). Although events involving other heat exchanger leaks (Subtree O) are not expected to have safety risk significance, further analysis to substantiate this contention is recommended.

6.2 Releases from Sources Outside the Reactor Building

Previous assessments (e.g., Ref. 6-3) strongly suggest that releases from sources outside the reactor building will contribute little to the MHTGR safety risk envelope. Nevertheless, so little information pertaining to the design and operation of these systems is currently available that future analyses are required (as the design evolves).

6.3 External Events

The safety characterization presented in Sections 6.1 and 6.2 is for events in which system failures are not strongly coupled. In order to complete the safety characterization, it is necessary to reexamine the MLD

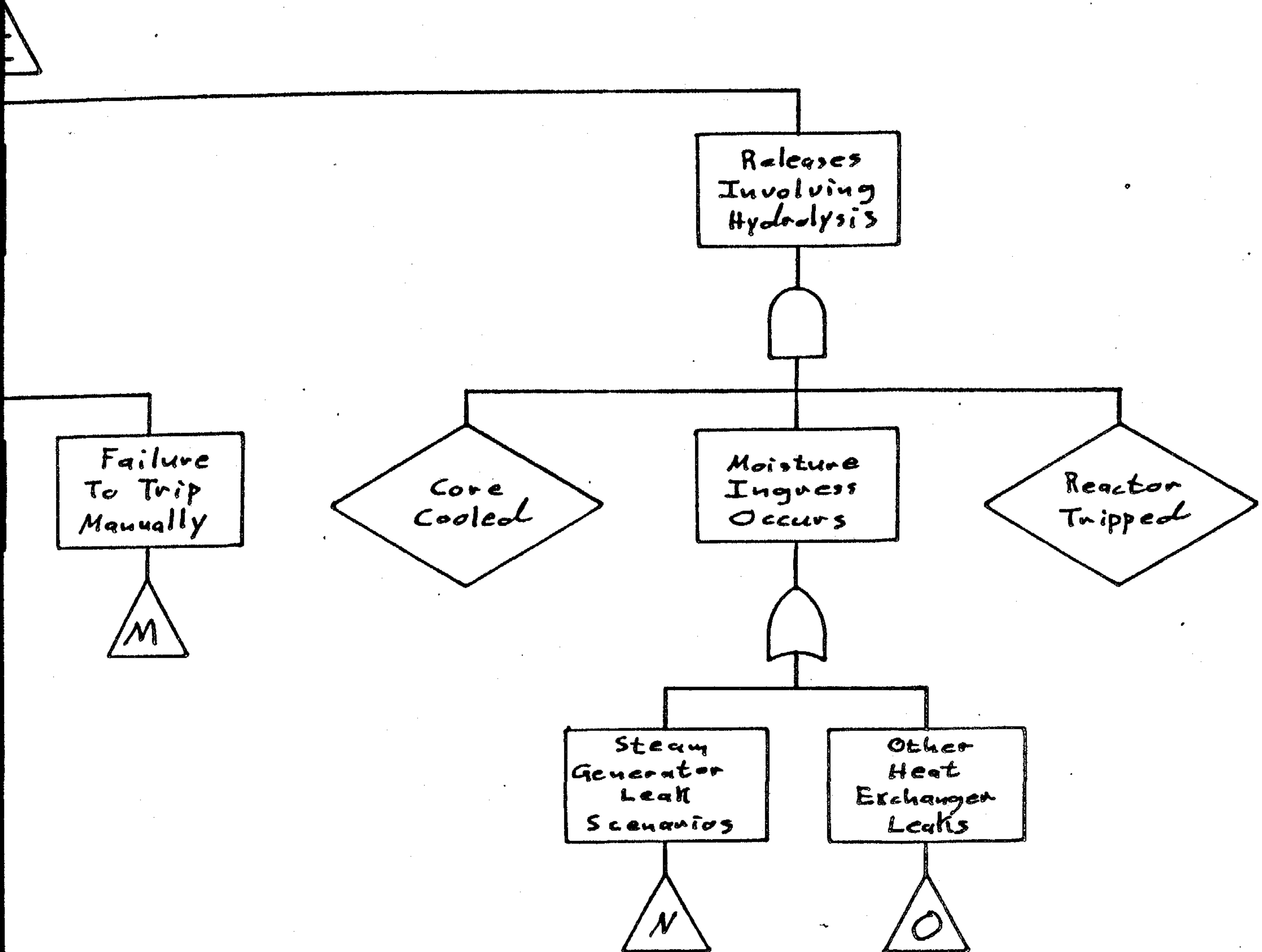
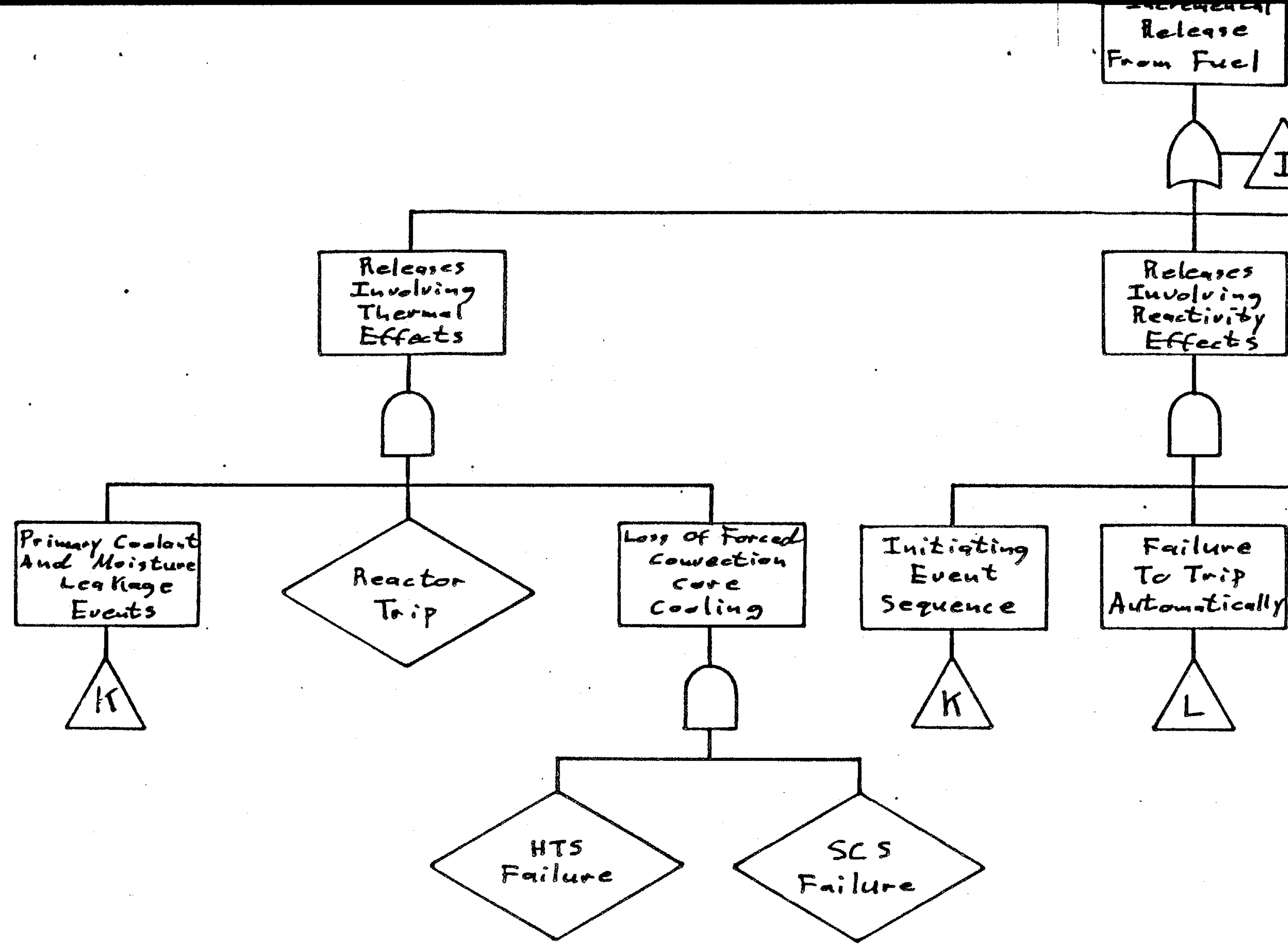


Figure 6-7. Subtree I for Events Involving Reactor Building Bypass and an Incremental Release from the Fuel


Release
From Fuel
I
Releases
Involving
Reactivity
Effects
Releases
Involving
Thermal
Effects
Primary Coolant
And Moisture
Leakage
Events
K
Reactor
Trip
Loss of Forced
Convection
Core
Cooling
HTS
Failure
SCS
Failure
Initiating
Event
Sequence
K
Failure
To Trip
Automatically
L

from the perspective of "external events" (i.e., disruptive events, such as fires, floods, and earthquakes, that can cause widely separated and normally independent plant systems to fail concurrently). Candidate external events for inclusion in a PRA are compiled in Table 6-3 (from Ref. 6-1). Due to limitations in the availability of external event assessments that are readily applicable to the MHTGR, the safety risk impact from external events cannot be characterized without additional analyses exceeding the scope of this section.

6.4 Summary of Events Recommended for Further Evaluation

Table 6-4 lists the events recommended for additional evaluation. The first four columns in the table correspond to the upper levels of the MLD (Fig. 6-1). These columns provide discernment as to whether the:

1. Release is from a source inside of the reactor building.
2. Reactor building functions properly.
3. Primary coolant boundary remains intact.
4. Event causes an incremental release from the fuel.

Additional information needed to fully characterize the event appears in the fifth column, while the last column identifies the section in this report where each event is considered further.

TABLE 6-3
EXTERNAL INITIATING EVENTS

Event	Remarks
Aircraft impact	Site specific; requires detailed study.
Avalanche	Can be excluded for most sites in the United States.
Coastal erosion	Included in the effects of external flooding.
Drought	Excluded because ultimate heat sink is not affected by drought (e.g., air cooling).
External flooding	Site specific; requires detailed study.
Extreme winds and tornadoes	Site specific; requires detailed study.
Fire	Plant specific; requires detailed study.
Fog	Could increase the frequency of man-made hazard involving surface vehicles or aircraft; accident data include the effects of fog.
Forest fire	Fire cannot propagate to the site because the site is cleared; plant design and fire-protection provisions are adequate to mitigate the effects.
Frost	Snow and ice govern.
Hail	Other missiles govern.
High tide, high lake level, or high river stage	Included under external flooding.
High summer temperature	Ultimate heat sink is designed to operate with air.
Hurricane	Included under external flooding; wind forces are covered under extreme winds and tornadoes.
Ice cover	Ice blockage of river included in flood. Potential loss of airflow passage is considered in plant design.
Industrial or military facility accident	Site specific; requires detailed study.
Internal flooding	Plant specific; requires detailed study.
Landslide	Can be excluded for most sites in the United States.
Lightning	Considered in plant design.
Low lake or river water level	Ultimate heat sink is designed for operation with air.

TABLE 6-3 (Cont.)

Event	Remarks
Low winter temperature	Thermal stresses and embrittlement are insignificant or covered by design codes and standards for plant design.
Meteorite	All sites have approximately the same frequency of occurrence.
Pipeline accident (gas, etc.)	Site specific; requires detailed study.
Intense precipitation	Included under external and internal flooding.
Release of chemicals in onsite storage	Plant specific; requires detailed study.
River diversion	Not applicable for air-cooling.
Sandstorm	Included under tornadoes and winds; potential blockage of air intakes with particulate matter is considered in plant design.
Seiche	Included under external flooding.
Seismic activity	Site specific; requires detailed study.
Snow	Plant designed for higher loading; snow melt causing river flooding is included under external flooding.
Soil shrink-swell consolidation	Site-suitability evaluation and site development for the plant are designed to preclude the effects of this hazard.
Storm surge	Included under external flooding.
Transportation accidents	Site specific; require detailed study.
Tsunami	Included under external flooding and seismic events.
Toxic gas	Site specific; requires detailed study.
Turbine-generated missile	Plant specific; requires detailed study.
Volcanic activity	Can be excluded for most sites in the United States.
Waves	Included under external flooding.

TABLE 6-4
EVENTS RECOMMENDED FOR FURTHER EVALUATION

Release From Sources Inside Reactor Building	Reactor Building Functions Properly	Primary Coolant Boundary Remains Intact	Incremental Release From The Fuel	Additional Event Characteristics	Section With Further Considera-tion of Event
Yes	Yes	Yes	-	Instrument line leakage	2.4
Yes	Yes	Yes	-	HPS leakage	2.4
Yes	Yes	Yes	-	Primary coolant over pressure transients	2.4
Yes	Yes	No	No	Primary coolant leakage from the primary coolant - reactor building atmosphere interface	7.3.1
Yes	Yes	No	Yes	Conduction cooldowns	7.3
Yes	Yes	No	Yes	Reactivity addition initiated by moisture inleakage	2.4
Yes	Yes	No	Yes	Large primary coolant leaks that prevent reactor trip	2.4
Yes	Yes	No	Yes	Reactivity addition initiated by a spurious response to control rod insertion	2.4

TABLE 6-4
EVENTS RECOMMENDED FOR FURTHER EVALUATION
(Cont.)

Release From Sources Inside Reactor Building	Reactor Building Functions Properly	Primary Coolant Boundary Remains Intact	Incremental Release From The Fuel	Additional Event Characteristics	Section With Further Considera-tion of Event
Yes	Yes	No	Yes	Reactivity addition initiated by spurious control rod withdrawl	7.3.7
Yes	Yes	No	Yes	Reactivity addition initiated by control rod ejection	2.4
Yes	Yes	No	Yes	Reactivity addition initiated by a spurious response to accidental RSS insertion	2.4
Yes	Yes	No	Yes	Reactivity addition due to fission product decay	7.3
Yes	Yes	No	Yes	Loss of reactivity control due to loss of core geometry	2.4
Yes	No	No	No	Primary coolant leakage from the primary coolant-reactor building atmosphere interface (reactor building not bypassed)	7.3.1

TABLE 6-4
EVENTS RECOMMENDED FOR FURTHER EVALUATION
(Cont.)

Release From Sources Inside Reactor Building	Reactor Building Functions Properly	Primary Coolant Boundary Remains Intact	Incremental Release From The Fuel	Additional Event Characteristics	Section With Further Considera-tion of Event
Yes	No	No	Yes	Large primary coolant leaks with oxidation and reactivity effects (reactor building not bypassed)	2.4
Yes	No	No	Yes	Large primary coolant leaks with oxidation only (reactor building not bypassed)	2.4
Yes	No	No	No	HPS leaks that bypass the reactor building	2.4
Yes	No	No	Yes	Failed fuel particle hydrolysis initiated by a steam generator leak (reactor building bypassed)	7.3.2
No	-	-	-	(Additional design and operating characteristics are needed)	2.4
-	-	-	-	(External events)	2.4 and 7.3.5

6.5 References

6-1. "PRA Procedures Guide," NUREG/CR-2300, Vol. 1, January 1983.

6-2. Everline, C. J., "Master Logic Diagram for the Modular HTGR," RGE 908667/0, April 25, 1986.

6-3. Pasternak, T., et al., "HTGR Accident Initiation and Progression Analysis Status Report Volume III. Preliminary Results (Including Design Options)," GA-A13617, Vol. III, November 1975.

6-4. Hansen, R. H., "Reliability Data Base," RGE 906551/3, October 1985.

6-5. Everline, C. J., et al., "Probabilistic Risk Assessment of the Modular HTGR Plant," HTGR-86-011/0 (908664/0), January 1986.

7. ACCIDENT FREQUENCY ASSESSMENT

The sequence probability quantification portion of the safety risk assessment methodology (Fig. 3-1) is applied in this section. Eight event trees are constructed and quantified. They address:

1. Primary coolant leaks.
2. Small steam generator leaks.
3. Moderate steam generator leaks.
4. Loss of HTS cooling.
5. Loss of offsite power.
6. Control rod withdrawal.
7. Anticipated transients requiring a reactor trip.
8. Earthquakes.

Some of these events (i.e., loss of HTS cooling and loss of offsite power) were shown to contribute negligibly to the MHTGR safety risk envelope in Section 6. Nevertheless, frequency assessments of these events are required by the bridging methods for standout HTGR licensing bases documented in Ref. 7-1.

7.1 Data Base

References 7-2 through 7-24 provided the data base utilized in assessing accident frequencies. Many of the data sources were compiled from operating experience involving hardware in LWR or nonnuclear applications. Employing such data in an HTGR risk assessment is justified because many components (e.g., condensate pumps, circuit breakers, water-to-water heat exchangers) are subjected to a comparable operating environment in the MHTGR. Hence, they should exhibit a comparable reliability.

Reliability data for some MHTGR components cannot be based directly upon LWR, or nonnuclear operating experience, either because the components are unique to the MHTGR design (e.g., the helium circulators), or operate in a dissimilar environment (e.g., the steam generators). In these situations, reliability data garnered during previous HTGR risk assessments (e.g., Refs. 7-2, 7-4, 7-5, 7-7, 7-9, 7-12, 7-15, 7-16, 7-17, and 7-18) are applied. When adequate design definition exists, application of these data result in highly design-specific reliability estimates. When details are sparse due to the current status of the MHTGR design, generic HTGR reliability data are utilized. This correctly implies that reliability estimates can change as the design evolves, which is precisely why periodic revisions to the MHTGR PRA are needed to track the impact of design evolution on Goal 3 compliance.

7.2 Initiating Events

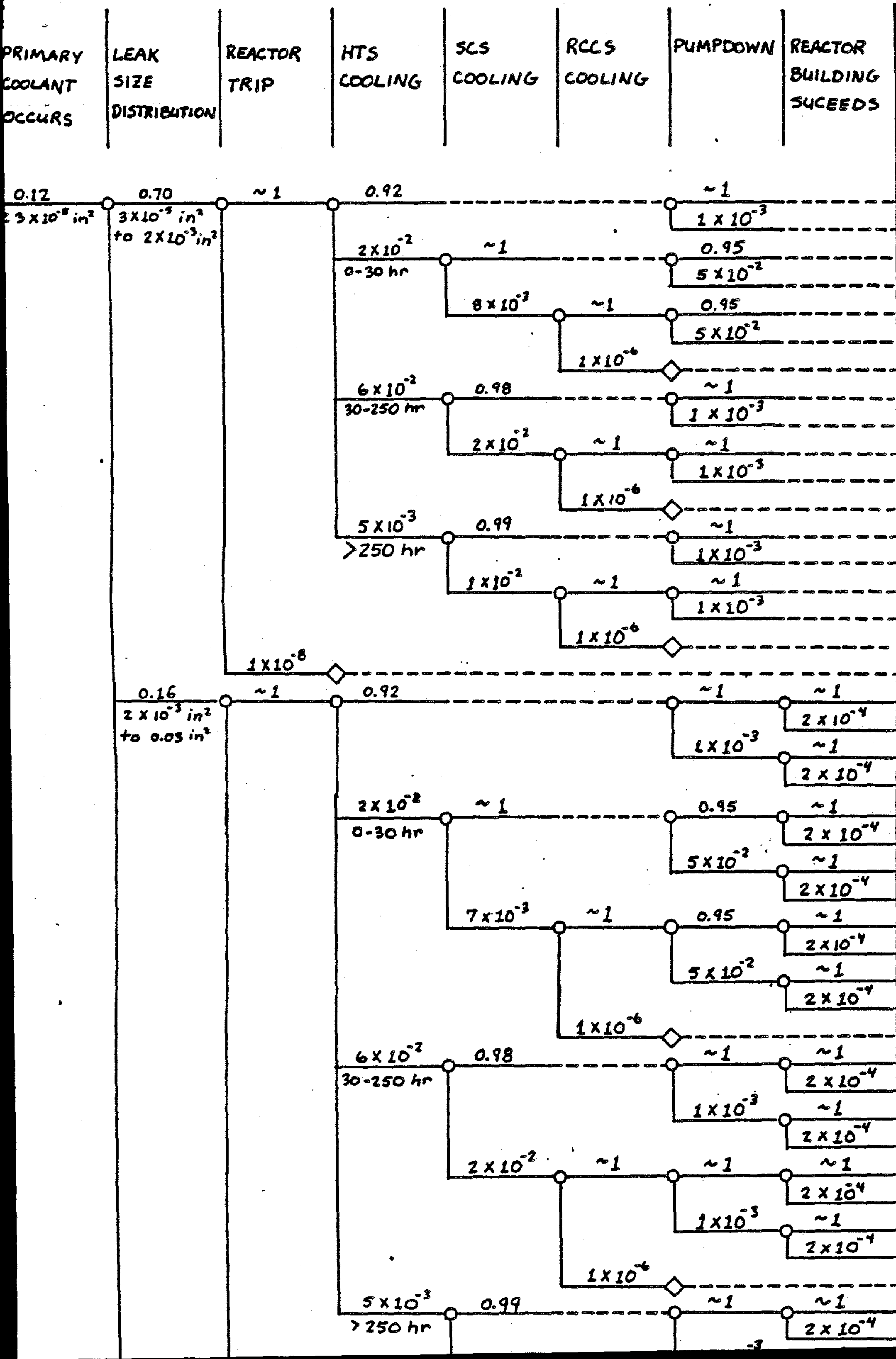
Events selected for sequence probability quantification must ultimately be those important for Goal 3 compliance. These events are those identified in Section 6, along with events important to bridging (Ref. 7-1). Not all of the important Goal 3 related events are evaluated in this section due to scope and schedule limitations. Instead, the eight initiating events included in this section are presently perceived as having the greatest potential impact on Goal 3 compliance. Other initiating events will be incorporated into future PRA revisions until the entire spectrum of salient events has been quantified.

7.3 Event Sequence Analysis

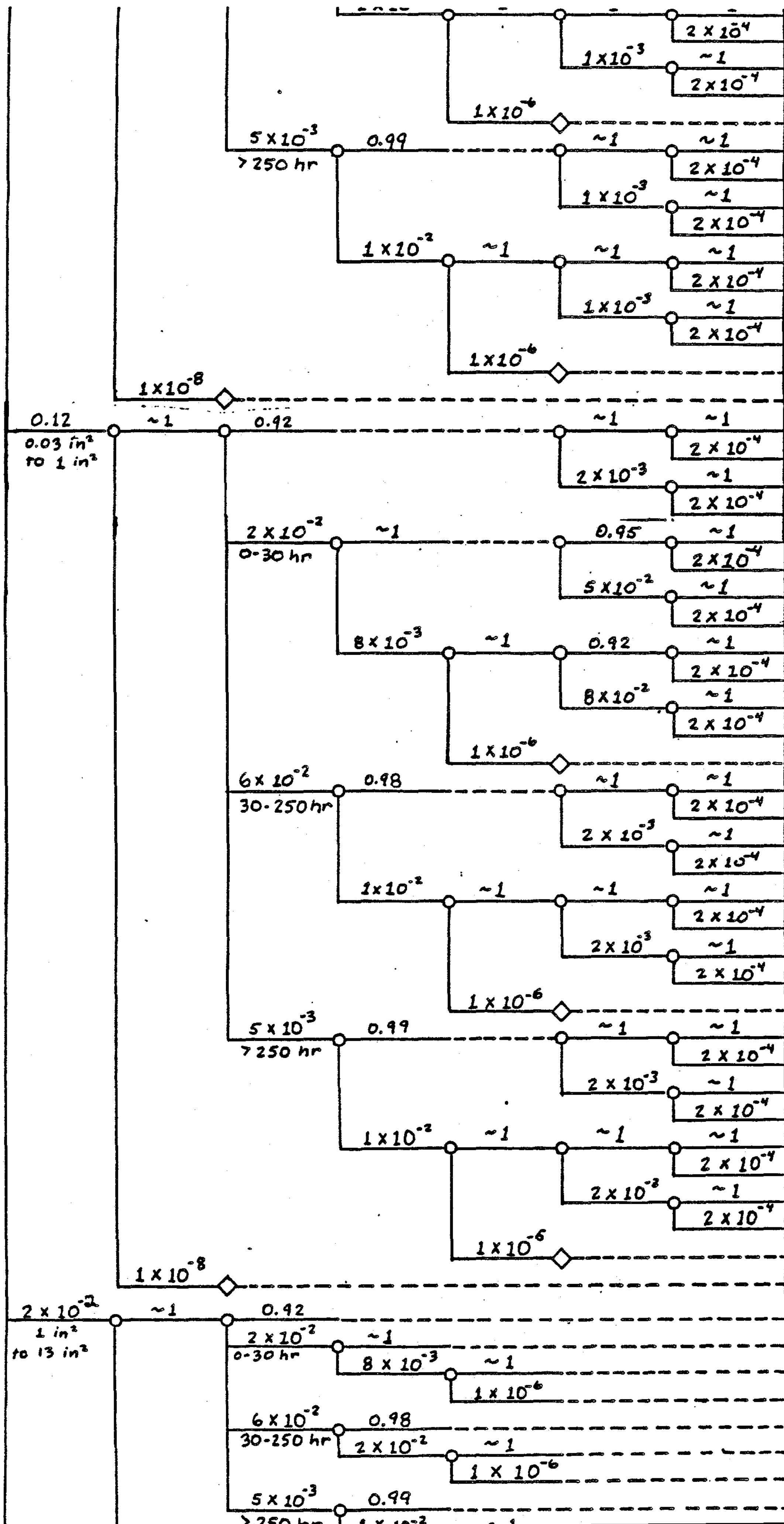
7.3.1 Accidents Initiated by Primary Coolant Leaks

Figure 7-1 is the event tree for accidents initiated by primary coolant leaks. The event descriptions across the top of the tree pertain to the various ways that plant systems respond to (or fail to respond to) the accident. Note that every event sequence in Fig. 7-1 leads to an offsite dose, even if all plant systems respond successfully to the primary coolant leak.

Event tree quantification was predicated upon Ref. 7-5 with the exception of three events. The initiating event frequency and leak size distribution



ID	MEDIAN FREQUENCY PER PLANT YEAR	ACCIDENT FAMILY
AA	8×10^{-2}	PC-10
AB	8×10^{-5}	PC-9
AC	2×10^{-3}	PC-10
AD	8×10^{-5}	PC-9
AE	1×10^{-5}	CCp-12
AF	7×10^{-7}	CCp-11
AG	ε	–
AH	5×10^{-3}	PC-10
AI	5×10^{-6}	PC-9
AJ	1×10^{-4}	CCp-12
AK	1×10^{-7}	CCp-11
AL	ε	–
AM	4×10^{-4}	PC-10
AN	4×10^{-7}	PC-9
AO	4×10^{-6}	CCp-12
AP	4×10^{-9}	CCp-11
AQ	ε	–
AR	ε	–
AS	2×10^{-2}	PC-10
AT	4×10^{-6}	PC-8
AU	2×10^{-5}	PC-9
AV	4×10^{-9}	PC-7
AW	4×10^{-4}	PC-10
AX	7×10^{-8}	PC-8
AY	2×10^{-5}	PC-9
AZ	4×10^{-9}	PC-7
BA	3×10^{-6}	CCp-7
BB	ε	–
BC	1×10^{-7}	CCp-4
BD	ε	–
BE	ε	–
BF	1×10^{-3}	PC-10
BG	2×10^{-7}	PC-8
BH	1×10^{-6}	PC-9
BI	ε	–
BJ	2×10^{-5}	CCp-7
BK	5×10^{-9}	CCp-2
BL	2×10^{-8}	CCp-4
BM	ε	–
BN	ε	–
BO	1×10^{-4}	PC-10
BP	2×10^{-8}	PC-8



BK	5×10^{-9}	CCp-2
BL	2×10^{-8}	CCp-4
BM	ε	—
BN	ε	—
BO	1×10^{-4}	PC-10
BP	2×10^{-8}	PC-8
BQ	1×10^{-7}	PC-9
BR	ε	—
BS	1×10^{-6}	CCp-7
BT	ε	—
BU	1×10^{-9}	CCp-4
BV	ε	—
BW	ε	—
BX	ε	—
BY	1×10^{-2}	PC-6
BZ	3×10^{-6}	PC-5
CA	3×10^{-5}	PC-6
CB	5×10^{-9}	PC-5
CC	3×10^{-4}	PC-6
CD	5×10^{-8}	PC-5
CE	1×10^{-5}	PC-6
CF	3×10^{-9}	PC-5
CG	2×10^{-6}	CCp-9
CH	ε	—
CI	2×10^{-7}	CCp-9
CJ	ε	—
CK	ε	—
CL	8×10^{-4}	PC-6
CM	2×10^{-7}	PC-5
CN	2×10^{-6}	PC-6
CO	ε	—
CP	2×10^{-5}	CCp-9
CQ	3×10^{-9}	CCp-10
CR	3×10^{-8}	CCp-9
CS	ε	—
CT	ε	—
CU	7×10^{-5}	PC-6
CV	1×10^{-8}	PC-5
CW	1×10^{-7}	PC-6
CX	ε	—
CY	7×10^{-7}	CCp-9
CZ	ε	—
DA	1×10^{-9}	CCp-9
DB	ε	—
DC	ε	—
DD	ε	—
DE	2×10^{-3}	PC-4
DF	5×10^{-5}	PC-4
DG	4×10^{-7}	CCp-8
DH	ε	—
DI	1×10^{-4}	PC-4
DJ	3×10^{-6}	CCp-8
DK	ε	—
DL	1×10^{-5}	PC-4

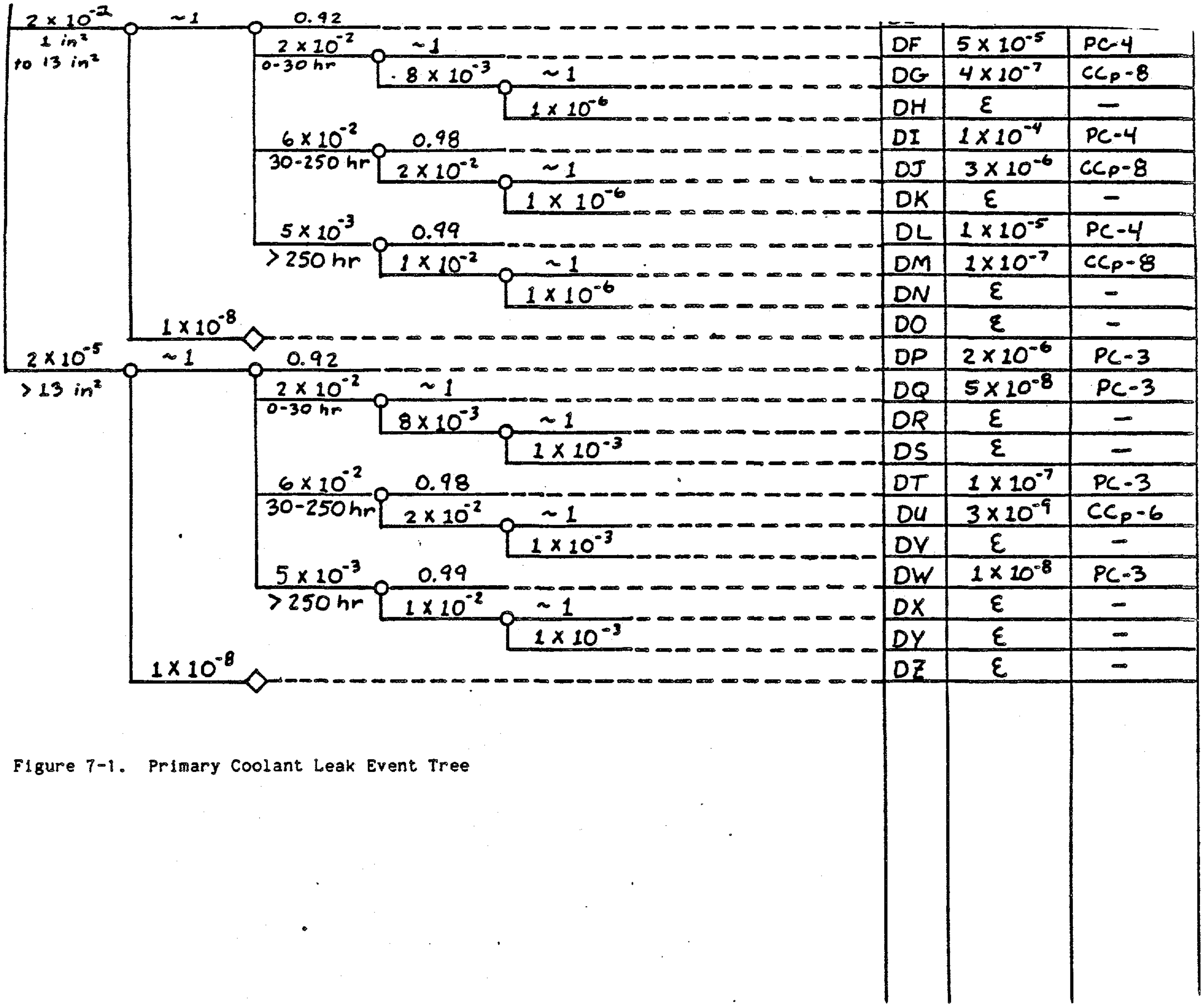


Figure 7-1. Primary Coolant Leak Event Tree

(Events 1 and 2 in Fig. 7-1) have been revised to reflect the impact of recent design modifications, as well as development of a new probabilistic model for primary coolant leaks (Ref. 7-24). The design modifications have resulted from the natural course of design evolution and simply furnish additional details regarding the primary coolant boundary configuration. Reference 7-24 provides a more mechanistic basis for evaluating leak size distribution probabilities.

The RCCS failure probability (Event 6 in Fig. 7-1) has also been revised to correspond to the passive air-cooling system recently adopted for the MHTGR and described in Section 5.

7.3.1.1 Primary Coolant Leak Frequency and Size Distribution. The initiating event in Fig. 7-1 is a primary coolant leak that engenders a module shutdown. Event 2 pertains to the distribution of depressurization areas. If the depressurization area is less than 3×10^{-5} in.^{2}, no module shutdown is required. If the leak size is greater than or equal to 2×10^{-3} in.^{2} and is located in the HTS circulator enclosure, preliminary analyses indicate that the depressurization can damage the circulator wiring. Thus, when the depressurization area A is in the range:

$$2 \times 10^{-3} \text{ in.}^2 \leq A < 3 \times 10^{-2} \text{ in.}^2 \quad ,$$

there is a possibility that the initiating event causes an HTS failure by damaging the HTS circulator. A leak size of 2×10^{-3} in.^{2} is important in that leaks greater than or equal to this size will lift the reactor building dampers, effectively increasing the building leakage rate. Consequently, this leak size is an important depressurization area with respect to reactor building response during the accident. Preliminary analyses also indicate that 3×10^{-2} in.^{2} is the critical leak size for SCS circulator damage. Hence, if

$$3 \times 10^{-2} \text{ in.}^2 \leq A < 1 \text{ in.}^2$$

and the leak is in the SCS circulator enclosure, the initiating event incapacitates the SCS.

If:

$$1 \text{ in.}^2 \leq A \text{ ,}$$

The primary coolant depressurization time is less than an hour. Since the primary coolant inventory decreases exponentially, an HPS pumpdown does not appreciably alter the accident consequences.

Depressurization through a 12 in.^{2} or greater opening is important due to concerns that the initiating event may be energetic enough to damage the RCCS cooling panels.

The assessment of Events 1 and 2 is predicated upon Ref. 7-24.

7.3.1.2 Reactor Trip. Since the initiating event is a primary coolant leak large enough to force a module outage, one of the first responses to the initiating event should be a reactor trip. The probability that the module fails to trip is taken from the Ref. 7-2 risk assessment.

7.3.1.3 Heat Transport System Cooling Maintained. One function of the HTS is to remove decay heat subsequent to a reactor trip. Two generic HTS failure modes are addressed:

1. Failure to survive the initiating event.

2. Failure to remove decay heat (given that the initiating event is survived).

The initiating event can fail the HTS by causing a spurious plant trip or damaging the HTS circulator. Circulator damage was discussed in Section 7.3.1.1. Since the initiating event leads to a forced module outage, an improper response to this loss of thermal power could cause the plant to trip and produce a loss of HTS cooling capability.

Given that the HTS survives the initiating event, the probability that it does not remove an adequate quantity of decay heat is the probability that helium flow in the primary system, water flow in the secondary system, or air flow in the tertiary system is lost.

HTS failure to remove decay heat can be caused by equipment failures that result in a direct loss of helium, water, or air flow (e.g., the circulator, feed pumps, and cooling tower fans), as well as support system failures (i.e., electric power, service air, and service water). Explicitly assessing support system failures is important because they are coupled to other systems besides the HTS. For example, the HPS cannot operate if the HTS fails due to a loss of electric power. Moreover, an electric power outage impacts the SCS reliability, even though the SCS can operate with emergency power.

7.3.1.4 Shutdown Cooling Maintained. Subsequent to an HTS failure, the standard response is to establish and maintain SCS cooling until the HTS is repaired. Two generic SCS failure modes are addressed:

1. Failure to survive the initiating event.

2. Failure to remove decay heat (given that the initiating event is survived).

As is discussed in Section 7.3.1.1, a primary coolant leak in the SCS circulator enclosure can damage the shutdown circulator if:

$$3 \times 10^{-2} \text{ in.}^2 \leq A < 1 \text{ in.}^2 \quad .$$

If the initiating event does not damage the shutdown circulator, its failure probability is the probability that it fails to start, plus the probability that it starts but fails to operate. Both probabilities are conditionally dependent upon the HTS failure mode. Thus, for example, if the HTS fails due to a loss of service water, the probability that the reserve service water system is unavailable contributes to the SCS failure to start probability. Conversely, if the HTS failure is caused by something other than a loss of service water (e.g., a circulator failure), then service water unavailability has a negligible impact on the SCS failure to start probability.

7.3.1.5 Reactor Cavity Cooling Maintained. Implicit in the RCCS reliability model are two generic failure modes:

1. Failure to survive the initiating event.

2. Failure to remove decay heat (given that the initiating event is survived).

The correlation between the initiating event and RCCS failure arises because highly energetic reactor vessel depressurizations (i.e., if $A \geq 13$ in.^{2}) can damage the RCCS cooling panels.

Given that the primary coolant leak does not damage the cooling panels, the RCCS needs only to operate until the HTS or SCS is repaired or the mission time for RCCS cooling elapses (whichever occurs first). Because the RCCS relies on natural air circulation through the cooling panels, its failure probability cannot be assessed by techniques applicable to systems composed of active components (e.g., pumps on valves). Moreover, there is no available operating data upon which to base a meaningful reliability estimate. Hence, in this assessment, 10^{-6} is adopted as the RCCS failure probability (predicated upon engineering judgment) for cases in which the cooling panels are not damaged by the initiating event.

7.3.1.6 Pumpdown. The planned response to a primary coolant leak is intentional vessel depressurization through the HPS. By pumping some primary coolant to the helium storage bottles, the amount of circulating and lifted-off activity released from the vessel is reduced. Recall from Section 7.3.1.1 that when

$$1 \text{ in.}^2 \leq A$$

the pumpdown rate is ineffectual. Below 1 $in.^2$ depressurization areas, the pumpdown failure probability diminishes with the leak size, becoming negligible for very small depressurization areas with concomitantly long depressurization times. This results because even if the pumpdown is delayed or interrupted, large fractions of primary coolant can still be transferred to the storage bottles if the leak area is small.

The HPS unavailability and failure to operate probability are predicated upon the Ref. 7-16 assessment.

Loss of electric power is also an HPS failure mode. Consequently, the pumpdown failure probability is conditionally dependent upon whether the HTS and SCS function successfully or fail. For example, if the HTS operates during the first 30 h following the reactor trip, then the probability that the HPS is deprived of power is zero because the HTS and HPS are both connected to the nonessential distribution system; and the intentional depressurization time is 30 h or less. However, if the HTS fails during the first 30 h, even if the SCS operates successfully, there is a chance that the pumpdown fails due to a loss of power because the HPS (unlike the SCS) is not connected to the emergency electrical system.

7.3.1.7 <u>Successful Reactor Building Response</u>. The module reactor building furnishes three dose attenuation mechanisms:

1. Settling
2. Plateout
3. Holdup (which allows decay)

The effectiveness of these three mechanisms depends upon the leak size (see Section 8.1) and reactor building response. One reactor building failure mode is identified: failure to isolate filters following transient initiation combined with failure to disengage the fans.

This failure mode increases offsite doses by forcing fission products that would normally have been retained in the building, directly to the atmosphere.

7.3.2 Accidents Initiated by Small Steam Generator Leaks

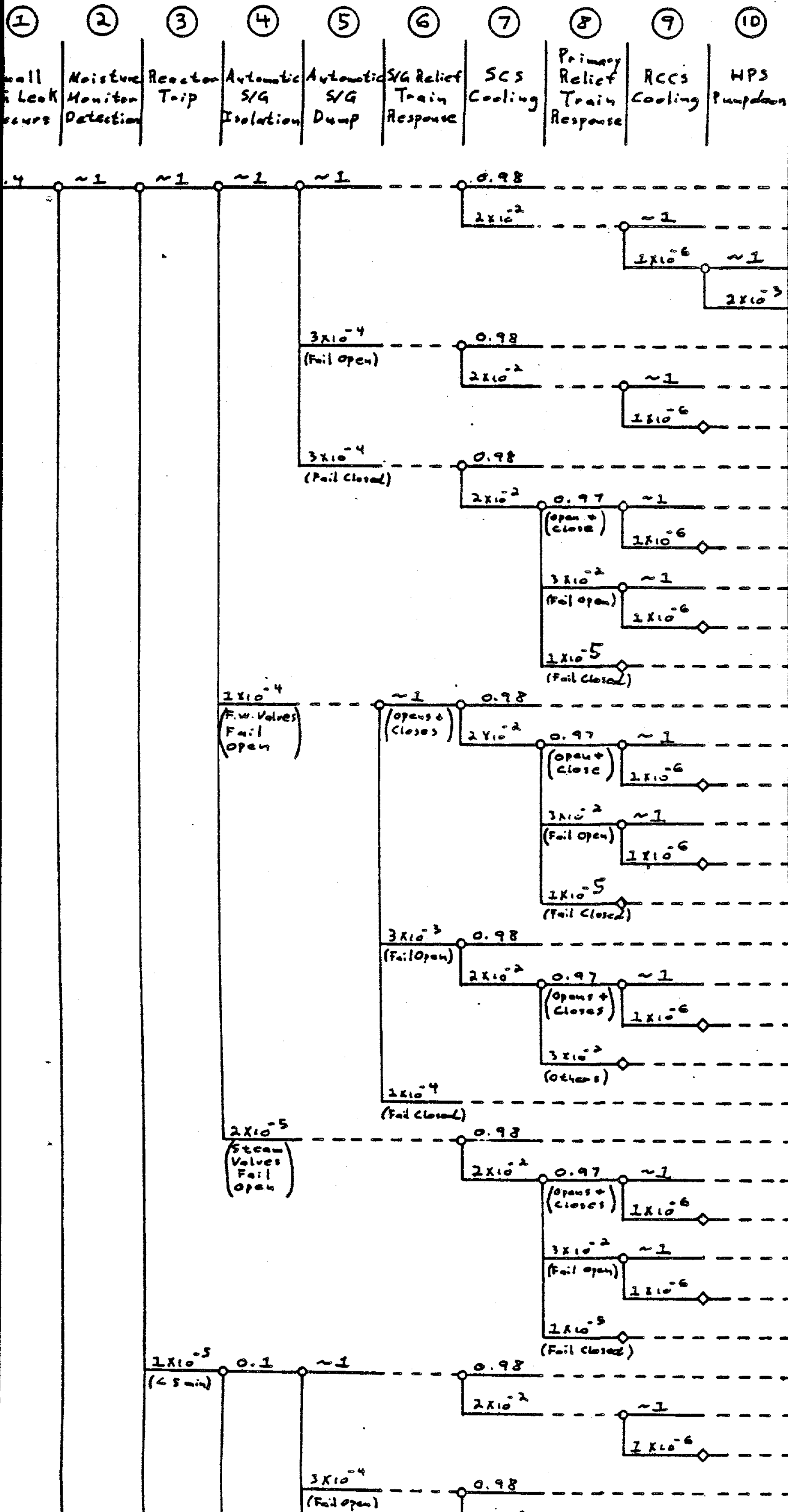
Figure 7-2 is the event tree for accidents initiated by small steam generator leaks. The event descriptions across the top of the tree pertain to the various ways that plant systems respond to (or fail to respond to) the accident. Note that many event sequences in Fig. 7-2 result in no offsite doses. Only in those sequences where certain protective functions fail subsequent to the leak are offsite doses incurred.

Event tree quantification is predicated upon Ref. 7-25 except for sequences in which operator intervention is important. The current design philosophy emphasizes a high degree of plant automation, with the operator assuming the role of a mission manager who monitors plant operations, but rarely influences them. To quantify the failure probability of an operator cast in this new role, the model for cognitive human errors given in Ref. 7-23 is utilized. The principal impact of this modeling revision is to increase the average operator response time in sequences where automated systems (e.g., the steam generator isolation system) fail.

7.3.2.1 Steam Generator Leak Frequency. The steam generator leak frequency was derived by applying the Ref. 7-25 methodology to the MHTGR steam generator design. The dominant initiating event contributors are:

1. Bimetallic weld failure (6×10^{-3} per steam generator year)
2. Corrosion (4×10^{-2} per steam generator year)
3. Similar weld failure (5×10^{-2} per steam generator year)
4. Mechanical damage (4×10^{-3} per steam generator year)

for total frequency of 0.1 per steam generator year. This is the frequency at which leaks of any size can occur. Since the MHTGR has a total of four steam generators, the initiating event frequency is 0.4 per plant year.



I.D.	Number of Modules Experiencing Event Sequence	Median Frequency Per Plant Year	Accident Family
SS-AA	1	0.4	SG-N1
AB	1	8×10^{-3}	CC_s-N1
AC	1	8×10^{-9}	CC_s-N3
AD	1	ϵ	—
AE	1	1×10^{-4}	SG-4
AF	1	2×10^{-6}	CC_s-10
AG	1	ϵ	—
AH	1	1×10^{-4}	SG-N5
AI	1	2×10^{-6}	CC_s-7
AJ	1	ϵ	—
AK	1	7×10^{-8}	CC_s-5
AL	1	ϵ	—
AM	1	ϵ	—
AN	1	4×10^{-5}	SG-N6
AO	1	8×10^{-7}	CC_s-4
AP	1	ϵ	—
AQ	1	2×10^{-8}	CC_s-6
AR	1	ϵ	—
AS	1	ϵ	—
AT	1	1×10^{-7}	SG-7
AU	1	2×10^{-9}	CC_s-3
AV	1	ϵ	—
AW	1	ϵ	—
AX	1	4×10^{-9}	SG-13
AY	1	8×10^{-6}	SG-N6
AZ	1	2×10^{-7}	CC_s-4
BA	1	ϵ	—
BB	1	5×10^{-9}	CC_s-6
BC	1	ϵ	—
BD	1	ϵ	—
BE	1	4×10^{-6}	SG-N1
BF	1	2×10^{-8}	CC_s-N1
BG	1	ϵ	—
BH	1	1×10^{-9}	SG-4

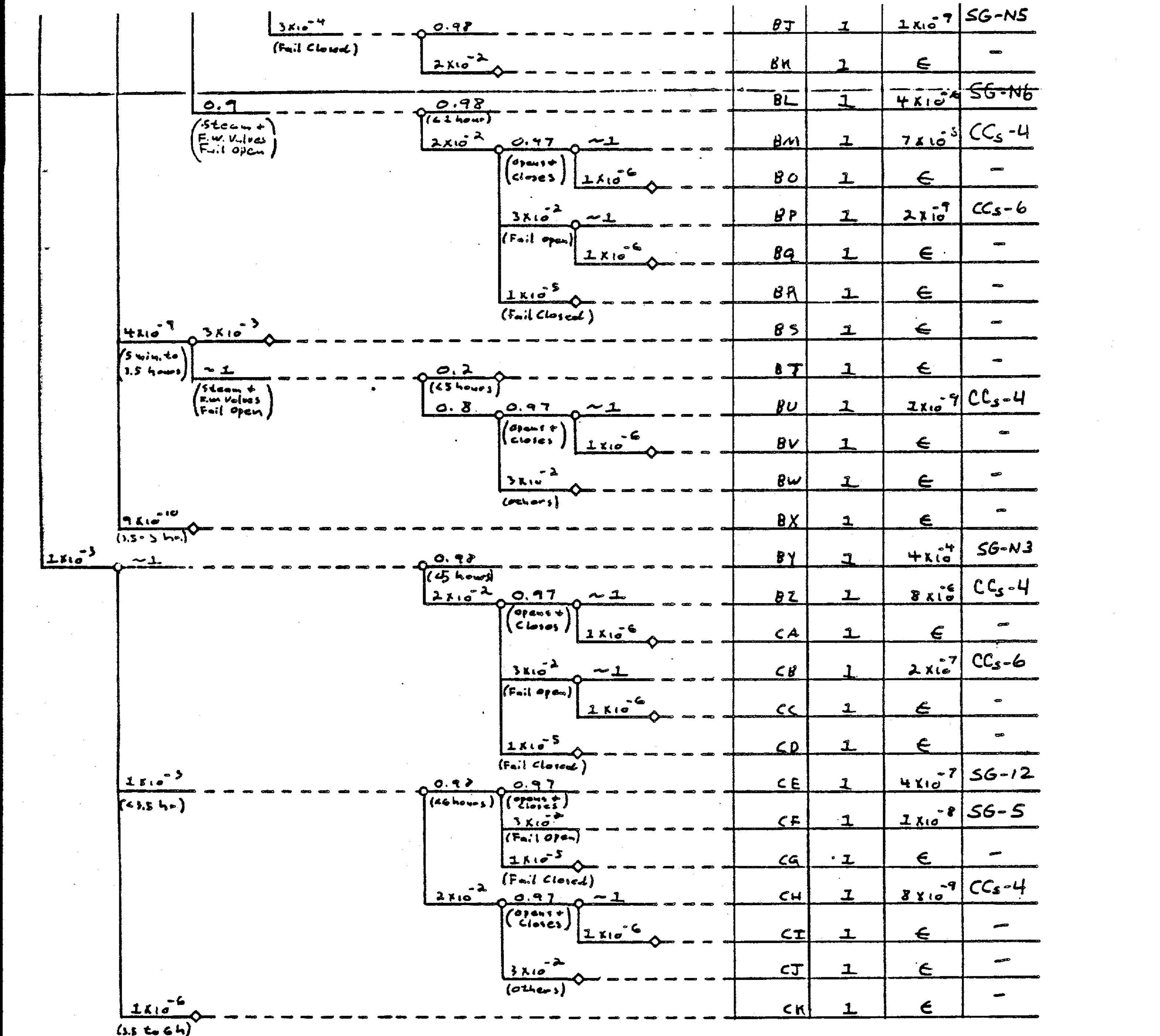


Figure 7-2. Small Steam Generator Leak Event Tree

7.3.2.2 Moisture Monitor Detection. In order to protect the reactor from water ingress, moisture monitors are installed in each module. If high moisture levels are detected, the module is supposed to experience an automatic reactor trip followed by steam generator isolation and dump. References 7-2 and 7-5 are sources for the moisture monitor failure probability and uncertainty.

If the moisture monitors fail to function adequately, other trip setpoints (e.g., high vessel pressure) will initiate an automatic reactor trip. However, the steam generator isolation and dump systems must be initiated manually under this condition.

7.3.2.3 Reactor Trip. The probability that the reactor is not automatically tripped on the outer control rods is predicated upon Ref. 7-2. Reference 7-26 is the basis for the RSS failure probability estimate. As is noted in Section 7.3.2, Ref. 7-23 is utilized to evaluate operator error probabilities.

7.3.2.4 Automatic Steam Generator Isolation. A function of the steam generator isolation system is to limit the amount of water that enters the primary circuit by closing a set of feedwater and steam outlet block valves. In addition to the set of block valves, the steam generator outlet can also be isolated (against reverse flow) by a check valve. Three system failure modes are considered:

1. Only the feedwater valves fail open.
2. Only the steam valves (including the check valve) fail open.
3. Both sets of isolation valves fail open.

The failure probabilities for this event are from Refs. 7-2 and 7-5 with two exceptions:

1. The probability that the steam valves fail open was obtained from the Ref. 7-2 failure probability and the conditional probability of check valve sticking.

2. Given that the reactor is not tripped within five or fewer minutes when the moisture monitors function properly, the conditional probability of an automatic steam generator isolation system failure is governed by the probability that a PPIS logic fault prevented outer control rod insertion (from Ref. 7-2).

7.3.2.5 Automatic Steam Generator Dump. Following a successful steam generator isolation, the dump valves are designed to open in order to transfer most of the steam generator water inventory to the dump tank. To prevent a primary coolant depressurization through the dump system, the dump valves normally close just before the steam generator pressure reaches primary coolant pressure. If the dump valves fail open, there is a primary coolant depressurization through the dump tank directly to the atmosphere. If the dump valves fail closed, a larger than intended amount of water will enter the primary circuit from the steam generator. Dump system failure probabilities are from Ref. 7-27.

7.3.2.6 Steam Generator Relief Train Response. Normally, the steam generator relief train operates at approximately twice primary coolant pressure. Thus, the probability of a spurious relief valve opening at primary coolant pressure is negligible. However, if the feedwater isolation valves fail open but the steam generator outlet valves close on demand, the resultant pressure transient will lift the steam generator relief valve. If this valve fails open, radioactivity in the vessel has a direct pathway to the atmosphere. Given that the steam generator relief valve opens, the probability that it fails open is from Ref. 7-4. The probability that the steam generator relief valve fails to open on demand is predicated upon Refs. 7-4 and 7-7.

7.3.2.7 Shutdown Cooling System Cooling Maintained. The purpose of the SCS in Fig. 7-2 is to prevent or mitigate offsite doses, depending upon which particular event sequence is under consideration. If the reactor trips and the steam generator is automatically isolated and dumped, the SCS serves to reduce the possibility of an initially pressurized conduction cooldown without

RCCS cooling (e.g., sequence SS-AC). In sequences involving isolation or dump system failure, the SCS can serve to prevent the combination of an increased primary circuit inventory (due to the added moisture) and higher than normal temperatures (which are produced during pressurized conduction cooldowns) from lifting the primary coolant relief train. Finally, given that the primary circuit depressurizes, the SCS can mitigate the resultant doses by preventing a thermally induced fission product release from the fuel.

7.3.2.8 Primary Relief Train Response. Preliminary calculations indicate that as long as SCS cooling is maintained, the probability of a small steam generator leak producing primary circuit pressures high enough to require relief is negligible. Thus the possibility of opening the primary relief train is only considered in sequences that include SCS failure. For these scenarios, the probability that the relief train remains closed is the probability that the operator intervenes before the relief valve setpoint is reached. Given that the relief valve setpoint is reached, the probability that the relief valve fails open is from Ref. 7-4. A Ref. 7-4 datum and an estimated common mode factor of 0.1 were combined to quantify the probability that both relief trains fail closed after being subjected to their setpoint pressure.

7.3.2.9 Reactor Cavity Cooling Maintained. The RCCS reliability is predicated upon Section 7.3.1.5.

7.3.2.10 Pumpdown. If both SCS and RCCS cooling are lost and the primary circuit is pressurized, the primary coolant boundary can experience significant stress at elevated temperatures. In order to protect the primary coolant boundary, it is anticipated that the operator attempts a pumpdown with the HPS. If the pumpdown attempt fails (see sequence AD in Fig. 7-2), it is postulated that the vessel ultimately depressurizes as a result of primary coolant boundary failure.

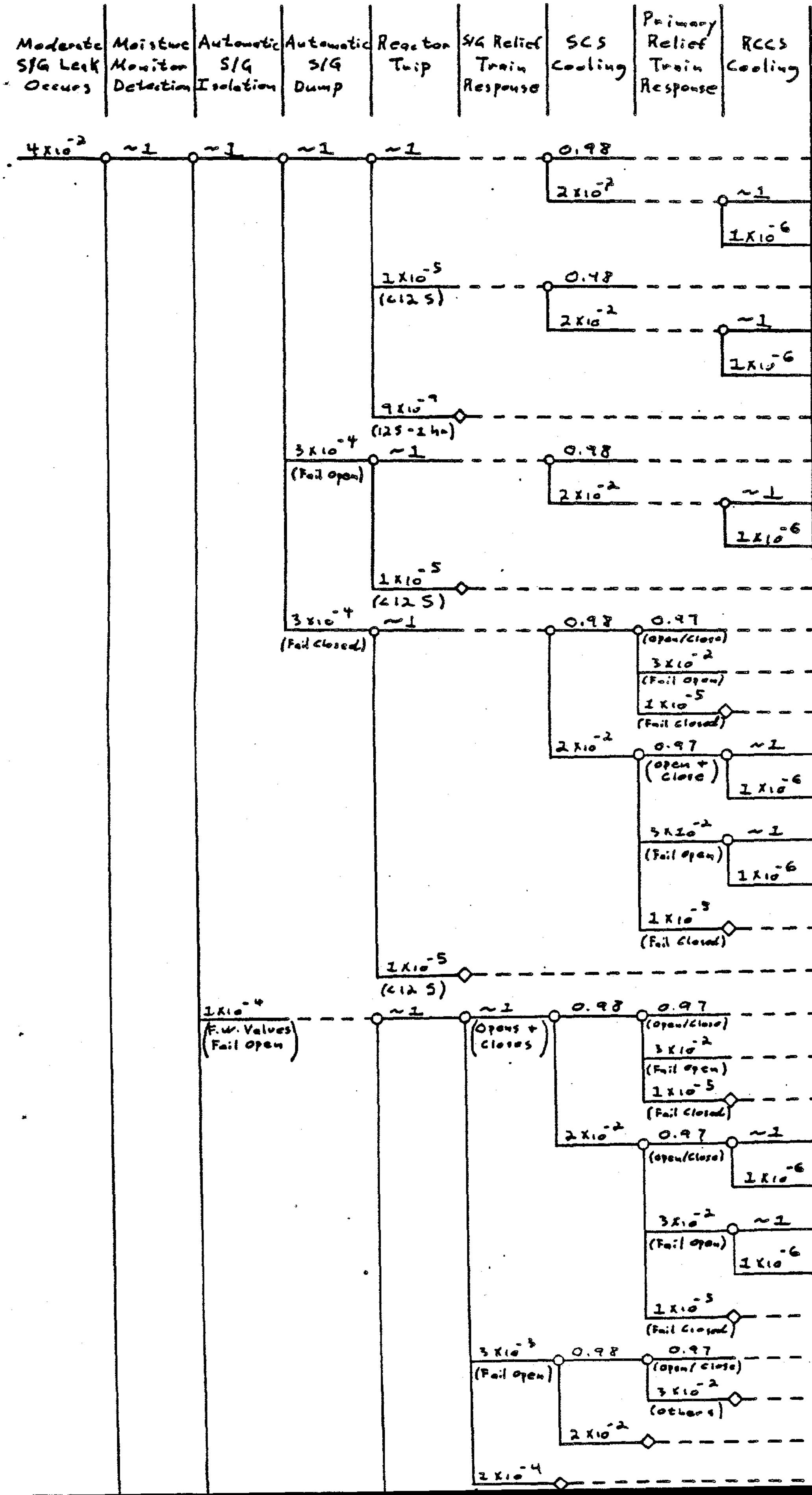
7.3.3 Accidents Initiated by Moderate Steam Generator Leaks

Figure 7-3 is the event tree for accidents initiated by moderate steam generator leaks. Any leak with an ingress rate between 0.1 and 12.5 lbm/s is classified as moderate. An upper bound leak rate of 12.5 lbm/s was selected because it corresponds to an offset steam generator tube rupture, and available data (Ref. 7-2) indicate that the probability of a larger size leak occurring is negligible. Predicated upon Ref.7-2, an average steam generator leak of moderate size has an ingress rate of approximately 2.6 lbm/s, and less than 30% of all moderate steam generator leaks exceed this mean value.

Distinguishing between small and moderate steam generator leaks is phenomenologically important due to inherent differences in occurrence rates and response times. Since small steam generator leak events progress slowly, there is a relatively long operator response time and a concomitantly high probability of successful operator intervention preventing or mitigating the offsite dose. Subsequent to a moderate steam generator leak, events progress more rapidly. Successful operator intervention is less likely to occur, and the potential for larger releases (than result from smaller leaks) exists. Therefore, to better model these differences, two steam generator leak event trees have been constructed. In comparing them, note that the event descriptions across the tops of Figs. 7-2 and 7-3 have dissimilarities caused by the variance in event progressions.

7.3.3.1 Moderate Steam Generator Leak Frequency. Reference 7-2 indicates that approximately 10% of all HTGR steam generator leaks exceed a 0.1 lbm/s ingress rate. Since only 10% of all MHTGR leaks are consequently expected to be of moderate size, the moderate steam generator leak frequency is 10% of the frequency calculated in Section 7.3.2.1.

7.3.3.2 Moisture Monitor Detection. The moisture monitor detection ability is relatively insensitive to the ingress rate. Hence, their reliability subsequent to a moderate steam generator leak is predicated upon Section 7.3.2.2.



Id.	Number of Modules Experiencing Event Sequence	Median Frequency Per Plant Year	Accident Family
MS-AA	1	4×10^{-2}	SG-N2
AB	1	8×10^{-4}	CCs-N2
AC	1	ε	—
AD	1	4×10^{-7}	SG-N2
AE	1	8×10^{-9}	CCs-N2
AF	1	ε	—
AG	1	ε	—
AH	1	1×10^{-5}	SG-8
AI	1	2×10^{-7}	CCs-9
AJ	1	ε	—
AK	1	ε	—
AL	1	1×10^{-5}	SG-10
AM	1	4×10^{-7}	SG-2
AN	1	ε	—
AO	1	2×10^{-7}	CCs-13
AP	1	ε	—
AQ	1	7×10^{-9}	CCs-11
AR	1	ε	—
AS	1	ε	—
AT	1	ε	—
AU	1	4×10^{-6}	SG-9
AV	1	1×10^{-7}	SG-3
AW	1	ε	—
AX	1	8×10^{-8}	CCs-14
AY	1	ε	—
AZ	1	2×10^{-9}	CCs-12
BA	1	ε	—
BB	1	ε	—
BC	1	1×10^{-8}	SG-11
BD	1	ε	—
BE	1	ε	—
BF	1	ε	—

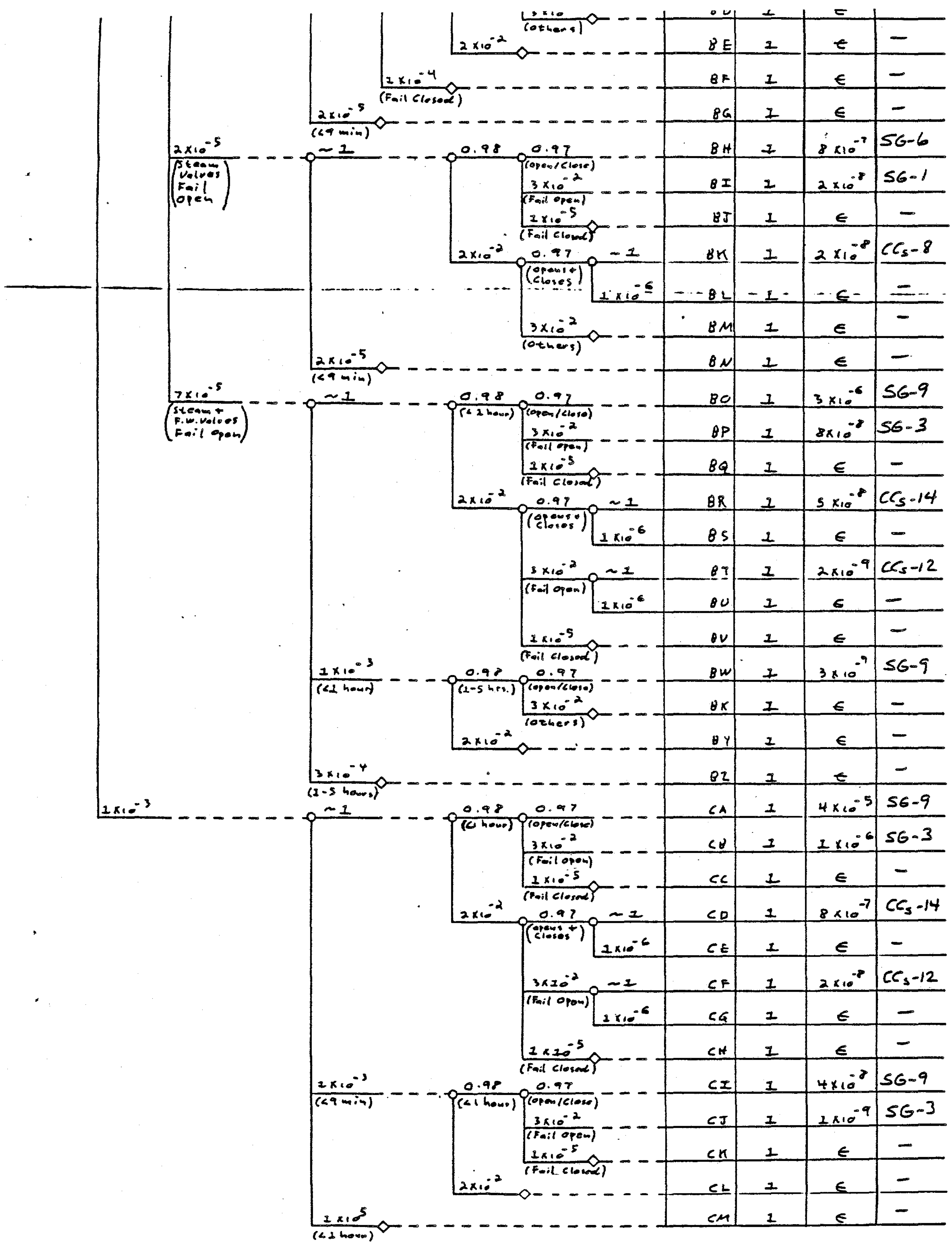


Figure 7-3. Moderate Steam Generator Leak Event Tree

7.3.3.3 Automatic Steam Generator Isolation. Subsequent to receiving a high moisture signal, the steam generator isolation system reliability is solely a function of its component reliabilities. Therefore, the automatic steam generator isolation conditional probabilities in Figs. 7-2 and 7-3 are equal.

7.3.3.4 Automatic Steam Generator Dump. The conditional probability of an automatic steam generator dump is independent of whether the leak is moderate or small. Thus, the automatic steam generator dump probabilities in Figs. 7-2 and 7-3 are also equal.

7.3.3.5 Reactor Trip. It is currently estimated that if 800 kg or more of water enter a MHTGR core, cold shutdown cannot be maintained by inserting either the outer control rods or RSS alone. With a small steam generator leak, being unable to trip on either the outer control rods or RSS alone is not an issue because a minimum of 98 h are required to accumulate 800 kg of water in a MHTGR core at a 0.1 lbm/s ingress rate (98 h is a minimum because it is postulated that adequate helium flow through the steam generator is maintained to efficiently transport moisture from the steam generator vessel to the reactor core). Thus, in Fig. 7-2 the reactor trip failure probability is the probability that neither the outer control rods nor the RSS are inserted.

At a 12.5 lbm/s ingress rate, 800 kg of moisture could (if adequate transport conditions are maintained) accumulate in a MHTGR core in less than 0.8 h. Therefore, given that the steam generator is not automatically isolated following a moderate leak and that the operator fails to intervene before a large quantity of water accumulates in the reactor core, the conditional reactor trip failure probability is the probability that either the outer control rods or RSS are not inserted. These phenomenological differences are responsible for the top events in Figs. 7-2 and 7-3 being sequenced differently, and the higher failure probabilities associated with some Fig. 7-3 scenarios.

7.3.3.6 Steam Generator Relief Train Response. The probabilistic assessment of the steam generator relief train response is described in Section 7.3.2.6.

7.3.3.7 Shutdown Cooling System Cooling Maintained. The mechanistic SCS response to small and moderate steam generator leaks is sufficiently similar that the SCS reliability given in Fig. 7-3 is from the Section 7.3.2.7 assessment.

7.3.3.8 Primary Relief Train Response. Another difference in the MHTGR response to small and moderate steam generator leaks is that maintaining SCS cooling subsequent to a moderate leak is not, by itself, sufficient to preclude primary relief train opening. However, given that the primary relief train setpoint is reached, the probability of each particular relief train response is predicated upon the Section 7.3.2.8 evaluation.

7.3.3.9 Reactor Cavity Cooling Maintained. Steam generator leaks do not impact the RCCS reliability. Hence, the RCCS failure probability from Section 7.3.1.5 is applicable to the moderate steam generator leak event tree.

7.3.4 Loss of Heat Transport System (LM).

The initiating event for the LM tree is any failure of the HTS, except from the loss of offsite electric power, which leads to a loss of HTS cooling.

7.3.4.1 Loss of HTS Cooling. The event sequence following a loss of normal cooling via the HTS is shown in Fig. 7-4. Event 1 gives the frequency of failure in the normal power conversion train that precludes further heat rejection either at power or shutdown. Of this failure rate, approximately 20% are due to failures within the BOP.

7.3.4.2 Reactor Trip. Upon loss of normal cooling the reactor is automatically tripped and the SCS started. Event 2 considers the possibility that reactor trip is unsuccessful. Both control rods and the RSS are considered.

7.3.4.3 SCS Cooling. Event 3 considers whether the SCS successfully starts and operates until the HTS is restored or adequate decay heat is removed from the core.

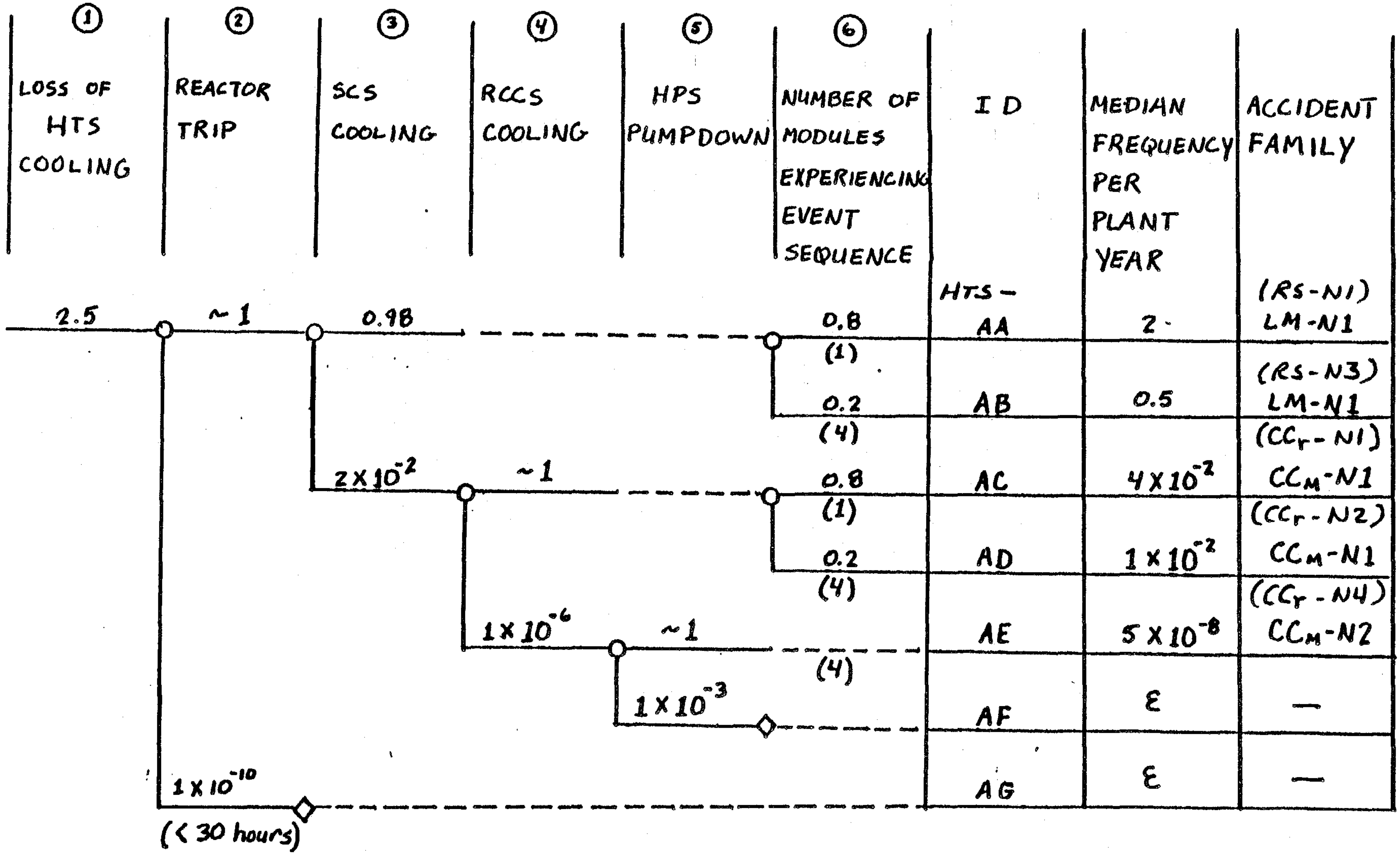


Fig. 7-4. Loss of HTS Cooling Event Tree

7.3.4.4 RCCS Cooling. Should the HTS and SCS fail, heat rejection is provided by conduction, local convection, and radiation to the RCCS (Event 4). In this mode of cooling the vessel remains pressurized.

7.3.4.5 Pumpdown Through HPS. The reactor is intentionally depressurized in a controlled manner by pumping the primary coolant through the HPS to the helium storage bottles (Event 5).

7.3.4.6 Number of Modules Experiencing Event Sequence. Given a loss of HTS cooling, there is a 20% chance that all four modules experience the loss because they are coupled to a common power conversion train (Section 7.3.4.1). There is also an 80% chance that a spurious HTS circulator trip or similar fault resulted in a loss of HTS cooling to only one module. If the SCS also fails, there is a possibility that all four modules are without HTS and SCS cooling. This can occur if the HTS failure resulted from a power conversion train fault, and the SCS cooling water system fails as well. To first order, other failure modes result in only one module having neither HTS nor SCS cooling.

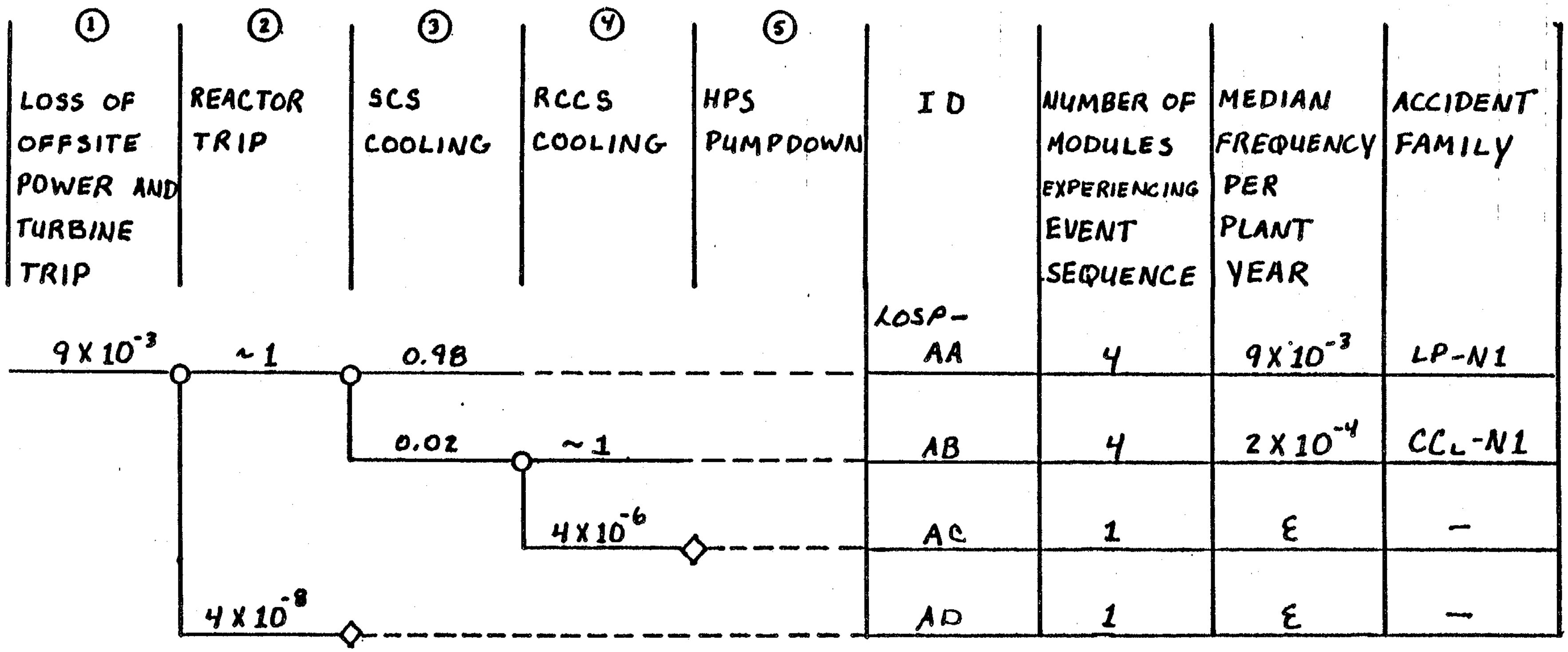
If the RCCS is additionally unable to furnish cooling, only one module is likely to experience the event sequence since each RCCS is independent of the others in the absence of disruptive external events.

7.3.5 Loss of Offsite Power and Turbine Trip

The initiating event is a loss of all offsite power coupled with inability to maintain house loads with the power conversion train.

7.3.5.1 Loss of Offsite Power and Turbine Trip Event. The event sequence following a loss of offsite power and turbine trip is shown in Fig. 7-5. The frequency of Event 1 is 9×10^{-3}/yr as developed in Ref. 7-28.

7.3.5.2 Reactor Trip. Upon losing offsite power and the power conversion train, the reactor is immediately tripped. Both control rods and the RSS are considered.



(ε DENOTES FREQUENCIES BELOW 10^{-9} PER MODULE YEAR)

Fig. 7-5. Loss of Offsite Power and Turbine Trip Event Tree

7.3.5.3 SCS Cooldown. The SCS heat removal is considered in Event 3. The SCS operation is dependent upon the startup of the diesel generators. If offsite power is restored, it is assumed that the HTS becomes available to cool the core. Failure of the diesels to run is a small contributor due to the short mission time to restore cooling.

7.3.5.4 RCCS Cooldown. In the event the SCS is not operational, heat rejection is provided by conduction, local convection, and radiation to the RCCS (Event 4). This mode of cooling was described in the loss of HTS tree.

7.3.5.5 Pumpdown Through HPS. In the event that the HTS, SCS, and RCCS fail, intentional depressurization through the HPS (Event 5) is attempted.

Since a pressurized conduction cooldown to the environment would lead to primary coolant boundary damage, the intentional pumpdown is employed to protect the primary coolant boundary. With the primary coolant boundary intact, no fission products are released to the reactor building or atmosphere during the event sequence.

7.3.6 Control Rod Bank Withdrawal

Figure 7-6 is the event tree for spurious control rod bank withdrawal from a module operating at power. This is one of three new (relative to Ref. 7-28) event trees and whose importance was recognized as part of the Section 6 safety characterization. The other two events are presented in Sections 7.3.7 and 7.3.8.

7.3.6.1 Spurious Control Rod Bank Withdrawal from A Module at Power. Reference 7-23 cites 2×10^{-2} per reactor year as the frequency of spurious control rod withdrawals. Since the MHTGR control strategy requires operating the rods in banks rather than individually, 2×10^{-2} was adopted as the frequency of a spurious control rod bank withdrawal per module year. By postulating that reactivity control in each module is relatively independent of the

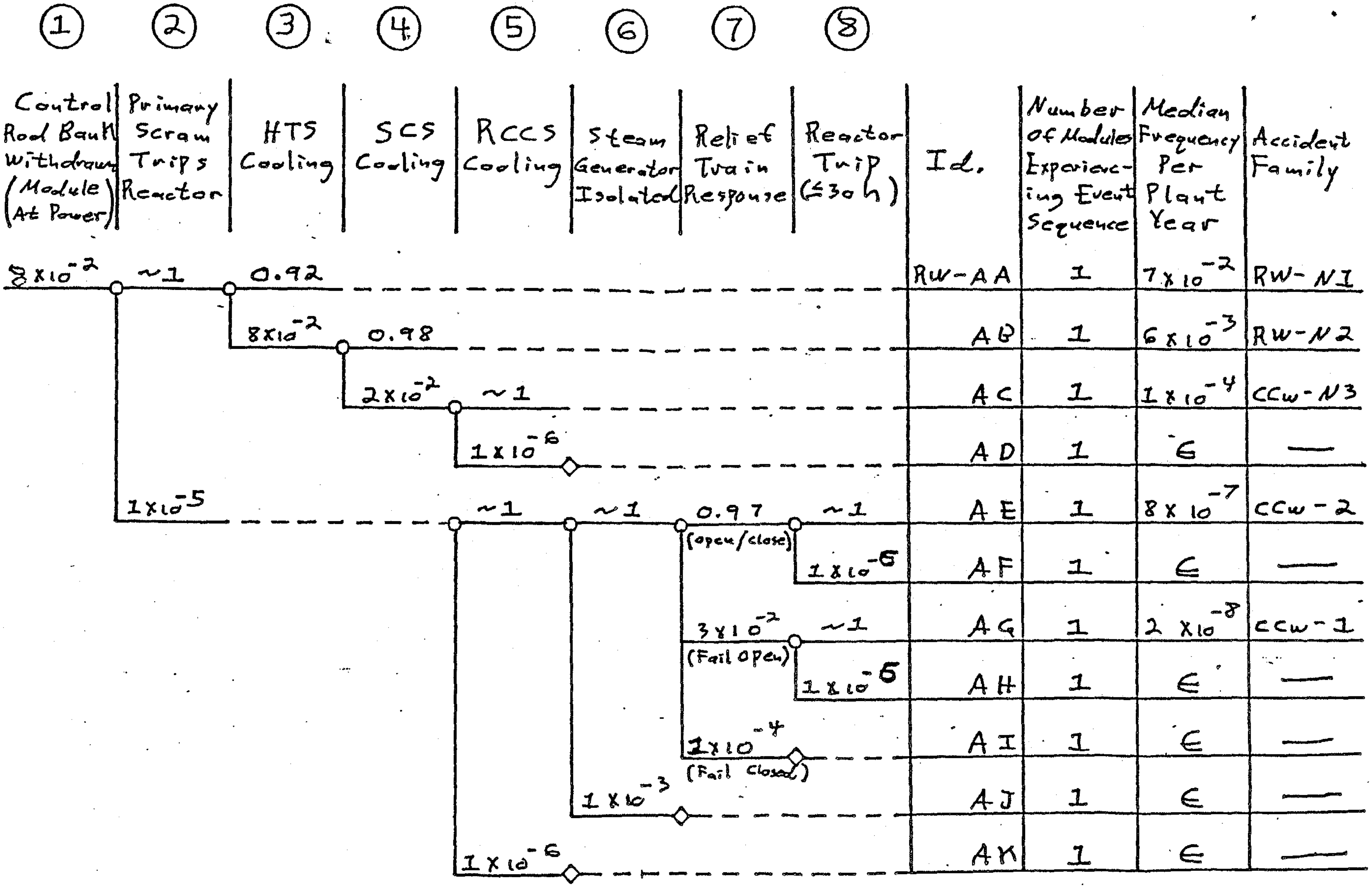


Figure 7-6. Control Rod Bank Withdrawal Event Tree

others, 8×10^{-2} is the frequency per plant year of a spurious control rod bank withdrawal from a module at power.

7.3.6.2 <u>Primary Scram Trips Reactor</u>. The probability that the module is not automatically tripped immediately following the spurious rod banks withdrawal is predicated upon Ref. 7-2.

7.3.6.3 <u>Heat Transport System Cooling Maintained</u>. The normal response to a spurious control rod bank withdrawal followed by an automatic trip is to continue cooling the module with the HTS. However, if the reactor is not tripped, core cooling with the HTS is precluded. This occurs because the reactor core generates heat (through radioactive decay and fission) at a rate that just compensates for the HTS energy removal rate in order to maintain a core temperature profile corresponding to the new control rod configuration.

In addition to being unable to lower the core temperature, the HTS will suffer extensive steam generator damage if it remains operational. Given that the primary scram fails, there is a high probability (approximately 90% according to Ref. 7-2) that no trip signals are transmitted to the HTS. Preliminary analyses indicate that without tripping the HTS, the 2-1/4 Cr-1 Mo steam generator tubes will fail in the vicinity of the bimetallic weld within a few hundred seconds after the spurious rod bank withdrawal is initiated. Since nearly all of the steam generator tubes are expected to fail, a massive water ingress can occur.

7.3.6.4 <u>Shutdown Cooling Maintained</u>. The SCS is designed for automatic actuation subsequent to a loss of HTS cooling. Thus, regardless of whether the HTS is damaged (approximately 90% probability) or automatically tripped (approximately 10% probability), some initial interval of SCS operation is expected to occur in scenarios without primary reactor scram. Since the SCS is also unable to lower the core temperature, there is concern that continued SCS operation could expose the primary coolant boundary to excessively high temperatures that eventually result in primary coolant boundary failure.

7.3.6.5 Reactor Cavity Cooling Maintained. It is anticipated that even if the primary scram fails, continued RCCS operation will maintain reactor vessel temperatures within acceptable limits if forced helium convection is terminated.

7.3.6.6 Steam Generator Isolated. Isolating the steam generator is only important if it is thermally damaged. Since water ingress is an immediate consequence of steam generator damage, isolation is needed to limit the amount of reactivity added to the core by the moisture and to prevent possible primary coolant boundary damage by an ingress rate that exceeds the relief train capacity. Failure to isolate the steam generator is dominated by moisture monitor failure to detect the ingress.

7.3.6.7 Shutdown Cooling System Trip. Section 7.3.6.4 states that there is a high probability of an initial interval of SCS operation following an HTS outage. In event sequences without primary scram, SCS operation could engender a primary coolant boundary failure. Preliminary assessments, however, indicate that before the primary coolant boundary is exposed to high temperatures, boiling in the shutdown heat exchanger is initiated. Since the SCS is designed to trip subsequent to detecting boiling, the likelihood of an extended SCS operating period is small.

7.3.6.8 Relief Train Response. The primary coolant relief train remains closed in event sequences having a reactor trip and successful HTS, SCS, or RCCS cooling. If a primary scram is not achieved, there is a 10% conditional probability that the relief train remains closed due to acceptable primary coolant pressure conditions. This corresponds to the conditional probability of an automatic HTS trip. If the HTS is not automatically tripped, the relief setpoint is exceeded by the water ingress resulting from the steam generator damage.

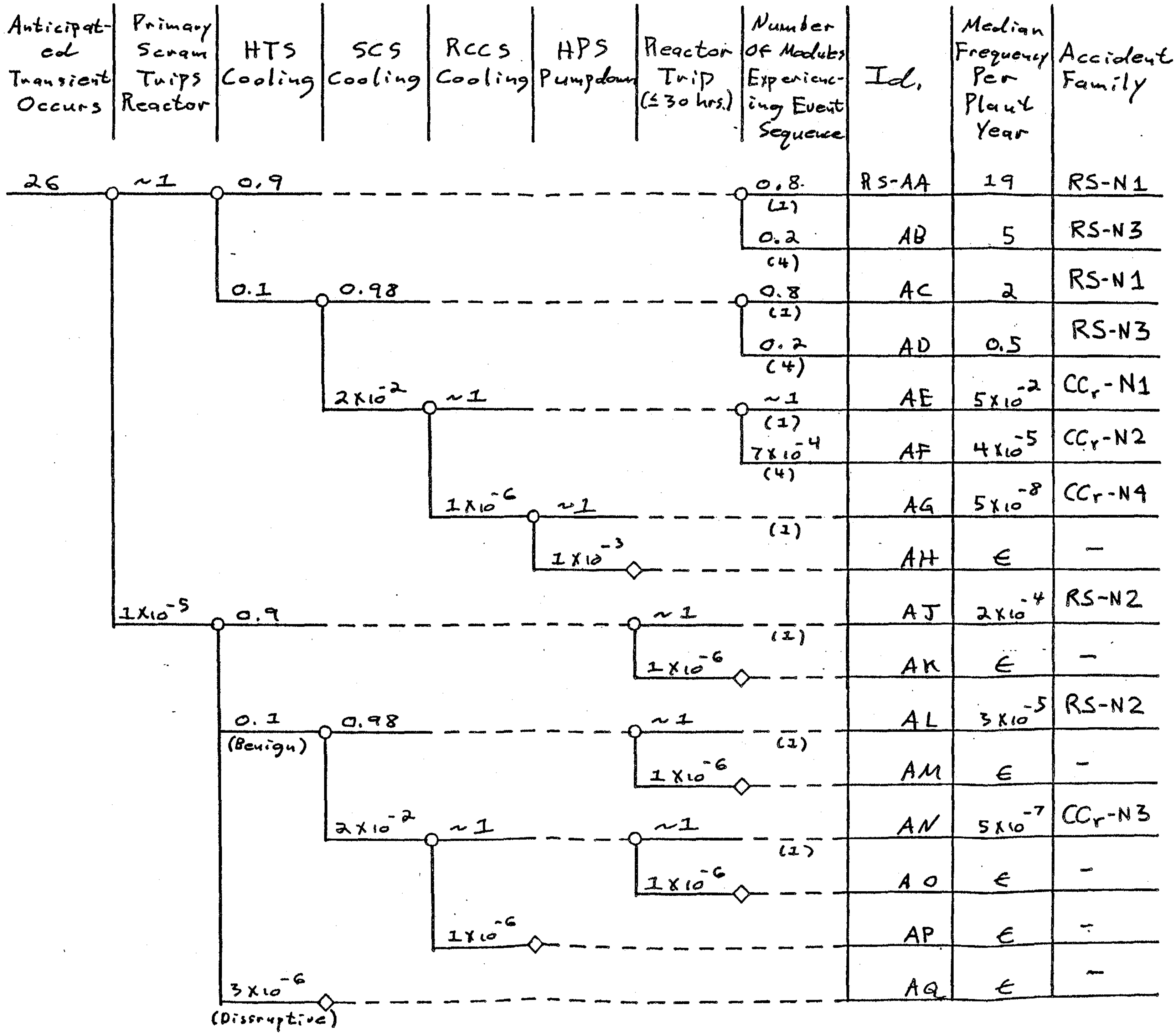
7.3.6.9 Reactor Trip Within 30 Hours. The probability of a reactor trip within 30 h following the rod bank withdrawal only needs evaluation for three Fig. 7-6 event sequences. All three sequences involve primary scram failure, continued RCCS cooling, steam generator isolation, and an SCS trip. They differ only in the primary coolant relief train response. For these event sequences, preliminary analyses indicate that if the module remains pressurized (either because the relief valve remains closed, or opens and then reseats) the thermal transient resembles a pressurized conduction cooldown to the RCCS for the first day. If the module is depressurized (because the primary coolant relief valve fails open), a thermal transient similar to a depressurized conduction cooldown to the RCCS is expected during the first day. In all three event sequences, therefore, no primary coolant boundary damage (excluding the relief valve) is envisaged.

The need to ultimately trip the reactor within approximately 30 h arises from the buildup and decay of Xe-135. After HTS cooling ceases, the core power level drops (even though the core temperature is above normal). This allows the Xe-135 inventory to increase, which concomitantly lowers the core reactivity and core temperature. The Xe-135 inventory eventually peaks and then starts to decay. After a period on the order of 30 h, the decay process will return the core reactivity and temperature to their levels reached immediately after the rod bank withdrawal. Although 30 h is not a rigorously derived number (perhaps only 20 or 25 h are available), its uncertainty has a negligible influence on the reactor trip probability. In Fig. 7-6 the probability that the reactor is not tripped before Xe-135 decay increases the core temperature above its initial maximum value is the conditional probability that both the control rods and RSS fail to trip due to faults in components located inside the reactor.

7.3.7 Anticipated Transients Without Scram

Figure 7-7 is the event tree for sequences initiated by anticipated transients that require at least one module to trip. Some of these sequences are identical to loss of HTS cooling sequences given in Fig. 7-4. Where such duplication exists, the same sequence identifications are utilized in both event trees.

Figure 7-7. ATWS Event Tree



7.3.7.1 Anticipated Transient Occurs. Estimates predicated upon Refs. 7-29 and 7-30 indicate that approximately 26 transients per plant year will cause reactor trip conditions to exist for at least one module. Approximately 20% of these transients create trip conditions in all four modules.

7.3.7.2 Primary Scram Trips Reactor. Reactor trip in the modules perturbed by the transient is the normal response to the initiating event. The failure to trip probability is from Ref. 7-2.

7.3.7.3 Heat Transport System Cooling Maintained. Continued HTS cooling is the normal response to a reactor trip. Even if the trip fails, continued HTS operation is expected. If the HTS fails subsequent to an ATWS, two failure modes can result.

1. The HTS failure mode includes an HTS circulator trip (the benign failure mode).

2. The HTS failure involves a loss of feedwater flow without a circulator trip (the disruptive failure mode).

The second failure mode is classified as disruptive because the inability to cool the helium exposes the primary coolant boundary to temperatures hot enough to engender a steam generator vessel rupture approximately 15 min following the initiating event.

7.3.7.4 Shutdown Cooling System Cooling Maintained. Subsequent to a benign loss of HTS cooling, the SCS is normally activated. However, in most cases where the HTS failure mode is disruptive, no initiation signal is transmitted to the SCS.

7.3.7.5 Reactor Cavity Cooling Maintained. Continued RCCS operation precludes offsite doses in scenarios with benign HTS failure coupled with a loss of SCS cooling. This result is believed applicable within the first day following an ATWS.

7.3.7.6 HPS Pumpdown. As is discussed in Section 7.3.2.10, an intentional primary coolant pumpdown is the normal response to a loss of HTS, SCS, and RCCS cooling.

7.3.7.7 Reactor Trip Within 30 Hours. The need to trip the reactor within approximately 30 h subsequent to an ATWS is due to the eventual increase in core temperature that follows the initial Xe-135 buildup and decay interval (see Section 7.3.6.9). Although the exact period available to trip the reactor without incurring a release from the vessel is not precisely known, the failure to trip probability is relatively insensitive to it within its anticipated range of uncertainty.

7.3.7.8 Number of Modules Experiencing Event Sequence. The probability distribution for the number of modules experiencing each event sequences is predicated upon models described in Section 7.3.4.6.

7.3.8 Earthquakes

Figure 7-8 is the event tree for the risk due to seismic activity occurring in the vicinity of the MHTGR. While seismic events capable of causing substantial plant damage are considered as part of the analysis, the assessment shows only limited radiological consequences for events as severe as ten times the SSE. Because events of this severity are very unlikely, the risk from seismic events is shown to be considerably less than that due to other events addressed in this report.

A user requirement imposed on the MHTGR is that the site characteristics envelope approximately 85% of prospective U.S. nuclear sites (Ref. 7-31). In order to comply with this requirement, a site-seismicity curve (ground acceleration as a function of frequency) that encompasses 85% of prospective U.S. sites is derived from data in Ref. 7-32.

In Ref. 7-32 a survey of the continental United States was made evaluating, for each of 21 zones, the maximum expected horizontal ground acceleration

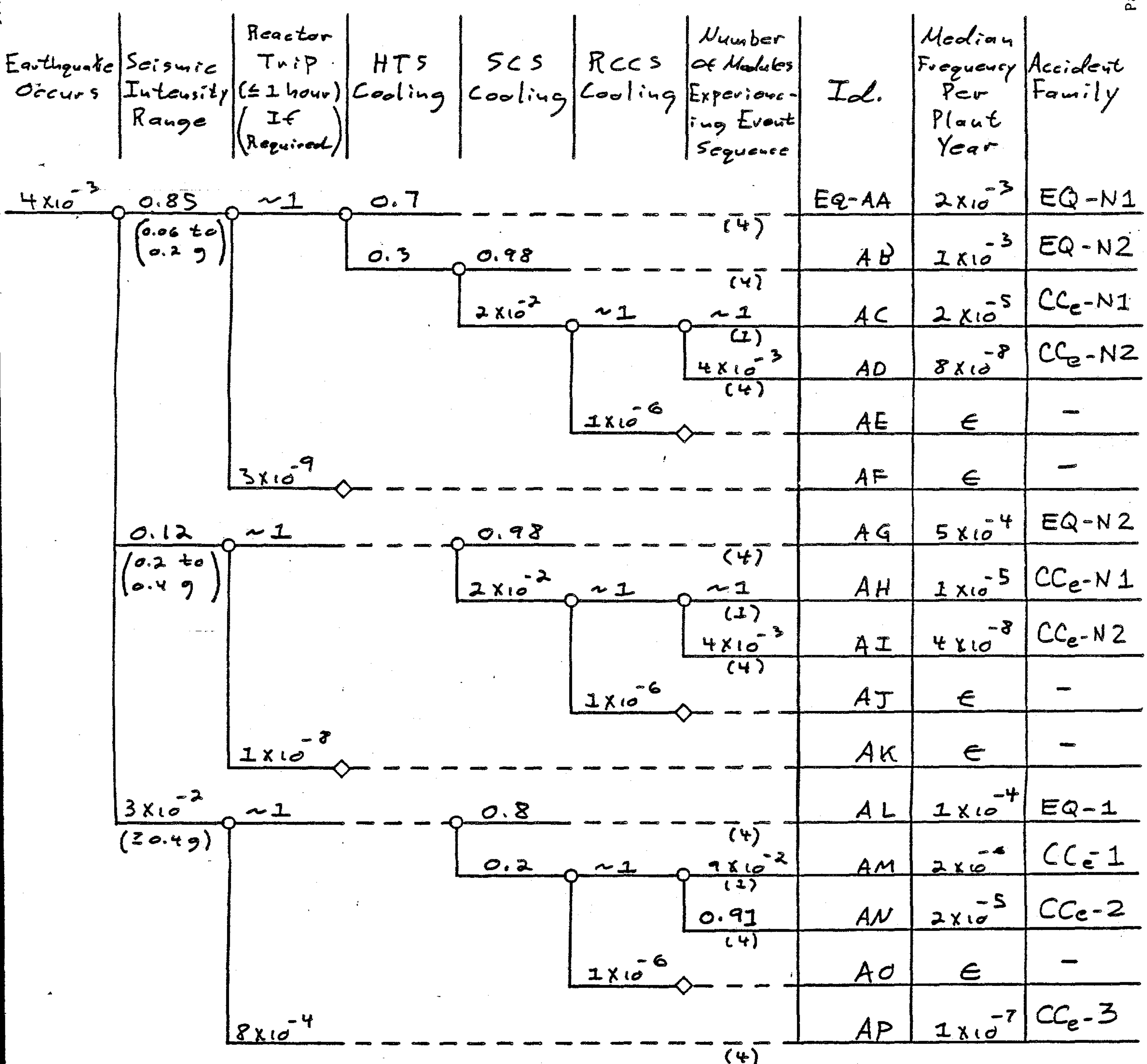
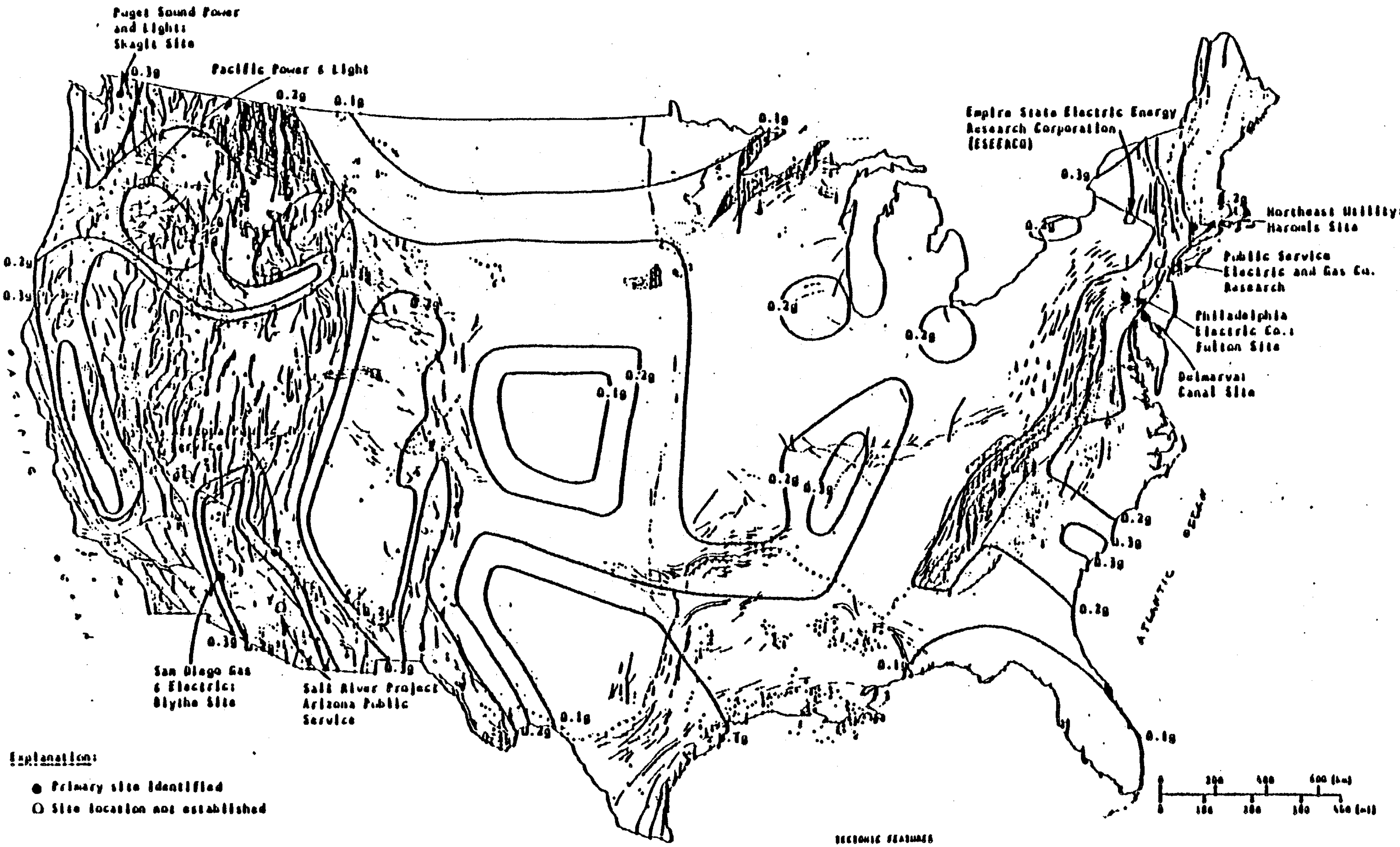


Figure 7-8. Earthquake Event Tree

for rock sites. This, in turn, can be viewed as the SSE acceleration likely to result from the licensing process. A contour map predicated upon this evaluation is shown in Fig. 7-9. Using this as a basis, it was then concluded that a plant designed for an SSE of 0.3 g would be certified at approximately 85% of the U.S. sites. Ideally, at this point one would refer to a full PRA of a plant located on a 0.3 g site and use its seismicity curve as that which would encompass 85% of the sites. As no such PRA was available, the extrapolation of available data described below was employed.

References 7-33 and 7-34 provide seismicity curves for the Zion and Seabrook nuclear power stations. These plants have SSE accelerations of 0.15 g and 0.2 g, respectively. Plots of the seismicity curves for these two plants can be seen in Fig. 7-10. Note that the two curves are similar, only displaced. That is, for any given frequency Zion is a constant factor less active than Seabrook. The assessment assumes that this constant factor is related to the difference in SSEs. It also is postulated that the "maximum expected horizontal acceleration" (SSE) corresponds to the ground acceleration predicted to occur at some low, but nonnegligible, frequency. While not exactly borne out by the data available, this assumption is reasonable; the SSEs falling between 10^{-3} and 10^{-4} per site year, and when combined with the first assumption, allows extrapolating a seismicity curve for the assumed site.

Noting from Fig. 7-10 that the Zion SSE occurs at approximately 1.3×10^{-4} and that the Seabrook SSE occurs at a frequency of 3.8×10^{-4}, the hypothetical site evaluated is assumed to have a ground acceleration corresponding to its SSE of 0.3 g at the geometric mean of these frequencies, 2.2×10^{-4}. Frequencies of 1.3×10^{-4} and 3.8×10^{-4} are assumed to be the 5th and 95th percentile bounds on uncertainty. Establishing the 0.3 g SSE at 2.2×10^{-4} and utilizing the simplifying assumption of similarly shaped seismicity curves at various sites, a seismicity curve for the evaluated HTGR site can be constructed. Such a curve is shown in Fig. 7-11 and provides the foundation for the risk evaluation.



Explanation:

● Primary site identified

○ Site location not established

NOTE: The contours shown indicate schematically the approximate level of maximum acceleration that would have to be considered in the siting of nuclear facilities. Contours for accelerations exceeding 0.3g are not shown.

TECTONIC FEATURES
(from Philip B. King, U.S. Geological Survey, [illegible])
Albers Equal Area Projection

FIGURE 1 CONTOURS OF MAXIMUM EXPECTED HORIZONTAL GROUND ACCELERATION FOR ROCK SITES

The contours and regions shown represent generalized estimates based on existing published data.

Fig. 7-9. Contours of Maximum Expected Horizontal Ground Accelerations for Rock Sites

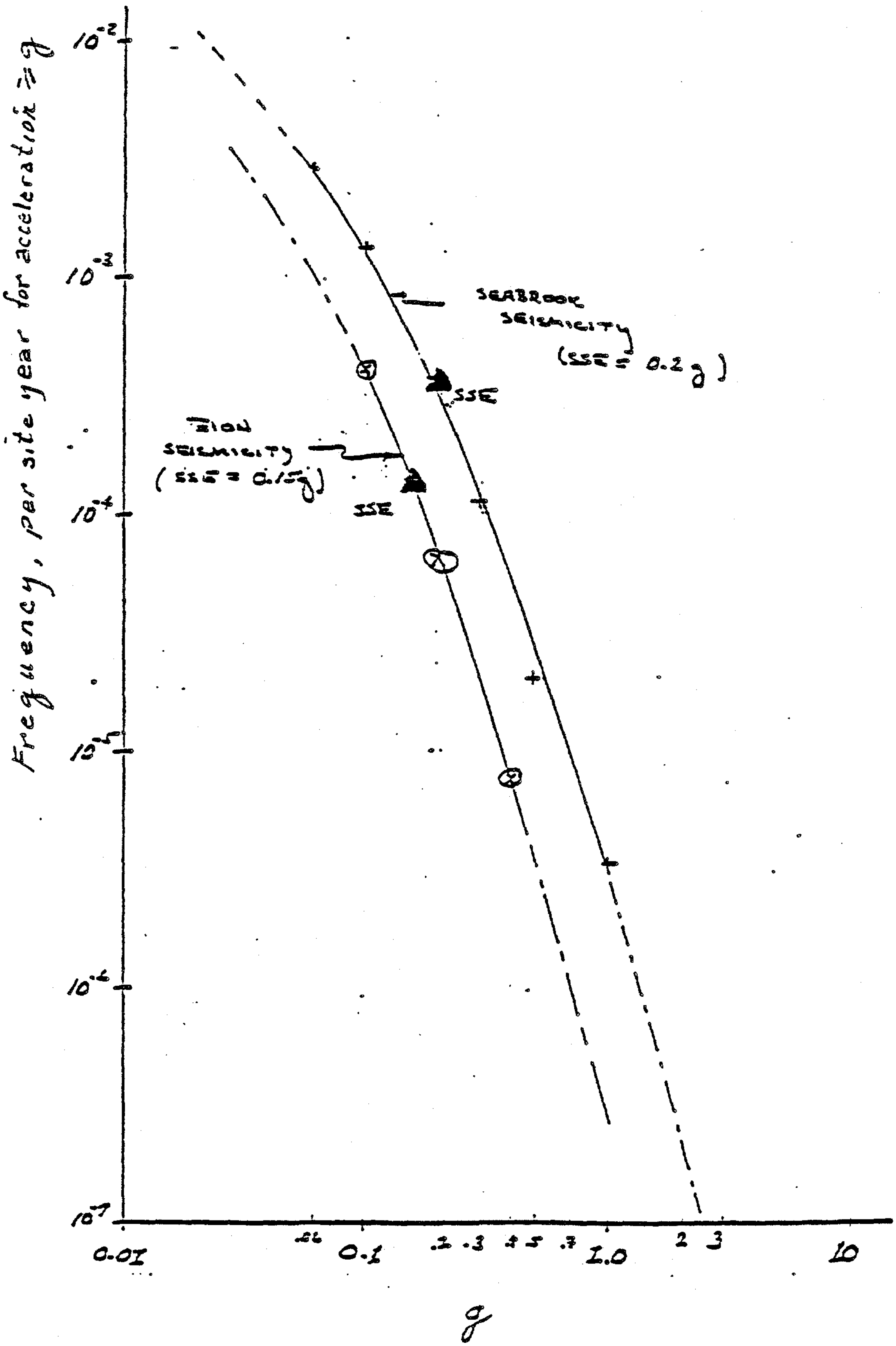


Figure 7-10. Zion and Seabrook Seismicity Curve

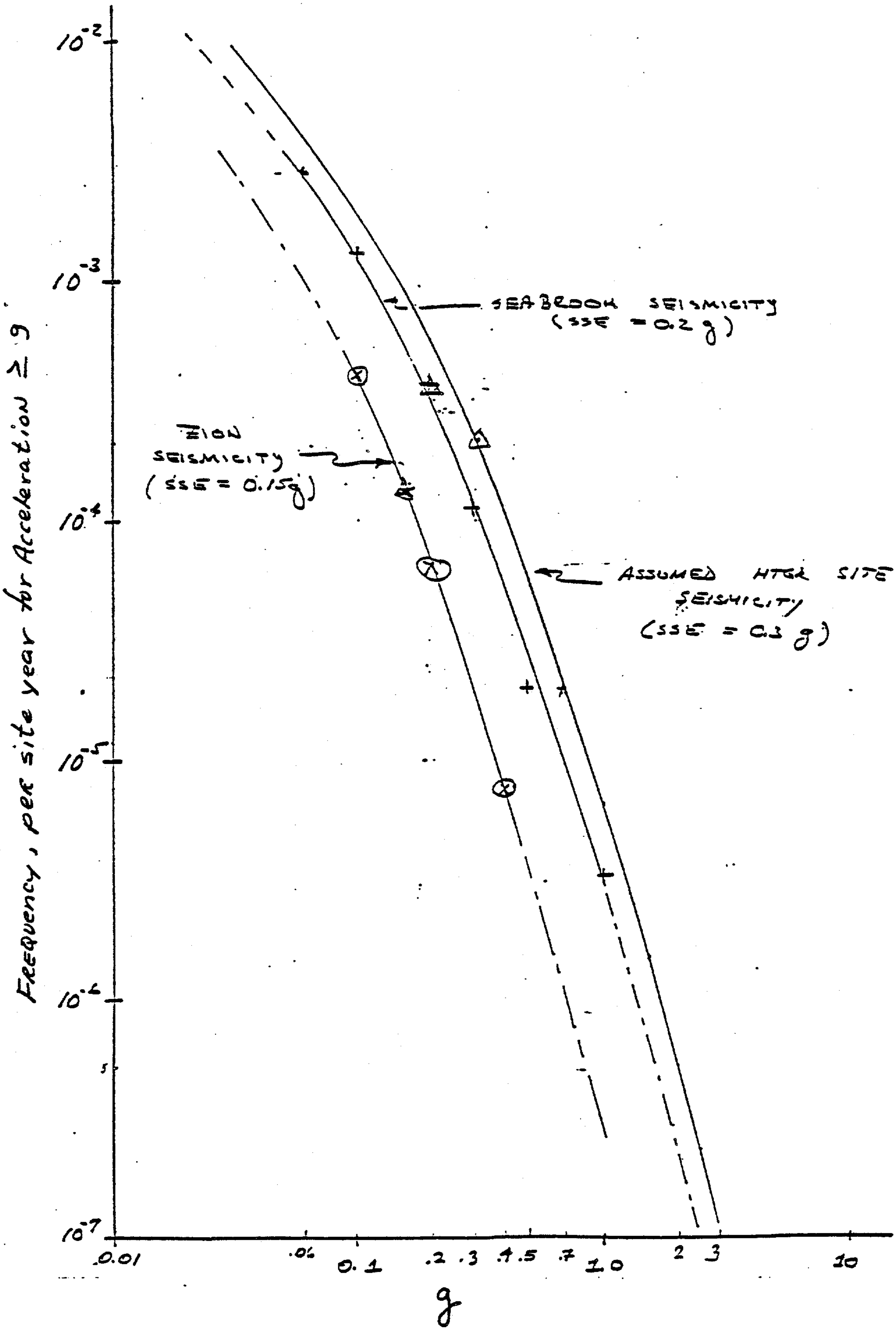


Figure 7-11. MHTGR Site Seismicity Curve

7.3.8.1 Earthquake Occurs. Seismographic data demonstrate that the ground is in constant motion. In order to differentiate between this normal seismic background and an "earthquake," a 6×10^{-2} g delimiter is introduced because no damage to typical commercial or residential structures is expected for earthquakes of 6×10^{-2} g or less intensity. From Fig. 7-11, the frequency of earthquakes having an intensity above 6×10^{-2} g is 4×10^{-3} per plant year.

7.3.8.2 Seismic Intensity Range. Three seismic intensity ranges are identified in Fig. 7-8:

1. 6×10^{-2} g to 0.2 g
2. 0.2 g to 0.4 g
3. 0.4 g or greater

These intensity ranges were selected because seismic activity contributes negligibly to plant component failure probabilities when the ground acceleration is below 0.2 g. Preliminary studies, utilizing Refs. 7-33 and 7-34, disclose that above approximately 0.2 g the HTS seismic failure probability is quite high. This basis is why HTS cooling is excluded from event sequences involving seismic intensities of 0.2 g or greater. Earthquakes are not, however, expected to impact the SCS or RCCS reliability at intensities below 0.4 g.

7.3.8.3 Heat Transport System Cooling Maintained. Below a 0.2 g seismic intensity, the normal plant response to an earthquake is to continue HTS cooling. For earthquakes less than the OBE (0.15 g), there is a reasonable probability that power production can be maintained. However, the HTS is not anticipated to remain operational if an approximate 0.2 g seismic intensity is exceeded.

7.3.8.4 Shutdown Cooling System Cooling Maintained. The standard plant response to an HTS outage is automatic SCS initiation. Less than 0.4 g seismic events only negligibly impact the SCS reliability. Above 0.4 g, the

SCS failure probability is sensitive to seismic intensity. Preliminary evaluations indicate that above 0.4 g, the SCS failure probability is approximately 20%. The contribution from earthquake-induced failures is 0.18, while the remaining contribution (2×10^{-2}) is due to other causes.

7.3.8.5 Reactor Cavity Cooling Maintained. Seismic events exceeding 0.4 g can also diminish the RCCS reliability, thus presenting the possibility of a conduction cooldown to the earth. A conditional failure probability of 10^{-4} is employed in Fig. 7-8, predicated upon engineering judgment. It is further judged that if the seismic event induces RCCS failure, the conditional probability of HPS or instrument line damage inside of the reactor building is virtually unity.

7.4 References

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8. ACCIDENT CONSEQUENCES

The consequences for each of the event sequences identified in the frequency assessment of Section 7 are evaluated in terms of fission product release to the atmosphere and resultant dose to an individual at the plant EAB. Many of the event sequences identified result in no offsite dose and will not be addressed here. They have been included in the PRA for completeness and as a basis for future licensing analyses. For those sequences that do result in an offsite release, the results are reported for whole body gamma and thyroid doses.

The accidents considered in Section 7 that result in dose consequences include fission product releases from vessel depressurization initiated by primary coolant leaks, seismic activity, and steam generator leaks; and releases from heatup of the core in conduction cooldown accidents. Conduction cooldown accidents may be initiated by primary coolant leaks, steam generator leaks, control rod bank withdrawal, or seismic activity. The consequences from primary coolant leaks are discussed in Section 8.1. The consequences resulting from a small steam generator leak accident are presented in Section 8.2. Moderate steam generator leak accident consequences are discussed in Section 8.3. Consequences of conduction cooldown accidents for all initiating events are presented in Section 8.4. Earthquake consequences are the topic of Section 8.5.

Results from the previous risk assessment for the MHTGR plant documented in Ref. 8-3 have been used as the basis for most of the consequences reported here. Pertinent design changes that have been made in the interim are considered in the current assessment. In terms of consequences, the improvement in fuel quality specifications has resulted in a reduction of the doses previously reported in Ref. 8-3.

8.1 Primary Coolant Leak Consequences

The primary coolant leak frequency assessment of Section 7.3.1 identified eight accident families as shown in Fig. 7-1 which result in an offsite dose to the public. The categories are labeled PC-3 through PC-10 where PC-3 is the largest leak identified and PC-10 the smallest engendering a plant shutdown. The consequence source term for primary coolant leaks includes circulating activity and some liftoff of material plated out on primary circuit surfaces. In all cases the reactor core is cooled by forced circulation of helium provided either by the HTS or the SCS which precludes any incremental release of radionuclides from the fuel body inventory.

Available activity is released from the breach in the primary coolant boundary into the reactor building. For smaller leak sizes, the consequences can be reduced by pumpdown of the circulating activity to storage bottles by the HPS. For larger leak sizes pumpdown becomes ineffective, and essentially 100% of the circulating activity is released into the reactor building. The fraction of material lifted off primary circuit surfaces increases for larger leaks as well because of the higher velocity helium flows. Once fission products have been released into the reactor building, they can be transported to the atmosphere through the building dampers or by building leakage if the egress rate from the vessel is small enough.

8.1.1 Data Base

The consequence data include that used in the transient models of plant thermodynamic and radiological response. These data are cited in Ref. 8-2.

8.1.2 Physical Phenomena

The primary coolant leak results in release of fission products to the reactor building and reduction in reactor vessel pressure. Upon detection of low primary coolant pressure, the reactor is tripped automatically on the outer control rods and core cooling continues on the HTS. Detection of high reactor building radiation and low primary coolant pressure initiates pumpdown of the primary coolant by the HPS.

The rate at which helium depressurizes through the leak area as a function of time is determined conservatively by assuming choked flow conditions which maximize the flow rate of helium from the reactor vessel. The amount of helium released is determined by integrating the time-dependent mass balance equation that includes the time-dependent rate of depressurization through the leak and rate of helium pumpdown. The time to depressurize the reactor vessel is shown in Fig. 8-1 as a function of leak size. For leak sizes greater than 1 in.^{2}, the time to depressurize the reactor vessel is less than 1 h and prevents the HPS from pumping any significant amount of helium to storage.

If the leak size is large enough, liftoff becomes a major source of released fission products in addition to the circulating activity. Liftoff refers to the mechanical removal of fission products plated out on reactor components and other primary circuit surfaces. The liftoff model developed for the Ref. 8-2 assessment has been used here as well. In general, the model provides an empirical approach to estimate liftoff fraction as a function of shear force ratio (ratio of shear stress during the accident condition to that at normal operating conditions).

The shear force ratio distribution in the primary coolant loop was calculated for various leak sizes and positions in the loop. The calculational method used for determining the shear force ratio distribution solves a set of ordinary differential equations and relations governing the modeled flow system. The analytical model assumes that the primary coolant system can be broken down into a series of subvolumes, or nodes, interconnected by flow paths. The transient forms of conservation of mass and energy, as well as the equation of state, are then applied to the nodes, and the transient conservation of momentum with the buoyancy term is applied to the interconnecting flow paths. Transient coolant pressure, temperature, and flow throughout the primary coolant system is calculated, taking into account the dynamic behavior of the circulators and valves, the actions of the plant protection system, and the heat transfer between the coolant, core, steam generator, shutdown cooling heat exchanger (SCHE), and reactor internals. These calculations were performed using the systems-dynamics computer code RATSAM (Ref. 8-1). The estimated shear force ratio was used to determine the distribution of the

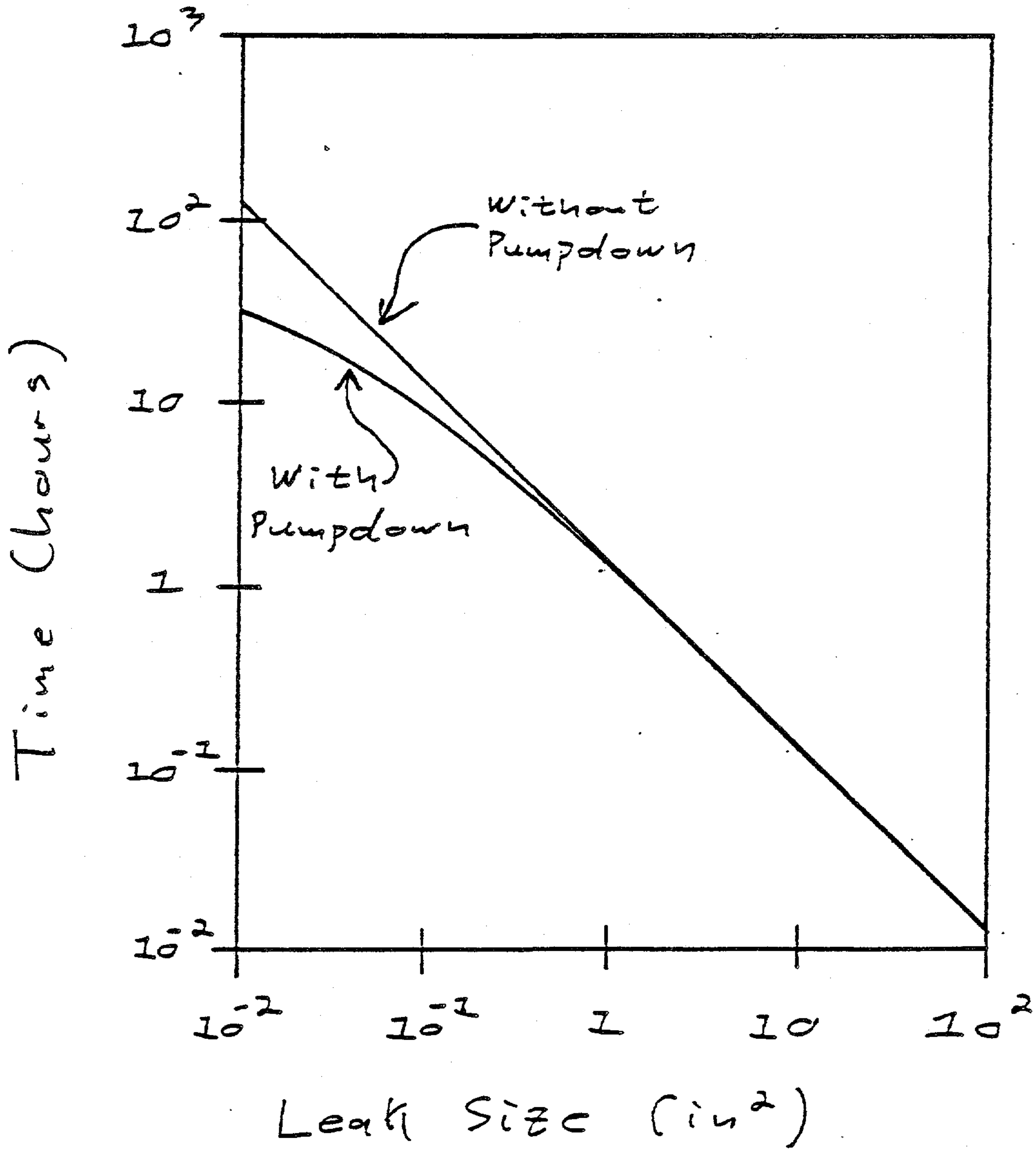


Figure 8-1. Depressurization Time with and without Pumpdown for the Modular HTGR

liftoff in the primary loop, and given the plateout distribution, the total liftoff of fission products from the primary loop surfaces into the circulating helium was estimated for a given leak size.

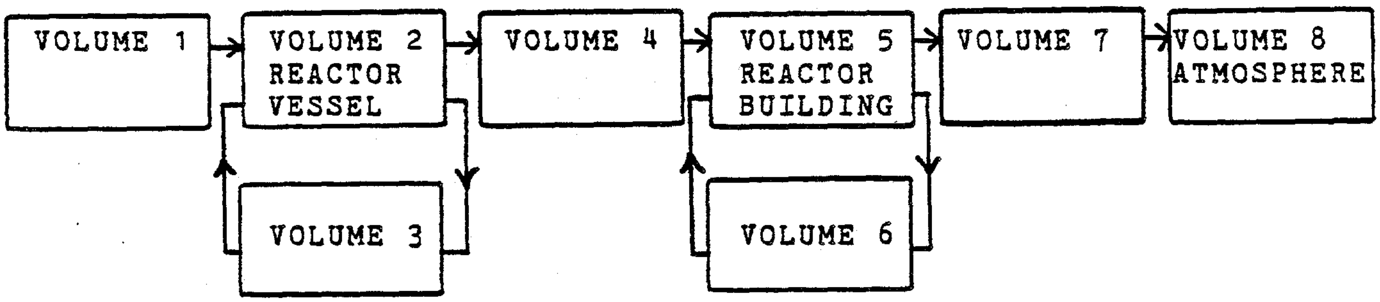
The total percent liftoff is given in Table 8-1 as a function of leak size for some of the important fission product nuclides. The subsequent transport of fission products that are lifted off will depend on the governing phenomena and could lead to retention within or loss from the reactor vessel. Conservatively, retention mechanisms in the reactor vessel are neglected, and all the lifted off fraction is assumed available for release as the vessel depressurizes. All liftoff is considered to be elemental rather than in the form of compounds.

The fission product transport in the reactor building, subsequent release to the atmosphere, and the resultant dose calculations were performed using the TDAC computer code developed at GA. The method used is based on the analytical solution of coupled linear differential equations governing the activity in different volumes representing the reactor vessel, reactor building, and the environment over time. The calculation of activity in each volume is based on the assumption of instantaneous homogeneous mixing. The calculation of radiological doses is based on the semi-infinite cloud approximation. The code allows up to 65 decay chains with up to 6 nuclides each. The TDAC model is shown in Fig. 8-2, which indicates the various volumes available and interconnecting flow paths. The release rate from the reactor vessel, retention by pumpdown of helium, attenuation of fission products due to plateout and settling in the reactor building, and release through the building dampers are represented in the TDAC model.

The building dampers will remain closed if the egress rate from the reactor vessel is lower than the building leakage rate. When the helium leakage rate from the vessel is larger than the reactor building leak rate, then the dampers open to relieve the excessive building pressure allowing fission products to escape to the atmosphere. After the pressure transient is complete and the dampers reclose, the remaining reactor building radionuclide inventory is released by normal building leakage.

TABLE 8-1
% LIFTOFF AS A FUNCTION OF LEAK SIZE

	% Liftoff From the Primary Loop		
Leak Size (in.^{2})	I-131	Sr-90	Cs-134
1	4.5-3	0.095	5.0-4
10	0.060	0.12	8.5-3
30	0.21	0.42	0.035
100	0.73	1.5	0.13



VOLUME 1 - Pseudo Volume to Simulate Time Dependent Fission Product Release from Core

VOLUME 2 - Reactor Vessel

VOLUME 3 - Not Used

VOLUME 4 - Simulates Removal by Helium Purification System

VOLUME 5 - Reactor Building Structure

VOLUME 6 - Simulates Removal by Plateout and Settling in Reactor Building

VOLUME 7 - Simulates Removal by Reactor Building Filters, if Available

VOLUME 8 - Atmosphere

Figure 8-2. TDAC Model

Table 8-2 shows the reactor building parameters and site data assumed for the MHTGR. As shown in Table 8-2, credit is taken for the physical processes of plateout and particulate settling in the reactor building. The uncertainty distribution for meteorology is discussed in Ref. 8-4.

The weather dispersion parameter,

$$\chi/Q = \frac{1}{\pi \Sigma_z \Sigma_y u}$$

where $\Sigma_z = \sigma_y^2 + CA/\pi$

$\Sigma_y = \sigma_y^2 + CA/\pi$

u = wind velocity

A = area of building

C = 0.585

σ_z = deviation in z direction, and

σ_y = deviation in y direction.

The χ/Q assessment included the probability of being in six different weather stability classes, as well as four different wind speeds, and the probability of being in any one of ten wind directions to account for the building wake factor. The values of σ_y and σ_z were taken from Regulatory Guides 1.145 and 1.111, respectively. The probability distribution was taken from Ref. 8-5. The χ/Q distribution is shown in Fig. 8-3 for the EAB distance of 425 m. χ/Q was not varied as a function of time in this assessment.

The frequency assessment for primary coolant leaks assigns an accident family designation for a given range of leak sizes. For the purposes of consequence assessment, a representative size is selected for each range identified in the frequency assessment. Proceeding from the smallest to the largest leak sizes, the following paragraphs describe for each accident family the dominant event sequence, radionuclide release mechanism, and family consequences.

TABLE 8-2
REACTOR BUILDING PARAMETERS AND SITE DATA

A. Reactor Building Parameters

	Medians	Uncertainty Factor
Volume:	183,738 ft^3	
Settling:	$9.8\ h^{-1}$ $t \leq 3$ h, $1.13\ h^{-1}$ $T > 3$ h	10*
Plateout:	$8.7\ h^{-1}$	10
Dampers:	Open when $\dot{m}_{in} > \dot{m}_{out}$	
Leak rate:	1/day	

B. Dose Parameters

EAB distance: 425 m

Breathing rate, median, m^3/s 2.32×10^{-4}

χ/Q See Fig. 8-3.

*Uncertainty factor = ratio of 95th percentile to median; assumed distribution is lognormal.

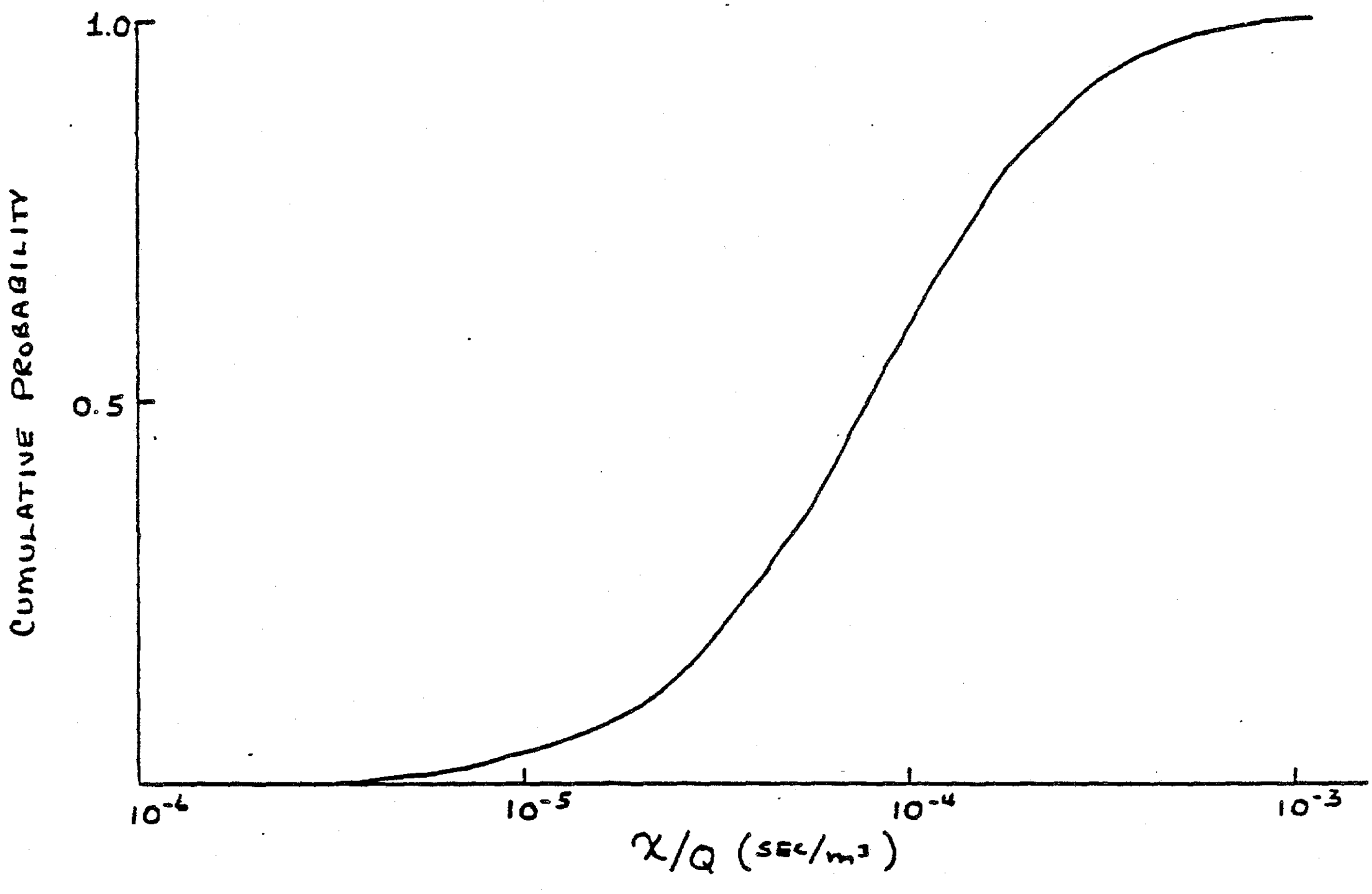


Figure 8-3. χ/Q Probability Distribution Curve

Accident families PC-10 and PC-9 describe leaks in the range of 3×10^{-5} to 0.03 in.^{2}. For purposes of consequence assessment, a leak size of 0.01 in.^{2} has been selected as typical of leaks in this range. In the dominant sequence for both families, core cooling on the HTS continues. In other sequences, the HTS may fail and the SCS provides forced core cooling. Both systems, however, perform the same function. In both families the reactor building responds as planned by isolating the filtration system used during normal operating conditions and turning off the reactor building HVAC fans. The difference in the accident families is that in PC-10 pumpdown by the HPS is successful, whereas in PC-9 it fails. The doses of PC-10 are therefore less than those of PC-9 because of the retention of some primary coolant by the HPS. Liftoff of plated-out material for these accident families is negligible because of the small leak size. Consequences of PC-10 and PC-9 are based on Ref. 8-3 results. Offsite doses have been reduced to account for improvement in fuel quality specifications. This design improvement affects the circulating activity in addition to the plated-out material on primary circuit surfaces.

Accident families PC-8 and PC-7 describe leaks in the range of 2×10^{-3} to 0.03 in.^{2}. Similar to PC-10 and PC-9 described above, a typical leak size of 0.01 in.^{2} has been selected for this leak size range. The dominant event sequences again include continued operation of the HTS following the detection of the leak and subsequent reactor trip on outer control rods. The event sequence describing PC-8 involves successful pumpdown of primary coolant, but failure to isolate the reactor building filters and disengage the building HVAC fans. The fission products are therefore released into the environment via the building dampers at a rate in excess of what normally would be seen. The effects of reactor building holdup as well as retention of halogens and particulates due to plateout and settling in the building are not available in this case to reduce radionuclide release to the atmosphere. Accident family PC-7 is similar to PC-8, except pumpdown of primary coolant fails and more fission products are released to the reactor building. Subsequent failures in the reactor building are the same as for category PC-8. For family PC-7 essentially all of the primary coolant circulating activity is released into the reactor building. In PC-8, however, a significant fraction of the available circulating activity is retained by the HPS. Liftoff of plated-out material is not important for these accident families because of the small leak size. Consequences for accident families PC-8 and PC-7 are based on families PC-10 and PC-9, respectively. Whole body gamma doses are similar because of the negligible effect of plateout, settling and building holdup on noble gases which typically are the dominant whole body dose contributors. Thyroid doses, however, are increased by approximately a factor of 20 because of the inability to retain iodines which are the dominant thyroid dose contributors.

Accident families PC-6 and PC-5 describe leaks in the range of 0.03 to 1 $in.^2$. A representative size of 1 $in.^2$ was selected for analysis in this size range. For both accident families, the dominant event sequence begins by a reactor trip on the outer control rods following detection of low primary coolant pressure. Forced core cooling is maintained using the HTS. In either accident family, pumpdown of the primary coolant is ineffective as depicted in Fig. 8-1 for a leak size of 1 $in.^2$. For accident family PC-6, the reactor building responds as planned. In family PC-5, however, the building filters are not isolated, and the HVAC fans continue to run forcing fission products out the building dampers. The consequence source term for these two families consists of the circulating primary coolant and some liftoff of plated-out material. Consequences for PC-6 have been derived from the Ref. 8-3 assessment and modified to account for the improved fuel quality specifications. PC-5 consequences are based on Ref. 8-3 results as well with the appropriate reduction in offsite dose to account for the fuel quality improvement. Whole body gamma doses are the same for both families because noble gas retention in the reactor building is negligible. The thyroid doses, however, vary by a factor of 20 because of the significant retention of iodine the reactor building provides.

Accident family PC-4 represents a leak in the size range of 1 to 13 $in.^2$. A representative size of 10 $in.^2$ has been selected for analysis. Following detection of the leak, the reactor is tripped on outer control rods and the core continues to be cooled on the HTS. Neither HPS pumpdown of primary coolant or attenuation in the reactor building can mitigate the consequences for this accident family because of the large leak size. All of the primary coolant circulating activity is released as well as a significant fraction of plated-out material. The release into the reactor building is essentially released immediately to the atmosphere through the building dampers. Consequences for this category are based on Ref. 8-3 results reduced to account for improved fuel quality.

The final accident family considered here is designated PC-3 and represents any leak in excess of 13 in.^{2}. The representative size is taken to be 30 in.^{2}. System response is the same as that described for PC-4 above. The differences in these two families is in the rate the reactor vessel depressurizes and the fraction of material lifted off primary circuit surfaces. Consequences for PC-3 are taken from Ref. 8-3 and reduced to account for fuel quality improvement.

To summarize the relative importance of liftoff for the various accident families discussed in the preceding paragraphs, Table 8-3 shows the fractional release contributions of circulating activity and liftoff of plated-out activity to the atmosphere for some of the important isotopes. Sizes smaller than 1 in.^{2} are not listed because the release for those leak sizes is essentially 100% circulating activity.

To summarize dose as a function of leak size, Table 8-4 gives whole body gamma and thyroid doses at the plant EAB for the representative sizes selected. For each accident family, the HPS and reactor building function as designed. Other accident family consequences not given in the table are summarized in Section 9. Figures 8-4 and 8-5 show the median thyroid and whole body gamma dose, respectively, as a function of primary coolant leak size. Also shown in the figures are the 95th percentile and 5th percentile dose estimates. The thyroid dose is sensitive to leak size and increases with higher liftoff of halogens at larger leak sizes. The whole body gamma dose is comparatively insensitive to liftoff as it is determined essentially by the noble gas fission products in the circulating activity.

TABLE 8-3
FRACTIONAL CONTRIBUTION FROM CIRCULATING AND
LIFTOFF TO THE ACTIVITIES RELEASED TO THE ATMOSPHERE

	% Contribution to Atmospheric Release					
	Circulating			Liftoff		
Leak Size (in.^{2})	I-131	Sr-90	Cs-134	I-131	Sr-90	Cs-134
1	94.8	0.	3.9	5.2	100.0	96.1
10	58.7	0.	1.2	41.3	100.0	98.8
30	28.8	0.	0.	71.2	100.0	100.0
100	11.0	0.	0.	89.0	100.0	100.0

TABLE 8-4
PRIMARY COOLANT LEAK DOSES AT THE EAB FOR VARYING LEAK SIZES

		Median Dose			
		WBG		Thy	
Accident Family	Leak Size (in.^{2})	Rem	f*	Rem	f
PC-10	0.01	3.8-5	10	1.9-6	10
PC-6	1	5.0-4	10	8.2-4	10
PC-4	10	5.8-4	10	1.5-3	10
PC-3	30	6.2-4	10	2.8-3	10

*f = Ratio of 95th percentile to median; assumed distribution is lognormal.

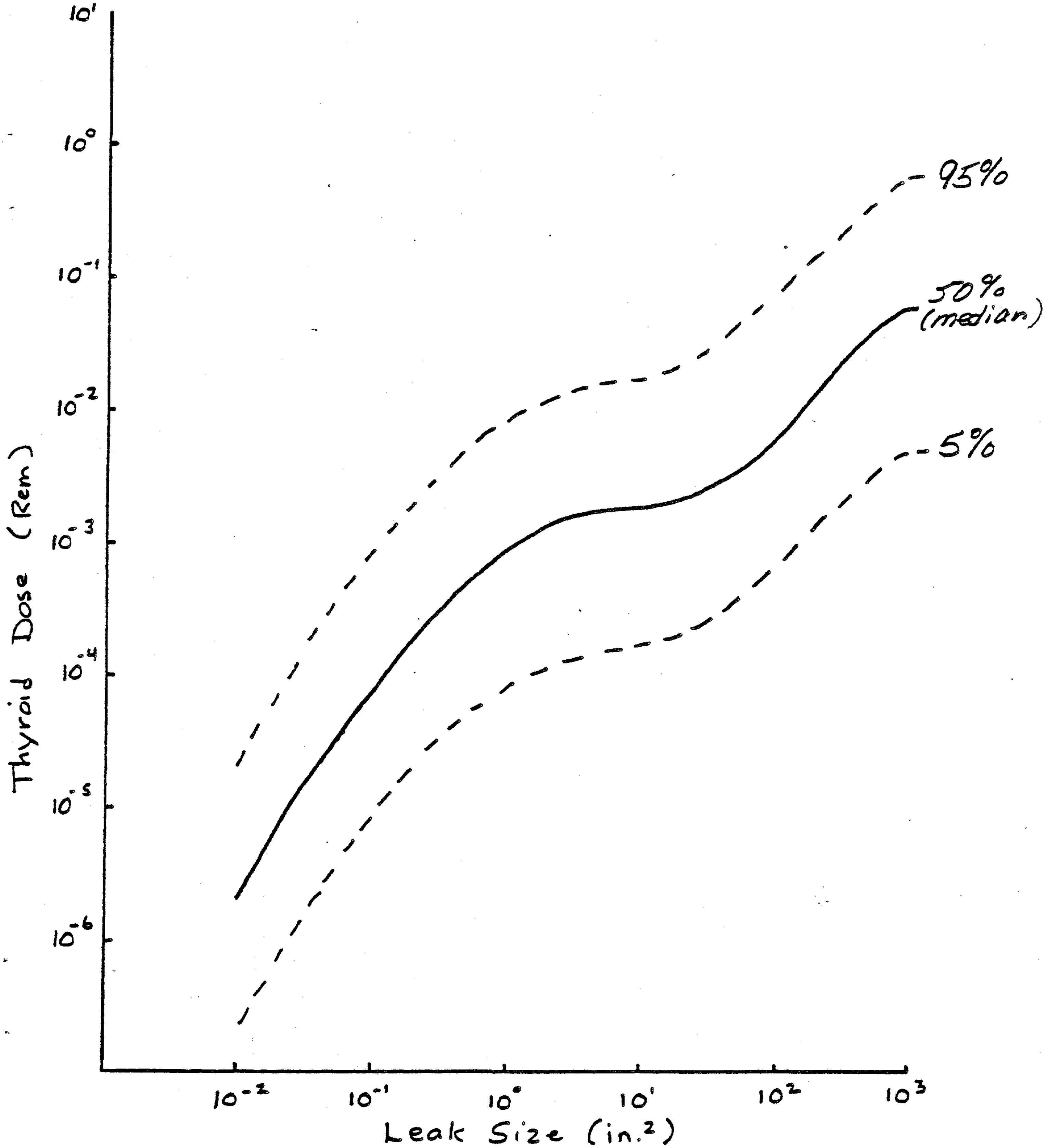


Fig. 8-4. Offsite Thyroid Dose as a Function of Primary Coolant Leak Size (Intentional Pumpdown Functions)

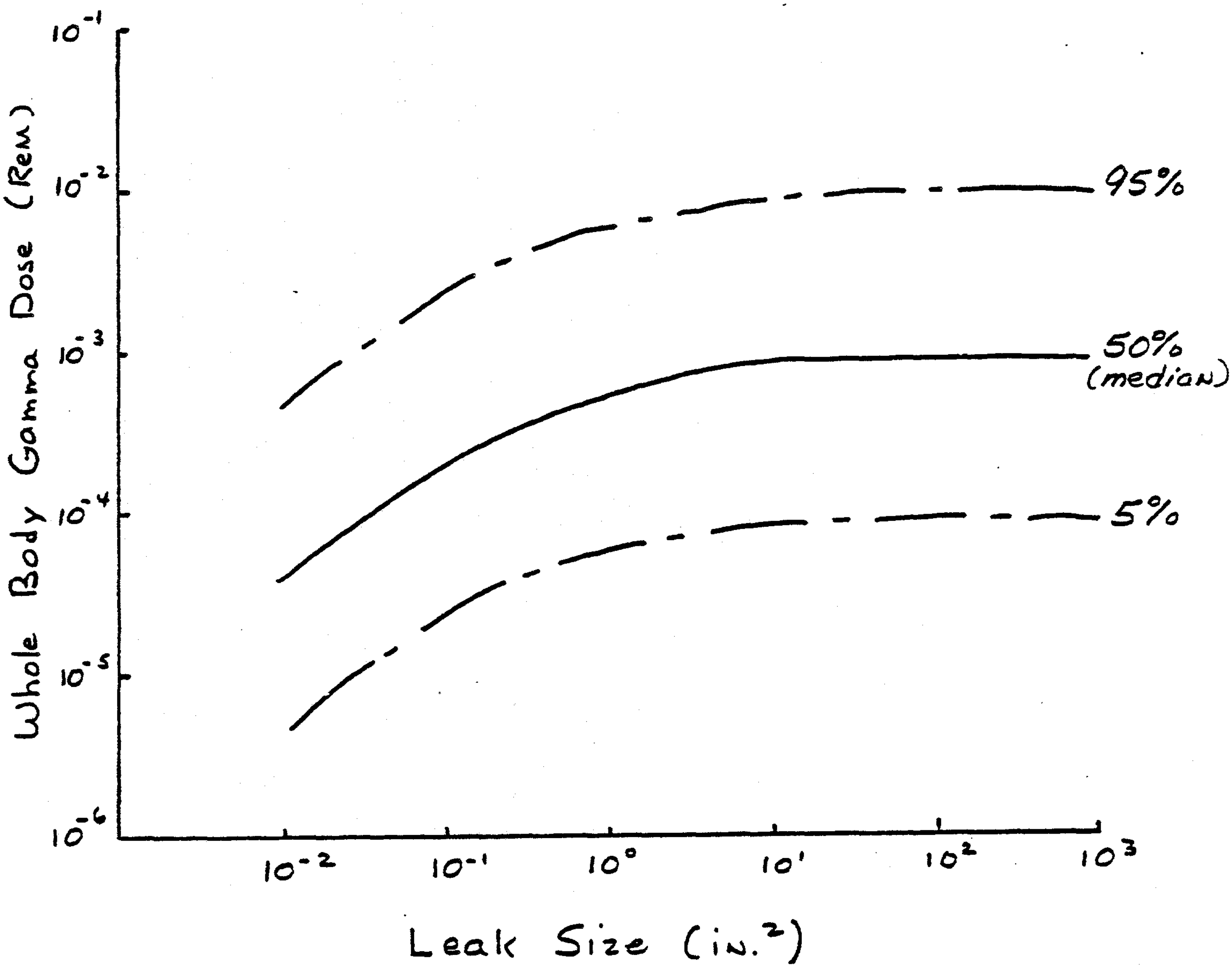


Figure 8-5. Offsite Whole Body Gamma Dose as a Function of Primary Coolant Leak Size (Intentional Pumpdown Functions)

8.1.3 Uncertainty Analysis

A method for assessing the uncertainties in consequence prediction was developed by GA during the AIPA safety assessment documented in Ref. 8-5. The method uses a simplified mathematical algorithm describing the consequence controlling phenomena. The algorithms are used in a Monte Carlo error propagation program to simulate many safety risk consequence assessments. Cumulative probability distributions of independent variables are specified as input to the program. The algorithms used for the primary coolant leak dose consequences for the MHTGR are identical to those used in the Ref. 8-2 assessment. The Ref. 8-2 uncertainty distributions were modified by replacing the median consequences calculated for the Ref. 8-2 assessment by the current consequences estimated here in Section 8.1.2.

8.2 Small Steam Generator Leak Consequences

A small steam generator leak with a subsequent primary coolant boundary failure releases fission products that result in a dose to the public. A number of event sequences as pictured in Fig. 7-2, however, result in accident families that do not have an offsite dose as primary coolant pressure boundary integrity is maintained. These accident families will not be addressed here as they have no bearing on the safety risk of the MHTGR. They are included in the risk assessment in order to provide a basis for licensing work. Accident families that exhibit doses have release paths that are either directly to the environment through the steam generator dump system or the steam generator relief valve or that vent to the reactor building through the primary relief valves before reaching the environment. The consequence source term may consist of circulating activity, radionuclides washed off wetted primary circuit surfaces, release from hydrolyzed fuel, release from oxidized graphite, or any liftoff of plated-out activity.

8.2.1 Data Base

The thermodynamic and fluid dynamic data used are those typically used to analyze the plant and reactor core response to moisture ingress. Pertinent details are cited in Ref. 8-2.

8.2.2 Physical Phenomena

The frequency assessment in Section 7.3.2 for small steam generator leaks covers a spectrum of leak sizes ranging from pinhole to approximately 8×10^{-3} in.^{2}. The maximum size considered for small steam generator leaks corresponds to a flow rate of 0.1 lbm/s which will be used in the consequence assessment for all small leaks.

The planned response to a moisture ingress event begins with the detection of moisture at the 1000 ppm level by the moisture monitors. For a small leak, this level is not reached for approximately 5 min. The moisture sampling process takes another 20 s following which the PPIS is signaled. The PPIS initiates a reactor trip on the outer control rods and steam generator isolation valve closure. Following the signal to isolate, the main circulator is tripped, and the SCS is started and cools the reactor core by forced helium circulation. Following isolation, the steam generator dump system valves are opened and the steam generator inventory released into the dump system tanks. Just prior to releasing primary coolant through the dump system, the valves are reclosed. The increase in system pressure resulting from the ingress of moisture is not large enough to lift the primary relief valves. There is no fission product release as the primary coolant boundary remains intact. This sequence of events is observed to correspond to sequence SS-AA in Fig. 7-2.

For fission product release to occur, additional failures are required that result in failure of the primary coolant boundary to contain the fission products. As shown in Fig. 7-2, failures in addition to the small steam generator leak may result in a number of sequences that result in fission product release. Failure of the steam generator dump system or failure to isolate precedes each event sequence where an offsite dose occurs. Each of these sequences is labeled with an accident family designation. Those under consideration in this section are S/G-4, S/G-5, S/G-7, S/G-12, and S/G-13. The following paragraphs describe for each accident family the dominant event sequence, fission product release path, contributors to the source term and the basis for assessment of the family consequences.

Accident family S/G-4 results in fission product release to the atmosphere through the steam generator dump system. Response to the moisture inleakage proceeds as planned until following steam generator dump; the dump valves are signaled to reclose and they do not. It was assumed in this assessment that the steam generator dump system tanks are not designed for primary coolant pressure retention. Therefore, primary coolant depressurizes slowly through the open dump system to the atmosphere via the dump tank relief valves. Core cooling is provided by the SCS. Fission product release to the atmosphere consists of primary coolant circulating activity and activity released due to hydrolysis of initially failed fuel particles. Consequences for this accident family are based on the results of Ref. 8-3. The consequences were reduced to account for improvement in fuel quality and a smaller ingress rate. Improvement in the fuel quality has resulted in lower levels of primary circuit activity as well as a reduction in the fraction of initially failed fuel important in hydrolysis.

Accident family S/G-5 results in fission product release to the reactor building and subsequently to the atmosphere. Following the initiating event of a small leak in the steam generator, the moisture monitors fail to detect the moisture. The plant continues to operate, being cooled on the HTS. Because of continued circulator operation, all moisture entering the system is transported to the hot reactor core. If the moisture is in the form of superheated steam, it remains as such; and if it is in the form of saturated steam or subcooled liquid, it will be evaporated upon contact with hot surfaces. Reactor trip occurs on high primary pressure between 3.5 and 6.0 h into the transient. The operator subsequently responds by isolating and dumping the steam generator and initiating a main loop trip. The main circulator coasts down and the SCS is started to provide core cooling by forced helium circulation. The moisture that has entered the primary system as a result of failure to initially isolate the steam generator results in lifting of the primary relief valve. The relief valve fails to reclose as designed following pressure relief, and the primary coolant is rapidly depressurized into the reactor building. Fission product release to the atmosphere includes primary coolant circulating activity, hydrolysis products, and some liftoff of material plated out on primary circuit surfaces. This liftoff is due to the high-velocity helium flows and shear force ratios in excess of unity experienced due to the rapid depressurization through the relief train. Consequences for S/G-5 are based on Ref. 8-2 results for failure to isolate the steam outlet line following a moisture ingress. These results have been reduced to account for the lower ingress rate and improved fuel quality as noted earlier.

Accident family S/G-12 is identical to family S/G-5 described above with the exception that the primary relief train responds as planned and recloses following relief of excess pressure. Primary coolant activity is released during the relief valve cycle into the reactor building and subsequently to the atmosphere. Contributors to the radionuclide source term are circulating activity, hydrolysis products, and some liftoff of plated-out material. The relief valve will lift in this scenario at approximately 7 h into the transient. The probability that the valve will lift a second time in the event sequence is less than 10^{-9} per plant year and has been truncated from the event tree of Fig. 7-2. Cycling of the relief valve will release approximately 15% of the mass present in the system to the reactor building. The consequences of S/G-12 are therefore estimated to be 15% of S/G-5 consequences.

Accident family S/G-7 results in release of fission products through the steam generator relief train directly to the atmosphere. Following detection of the leak by the moisture monitors, the PPIS signals the isolation valves to close. Steam line valves close, but the feedwater valves do not. The steam generator relief train normally is exposed to steam pressure of 2515 psia but is exposed to feedwater pressure of 3000 psia in this accident scenario. The steam generator relief train opens to relieve the excessive feedwater pressure but subsequently fails to reclose as designed. Core cooling is provided by the SCS. The consequence source term consists of circulating activity and hydrolysis products. Offsite doses are estimated to be similar to those for S/G-5 described earlier. Although S/G-5 results in a depressurization to the reactor building before reaching the atmosphere, the amount of moisture available for fuel hydrolysis is similar. It is estimated that fission product attenuation in the reactor building will resemble the attenuation provided by depressurization through the 8×10^{-3} in.^{2} leak area in the steam generator before fission products are released into the atmosphere.

Accident family S/G-13 results in release of fission products through the primary relief train to the reactor building and subsequently to the atmosphere. The scenario is similar to S/G-7 except the steam generator relief train fails to open. This results in a massive overpressure of the steam generator by the incoming feedwater. Multiple steam generator tube ruptures ensue causing a large ingress rate of approximately 300 lbm/s. The corresponding leak area for such a large rate is on the order of 24 in.^{2}. As moisture enters the primary circuit, both primary relief train valves will open, affording an egress area from the reactor vessel of 26 in.^{2}. It is postulated that since the relief train can accommodate the incoming moisture, catastrophic vessel failure can be avoided. Consequences for S/G-13 are based on those for S/G-5 described earlier. The hydrolysis fraction has been reduced because of less moisture being transported to the core in the case of S/G-13.

For all accident families considered, meteorological conditions and reactor building parameters are as given in Table 8-2. For those event sequences that result in releases through the steam generator secondary side, no attenuation is assumed with the exception of removal of particulates and halogens by water in the dump system tanks. A summary of offsite dose consequences for each family considered is given in Section 9 on risk assessment results.

8.2.3 Uncertainty Analysis

The consequence uncertainty model used for small steam generator leak initiated events is the same as the model given in Section 8.1.3 for primary coolant leak initiated events. The uncertainty distributions for steam generator leaks as given in Ref. 8-2 were modified by replacing the median consequences for the Ref. 8-2 assessment by the current assessment consequences. For those events that were not previously analyzed in Ref. 8-2, uncertainty distributions for similar events that were analyzed in Ref. 8-2 were applied.

8.3 Moderate Steam Generator Leak Consequences

A moderate steam generator leak with a subsequent primary coolant boundary failure releases fission products that result in a dose to the public. The release path can be directly to the atmosphere through the steam generator secondary side or to the reactor building if the reactor vessel relief valves lift. Releases through the secondary side can be either through an open steam generator dump system or through the steam generator relief valve. The consequence source term may consist of circulating activity, radionuclides washed off wetted primary circuit surfaces, release from hydrolyzed fuel, release from oxidized graphite, or any liftoff of plated-out activity in the event the reactor vessel relief valves lift.

8.3.1 Data Base

The thermodynamic and fluid dynamic data used are those typically used to analyze the plant and reactor core response to moisture ingress. Pertinent details are cited in Ref. 8-2.

8.3.2 Physical Phenomena

The frequency assessment in Section 7.3.3 for moderate steam generator leaks covers a spectrum of leak sizes. The flow rates may range from 0.1 to

12.5 lbm/s with the latter corresponding to a flow rate equivalent to a single tube offset rupture. The consequence assessment for moderate steam generator leaks has been based realistically on an average leak rate of 2.6 lbm/s.

The planned response to a moisture ingress event begins with the detection of moisture at the 1000 ppm level by the moisture monitors. For a moderate steam generator leak this moisture level is reached in about 10 s. The moisture monitor sampling process takes approximately 20 s following which the PPIS is signaled. The PPIS initiates a reactor trip on the outer control rods and steam generator isolation valve closure. Following the signal to isolate, the main circulator is tripped, and the SCS is subsequently used to cool the core by forced circulation of helium. Following isolation, the steam generator dump system valves are signaled to open, releasing the steam generator inventory to the dump system tanks. Just prior to releasing primary coolant through the dump system, the valves are reclosed. The increase in system pressure resulting from the ingress of moisture is not large enough to lift the reactor vessel relief valves. There is no fission product release as the primary coolant boundary remains intact. This sequence of events is observed to correspond to sequence MS-AA in Fig. 7-3.

For fission product release to occur, additional failures are required that result in failure of the primary coolant boundary to contain the fission products. As shown in Fig. 7-3, failures in addition to the steam generator leak may result in a number of sequences that result in fission product release. Failure of the steam generator dump system or failure to isolate precedes each event sequence where an offsite dose occurs. Sequences where fission product release occurs have been labeled with an accident family designation. Those under consideration in this section are labeled S/G-1 to S/G-3, S/G-6, and S/G-8 to S/G-11. The following paragraphs describe for each accident family the dominant event sequence, fission product release path, contributors to the source term, and the basis for the consequence assessment.

Accident family S/G-8 results in fission product release to the atmosphere through the steam generator dump system. Response to the moisture inleakage proceeds as planned, with the exception that after dumping the steam generator inventory the dump system valves fail to reclose. It has been assumed in this assessment that the dump system tanks are not designed to contain primary coolant pressure. The primary coolant therefore depressurizes through the steam generator leak into the dump system, through the tank relief valves, and into the atmosphere. Core cooling is provided by the SCS. Fission product release to the atmosphere consists of primary coolant circulating activity and activity released due to hydrolysis of initially failed fuel. The dose assessment for this accident family was based upon the results of Ref. 8-3. The consequences were reduced to account for an improvement in fuel quality and a realistic, instead of bounding, moisture ingress rate. In addition to lower levels of circulating activity, the fraction of initially failed fuel particles has been substantially decreased in the design improvement. Since hydrolysis only occurs in these initially failed particles, the activity released due to hydrolysis is directly proportional to the reduction in the failure fraction.

Accident family S/G-2 results in fission product release to the reactor building and subsequently to the atmosphere. The sequence of events following the leak proceeds as planned until the dump system is called upon to open. The dump system valves fail to open which presents a potential ingress into the primary system of the portion of the steam generator inventory above the leak area. Because of the location of the dump valves, it has been assumed that repair cannot take place in time to prevent the continued ingress of moisture. The system pressure is increased in the primary system to the extent that the relief valve lifts. In S/G-2 the relief valve fails to reclose as designed, thereby rapidly depressurizing the primary circuit inventory to the reactor building. Core cooling is provided by the SCS. The source term in S/G-2 includes primary coolant circulating activity, release due to failed fuel hydrolysis, and some liftoff of material plated out on primary circuit surfaces. The liftoff term becomes a contributor in those accident sequences where the reactor vessel relief valve lifts and shear force ratios within the primary circuit exceed unity because of high velocity helium flows. Radiological consequences have been taken from the Ref. 8-3 analysis and reduced to account for improved fuel quality and a realistic leakage rate as noted above.

Accident family S/G-1 results in fission product release to the reactor building and subsequently to the atmosphere. Following detection of excessive moisture in the primary system, the reactor is tripped and an attempt is made to close the steam generator isolation valves. Although the signal is sent to close, the main steam outlet line fails to be isolated. Steam continues to ingress into the primary system from other modules until operator action terminates the event. Core cooling is provided by the SCS. Excessive primary system pressure opens the primary relief valve, venting primary circuit radionuclides into the reactor building. Once the relief valve lifts, it fails to reclose as designed. The event is essentially terminated at this time even though steam continues to ingress into the system after the relief valve has opened. Since the location of the primary relief valves is the steam generator vessel head, continued ingresses will not contact the reactor core and cause further fuel hydrolysis; rather, steam will vent directly through the open relief valve to the reactor building. The source term for this accident family is composed of circulating activity, hydrolysis products, and some liftoff of plated-out material. Consequences have been taken from Ref. 8-2 and reduced to account for improved fuel quality and a realistic leakage rate as noted earlier.

Accident family S/G-3 results in fission product release to the reactor building and subsequently to the atmosphere. It is similar to accident family S/G-1 in that isolation fails. In this case, however, no signal is sent by the PPIS to isolate the steam generator or to trip the reactor. The reactor is eventually tripped on high pressure, but moisture continues to enter the primary system. The core continues to be cooled on the HTS until operator intervention trips the main circulator and isolates the steam generator. The moisture entering the primary system in this event is of lower quality than in S/G-1 because both the steam and feedwater lines are not isolated. Moisture will, however, be entrained and transported to the reactor core by continued operation of the main circulator. Upon reaching the core, the moisture will contact hot regions and be evaporated. The amount of steam available for failed fuel hydrolysis will therefore be similar to that if only steam line isolation had failed. The ingress is sufficient to lift the primary relief valves, following which reclosure fails. The primary coolant activity rapidly depressurizes through the open relief valve into the reactor building and is subsequently released to the atmosphere. The source term contributors are circulating activity, hydrolysis products, and some liftoff of plated-out radionuclides. The offsite dose consequences of this accident family are assumed to resemble those of S/G-1 because of similarities in the amount of steam available to react and the radionuclide release paths.

Accident family S/G-6 is identical to family S/G-1 with the exception that the primary relief valve recloses after relieving the primary system pressure to a value 15% below the opening pressure of 1041 psia. With steam ingressing into the primary system at a rate of 2.6 lbm/s, the primary relief valve will open at approximately 22.5 min into the transient. When the pressure has been reduced to 885 psia, the valve recloses. The probability of the primary relief valve opening a second time in this event sequence is less than 10^{-9} per plant year and has been truncated in the event tree of Fig. 7-3. The consequences of this accident family are therefore estimated to be 15% of the consequences of S/G-1.

Accident family S/G-9 is identical to family S/G-3 except the primary relief valve recloses after relieving primary system pressure. The consequences are the same as those for S/G-6 described above, since S/G-1 and S/G-3 consequences have been identified as being approximately equal. Operator response time, moisture ingress rate, and gas conditions are assumed to be the same as those for S/G-6; therefore, yielding the same release fraction.

Accident family S/G-10 is identical to family S/G-2 except the primary relief valve does not fail open but successfully recloses. Recall that in S/G-2, the plant responds as planned except the dump system valves do not open on demand which results in an overpressure of the primary system due to excessive moisture ingress. Operator intervention in this case cannot be assumed because valve failure is mechanical rather than resulting from the absence of a signal from the PPIS. The consequences of S/G-10 are a fraction of those associated with S/G-2. The moisture ingress in this case is not terminated until approximately one-half of the steam generator inventory is placed into the primary circuit (assuming a median location of the leak area at the steam generator midplane). Once the relief valve opens, it relieves primary system pressure until 885 psia is reached. As noted earlier, this is 15% of the opening pressure of 1041 psia. The remaining available moisture continues to ingress into the system, causing the relief valve to cycle open and closed for a second time. Based on these considerations, the calculated release fraction was found to be 0.3. The consequences of family S/G-10 are therefore approximately 30% of those associated with family S/G-2.

The final moderate steam generator leak release category is designated S/G-11. This accident family results in the release of fission products directly to the atmosphere through a failed open steam generator relief train. Following detection of the leak by the moisture monitors, the PPIS signals the isolation valves to close. Steam line valves close but the feedwater valves do not. The steam generator relief train normally is exposed to steam pressure of 2515 psia, but in this accident scenario it is exposed to feedwater pressure of 3000 psia. The steam generator relief train opens to relieve the excessive pressure and fails to reclose. Core cooling is provided by the SCS. Excessive moisture in the primary system lifts the primary relief valve which successfully reseats after pressure relief. The consequences of S/G-11 have been approximated based on previously evaluated accident families. The contribution to the dose from radionuclides released through the primary relief train is estimated to be similar to family S/G-9. The remaining primary coolant activity will then depressurize through the leak in the steam generator to the atmosphere through the open steam generator relief train. The consequences of this portion of the release are approximated as resembling those for accident family S/G-8, which has a depressurization through the dump system tank relief valve to the atmosphere, adjusting for the water retention factor applied when depressurizing through the dump system.

For all accident families, the reactor building parameters and meteorological data are as given in Table 8-2. If fission product release is through the steam generator secondary side, no attenuation has been assumed with the exception of the case were depressurization is through the steam generator dump system tanks. For this case, attenuation of halogens and particulates by the dump tank water inventory is accounted for. The offsite dose consequences for each accident family are summarized in the results of Section 9.

8.3.3 Uncertainty Analysis

The consequence uncertainty model used for moderate steam generator leak initiated events is the same as the model given in Section 8.1.3 for primary coolant leak initiated events. The uncertainty distributions for steam generator leaks as given in Ref. 8-2 were modified by replacing the median consequences for the Ref. 8-2 assessment by the current assessment consequences. For those events that were not previously analyzed in Ref. 8-2, uncertainty distributions for similar events that were analyzed in Ref. 8-2 were applied.

8.4 Conduction Cooldown Accident Consequences

Conduction cooldown accidents result from the loss of both the HTS and SCS core cooling systems. Core decay heat removal is then accomplished by conductive and radiative heat transport to the RCCS cooling panels. Conduction cooldown accidents can occur with the reactor vessel either pressurized or depressurized. In the event the reactor vessel remains pressurized and there is no ingress of moisture or a reactivity excursion, primary coolant boundary integrity is maintained and there is no offsite dose. However, a pressurized conduction cooldown accompanied by an ingress of moisture or reactivity excursion may lead to lifting and subsequent reclosure of the primary relief train valve. In this case, there will be some release of fission products while the relief valve is open. Alternatively, in the depressurized conduction cooldown accident, the primary coolant boundary remains open, allowing the release of a larger fraction of radionuclides from the reactor vessel. For the purposes of safety risk quantification, depressurized conduction cooldown transients as well as pressurized transients where the relief valve lifts are considered. Pressurized conduction cooldown transients that do not result in lifting the primary relief train valves will not be considered here because they have no impact on the safety risk of the plant. They have been identified in the frequency assessment portion of this report for the purposes of future licensing analyses.

Conduction cooldown accidents subsequent to or concurrent with failure to maintain primary coolant pressure boundary integrity result in fission product release and offsite dose consequences. The release of radionuclides may consist of primary coolant circulating activity, hydrolysis products, liftoff or washoff of plated-out material, if any, or partial fuel body activity release due to the thermal transient. Initiating events for conduction cooldown accidents that will be addressed here include primary coolant leaks, small and moderate steam generator leaks, seismic activity, and control rod bank withdrawal.

8.4.1 Data Base

The thermodynamic data used are those that are typically used to analyze the thermal and fission product release response of the reactor core during conduction cooldown accidents. Pertinent details are cited in Ref. 8-2.

8.4.2 Physical Phenomena and Consequence Assessment

As noted in the Section 7 frequency assessment, conduction cooldown transients are most likely to be initiated by primary coolant leaks, less likely to be initiated by steam generator leaks, and least likely to be initiated by either seismic activity or a spurious control rod bank withdrawal.

Regardless of whether the reactor vessel remains pressurized for some period of time or depressurizes immediately, failure of all forced core cooling systems leads to a slow heatup of the core with decay heat removal accomplished by conduction and radiation to the RCCS cooling panels. Peak and average core temperatures will be lower in the pressurized condition due primarily to the redistribution of heat from hotter to cooler portions of the core by natural convective flows established within the core. The thermal transient for a depressurized conduction cooldown was modeled using a two-dimensional finite difference method and a series of nodal points describing the system. A finite difference equation is formulated for each nodal point in terms of its capacitance, heat generation, and heat flow paths to neighboring nodal points. A system of these equations is solved by an implicit method to generate local temperatures at given points in time. The modeled system encompasses the reactor core, internals, vessel, and the RCCS as the heat sink. The transient was analyzed using the TAC2D computer code (Ref. 8-6) for a time period of 1000 h. The results of the analysis indicate a thermal transient experienced by the core as shown in Fig. 8-6 for both the peak fuel and average active core temperatures over time. This particular transient assumes immediate depressurization. Figure 8-7 shows an isothermal plot at the time of temperature peaking, demonstrating that the peak fuel temperature is experienced in only a small fraction of the core.

For cases in which the reactor vessel remains pressurized for some period of time before pressure relief, the thermal transient was modeled using a finite difference method to calculate transient temperatures in a network of thermal capacitances and conductors. The model allows both solid and fluid nodes to determine the effect of natural circulation of helium within the reactor core. This model was also used to determine the temperatures associated with a time-dependent pressure relief such as in the case of a small primary coolant leak which depressurizes slowly over a long time period. Analysis was done using the GA-developed computer code, PANTHER. Figure 8-8 shows the resultant temperature profile for the case where the reactor vessel remains pressurized.

Transient temperature profiles for both the pressurized and depressurized conduction cooldown cases were used to determine fuel failure and subsequent

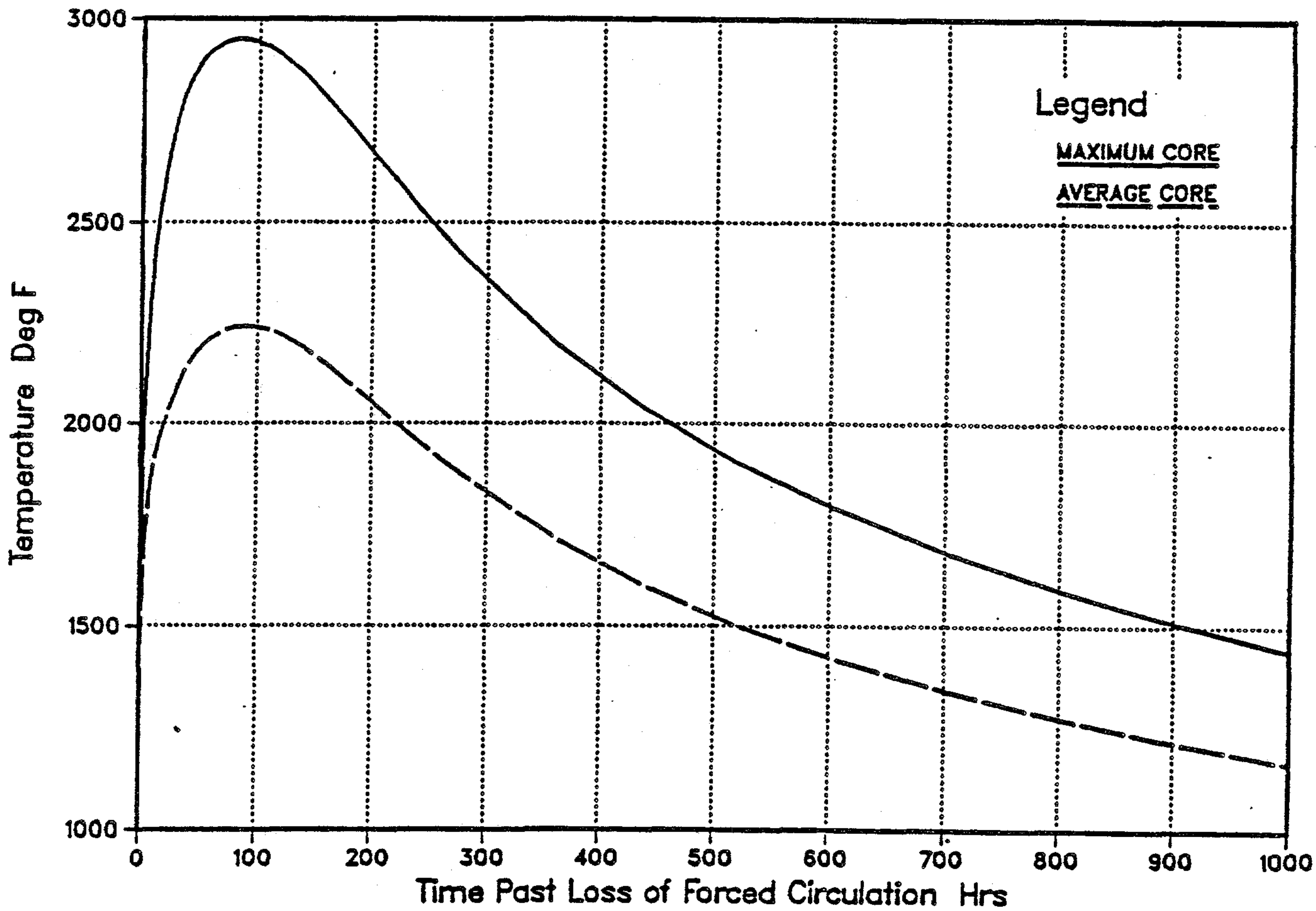


Figure 8-6. Temperature Transients During a Depressurized Conduction Cooldown with RCCS Cooling for the Modular HTGR

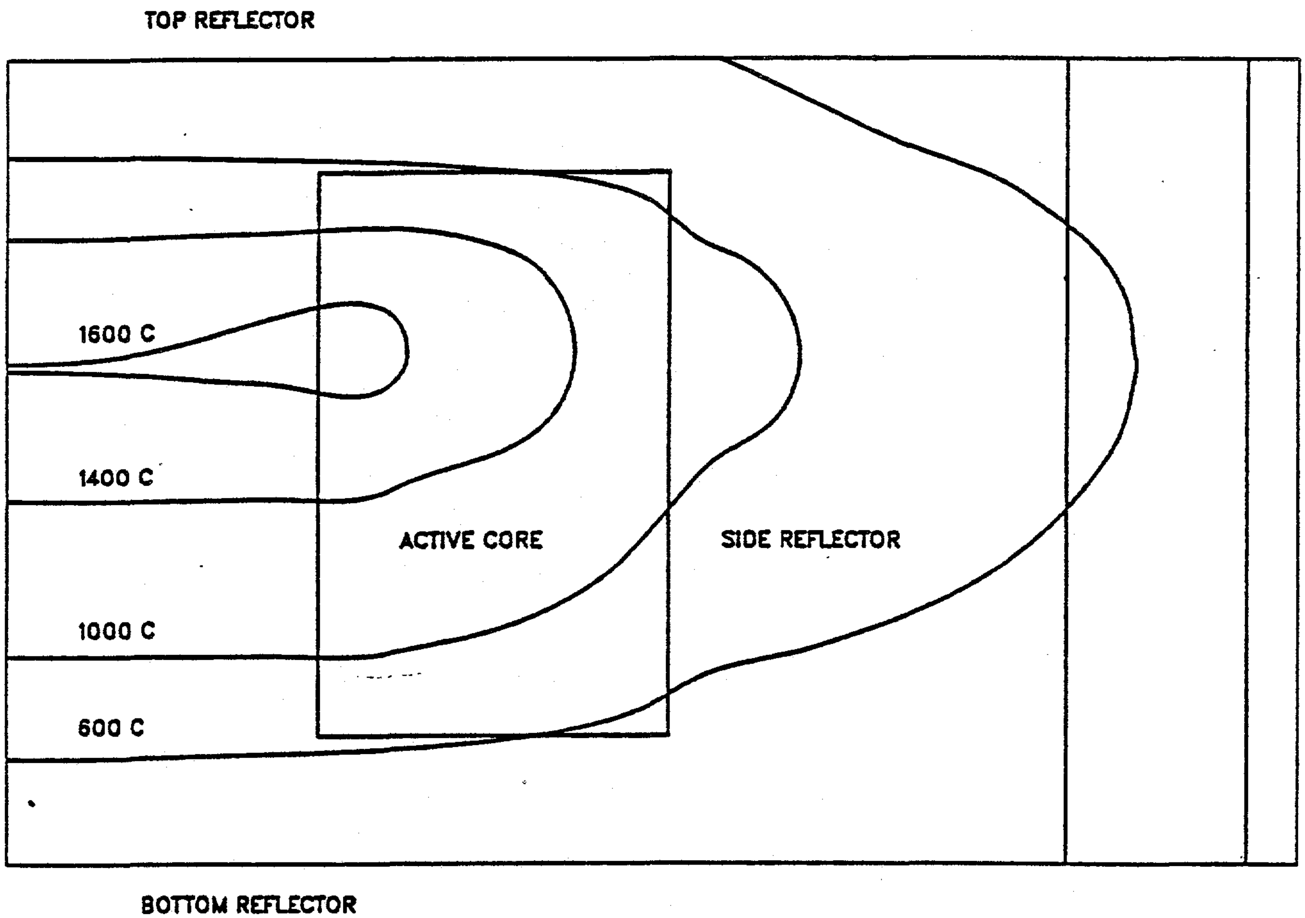


Figure 8-7. Isothermal Plot at Time of Temperature Peaking During a Depressurized Conduction Cooldown for the Modular HTGR

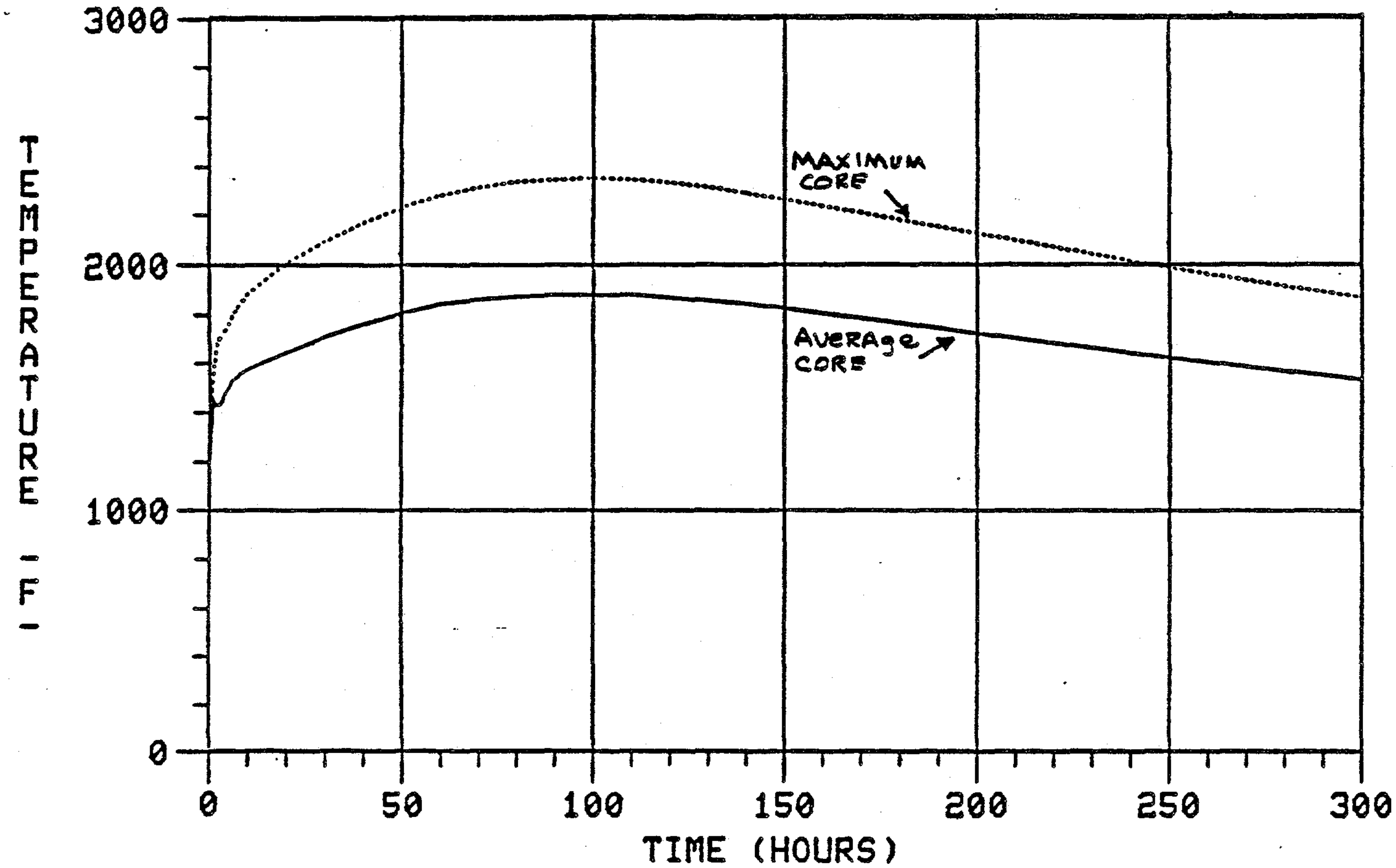


Figure 8-8. Temperature Transients During a Pressurized Conduction Cooldown with RCCS Cooling for the Modular HTGR

fission product release over time. A segmented cylindrical core spatial model with associated values of material quantities and temperature conditions was used to determine the time-dependent amounts of fission product nuclides that escape the from the core in the gas phase. Fission product transport and decay chain behavior are described by a set of differential equations which describe the entire core radionuclide inventory by means of calculated parameters based on the detailed spatial core conditions. The effects of sorption within the reactor core are considered for all nongaseous species. Behavior of nonfission product core materials such as the core graphite are also considered. This model has been applied by the computer code SORS (Ref. 8-7) developed at GA specifically for HTGRs. The results of the SORS analysis indicate that as the core heats up above normal operating temperatures, fission products are slowly released as a function of time from the fuel. Cumulative curie releases from the core as a function of time are shown in Figs. 8-9 and 8-10, respectively, for pressurized and depressurized conduction cooldown accidents. Included in the figures are releases for the dominant dose contributors for thyroid and whole body gamma doses.

Metallic fission products such as cesium and silver are released primarily from intact fuel particles by diffusion. This occurs in the small central core region where peak fuel temperatures are experienced as depicted in Fig. 8-7. This release of metallic fission products is, however, quickly readsorbed on cooler graphite surfaces in the core. The SORS computer code therefore predicts essentially no release of metallic fission products from the core.

8.4.2.1 Primary Coolant Leak Initiated Conduction Cooldown Accidents. Nine conduction cooldown accident families initiated by primary coolant leaks were identifies in the frequency assessment of Section 7.3.1. These accident families have been designated CC_p-2, CC_p-4, and CC_p-6 through CC_p-12.

Following detection of a primary coolant leak, the reactor is tripped on low primary coolant pressure. Core cooling is not provided by the HTS for a

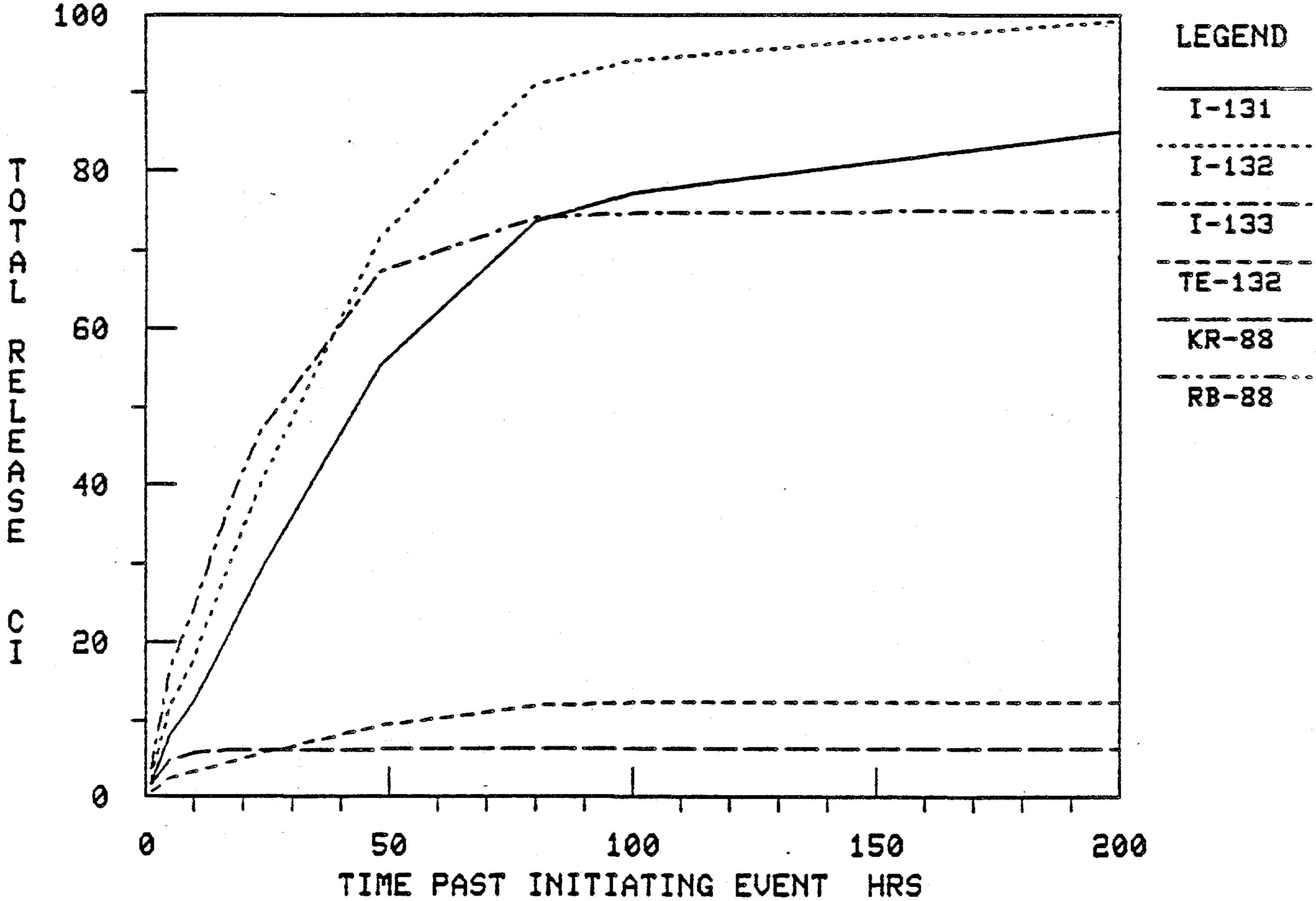


Figure 8-9. Fission Product Release from the Core During a Pressurized Conduction Cooldown for the Modular HTGR

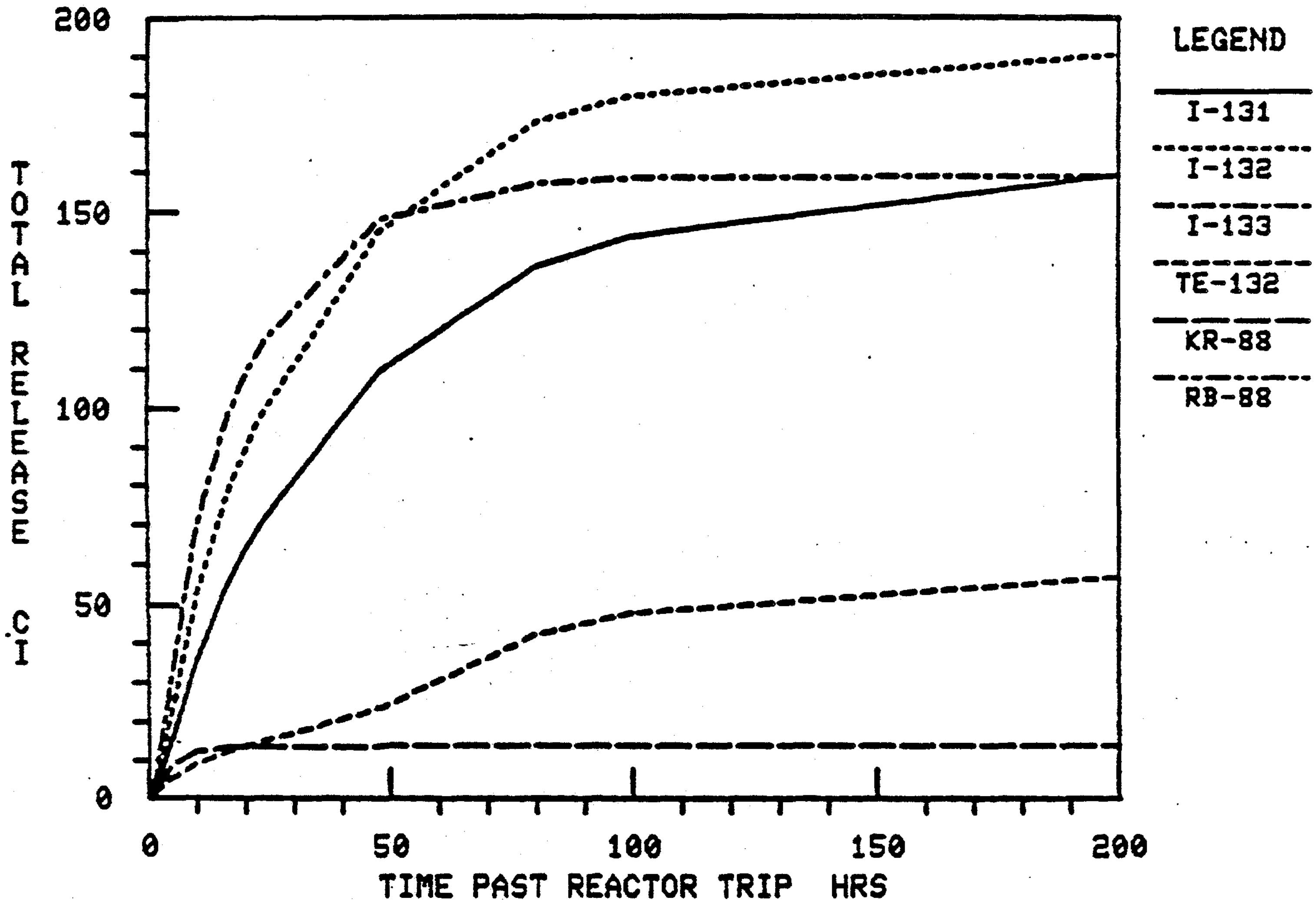


Figure 8-10. Fission Product Release from the Core During a Pressurized Conduction Cooldown for the Modular HTGR

sufficient amount of time to mitigate heatup of the core. Attempts to provide cooling by the SCS fail. Core decay heat is rejected by conductive and radiative heat transport to the RCCS cooling panels. During the initial phase of the transient where depressurization of the primary coolant is taking place, the physical phenomena and fission product release characteristics are as discussed in Section 8.1.2 for primary coolant leaks where forced core cooling succeeds and there is no conduction cooldown. Circulating activity and any liftoff of plated-out material is essentially released during the initial depressurization, followed by the slow release of fuel body activity as the core heats up during the conduction cooldown transient. The transport mechanisms in the reactor vessel consist of the initial helium depressurization, hydrostatic displacement of helium, if any, and slow thermal expansion of gases during the heatup of the core. Upon completion of the initial primary coolant depressurization and the core heatup phase, the core begins to cool down and fission product release from the reactor vessel is essentially terminated due to a lack of any transport mechanism.

The following paragraphs describe the identified accident families given above and the governing fission product release mechanisms for each representative leak size.

Accident families CC_p-12 and CC_p-11 are represented by a characteristic leak area of 0.001 in.^{2}. The size may range from 3×10^{-5} to 2×10^{-3} in.^{2}. For sizes this small, the depressurization time for the reactor vessel is on the order of hundreds of hours. The transient temperature profile is therefore approximated by that given in Fig. 8-8 for pressurized conduction cooldowns. These accident families are similar except that pumpdown of primary coolant fails in the case of CC_p-11. Since a pumpdown time of 30 h is used for this assessment, the sequence defining CC_p-12 is terminated in 30 h instead of the several hundreds of hours required to depressurize if pumpdown fails as in CC_p-11. Dose consequences for these categories are based on their companion primary coolant leak accident families as shown in Fig. 7-1. The dose consequences for these accident families are assumed to be similar to the primary coolant leak accident family consequences because of the lower core temperatures during the pressurized conduction cooldown.

For leak sizes between 2 × 10^{-3} in.^{2} and 0.03 in.^{2}, a representative leak area of 0.01 in.^{2} has been selected. Conduction cooldown accident families in this size range are designated CC_p-2, CC_p-4, and CC_p-7. For this leak size range, the depressurization time for the reactor vessel is approximately 100 h in the absence of HPS pumpdown. The transient temperature profile for these leaks is approximated by Fig. 8-6 for a depressurized conduction cooldown. Accident family CC_p-7 is the least consequential of the three in this size range. In the event sequence describing this family, pumpdown of the primary coolant is successful, and the reactor building responds as planned by disengaging HVAC fans and isolating the filter system. CC_p-4 is the same as CC_p-7 except the HPS fails to pumpdown primary coolant to storage. The result is a depressurization over 100 h where fuel body activity released during that time period has a mechanism to be transported out of the reactor vessel. For CC_p-7, pumpdown occurs and releases from the fuel are considered only up to 30 h when pumpdown is complete. Family CC_p-2 considers sequences where pumpdown is successful but the reactor building does not respond properly. Failure to isolate filters and turn off the building HVAC fans results in fission product release from the reactor building in excess of what normally would be expected. Fission product attenuation in the reactor building by means of plateout, settling, and volumetric holdup does not have adequate time to take place in this event. Consequences for these accident families were taken from the Ref. 8-3 assessment and reduced to account for improved fuel quality specifications.

Four accident families have been identified that result in depressurization areas of 1 in.^{2} or greater. These families have been designated CC_p-10, CC_p-9, CC_p-8, and CC_p-6. For leaks of this size, the initial depressurization and hydrostatic displacement, if any, of helium occur within the first hour of the transient. During this time, the fission products released by the depressurization include those from circulating activity, fractional liftoff of plated-out materials, and any releases from the core due to the conduction cooldown. At times later than 1 h, the transport mechanism for fission products released from the core is by thermal expansion of gases. This release mechanism is available up to 80 h, at which time the core temperatures begin to decrease and fission product release is essentially terminated. A review of Figs. 8-6 and 8-10 indicate a high thermal expansion rate and low fission product release rate at earlier times. As the rate of fission product release from the core increases with time, the thermal expansion rate decreases which results in a much lower fission product release from the reactor vessel. The effect of primary coolant pumpdown on leaks in excess of 1 $in.^2$ is negligible as shown in Fig. 8-1. For leaks of 10 $in.^2$ or greater, the response of the reactor building is not important because the high depressurization rate of primary coolant through the leak area is such that all releases go directly through the building dampers with very little time for fission product attenuation. Accident families CC_p-9 and CC_p-10 encompass leaks in the range of 0.03 to 1 $in.^2$. The representative size was selected as 1 $in.^2$. Following reactor trip and loss of forced circulation systems, the reactor building responds as planned for CC_p-9 but fails in the case of CC_p-10. CC_p-8 describes leaks in the range of 1 to 13 $in.^2$ with 10 $in.^2$ being selected as representative. CC_p-8 results in more liftoff of plated-out material and a shorter depressurization time than previously described categories. The final category of CC_p-6 describes leaks in excess of 13 $in.^2$. Analysis was done for a leak area of 30 $in.^2$. An increased liftoff fraction and reduced depressurization time describe CC_p-6. Consequences for these accident families have been derived from Ref. 8-3 results and reduced to account for improved fuel quality specifications.

For all conduction cooldown categories initiated by primary coolant leaks, plateout and settling in the reactor building on surfaces cooled by the RCCS have been considered. Meteorological conditions and reactor building parameters are as given in Table 8-2. A summary of the offsite dose consequences for each representative leak size considered is given in Table 8-5. For each size, the HPS and reactor building are assumed to function as designed. Median dose consequences at the plant EAB are given along with the associated uncertainty factors for both the whole body and thyroid. The highest dose consequences are for the case of the small 0.01 $in.^2$ leak where the slow depressurization releases significantly more of the fractional fuel

TABLE 8-5
PRIMARY COOLANT LEAK INITIATED CONDUCTION COOLDOWN
DOSES AT THE EAB FOR VARYING LEAK SIZES

		Median Dose			
		Whole Body Gamma		Thyroid	
Accident Family	Leak Size (in.^{2})	Rem	f*	Rem	f
CC_p-12	0.001	3.8-5	10	1.9-6	10
CC_p-7	0.01	7.0-4	10	1.2-2	12
CC_p-9	1	5.6-3	10	1.3-2	10
CC_p-8	10	6.2-4	10	2.2-3	10
CC_p-6	30	6.4-4	10	6.8-3	10

*f = Ratio of 95th percentile to median; assumed distribution is lognormal.

body inventory released during the slow heatup of the core. Other accident family consequences not given in the table are summarized in Section 9.

8.4.2.2 Small Steam Generator Leak Initiated Conduction Cooldowns. Small steam generator leaks with a subsequent primary coolant boundary failure and loss of forced core cooling result in conduction cooldowns with an offsite dose to the public. The fission product release pathway may be either through the steam generator secondary side or to the reactor building if the primary relief train valves lift. Releases may consist of circulating activity, radionuclides washed off wetted primary circuit surfaces, release from hydrolyzed fuel, release from oxidized graphite, release from liftoff of plated-out material, or release from the fuel body inventory due to the thermal transient. The frequency assessment of Section 7.3.2 for small steam generator leaks covers a spectrum of leak sizes ranging from pinhole to approximately 8×10^{-3} in.^{2}. The maximum size considered corresponds to a leak rate of about 0.1 lbm/s and will be used for the consequence assessment in this section.

The planned response to a moisture ingress event is described in Section 8.2 for small steam generator leaks with forced circulation. In this case, however, forced circulation by the SCS is lost and core heat removal is by conduction and radiation to the RCCS cooling panels. The resulting transient is a pressurized conduction cooldown with pressures low enough that the primary relief train valve is not lifted. This sequence of events results in no fission product release as the primary coolant pressure boundary remains intact and is observed to correspond to sequence SS-AB in Fig. 7-2.

For fission product release to occur, additional failures are required that result in failure of the primary coolant boundary to contain the fission products. As shown in Fig. 7-2, failures in addition to the small steam generator leak and loss of forced circulation may result in a number of accident sequences that result in fission product release. Failure of the steam generator dump system or failure to isolate precedes each event sequence where an offsite dose occurs. A total of six accident families are under consideration in this section, and they are labeled CC_s-3 through CC_s-7 and CC_s-10. The following paragraphs describe for each accident family the dominant sequence, fission product release path, contributors to the release, and basis for the consequence assessment.

Accident family CC_s-10 results in fission product release through the steam generator dump system to the atmosphere. System response is as planned until the steam generator dump valves are signaled to reclose following the dump of the steam generator inventory to the dump system tanks. The valves fail to reclose, thereby opening a pathway for fission products to the atmosphere. Since it has been assumed that the dump tanks are not designed to contain primary coolant pressure, the helium inventory depressurizes slowly through the open dump system, through the tank relief valves to the atmosphere. Forced core cooling by the SCS fails, and core decay heat is removed by conduction and radiation to the RCCS. Releases to the environment include circulating activity, hydrolysis products, and fuel releases due to thermal effects. Consequences for this accident family have been estimated to be similar to those of families S/G-4 and CC_p-7 evaluated in earlier sections. Recall from Section 8.2 that S/G-4 is the same as CC_s-10 except forced circulation is successful. Family CC_p-7 is described in Section 8.4.2.1 for primary coolant leak initiated conduction cooldowns and corresponds to a leak size resembling that of a small steam generator tube leak. Summing the consequences for these two accident families is used as an approximation of the consequences for CC_s-10.

Accident family CC_s-3 results in fission product release through an open steam generator relief train directly to the atmosphere. Following detection of the leak by the moisture monitors, the PPIS signals the steam generator isolation valves to close. Steam line isolation is successful, but feedwater line isolation is not. The steam generator relief valve is subsequently exposed to 3000 psia feedwater pressure which is in excess of its setpoint value. The valve opens to relieve pressure and does not reseat as designed. Core cooling by the SCS fails, and heat is transported to the RCCS by conduction and radiation. The total amount of moisture ingressed into the primary circuit in this event is small enough that negligible hydrolysis will take place. The probability that a significant amount of moisture is ingressed in this event sequence places the total sequence frequency below 10^{-9} per plant year and is not considered. The release for this accident is constituted by circulating activity and fuel body activity released during the thermal transient. The consequences for CC_s-3 are estimated to be similar to CC_p-7 evaluated earlier. CC_p-7 corresponds to a leak in the primary coolant pressure boundary with a size approximating that of a small steam generator leak.

Accident family CC_s-6 results in fission product release to the reactor building and subsequently to the atmosphere. Following the initiating event of the small steam generator leak, isolation of the steam generator fails. Core cooling by forced circulation is not successful, and the core experiences a pressurized conduction cooldown transient. Moisture continues to ingress into the primary circuit causing the primary relief train valves to open in approximately 4.5 h. After pressure relief the valve does not reclose, allowing the primary circuit to depressurize rapidly into the reactor building and through the building dampers into the atmosphere. Continued ingresses of moisture into the primary system are at a very slow rate and do not add appreciably to the releases from the reactor vessel. Releases from the reactor vessel consist of primary coolant circulating activity, liftoff of plated-out material, hydrolysis products, and fuel body releases due to the pressurized conduction cooldown. Consequences for CC_s-6 have been based on previously evaluated scenarios modified to account for differences in total moisture available for hydrolysis and releases from the fuel during the conduction cooldown.

Accident family CC_s-4 is the same as CC_s-6 except the primary relief valve responds as designed and successfully recloses following pressure relief. Primary circuit activity is released into the reactor building during the relief valve cycle, through the building dampers, and into the atmosphere. The probability that the relief valve will lift a second time in this event sequence is less than 10^{-9} per plant year and therefore has been truncated in the event tree of Fig. 7-2. One cycle of the relief valve releases approximately 15% of the mass present in the system to the reactor building. The consequences of CC_s-4 are therefore estimated to be 15% of CC_s-6 consequences.

Accident family CC_s-5 results in fission product release to the reactor building and subsequently to the atmosphere. Moisture monitors successfully detect moisture in this scenario, initiating a reactor trip, main circulator trip, and steam generator isolation. The steam generator dump valves fail to open following steam generator isolation, posing a potential ingress into the primary system of that fraction of the steam generator inventory located above the leak area. It has been assumed in this assessment that the leak location is at the steam generator midplane, therefore allowing 50% of the steam generator inventory to ingress into the primary system. Primary system pressure continues to increase, and the primary relief train valve opens at approximately 4.5 h into the transient. The valve fails to reseat following pressure relief, depressurizing the primary circuit inventory into the reactor building, through the building dampers, and into the atmosphere. Moisture continues to ingress until the steam generator inventory falls to 50% but, because of the slow inleakage rate, does not add appreciably to the vessel releases. The consequences of CC_s-5 are assumed to resemble those of CC_s-6 because of similarities in the total moisture ingressed, the failure to reseat the relief valve, and in the valve opening time.

The final accident family under consideration in this section is designated CC_s-7. CC_s-7 is identical to CC_s-5 up until the time the relief valve opens. In accident family CC_s-7, the relief valve reseats and cycles open and closes a second time during the transient. It has been assumed that the dump system valves cannot be opened manually by operator intervention due to their inaccessibility. After the valve cycles open and closed at approximately 4.5 h into the transient, temperature increases caused by the pressurized conduction cooldown transient increase system pressure a second time to relief valve setpoint pressure. The valve opens once again at approximately 15 h and successfully reseats. The relief valve remains closed following the second relief due to termination of the ingress once 50% of the steam generator has been emptied. Consequences for CC_s-7 have been approximated as being similar to those of CC_s-4 with the addition of consequences for activity released during the second relief valve cycle. Consequences associated with the second relief valve cycle have been estimated, based on the results of previously evaluated scenarios with corrections made for the amount of moisture present and activity released from the fuel due to the thermal transient up until the time the relief valve lifts for the second time. Of the available activity in the system at the time the relief valve lifts the second time, 15% is assumed to be released into the reactor building and contribute to the offsite dose.

For all accident families, the reactor building and site data are given in Table 8-2. Attenuation through the steam generator secondary side was not considered except in the case of depressurization through the steam generator dump system. In this event attenuation of halogens and particulates by the dump tank water inventory was considered. A summary of offsite dose consequences for each accident family is found in the risk assessment results of Section 9.

8.4.2.3 Moderate Steam Generator Leak Initiated Conduction Cooldowns. Moderate steam generator leaks with a subsequent primary coolant boundary failure and loss of forced core cooling result in conduction cooldowns with an offsite dose to the public. The fission product release pathway may be either through the steam generator secondary side or to the reactor building if the primary relief train valves lift. Releases may consist of circulating activity, radionuclides washed off wetted primary circuit surfaces, release from hydrolyzed fuel, release from oxidized graphite, release from liftoff of plated-out material, or release from the fuel body inventory due to the thermal transient. The frequency assessment of Section 7.3.3 for moderate steam generator leaks covers a spectrum of flow rates ranging from 0.1 to 12.5 lbm/s. The consequence assessment has been based realistically on an average leak rate of 2.6 lbm/s.

The planned response to a moisture ingress event is described in Section 8.3 for moderator steam generator leaks with forced circulation. In this case, forced circulation is lost and core heat removal is by conduction and radiation to the RCCS cooling panels. The resulting transient is a pressurized conduction cooldown with pressures low enough that the primary relief train valve is not lifted. This sequence of events results in no fission product release as the primary coolant pressure boundary remains intact and is observed to correspond to sequence MS-AB in Fig. 7-3.

For fission product release to occur, additional failures are required that result in failure of the primary coolant boundary to contain the fission products. As shown in Fig. 7-3, failures in addition to the steam generator leak and loss of forced cooling may result in a number of sequences that result in fission product release. Failure of the steam generator dump system or failure to isolate precedes each event sequence where an offsite dose occurs. A total of six accident families are under consideration in this section and are labeled CC_s-8, CC_s-9, and CC_s-11 through CC_s-14. The following paragraphs describe for each accident family the dominant sequence, fission product release path, contributors to the consequence source term, and basis for the consequence assessment.

Accident family CC_s-9 is the only identified family whose release path is through the steam generator secondary side. Response to the moisture inleakage proceeds as planned until the dump system valves are signaled to reclose after dumping the steam generator inventory to the dump system tanks. Core cooling by the SCS fails, and heat is removed by conduction and radiation to the RCCS. The primary circuit is depressurized through the steam generator leak into the dump system, through the tank relief valves and into the atmosphere. The release consists of circulating activity and releases from the fuel due to the conduction cooldown. Negligible hydrolysis takes place during this transient because the high fuel and graphite temperatures induce graphite oxidation, leaving very little water available for reaction with the fuel. The transient is complete in an hour after which time any additional fuel body releases are contained within the reactor vessel due to the lack of sufficient transport mechanisms out of the vessel. The dose assessment for this accident family was based on the Ref. 8-3 results. The consequences were reduced to account for improved fuel quality specifications and a realistic, instead of bounding, moisture ingress rate. Improved fuel quality results in a reduction in both the number of fuel particles available for hydrolysis and the fission product release from the core during a conduction cooldown transient.

Accident family CC_s-11 results in fission product release to the reactor building and subsequently to the atmosphere. In addition to losing forced cooling by the SCS, the dump system valves fail to open following isolation of the steam generator. One-half of the steam generator inventory is available for ingress into the primary system (assuming the leak is located at the steam generator midplane). It has been assumed that the valves cannot be manually opened to mitigate consequences because of their inaccessibility. The pressure continues to increase in the reactor vessel until the primary relief valve lifts. The relief valve fails to reclose following pressure relief and the primary circuit inventory rapidly depressurizes into the reactor building. Fission product release consists of circulating activity, hydrolysis products, and liftoff of plated-out material. Very little fuel release is incurred before the relief valve fails open because temperatures have not increased significantly. Moisture will tend to preferentially hydrolyze fuel instead of oxidize graphite until temperatures have increased. Radiological consequences have been based on the Ref. 8-3 assessment and reduced to account for improved fuel quality and a realistic ingress rate as noted above.

Accident family CC_s-12 results in fission product release to the reactor building and, subsequently, to the atmosphere. High moisture levels are not detected by the moisture monitors resulting in no isolation and trip signal by the PPIS. Reactor trip eventually occurs on high pressure but moisture continues to enter the primary system. The core continues to be cooled on the HTS until operator intervention trips the main circulator and isolates the steam generator. Following the main loop trip, forced core cooling on the SCS fails, and decay heat is subsequently removed by conduction and radiation to the RCCS cooling panels. The moisture entering the primary system is entrained and transported to the reactor core by continued operation of the main circulator. Upon reaching the core, the moisture will contact hot regions and be evaporated. This ingress is sufficient to lift the primary relief valves, following which reclosure fails. Primary coolant activity rapidly depressurizes through the open relief valve into the reactor building and is subsequently released to the atmosphere. Since the conduction cooldown transient does not begin until after the main circulator is tripped, releases at the time the relief train valve opens will not include any significant contribution from the thermal transient. Fuel hydrolysis will, however, occur and is estimated to be similar to that described for accident family S/G-3 as discussed in Section 8.3. Recall that accident family S/G-3 is identical to family CC_s-12 being discussed here with the exception that the SCS is available to provide core cooling. The offsite dose consequences of CC_s-12 are, therefore, assumed to resemble those of S/G-3 because of similarities in the amount of steam available to react, core temperatures up to the time of pressure relief, and the radionuclide release path.

Accident family CC_s-14 is identical to family CC_s-12 with the exception that the primary relief valve recloses after relieving the primary system pressure. The opening pressure is assumed to be 1041 psia and the reclosing pressure 885 psia. With steam ingressing into the primary system at a rate of 2.6 lbm/s, the primary relief valve opens at approximately 22.5 min into the transient and then recloses. The probability of the primary relief valve opening a second time in this event sequence is less than 10^{-9} per plant year and has been truncated in the event tree of Fig. 7-3. The consequences of this accident family are estimated to be 15% of CC_s-12, 15% being the fraction of material in the primary circuit released during the relief valve cycle.

Accident family CC_s-8 results in fission product release to the reactor building and subsequently to the atmosphere. Following successful detection of high moisture levels, the reactor is tripped, and isolation of the steam generator is signaled by the PPIS. Feedwater isolation succeeds but isolation of the steam outlet line fails. Steam continues to ingress into the primary system from other modules until operator action terminates the event. Attempts to cool the core on the SCS fail and heat removal is by conduction and radiation to the RCCS. Excessive primary system pressure opens the primary relief valve, venting primary circuit radionuclides into the reactor building. The relief valve then recloses as designed. The consequences of this family are approximated by those of CC_s-14 because of similarities in the amount of steam that reaches the core and radionuclide release path. Although in CC_s-14 the moisture is initially of a lower quality than that in CC_s-8, the moisture is eventually vaporized because of continued HTS circulator operation. The probability of the relief valve opening a second time is truncated in this event sequence as well.

The final accident family to be considered in this section is designated CC_s-13. This family is identical to family CC_s-11 except the relief valve successfully recloses. Recall that in CC_s-11, the plant responds as planned with the exception that dump valves fail to open and SCS cooling fails. Moisture ingress is not terminated until approximately one-half of the steam generator inventory ingresses into the primary system. Because of continued ingresses, the relief valve cycles open and close twice before the steam generator is emptied. This results in a release of 30% of the available activity in the primary system to the reactor building. Fuel body releases are not significant contributors because of the short time involved to empty the steam generator and terminate the transient. The consequences of CC_s-13 are approximated as 30% of those for CC_s-11.

Site data and reactor building parameters as given in Table 8-2 were used for the consequence assessment for all accident families. Accident sequences which resulted in release through the steam generator secondary side did not consider attenuation of fission products except where depressurization was through the dump system. In this case, attenuation of halogens and particulates by the tank water inventory was considered. A summary of the offsite doses for each accident family is given in Section 9.

8.4.2.4 Earthquake Initiated Conduction Cooldowns. The frequency assessment of Section 7.3.8 identified five conduction cooldown categories initiated by earthquakes, two of which result in no offsite dose consequences as primary coolant boundary integrity is maintained.

Consequence assessment over ground accelerations ranging from 0.3 to 3 g was performed following the structure of the MLD discussed in further detail in Section 6. Each branch of the diagram was reviewed and an estimate made of the acceleration required to cause such a failure. Fragility models for making these estimates were scaled from the component fragility models in Refs. 8-8 and 8-9. Generally, though not always, the ground acceleration required to cause equipment failure was scaled from the ratio of the site SSE intensities. For example, for the MHTGR designed to an SSE of 0.3 g, the acceleration required to fail a certain piece of equipment would be assumed to be 150% of that required to fail a similar piece of equipment at Seabrook which was designed to a 0.2 g SSE. The most common variation from this relationship occurs when a comparison of the Seabrook and Zion fragilities for a certain piece of equipment shows little or no difference, suggesting that the item might be a standard "off-the-shelf" item since the Seabrook and Zion plants were designed to SSEs of different magnitudes. In these cases, an identical equipment fragility model is used for the MHTGR.

Scanning the fragility models in Refs. 8-8 and 8-9 quickly reveals that for large fractions of the Seismic 1 NSSS equipment, failure does not occur until ground acceleration exceeds five to ten times the SSE acceleration. These large margins for Seismic 1 equipment have also been observed by others (Ref. 8-10). For the MHTGR, with its high reliance on passive features to accomplish safety functions, this means that extensive NSSS failures are not predicted to occur below 3 g. Structural failures such as failure of the reactor cavity walls or surrounding buildings are expected to require even greater accelerations but were not evaluated in this assessment.

In Fig. 8-11, the sequential failure of selected major equipment in the plant as ground acceleration increases is shown. Within the range of ground accelerations considered, the most severe consequence resulting from these failures was that due to a loss of forced circulation and a small failure in the primary coolant pressure boundary leading to a depressurized conduction

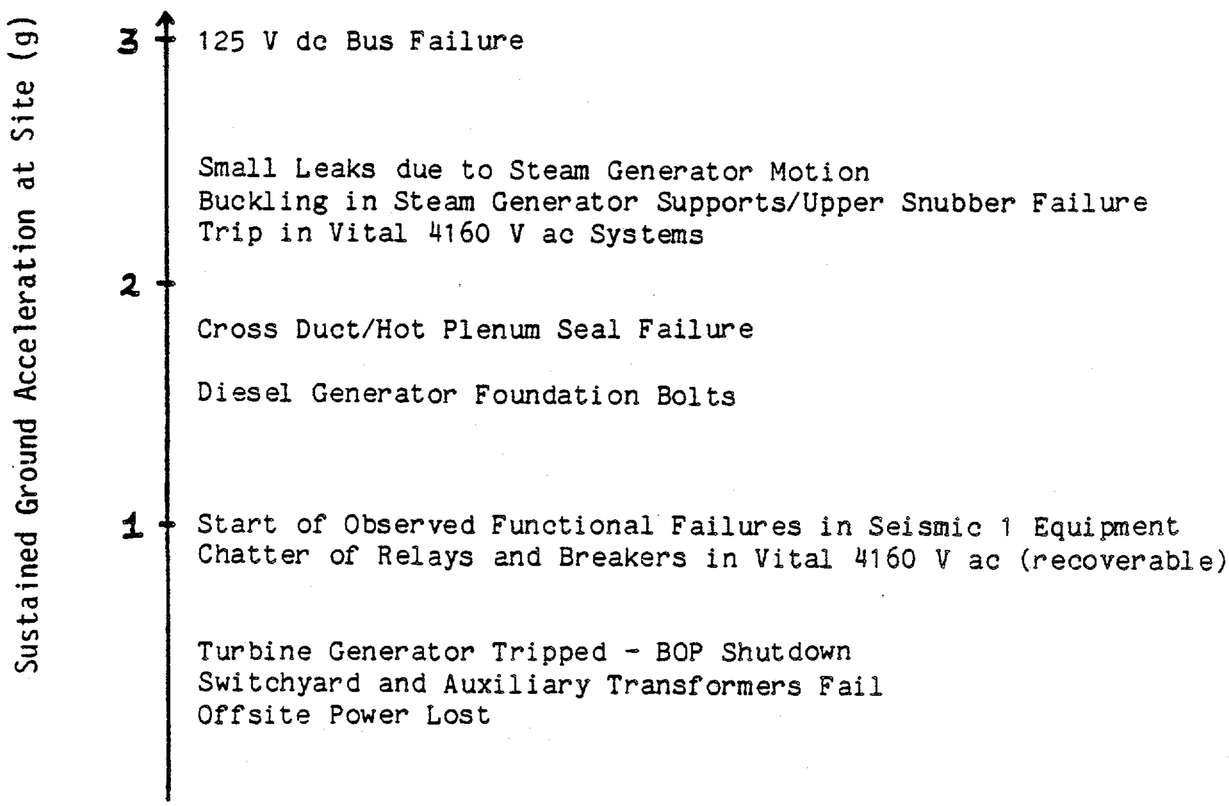


Fig. 8-11. Selected Equipment Response to Increasing Ground Acceleration

cooldown. Specifically, the failure scenario leading to this release is as follows:

Due to failures in the 4160 V ac distribution system, forced circulation from either the main or shutdown cooling system is lost for seismic events larger than about 1.5 g. At about 0.15 g, the reactor is assumed to be manually tripped, and the power plant shut down in an effort to limit equipment damage from accelerations greater than the operating basis earthquake (OBE). Offsite power connections are lost at approximately 0.5 g, necessitating operation of the diesel generator set. However, for accelerations greater than about 1.5 g, diesel generator foundation bolting fails and this last available power source is lost. Also, at about this same g loading it has been estimated that the coaxial cross duct connection to the lower core plenum fails. Such a failure would allow coolant flow to bypass the core and reduce or preclude further forced convection through the core. As acceleration becomes greater, further damage occurs. Scaling from Ref. 8-8, it is judged that steam generator support buckling and snubber failure occur at 2.55 g. The resultant vessel motion is assumed to lead to a small primary coolant leak such as that resulting from a failed, unisolable instrument line. Fragility curves for instrument line and diesel generator failure, typifying the failure modeling, are shown in Figs. 8-12 and 8-13. Further accelerations up to 3 g are not predicted to cause further damage.

Three event sequences have been identified that result in depressurized conduction cooldowns initiated by seismic activity and have been labeled CC_e-1, CC_e-2, and CC_e-3 as indicated in Fig. 7-8. For all release categories, a 0.3 in.^{2} nominal instrument line failure was assumed to occur in the affected modules. Forced convection core cooling is lost in all event sequences.

Release category CC_e-1 involves one affected module in which RCCS cooling functions properly. This event was previously analyzed in Section 8.4.2.1 for primary coolant leak initiated conduction cooldowns. The consequences for CC_e-1 have been approximated as being similar to those for CC_p-9 which encompasses leaks in the range of 0.03 to 1 in.^{2}.

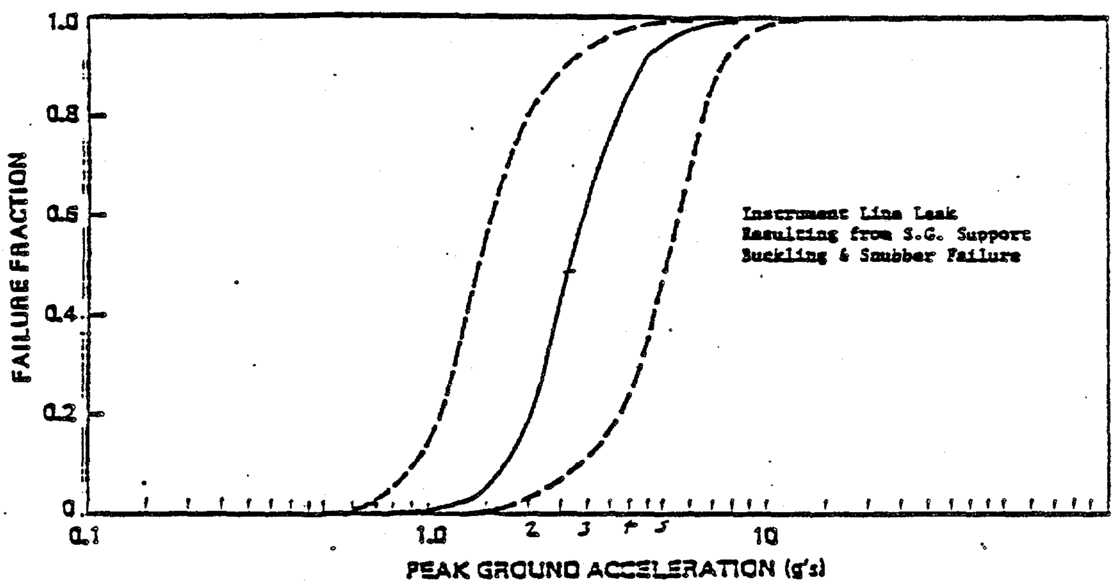


Figure 8-12. Fragility Model for Instrument Line Failure

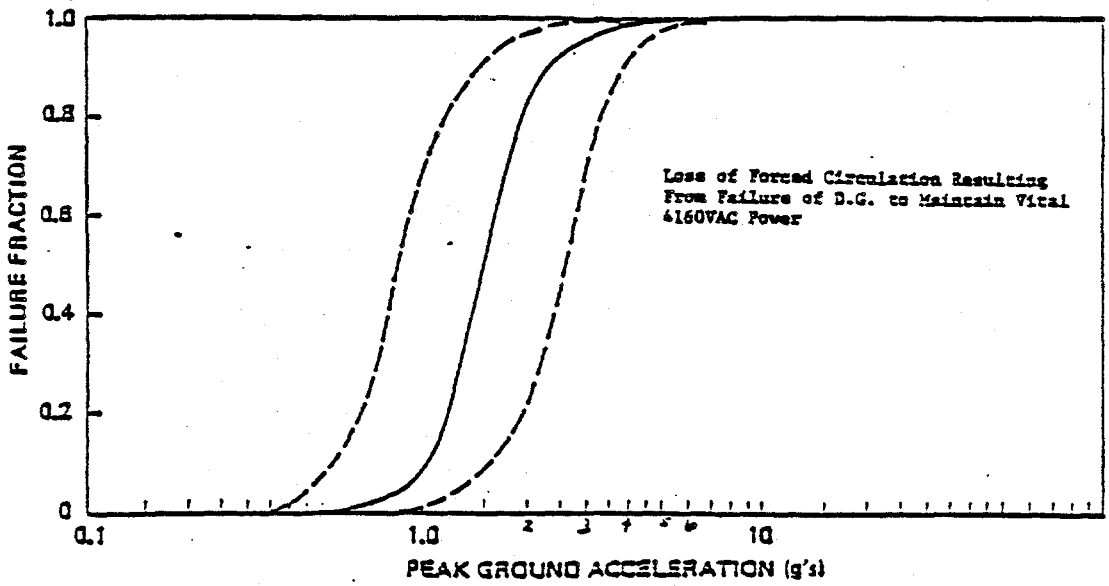


Figure 8-13. Fragility Model for Diesel Generator Failure

The event sequence leading to release category CC_e-2 is the same as given for CC_e-1, except all four modules are affected. The consequences of CC_e-2 are therefore four times those of CC_e-1.

Release category CC_e-3 is the final category under consideration in this section. In this event sequence, reactor trip of all modules is unsuccessful. In addition, because of the severity of the initiating event, SCS and RCCS cooling are unavailable. The thermal transient experienced by the core results in temperatures in excess of a normal conduction cooldown transient because of the lack of reactivity insertion and RCCS cooling. Consequences for this event sequence have been estimated, based on prior analyses for loss of RCCS cooling and loss of reactivity systems. Fission product release from these events was compared to the fission product release from categories with known offsite doses and scaled accordingly to obtain offsite doses for CC_e-3.

8.4.2.5 Control Rod Bank Withdrawal Initiated Conduction Cooldowns. Two accident families have been identified in the Section 7.3.6 frequency assessment that result in conduction cooldowns initiated by a control rod bank withdrawal. The accident families have been designated CC_w-1 and CC_w-2.

The normal system response to a spuriously withdrawn rod bank is to initiate a reactor trip on the outer control rods following a high power to flow signal to the PPIS. Core cooling by the HTS is expected to continue. Significant fuel failure does not occur in this event due to the insertion of control rods shortly after the initiation of the transient and continued operation of the HTS.

Failure to trip the reactor by control rod insertion results in two event sequences where fission product release occurs with a nonnegligible frequency. The withdrawal of a control rod bank causes a rapid increase in core power in a very short time period along with a rapid temperature increase. The HTS continues to operate at a level commensurate with the new core power level attained by the rod bank withdrawal and transports excessively hot helium gas to the steam generator and subsequently overheats the steam generator tubes in a matter of a few hundred seconds. Multiple steam generator tube failures near the bimetallic weld in the superheater region present the potential for a severe moisture ingress into the primary system. Isolation of the steam generator is accomplished when moisture levels increase rapidly in the primary system and are detected by the moisture monitors. Concurrent with isolation, the main circulator is tripped, and the SCS is started as designed to remove core heat. Because of high core temperatures, the SCS is unable to perform its function and boiling in the shutdown heat exchanger is initiated. Subsequent to detecting the boiling condition, the SCS is tripped, precluding exposure of the primary coolant boundary to excessively high temperatures. The RCCS continues to operate, maintaining vessel temperatures at acceptable limits. The primary relief train valve setpoint is reached in approximately 15 min and lifts to relieve the excessive pressure buildup due to the water ingress resulting from the steam generator damage.

Accident family CC_W-1 results from the above described conditions with the added failure of the relief valve to reseat. The thermal transient experienced by the core following relief valve failure is expected to approximate that for a depressurized conduction cooldown as depicted in Fig. 8-6. Prior to relief valve failure, the temperature transient is as shown in Fig. 8-14 which is similar to a pressurized conduction cooldown transient. The reactor is ultimately tripped before approximately one day has expired and the effects of Xe-135 poisoning are diminished. The ingress of moisture is not expected to significantly hydrolyze failed fuel because of the excessive temperatures and preferential oxidation of graphite; therefore, the consequences for this accident family are dominated by the release of circulating activity, liftoff of plated-out material, and releases from the fuel due to thermal effects. Approximations have been made to determine the offsite dose consequences based on the temperatures given in Fig. 8-14 and the results of previously analyzed pressurized conduction cooldown accident scenarios.

Accident family CC_W-2 is identical to CC_W-1 with the exception that the relief valve successfully recloses following the initial pressure relief. The relief valve reclosure setpoint has been assumed to be 885 psia, a value 15% below the opening setpoint. The probability that the valve will open a second

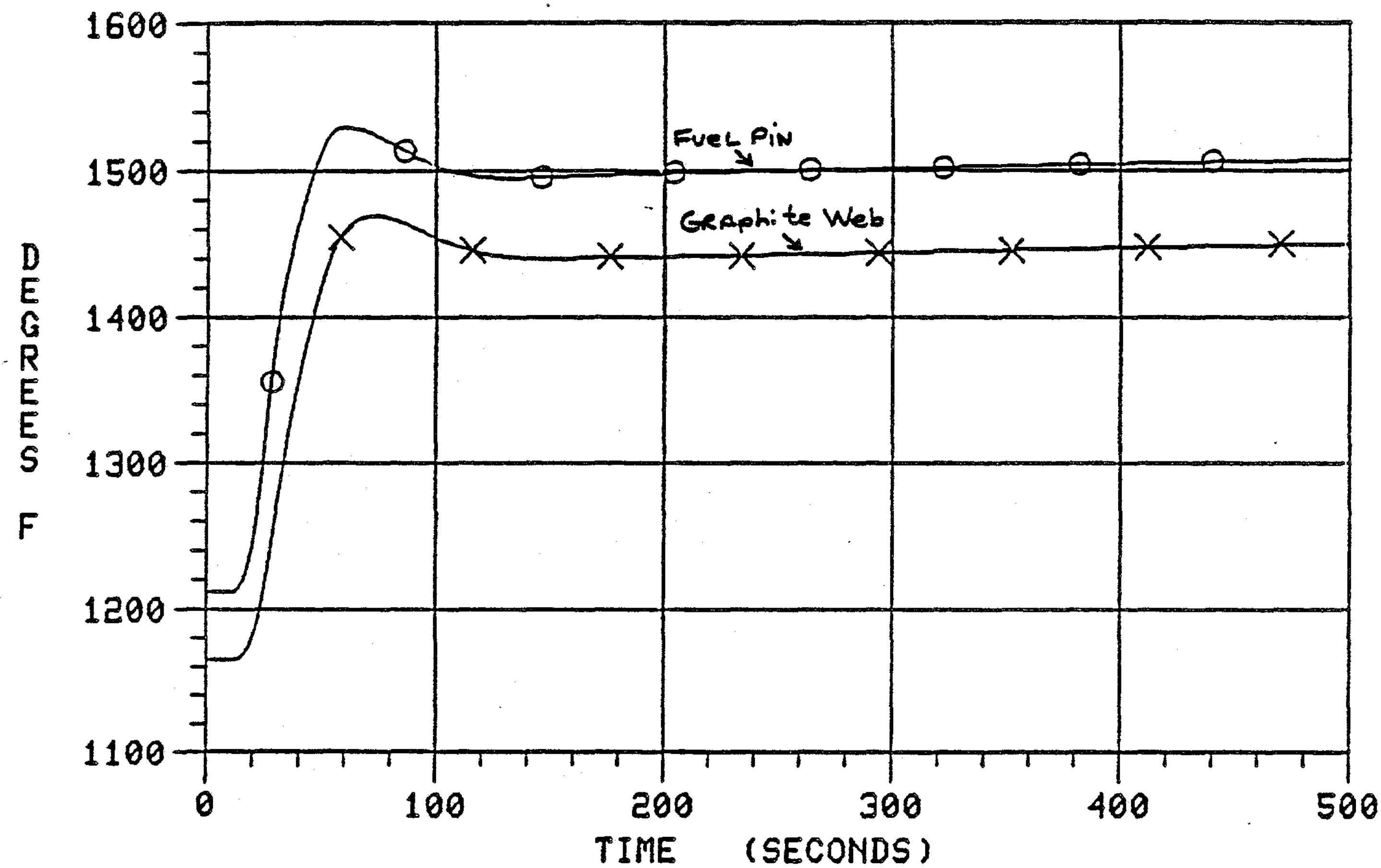


Figure 8-14. Temperature Transients During a Pressurized Conduction Cooldown with Control Rod Bank Withdrawal

time in this event sequence before the reactor is tripped is less than 10^{-9} per plant year and has been truncated in the event tree of Fig. 7-6. The consequences of CC_W-2 are approximated as being 15% of those for CC_W-1.

8.4.3 Uncertainty Analysis

The consequence uncertainty model used for conduction cooldown transients is the same as the model given in Section 8.1.3 for primary coolant leak initiated events. The uncertainty distributions for conduction cooldowns as given in Ref. 8-2 were modified by replacing the median consequences for the Ref. 8-2 assessment by the current assessment results. For those event sequences that were not previously analyzed in Ref. 8-2, uncertainty distributions for similar events that were analyzed in Ref. 8-2 were applied.

8.5 Earthquake Consequences

The earthquake frequency assessment of Section 7.3.8 identified one accident family as shown in Fig. 7-8 which results in an offsite dose to the public when forced convection core cooling is available. The release category belonging to this accident family is labeled EQ-1. Two accident families were identified which result in no offsite doses at lower seismic intensity ranges where no seismic-induced failures are expected to occur with the exception of loss of HTS cooling. These last two release categories will not be discussed here as they have no impact on the safety risk of the MHTGR.

8.5.1 Data Base

The data base for earthquake consequence analysis is outlined in Section 8.4.2.4 for earthquake initiated conduction cooldowns and Section 8.1 for primary coolant leaks.

8.5.2 Physical Phenomena

The release category under consideration in this section encompasses seismic intensities greater than or equal to a 0.4 g ground acceleration. As indicated in Section 8.4.2.4, large ground accelerations are expected to engender failures precluding the operation of the HTS. The event sequence describing EQ-1, therefore, does not consider the availability of the HTS. The seismic event does not, however, result in the loss of SCS cooling. Due to the large intensity of the earthquake, instrument line failure is assumed to occur in all four modules resulting in leakage of primary coolant from the reactor vessels. The size of the leakage area was assumed to correspond to that of a nominal instrument line which is on the order of 0.3 $in.^2$. The consequences of this event are approximated by release category PC-6 which is described in Section 8.1.2 for primary coolant leaks. The PC-6 consequences are multiplied by four to account for leakage in all four modules.

8.5.3 Uncertainty Analysis

The consequence uncertainty model used for EQ-1 is given in Section 8.1.3 for primary coolant leak initiated events. The Ref. 8-2 uncertainty distribution for PC-6 was modified by replacing the median consequences calculated for the Ref. 8-2 assessment by the consequences calculated for category EQ-1.

8.6 References

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8-2. Everline, C. J., and S. B. Inamati, "Safety Risk Assessment of 250 MW(t) Side-by-Side Modular HTGR Plant," HTGR-85-097 (908246/0), August 1985.

8-3. Everline, C. J., et al., "Probabilistic Risk Assessment of the Modular HTGR Plant," HTGR-86-011 (908664/0), January 1986.

8-4. Slade, D. H., ed., Meteorology and Atomic Energy 1968, USAEC, 1968.

8-5. Project Staff, "HTGR Accident Initiation and Progression Analysis Status Report - Phase II Risk Assessment," GA-A15000, April 1978.

8-6. "TAC2D - A General Purpose Two-Dimensional Heat Transfer Computer Code," GA Document GA-A14032, July 1976.

8-7. "SORS - Computer Program for Analyzing Fission Product Release from HTGR Cores During Transient Temperature Excursions," GA-A12462, April 1974.

8-8. Project Staff, Pickard, Lowe, and Garrick Inc., "Zion Probabilistic Safety Study." Prepared for Commonwealth Edison Co., 1981.

8-9. Project Staff, Pickard, Lowe, and Garrick Inc., "Seabrook Station Probabilistic Safety Assessment." Prepared for Public Service Co. of New Hampshire and Yankee Atomic Electric Co., 1983.

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9. RISK ASSESSMENT RESULTS

Accident frequencies and their uncertainties from Section 7 are combined with the consequences and their uncertainties from Section 8 to render complementary cumulative risk envelopes, point estimates of mean frequency and consequence, and mean risk estimates. Risk envelopes for whole body gamma and thyroid doses at the EAB are developed in Section 9.1. Section 9.2 presents point value plots of mean frequency and consequence for all accident families that result in an offsite dose. The mean risk of acute and latent fatalities is the topic of Section 9.3.

9.1 Risk Envelopes

Figures 9-1 through 9-8 display the MHTGR risk envelopes. Figures 9-1 and 9-2 are for whole body gamma and thyroid doses resulting from primary coolant leaks. The frequency for primary coolant leaks is approximately 0.1 per plant year as cited in Section 7.3.1. At high frequencies, Fig. 9-1 indicates that the accidents belonging to release category PC-10 dominate the overall risk envelope for whole body gamma doses, while PC-6 governs at lower frequencies. As described in Section 3, event sequences leading to similar consequences are customarily grouped into release categories for convenience. Table 9-1 defines the release categories for all primary coolant leak initiated accidents (including those that do not contribute appreciably to the risk envelope).

The risk envelope for thyroid doses caused by primary coolant leaks is dominated by PC-4 at higher consequences. According to Table 9-1, PC-4 is initiated by primary coolant depressurization through an area, A, between 1 and 13 in.^{2}. Falling within this range of sizes is the area corresponding to that of one primary relief train. As depicted in Fig. 9-2, the lower bound frequency of 5×10^{-7} per plant year is crossed at approximately 0.3 rem, well below the PAG limit of 5 rem. Extremely large leak areas associated with vessel ruptures, such as in PC-1, have been judged to not contribute appreciably to the

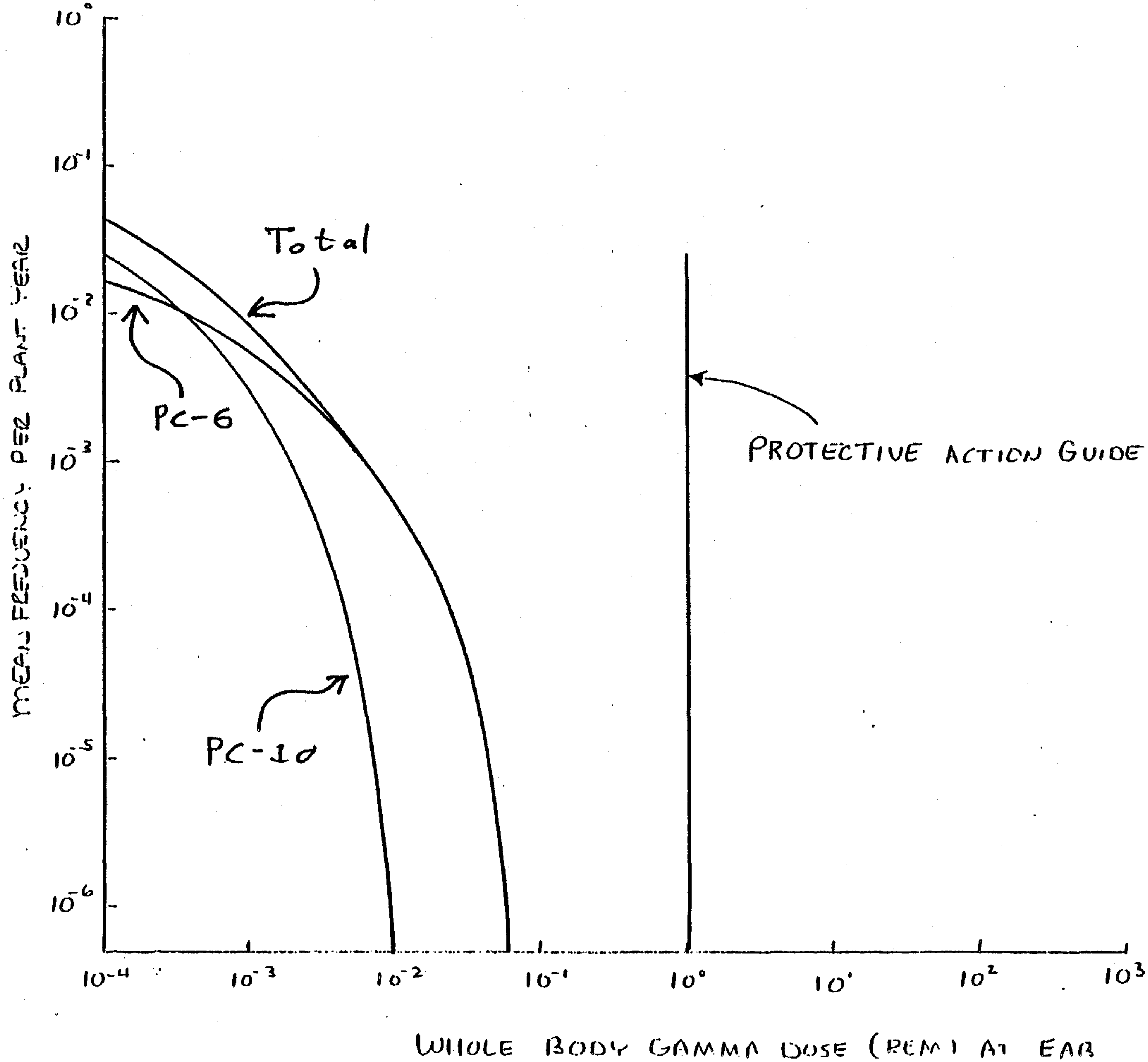

Fig. 9-1. Risk Envelope for Whole Body Gamma Doses Resulting from Primary Coolant Leaks

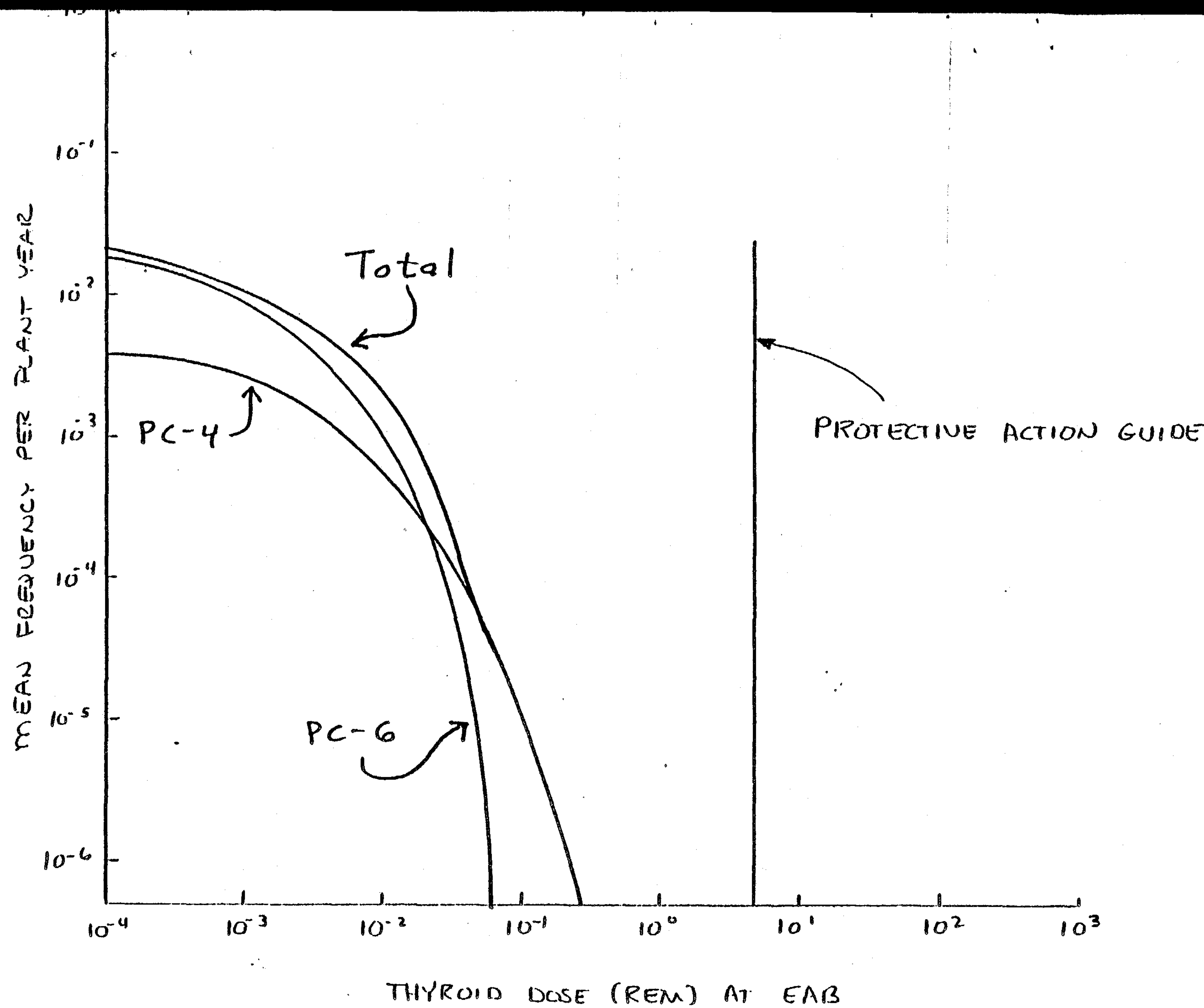

Fig. 9-2. Risk Envelope for Thyroid Doses Resulting from Primary Coolant Leaks

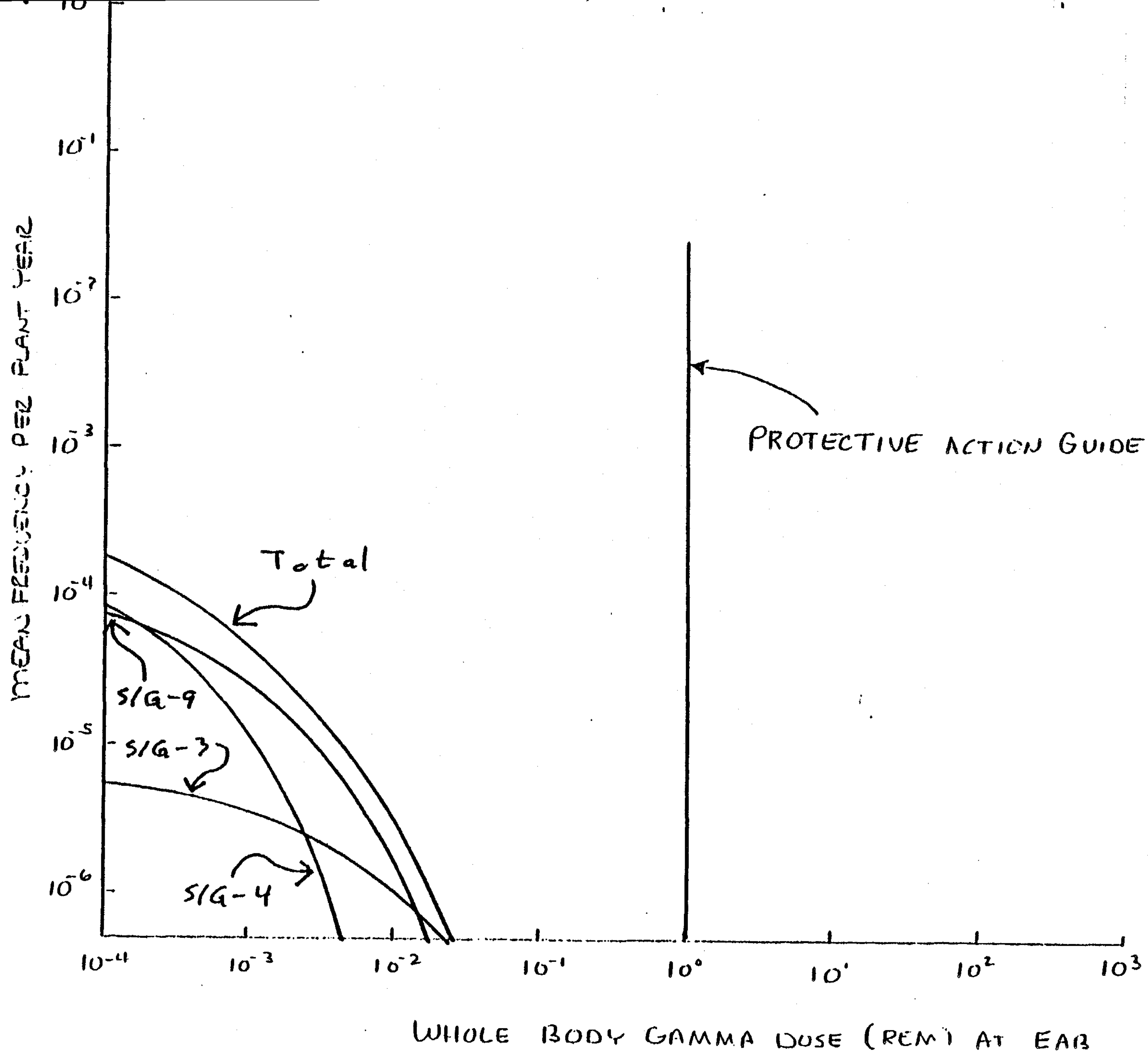

Fig. 9-3. Risk Envelope for Whole Body Gamma Doses Resulting from Steam Generator Leaks

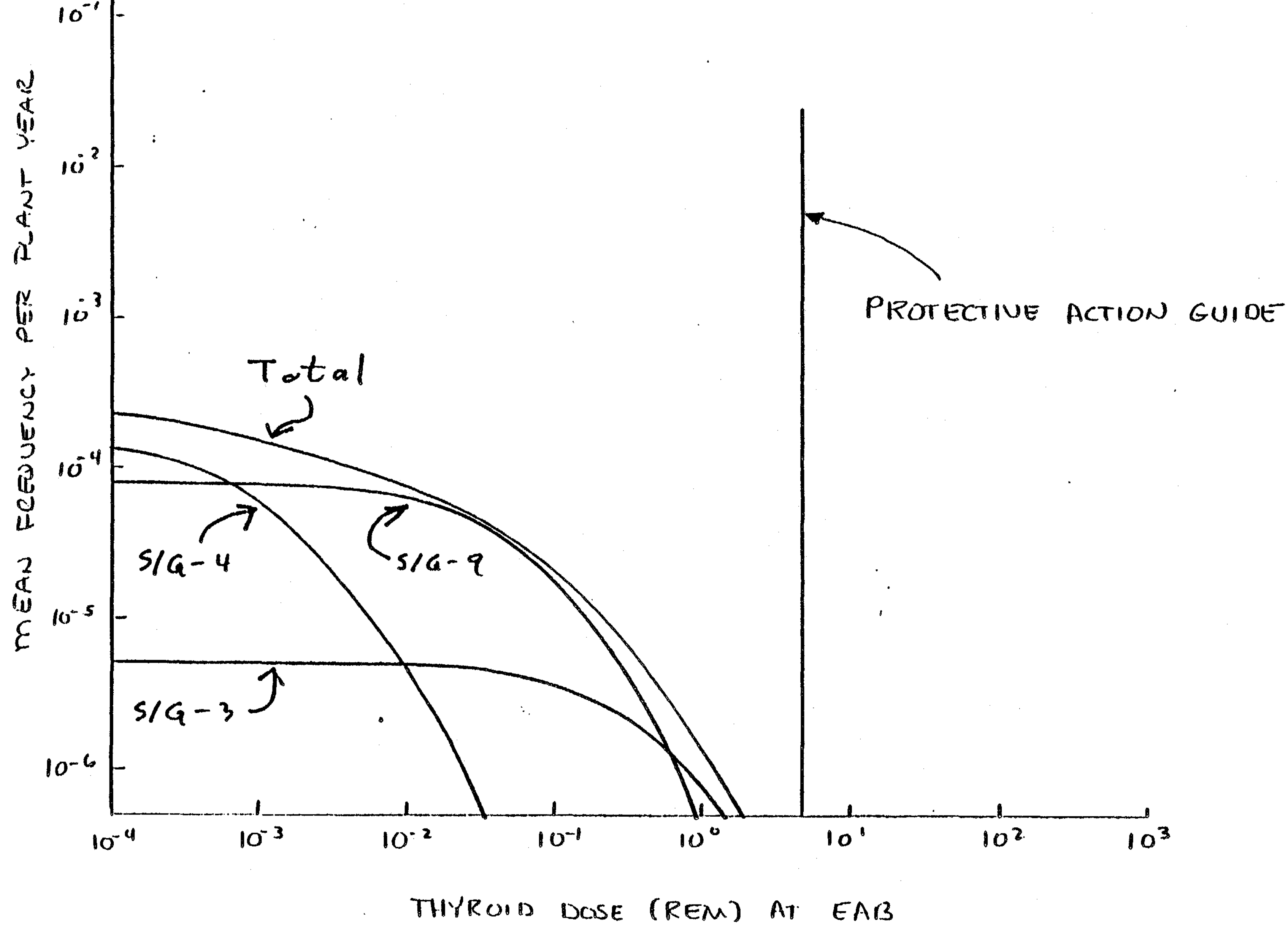

Fig. 9-4. Risk Envelope for Thyroid Doses Resulting from Steam Generator Leaks

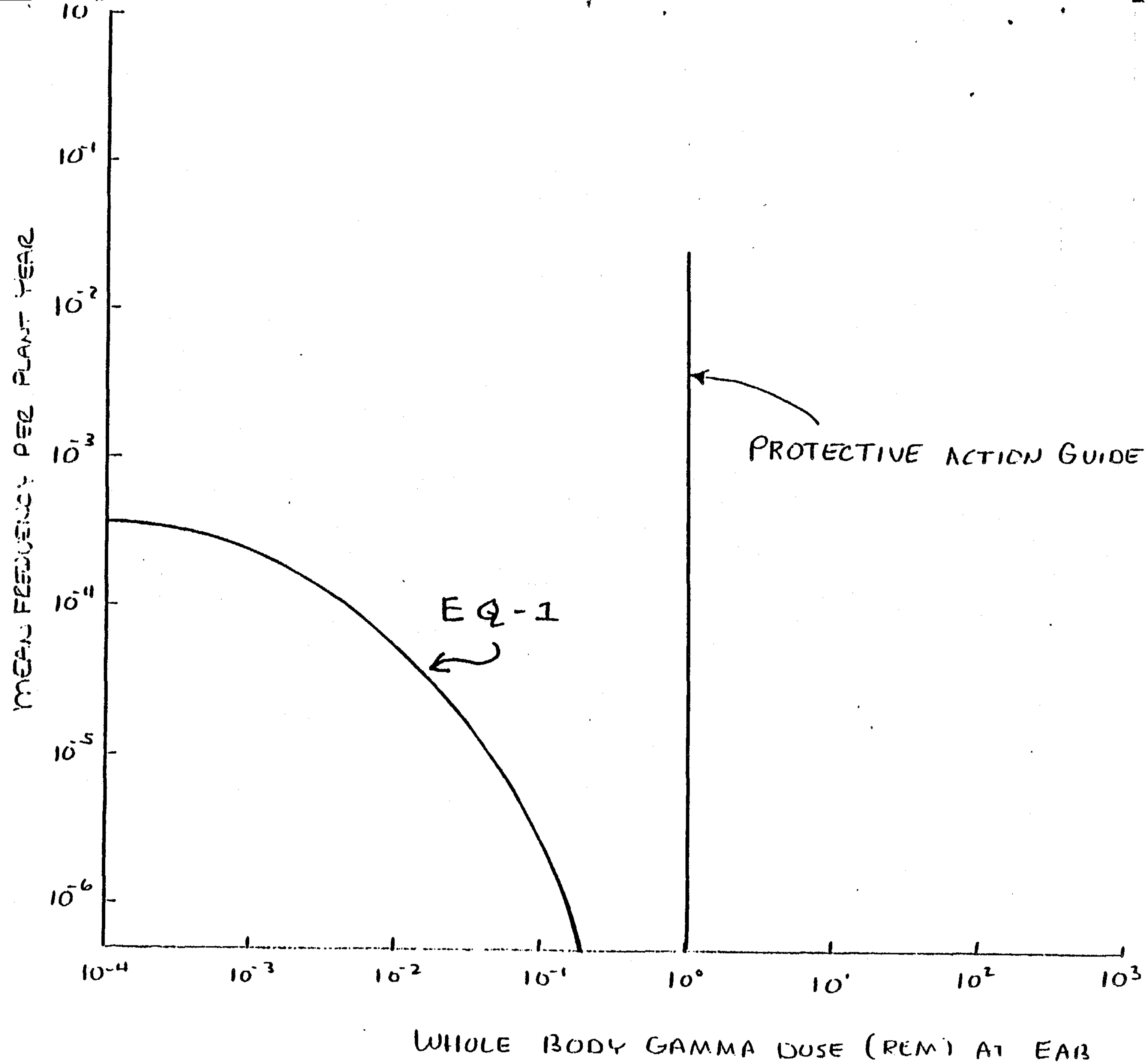

Fig. 9-5. Risk Envelope for Whole Body Gamma Doses Resulting from Earthquakes

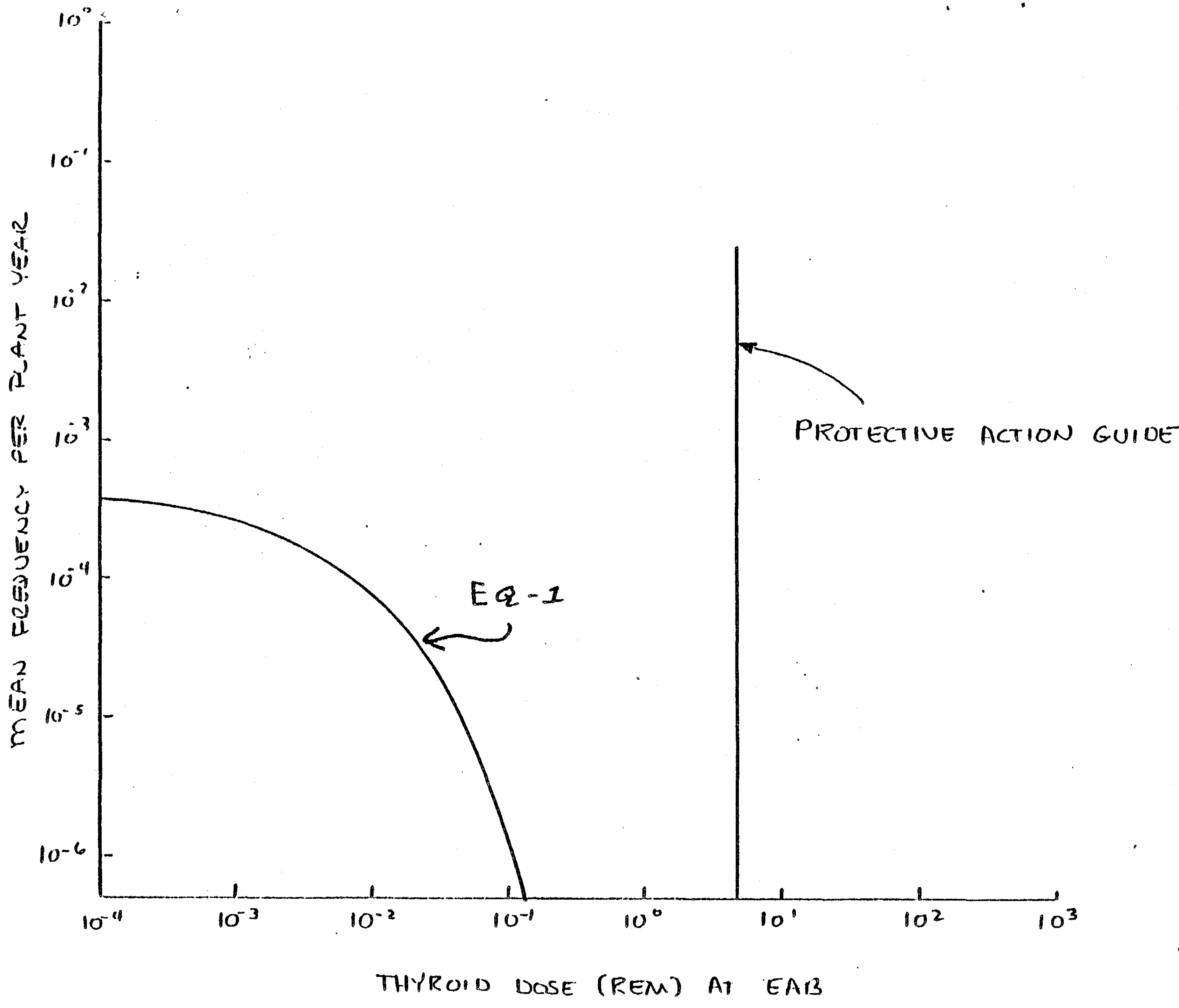

Fig. 9-6. Risk Envelope for Thyroid Doses Resulting from Earthquakes

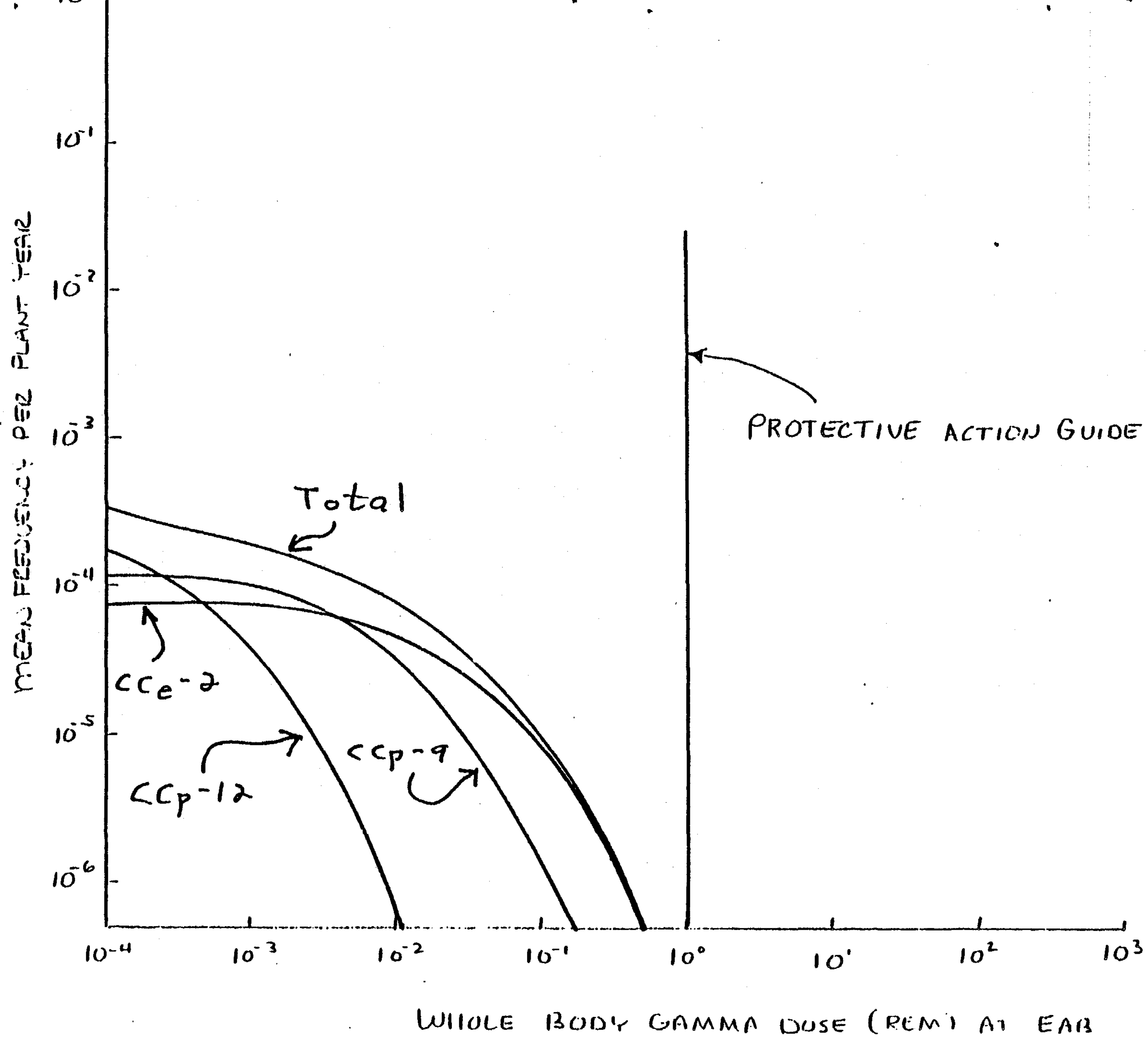

Fig. 9-7. Risk Envelope for Whole Body Gamma Doses Resulting from Conduction Cooldowns

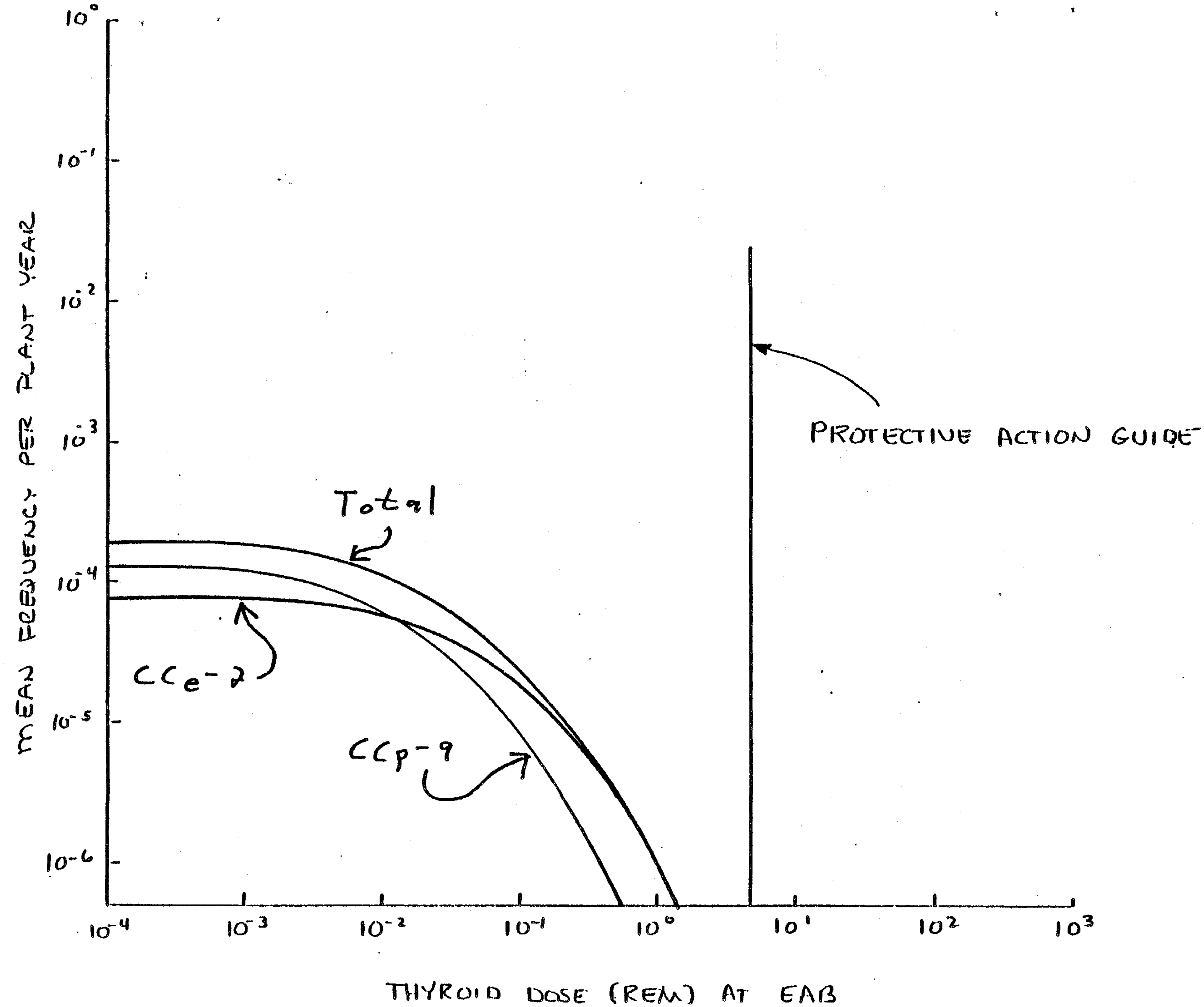

Fig. 9-8. Risk Envelope for Thyroid Doses Resulting from Conduction Cooldowns

TABLE 9-1
PRIMARY COOLANT LEAK INITIATED RELEASE CATEGORY DESIGNATIONS

Release Category	Description
PC-1	Major primary coolant leak where $A \geq 550$ in.^{2} occurs. HTS or SCS core cooling is maintained. HPS pumpdown and reactor building are ineffective. Release is through the dampers.
PC-2	Very large primary coolant leak where 30 in.$^2 \leq A <$ 550 in.^{2} occurs. HTS or SCS core cooling is maintained. HPS pumpdown and reactor building are ineffective. Release is through the dampers.
PC-3	Large primary coolant leak where 13 in.$^2 \leq A <$ 30 in.^{2} occurs. HTS or SCS core cooling is maintained. HPS pumpdown and reactor building are ineffective. Release is through the dampers.
PC-4	Moderate primary coolant leak where 1 in.$^2 \leq A <$ 13 in.^{2} occurs. HTS or SCS core cooling is maintained. HPS pumpdown and reactor building are ineffective. Release is through the dampers.
PC-5	Small primary coolant leak where 3×10^{-2} in.$^2 \leq A <$ 1 in.^{2} occurs. HTS or SCS core cooling is maintained. Reactor building fans fail to disengage, filters are not isolated. Release is through the dampers.
PC-6	Small primary coolant leak where 3×10^{2} in.$^2 \leq A <$ 1 in.^{2} occurs. HTS or SCS core cooling is maintained. HPS pumpdown occurs. Reactor building functions properly. Release is through the dampers.
PC-7	Small primary coolant leak where 2×10^{-3} in.$^2 \leq A < 3 \times 10^{-2}$ in.^{2} occurs. HTS or SCS core cooling is maintained. HPS pumpdown fails. Reactor building fans fail to disengage, filters are not isolated. Release is through the dampers.
PC-8	Very small primary coolant leak where 2×10^{-3} in.$^2 \leq A < 3 \times 10^{-2}$ in.^{2} occurs. HTS or SCS core cooling is maintained. HPS pumpdown occurs. Reactor building fans fail to disengage, filters are not isolated. Release is through the dampers.

TABLE 9-1
(Continued)

Release Category	Description
PC-9	Very small primary coolant leak where 3×10^{-5} in.^{2} $\leq A < 3 \times 10^{-2}$ in.^{2} occurs. HTS or SCS core cooling is maintained. HPS pumpdown fails. Reactor building functions properly. Release is initially through the dampers. Subsequent release is by building leakage.
PC-10	Very small primary coolant leak where 3×10^{-5} in.^{2} $\leq A < 3 \times 10^{-2}$ in.^{2} occurs. HTS or SCS core cooling is maintained. HPS pumpdown fails. Reactor building functions properly. Release is initially through the dampers. Subsequent release is by building leakage.
CC_p-2	Very small primary coolant leak where 2×10^{-3} $\leq A < 3 \times 10^{-2}$ in.^{2} occurs. HTS and SCS core cooling fail, RCCS cooling succeeds. HPS pumpdown occurs. Reactor building fans fail to disengage, filters are not isolated. Release is through the dampers.
CC_p-4	Very small primary coolant leak where 2×10^{-3} $\leq A < 3 \times 10^{-2}$ in.^{2} occurs. HTS and SCS core cooling fail, RCCS cooling succeeds. HPS pumpdown fails. Reactor building functions properly. Release is initially through the dampers. Subsequent release is by building leakage.
CC_p-6	Very large primary coolant leak where 13 in.^{2} $\leq A < 30$ in.^{2} occurs. HTS and SCS core cooling fail, RCCS cooling succeeds. HPS pumpdown and reactor building are ineffective. Release is through the dampers.
CC_p-7	Very small primary coolant leak where 2×10^{-3} $\leq A < 3 \times 10^{-2}$ in.^{2} occurs. HTS and SCS core cooling fail, RCCS cooling succeeds. HPS pumpdown occurs. Reactor building functions properly. Release is initially through the dampers. Subsequent release is by building leakage.

TABLE 9-1
(Continued)

Release Category	Description
CC_p-8	Large primary coolant leak where 1 in.^{2} $\leq$ A < 13 in.^{2} occurs. HTS and SCS core cooling fail, RCCS cooling succeeds. HPS pumpdown and reactor building are ineffective. Release is through the dampers.
CC_p-9	Moderate primary coolant leak where $3 \times 10^{-2} \leq A < 1$ in.^{2} occurs. HTS and SCS core cooling fail, RCCS cooling succeeds. Reactor building functions properly. Release is through the dampers.
CC_p-10	Moderate primary coolant leak where 3×10^{-2} in.^{2} $\leq A < 1$ in.^{2} occurs. HTS and SCS core cooling fail, RCCS cooling succeeds. HPS pumpdown occurs. Reactor building fans fail to disengage, filters are not isolated. Release is through the dampers.
CC_p-11	Very small primary coolant leak where 3×10^{-5} in.^{2} $\leq A < 2 \times 10^{-3}$ in.^{2} occurs. HTS and SCS core cooling fail, RCCS cooling succeeds. HPS pumpdown fails. Release is by leakage through the building.
CC_p-12	Very small primary coolant leak where 3×10^{-5} in.^{2} $\leq A < 2 \times 10^{-3}$ in.^{2} occurs. HTS and SCS core cooling fail, RCCS cooling succeeds. HPS pumpdown occurs. Release is by leakage through the building.

primary coolant leak risk envelope. The frequency of such events has been estimated to be approximately 2×10^{-6} per plant year. This frequency estimate is predicated upon data in Refs. 9-1 and 9-2. The present assessment for such large leak sizes is preliminary, requiring more detailed frequency and consequence analyses to fully quantify the risk from this type of accident.

Figures 9-3 and 9-4 are the risk envelopes for whole body gamma and thyroid doses resulting from both small and moderate steam generator leaks. Release categories for steam generator leak initiated accidents are defined in Tables 9-2 and 9-3. Three accident families, S/G-3, S/G-4, and S/G-9 dominate both the whole body gamma and thyroid risk envelopes. S/G-4 is a small steam generator leak and tends to dominate at higher frequencies. At higher consequences, S/G-3 and S/G-9, which are moderate steam generator leaks, tend to dominate.

Risk envelopes for whole body gamma and thyroid doses resulting from earthquake initiated events are shown in Figs. 9-5 and 9-6. Only one accident family was identified that resulted in an offsite dose and is designated EQ-1. It is therefore the only contributor to both risk envelopes. As shown in the figures, both PAG dose limits are met with margin at the lower bound frequency of 5×10^{-7} per plant year. Release category EQ-1 is described in Table 9-4.

Figures 9-7 and 9-8 depict the risk envelopes for whole body gamma and thyroid doses resulting from conduction cooldown accidents. Conduction cooldowns initiated by primary coolant leaks and seismic activity dominate both risk envelopes. These release categories are described in Tables 9-1 and 9-4, respectively. Release category CC_e-2 dominates at high consequences for both whole body gamma and thyroid doses. The seismic activity initiating the event sequence of CC_e-2 is of such a magnitude that instrument line failure occurs in all four modules. Consequences of this family are, therefore, high relative to other accident families identified which generally involve only one affected module. For example, CC_p-9 is initiated by primary coolant leakage through an area comparable to that in CC_e-2, but only one module is affected in the event sequence. The whole body gamma risk envelope includes an additional release

TABLE 9-2
SMALL STEAM GENERATOR LEAK INITIATED RELEASE CATEGORY DESIGNATIONS

Release Category	Description
SG-N1	Small steam generator tube leak occurs. Moisture monitors detect ingress. Automatic steam generator isolation succeeds. Automatic steam generator dump succeeds. SCS core cooling is maintained. No release occurs.
SG-N3	Small steam generator tube leak occurs. Moisture monitor detection fails. Reactor trip occurs in less than 3.5 h. Operator intervention initiates main circulator trip. Isolation and dump succeed. SCS core cooling succeeds. No release occurs.
SG-N5	Small steam generator tube leak occurs. Moisture monitors detect ingress. Automatic steam generator isolation succeeds. Steam generator dump system valves fail to open. SCS core cooling is maintained. No release occurs.
SG-N6	Small steam generator tube leak occurs. Moisture monitors detect ingress. Isolation of feedwater fails. Steam generator relief train opens and successfully recloses. SCS core cooling is maintained. Operator intervention succeeds in terminating feedwater flow. Insignificant release occurs.
SG-4	Small steam generator tube leak occurs. Moisture monitors detect ingress. Automatic steam generator isolation succeeds. Dump system valves do not successfully reclose. SCS core cooling is maintained. Release path is through the dump system tank relief valves.
SG-5	Small steam generator tube leak occurs. Moisture monitor detection fails. Reactor trip occurs on high primary coolant pressure. Primary relief valve opens and fails to reclose. SCS core cooling is maintained. Release path is through the reactor building dampers.

TABLE 9-2
(Continued)

Release Category	Description
SG-7	Small steam generator tube leak occurs. Moisture monitors detect ingress. Isolation of feedwater fails. Steam generator relief valve opens and fails to reclose. SCS core cooling is maintained. Operator intervention succeeds in terminating the feedwater flow. Release path is directly to the atmosphere via the failed valve.
SG-12	Small steam generator tube leak occurs. Moisture monitor detection fails. Reactor trip occurs on high primary coolant pressure. Primary relief valve opens and successfully recloses. SCS core cooling is maintained. Release path is through the reactor building dampers.
SG-13	Small steam generator tube leak occurs. Moisture monitors detect the ingress. Isolation of feedwater fails. Steam generator relief valve fails to open. Multiple steam generator tube failures occur. Primary relief valve opens and remains open. Operator intervention succeeds in terminating the feedwater flow. Release path is through the reactor building dampers.
CC_s-N1	Small steam generator tube leak occurs. Moisture monitors detect the ingress. Automatic steam generator isolation succeeds. Automatic steam generator dump succeeds. SCS core cooling fails, RCCS cooling succeeds. No release occurs.
CC_s-N3	Small steam generator tube leak occurs. Moisture monitors detect the ingress. Automatic steam generator isolation succeeds. Automatic steam generator dump succeeds. SCS core cooling fails, RCCS cooling fails. Pumpdown of the primary coolant by the HPS succeeds. No release occurs.
CC_s-3	Small steam generator tube leak occurs. Moisture monitors detect the ingress. Isolation of feedwater fails. Steam generator relief valve opens and fails to reclose. SCS core cooling fails, RCCS cooling succeeds. Primary relief valve opens and successfully recloses. Partial release is directly to the atmosphere via the open valve. Partial release is through the reactor building dampers.

TABLE 9-2
(Continued)

Release Category	Description
CC_s-4	Small steam generator tube leak occurs. Moisture monitor detection fails. Reactor trip occurs in less than 3.5 h. Operator intervention initiates main circulator trip. Isolation and dump succeed. SCS cooling fails, RCCS cooling succeeds. Primary relief valve opens and successfully recloses. Release is through the reactor building dampers.
CC_s-5	Small steam generator tube leak occurs. Moisture monitors detect the ingress. Automatic steam generator isolation succeeds. Steam generator dump system valves fail to open. SCS cooling fails, RCCS cooling succeeds. Primary relief valve opens and fails to reclose. Release is through the reactor building dampers.
CC_s-6	Small steam generator tube leak occurs. Moisture monitor detection fails. Reactor trip occurs in less than 3.5 h. Operator intervention initiates main circulator trip. Isolation and dump succeed. SCS core cooling fails, RCCS cooling succeeds. Primary relief valve opens and fails to reclose. Release is through the reactor building dampers.
CC_s-7	Small steam generator tube leak occurs. Moisture monitors detect the ingress. Automatic steam generator isolation succeeds. Steam generator dump system valves fail to open. SCS cooling fails, RCCS cooling succeeds. Primary relief valve opens and successfully recloses. Release is through the reactor building dampers.
CC_s-10	Small steam generator tube leak occurs. Moisture monitors detect the ingress. Automatic steam generator isolation succeeds. Steam generator dump system valves fail to reclose. SCS cooling fails, RCCS cooling succeeds. Release path is to the atmosphere via the open dump system.

TABLE 9-3
MODERATE STEAM GENERATOR LEAK INITIATED RELEASE CATEGORY DESIGNATIONS

Release Category	Description
SG-N2	Moderate steam generator tube leak occurs. Moisture monitors detect ingress. Automatic steam generator isolation succeeds. Automatic steam generator dump succeeds. SCS core cooling is maintained. No release occurs.
SG-1	Moderate steam generator tube leak occurs. Moisture monitors detect ingress. Steam side isolation fails. SCS core cooling succeeds. Primary relief train valve opens and fails to reclose. Release is through the reactor building dampers.
SG-2	Moderate steam generator tube leak occurs. Moisture monitors detect ingress. Automatic steam generator isolation succeeds. Steam generator dump system valves fail to open. SCS core cooling succeeds. Primary relief train valve opens and fails to reclose. Release is through the reactor building dampers.
SG-3	Moderate steam generator tube leak occurs. Moisture monitor detection fails. Reactor trip occurs on high primary coolant pressure. Main circulator trip occurs. Operator intervention succeeds in isolating the steam generator. SCS core cooling succeeds. Primary relief train valve opens and fails to reclose. Release is through the reactor building dampers.
SG-6	Moderate steam generator leak occurs. Moisture monitors detect ingress. Steam side isolation fails. SCS core cooling is maintained. Primary relief train valve opens and successfully recloses. Release is through the reactor building dampers.
SG-8	Moderate steam generator leak occurs. Moisture monitors detect ingress. Automatic steam generator isolation succeeds. Steam generator dump system valves fail to reclose. SCS core cooling is maintained. Release is through the dump system tank relief valves.

TABLE 9-3
(Continued)

Release Category	Description
SG-9	Moderate steam generator tube leak occurs. Moisture monitor detection fails. Reactor trip occurs on high primary coolant pressure. Main circulator trips. Operator intervention succeeds in isolating the steam generator. SCS core cooling succeeds. Primary relief train valve opens and successfully recloses. Release is through the reactor building dampers.
SG-10	Moderate steam generator tube leak occurs. Moisture monitors detect ingress. Automatic steam generator isolation succeeds. Steam generator dump system valves fail to open. SCS core cooling is maintained. Primary relief train valve opens and successfully recloses. Release is through the reactor building dampers.
SG-11	Moderate steam generator tube leak occurs. Moisture monitor detection succeeds. Isolation of feedwater fails. Steam generator relief valve opens and fails to reclose. SCS core cooling is maintained. Primary relief train valve opens and successfully recloses. Partial release is directly to the atmosphere via the open valve. Partial release is through the reactor building dampers.
CC_s-N2	Moderate steam generator tube leak occurs. Moisture monitor detection succeeds. Automatic steam generator isolation succeeds. Automatic steam generator dump succeeds. SCS core cooling fails, RCCS cooling succeeds. No release occurs.
CC_s-8	Moderate steam generator tube leak occurs. Moisture monitor detection succeeds. Steam side isolation fails. SCS core cooling fails, RCCS cooling succeeds. Primary relief train valve opens and successfully recloses. Release is through the reactor building dampers.

TABLE 9-3
(Continued)

Release Category	Description
CC_s-9	Moderate steam generator tube leak occurs. Moisture monitor detection succeeds. Automatic steam generator isolation succeeds. Steam generator dump system valves fail to reclose. SCS core cooling fails, RCCS cooling succeeds. Release is through the dump system tank relief valves.
CC_s-11	Moderate steam generator tube leak occurs. Moisture monitor detection succeeds. Automatic steam generator isolation succeeds. Steam generator dump system valves fail to open. SCS core cooling fails, RCCS cooling succeeds. Primary relief train valve opens and fails to reclose. Release is through the reactor building dampers.
CC_s-12	Moderate steam generator tube leak occurs. Moisture monitor detection fails. Reactor trip occurs on high primary coolant pressure. Main circulator trip occurs. Operator intervention succeeds in isolating the steam generator. SCS core cooling fails, RCCS cooling succeeds. Primary relief train valve opens and fails to reclose. Release is through the reactor building dampers.
CC_s-13	Moderate steam generator tube leak occurs. Moisture monitor detection succeeds. Automatic steam generator isolation succeeds. Steam generator dump system valves fail to open. SCS core cooling fails, RCCS cooling succeeds. Primary relief train valve opens and successfully recloses. Release is through the reactor building dampers.
CC_s-14	Moderate steam generator tube leak occurs. Moisture monitor detection fails. Reactor trip occurs on high primary coolant pressure. Main circulator trip occurs. Operator intervention succeeds in isolating the steam generator. SCS core cooling fails, RCCS cooling succeeds. Primary relief train valve opens and fails to reclose. Release is through the reactor building dampers.

TABLE 9-4
EARTHQUAKE INITIATED RELEASE CATEGORY DESIGNATIONS

Release Category	Description
EQ-N1	An earthquake occurs with an intensity between 0.06 and 0.2 g. Reactor trip of all four modules occurs if required. HTS core cooling continues No release occurs.
EQ-N2	An earthquake occurs with an intensity between 0.06 and 0.4 g. Reactor trip of all four modules occurs. HTS cooling is unavailable. SCS core cooling succeeds. No release occurs.
EQ-1	An earthquake occurs with an intensity greater than 0.4 g. Reactor trip of all four modules occurs. HTS cooling is unavailable. SCS cooling succeeds. Instrument line failure causes leakage in all four modules. Release is through the reactor building.
CC_e-N1	An earthquake occurs with an intensity between 0.06 and 0.4 g. Reactor trip occurs in one affected module. HTS cooling is unavailable. SCS cooling fails. RCCS cooling succeeds. No release occurs.
CC_e-N2	An earthquake occurs with an intensity between 0.06 and 0.4 g. Reactor trip of all four modules occurs. HTS cooling is unavailable. SCS cooling fails. RCCS cooling succeeds. No release occurs.
CC_e-1	An earthquake occurs with an intensity greater than 0.4 g. Reactor trip occurs in one affected module. HTS and SCS cooling are unavailable. RCCS cooling succeeds. Instrument line failure causes leakage in the affected module. Release is through the reactor building.

TABLE 9-4
(Continued)

Release Category	Description
CC_e-2	An earthquake occurs with an intensity greater than 0.4 g. Reactor trip of all four modules occurs. HTS and SCS cooling are unavailable. RCCS cooling succeeds. Instrument line failure causes leakage in all four modules. Release is through the reactor building.
CC_e-3	An earthquake occurs with an intensity greater than 0.4 g. Reactor trip of all four modules fails. HTS and SCS cooling are unavailable. RCCS cooling fails. Instrument line failure causes leakage in all four modules. Release is through the reactor building.

category initiated by a primary coolant leak designated CC_p-12. This release category dominates at higher frequencies because of the very small leak area involved. The consequences are, however, small compared to other dominant release categories.

Event sequences leading to conduction cooldowns may be initiated by events other than primary coolant leaks and earthquakes, but do not contribute appreciably to the risk envelopes. Conduction cooldowns initiated by small steam generator leaks, moderate steam generator leaks, and control rod bank withdrawal are described in Tables 9-2, 9-3, and 9-5, respectively.

Event sequences that result in no offsite dose do not contribute to the safety risk of the MHTGR. These sequences are, however, described for licensing purposes in Tables 9-6, 9-7, and 9-8 for events initiated by anticipated transients requiring reactor scram, loss of HTS cooling, and loss of offsite power with inadvertent turbine trip, respectively.

9.2 Point Value Plots

Figures 9-9 and 9-10 present the results of the PRA for the MHTGR in the form of point value risk plots. The mean frequency of each release category is plotted against the mean consequence. Figure 9-9 is for whole body gamma dose, and Fig. 9-10 is for thyroid dose. Included in the figures are points for all release categories identified that have an offsite dose and a mean frequency at or above 5×10^{-7} per plant year.

The intention of the point value plots is not to demonstrate goal compliance which requires the additional consideration of uncertainties in dose and frequency estimates. They are intended, rather, to complement the risk envelopes given in Section 9.1 to present a more clear understanding of the relationships among various release categories as well as to present in graphical form the release categories not pictured on the risk plots.

TABLE 9-5
CONTROL ROD BANK WITHDRAWAL INITIATED RELEASE CATEGORY DESIGNATIONS

Release Category	Description
RW-N1	Spurious control rod bank withdrawal occurs. Primary scram trips the reactor. HTS core cooling is maintained. No release occurs.
RW-N2	Spurious control rod bank withdrawal occurs. Primary scram trips the reactor. HTS core cooling fails. SCS core cooling succeeds. No release occurs.
CC_W-N3	Spurious control rod bank withdrawal occurs. Primary scram trips the reactor. HTS and SCS core cooling fail. RCCS cooling succeeds. No release occurs.
CC_W-1	Spurious control rod bank withdrawal occurs. Primary reactor scram fails. HTS cooling is ineffective . RCCS cooling succeeds. Multiple steam generator tube failures occur. Moisture monitors detect the ingress. Steam generator isolation succeeds. Main circulator trip succeeds. SCS cooling is ineffective. SCS circulator trip occurs. Primary relief valves lift due to excessive pressure induced by the moisture. Primary relief valves fail open. Reactor trip occurs. Release path is through the dampers.
CC_W-2	Spurious control rod bank withdrawal occurs. Primary reactor scram fails. HTS cooling is ineffective . RCCS cooling succeeds. Multiple steam generator tube failures occur. Moisture monitors detect the ingress. Steam generator isolation succeeds. Main circulator trip succeeds. SCS cooling is ineffective. SCS circulator trip occurs. Primary relief valves lift due to excessive pressure induced by the moisture. Primary relief valves reseat. Reactor trip occurs. Release path is through the dampers.

TABLE 9-6
ANTICIPATED TRANSIENT INITIATED RELEASE CATEGORY DESIGNATIONS

Release Category	Description
RS-N1	Anticipated plant transient occurs which affects one module. Primary scram trips the module. HTS or SCS core cooling is maintained. No release occurs.
RS-N2	Anticipated plant transient occurs which affects one module. Primary reactor scram of the module fails. HTS or SCS core cooling is maintained. Reactor trip occurs in less than 30 h. No release occurs.
RS-N3	Anticipated plant transient occurs affecting four modules. Primary scram trips all modules. HTS or SCS core cooling is maintained. No release occurs.
CC_r-N1	Anticipated plant transient occurs affecting one module. Primary scram trips the module. HTS or SCS core cooling fail. RCCS cooling succeeds. No release occurs.
CC_r-N2	Anticipated plant transient occurs affecting all four modules. Primary scram trips all modules. HTS or SCS core cooling fail. RCCS cooling succeeds. No release occurs.
CC_r-N3	Anticipated plant transient occurs affecting one module. Primary reactor scram of the module fails. HTS or SCS core cooling fail. RCCS cooling succeeds. Reactor trip occurs in less than 30 h. No release occurs.
CC_r-N4	Anticipated plant transient occurs affecting one module. Primary scram trips the module. HTS, SCS, and RCCS cooling fail. HPS pumpdown succeeds. No release occurs.

TABLE 9-7
LOSS OF HTS COOLING INITIATED RELEASE CATEGORY DESIGNATIONS

Release Category	Description
LM-N1	HTS cooling is lost in one or all modules. Reactor trip occurs in the affected modules. SCS cooling succeeds. No release occurs.
CC_m-N1	HTS cooling is lost in one or all modules. Reactor trip occurs in the affected modules. SCS cooling fails. RCCS cooling succeeds. No release occurs.
CC_m-N2	HTS cooling is lost in all four modules. Reactor trip of the modules occurs. SCS and RCCS cooling fail. Pumpdown of the primary coolant by the HPS succeeds. No release occurs.

TABLE 9-8
LOSS OF OFFSITE POWER AND TURBINE TRIP
INITIATED RELEASE CATEGORY DESIGNATIONS

Release Category	Description
LP-N1	A loss of offsite power and inadvertent turbine trip occur. Primary scram trips all four modules. SCS cooling is maintained by the back-up power supply. No release occurs.
CC_L-N1	A loss of offsite power and inadvertent turbine trip occur. Primary scram trips all four modules. SCS cooling fails. RCCS cooling succeeds. No release occurs.

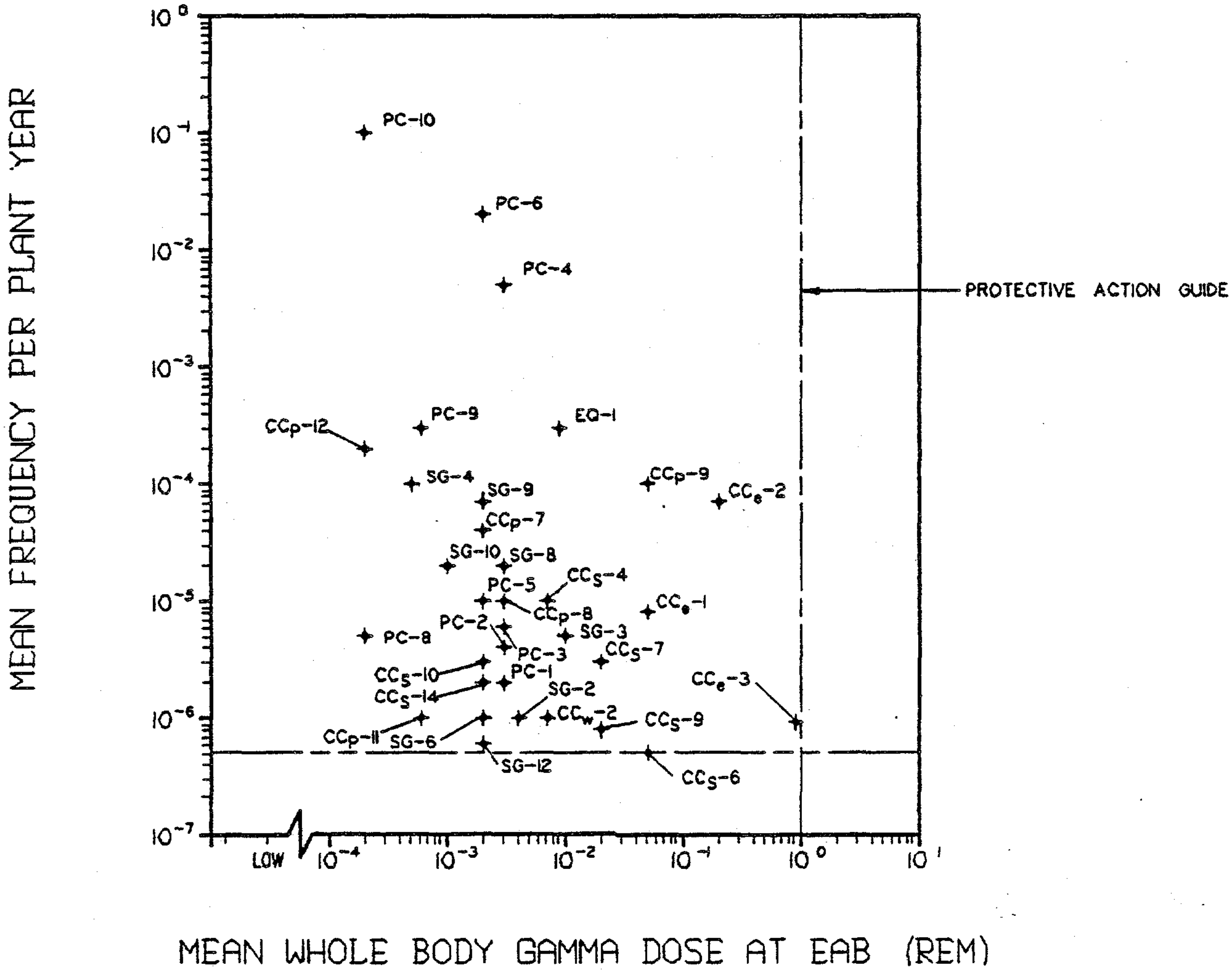

Figure 9-9. Point Value Whole Body Gamma Risk Plot

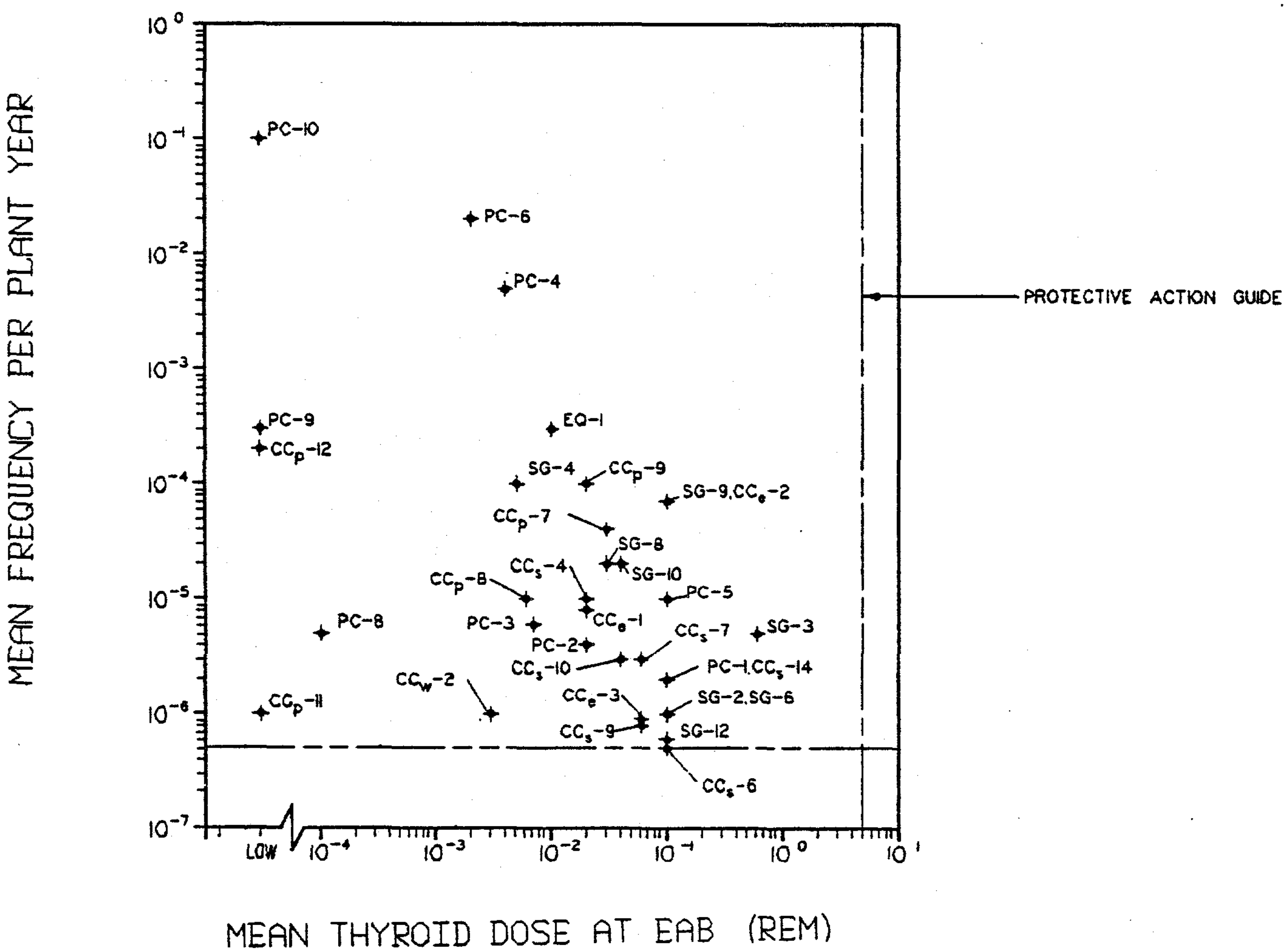

Figure 9-10. Point Value Thyroid Risk Plot

The values for mean frequency and consequence for each release category given on the point value plots is given in Table 9-9 which summarizes the pertinent results for all the accident families identified in this study. The table also indicates the dominant event tree sequence for each release category. Events which resulted in no offsite doses and therefore do not contribute to the safety risk of the MHTGR were included in the table as they become a needed factor in the licensing efforts of the HTGR program.

9.3 Mean Risk Estimates

Section 9.1 addressed the degree of confidence with which the MHTGR satisfies the user-imposed public sheltering/evacuation requirements. This was accomplished by comparing the complementary cumulative frequency distributions for whole body and thyroid doses to the PAG limits. The present section is concerned with mean risk (defined as the product of release category mean frequency and consequence) estimates. This differs from the Section 9.1 assessment because mean risk signifies what is expected to result if an accident occurs, while Section 9.1 was concerned with what doses might result if an accident occurred under statistically unlikely conditions (e.g., higher than expected fission product inventories or worse than expected weather). Table 9-10 lists the mean risk (in rem per plant year) for the whole body and thyroid, as well as the expected whole body gamma and thyroid doses to the maximum exposed individual at the plant EAB. The highest whole body and thyroid risks are due to release category PC-6. Although this release category does not result in exceptionally high doses, it is risk-dominant because it also has a relatively high occurrence frequency. The highest mean consequences are 0.9 rem and 0.8 rem to the whole body and thyroid, respectively. The highest whole body gamma dose results from release category CC_e-3 which is an earthquake-initiated conduction cooldown with instrument line failure, failure to trip, and RCCS failure in all four modules. The primary coolant leak initiated conduction cooldown release category CC_p-2 results in the highest thyroid dose. The depressurization area representative of CC_p-2 is 0.01 in.^{2}. Depressurization time through this small area is relatively long, therefore

TABLE 9-9
SUMMARY TABLE OF RESULTS FOR ALL ACCIDENT FAMILIES

Release Category (Dominant Branch)	Frequency (Per Plant Year) Mean	Median	Uncertainty Factor	Consequence (Rem) Mean	Median	Uncertainty Factor
PC-1	1.6×10^{-6}	1.0×10^{-6}	5	2.8×10^{-3} 0.14	6.6×10^{-4} 5.0×10^{-2}	10 (W.B.) 10 (Thy.)
PC-2	3.5×10^{-6}	1.3×10^{-6}	10	2.8×10^{-3} 2×10^{-2}	6.4×10^{-4} 6.8×10^{-3}	10 (W.B.) 10 (Thy.)
PC-3 (PC-DP)	5.9×10^{-6}	2.2×10^{-6}	10	2.8×10^{-3} 7.2×10^{-3}	6.2×10^{-4} 2.8×10^{-3}	10 (W.B.) 10 (Thy.)
PC-4 (PC-DE)	5.4×10^{-3}	2.2×10^{-3}	9	2.6×10^{-3} 3.8×10^{-3}	5.8×10^{-4} 1.5×10^{-3}	10 (W.B.) 10 (Thy.)
PC-5 (PC-BZ)	1.2×10^{-5}	3.3×10^{-6}	14	2.2×10^{-3} 4.8×10^{-2}	5.0×10^{-4} 1.7×10^{-2}	10 (W.B.) 10 (Thy.)
PC-6 (PC-BY)	2.1×10^{-2}	1.1×10^{-2}	7	2.2×10^{-3} 2.4×10^{-3}	5.0×10^{-4} 8.2×10^{-4}	10 (W.B.) 10 (Thy.)
PC-7 (PC-AV)	1.2×10^{-8}	8×10^{-9}	5	5.6×10^{-4} 4.8×10^{-4}	1.3×10^{-4} 1.7×10^{-4}	10 (W.B.) 10 (Thy.)
PC-8 (PC-AT)	5.4×10^{-6}	4.3×10^{-6}	3	1.7×10^{-4} 1.4×10^{-4}	3.8×10^{-5} 4.8×10^{-5}	10 (W.B.) 10 (Thy.)
PC-9 (PC-AB)	2.6×10^{-4}	2.1×10^{-4}	3	5.6×10^{-4} 2.8×10^{-5}	1.3×10^{-4} 9.6×10^{-6}	10 (W.B.) 10 (Thy.)
PC-10 (PC-AA)	0.11	0.11	~1	1.7×10^{-4} 5.6×10^{-6}	3.8×10^{-5} 1.9×10^{-6}	10 (W.B.) 10 (Thy.)
EQ-1 (EQ-AL)	3.4×10^{-4}	1×10^{-4}	13	8.8×10^{-3} 9.6×10^{-3}	2×10^{-3} 3.3×10^{-3}	10 (W.B.) 10 (Thy.)
SG-N1 (SS-AA)	0.4	0.4	~1	0	0	-
SG-N2 (MS-AA)	5×10^{-2}	4×10^{-2}	3	0	0	-
SG-N3 (SS-BY)	5×10^{-4}	4×10^{-4}	3	0	0	-

TABLE 9-9
(Continued)

Release Category (Dominant Branch)	Frequency (Per Plant Year) Mean	Median	Uncertainty Factor	Mean	Consequence (Rem) Median	Uncertainty Factor
SG-N5 (SS-AH)	1.3×10^{-4}	1×10^{-4}	3	0	0	-
SG-N6 (SS-AN)	6.5×10^{-5}	5.2×10^{-5}	3	0	0	-
SG-1 (MS-BI)	3.9×10^{-8}	2×10^{-8}	7	1.4×10^{-2} 0.64	4.8×10^{-3} 0.32	8 (W.B.) 7 (Thy.)
SG-2 (MS-AM)	9.8×10^{-7}	4×10^{-7}	9	4×10^{-3} 0.14	1.4×10^{-3} 7.0×10^{-2}	8 (W.B.) 7 (Thy.)
SG-3 (MS-CB)	4.7×10^{-6}	1.2×10^{-6}	15	1.4×10^{-2} 0.64	4.8×10^{-3} 0.32	8 (W.B.) 7 (Thy.)
SG-4 (SS-AE)	1.3×10^{-4}	1×10^{-4}	3	5.4×10^{-4} 5.4×10^{-3}	1.9×10^{-4} 9.9×10^{-4}	8 (W.B.) 21 (Thy.)
SG-5 (SS-CF)	2×10^{-8}	1×10^{-8}	7	1.4×10^{-2} 0.64	4.8×10^{-3} 0.32	8 (W.B.) 7 (Thy.)
SG-6 (MS-BH)	1.2×10^{-6}	8×10^{-7}	5	2.1×10^{-3} 9.6×10^{-2}	7.2×10^{-4} 4.8×10^{-2}	8 (W.B.) 7 (Thy.)
SG-7 (SS-AT)	1.6×10^{-7}	1×10^{-7}	5	5.4×10^{-4} 2.5×10^{-2}	1.8×10^{-4} 1.2×10^{-2}	8 (W.B.) 7 (Thy.)
SG-8 (MS-AH)	1.6×10^{-5}	1×10^{-5}	5	2.8×10^{-3} 2.8×10^{-2}	9.7×10^{-4} 5.1×10^{-3}	8 (W.B.) 21 (Thy.)
SG-9 (MS-CA)	7.3×10^{-5}	4.7×10^{-5}	5	2.1×10^{-3} 9.6×10^{-2}	7.2×10^{-4} 4.8×10^{-2}	8 (W.B.) 7 (Thy.)
SG-10 (MS-AL)	1.6×10^{-5}	1×10^{-5}	5	1.2×10^{-3} 4.2×10^{-2}	4.2×10^{-4} 2.1×10^{-2}	8 (W.B.) 7 (Thy.)
SG-11 (MS-BC)	2×10^{-8}	1×10^{-8}	7	4.5×10^{-3} 0.57	1.5×10^{-3} 0.13	11 (W.B.) 17 (Thy.)
SG-12 (SS-CE)	6.2×10^{-7}	4×10^{-7}	5	2.1×10^{-3} 9.6×10^{-2}	7.2×10^{-4} 4.8×10^{-2}	8 (W.B.) 7 (Thy.)
SG-13 (SS-AX)	6.2×10^{-9}	4×10^{-9}	5	4.4×10^{-3} 9.6×10^{-2}	1.2×10^{-3} 4.7×10^{-2}	14 (W.B.) 7 (Thy.)

TABLE 9-9
(Continued)

Release Category (Dominant Branch)	Frequency (Per Plant Year) Mean	Median	Uncertainty Factor	Mean	Consequence (Rem) Median	Uncertaint Factor
RS-N1 (RS-AA)	21	21	~1	0	0	-
RS-N2 (RS-AJ)	2.9×10^{-4}	2.3×10^{-4}	3	0	0	-
RS-N3 (RS-AB)	5.5	5.5	~1	0	0	-
RW-N1 (RW-AA)	8.8×10^{-2}	7×10^{-2}	3	0	0	-
RW-N2 (RW-AB)	9.4×10^{-3}	6×10^{-3}	5	0	0	-
LP-N1 (LOSP-AA)	1.1×10^{-2}	9×10^{-3}	3	0	0	-
LM-N1 (HTS-AA)	2.5	2.5	~1	0	0	-
EQ-N1 (EQ-AA)	2.5×10^{-3}	2×10^{-3}	3	0	0	-
EQ-N2 (EQ-AB)	1.9×10^{-3}	1.5×10^{-3}	3	0	0	-
CC_r-N1 (RS-AE)	7.8×10^{-2}	5×10^{-2}	5	0	0	-
CC_r-N2 (RS-AF)	7.8×10^{-5}	4×10^{-5}	7	0	0	-
CC_r-N3 (RS-AN)	9.8×10^{-7}	5×10^{-7}	7	0	0	-
CC_r-N4 (RS-AG)	9.8×10^{-8}	5×10^{-8}	7	0	0	-
CC_w-N3 (RW-AC)	1.6×10^{-4}	1×10^{-4}	7	0	0	-

TABLE 9-9
(Continued)

Release Category (Dominant Branch)	Frequency (Per Plant Year) Mean	Median	Uncertainty Factor	Mean	Consequence (Rem) Median	Uncertaint Factor
CC_L-N1 (LOSP-AB)	3.1×10^{-4}	2×10^{-4}	5	0	0	-
CC_m-N1 (HTS-AC)	6.3×10^{-2}	5×10^{-2}	3	0	0	-
CC_m-N2 (HTS-AE)	7.8×10^{-8}	5×10^{-8}	5	0	0	-
CC_e-N1 (EQ-AC)	4.7×10^{-5}	3×10^{-5}	5	0	0	-
CC_e-N2 (EQ-AD)	2.3×10^{-7}	1.2×10^{-7}	7	0	0	-
CC_s-N1 (SS-AB)	1×10^{-2}	8×10^{-3}	3	0	0	-
CC_s-N2 (MS-AB)	1.2×10^{-3}	8×10^{-4}	5	0	0	-
CC_s-N3 (SS-AC)	1.2×10^{-8}	8×10^{-9}	5	0	0	-
CC_e-1 (EQ-AM)	8.3×10^{-6}	2×10^{-6}	16	5.4×10^{-2} 2.4×10^{-2}	5.6×10^{-3} 1.3×10^{-2}	10 (W.B. 10 (Thy.
CC_e-2 (EQ-AN)	6.7×10^{-5}	2×10^{-6}	13	0.22 9.6×10^{-2}	2.2×10^{-2} 5.2×10^{-2}	10 (W.B. 10 (Thy.
CC_e-3 (EQ-AP)	8.8×10^{-7}	1×10^{-7}	31	0.92 6.0×10^{-2}	9.2×10^{-2} 3.2×10^{-2}	34 (W.B.) 6 (Thy.)
CC_w-1 (RW-AG)	3.9×10^{-8}	2×10^{-8}	7	4.3×10^{-2} 2.1×10^{-2}	4.5×10^{-3} 1.1×10^{-2}	33 (W.B.) 6 (Thy.)
CC_w-2 (RW-AE)	1.2×10^{-6}	8×10^{-7}	5	6.5×10^{-3} 3.2×10^{-3}	6.8×10^{-4} 1.7×10^{-3}	33 (W.B. 6 (Thy.

TABLE 9-9
(Continued)

Release Category (Dominant Branch)	Frequency (Per Plant Year) Mean	Median	Uncertainty Factor	Mean	Consequence (Rem) Median	Uncertaint Factor
CC_p-2 (PC-BK)	9.6×10^{-9}	5×10^{-9}	6	1.8×10^{-3} 0.8	7×10^{-4} 0.28	10 (W.B. 12 (Thy.
CC_p-4 (PC-BC)	2.4×10^{-7}	1.2×10^{-7}	7	2.4×10^{-3} 6.8×10^{-2}	1.0×10^{-3} 2.8×10^{-2}	10 (W.B. 10 (Thy.
CC_p-6 (PC-DU)	8.3×10^{-9}	3×10^{-9}	10	2.8×10^{-3} 2.0×10^{-2}	6.4×10^{-4} 6.8×10^{-3}	10 (W.B. 10 (Thy.
CC_p-7 (PC-BJ)	4.0×10^{-5}	2.4×10^{-5}	5	1.8×10^{-3} 3.2×10^{-2}	7×10^{-4} 1.2×10^{-2}	10 (W.B. 12 (Thy.
CC_p-8 (PC-DJ)	1.4×10^{-5}	3.5×10^{-6}	15	2.8×10^{-3} 5.6×10^{-3}	6.2×10^{-4} 2.2×10^{-3}	10 (W.B. 10 (Thy.
CC_p-9 (PC-CP)	1.1×10^{-4}	2.3×10^{-5}	19	5.4×10^{-2} 2.4×10^{-2}	5.6×10^{-3} 1.3×10^{-2}	10 (W.B. 10 (Thy.
CC_p-10 (PC-CQ)	5×10^{-9}	3×10^{-9}	5	5.4×10^{-2} 0.48	5.6×10^{-3} 0.25	10 (W.B. 10 (Thy.
CC_p-11 (PC-AF)	1.3×10^{-6}	8×10^{-7}	5	5.6×10^{-4} 2.8×10^{-5}	1.3×10^{-4} 9.6×10^{-6}	10 (W.B. 10 (Thy.
CC_p-12 (PC-AJ)	1.8×10^{-4}	1.1×10^{-4}	5	1.7×10^{-4} 5.6×10^{-6}	3.8×10^{-5} 1.9×10^{-6}	10 (W.B. 10 (Thy.
CC_s-3 (SS-AU)	3.9×10^{-9}	2×10^{-9}	7	1.8×10^{-3} 3.2×10^{-2}	7×10^{-4} 1.2×10^{-2}	10 (W.B. 12 (Thy.
CC_s-4 (SS-BZ)	1.4×10^{-5}	9.1×10^{-6}	5	6.8×10^{-3} 1.5×10^{-2}	7.8×10^{-4} 8.6×10^{-3}	30 (W.B. 6 (Thy.
CC_s-5 (SS-AK)	1.4×10^{-7}	7×10^{-8}	7	4.5×10^{-2} 0.1	5.2×10^{-3} 5.7×10^{-2}	30 (W.B. 6 (Thy.

TABLE 9-9
(Continued)

Release Category (Dominant Branch)	Frequency (Per Plant Year) Mean	Median	Uncertainty Factor	Consequence (Rem) Mean	Median	Uncertainty Factor
CC_s-6 (SS-CB)	4.5×10^{-7}	2.3×10^{-7}	7	4.5×10^{-2} 0.1	5.2×10^{-3} 5.7×10^{-2}	30 (W.B.) 6 (Thy.)
CC_s-7 (SS-AI)	3.1×10^{-6}	2×10^{-6}	5	2.3×10^{-2} 5.5×10^{-2}	2.6×10^{-3} 2.9×10^{-2}	31 (W.B.) 6 (Thy.)
CC_s-8 (MS-BK)	3.9×10^{-8}	2×10^{-8}	7	2.1×10^{-3} 9.6×10^{-2}	7.2×10^{-4} 4.8×10^{-2}	8 (W.B.) 7 (Thy.)
CC_s-9 (MS-AI)	7.8×10^{-7}	2×10^{-7}	15	1.8×10^{-2} 6.4×10^{-2}	4.2×10^{-3} 1.4×10^{-2}	8 (W.B.) 18 (Thy.)
CC_s-10 (SS-AF)	3.1×10^{-6}	2×10^{-6}	5	2.3×10^{-3} 3.7×10^{-2}	8.9×10^{-4} 1.3×10^{-2}	10 (W.B. 11 (Thy.
CC_s^--11 (MS-AQ)	1.7×10^{-8}	7×10^{-9}	9	4×10^{-3} 0.14	1.4×10^{-3} 7×10^{-2}	8 (W.B. 7 (Thy.
CC_s^--12 (MS-CF)	9.3×10^{-8}	2.4×10^{-8}	15	1.4×10^{-2} 0.64	4.8×10^{-3} 0.32	8 (W.B. 7 (Thy.
CC_s^--13 (MS-AO)	3.9×10^{-7}	2×10^{-7}	7	1.2×10^{-3} 4.2×10^{-2}	4.2×10^{-4} 2.1×10^{-2}	8 (W.B.) 7 (Thy.)
CC_s-14 (MS-CD)	1.8×10^{-6}	9.3×10^{-7}	7	2.1×10^{-3} 9.6×10^{-2}	7.2×10^{-4} 4.8×10^{-2}	8 (W.B.) 7 (Thy.)

TABLE 9-10
MEAN RISK ESTIMATES

Release Category	Mean Risk* (Rem/Plant Year)	Mean Consequence* (Rem)
PC-1	4×10^{-9} (W.B.) 2×10^{-7} (Thy.)	3×10^{-3} (W.B.) 0.1 (Thy.)
PC-2	1×10^{-8} (W.B.) 7×10^{-8} (Thy.)	3×10^{-3} (W.B.) 2×10^{-2} (Thy.)
PC-3	2×10^{-8} (W.B.) 4×10^{-8} (Thy.)	3×10^{-3} (W.B.) 7×10^{-3} (Thy.)
PC-4	1×10^{-5} (W.B.) 2×10^{-5} (Thy.)	3×10^{-3} (W.B.) 4×10^{-3} (Thy.)
PC-5	3×10^{-8} (W.B.) 6×10^{-7} (Thy.)	2×10^{-3} (W.B.) 5×10^{-2} (Thy.)
PC-6	5×10^{-5} (W.B.) 5×10^{-5} (Thy.)	2×10^{-3} (W.B.) 2×10^{-3} (Thy.)
PC-7	7×10^{-12} (W.B.) 6×10^{-12} (Thy.)	6×10^{-4} (W.B.) 5×10^{-4} (Thy.)
PC-8	9×10^{-10} (W.B.) 8×10^{-10} (Thy.)	2×10^{-4} (W.B.) 1×10^{-4} (Thy.)
PC-9	1×10^{-7} (W.B.) 7×10^{-9} (Thy.)	6×10^{-4} (W.B.) 3×10^{-5} (Thy.)
PC-10	2×10^{-5} (W.B.) 6×10^{-7} (Thy.)	2×10^{-4} (W.B.) 6×10^{-6} (Thy.)
EQ-1	3×10^{-6} (W.B.) 3×10^{-6} (Thy.)	9×10^{-3} (W.B.) 1×10^{-2} (Thy.)
SG-1	5×10^{-10} (W.B.) 2×10^{-8} (Thy.)	1×10^{-2} (W.B.) 0.6 (Thy.)
SG-2	4×10^{-9} (W.B.) 1×10^{-7} (Thy.)	4×10^{-3} (W.B.) 0.1 (Thy.)
SG-3	7×10^{-8} (W.B.) 3×10^{-6} (Thy.)	1×10^{-2} (W.B.) 0.6 (Thy.)

TABLE 9-10
(Continued)

Release Category	Mean Risk* (Rem/Plant Year)	Mean Consequence* (Rem)
SG-4	7×10^{-8} (W.B.) 7×10^{-7} (Thy.)	5×10^{-4} (W.B.) 5×10^{-3} (Thy.)
SG-5	3×10^{-10} (W.B.) 1×10^{-8} (Thy.)	1×10^{-2} (W.B.) 0.6 (Thy.)
SG-6	3×10^{-9} (W.B.) 1×10^{-7} (Thy.)	2×10^{-3} (W.B.) 0.1 (Thy.)
SG-7	9×10^{-11} (W.B.) 4×10^{-9} (Thy.)	5×10^{-4} (W.B.) 3×10^{-2} (Thy.)
SG-8	4×10^{-8} (W.B.) 4×10^{-7} (Thy.)	3×10^{-3} (W.B.) 3×10^{-2} (Thy.)
SG-9	2×10^{-7} (W.B.) 7×10^{-6} (Thy.)	2×10^{-3} (W.B.) 0.1 (Thy.)
SG-10	2×10^{-8} (W.B.) 7×10^{-7} (Thy.)	1×10^{-3} (W.B.) 4×10^{-2} (Thy.)
SG-11	9×10^{-11} (W.B.) 1×10^{-8} (Thy.)	5×10^{-3} (W.B.) 0.6 (Thy.)
SG-12	1×10^{-9} (W.B.) 6×10^{-8} (Thy.)	2×10^{-3} (W.B.) 0.1 (Thy.)
SG-13	3×10^{-11} (W.B.) 6×10^{-10} (Thy.)	4×10^{-3} (W.B.) 0.1 (Thy.)
CC_e-1	4×10^{-7} (W.B.) 2×10^{-7} (Thy.)	5×10^{-2} (W.B.) 2×10^{-2} (Thy.)
CC_e-2	1×10^{-5} (W.B.) 6×10^{-6} (Thy.)	0.2 (W.B.) 0.1 (Thy.)
CC_e-3	8×10^{-7} (W.B.) 5×10^{-8} (Thy.)	0.9 (W.B.) 6×10^{-2} (Thy.)
CC_w-1	2×10^{-9} (W.B.) 8×10^{-10} (Thy.)	4×10^{-2} (W.B.) 2×10^{-2} (Thy.)
CC_w-2	8×10^{-9} (W.B.) 4×10^{-9} (Thy.)	7×10^{-3} (W.B.) 3×10^{-3} (Thy.)

TABLE 9-10
(Continued)

Release Category	Mean Risk* (Rem/Plant Year)	Mean Consequence* (Rem)
CC_p-2	2×10^{-11} (W.B.) 8×10^{-9} (Thy.)	2×10^{-3} (W.B.) 0.8 (Thy.)
CC_p-4	6×10^{-10} (W.B.) 2×10^{-8} (Thy.)	2×10^{-3} (W.B.) 7×10^{-2} (Thy.)
CC_p-6	2×10^{-11} (W.B.) 2×10^{-10} (Thy.)	3×10^{-3} (W.B.) 2×10^{-2} (Thy.)
CC_p-7	7×10^{-8} (W.B.) 1×10^{-6} (Thy.)	2×10^{-3} (W.B.) 3×10^{-2} (Thy.)
CC_p-8	4×10^{-8} (W.B.) 8×10^{-8} (Thy.)	3×10^{-3} (W.B.) 6×10^{-3} (Thy.)
CC_p-9	6×10^{-6} (W.B.) 3×10^{-6} (Thy.)	5×10^{-2} (W.B.) 2×10^{-2} (Thy.)
CC_p-10	3×10^{-10} (W.B.) 2×10^{-9} (Thy.)	5×10^{-2} (W.B.) 0.5 (Thy.)
CC_p-11	7×10^{-10} (W.B.) 4×10^{-11} (Thy.)	6×10^{-4} (W.B.) 3×10^{-5} (Thy.)
CC_p-12	3×10^{-8} (W.B.) 1×10^{-9} (Thy.)	2×10^{-4} (W.B.) 6×10^{-6} (Thy.)
CC_s-3	7×10^{-12} (W.B.) 1×10^{-10} (Thy.)	2×10^{-3} (W.B.) 3×10^{-2} (Thy.)
CC_s-4	1×10^{-7} (W.B.) 2×10^{-7} (Thy.)	7×10^{-3} (W.B.) 2×10^{-2} (Thy.)
CC_s-5	6×10^{-9} (W.B.) 1×10^{-8} (Thy.)	5×10^{-2} (W.B.) 0.1 (Thy.)
CC_s-6	2×10^{-8} (W.B.) 5×10^{-8} (Thy.)	5×10^{-2} (W.B.) 0.1 (Thy.)
CC_s-7	7×10^{-8} (W.B.) 2×10^{-7} (Thy.)	2×10^{-2} (W.B.) 6×10^{-2} (Thy.)
CC_s-8	8×10^{-11} (W.B.) 4×10^{-9} (Thy.)	2×10^{-3} (W.B.) 0.1 (Thy.)

TABLE 9-10
(Continued)

Release Category	Mean Risk* (Rem/Plant Year)	Mean Consequence* (Rem)
CC_s-9	1×10^{-8} (W.B) 5×10^{-8} (Thy.)	2×10^{-2} (W.B.) 6×10^{-2} (Thy.)
CC_s-10	7×10^{-9} (W.B.) 1×10^{-7} (Thy.)	2×10^{-3} (W.B.) 4×10^{-2} (Thy.)
CC_s-11	7×10^{-11} (W.B.) 2×10^{-9} (Thy.)	4×10^{-3} (W.B.) 0.1 (Thy.)
CC_s-12	1×10^{-9} (W.B.) 6×10^{-8} (Thy.)	1×10^{-2} (W.B.) 0.6 (Thy.)
CC_s-13	5×10^{-10} (W.B.) 2×10^{-8} (Thy.)	1×10^{-3} (W.B.) 4×10^{-2} (Thy.)
CC_s-14	4×10^{-9} (W.B.) 2×10^{-7} (Thy.)	2×10^{-3} (W.B.) 0.1 (Thy.)

providing a vital transport mechanism for conveying fission products released from the fuel from the reactor vessel into the reactor building. The additional failure to disengage reactor building HVAC fans results in increased fission product transport to the atmosphere.

The latent fatality risk to the maximum exposed individual at the EAB is obtained from the whole body risk by using a conversion factor of one fatality per 10^4 rem. No acute fatalities are expected because doses are well below the acute fatality threshold.

9.4 References

9-1. Thomas, H. M., "Pipe and Vessel Failure Probability," Reliability Engineering, 1981.

9-2. Smith, T. A., and R. G. Warwick, "A Survey of Defects in Pressure Vessels in the U.K. for the Period 1982-1978 and Its Relevance to Nuclear Primary Circuits," International Journal of Pressure Vessels and Piping, 1983.

APPENDIX A

PRA DATA BASE

[LATER]