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DOE/HTGR--86-034

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LICENSING BASIS EVENTS

FOR THE MODULAR HTGR

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DOE Contract No. DE-AC03-84SF11963

GA Project 6300

APRIL 1986

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GA 1485 (REV. 10/82)

GA Technologies Inc.

INFO

ISSUE SUMMARY

TITLE			
LICENSING BASIS EVENTS FOR THE MODULAR HTGR	☐ R & D ☐ DV & S ☒ DESIGN	APPROVAL LEVEL 5	

DISCIPLINE	SYSTEM	DOC. TYPE	PROJECT	DOCUMENT NO.	ISSUE NO./LTR.
0	01-00	RGE	6300	908814	0

QUALITY ASSURANCE LEVEL	SAFETY CLASSIFICATION	SEISMIC CATEGORY	ELECTRICAL CLASSIFICATION
N/A	N/A	N/A	N/A

ISSUE	DATE	PREPARED BY	APPROVAL				ISSUE DESCRIPTION/ CWBS NO.
			ENGINEERING	QA	FUNDING PROJECT	APPLICABLE PROJECT	
0	APR 1 8 1986	E.A.Bellis E.A.Bellis	S.B.Inamati S.B.Inamati F.A.Silady F.A.Silady D.S.Sheng Interface Assurance	G.P.Connors G.P.Connors		G.C.Bramblett	Initial Issue WBS 6351060101 HTGR-86-034

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Cover Sheets	= 2	(Unnumbered)
Pages 1 thru 59	= 59	
Total	61	

NEXT INDENTURED DOCUMENTS

908397

HTGR-86-004

REV																												
SH																												
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SH	29	30	31	32	33	34	35	36	37	38	39	40	41	42	43	44	45	46	47	48	49	50	51	52	53	54	55	56
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CWINTER-AAWOR-80 (SR-6934)

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TABLES

1. INTRODUCTION

The licensing approach of the MHTGR program may be summarized in three steps:

1. Identify top level criteria generic to all reactor types as a starting point.

2. Develop a process to derive licensing bases specific to the MHTGR which insure that top level criteria are met.

3. Apply the process to identify MHTGR licensing bases.

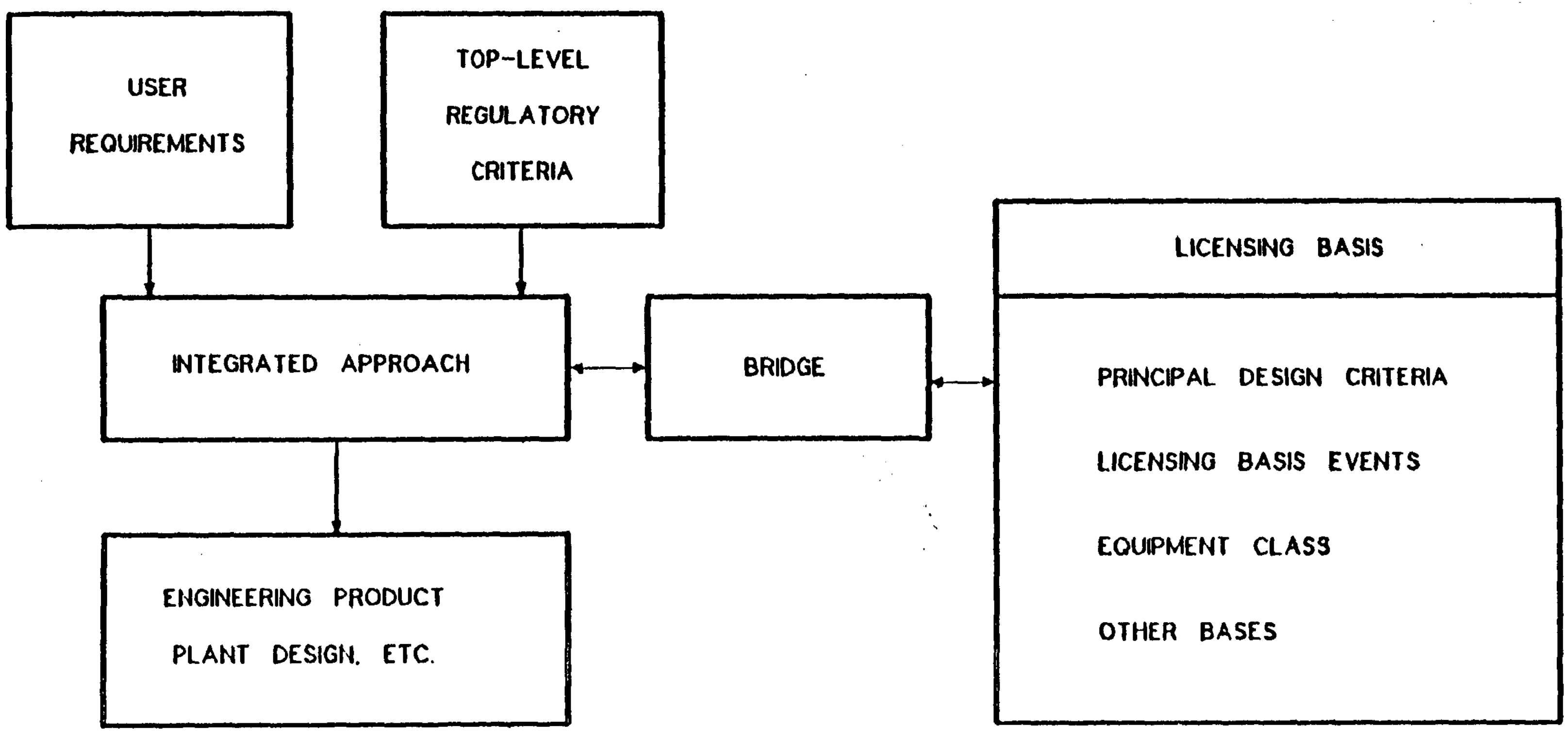
The design of the MHTGR is being conducted in a top down, goal-oriented manner using the DOE-sponsored Integrated Approach. Methods have previously been presented (Ref. 1) that enable bridging between this design process and the NRC's licensing framework. In turn, this structured method for selecting licensing bases supports one of the objectives of the Integrated Approach process, promoting stability in licensing. In this document these bridging methods are applied to derive licensing basis events for the MHTGR.

At each design phase (i.e., preconceptual, conceptual, preliminary, and final) the bridging methods will be used to derive licensing bases from Top-level Regulatory Criteria (Ref. 2). This is illustrated in Fig. 1-1. On the left hand side of the figure are seen the Top-Level Regulatory Criteria feeding the Integrated Approach process. This process, also having user requirements as input, yields the engineering product of the plant design, etc. On the right is a list of licensing bases.

The subject of this document is quantification of the licensing basis event portion of the overall licensing basis for the MHTGR.

FIGURE 1-1

BRIDGING METHOD: A BRIDGE BETWEEN THE INTEGRATED APPROACH AND THE LICENSING BASIS



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Since this is the first application for a series of design phases in which the design concept is being developed, the licensing basis events identified in this report can be expected to be adjusted as time passes in response to design evolution and further analyses. However, the report provides the status of licensing basis events identified for the MHTGR to date.

2. BRIDGING METHOD FOR LICENSING BASIS EVENT SELECTION

The method for bridging between the Integrated Approach design process and the licensing basis, with which the NRC is familiar, has been documented in Ref. 1. A series of steps is outlined in the bridging method which will be used here to choose licensing basis events for the current MHTGR design. These steps are summarized in Table 2-1 to illustrate the flow of the process used to select licensing basis events. The first part of this section summarizes the definition of licensing basis events resulting from application of the bridging method. In the second half, quantitative guidelines established to select licensing basis events are summarized. A more detailed description of the selection criteria is given in Ref. 3.

2.1 Licensing Basis Event Definition

Licensing Basis Events (LBEs) are one of the vehicles by which regulators evaluate the licensability of a plant. The LBEs need to be chosen in the process of defining the plant concept.

The approach herein is to determine the LBEs utilizing probabilistic risk assessment. This provides a basis for judging, in a quantitative manner, the frequency of the entire event sequence. The event frequency is used to determine the appropriate dose or risk criteria.

TABLE 2-1
STEPS FOR LICENSING BASIS EVENT SELECTION

Step 1. Define three regions on a frequency-consequence risk plot bounded by three agreed upon mean frequencies and related to Appendix I, 10CFR100 or the PAGs.

Step 2. Compare the results of a risk assessment of the established design to the frequency-consequence risk plot.

Step 3. Identify as AOOs those families of events within the Appendix I region that would violate the dose criteria were it not for design selections that control radioactivity release.

Step 4. Evaluate the consequences of the selected AOOs realistically against the Appendix I annualized dose criteria. Provide in PSID Chapter 11 Analyses.

Step 5. Identify as DBEs those families of events within the design basis region that would violate the dose criteria were it not for design selections that control radioactivity release.

Step 6. Identify as DBEs those events with agreed upper margin frequencies that lie within the design basis region and satisfy step 5.

Step 7. Evaluate the consequences of the selected DBEs conservatively against the 10CFR100 dose criteria. Provide the evaluation in PSID Chapter 15 analyses.

Step 8. Identify as emergency planning basis events (EPBEs) the dose-dominant events whose upper margin frequency is within the emergency planning basis region.

Step 9. Evaluate the consequences of the selected EPBEs realistically for emergency planning and environmental assessments.

This process can be illustrated with a figure, such as Fig. 2-1, which is basically a "risk plot" where accident families are plotted. The ordinate is the frequency with which an accident family is predicted to occur. The abscissa is the dose for the family evaluated at the Exclusion Area Boundary (EAB) of the plant.

The steps applied in the bridging method yield the following definitions for licensing basis events.

1. Licensing Basis Eventss (LBEs)

   - The off-normal or accident events used for demonstrating design compliance with the top-level regulatory criteria.

   - Collectively LBEs are analyzed in PRAs for demonstrating compliance with interim safety risk goals.

   - LBEs encompass the following three event catagories:

   A. Anticipated Operational Occurrences (AOOs)

      - Events Expected once or more in the plant lifetime.

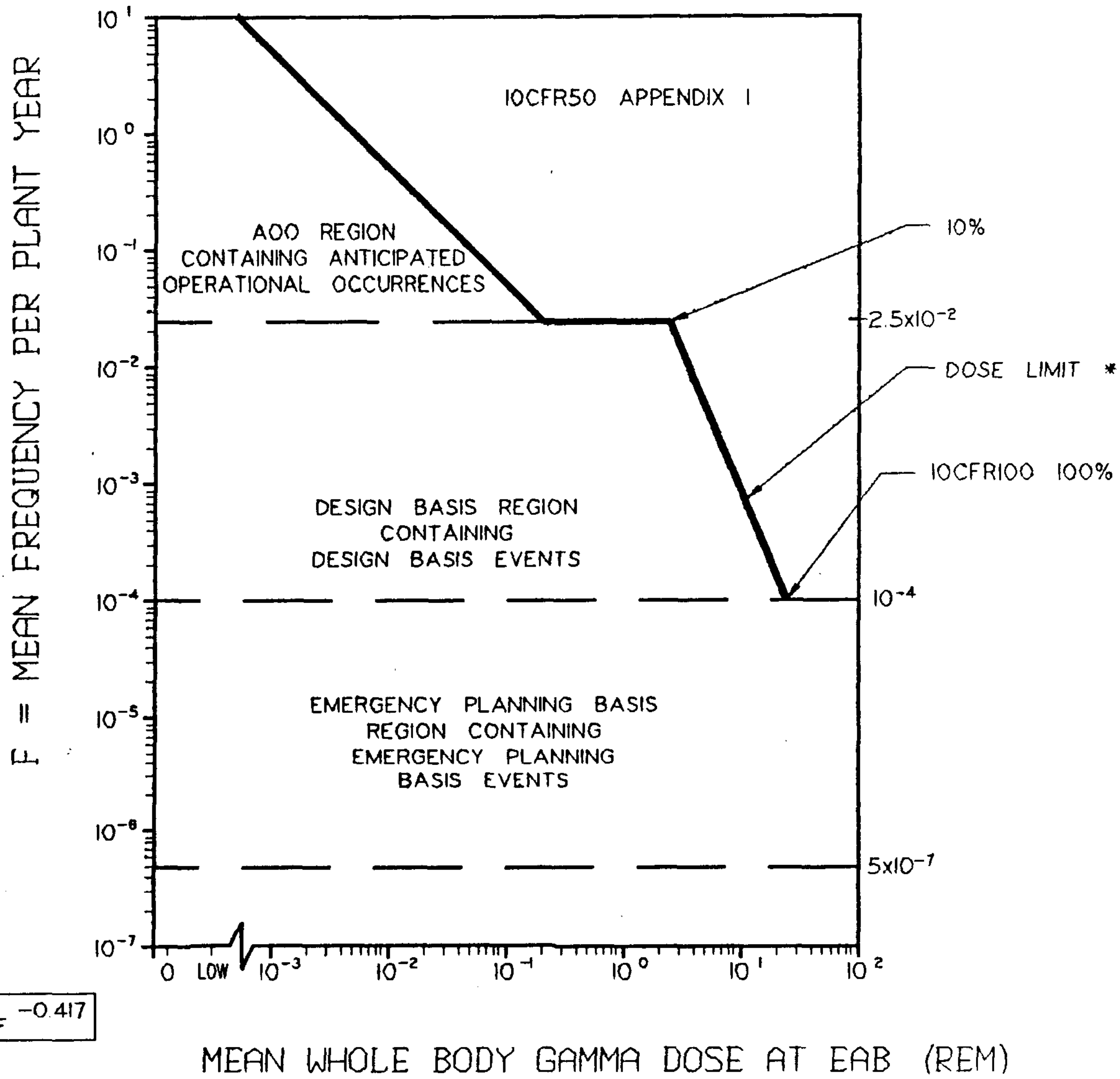
      - Consequences realistically analyzed in Chapter 11 of PSID for compliance with 10CFR50 Appendix I

   B. Design Basis Events (DBEs)

      - Events of lower frequency than AOOs, not expected to occur in the lifetime of the plant

      - Consequences conservatively analyzed in Chapter 15 of PSID for compliance with 10CFR100

FIGURE 2-1
LICENSING BASIS REGIONS FOR
LICENSING BASIS EVENT SELECTION



* FRACTIONS OF 10CFR100

$F = 6.31(\text{FRACTION})^{-2.4}$

$\text{FRACTION} = (6.31/F)^{0.417} = \boxed{2.16F^{-0.417}}$

C. Emergency Planning Basis Events (EPBEs)

- Events of lower frequency than DBEs, not expected to occur in the lifetime of all modular HTGRs

- Consequences realistically analyzed in PRA and Chapter 13 of PSID for compliance with PAG dose limits

2.2 Licensing Basis Event Selection Criteria

The first step of the method for choosing LBEs defines three regions on a frequency-consequence risk plot bounded by three agreed upon mean frequencies. In Fig. 2-1 these three regions are shown along with their bounding frequencies. Development of the numerical definitions for these boundaries is given in Ref. 3.

Families of events which plot near the lower boundary of some region may have significant uncertainties in the estimate of their frequencies, as acknowledged in the LBE selection process. The consideration of these uncertainties is necessary to allow a well balanced choice of events that will be under the rules of the appropriate region. The mean value of frequency, which involves an integral over the complete uncertainty spectrum, is the proposed function for accounting for frequency uncertainties. An agreed upon factor would then be placed on the mean frequency to provide margin and to dispel concern over event families falling just barely below the frequency boundaries of a region and therefore not being included in the region. A factor of 2 is used in this report.

3. SELECTION OF LICENSING BASIS EVENTS

Having defined the Top-level Regulatory Criteria and the regions over which these criteria will be applied, the licensing basis events are selected.

The next step is to compare the results of a risk assessment of a plant design meeting the established requirements to the regulatory dose limits previously described. In Fig. 3-1 the results of the risk assessment for the MHTGR are compared to the regulatory criteria as shown on a frequency-consequence risk plot. These results are based upon analyses performed for the 350 MW(t) plant (Ref. 4) Each point on the risk plot is labelled with an accident family designation from the PRA. A general definition of these families is given in Table 3-1. The points shown on the plot are for mean frequency and mean values of dose. The conservative values of dose are developed in the bridging process by doing conservative dose calculations.

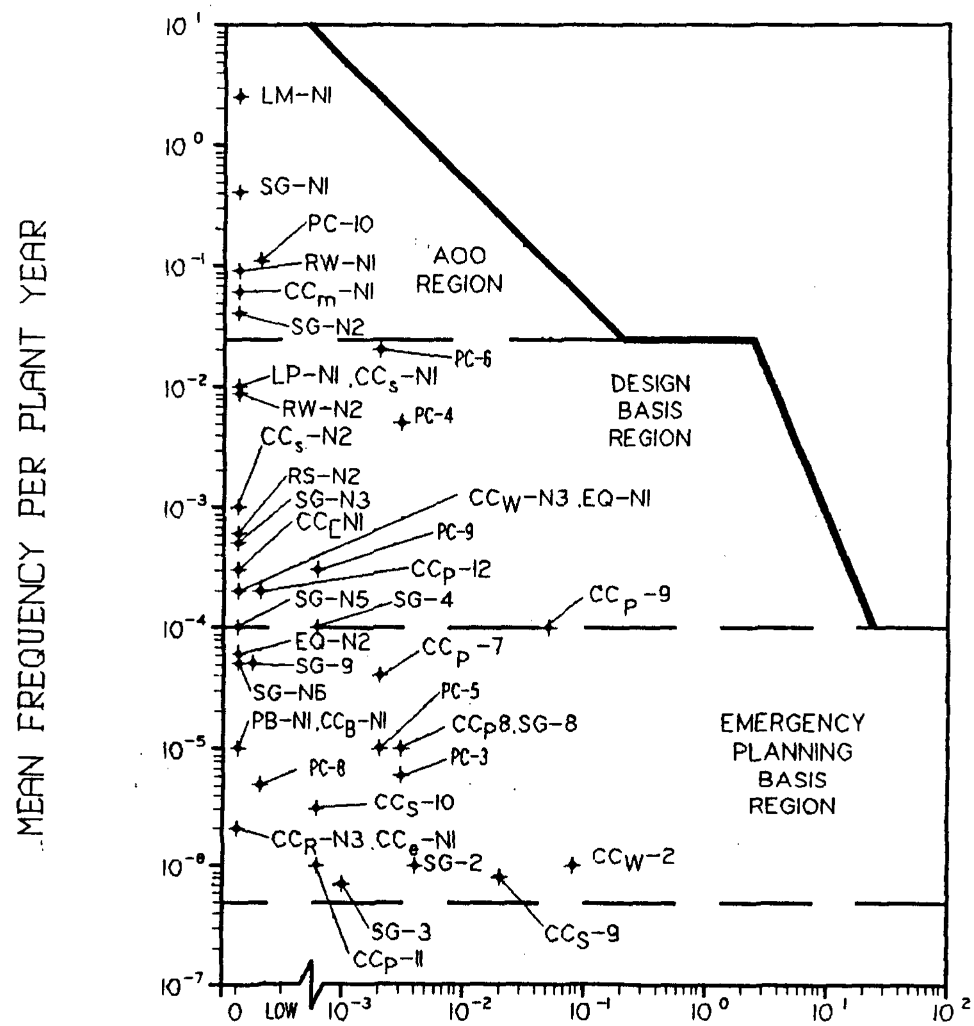
Many of the event families of interest result in no or essentially no dose to the public. These events are shown lying on the left of the figure. Other events, to the right of these, represent those event families that lead to significant predicted public exposure and still fall within the range of frequencies considered in the licensing bases. Because the plant was designed from the top down to meet all the top-level requirements of the Integrated Approach, both user and regulatory, all these events can be seen to fall within the allowable regulatory dose limits.

3.1 Anticipated Operational Occurrences

3.1.1 Selection of AOOs

AOOs are identified as those families of events whose mean frequencies fall within the AOO region and would violate the dose criteria were it not for design selections that control radioactivity release. In Fig. 3-2 those event families with a frequency greater than 0.025 (once in 40 years) and therefore falling within the AOO region are shown.

FIG 3-1
MHTGR PRELIMINARY SAFETY RISK ASSESSMENT
COMPARED TO LICENSING BASIS REGIONS



MEAN WHOLE BODY GAMMA DOSE AT EAB (REM)

TABLE 3-1
ACCIDENT FAMILY DESIGNATION DEFINITIONS*

PC-X	Accident family is initiated by a leak in the primary coolant pressure boundary. Forced core cooling is maintained.
RW-X	Accident family is initiated by a control rod bank withdrawal. Forced core cooling is maintained.
RS-X	Accident family is initiated by a main loop transient which requires a reactor trip. Forced core cooling is maintained.
EQ-X	Accident family is initiated by an earthquake. Forced core cooling is maintained.
LP-X	Accident family is initiated by a loss of offsite power and turbine generator trip. Forced core cooling is maintained.
LM-X	Accident family is initiated by a loss of main loop cooling caused by some event other than a loss of offsite power or an event external to the plant. Forced core cooling is maintained.
SG-X	Accident family is initiated by a leak in one or more steam generator tubes. Forced core cooling is maintained.
PB-X	Accident family is initiated by a main steam or feedwater pipe break. Forced core cooling is maintained.
CC_p-X	Accident family is initiated by a leak in the primary coolant pressure boundary and results in a conduction cooldown (cc) transient.
CC_s-X	Accident family is initiated by a leak in one or more steam generator tubes and results in a conduction cooldown transient.
CC_e-X	Accident family is initiated by an earthquake and results in a conduction cooldown transient.
CC_w-X	Accident family is initiated by a control rod bank withdrawal and results in a conduction cooldown transient.
CC_r-X	Accident family is initiated by a main loop transient requiring a reactor trip and results in a conduction cooldown transient.
CC_L-X	Accident family is initiated by a loss of offsite power and turbine generator trip and results in a conduction cooldown transient.

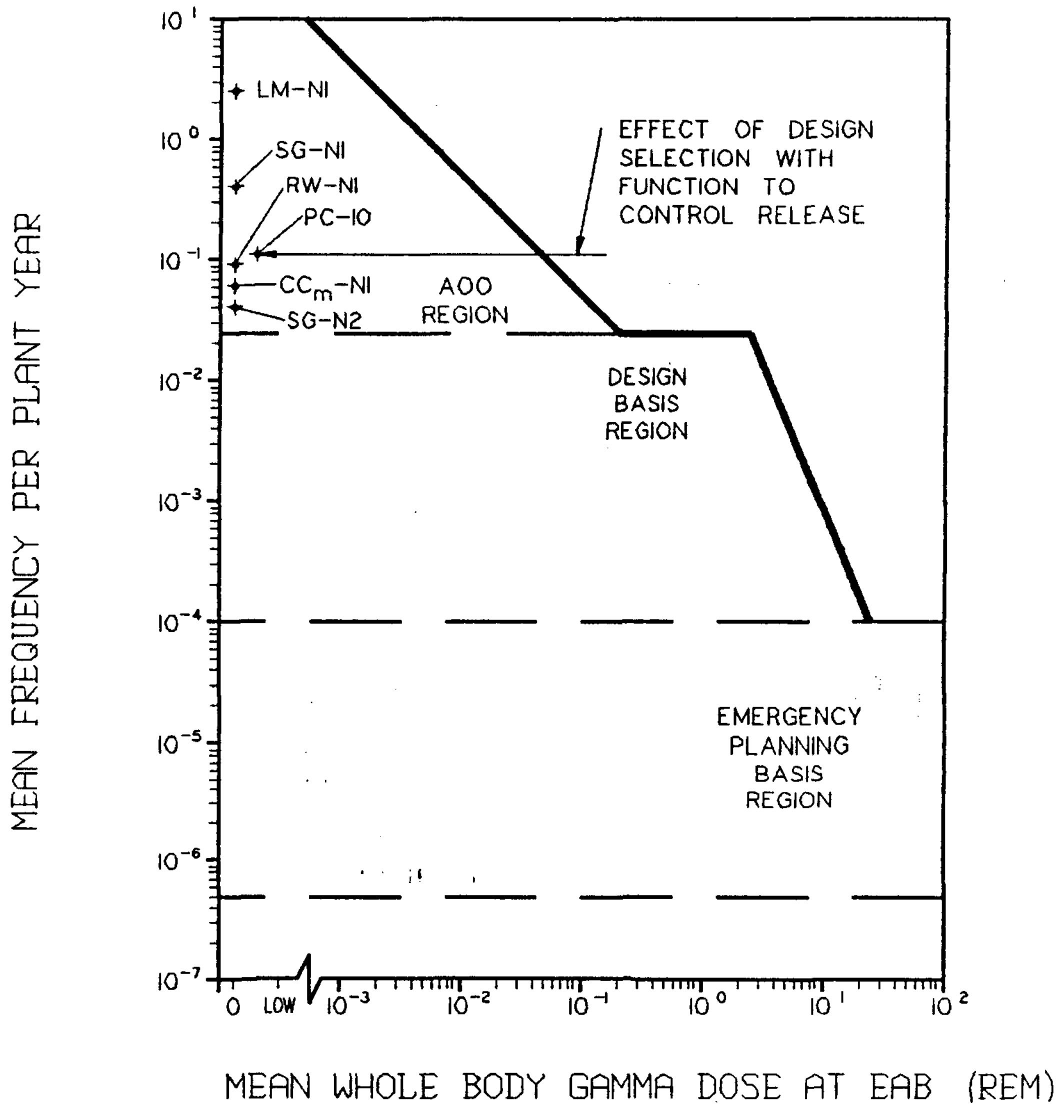
TABLE 3-1 (Cont'd.)

CC_m-X Accident family is initiated by a loss of main loop cooling caused by some event other than a loss of offsite power and turbine trip or an event external to the plant and results in a conduction cooldown transient.

CC_b-X Accident family is initiated by a main steam or feedwater pipe break and results in a conduction cooldown transient.

* Numerical designations generally proceed from the most consequential to the least consequential with some exceptions. For example, PC-6 would have higher consequences than PC-10. Those families labeled with an "N" designate a no dose event such as in SG-N1.

FIG 3-2
IDENTIFY PC-10 AS AN ANTICIPATED OPERATIONAL OCCURRENCE



As an example of the AOO selection process, consider the accident family PC-10. With its predicted mean frequency of .1 per plant year, PC-10 is included on Fig. 3-2. PC-10 is a very small leak in the primary coolant pressure boundary predicted to occur about once every ten years of plant operation. A description of the event is as follows.

1. The coated fuel particles retain most of the generated fission products and thereby keep the level of primary coolant activity low.

2. The helium purification system (HPS) operates and further reduces the primary coolant activity.

3. The reactor is at equilibrium 100% power.

4. A very small leak (between 3×10^{-5} to 0.03 sq in.) occurs in the primary coolant pressure boundary allowing primary coolant to escape.

5. The leak is detected and the reactor is shut down.

6. Core cooling is maintained using the Heat Transport System (HTS).

7. Some of the primary coolant is pumped to storage through the HPS.

8. The remainder of the primary coolant leaks to the reactor building and ultimately, after some hold up, to the environment.

While PC-10 clearly lies within the allowable dose limits, the line extending from the right of the event indicates what the dose from such a leak could be were it not for the performance of certain design selections which control the release of radioactivity. For instance, if the design selection for fuel particle coatings allowed much higher fission product releases during normal operation, the primary coolant activity levels could be higher than currently anticipated. Then if a leak were to occur the activity released and the resultant offsite doses could be higher than now predicted and could exceed the dose limits as pictured. Therefore, PC-10 is included as an AOO and is numbered AOO-5 in Table 3-2. The other AOOs identified in this manner are also listed in this table.

Actually, since the analyses to date deal only with potential releases of radioactivity from the primary coolant and the reactor core, all the candidate events shown in Fig. 3-2 are, in fact, AOOs. The radioactivity in the reactor core is also retained because of design selections. Therefore, the events which challenge the retention of primary coolant and core radionuclides involve design selections and are, therefore, events which become AOOs. Other events to be studied in the future may not qualify as anticipated operational occurrences. The simplest example of this is the neutron sources used in the startup of a reactor which even if damaged would not lead to a violation of Appendix I for the public (though, of course, the radioactivity has to be controlled to limit the potential doses to the power plant crew).

Five AOOs are listed in Table 3-2. Along with a number assigned to each of the AOOs, the table contains two other pieces of information about each of the AOOs. In the second column the dominant accident family, also called a consequence category, is listed. This designation comes from the PRA and is chosen to identify a group of events with a certain type of radionuclide release path. Knowing what the accident family category is, one can go back to the PRA event trees and determine which sequences lead to the AOO. So, listing the accident family assists in cross referencing between the AOOs and the PRA on which they're based. In the third column a brief descriptive name is provided. Further description of these AOOs is provided in the following paragraphs.

TABLE 3-2
ANTICIPATED OPERATIONAL OCCURRENCE (AOO)
SUMMARY TABLE

AOO Number	Accident Family	Anticipated Operational Occurrence
AOO-1	LM-N1	Main loop transient with forced core cooling
AOO-2	CC_m-N1	Loss of main and shutdown cooling loops
AOO-3	RW-N1	Control rod bank withdrawal with scram
AOO-4	SG-N1	Small steam generator leak
AOO-5	PC-10	Small primary coolant leak

Some AOOs include other accident families in addition to the dominant family selected. Several families have been grouped together under one AOO classification because of similarities in the scenarios and radionuclide release paths. Descriptions are provided for each secondary family as well as the dominant family.

AOO-1 is represented by a main loop transient where forced core cooling is available. The dominant accident family is designated LM-N1 in which a loss of main loop cooling occurs. It is intended to encompass most failures, originating within the plant, that preclude the Heat Transport System (HTS) from continuing to perform the function of removing reactor heat. The event is terminated by reactor shutdown and startup of the Shutdown Cooling System (SCS). Failures typifying the event include circulator trips and losses of feedwater.

A second contributor to AOO-1 is accident family LP-N1 which is a loss of offsite power and turbine trip. The event leads to a loss of the HTS and is terminated by reactor shutdown and startup of the SCS. The failure inititaing the event sequence, a loss of offsite power, originates outside the plant and in that respect differs from accident family LM-N1. Startup of the SCS is predicated upon successful start of a backup power supply. This accident has been included as portion of AOO-1 because of its similarity in terms of event sequence to other contributors to AOO-1. Its mean frequency, however, as pictured in Fig. 3-1 does not place it in the AOO region.

A third contributor to AOO-1 is accident family RS-N1 which is a plant transient calling for a reactor trip. The event differs from LM-N1 in that forced core cooling continues on the HTS rather than requiring SCS cooling. The events are grouped together because either the HTS or SCS can perform the function of removing reactor heat by forced helium circulation.

A final contributor to AOO-1 is a main loop transient which does not call for a reactor trip. Core cooling continues on the HTS with the circulator speed reduced to compensate for either a reduction in core power level or feedwater flow. Typical initiating events are a turbine trip, load rejection, sudden reduction in feedwater flow, inadvertent control rod insertion, main loop overcooling or loss of SCS heat exchanger cooling water during normal operation conditions.

AOO-2 encompasses failures similar to event LM-N1 in AOO-1 except that in AOO-2 the SCS fails to run. The event sequence is terminated with core cooling in the still pressurized module being provided by conductive and radiative heat transport to the Reactor Cavity Cooling System (RCCS). Convective flows developed within the reactor core aid in distributing heat from hotter to cooler core regions. Some of the failures in AOO-2 are failures originating within the plant that render both the HTS and SCS unable to perform their heat removal functions.

AOO-3 is an accident initiated by a spurious control rod bank withdrawal, but responded to properly with a scram of the control rods. This is designated accident family RW-N1. The response of the Plant Protection and Instrumentation System (PPIS) and the control rods is sufficiently prompt that the HTS remains on-line but the circulator speed is reduced in order to respond to the reduction in power level of the core.

AOO-4 is a leak in a one or more steam generator tubes. The event is similar to AOO-1 in that it leads to a loss of the HTS but because the failure leads to different design bases for certain design selections, it is treated as a separate event. In AOO-4 the steam generator leak is followed by reactor and main loop shutdown. The leaking steam generator is isolated from the feedwater and steam headers and its inventory dumped. Core cooling in this event is provided by the SCS. Two accident families, SG-N1 which is a small leak and SG-N2 which is a moderate leak, contribute to AOO-4.

A00-5 is a leak in the primary coolant pressure boundary. The dominant contributor is PC-10 which was discussed in some detail earlier. A second contributor to A00-5 is accident family PC-6. A factor of two on the mean frequency of PC-6 places its upper bound in the A00 region. This family is also addressed as a design basis event. The full area of the leak in this case ranges between .03 and 1 square inch. It is otherwise similar to PC-10 in that the HTS or SCS provide forced core cooling and pumpdown of the primary coolant by the helium purification system (HPS) is successful. Since the flow rate of primary coolant out the leak is higher in this case, the HPS is able to retain a smaller fraction of the primary coolant within the plant. The remainder of the primary coolant is released from the reactor building over a period of time. A00-5 is the only anticipated operational occurrence in which an offsite dose is incurred.

3.1.2 A00 Event Descriptions

Detailed descriptions of the A00s are summarized in Table 3-3 for ease of reference. The event descriptions are from the draft PRA (Ref. 4). The accident family is a consequence family which may include a number of individual event sequences from the event trees of the PRA. The dominant event sequence within a family is the one that contributes the most to the total frequency of the family. This dominant sequence is listed on the table in a form which indicates first the event tree symbol and then the alphabetic indicator of the sequence within the event tree. The family mean frequency is obtained by summing all sequence frequencies for the family from the event trees. This sum is multiplied by four, if appropriate, to obtain the total median frequency for a four module plant and then converted from median to mean by the appropriate conversion factor. Other accident families which contribute to a given A00 are listed as well along with their PRA identification information.

TABLE 3-3
AOO EVENT DESCRIPTIONS

LBE IDENTIFICATION NUMBER

AOO-1

EVENT DESCRITION

- Main loop cooling is lost either due to a failure originating within the plant or due to a loss of offsite power accompanied by a turbine generator trip.
- The PPIS signals a reactor trip and outer control rods are successfully inserted into the reactor core.
- Shutdown cooling is started either by the primary power supply or a backup power supply if all offsite and in-house power has been lost.
- The reactor vessel remains pressurized and heat is removed by the SCS via forced helium circulation.
- Radionuclides are retained within the fuel particles and reactor core graphite.
- Primary circuit radionuclides are retained within the primary coolant pressure boundary.

PRA IDENTIFICATION

Accident Initiating Event: Loss of main loop cooling
Accident Family: LM-N1
Dominant Event Sequence Within Family: HTS-AA
Family Mean Frequency: 2.5 per plant year

Other Accident Families: LP-N1, RS-N1
Dominant Event Sequences Within Families: LOSP-AA, RS-AA
Family Mean Frequencies: 1×10^{-2} per plant year, 40 per plant yr

LBE IDENTIFICATION NUMBER

A00-2

EVENT DESCRITION

- Main loop (HTS) cooling is lost due to some failure originating within the plant with the exception of a main circulator trip. This excludes a loss of offsite power as well as external initiating events.
- Immediate or delayed loss of feedwater flow to the steam generator is detected.
- The PPIS signals a main circulator trip in conjunction with a reactor trip signal.
- The reactor is successfully scrammed on outer control rods and the main circulator is tripped.
- The shutdown cooling system fails to provide core cooling due to either an independent failure or failure of a system common to both the HTS and SCS.
- The core is cooled by conductive and radiative heat transport to the RCCS.
- The reactor vessel remains pressurized during the cooldown transient.
- Radionuclides are retained within the fuel particles and reactor core graphite.
- Primary circuit radionuclides are retained within the primary coolant pressure boundary.

PRA IDENTIFICATION

Accident Initiating Event: Loss of main loop cooling
Accident Family: CC_m-N1
Dominant Event Sequence Within Family: HTS-AB
Family Mean Frequency: 6×10^{-2} per plant year

LBE IDENTIFICATION NUMBER

AOO-3

EVENT DESCRITION

- An outer control rod bank is spuriously withdrawn from the reactor core at the maximum withdrawal speed (1.2 in./s) to fullout.
- A reactor trip signal is initiated upon detection of high core power to helium flow.
- The reactor successfully scrams by outer control rod insertion.
- The HTS remains on line to provide core cooling, but is runback to respond to the reduced power level of the core following the reactor trip.
- Radionuclides are retained within the fuel particles and reactor core graphite.
- Primary circuit radionuclides are retained within the primary coolant pressure boundary.

PRA IDENTIFICATION

Accident Initiating Event: Control rod bank withdrawl
Accident Family: RW-N1
Dominant Event Sequence Within Family: RW-AA
Family Mean Frequency: 9×10^{-2} per plant year

LBE IDENTIFICATION NUMBER

AOO-4

EVENT DESCRITION

- A small steam generator leak occurs.
- Moisture ingresses into the primary system at a rate of 0.1 lbm/s.
- Moisture detection by the moisture monitors succeeds.
- The PPIS initiates a main circulator and reactor trip after receiving a signal from the moisture monitors.
- The reactor successfully scrams by outer control rod insertion and the main circulator trips.
- The PPIS initiates closure of the main steam and feedwater isolation valves.
- Following successful isolation, the steam generator dump system valves open and the water/steam inventory flows into the dump system tanks.
- The dump system valves reclose when the line pressure is slightly above primary coolant pressure.
- Core cooling is provided by the shutdown cooling system.
- Some core radionuclides are released into the primary circuit due to fuel hydrolysis and graphite oxidation.
- Primary circuit radionuclides are retained within primary coolant pressure boundary.

PRA IDENTIFICATION

Accident Initiating Event: Small steam generator leak
Accident Family: SG-N1
Dominant Event Sequence Within Family: SS-AA
Family Mean Frequency: 0.40 per plant year

Other Accident Families: SG-N2
Dominant Event Sequence Within Family: MS-AA
Family Mean Frequency: 4×10^{-2} per plant year

LBE IDENTIFICATION NUMBER

A00-5

EVENT DESCRITION

- A leak occurs in the primary coolant pressure boundary, the full flow area of which may range from 3×10^{-5} in.^{2} up to 1 in.^{2}.
- The leak is detected and a reactor trip initiated by the PPIS on low primary coolant pressure.
- Core cooling is maintained on either the HTS or the SCS.
- The helium purification system pumps down to storage some fraction of the primary coolant depending on the leak area.
- Radionuclides are retained within the fuel particles and reactor core graphite.
- Primary coolant circulating activity is released to the reactor building.
- Radionuclides are released to the environment after some retention in the reactor building due to the effects of deposition and volumetric holdup.

PRA IDENTIFICATION

Accident Initiating Event: Small primary coolant leak
Accident Family: PC-10, PC-6
Dominant Event Sequence Within Family: PC-AA, PC-BY
Family Mean Frequency: PC-10: .1 per plant year
PC-6: 2×10^{-2} per plant year

3.2 Design Basis Events

3.2.1 Selection of DBEs

A similar approach is utilized to select Design Basis Events (DBEs). DBEs are identified as those families of events whose mean frequencies fall within the design basis region and would violate the dose criteria were it not for design selections that control radioactivity release.

Those events identified in the PRA and falling within the design basis region, that is those with mean frequencies between 0.025 and 1×10^{-4} per plant year, are shown in Fig. 3-3. Each of these event families is considered as a potential DBE. For example, consider the event family designated on the figure as CC_L-N1.

CC_L-N1 is a station blackout in which all ac electrical power is lost. This event is predicted to occur at a mean value of 3×10^{-4} per plant year. The specific events involved are as follows.

1. The reactor is at equilibrium 100% power.

2. Offsite electrical power is lost.

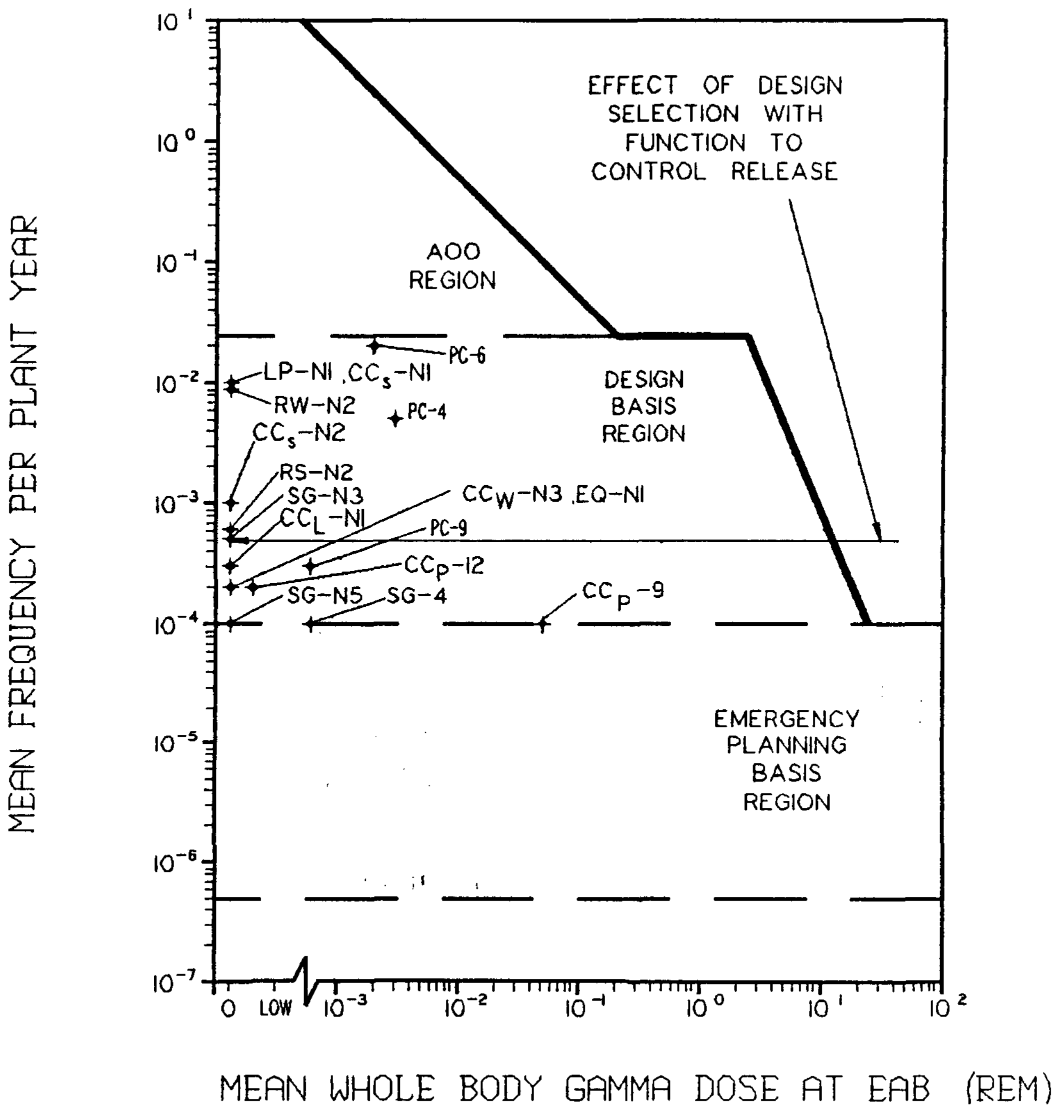
3. Turbine-generator fails to remain on-line. Normal, in-house power is lost. HTS is unavailable.

4. The main circulator trips and outer control rods are inserted into the core.

5. Standby power source fails to start. No power is available to operate the Shutdown Cooling System.

6. Primary coolant remains at pressure.

FIG 3-3
IDENTIFY CC_L -NI AS A
DESIGN BASIS EVENT



7. Decay heat removal from the core is accomplished via localized convection, conduction and radiation. Fuel temperatures remain below maximum normal operating limits.

8. Vessel temperatures are kept within design limits by the passive reactor cavity cooling system (natural circulation of air).

9. Primary coolant and fuel inventories are retained. Therefore, there is no radioactivity release.

As seen in Fig. 3-3 the frequency of this event family, being in excess of 10^{-4}, puts CC_L-N1 in the design basis region. Additionally, while the event leads to no public exposure, and therefore falls well within the dose limit for this region, this limit could be exceeded were it not for certain features selected in the design of the plant. The horizontal arrow extending from the right of CC_L-N1, in the figure, is intended to show an estimate of what the dose from this sequence of failures might be if the plant was not designed to reject decay heat via natural convection, conduction and radiation and not designed with high temperature fuel. Without the "conduction cooldown" feature this sequence of events, described by CC_L-N1, would result in a total loss of core cooling and loss of fuel integrity and the dose limits could be exceeded as shown. Because of this, CC_L-N1 is included as a DBE and labeled DBE-1.

Other events, not shown on Fig. 3-3 but visible on 3-1, might fall very close to but not actually within the frequencies bounding the design basis region. Also identified as DBEs are any additional event families whose agreed upper margin frequencies fall within the design basis region and who otherwise satisfy the requirements as described above. A review of Fig. 3-1 shows SG-9 satisfies the frequency requirement, taking a factor of two to cover the margin.

A listing of all the event families identified as DBEs is provided in Table 3-4. For each DBE within the table a convenient DBE identifying number is first given. As was done for the table of AOOs, this is then followed by the accident family category and a brief descriptive name. Additional descriptions of the DBEs are given in the following paragraphs.

DBE-1 is a loss of offsite power followed by inadvertent turbine trip and failure to successfully start the SCS. DBE-1 is generally similar to AOO-2 in that both encompass a loss of forced core cooling with the event sequence terminated by a pressurized conduction cooldown and no offsite release. In the case of DBE-1, however, the course of the event sequence is strongly influenced by the loss of offsite power and the failure of the SCS to start which is typically attributable to a failure of the standby power source to start.

DBE-2 is an HTS transient which requires a reactor trip in which the automatic scram system fails to insert the outer control rods. Reactor scram is, however, accomplished by insertion of the Reserve Shutdown System (RSS) control poison. The main circulator is tripped and the SCS subsequently serves to remove heat from the reactor core.

DBE-3 is initiated by a spurious control rod bank withdrawal with successful scram of the reactor. It is similar to AOO-3 with the exception that the SCS provides forced core cooling rather than the HTS.

DBE-4 is also initiated by a spurious control rod bank withdrawal but in this case both the HTS and SCS fail to provide forced helium circulation. The reactor is scrammed on the control rods and heat is removed by conduction and radiation to the RCCS.

TABLE 3-4
DESIGN BASIS EVENT (DBE) SUMMARY TABLE

DBE Number	Accident Famiy	Design Basis Event
DBE-1	CC_L-N1	Loss of HTS and SCS cooling
DBE-2	RS-N2	HTS transient without control rod trip
DBE-3	RW-N2	Control rod withdrawal without HTS cooling
DBE-4	CC_W-N3	Control rod withdrawal without HTS and SCS cooling
DBE-5	EQ-N1	Large earthquake with SCS cooling
DBE-6	SG-N2	Moisture inleakage with SCS cooling
DBE-7	CC_S-N2	Moisture inleakage without SCS cooling
DBE-8	SG-N3	Moisture inleakage with moisture monitor failure
DBE-9	SG-4	Moisture inleakage with isolation and failure to reclose the dump system
DBE-10	PC-4	Primary coolant leak with HTS cooling
DBE-11	CC_P-9	Primary coolant leak without HTS and SCS cooling

DBE-5 is initiated by a large earthquake with a ground acceleration of between 0.3 and 1.5 g. Forced core cooling is provided by the SCS. The analyses have assumed that the nonseismic category I codes are used in most of the plant systems, such as the electrical distribution system, HTS and SCS. The assessments indicate that these principal nonseismic category I systems will survive the event and cool the core. The core and control rods and associated systems were assumed to be seismic category I and therefore are found to have much greater margin for survival in this accident.

DBE-6 is a moderate leak in the steam generator followed by a reactor and main loop shutdown. The leaking steam generator is isolated from the feedwater and steam headers and its inventory dumped. Core cooling is provided by the SCS. This DBE comes from the SG-N2 family. Recall that SG-N2 was included in AOO-4 as well. Because the agreed lower margin frequency falls within the DBE region, this accident family is evaluated with respect to the dose limits of both the AOO and DBE regions.

A second contributor to DBE-6 is a small steam generator leak designated SG-N5 in the PRA. Following the leak, the steam generator is successfully isolated but the dump valves fail to open. Core cooling continues on the SCS. Because the leak is small and the core continues to be cooled, the relief valves are not expected to open.

Another small steam generator leak also contributes to DBE-6. This event has been designated SG-N6 in the PRA. In the event sequence, moisture is detected and isolation of the steam generator is attempted. The feedwater block valves, however, fail to close and moisture continues to ingress into the primary system until operator intervention eventually terminates the leak. The core is cooled on the SCS. Because the leak is small and the core is being cooled down, the primary relief valves are again not expected to lift.

The final contributor to DBE-6 is a moderately sized steam generator leak in which moisture monitor detection fails. The reactor is tripped on high pressure. Isolation of the steam generator is initiated by the operator and the core is cooled on the SCS. Because of the delay in isolating the steam generator, sufficient moisture ingresses into the system to lift the primary relief valves. The valves do, however, reseat following pressure relief. This accident family is designated SG-9 in the PRA.

DBE-7 is similar to DBE-6 with the exception that SCS cooling fails. Heat removal is therefore by conduction and radiation to the RCCS and the affected module experiences a pressurized conduction cooldown. The leak size range of DBE-7 includes moderate as well as small steam generator leaks.

DBE-8 is a small steam generator leak followed by failure of the moisture monitors to detect the inleakage into the primary circuit. The reactor is tripped on high primary system pressure using both the outer control rods and RSS, both being required to offset the positive reactivity effect of the moisture inleakage into the reactor core. Forced core cooling continues on the HTS. The leaking steam generator is isolated and dumped.

DBE-9 is a small steam generator leak in which the steam generator dump system fails to operate properly. After successful moisture detection and steam generator isolation, the steam generator inventory is dumped, but the dump system valves fail to reclose. A release path to the environment is available through the open dump system. Forced core cooling is provided by the SCS.

DBE-10 is characterized by primary coolant leaks which were found in the safety assessment of the modular HTGR to dominate risk to the public. Leak sizes range from very small to the largest size expected to occur in the design basis region. In all cases the reactor is shut down following detection of the leak. A portion of the primary coolant leaks into the reactor building while the remainder may be pumped to storage by the HPS. The moderate, though less likely leaks differ from the smaller leaks in four ways. The higher flow rates can give rise to greater liftoff of plated out material in the primary coolant loop. The shorter transient time limits or completely precludes the extent to which the coolant can be pumped to storage, thus increasing that released. The greater differential pressures associated with the large leaks increase the loads on various structures within the primary circuit. Finally, the reactor building response to different sized leaks can vary.

DBE-10 is dominated by a moderate primary coolant leak with forced helium circulation. It comes from the PRA accident family PC-4. This event has a range of leak flow areas running between 1 and 13 square inches. The 13 square inch leak size is chosen as representative on the basis that it is the largest leak that will occur with a frequency in the design basis region. The reactor is shut down following detection of the leak. Because of the large leak area involved, all of the primary coolant depressurizes quickly into the reactor building and subsequently to the environment.

A second contributor to DBE-10 is a smaller primary coolant leak designated PC-9 in the PRA. PC-9 occurs more frequently than PC-4 and encompasses flow areas between 3×10^{-5} and 0.03 square inches. Pumpdown of the primary coolant is unsuccessful, allowing all of the primary coolant to depressurize into the reactor building and subsequently to the atmosphere. Core cooling continues on either the HTS or SCS following a reactor trip on low primary coolant pressure.

The final contributor to DBE-10 is PC-6. This accident family was included as a contributor to AOO-5 as well because the upper margin frequency falls within the AOO region. Recall that PC-6 has a leak area ranging from 0.03 to 1 square inch. HPS pumpdown is successful and core cooling is provided by either the HTS or SCS following reactor trip.

DBE-11 is initiated by a primary coolant leak and in that respect is similar to DBE-10. However, DBE-11 encompasses single, common mode and multiple failures that preclude both the HTS and SCS from providing core cooling. In this case the events are terminated with core cooling being provided by conduction and radiation to the RCCS.

DBE-11 is dominated by accident family CC_p-9 in which a moderate primary coolant leak occurs with a flow area of between 0.03 and 1 square inch. Forced helium circulation is not provided by either the HTS or SCS. Pumpdown of the primary coolant is successful but because of the size of the leak, retention of radionuclides is minimal.

A second contributor to DBE-11 is CC_p-12 in which a very small leak in the primary coolant pressure boundary occurs. The leak flow area is between 3×10^{-5} and 2×10^{-3} square inches. Pumpdown in this case also succeeds but because of the smaller leak area, retention of radionuclides is significant.

3.2.2 DBE Event Description

Table 3-5 provides a summary sheet for each DBE which provides additional detail in the style similar to the summary sheets for the AOOs. Again, accident families, dominant event sequences and family mean frequencies are from the PRA. For each DBE, a detailed description of the dominant accident family is included. Other contributing families are not described in detail but their PRA identification is given.

3.3 Emergency Planning Basis Events

3.3.1 Selection of EPBEs

Emergency Planning Basis Events (EPBEs) are the dose dominant events whose upper margin frequencies fall within the frequency range encompassed by the emergency planning basis region.

TABLE 3-5
DBE EVENT DESCRIPTIONS

LBE IDENTIFICATION NUMBER

DBE-1

EVENT DESCRITION

- Loss of offsite power and a turbine generator trip initiate a loss of the primary coolant loop.
- Loss of the main loop initiates a reactor trip by the PPIS.
- Outer control rods are successfully inserted into the reactor core.
- The shutdown cooling system fails to start due to failure of the backup power supply.
- The reactor cavity cooling system (RCCS) succeeds in functioning to remove decay heat from the reactor core.
- Heat is transported to the RCCS via conductive and radiative heat transport. Heat is redistributed from hotter to cooler portions of the reactor core due to natural convective flows being esablished.
- The reactor vessel remains pressurized and primary coolant boundary integrity is maintained.
- Radionuclides are retained within the fuel particles and reactor core graphite.
- Primary circuit radionuclides are retained within the primary coolant pressure boundary.

PRA IDENTIFICATION

Accident Initiating Event: Loss of offsite power and turbine trip
Accident Family: CC_L-N1
Dominant Event Sequence Within Family: LOSP-AB
Family Mean Frequency: 3.0×10^{-4} per plant year

LBE IDENTIFICATION NUMBER

DBE-2

EVENT DESCRIPTION

- An HTS transient event occurs requiring a reactor trip.
- The automatic scram system using outer control rods fails to successfully operate.
- A delayed reactor trip on the reserve shutdown system succeeds.
- The SCS is successfully started following the main loop trip.
- Core heat is removed by forced circulation of helium using the SCS.
- The reactor vessel remains pressurized and primary coolant boundary integrity is maintained.
- Radionuclides are retained within the fuel particles and reactor core graphite.
- Primary circuit radionuclides are retained within the primary coolant pressure boundary.

PRA IDENTIFICATION

Accident Initiating Event: Plant transient requiring reactor scram
Accident Family: RS-N2
Dominant Event Sequence Within Family: RS-AB
Family Mean Frequency: 6×10^{-4} per plant year

LBE IDENTIFICATION NUMBER

DBE-3

EVENT DESCRIPTION

- An outer control rod bank is spuriously withdrawn from the reactor core at the maximum withdrawal speed (1.2 in./sec.) to full out.
- A reactor trip signal is initiated upon detection of high power to flow.
- The reactor is scrammed on outer control rods.
- The HTS fails to remain on line following the transient.
- The SCS starts following failure of the HTS.
- Core heat removal is accomplished via forced helium circulation by the SCS.
- The reactor vessel remains pressurized and primary coolant boundary integrity is maintained.
- Radionuclides are retained within the fuel particles and reactor core graphite.
- Primary circuit radionuclides are retained within the primary coolant pressure boundary.

PRA IDENTIFICATION

Accident Initiating Event: Control Rod Bank Withdrawal
Accident Family: RW-N2
Dominant Event Sequence Within Family: RW-AB
Family Mean Frequency: 9×10^{-3} per plant year

LBE IDENTIFICATION NUMBER

DBE-4

EVENT DESCRIPTION

- An outer control rod bank is spuriously withdrawn from the reactor core at the maximum withdrawal speed (1.2 in./sec.) to full out.
- A reactor trip signal is initiated upon detection of high power to flow.
- The reactor successfully scrams by outer control rod insertion.
- Both HTS and SCS cooling is unsuccessful.
- Core heat removal is accomplished by conductive and radiative heat transport to the RCCS.
- Convective flows established within the reactor core serve to redistribute heat from hotter to cooler regions.
- The reactor vessel remains pressurized and primary coolant boundary integrity is maintained.
- Radionuclides are retained within the fuel particles and reactor core graphite.
- Primary circuit radionuclides are retained within the primary coolant pressure boundary.

PRA IDENTIFICATION

Accident Initiating Event: Control Rod Bank Withdrawal
Accident Family: CC_W-N3
Dominant Event Sequence Within Family: RW-AC
Family Mean Frequency: 2×10^{-4} per plant year

LBE IDENTIFICATION NUMBER

DBE-5

EVENT DESCRIPTION

- A large earthquake occurs with an acceleration of 0.3 g.
- The main loop trips.
- The reactor is successfully tripped on the outer control rods following a main circulator trip.
- The shutdown cooling system provides forced core cooling.
- The reactor vessel remains pressurized and primary coolant boundary integrity is maintained.
- Radionuclides are retained within the fuel particles and reactor core graphite.
- Primary circuit radionuclides are retained with the primary coolant pressure boundary.

PRA IDENTIFICATION

Accident Initiating Event: Earthquake between 0.3 and 0.5 g
Accident Family: EQ-N1
Dominant Event Sequence Within Family: EQ-AA
Family Mean Frequency: 2×10^{-4} per plant year

Other Accident Families: EQ-N2
Dominant Event Sequence Within Family: EQ-AC
Family Mean Frequency: 6×10^{-5} per plant year

LBE IDENTIFICATION NUMBER

DBE-6

EVENT DESCRIPTION

- A moderate steam generator tube leak occurs.
- Moisture ingresses into the primary system at a rate equivalent to that associated with an offset tube rupture (Approximately 12 lbm/sec).
- Moisture detection by the moisture monitors succeeds.
- The PPIS initiates a main circulator and reactor trip after receiving a signal from the moisture monitors.
- The main circulator successfully trips.
- The reactor successfully scrams by outer control rod insertion.
- The PPIS initiates closure of the main steam and feedwater isolation valves.
- Following successful isolation, the steam generator dump system valves open and the water/steam inventory flows into the dump system tanks.
- The dump system valves reclose when the line pressure is slightly above primary coolant pressure.
- Core cooling is provided by the SCS which successfully starts and runs when called upon following the main circulator trip.
- The reactor vessel remains pressurized and primary coolant boundary integrity is maintained.
- Some radionuclides are released into the primary circuit due to fuel hydrolysis and graphite oxidation.
- Primary circuit radionuclides are retained within the primary coolant pressure boundary.

PRA IDENTIFICATION

Accident Family Initiating Event: Moderate steam generator leak
Accident Family: SG-N2
Dominant Event Sequence Within Family: MS-AA
Family Mean Frequency: 4×10^{-2} per plant year

Other Accident Families: SG-N5, SG-N6, SG-9
Dominant Event Sequences Within Families: SS-AH, SS-AN, MS-CA
Family Mean Frequencies: 1×10^{-4}, 5×10^{-5}, 5×10^{-5} per plant year

LBE IDENTIFICATION NUMBER

DBE-7

EVENT DESCRIPTION

- A moderate steam generator tube leak occurs
- Moisture ingresses into the primary system at a rate of approximately 12 lbm/sec.
- Moisture detection by the moisture monitors succeeds.
- The PPIS initiates a main circulator and reactor trip after receiving a signal from the moisture monitors.
- The main circulator successfully trips.
- The reactor successfully scrams by outer control rod insertion.
- The PPIS initiates closure of the main steam and feedwater isolation valves.
- Following successful isolation, the steam generator dump system valves open and the water/steam inventory flows into the dump system tanks.
- The dump system valves reclose when the line pressure is slightly above primary coolant pressure.
- The SCS does not start on demand or does not run for a sufficient amount of time to adequately cool the core.
- Heat removal is provided by conduction and radiation to the RCCS.
- The reactor vessel remains pressurized and primary coolant boundary integrity is maintained.
- Some radionuclides are released into the primary circuit due to graphite oxidation, fuel hydrolysis and thermal effects.
- Primary circuit radionuclides are retained within the primary coolant pressure boundary.

PRA IDENTIFICATION

Accident Family Initiating Event: Moderate steam generator leak
Accident Family: CC_S-N2

Dominant Event Sequence Within Family: MS-AB
Family Mean Frequency: 1×10^{-3} per plant year
Other Accident Families: CC_S - N1

Dominant Event Sequence Within Family: SS-AB
Family Mean Frequency: 1×10^{-2}

LBE IDENTIFICATION NUMBER

DBE-8

EVENT DESCRIPTION

- A small steam generator leak occurs
- Moisture ingresses into the primary system at a rate of 0.1 lbm/sec.
- Moisture monitors fail to detect the moisture.
- High primary system pressure causes a reactor trip on outer control rods and RSS.
- The HTS is ramped down to compensate for the decreased heat load.
- Feedwater flow to the steam generator is decreased due to the lower primary coolant flow rate.
- Heat removal is accomplished via forced circulation on the HTS.
- The steam generator is isolated and dumped by the operator.
- The SCS is automatically started to cool the core.
- The reactor vessel remains pressurized and primary coolant boundary integrity is maintained.
- Some radionuclides are released into the primary circuit due to fuel hydrolysis and graphite oxidation.
- Primary circuit radionuclides are retained within the primary coolant pressure boundary.

PRA IDENTIFICATION

Accident Initiating Event: Small steam generator leak
Accident Family: SG-N3
Dominant Event Sequence Within Family: SS-BY
Family Mean Frequency: 5×10^{-4} per plant year

LBE IDENTIFICATION NUMBER

DBE-9

EVENT DESCRIPTION

- A small steam generator leak occurs.
- Moisture ingresses into the primary system at a rate of 0.1 lbm/sec.
- Moisture detection by the moisture monitors succeeds.
- The PPIS initiates a main circulator trip and reactor trip after receiving a signal from the moisture monitors.
- The main circulator successfully trips.
- The reactor successfully scrams by outer control rod insertion.
- The PPIS initiates closure of the main steam and feedwater isolation valves.
- Following successful isolation, the steam generator dump system valves open and the water/steam inventory flows into the dump system tanks.
- The dump system valves fail to reclose when the line pressure is slightly above primary coolant pressure thereby providing a release pathway to the atmosphere.
- Core cooling is provided by the SCS which successfully starts and runs when called upon following the main circulator trip.
- Some radionuclides are released from the core into the primary circuit due to fuel hydrolysis and graphite oxidation.
- Radionuclide release due to fuel hydrolysis and graphite oxidation as well as the primary coolant circulating radionuclide inventories are released through the open steam generator dump system, through the tank relief valve and finally to the environment.

PRA IDENTIFICATION

Accident Initiating Event: Small steam generator leak
Accident Family: SG-4
Dominant Event Sequence Within Family: SS-AE
Family Mean Frequency: 1×10^{-4} per plant year

LBE IDENTIFICATION NUMBER

DBE-10

EVENT DESCRIPTION

- A moderate sized leak (approximately 13 $in.^2$) in the primary coolant pressure boundary occurs.
- Low primary coolant pressure initiates a PPIS reactor trip signal.
- The reactor is successfully tripped using the primary scram system and outer control rods.
- Forced core cooling of the reactor core is maintained using the HTS.
- Pumpdown of the primary coolant through the HPS is ineffective.
- Radionuclides are retained within the fuel particles and reactor core graphite.
- Primary coolant circulating activity is released to the reactor building through the breach in the primary coolant pressure boundary.
- Some plated out radionuclides are lifted off primary circuit surfaces due to high helium velocities and released into the reactor building.
- Radionuclides in the reactor building are subsequently released to the atmosphere through the building dampers.

PRA IDENTIFICATION

Accident Initiating Event: Moderate primary coolant leak
Accident Family: PC-4
Dominant Event Sequence Within Family: PC-DE
Family Mean Frequency: 5×10^{-3} per plant year

Other Accident Families: PC-6, PC-9
Dominant Event Sequences: PC-BY, PC-AB
Family Mean Frequencies: 2×10^{-2}, 3×10^{-4} per plant year

LBE IDENTIFICATION NUMBER

DBE-11

EVENT DESCRIPTION

- A small leak (0.05 in.^{2}) in the primary coolant pressure boundary occurs.
- Low primary coolant pressure initiates a PPIS reactor trip signal.
- The reactor is tripped using the primary scram system and outer control rods.
- Forced circulation of helium is provided by the HTS for 15 hours before the system fails.
- Attempts to cool the core via the SCS are unsuccessful.
- Core cooling is provided by conductive and radiative heat transport to the RCCS.
- A fraction of the primary coolant is pumped down to storage through the HPS.
- Some release of radionuclides from the fuel particles and reactor core graphite is incurred due to excessive temperatures.
- Radionuclides released from the fuel as well as primary coolant circulating activity are released through the leak area into the reactor building until the reactor vessel pressure equilibrates with the reactor building pressure and core temperatures begin to decrease.
- Radionuclides in the reactor building are subsequently released to the atmosphere.

PRA IDENTIFICATION

Accident Initiating Event: Small primary coolant leak
Accident Family: CC_p-9
Dominant Event Sequence Within Family: PC-CP
Family Mean Frequency: 1 x 10^{-4} per plant year

Other Accident Families: CC_p-12
Dominant Event Sequences Within Families: PC-AJ
Family Mean Frequencies: 2 x 10^{-4} per plant year

As an example of the application of the definition of an EPBE consider Fig. 3-4. Shown in the figure are all those accident families evaluated in the PRA and whose expected frequencies fall within the emergency planning basis region. As can be seen, the highest dose family in the region is the spurious control rod withdrawal category designated CC_w-2. Details of this event are as described below.

1. Reactor is at equilibrium 100% power.

2. A spurious control rod withdrawal begins.

3. An attempt is made by the primary scram system to trip the control rods and they do not succeed.

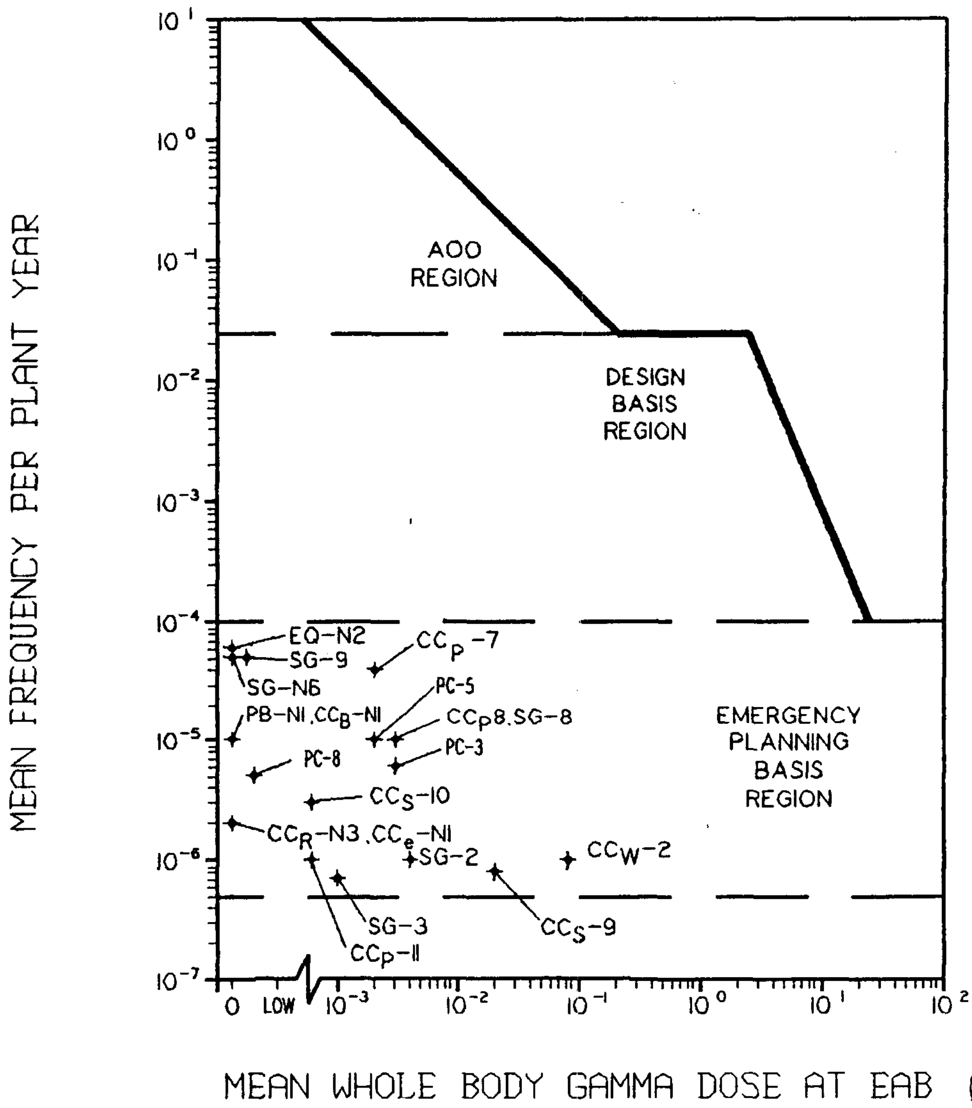
4. The rapid increase in core temperature causes the HTS and SCS to not be utilized.

5. The RCCS cooling of the reactor continues.

6. The pressure rise within the primary coolant system is sufficiently great because of temperature increase to lift the relief valves, therefore, releasing some of the primary coolant and some of the fission products which are released from the hot spots in the reactor core which were created during the initial reactivity transient. Following relief of excessive pressure, the valves reclose.

7. The reserve shutdown system is inserted.

FIG 3-4
IDENTIFY CCw -2 AS AN
EMERGENCY PLANNING BASIS EVENT



Since it is the largest dose event in the range of frequencies between 1×10^{-4} and 5×10^{-7}, CC_W-2 is selected as an EPBE and is designated as EPBE-2.

Other events of lesser dose but which still contribute significantly to the emergency planning basis region are included as EPBEs. Judging what constitutes a significant contribution has been done by including any event that has a dose of greater than about .1% of the PAG dose limits of 1 rem whole body and 5 rem thyroid. Table 3-6 lists all the events identified as EPBEs.

Five events are listed in the table. EPBE-2 is the dominant dose contributor in the region and has already been addressed in some length. Other EPBEs within the table are summarized below.

EPBE-1 is initiated by a large primary coolant leak. This EPBE is derived from accident family PC-3. This family in the PRA is for a range of leak areas above 13 $in.^2$. The frequency of this range of leak sizes is dominated by the smaller leaks and is, therefore, near 10^{-4}. In fact, the leak size of 13 $in.^2$ is taken into the DBE region. In choosing an EPBE which then becomes an event in the licensing basis work, an event is chosen at the bottom of a frequency range and the largest area at the bottom of that range. The leak area for EPBE-1 is, therefore, chosen as being 30 $in.^2$ and representative of a frequency of 6×10^{-6}. The reactor is tripped in this event and the HTS or SCS provides the core cooling.

Included as a contributor to EPBE-1 is a moderately sized primary coolant leak between 0.03 and 1 sq in. in which reactor building ventilation fails to operate as designed. This event includes the failure to disengage the reactor building HVAC fans. This failure reduces the building's ability to holdup released radioactivity and results in higher doses than would otherwise occur for leaks of this size. In this case, the release is increased by the ineffectiveness of the HPS in pumping the primary coolant to storage.

TABLE 3-6
EMERGENCY PLANNING BASIS EVENT (EPBE)
SUMMARY TABLE

EPBE Number	Accident Family	Emergency Planning Basis Event
EPBE-1	PC-3	Large primary coolant leak
EPBE-2	CC_w-2	Control rod withdrawal without scram
EPBE-3	CC_s-9	Moisture inleakage with open dump system and no forced cooling
EPBE-4	CC_p-8	Primary coolant leak with no forced cooling
EPBE-5	SG-8	Moisture inleakage with SCS cooling and dump system failure

A third contributor to EPBE-1 is a smaller primary coolant leak in the range of 2×10^{-3} to 0.03 in.^{2}. This event is designated PC-8 in the PRA. Forced core cooling is provided in this case by either the HTS or SCS. Pumpdown of the primary coolant is successful in retaining a fraction of the radionuclides present as circulating activity. Release to the environment is increased, however, in that the reactor building ventilation fans are not disengaged following detection of the leak. Normally, because the leak area is so small, a significant fraction of radionuclides released to the reactor building would be retained due to deposition effects and holdup within the building.

EPBE-2 was previously discussed.

EPBE-3 is similar to DBE-9 but is a moderate leak and includes a loss of forced circulation. Like DBE-9 a leak occurs in which the mutiple dump valves fail to reshut after discharging the steam generator water inventory. However, in EPBE-3 the SCS fails to successfully provide core cooling until the HTS is restored to service. Core cooling is via conduction and radiation to the RCCS. These failures allow the primary coolant and any incremental fuel releases, due to the conduction cooldown temperature transient, to escape to the atmosphere and bypass the reactor building. This event is CC_s-9 in the PRA.

A second contributor to EPBE-3 is identical to CC_s-9 except that the leak is small. This accident family is designated CC_s-10 in the PRA. The effect of the smaller leak size is the significantly longer time required for depressurization through the leak in the tube. The consequences of this event are, therefore, very small compared to CC_s-9.

EPBE-4 is initiated by a leak in the primary coolant pressure boundary in which forced helium circulation fails to be provided. The size of the leak ranges from 1 to 13 in.^{2} and is designated as accident family CC_p-8. Because the leak is so large, pumpdown of the primary coolant is ineffective and releases into the reactor building are almost immediately released to the environment through the building dampers. Core cooling is provided by conduction and radiation of heat to the RCCS.

A smaller primary coolant leak in the range of 2×10^{-3} to 0.03 in.^{2} contributes to EPBE-4 as well. It is designated CC_p-7 in the PRA. In this case, pumpdown of the primary coolant does provide some fission product attenuation. Forced core cooling is lost and heat removal is by conduction and radiation to the RCCS.

A third contributor to EPBE-4 is PRA accident family CC_p-11. This family consists of primary coolant leaks with loss of forced core cooling in the size range of 3×10^{-5} to 2×10^{-3} in.^{2}. Pumpdown by the HPS in this case fails. Core cooling is provided by conduction and radiation to the RCCS.

EPBE-5 is similar to DBE-9 except the event is initiated by a moderator steam generator leak rather than a small leak. The reactor is tripped following detection of the leak by the moisture monitors and the steam generator is successfully isolated. The dump valves open and the steam generator water/steam inventory is placed into the dump tanks. The valves, however, fail to reclose and the primary coolant radionuclide inventory is depressurized through the leaking tube to the environment.

EPBE-5 also includes a moderate steam generator leak in which the steam generator dump valves do not successfully dump the water inventory from the affected module. This failure allows more water/steam ingress to the primary coolant system than otherwise would occur which leads to lifting of the primary coolant relief valve. The relief valve does not reseat after successfully opening. This in turn leads to venting the primary coolant and some liftoff of plated out material to the reactor building.

Also included in EPBE-5 is a moderate steam generator leak in which feedwater is not successfully isolated from the affected module. This failure allows more water/steam ingress to the primary coolant system than otherwise would occur which leads to lifting of the primary coolant relief valve. The relief valve does not reseat after successfully opening. This is turn leads to venting the primary coolant and some liftoff to the reactor building.

3.3.2 EPBE Event Descriptions

Summary sheets for the EPBEs are provided in Table 3-7. They follow a format similar to those for the AOOs and the DBEs.

4. REQUESTED NRC RESPONSE

This document has been prepared for submittal to the Nuclear Regulatory Commission (NRC) in support of the HTGR Licensing Plan (Ref. 5). It is submitted for their information and review. The NRC is requested to address and respond to the following questions on the selection of the Licensing Basis Events for the MHTGR given in Section 4:

1. Does the NRC agree that the choice of licensing basis events follows the bridging method?

2. Are the licensing basis events chosen a complete selection?

3. Are the anticipated operational occurrences and the design basis events chosen the appropriate and complete set of events to be included in the PSID?

4. Are the emergency planning basis events chosen the appropriate set of events to be emphasized in choosing the radius of the emergency planning zone for a plant?

TABLE 3-7
EPBE EVENT DESCRIPTIONS

LBE IDENTIFICATION NUMBER

EPBE-1

EVENT DESCRIPTION

- o A large leak occurs in the primary coolant pressure boundary, the full flow area of which is between 13 $in.^2$ and 30 $in.^2$.
- o The leak is detected and a reactor trip initiated by the PPIS on low primary coolant pressure.
- o Core cooling via forced helium circulation is maintained by either the HTS or SCS.
- o The helium purification system is ineffective in pumping down any primary coolant to storage due to the large leakage area and associated high velocity helium flow.
- o Radionuclides are retained in the fuel particles and core graphite.
- o Primary coolant circulating activity is released to the reactor building through the breach in the primary coolant pressure boundary.
- o Some plated out radionuclides are lifted off of primary system surfaces due to high helium velocities and released into the reactor building.
- o Radionuclides in the reactor building are released directly to the environment through the open building dampers.

PRA IDENTIFICATION

Accident Initiating Event: Large primary coolant leak
Accident Family: PC-3
Dominant Event Sequence Within Family: PC-DP
Family Mean Frequency: 6×10^{-6} per plant year

Other Accident Families: PC-5, PC-8
Dominant Event Sequences: PC-BZ, PC-AT
Family Mean Frequencies: 1×10^{-5}, 5×10^{-6} per plant year

LBE IDENTIFICATION NUMBER

EPBE-2

EVENT DESCRIPTION

- A control rod bank is spuriously withdrawn from the reactor core at the maximum withdrawal speed (1.2 in./sec) to full out.
- A reactor trip signal is initiated upon detection of high power to helium flow.
- Control rods fail to be successfully inserted into the reactor core.
- Rapid temperature increases due to the positive reactivity insertion preclude the operation of either the HTS or SCS.
- Core cooling is maintained by conductive and radiative heat transport to the RCCS.
- The primary relief train opens to relieve the increase in pressure in the reactor vessel.
- Reactivity control is restored due to insertion of the RSS.
- Some radionuclides are released from the fuel due to excessive temperature conditions resulting from the initial rod bank withdrawal and subsequent failure to provide forced helium circulation.
- Some fuel particle releases and primary coolant circulating activity are released into the reactor building through the primary relief train.
- The relief train recloses when the closure setpoint is reached.
- Radionuclides in the reactor building are subsequently released to the environment.

PRA IDENTIFICATION

Accident Initiating Event: Control Rod Withdrawal
Accident Family: CC_W-2

Dominant Event Sequence within Family: RW-AE
Family Mean Frequency: 1×10^{-6} per plant year

LBE IDENTIFICATION NUMBER

EPBE-3

EVENT DESCRIPTION

- A moderate steam generator tube leak occurs.
- Moisture ingresses into the primary system at a rate equivalent to that associated with an offset tube rupture (12 lbm/sec).
- Moisture detection by the moisture monitors succeeds.
- The PPIS initiates a main circulator and reactor trip after receiving a signal from the moisture monitors.
- The main circulator successfully trips.
- The reactor successfully scrams by outer control rod insertion.
- The PPIS initiates closure of the main steam and feedwater isolation valves.
- Following successful isolation, the steam generator dump system valves open and the water/steam inventory flows into the dump system tanks.
- The dump system valves fail to reclose when the line pressure is slightly above the primary coolant pressure.
- The SCS does not start on demand or does not operate long enough to adequately cool the core.
- Heat removal from the reactor core is accomplished by conduction and radiation to the RCCS.
- Some core radionuclides are released into the primary circuit due to fuel hydrolysis, graphite oxidation and fuel particle failure resulting from excessive temperatures.
- Primary circuit radionuclides are released through the steam generator dump system, through the tank relief valve and finally to the environment.

PRA IDENTIFICATION
Accident Family Initiating Event: Moderate steam generator leak
Accident Family: CC_s-9

Dominant Event Sequence Within Family: MS-AI
Family Mean Frequency: 8×10^{-7} per plant year
Other Accident Families: CC_s-10
Dominant Event Sequences Within Family: SS-AF
Family Mean Frequency: 3×10^{-6} per plant year

LBE IDENTIFICATION NUMBER

EPBE-4

EVENT DESCRIPTION

- A moderate sized leak (between 1 and 13 $in.^2$) in the primary coolant pressure boundary occurs.
- Low primary coolant pressure initiates a PPIS reactor trip signal.
- The reactor trips using the outer control rods.
- The HTS and SCS are unavailable to cool the core by forced helium circulation.
- Core cooling is provided by conduction and radiation of heat to the RCCS.
- Pumpdown of the primary coolant through the HPS in ineffective.
- Some release of radionuclides from the fuel particles and reactor core graphite is incurred due to excessive temperatures.
- Radionuclides released from the fuel, primary coolant circulating activity and some plated out radionuclides lifted off primary circuit surfaces due to high helium velocities are released to the reactor building.
- Fuel particle releases continue to egress into the reactor building after the initial depressurization at a slow rate until the core temperatures begin to decrease.
- Radionuclides in the reactor building are subsequently released to the atmosphere.

PRA IDENTIFICATION

Accident Initiating Event: Primary coolant leak
Accident Family: CC_p-8
Dominant Sequence Within Family: PC-DJ
Family Mean Frequency: 1×10^{-5} per plant year

Other Accident Families: CC_p-7, CC_p-11
Dominant Event Sequences Within Families: PC-BJ, PC-AF
Family Mean Frequencies: 4×10^{-5}, 1×10^{-6} per plant year

LBE IDENTIFICATION NUMBER

EPBE-5

EVENT DESCRIPTION

- A moderate steam generator leak occurs.
- Moisture ingresses into the primary system at a rate of 12 lbm/sec.
- Moisture is detected by the moisture monitors.
- The PPIS initiates a main circulator trip and reactor trip after receiving a signal from the moisture monitors.
- The main circulator trips and the reactor scrams by outer control rod insertion.
- The PPIS initiates closure of the main steam and feedwater isolation valves.
- Following successful isolation, the steam generator dump system valves open and the water/steam inventory flows into the dump system tanks.
- The dump system valves fail to reclose when the line pressure is slightly above primary coolant pressure, thereby providing a release pathway to the atomsphere.
- Core cooling is provided by the SCS which starts following the main circulator trip.
- Some radionuclides are released from the core into the primary circuit due to fuel hydrolysis and graphite oxidation.
- Primary circuit radionuclide inventories are released through the open steam generator dump system, through the tank relief valve and finally to the environment.

PRA IDENTIFICATION

Accident Initiating Event: Moderate steam generator leak
Accident Family: SG-8
Dominant Event Sequence Within Family: MS-AH
Family Mean Frequency: 1×10^{-5} per plant year

Other Accident Families: SG-2, SG-3
Dominant Event Sequences Within Families: MS-AM, MS-AV
Family Mean Frequencies: 1×10^{-6}, 7×10^{-7} per plant year

5. REFERENCES

1. Houghton, W. J., "Bridging Methods for Standard HTGR Licensing Bases," HTGR-86-039 Rev. 2, GA Document PC-000194/2, February 1986.

2. "Top-level Regulatory Criteria for the Standard HTGR," HTGR-85-002, Rev. 1, GA Document PC-000169, June 1985.

3. Houghton, W. J., "Licensing Basis Event Selection Criteria," HTGR-86-001, Rev. 1, GA Document 908418/1, February 1986.

4. Everline, C. J., Probabilistic Risk Assessment of the Modular HTGR Plant," HTGR-86-011 Rev. 1, GA Document 908664/1, to be published.

5. "Licensing Plan for the Standard HTGR," HTGR-85-001 Rev. 3, Draft, January 1985.