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Authorizing Official
Date: 12/28/10

APPLICATION OF BRIDGING METHODS
FOR STANDARD HTGR LICENSING BASES

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NEXT INDENTURED DOCUMENTS

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HTGR-86-004

REV																												
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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
REV																												
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1. SUMMARY

A review of the bridging methods is accomplished by providing an application of the process to the standard HTGR. A representative PRA, of limited range, is used in the process, and representative deterministic licensing bases are obtained.

The PRA is presented as a risk plot, and the characteristics of the events and the equipment in the PRA are used in the bridging application.

Representative licensing basis events are identified consisting of anticipated operational occurrences, design basis events, and emergency planning basis events.

Representative safety-related functions and safety-related structures, systems and components are identified.

The licensing basis events and safety-related functions and equipment are representative of the deterministic licensing bases that will be included in the PSID and SARs.

A response is requested from NRC on the acceptability of the application of the method, considering the range of the representative PRA presented.

2. INTRODUCTION

The design of the HTGR 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 demonstrate the derivation of licensing basis events and safety classification of structures, systems and components.

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. 2-1. 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 the centrally located box in the figure, the "bridge," connecting the Integrated Approach and the licensing basis.

This application is based on the 350 MW(t) HTGR. Since this is the first of several applications on a design concept that is not fully developed, the licensing selections developed in this report should be considered representative and can be expected to change in response to design evolution and further analyses. However, the report serves as a basis for discussion of the application of the bridging methods.

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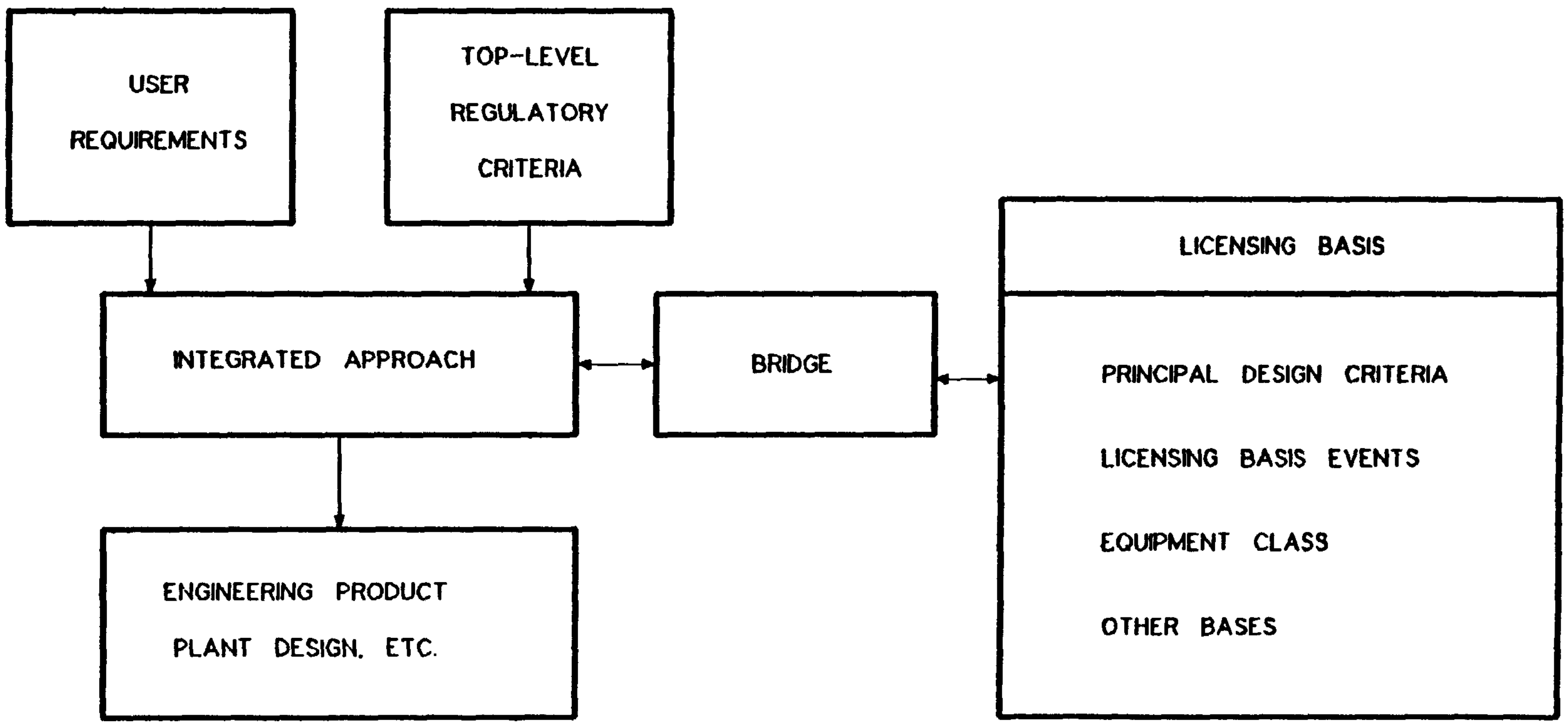
FIGURE 2.1

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



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3. BRIDGING METHOD

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. The steps in the method are summarized in the first part of this section. In the second half, quantitative guidelines used in implementing the first of these steps are discussed. Further discussion and examples of the implementation of each of the succeeding steps is given in Sections 4 and 5.

3.1 Steps in the Bridging Method

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 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.

This process can be illustrated with a figure, such as figure 3.1, 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 for choosing Licensing Basis Events are as follows.

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

Step 2. Compare the results of a risk assessment of a plant designed to Goals 1, 2, and 3 to the frequency-consequence risk plot.

Step 3. Identify as Anticipated Operational Occurrences (AOOs) those families of events 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 in for the Safety Analysis Reports Chapter 11 analyses.

Step 5. Identify as Design Basis Events (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 otherwise satisfy Step 5.

Step 7. Evaluate the consequences of the selected DBEs conservatively against the 10CFR100 dose criteria for the Safety Analysis Reports Chapter 15 analyses.

Step 8. Identify as Emergency Planning Basis Events (EPBEs) the dose-dominant events whose upper margin frequencies lie within the emergency planning basis region.

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

Step 10. Compare the risk assessment of the Goal 1-2-3 Design to the Goal 0 Top-Level Regulatory Criteria (Interim Risk Goals).

In addition to LBEs, another licensing basis with which regulators evaluate plant licensability and assure compliance with the Top-Level Regulatory Criteria is safety classification. Certain structures, systems and components (SSCs) capable of performing those radioactivity control functions related to public safety are classified as being safety-related. These SSCs which are so classified are subjected to various quality standards regulated by the NRC.

The steps for choosing which SSCs are to be classified as safety-related are as follows.

Step 1. For each DBE, classify as safety-related those SSC design selections needed to meet the design basis region dose criteria.

Step 2. For each EPBE with consequences greater than that specified by 10CFR100, classify as safety-related those SSC design selections chosen to assure that the event frequency is below the design basis region.

Step 3. For each SSC classified as safety-related, determine the design conditions for its operation by examining all of its associated DBEs and EPBEs.

3.2 Licensing Basis Event Selection Criteria

In Step 1 of the method for choosing LBEs it is necessary to define three regions on a frequency-consequence risk plot bounded by three agreed upon mean frequencies. In Fig. 3-1 these three regions are shown along with their bounding frequencies 10^{-W}, 10^{-X}, and 10^{-Y}. The development of numerical definitions for these boundaries is given in Ref. 3. The values for "W", "X," and "Y" arrived at in Ref. 3 and listed in Table 3-1 will be used in this document.

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 Steps 6 and 8 of the LBE selection process. The consideration of these is necessary to allow a well balanced choice of events that will be

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FIGURE 3.1
LICENSING BASIS REGIONS FOR HTGR BRIDGING METHODS

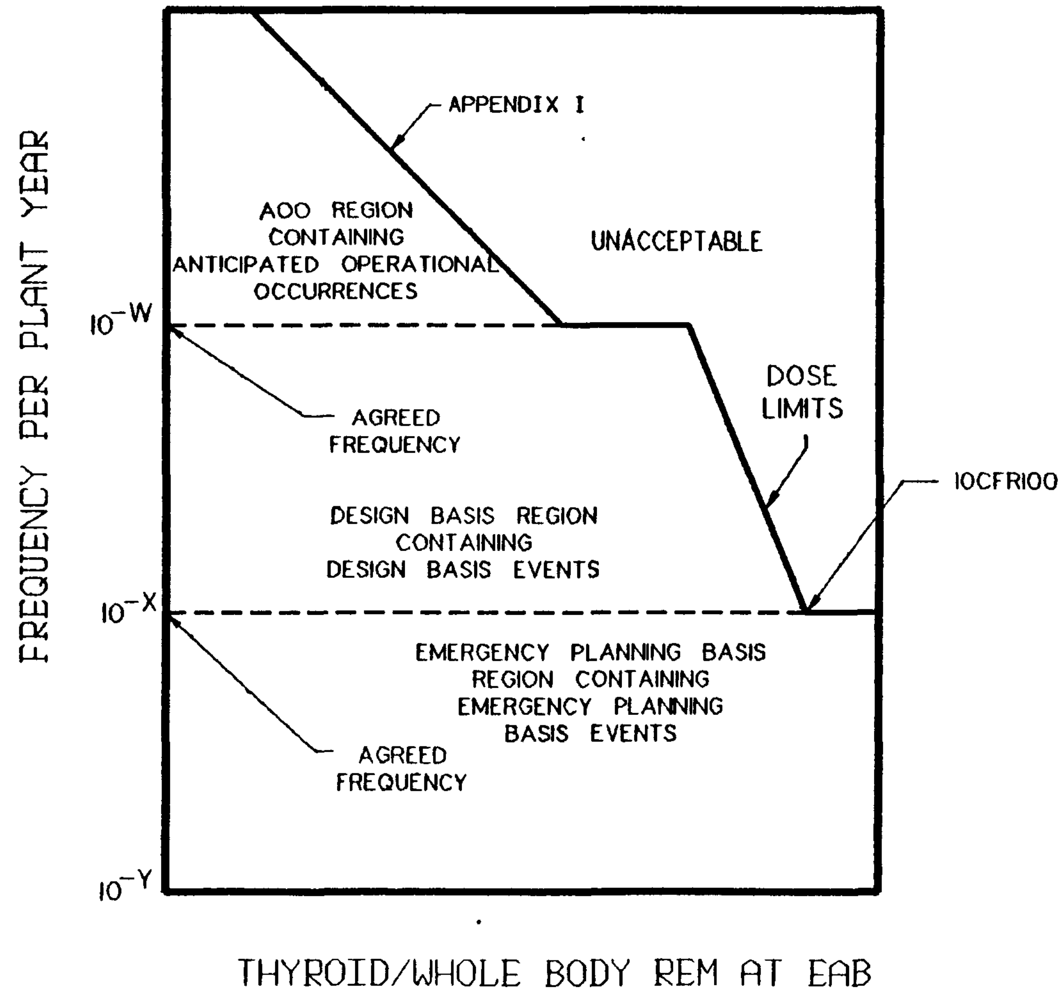

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TABLE 3-1
QUANTITATIVE CRITERIA FOR BRIDGING
PROPOSED REGION BOUNDARIES

	Criterion Symbol	Region Bounded by Guideline	Mean Frequency (per Plant Year)
!	10^{-W}	A00 region lower boundary	0.025
!	10^{-X}	Design basis region lower boundary	1×10^{-4}
!	10^{-Y}	Emergency planning basis region lower boundary	5×10^{-7}

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.

4. SELECTION OF LICENSING BASIS EVENTS

Having defined the Top-level Regulatory Criteria and the regions over which these criteria will be applied, a demonstration of the selection of events can be started.

Step 2 of the Bridging Method is to compare the results of a risk assessment of a plant design meeting the requirements of Goals 1, 2, and 3 to the regulatory dose limits previously described. In Fig. 4-1 the results of a representative risk assessment of the 350 MW(t) HTGR are compared to the regulatory criteria as shown on a frequency-consequence risk plot. These results are largely based upon analyses performed for the 250 MW(t) HTGR (Ref. 4) and scaled, as required, to provide representative results for the 350 MW(t) plant.

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.

4.1 Anticipated Operational Occurrence Selection

Step 3 of the Bridging Method is to identify as AOOs 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. 4-2 those event families with a frequency greater than 0.025 (once in 40 years) and therefore falling within the AOO region are shown.

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FIGURE 4.1
4X350 MW(t) PLANT PRELIMINARY
SAFETY RISK ASSESSMENT
COMPARED TO ACCIDENT SELECTION CRITERIA

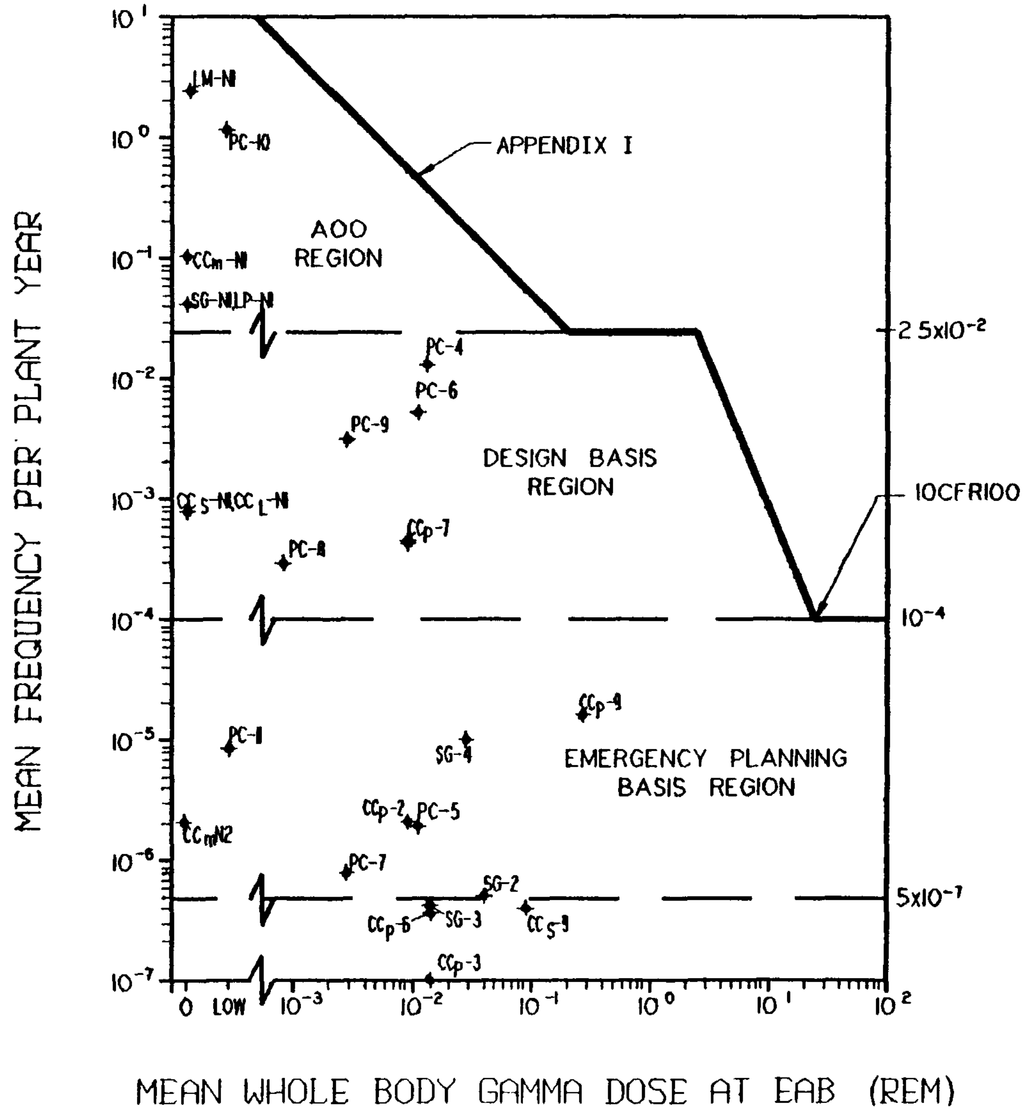

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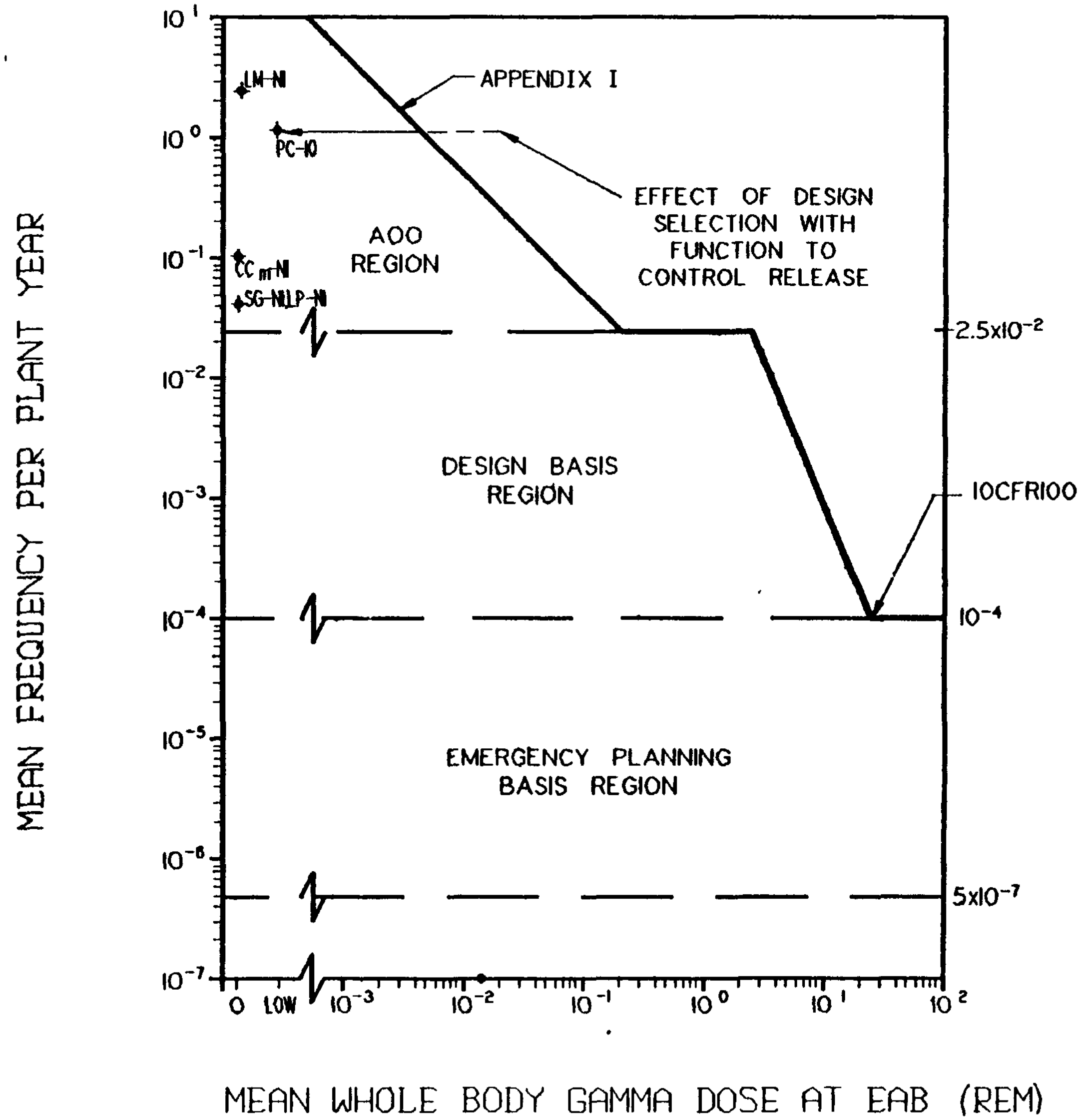
FIGURE 42
IDENTIFY PC-10 AS ANTICIPATED OPERATIONAL OCCURRENCE



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As an example of the selection process, consider the accident family PC-10. With it's predicted mean frequency of once per plant year, PC-10 is included on Fig. 4-2. PC-10 is a very small leak in the primary coolant pressure boundary predicted to occur about once every year 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 operates and further reduces the primary coolant activity.

3. Reactor is at equilibrium 100% power.

4. A very small leak occurs in the primary coolant pressure boundary allowing coolant to escape (between 3×10^{-5} to 0.05 sq in.)

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.

8. The remainder of the 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, for convenience, is numbered AOO-3 in Table 4-1. The other AOOs, identified in this manner, are also listed in this table.

Actually, since the analyses deal only with potential releases of radioactivity from the primary coolant and the reactor core, all the candidate events shown in Fig. 4-2 are, in fact, AOOs.

Five AOOs are listed in Table 4-1. 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 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, we can go back to the PRA event trees and determine which sequences lead to the AOO. So, listing the accident family assists in cross refe. encing 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 below.

AOO-1 is a loss of main loop cooling. It is intended to encompass most failures, originating within the plant, that preclude the Heat Transport System from continuing to perform the function of removing reactor heat. The event is terminated by reactor shutdown and startup of the Shutdown Cooling System. Failures typifying the event include circulator trips and losses of feedwater.

AOO-2 encompasses failures similar to those in AOO-1 except that in AOO-2 the Shutdown Cooling System fails to run. The event sequence is terminated with core cooling in the still pressurized module provided by convective loops developed within the reactor core, conduction, and radiation heat transport (pressurized conduction cooldown). Typical of the failures included in AOO-2 are failures originating within the plant that render both the HTS and SCS unable to perform their heat removal functions such as a failure of the service water system.

TABLE 4-1
ANTICIPATED OPERATIONAL OCCURRENCE (AOO)
SUMMARY TABLE

AOO Number	Accident Family	Anticipated Operational Occurrence
AOO-1	LM-N1	Loss of main loop cooling
AOO-2	CC_m-N1	Loss of main and shutdown cooling loops
AOO-3	PC-10	Very small primary coolant leak
AOO-4	LP-N1	Loss of offsite power & turbine trip
AOO-6	SG-N1	Steam generator tube rupture

AOO-3 is the small primary coolant leak (PC-10) discussed in some detail above.

AOO-4 is a loss of offsite power and turbine trip. The event is similar to AOO-1, in so far as that it leads to a loss of the HTS and the event is terminated by reactor shutdown and startup of the SCS. The event differs from AOO-1 in the failure sequence involved. For AOO-4 the failure initiating the event sequence, a loss of offsite power, occurs outside the plant and creates a significantly different set of conditions for analysis and design of the plant. For instance, in AOO-4 startup of the SCS is predicated upon successful start of a backup power supply.

AOO-6 is a steam generator tube rupture. The event is similar to AOO-1 in that it leads to 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-6 the tube rupture is followed by reactor and main loop shutdown. The leaking steam generator is isolated from the feedwater and steam headers and its inventory is dumped. Core cooling in this event is provided by the SCS.

Step 4 in the method is to evaluate the consequences of each of the selected AOOs realistically against the Appendix I annualized dose criteria for Chapter 11 of the Safety Analysis Reports. Because the doses from the PRA plotted in Fig. 4-2 are calculated realistically, they can be considered as indicative of what might be seen in Chapter 11. Showing that these doses are acceptable relative to the limits of Appendix I requires that the doses from each AOO be expressed on an annualized or risk bases of mRem/year, as are the limits of Appendix I. Then the risk from all of the events plus that from planned releases can be summed and compared to the Appendix I limits. Such a comparison is made in Table 4-2.

In the table both the frequency and resultant dose from each of the AOOs is listed. The product of these is the annualized dose and is listed in the right most column. Since AOO-3, the small primary coolant leak discussed

TABLE 4-2
COMPARISON OF RELEASES TO APPENDIX I LIMITS

AOO	Frequency (Per Year)	Dose (mRem) Whole Body	Annualized Dose (mRem Per Year)
	Normal Operation	0.05	0.05
AOO-1	2.5	0	0
AOO-2	0.1	0	0
AOO-3	1.2	0.84	1.01
AOO-4	0.04	0	0
AOO-6	0.04	0	0
		Total Annual Dose	1.06
		Appendix I Limit	5.0

above, is the only AOO which has a calculated dose it is the only AOO to contribute to the total. The planned releases from normal operation have not as of yet been evaluated but for the purposes of this example it is assumed to be 1% of the releases allowed.

4.2 Design Basis Event Selection

In Step 5 of the method a similar approach is utilized to to select Design Basis Events (DBEs). The method identifies as DBEs 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}, are shown in Fig. 4.3. 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 8×10^{-4} per plant year. The specific events involved are as follows.

1. 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. Reactor is shutdown.

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

6. Primary coolant remains at pressure.

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

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

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

As seen in Fig. 4-3 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 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 within the core, 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-8.

Other events, not shown on Fig. 4-3 but visible on 4-1, might fall very close to but not actually within the frequencies bounding the design basis region. Step 6 of the method identifies as DBEs any additional event families whose agreed upper margin frequencies fall within the design basis region and who otherwise satisfy Step 5 as described above. A review of Fig. 4-1 shows no additional DBEs from Step 6, assuming that a factor of two covers the agreed upon margin.

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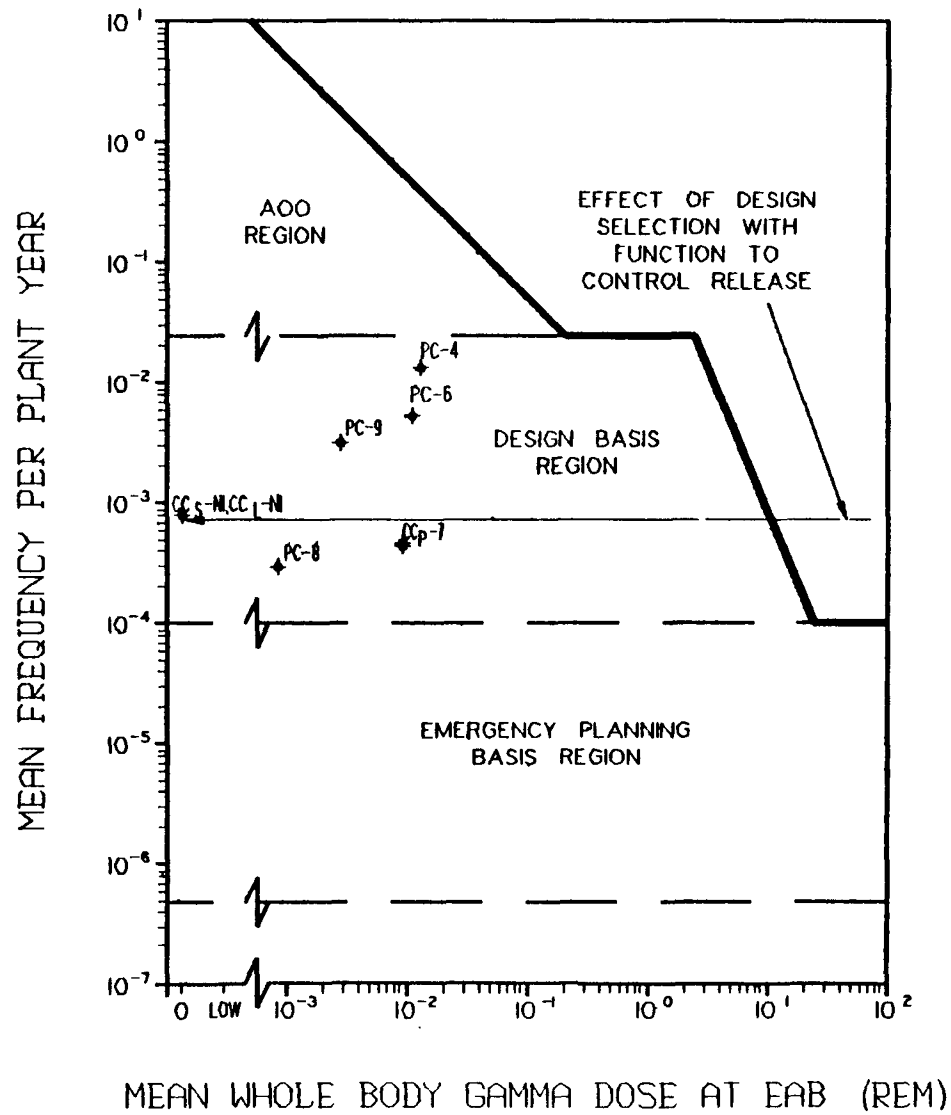
FIGURE 4.3
IDENTIFY CC_L -NI AS DESIGN BASIS EVENT



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A listing of all the event families, identified as DBEs in Steps 5 and 6, is provided in Table 4-3. 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.

Many of the DBEs listed in the table are aimed at primary coolant leaks which were found in the safety assessment of the plant to dominate risk to the public. DBEs 3 and 1 characterize leaks of increasingly large size in which the plant responds as planned. The reactor is shut down following detection of the leak. A portion of the primary coolant leaks into the reactor building while the remainder is pumped to storage. Core cooling is provided by the Heat Transport System operating in a shutdown mode. The moderate, though less likely, leaks, because of their more rapid depressurization, 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.

While the events described above all had the plant responding as it was designed to do, the method places no limit on the number of failures, subsequent to the leak, that must be included in the DBEs except that the predicted likelihood of the event sequence must fall within the range of frequencies included by the design basis region. DBEs 5 through 7 describe very small leaks in which the plant does not respond as planned. DBE-5 encompasses any failures that prevent pumping the primary coolant to storage. DBE-6 encompasses single common mode and multiple failures that preclude both the HTS and the SCS from providing core cooling. In this case the event is terminated with core cooling being provided by conduction and radiation and cavity

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

DBE Number	Accident Family	Design Basis Event (DBE) Name
DBE-1	PC-4	Moderately sized primary coolant leak
DBE-3	PC-6	Small primary coolant leak
DBE-5	PC-9	Very small primary coolant leak without pumpdown
DBE-6	CC_p-7	Very small primary coolant leak without shutdown cooling
DBE-7	PC-8	Very small leak without confinement Filters
DBE-8	CC_L-N1	Loss of offsite power, turbine trip and Failure of the SCS to Start
DBE-9	CC_s-N1	Large steam generator leak with failure of the SCS to start

cooling. DBE-7 encompasses failures that prevent use of the reactor building filters.

DBE-8 is a loss of offsite power followed by inadvertent turbine trip and failure to successfully start the SCS. DBE-8 is generally similar to AOO-2. 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-8, 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 9 encompasses large steam generator leaks, such as those described by AOO-6 but with the addition that the shutdown cooling system fails to provide forced convection cooling.

In Step 7 the consequences of each selected DBE is conservatively evaluated and compared to the 10CFR100 dose criteria for Chapter 15 of the Safety Analysis Reports. The doses shown on the figures for each of the DBE selected are realistic. Conservative calculations have not yet been completed, but representative uncertainty bars showing conservative values on the right end are shown on Fig. 4-3.

4.3 Emergency Planning Basis Event Selection

Step 8 of the LBE selection process identifies as Emergency Planning Basis Events (EPBEs) the dose dominant events whose upper margin frequencies fall within the frequency range encompassed by the emergency planning basis region.

As an example of the application of this step consider Fig. 4-4. 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 small primary coolant leak category designated CC_p-9. Details of this event are as described below.

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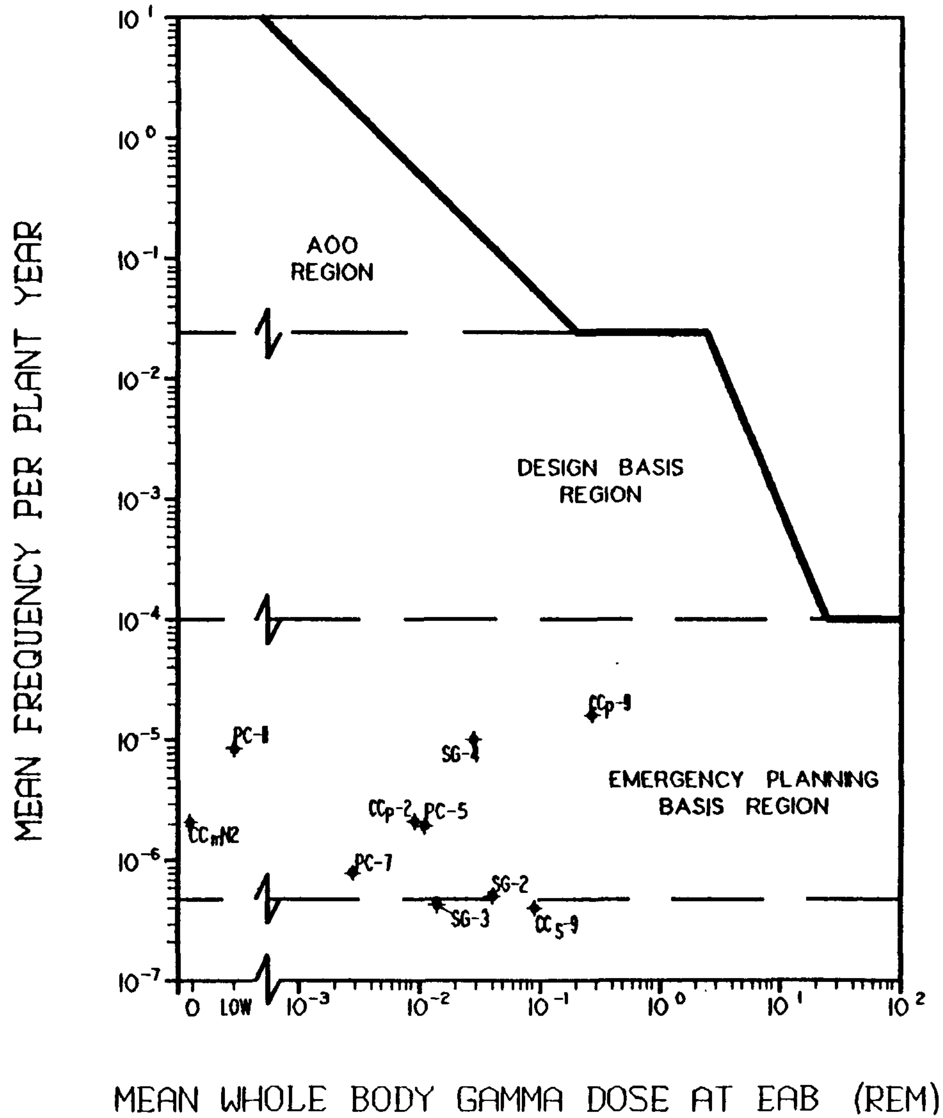
FIGURE 4.4
IDENTIFY CC_P −9 AS
EMERGENCY PLANNING BASIS EVENT



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1. Fuel particles retain most fission products keeping the primary coolant activity level low.

2. Helium purification system operation further reduces primary coolant activity.

3. Reactor is at equilibrium 100% power.

4. Small primary coolant leak occurs.

5. Leak is detected by plant control systems. Reactor is automatically shut down.

6. HTS fails to maintain core cooling.

7. Shutdown Cooling System fails to start and maintain forced convection cooling of core until HTS cooling restored.

8. Core temperatures limited by conduction and radiative heat transfer and cavity cooling.

9. Pumpdown of primary coolant fails.

10. Primary coolant continues leaking to reactor building throughout earlier portion of conduction cooldown temperature transient transporting both circulating activity and incremental fuel releases.

Being the largest dose event in the range of frequencies between 1×10^{-4} and 5×10^{-7}, $\bar{CC}_p$-9 is selected as an EPBE and is designated as EPBE-4.

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 event with the largest dose, in this case EPBE-4. Table 4-4 lists all the events identified as EPBEs.

Eight events are listed in the table. EPBE-4 is the dominant risk contributor in the region and has already been addressed in some length. Other EPBEs with in the table are summarized below.

EPBEs 1 and 2 are moderately sized primary coolant leaks of less than 1 sq in. in which reactor building ventilation fails to operate as designed. These events include the failure to divert ventilation discharge through exhaust filters. 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 the case of EPBE-2 the release is increased by a subsequent failure to pumpdown the primary coolant system to storage.

EPBEs 3, and 4 are small primary coolant leaks followed by failures in forced circulation. EPBE-4 has already been discussed in some detail. EPBE-3 is identical but the leak is smaller and the corresponding dose is lower. EPBE-9 is a very large primary coolent leak followed by failure in forced circulation.

EPBE-5 is a large 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. In EPBE-5 the relief valve does not reseat after successfully opening. This in turn leads to venting the primary coolant and some liftoff to the reactor building.

EPBE-6 is a large 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. In EPBE-6 the relief valve does not reseat after successfully opening. This in turn leads to venting the primary coolant and some liftoff to the reactor building.

TABLE 4-4
EMERGENCY PLANNING BASIS EVENT (EPBE)
SUMMARY TABLE

EPBE Number	Accident Family	Emergency Planning Basis Event (EPBE)
EPBE-1	PC-5	Moderate primary coolant leak with reactor bldg. fans operating without filters
EPBE-2	PC-7	Moderate primary coolant leak with failure to Pumpdown and Reactor Bldg. Fans operating without filters
EPBE-3	CC_p-2	Very small primary coolant leak without forced core cooling
EPBE-4	CC_p-9	Small primary coolant leak without forced core cooling
EPBE-5	SG-2	Large steam generator leak with failure to dump and stuck open primary relief valve
EPBE-6	SG-3	Large steam generator leak with failure to isolate feedwater and stuck open primary relief valve
EPBE-7	SG-4	Large steam generator leak with failed open dump valves
EPBE-8	CC_s-9	Large steam generator leak with failed open dump valves and without forced core cooling
EPBE-9	CC_p-6	Very large primary coolant leak without forced core cooling.

EPBE-7 is a large steam generator leak in which the mutiple dump valves fail to reshut after discharging the steam generator water inventory. This failure allows the primary coolant to escape through the ruptured steam generator tube, through the steam generator tubing and out to the atmosphere, via the open dump valves, and in so doing to bypass the reactor building.

EPBE-8 is similar to EPBE-7 but includes a loss of forced circulation. Like EPBE-7 a leak occurs in which the mutiple dump valves fail to reshut after discharging the steam generator water inventory. However, in EPBE-8 the SCS fails to successfully provide core cooling until the HTS is restored to service. Core cooling is via conduction and radiation. 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.

As stated in Step 9 of the Bridging Method the consequences of the selected EPBEs are evaluated realistically for emergency planning, and environmental assessments. These realistic dose evaluations, performed as a part of the preliminary PRA, are shown in Fig. 4-4 for the selected EPBEs.

4.4 Overall Regulatory Compliance

The preceding steps of the LBE selection method have shown how the choice of these events is used to demonstrate compliance with various portions of the Top-Level Regulatory Criteria. AOOs are used as a part of showing compliance with the annualized dose criteria of 10CFR50 Appendix I. DBEs are used to show compliance with the dose limits specified in 10CFR100. In Step 10 AOOs, DBEs and EPBEs are used together to show compliance with the Top-Level Regulatory Criteria consisting of the interim safety goals. The step requires that the results of a risk assessment of the plant be compared to that portion of the Interim Risk Goals contained in the Top-level Regulatory Criteria. In so doing plant conformance with these goal 0 regulatory criteria is confirmed.

Table 4-5 demonstrates how such a comparison is made using those LBEs selected in the previous sections. For each LBE the table notes both the event frequency and consequence, both of which can be read off the frequency-consequence plots previously seen. The product of these factors gives a risk. These risks have been calculated for each event and the result given in the right-hand column of the table. Summing up these risks would give a resultant risk due to all unplanned releases. However, the risk limits specified in the interim safety goals is for all releases, whether planned or unplanned. Therefore, one additional entry, listed as "normal operation" in the event column, can be seen in the top line of the table. Summing all the risks listed in the right hand column of the table yields a total risk which is compared to the risk goal.

Because of the preconceptual stage of the design of the 350 MW(t) HTGR actual calculations have not been performed on the predicted annual planned releases from the plant. However, to illustrate this demonstration of compliance with the interim safety goals, a value of 1% of that allowed by Appendix I is assumed. It can be seen then, by reviewing Table 4-5, that significant margin to the latent cancer risk goal (2×10^{-6} latent cancer deaths per plant year per person) is predicted. Furthermore, it should be noted that the total risk is dominated by the smaller, more frequent releases characterized by the AOOs and higher frequency DBEs.

TABLE 4-5
COMPARISON OF PRELIMINARY RISK ASSESSMENT TO LATENT CANCER RISK GOAL

	Region	Event	Frequency: Events Per year (Mean)	Consequence: Cancers Per Event Per Person	Risk: Cancers Per Year Per Person
		Normal Operation	1.0	5×10^{-9}	5×10^{-9}
!	A00	PC-10	0.95	9×10^{-9}	8×10^{-9}
!	Design Basis	PC-4	1×10^{-2}	1×10^{-6}	1×10^{-8}
!	Design Basis	PC-6	5×10^{-3}	1×10^{-6}	6×10^{-9}
!	Design Basis	PC-9	3×10^{-3}	3×10^{-7}	9×10^{-10}
!	Design Basis	CCp-7	4×10^{-4}	9×10^{-7}	4×10^{-10}
!	Design Basis	PC-8	3×10^{-4}	8×10^{-8}	2×10^{-11}
!	Emergency Planning Basis	PC-5	2×10^{-6}	1×10^{-6}	2×10^{-12}
!	Emergency Planning Basis	PC-7	8×10^{-7}	3×10^{-7}	2×10^{-13}
!	Emergency Planning Basis	CC_p-2	2×10^{-6}	9×10^{-7}	2×10^{-12}
!	Emergency Planning Basis	CC_p-9	2×10^{-5}	3×10^{-5}	4×10^{-10}
!	Emergency Planning Basis	SG-3	4×10^{-7}	9×10^{-6}	4×10^{-12}
!	Emergency Planning Basis	SG-4	1×10^{-5}	3×10^{-6}	3×10^{-11}
!	Emergency Planning Basis	CC_s-9	4×10^{-7}	9×10^{-6}	4×10^{-12}
				Total risk	3×10^{-8}
				Risk goal	2×10^{-6}

5. SELECTION OF SAFETY-RELATED STRUCTURES, SYSTEMS AND COMPONENTS

A three-step method for classifying certain structures systems and components (SSCs) as safety-related is described in Ref. 1. The method is linked to the selection of DBEs and leads to classifying as safety-related a set of plant features which perform the functions needed for compliance with the dose limits of 10CFR100. This method, along with the selection of DBEs given above, has been used to develop a representative classification of safety-related SSCs for the 4 x 350 MW(t) HTGR. The following section, in addition to listing the SSCs so classified, describes the method used and provides an example of its application.

Step 1 specifies that for each DBE, classify as safety-related those SSC design selections chosen for compliance with the dose criteria of the design basis region. More exactly, for each DBE selected various functions can be identified which must be performed if the consequence of the event is to remain within that allowed by the dose criteria. These functions, for a typical DBE, might be such things as removing core heat or controlling core reactivity. Step 1 requires that a set of SSCs which are capable of performing these functions be classified as safety-related. Note that this does not require that the SSC(s) performing these functions as described by the DBE be classified as safety-related. The method only requires that an SSC capable of adequately performing the required function(s) be so classified. This distinction will be further clarified in the example.

Step 2 specifies that for each EPBE, with consequences greater than 10CFR100, classify as safety-related those SSC design selections chosen to assure that the event frequency is below the design basis region. For this step too various functions can be identified which must be performed to assure that the frequency of an EPBE, with a consequence greater than that allowed by 10CFR100, falls below the design basis region. Step 2 requires that a set of SSCs capable of performing these functions be classified as safety-related. Again, these SSCs need not be those in the EPBE description.

Finally, Step 3 of the method specifies that for each SSC classified as safety-related, determine the design conditions for its operation by examining all it's associated DBEs and EPBEs.

Illustrating how this process has been used to classify various SSCs as safety-related, observe Table 5-1. In this table a matrix has been constructed showing, for each DBE, which functions must be performed if the dose limits of 10CFR100 are to be met. Because there are no EPBEs yet identified which exceed the 10CFR100 limits, there is no EPBE table.

From Steps 1 and 2 each of the functions listed in Table 5-1 an SSC, or set of SSCs, capable of performing the function will be classified as safety-related. Viewing the table it is immediately apparent that many of the DBEs rely on satisfactory performance of the same functions in order to result in allowable doses. Thus in selecting an SSC to classify as safety-related for some function, it is advisable to consider simultaneously all the events during which the function is required.

For example, consider function number "B" in the table, Remove Core Heat. This function is relied upon by all the DBEs listed. Furthermore, one can quickly think of several SSCs which are capable of providing the function and might be candidates for being classified as safety-related. In Table 5-2 a second matrix has been constructed showing, for each DBE, which SSCs in the plant design are available to perform the function of removing decay heat. In addition to this, the table, in the second from the right hand column, gives an estimate of the relative cost impact of classifying each of these SSCs safety-related. So in choosing which SSC to classify one first determines which SSCs can satisfy the needs of all the DBEs. Then, from among these possibilities, one chooses the SSC with the lowest cost impact. In this case the choice is somewhat more involved because while option 5 (Reactor Cavity and Surroundings) may at first appear more attractive, this choice would not be optimal when other safety-related functions are considered. Referring to Table 5-3 it is noted that the cavity and surroundings are not by themselves able to satisfy the function of preventing chemical attack and so option 4 of

TABLE 5-1
PRELIMINARY SELECTION OF SAFETY RELATED FUNCTIONS

Safety-Related Function To Meet 10CFR100	DBE-1 PC-4	DBE-3 PC-6	DBE-5 PC-9	DBE-6 CC_p-7	DBE-7 PC-8	DBE-8 CC_L-N1	DBE-9 CC_S-N1
A. Retain Radionuclides within Coated Fuel Particles	Yes	Yes	Yes	Yes	Yes	Yes	Yes
B. Remove Core Heat to Assist in Radionuclide Retention	Yes	Yes	Yes	Yes	Yes	Yes	Yes
C. Prevent Chemical Attack	-	-	-	-	-	Yes	Yes

TABLE 5-2
REPRESENTATIVE CLASSIFICATION OF AN SSC AS SAFETY-RELATED
FOR THE 350 MW(t) HTGR

<u>SAFETY-RELATED FUNCTION:</u> Remove Core Heat to Assist in Radionuclide Retention (Function "B" in Table 5-1)

SSC Available to Perform Function	DBE-1 PC-4	DBE-3 PC-6	DBE-5 PC-9	DBE-6 CC_p-7	DBE-7 PC-8	DBE-8 CC_L-N1	DBE-9 CC_s-N1	Related Cost	Class.
1. Heat Transport Sys	Yes	Yes	-	-	Yes	-	-	High	
2. Shutdown Cooling System	Yes	Yes	Yes	-	Yes	-	-	Medium	
3. Conduction C/D to Active RCCS	Yes	Yes	Yes	Yes	Yes	-	Yes	Medium	
4. Conduction C/D to Passive RCCS	Yes	Yes	Yes	Yes	Yes	Yes	Yes	Medium	X
5. Conduction C/D to Reactor Cavity and Surroundings	Yes	Yes	Yes	Yes	Yes	Yes	Yes	Medium	X

TABLE 5-3
FURTHER CONSIDERATIONS IN THE CLASSIFICATION OF AN SSC AS SAFETY-RELATED

SAFETY-RELATED FUNCTION: Prevent Chemical Attack
(Function "C" in Table 5-1)

SSC Available to Perform Function	DBE-8 CC_L-N1	DBE-9 CC_S-N1	Related Cost	Class.
1. HPS Pumpdown	–	Yes	Medium	
2. Conduction C/D to Passive RCCS	Yes	Yes	Low	X
3. Conduction C/D to Reactor Cavity and Surroundings	–	–		

Table 5-2, Conduction Cooldown to the RCCS in the passive mode, is classified as safety related. If no one SSC had been able to satisfactorily perform the function for all the events of concern then the more difficult, though conceptually similar, task of finding the most cost effective set of SSCs would be undertaken. A redesign introducing new SSCs could also be considered.

Other SSCs which have been classified as safety-related using this method can be found in Table 5-4. In the table are first listed, in the left-hand column, those functions which must be performed in order to limit any releases to within that allowed by 10CFR100. On the right are listed those SSCs which are both capable of performing the function and have been chosen to be classified as safety-related.

The first function listed in the table is the retention of fission products within the fuel. Accomplishment of this function assures low activity levels within the primary coolant such that should the primary coolant escape the resulting doses fall within acceptable limits. Furthermore, the retention function is also key to the HTGRs performance during transients and is related to function B. That is, the conditions under which this retention must be accomplished is determined in part by the degree to which function B, heat removal, is accomplished. The degree of retention required is that needed to limit any releases to levels consistent with the dose criteria of 10CFR100. While several design options are available to do this, the SSC capable of accomplishing this and chosen to be classified is the coated, ceramic HTGR fuel.

The second of the functions ("B"), Remove Core Heat, has been discussed in some depth previously and is not further discussed here.

The last function dealt with in the table, Function C, is preventing chemical attack. Accomplishment of this function involves limiting the ingress of any graphite oxidants to levels that would not lead to releases in excess of those compatible with meeting the Top-Level Regulatory Criteria.

Several SSCs have been selected as being safety related and together can perform this function. These include the reactor pressure vessel, the primary coolant relief valves and the reactor cavity cooling system.

TABLE 5-4
FUNCTIONS REQUIRED TO MEET 10CFR100 AND SSCs CLASSIFIED AS SAFETY-RELATED

Safety-Related Function	Safety-Related SSC
A. Retain radionuclides within fuel	Leaktight fuel, high temperature and moisture resistant fuel Reactor Core (18) especially particle coatings
B. Remove core heat to assist radionuclide retention	SSCs for conduction cooldown in (conduction and radiation heat transport from core) Reactor Core (18) Reactor Internals (17) Reactor Vessel (11-6) Reactor Enclosure Space
C. Prevent chemical attack	SSCs preventing large air ingress Reactor Cavity Cooling (56) in passive mode only Primary Coolant Pressure Relief Valves (11-5)
D. Control reactivity	(To be determined)

6. REQUESTED NRC RESPONSE

This document has been prepared for presentation to the Advanced Reactor Group of the Nuclear Regulatory Commission (NRC) in support of the HTGR Licensing Plan (Ref. 6). It is submitted for their information and review. The NRC is requested to address and respond to the following questions on the review of the bridging method based on the representative application given in Sections 4 and 5:

1. Does the NRC agree that the choice of licensing basis events follows the bridging method, within the range of the representative PRA results presented in the review?

2. Are the licensing basis events chosen a complete selection, within the range of the representative PRA results presented?

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 and SARs, within the range of the representative PRA results?

4. Are the emergency planning basis events chosen the appropriate set of events to be emphasized in the PRA report, within the range of the representative results?

5. Does the NRC agree that the choice of safety-related functions follows the bridging method, within the range of the representative PRA results presented?

6. Are the safety-related functions chosen a complete selection, within the range of the representative results?

7. Does the NRC agree that the choice of safety-related structures, systems and components are sufficient to perform the chosen safety-related functions?

8. Are the chosen safety-related structures, systems and components an appropriate and complete set to be included in the PSID and SARs as safety-related equipment, within the range of the representative results?

7. REFERENCES

1. Houghton, W. J., "Bridging Methods for Standard HTGR Licensing Bases," HTGR-85-039 Rev. 1, GA Document PC-000194, October 1985.

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

3. Houghton, W. J., Licensing Basis Event Selection Criteria," HTGR-86-001, GA Document 908418, to be published.

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

5. Houghton, W. J., "Bridging Manual," GA Doc 908395, September 1985.

6. "Licensing Plan for the Standard HTGR," HTGR-85-002, GA Document PC-000169, Rev. 1, June 1985.