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Exelon Generation

Project No.: 713

March 15, 2002

U.S. Nuclear Regulatory Commission
Attn: Document Control Desk
Washington, D.C. 20555

Subject: Revision of Exelon Generation Company's Proposed Licensing Approach for the Pebble Bed Modular Reactor in the United States

In a letter to the NRC dated November 15, 2001, Exelon Generation Company (EGC), LLC committed to provide a preliminary description of the Pebble Bed Modular Reactor (PBMR) safety grade systems, structures, and components (SSC). The attached revision of the "Proposed Licensing Approach for the Pebble Bed Modular Reactor in the United States," satisfies this commitment.

The original revision of this paper was submitted to the NRC by letter dated August 31, 2001, revised and resubmitted to the NRC by letter dated January 31, 2002. Specifically, the attached paper responds to the request in NRC letter dated September 26, 2001 designated A) 2 "What systems, structures, and components will be considered "safety grade" and what requirements will be imposed as a result of this classification?" Specifically Section 5.2 "Method for Equipment Safety Classification" of the attached document has been revised to include the preliminary SSC listing based on an initial PBMR Probabilistic Risk Assessment, and preliminary identification of the PBMR's licensing basis events. Section 5.4 "Requirements for Safety-Related Equipment" describes the method in which requirements will be imposed as a result of the classification. However, no specific requirements will been identified for the preliminary SSCs during this stage of the design.

If you have any questions or concerns regarding this matter, please contact R. M. Krich or me.

Sincerely,

Kevin F. Borton
Manager, Licensing

Attachment

cc: Farouk Eltawila, Office of Nuclear Reactor Research
James Lyons, Office of Nuclear Reactor Regulation
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D064.

ATTACHMENT

Project Number: 713

Exelon Generation Company

"Proposed Licensing Approach
For The
Pebble Bed Modular Reactor
In The United States"

March 15, 2002

WORKING DRAFT

Exelon Generation Company

PROPOSED LICENSING APPROACH

FOR THE

PEBBLE BED MODULAR REACTOR

IN THE UNITED STATES

MARCH 15, 2002

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1 INTRODUCTION

The Advanced Reactor Policy Statement (Reference 1) states that for advanced reactors the Commission expects, as a minimum, at least the same degree of protection of the public and the environment that is required for current generation LWR. Thus, the Commission expects that advanced reactor designs will comply with the Commission's safety goal policy statement (Reference 2). Furthermore, the Commission expects that advanced reactors will provide enhanced margins to safety and/or utilize simplified, inherent, passive, or other innovative means to accomplish their safety function. Advanced reactor designers are encouraged as part of their design submittals to propose specific review criteria or novel regulatory approaches which NRC might apply to their designs.

The purpose of this document is to present the proposed licensing approach for evaluating the Pebble Bed Modular Reactor (PBMR). Exelon Generation Company (Exelon) intends to utilize current regulations and is not seeking any new rule making as part of its Combine Operating License application. Exelon's approach incorporates risk-informed elements and insights. It also fits within and fully meets the existing, applicable NRC regulations that have been developed on a largely deterministic basis for LWR. The process is based on methods developed in the mid-80s for the Modular High Temperature Gas Cooled Reactor (MHTGR) and modified to reflect the advances that have been made since then in risk-informed regulation.

Exelon's proposed approach to the acceptance and licensing of the PBMR has a clear link between NRC's regulatory missions and the specific regulatory requirements to be applied to the design. Figure 1-1 displays: 1) the NRC mission, 2) the public safety objective for nuclear power plants as contained in 10CFR50.57, 10CFR50 Appendix A-Introduction and the Safety Goals, and 3) the means for meeting those objectives by limiting radiation exposures during normal operation, preventing and mitigating accidents, and protecting the plant against sabotage and safeguards threats.

Based on these fundamental objectives, a top-down licensing approach for the PBMR has been developed. Certain regulatory objectives are not amenable to probabilistic treatment in the present regulatory environment. These include occupational exposure minimization, environmental impacts other than radiological, and security and safeguards. These objectives will be met in the conventional manner as consistent with existing practice. For the remaining objectives (limiting public exposures during normal operation, and preventing and mitigating accidents), Exelon has developed a risk-informed licensing approach as described in this paper.

Exelon is proposing this risk-informed licensing approach for the PBMR in order to bridge a gap in NRC's existing regulations governing the design of reactors. Specifically, most of NRC's existing design-related regulations explicitly pertain only to light water reactors (LWRs). While Exelon intends to use those regulations as guidance to the extent that they relate to the design functions of the PBMR, some of those regulations are not relevant and others are only partially applicable to the design.

Furthermore, other aspects of the PBMR design are not addressed by NRC's existing regulations, and Exelon has identified a need to develop criteria to control the design in those areas. As result, Exelon believes that a risk-informed licensing approach is necessary to help identify the extent to which LWR-based regulations should be applied to the PBMR as guidance and to develop new criteria for the PBMR where existing regulations are silent. This approach will help the PBMR designer and reviewer navigate through the existing regulations.

The specific objectives for developing the PBMR licensing approach are as follows.

- Establish agreed upon quantitative top-level regulatory criteria
- Establish an agreed upon risk-informed method for selecting licensing basis events
- Establish a design-specific method to select and determine special treatment of safety-related systems, structures and components
- Establish a process to determine which regulatory requirements and guidance are applicable and to what extent they need to be supplemented for the PBMR.

This document provides a discussion of the logic and methods at an introductory level so that fundamental concepts may be discussed and a path to agreement can develop. Reaching agreement on the PBMR licensing approach is essential before moving ahead with design finalization, application preparation and specific design reviews in the application phase.

It is envisioned that once the specific licensing approach objectives (outlined above) are reached, design decisions and regulatory reviews regarding the PBMR will be better focused.

The proposed licensing approach (Figure 1-2) results are contingent upon identifying the following elements:

- Top Level Regulatory Criteria
- Identification of Applicable Regulations and Guidance
- Selection of Licensing Bases Events
- Development of Regulatory Design Criteria and Selection of Safety-Related Equipment

The first element is within the NRC Mission and Safety Goal box of Figure 1-2, the second element is shown on the left side of the figure, and the last two elements are shown on the right side. As shown, a comparison is performed to determine the applicability of the existing regulations and guidance leading to the development of a set of regulatory references that define the content of the application.

This report covers each of the elements: Section 2 discusses the Top Level Regulatory Criteria, which state ***what*** must be satisfied; Section 3 discusses the processing for identifying applicable regulations and guidance; Section 4 discusses the selection of Licensing Basis Events (LBE) and describes ***when*** the criteria must be met; Section 5 discusses Equipment Classification and Regulatory Design Criteria and discusses ***how*** and the ***how well*** the criteria will be met; and Section 6 addresses the marriage of the

PBMR licensing use of design specific information and risk insights from the PBMR PRA with that of current regulatory practice to develop a focused set of regulatory references to use in completing the design, preparing the application and guiding the NRC review of the application.

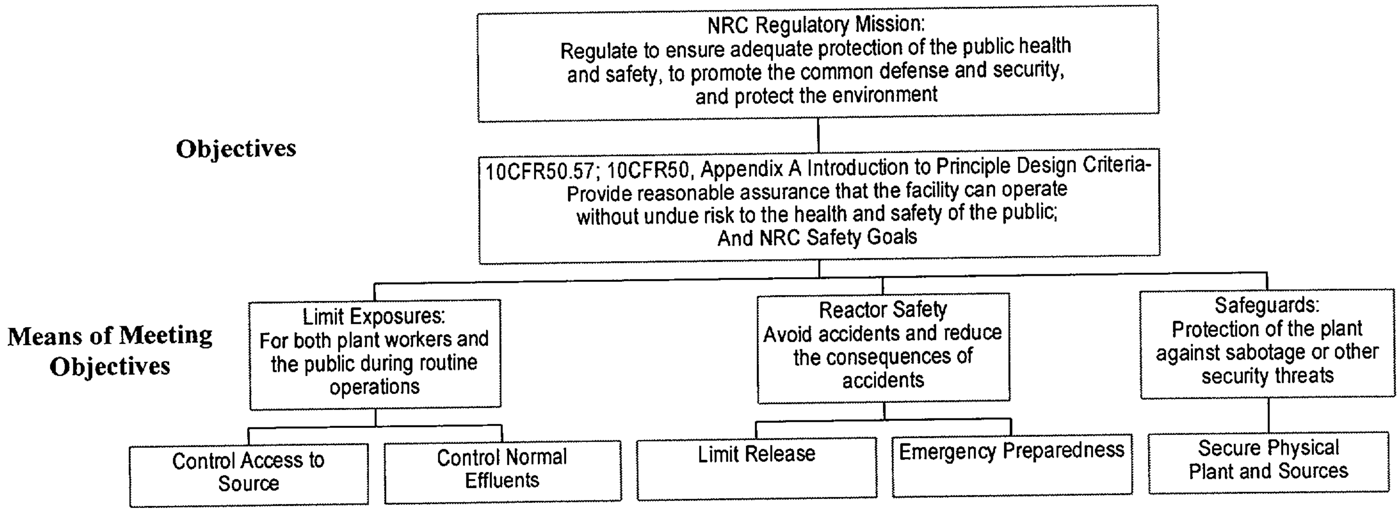

Figure 1-1

LINKAGE OF PBMR LICENSING APPROACH TO NRC REGULATORY MISSION

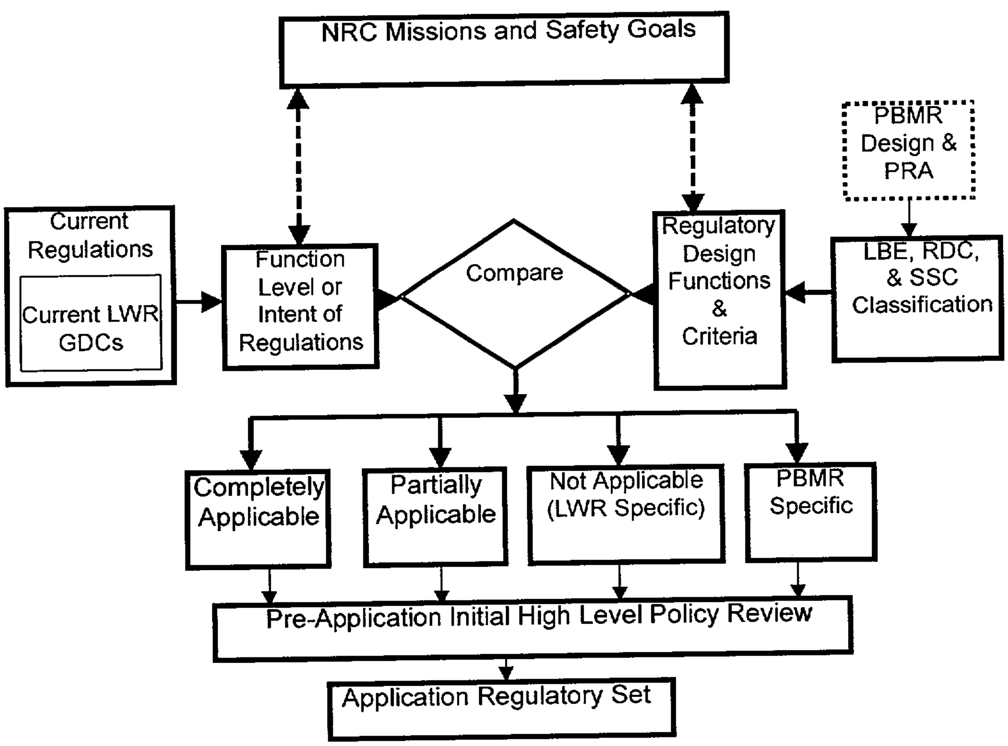

Figure 1-2

COMPARISON OF RISK-INFORMED LICENSING BASES WITH CURRENT REGULATIONS

2 TOP LEVEL REGULATORY CRITERIA

In support of the development and preapplication licensing of the Modular High Temperature Gas Cooled Reactor (MHTGR), Reference 3 presented a listing of Top Level Regulatory Criteria (TLRC) to be utilized as standards for judging nuclear power plant licensability related to the retention of radionuclides. These criteria directly specify acceptable numerical limits on radionuclide releases for the protection of public health and safety and the environment. This section updates the previously developed TLRC in support of planned efforts directed toward the licensing of the Pebble Bed Modular Reactor (PBMR).

The purpose of the TLRC is to establish a fundamental and quantitative basis that is both consistent and unambiguous for judging the current acceptability of potential radionuclide releases such that protection of the public health and safety and the environment is adequately maintained. The quantitative regulatory criteria are established to bound and ensure an acceptable level of health and safety as measured by the risks of radiological consequences to individuals and the environment. The TLRC are based upon existing NRC regulations, safety goals, and guidance.

This section covers the bases for selection of the TLRC, identification of risk criteria (both consequence and frequency), and assessment of the applicability of the TLRC to the PBMR

2.1 BASES FOR SELECTION

The following are the bases for the selection of the TLRC:

- The TLRC should be a necessary and sufficient set of direct statements of acceptable public health and safety as measured by the risks of radiological consequences to individuals and the environment[1].
- The TLRC should be independent of reactor type and site.
- The TLRC should use well defined and quantifiable risk metrics

Basis 1 ensures that the criteria are fundamental to the protection of the public health and safety and the environment.

[1] The term risk as used here implies the definition of a sufficiently complete set of event sequences or scenarios, estimates of their frequencies and consequences, and thorough understanding and quantification of uncertainties in these frequency and consequence estimates. Regulatory criteria may be framed in terms of limits on the aggregate risk levels, by limiting the frequencies of event sequences grouped by similar consequences, by limiting the consequences of events in different frequency ranges, or some combination of these approaches.

Basis 2 requires that the criteria be stated in terms that are generic and applicable to all reactor types and sites.

Basis 3 ensures that the achievement of the criteria can be measured or calculated and that the results of these calculations can be used to make unambiguous conclusions that the criteria have been met.

2.2 RISK CRITERIA

Based upon the top level regulatory criteria selection bases above and a review of current Federal regulations and other pertinent documents, the following regulatory sources have been identified as containing top-level criteria:

- **Reactor Safety Goal Policy Statement** – As documented in Federal Register, Vol. 51, No. 149, pp. 28044-28049, August 4, 1986 (References 2 and 4).

- **10CFR20, Standards For Protection Against Radiation, Subpart D -- Radiation Dose Limits for Individual Members of the Public.** These criteria limit the dose consequences of relatively high frequency events and releases that occur as part of normal plant operations.

- **10CFR50, Appendix I, Numerical Guides for Design Objectives and Limiting Conditions for Operation to Meet the Criterion "As Low as is Reasonably Achievable" for Radioactive Material in Light-Water-Cooled Nuclear Power Reactor Effluents.** These contain cost benefit criteria for normal releases.

- **40CFR190, Environmental Radiation Protection Standards For Nuclear Power Operations.** These are criteria that limit the radiological exposures to workers from routine plant operations.

- **10CFR100, Reactor Site Criteria, Subpart B, Evaluation Factors for Stationary Power Reactor Site Applications on or After January 10, 1997.** These criteria impose limits on the radiological consequences associated with low frequency design basis accidents and hypothetical scenarios selected to qualify the location of the site and the site boundary in relation to nearby population zones.

- **10CFR50.34 (a) (1) Content of Applications; Technical Information: Radiological Dose Consequences.** These dose limits are similar to those in 10CFR100 but expressed in Total Effective Dose Equivalent (TEDE) units.

- **EPA-400-R-92-001, October 1991, U.S. EPA, Manual of Protective Action Guides and Protective Actions for Nuclear Incidents** (Reference 5). These criteria set conditions for initiating offsite emergency protective actions in the

event of a threat of significant radiological exposure that would exceed 10CFR20 and Appendix I limits.

The above consequence limits can be transformed into risk criteria, because the event sequences against which they are applied have understood frequency ranges. For example, 1) the limits in Part 20, Appendix I, and 40 CFR 190 all pertain to normal operations and anticipated operational occurrences; 2) Part 100 and 10 CFR 50.34 pertain to design basis events; and 3) the reactor safety goals and protective action guides generally pertain to severe accidents.

2.2.1 Reactor Safety Goal Policy Statement

The policy statement on reactor safety goals was initiated because of recommendations of the President's Commission on the Accident at Three Mile Island. The content of the policy statement was discussed in many forums before the Commission issued Safety Goals for the Operation of Nuclear Power Plants Policy Statement in 1986, (Reference 2). The Safety Goal Policy Statement expressed the Commission's policy regarding the acceptable level of radiological risk from nuclear power plant operation as follows:

> Individual members of the public should be provided a level of protection from the consequences of nuclear power plant operation such that individuals bear no significant additional risk to life and health.
>
> Societal risks to life and health from nuclear power plant operation should be comparable to or less than the risks of generating electricity by viable competing technologies and should not be a significant addition to other societal risks.

The following quantitative objectives are used in determining achievement of the above safety goals:

> The risk to an average individual in the vicinity of a nuclear power plant of prompt fatalities that might result from reactor accidents should not exceed one-tenth of one percent (0.1 percent) of the sum of prompt fatality risks resulting from other accidents to which members of the U.S. population are generally exposed.
>
> The risk to the population in the area near a nuclear power plant of cancer fatalities that might result from nuclear power plant operation should not exceed one-tenth of one percent (0.1 percent) of the sum of cancer fatality risks resulting from all other causes.

NUREG-0880 provides quantitative data for the determination of incremental risk. This report cites data showing that the individual mortality risk of prompt fatality in the United States is about 5×10^{-4} per year for all accidental causes of death. Applying the prompt mortality risk safety goal for increased incremental risk of no more than 0.1% results in an increase in the individual's annual risk of accidental death by an increment of no more

than 5×10^{-7} per year. This is applicable to the average risk for those individuals who reside at a location within 1 mile of the plant site boundary.

NUREG-0880 also notes that, on average, roughly 19 persons per 10,000 population die annually in the United States as a result of cancer, although the geographic and demographic variation is large. Taking the average rate to be 2×10^{-3} per year and applying the delayed mortality risk safety goal of 0.1% would limit the increase in an individual's annual risk of cancer death to an increment of no more than 2×10^{-6} per year. This is applicable to the average risk for those individuals who reside at a location within 10 miles of the plant.

Based upon the above, it can be seen that the prompt mortality risk safety goal is more restrictive than the delayed mortality risk safety goal. Therefore, the licensing approach conservatively uses the prompt mortality risk safety goal.

2.2.2 10CFR20

This regulation specifies permissible exposure rates, dose levels and activity concentration in restricted and unrestricted areas.

2.2.3 10CFR50 Appendix I

This regulatory requirement identifies numerical guidelines for implementing the objective of as low as reasonably achievable (ALARA). The stated dose values presented are based upon light water reactor operating experience and design features in order to be consistent with the objective of being ALARA. The dose values stated in 10CFR50 Appendix I represent suitable power plant allocations of the overall fuel cycle limits stated by the Environmental Protection Agency in 40CFR190 and in this sense are representative of TLRC.

It should be noted that the cost benefit guideline for judging the necessity for additional radioactive waste system improvements is not included as a top level regulatory criterion since this guidance is not consistent with the criteria selection basis in Section 2. It is not a direct statement of acceptable risk to the public health and safety or the environment.

2.2.4 40CFR190

This regulation specifies both numerical dose criteria intended to protect the health and safety of the public and numerical radionuclide release criteria intended to protect the environment from the consequences of all normal uranium fuel cycle operations. Both limits are consistent with all of the selection bases and are included as TLRC.
The numerical criteria of 40CFR190 and 10CFR50 Appendix I are complementary and the PBMR would be assessed against both. Appendix I provides limits on the dose due to effluents from an individual reactor, including the allocations from shared facilities. In contrast, 40CFR190 sets a limit on exposure from all sources both effluent and direct

from the plant fuel cycle. On a site specific basis, one or the other may prove to be more limiting, depending on the existence of any other contributing plants or uranium fuel facilities in the vicinity and the expected types and levels of effluents. Accordingly both Appendix I and 40CFR190 are included as TLRC and the maximum allowable dose to any member of the public shall be the lower of the limits established by their application.

2.2.5 10CFR50.34

This regulation provides the guidance for determining site suitability for accident radioactive releases and is consistent with all the Section 2 identified bases and therefore qualifies as TLRC. The dose guidance specifies a limit for Total Effective Dose Equivalent (TEDE) values.

However, the analysis assumptions used in implementing these dose guidelines needs to be oriented to the characteristics of the specific reactor type and design. In particular the source term guidance given in TID 14844 (Reference 7) or 10CFR50.67 Accident Source Term as defined in NUREG 1465 (Reference 8) are applicable for light water reactors and are not appropriate for the PBMR.

This technical content of applications section of the regulation specifies the dose consequence criteria following design basis accidents as:

- An individual located at any point on the boundary of the exclusion area for any 2 hour period following the onset of the postulated fission product release, would not receive a radiation dose in excess of 25 rem total effective dose equivalent (TEDE).

- An individual located at any point on the outer boundary of the low population zone, who is exposed to the radioactive cloud resulting from the postulated fission product release (during the entire period of its passage) would not receive a radiation dose in excess of 25 rem TEDE.

2.2.6 EPA-400-R-92-001

This EPA manual (Reference 5) provides updated guidance for emergency planning and Protective Action Guides (PAGs), replacing those previously given in EPA-520/1-75-001, for exposure to airborne radioactive materials due to a nuclear incident. The PAGs for responses during the early and intermediate phases following an incident are now expressed in terms of the projected sum of the effective dose equivalent from external radiation and the committed effective dose equivalent incurred from inhalation of radioactive materials from exposure and intake. Supplementary guides are also specified in terms of committed dose equivalent to the thyroid and dose equivalent to the skin. This more complete guidance updates and replaces previous values, expressed in terms of whole body dose equivalent from external gamma exposure and thyroid dose equivalent from inhalation of radioactive iodine. However, Reference 5 incorporates directly the PAGs for contaminated foodstuffs previously published by the FDA in 1982 (Reference

10). As before the rationale for the selection of these dose guides is not reactor design specific.

The NRC implementation requirements in 10CFR50 Section 50.47 and Appendix E for emergency planning generally specify a plume exposure pathway Emergency Planning Zone (EPZ) of 10 miles in radius and an ingestion pathway EPZ of 50 miles in radius provide an adequate planning basis. The technical basis for the selection of these EPZ distances is given in NUREG-0396 (Reference 11) wherein it is found for the majority of light water reactors (LWRs) that for all but the most improbable events, the PAGs would not be expected to be exceeded beyond these distances.

As noted earlier, even though the above criteria appear to limit consequences and are not framed in the context of risk, they are still regarded as risk criteria because there are implied scenarios and frequency ranges that are used with the consequence criteria.

2.3 FREQUENCY REGIONS

The development of Frequency Regions is performed in the manner described in Reference 3 and consists of a spectrum of releases covering a frequency range from normal operation to very low probability off-normal events. The spectrum of potential accidental radioactive releases from a plant are divided in the following three regions in a scenario frequency vs. consequence chart.

- Anticipated Operational Occurrences (AOO)
- Design Basis Events (DBE)
- Emergency Planning Basis Events (EPBE)

An examination of the entire frequency range and the identification of one or more of the TLRC as being applicable for each Region provide assurance that the selected criteria are adequately established. A summary of the TLRC and their applicable frequency ranges are provided in Table 2-1.

2.3.1 Anticipated Operational Occurrences Region

Anticipated Operational Occurrences are those conditions of normal operation which are expected to occur one or more times during the life of the plant. Using a licensing basis design lifetime of 40 years yields a lower boundary for the AOO region of 2.5×10^{-2} per plant year. For this Region, 10CFR50, Appendix I is the applicable criteria as it specifies the numerical guidance to assure that releases of radioactive material to unrestricted areas during normal reactor operations, including AOOs, are maintained As Low As Reasonably Achievable (ALARA).

2.3.2 Design Basis Event Region

The Design Basis Event Region encompasses releases that are not expected to occur during the lifetime of one nuclear power plant. The frequency range covers events that are expected to occur during the lifetime of a population (several hundred) of nuclear power plants; and therefore a lower limit of 10^{-4} per plant year is chosen. This frequency is consistent with the frequency of DBEs for existing LWRs . Estimates of LWR core damage accidents, which exceed the design basis, have been in the range of $1x10^{-5}$ to greater than $1x10^{-4}$. For this region, 10CFR50.34 (a)(1) provides the quantitative dose guidance for accidental releases for siting a nuclear power plant to ensure that the surrounding population is adequately protected.

2.3.3 Emergency Planning Basis Event Region

The Emergency Planning Basis Event Region considers improbable events that are not expected to occur during the lifetime of several hundred nuclear power plants. This is to assure that the risk to the public from low probability events is acceptable, and that adequate emergency planning is developed to protect the public from undesirable exposure to radiation for improbable events. The frequency cutoff implicit in the acute fatality risk goal in NUREG-0880 is taken as the lower frequency boundary of the EPBE Region. NUREG-0880 notes that the individual mortality risk of prompt fatality in the U. S. is about 5×10^{-4} per year for all accidental causes of death. The prompt mortality risk design objective limits the increase in an individual's annual risk of accidental death to 0.1% of 5×10^{-4}, or an incremental increase of no more than 5×10^{-7} per year. If the frequency of a scenario or set of scenarios is at or below this value, it can be assured that the individual risk contributions from these scenarios would still be within the safety goal independent of the magnitude of consequences. Therefore this value is used as the lower frequency bound for the EPBE Region.

Table 2-1

TOP LEVEL REGULATORY CRITERIA

Policy Statement of Reactor Safety Goals

Purpose: To specify acceptable incremental risk of prompt and delayed fatality to an individual and the population due to local siting of a power plant.

Frequency Range: Normal Operation to 5×10^{-7} per plant year

1. Individual Prompt Mortality Risks:

   The risk to an average individual in the vicinity of a nuclear power plant of prompt fatalities that might result form reactor accidents should not exceed 0.1% of the sum of prompt fatality risks resulting from other accidents to which members of the U. S. population are generally exposed.

   Based on the quantitative regulatory guidance documents (References 2 and 6), the incremental risk to the average individual within 1 mile of a nuclear power plant site boundary shall be no more than 5×10^{-7} per plant year.

2. Individual Delayed Mortality Risk:

   The risk to the population in the area near a nuclear power plant of cancer fatalities that might result from nuclear power plant operation should not exceed 0.1% of the sum of cancer fatality risks resulting from all other causes.

   Based on the quantitative regulatory guidance documents (References 2 and 6), the incremental risk to the average individual within 10 miles of a nuclear power plant site boundary shall be no more than 2×10^{-6} per plant year.

10CFR20

Purpose: To specify acceptable occupational and public exposures and offsite releases in effluents.

Frequency Range: Normal Operation to 2.5×10^{-2} per plant year

1. Section 20.1201 - Occupational dose limits for adults.

   An annual limit, which is the more limiting of:

   (i) the total effective dose equivalent being equal to 5 rem (0.05 Sv); or

(ii) the sum of the deep-dose equivalent and the committed dose equivalent to any individual organ or tissue other than the lens of the eye being equal to 50 rem (0.5 Sv).

The annual limits to the lens of the eye, to the skin, and to the extremities, which are: (i) a lens dose equivalent of 15 rem (0.15 Sv), and (ii) a shallow-dose equivalent of 50 rem (0.50 Sv) to the skin or to any extremity.

Derived air concentration (DAC) and annual limit on intake (ALI) values are presented in Table 1 of Appendix B to Part 20 and may be used to determine the individual's dose (see §20.2106) and to demonstrate compliance with the occupational dose limits.

The dose that an individual may be allowed to receive in the current year shall be reduced by the amount of occupational dose received while employed by any other person (see §20.2104(e)).

2. Section 20.1301 –Dose limits for individual members of the public:

Total effective dose equivalent (TEDE) < 100 mrem per year.

Dose in unrestricted area from external sources < 2 mrem in any one hour.

3. Section 20.1302 – Compliance with dose limits for individual members of the public

Demonstrate by measurement or calculation that the TEDE to the individual likely to receive the highest dose does not exceed the annual dose limit or demonstrate that:
(i) the annual average concentrations of radioactive material released in gaseous and liquid effluents at the boundary of the unrestricted area do not exceed the values specified in Table 2 of Appendix B to Part 20 and
(ii) if an individual were continuously present in an unrestricted area, the dose from external sources would not exceed 2 mrem in an hour and 50 mrem in a year.

<u>10CFR50 Appendix I</u>

Purpose: To specify acceptable offsite exposures during normal operation and anticipated events for an individual reactor.

Frequency Range: Normal operation to 2.5×10^{-2} per reactor year.

Section II: Guides on design objectives for LWR:

1. Paragraph A:

Estimated annual dose from liquid effluents: ≤ 3 mrem total body, or ≤ 10 mrem to any organ.

2. Paragraph B:

   Estimated annual dose from gaseous effluents: ≤ 5 mrem total body or ≤ 15 mrem to the skin.

3. Paragraph C:

   Estimated annual dose from all radioactive iodine and radioactive material in particulate form in effluents to the atmosphere: ≤ 15 mrem to any organ.

<u>**40CFR190**</u>

Purpose: To specify acceptable offsite exposures and releases due to the entire uranium fuel cycle.

Frequency Range Normal operation to 2.5×10^{-2} per reactor year.

1. Section 190.10 (a) – Annual dose equivalent to a member of the general public from uranium fuel cycle operations (as defined in 190.02).

Whole body dose:	≤ 25 mrem
Thyroid dose:	≤ 75 mrem
Any other organ dose:	≤ 25 mrem

2. Section 190.10 (b) – Total quantity of radioactive materials entering the general environment from the entire uranium fuel cycle, per gigawatt-year of electrical energy produced by the fuel cycle:

   Kr-85 < 50,000 curies
   I-129 < 5 millicuries
   Pu and other α emitting and transuranic nuclides with half-lives:> 1 yr
   < 0.5 millicuries.

<u>**10CFR100/10CFR50.34 (a) (1)**</u>

Purpose: To specify acceptable offsite exposures resulting from unanticipated off-normal events.

Frequency Range: 2.5×10^{-2} to 1×10^{-4} per plant year

1. An individual located at any point on the boundary of the exclusion area for any 2 hour period following the onset of the postulated fission product release, would not receive a radiation dose in excess of 25 rem TEDE.
2. An individual located at any point on the outer boundary of the low population zone, who is exposed to the radioactive cloud resulting from the postulated fission product release (during the entire period of its passage) would not receive a radiation dose in excess of 25 rem TEDE.

Dose Protective Action Guides (PAGs) of EPA 400-R-92-001

Purpose: To specify offsite exposures at the plume exposure or ingestion pathway EPZ for initiating public protection due to airborne and food pathway radioactive materials resulting from unanticipated off-normal events.

Frequency Range: 2.5×10^{-2} to 5×10^{-7} per plant year

1. Protective Action Guides for Early Phase of Nuclear Incident (Exposure to Airborne Radioactive Materials):

Protective Action	**PAG Projected Dose**
Evacuation (or Sheltering) for general population if dose	> 1 to 5 rem[a]
Administration of stable iodine:	> 25 rem[b].

Dose Limits for Workers Performing Emergency Services (rem[c]):

All Activities	5
Protecting Valuable Property	10
Lifesaving or protection of large populations	25
(on a voluntary basis to persons fully aware of risks)	>25

[a] The sum of the effective dose equivalent resulting from exposure to external sources and the committed effective dose equivalent incurred form all significant pathways during the early phase. Committed dose equivalents to the thyroid and to the skin may be 5 and 50 times larger, respectively.

[b] Committed dose equivalent to the thyroid from radioiodine

[c] Sum of external effective dose equivalent and committed effective dose equivalent to non-pregnant adults from exposure and intake during an emergency situation. Workers performing services during emergencies should limit dose to the lens of the eye to 3 times the listed value and doses to any other organ (including skin and body extremities) to 10 times the listed value. These limits apply to all doses from an incident, except those received in unrestricted areas as members of the public during the intermediate phase of the incident.

2. Protective Action Guides for Exposure to Deposited Radioactivity During the Intermediate Phase of a Nuclear Incident

Protective Action	PAG Projected Dose[d]
Relocate the general population (beta dose to the skin may be up to 50 times higher).	≥ 2 rem
Apply simple dose reduction techniques	< 2 rem

[d] The projected sum of effective dose equivalent from external gamma radiation and committed effective dose equivalent from inhalation of resuspended materials, from exposure or intake during the first year. Projected dose refers to the dose that would be received in the absence of shielding from structures or the application of dose reduction techniques.

3. Protective Action Guides for Exposure from Materials via the Food Pathway

Protective Action	PAG Projected Dose
Preventive:	0.5 rem[e] 1.5 rem[f]

(Preventive action is to reduce the radioactive contamination of human food or animal feed)

Protective Action	PAG Projected Dose
Emergency:	5 rem[e] 15 rem[f]

(Emergency action is to isolate food containing radioactivity to prevent its introduction into commerce and the level at which the responsible officials should determine whether condemnation or another disposition is appropriate.)

[e] Dose to the whole body, bone marrow or any other organ.
[f] Dose to the thyroid

2.4 TOP LEVEL REGULATORY CRITERIA APPLIED TO PBMR

The regulatory criteria developed and presented in Section 2.3 are applicable to all nuclear power reactor types inclusive of the PBMR. The consequence criteria are based upon current regulation and regulatory guidance. The frequency criteria represent a reasonable quantification derived from current regulatory sources. Recognizing that some interpretation may be appropriate and that more stringent criteria may be imposed by the plant user and designer, specific applications of the criteria have been adopted for a PMBR.

2.4.1 Module Versus Plant Or Site

In the evaluation of the PBMR, the consequence criteria given in Table 2-1 (including 10CFR50 Appendix I) are numerically taken to be independent of a reactor plant, i.e., the criteria apply to a plant regardless of the number of reactors or modules. However, the criteria are for the licensing application of a new plant, which may or may not be on a site with existing, previously licensed reactors/plants. The existence of multiple modules would increase the frequency of single module related scenarios and create the potential for scenarios involving multiple modules concurrently. By contrast, in the case of LWR safety goals and other relevant criteria have been applied to each reactor unit independently. Thus, in the case of the PBMR, the total impact of installing multiple modules (up to 10) will be evaluated similar to the equivalent impact of adding a single, large LWR in the same location. In other words, in determining whether a 10-module PBMR facility satisfies the TLRC, the licensing approach considers the cumulative risk posed by the ten modules, rather than considering each module separately.

2.4.2 Distance Criteria

The dose guidelines of 10CFR50.34 (a) (1) apply to an individual located at any point on the boundary of the exclusion area for <u>any</u> 2 hour period following the onset of the postulated fission product release, and to an individual located at any point on the outer boundary of the low population zone, who is exposed to the radioactive cloud resulting from the postulated fission product release, during the entire period of its passage.

The safety goals for prompt and delayed mortality are applied at one and ten miles, respectively. However, for the PBMR, all offsite dose criteria and safety goals are to be met for the maximum exposed individual at the boundary of the exclusion area which is assumed to be 400 meters from the nearest module.

2.4.3 10CFR50 Appendix I

The dose criteria are expressed in terms of the expected annual dose at the site boundary along the plume centerline. Hence, for an event expected to occur twice per year the total dose from two events is compared to the Appendix I annual limit. This is used to derive an equivalent allowable dose for each event. For frequent events occurring more than once a year, this results in the sloped risk line shown in Figure 2-1. For less frequent

events within the plant lifetime, no single event may exceed the allowable dose as indicated by the vertical dose line in the figure. Appendix I is the most limiting requirement of those identified for normal operation and anticipated operational occurrences

2.4.4 10CFR50.34.(a) (1)

In the design basis region, acknowledgement that relatively more frequently occurring events should meet more stringent criteria leads to the sloping dose criteria line. At the lower end (i.e. 10^{-4} per plant year), the criteria are 100% of the limit dose. The criteria linearly decrease to the upper end where 10% of the limit is used. This is consistent with the NRC's qualitative criterion, as reflected in the Standard Review Plan guidance, that the dose limitations from more frequent accidents be a fraction of the dose guidelines. The dose criteria are expressed in TEDE at the site exclusion area boundary (EAB). The 10CFR50.34.(a) (1) criteria are more limiting than the Reactor Safety Goals.

2.4.5 Prompt Mortality Safety Goal

The incremental mortality cancer risk allowed by the safety goal is 5×10^{-7} fatalities per year. The illustration of the prompt mortality risk curve displayed in Figure 2-1 is approximated and presented in terms of whole body dose in rem. The prompt mortality risk is more limiting than the latent fatality risk. The use of the safety goals to draw the criteria line in this region is very conservative when applied to the dose at the site boundary along the plume centerline as a person at this point would be located at the point of maximum risk over the area within 1-mile of the site boundary in which the average individual risk must meet the safety goal. When the individual risk at this point meets the safety goal, the average individual risk within 1-mile of the site boundary would be much less than 5×10^{-7} per year value.

2.4.6 Dose Protective Action Guides

Protective Action Guidelines (PAG) from EPA-400-R-92-001 are shown as a dose limit as expressed in TEDE at the emergency planning zone (EPZ). The PAG apply to the design and emergency planning basis regions. Depending on the size of the EPZ, the PAG can be the most limiting criteria. The PBMR will be designed with the option to preclude the need for offsite sheltering, that is, the EPZ would be at the EAB (for sites without existing nuclear power plants with larger EPZ).

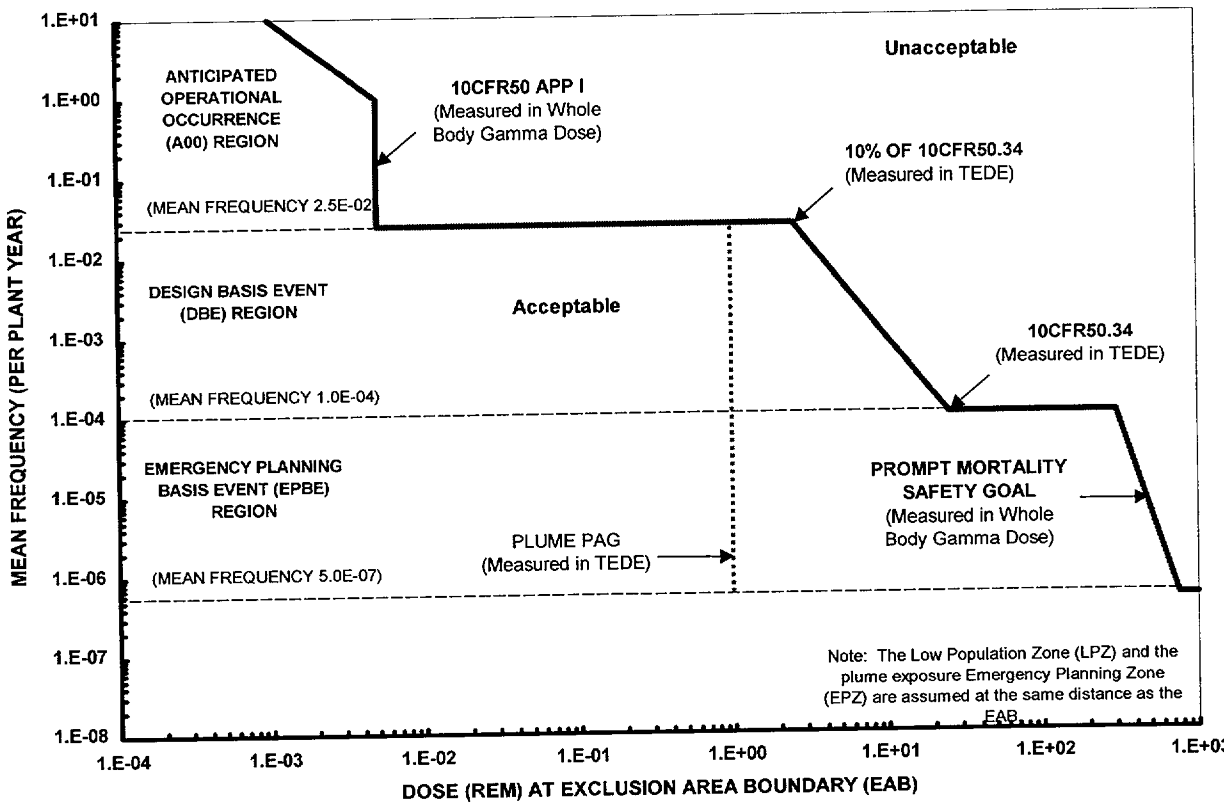

Figure 2-1

APPLICATION OF TOP LEVEL REGULATORY CRITERIA FOR THE PBMR

3 IDENTIFICATION OF APPLICABLE REGULATIONS

In order to effectively implement the licensing approach, two steps will be used to create a focused list of documents as the regulatory set for PBMR. The first step is a rough screening of documents for applicability, and the second step is to refine the focus of the applicable regulations using available design and risk-informed insights to develop the proposed regulatory requirements set. The purpose of this section is to define the first step. Sections 2, 4, and 5 of this document define a process that leads to the second step.

Given the very limited regulatory experience with gas reactor technology in the US, there is not an existing body of regulations directly suited to the PBMR design. Consequently, for a license application to be prepared, a different set of regulations, regulatory guides and standard review plans will have to be crafted out of the existing regulatory body to guide both the applicant in preparation of their license applications and NRC in their review of them. This situation is recognized in the Introduction to Appendix A to 10 CFR 50 and NRC's Policy on Advanced Reactors. Appendix A states that the general design criteria (GDC) were developed for LWRs and are intended to provide guidance in establishing the principal design criteria for other types of reactors. In the policy statement, the Commission encourages applicants to interact very early with the NRC Staff and to propose ways to better guide the application development and review of the advanced design. That is the objective of this regulatory screening process.

A top-down, safety focused method to create a specific regulatory set using a combination of deterministic and risk-informed techniques to decide specific requirements for the design has been discussed above. Using this process, it has been recognized by Exelon and NRC that some of the current design-related regulations are fully applicable to any design, some are not applicable to gas reactors, and many may be partially applicable. There may also be some features of the PBMR design that cannot be addressed by any current regulatory document, thus requiring new guidance documents to be developed or other agreements reached between Exelon and the NRC during the pre-application period. The process of addressing deterministic and risk-informed objectives is represented in Figure 3-1 (Same figure as Figure 1-2).

As stated above implementation of the proposed licensing approach consists of two steps. The acceptability of the end products will be determined with the NRC in the pre-application period. The outcome expected is to create early agreements where possible and a greater confidence that the license applications will provide the proper information for an efficient and effective regulatory review. Additionally, by determining early on which regulations apply to the PBMR design and which do not, Exelon and the NRC will have a much better understanding of how to navigate through the legal and procedural steps to obtaining exemptions or other suitable relief from existing regulations geared to light water reactors. The process described below defines how a Exelon preliminary screening of the current regulations was conducted.

3.1 Purpose of Regulatory Document Screening

The purpose of a pilot regulatory document screening was to develop a method for preliminarily determining applicability of current regulatory documents to the PBMR design. The pilot examined a large sample of regulations that could apply to the licensing of the PBMR in the US. The pilot project also provided:

- A greater sense of the number of exemptions that could be required in the process of reviewing the PBMR design;
- A greater sense of what the key questions and logic are for making decisions regarding applicability of current regulatory guidance documents at the regulation level and at lower levels;
- A beginning point for applying risk-informed insights to help shape the changes or interpretations that will be needed to address partially applicable regulations.
- Confidence that a logical, repeatable, reliable and defendable decision process can be defined for addressing the remaining large set of regulatory guidance documents in existence today.

3.2 Pilot Process

A Delphi process was utilized, i.e., subject matter experts to screen the sample set and develop the common logic from the group process. The expert panel consisted of a group of individuals with diverse backgrounds. The panel consisted of six participants with the following industry perspectives; owner, regulator, designer and legal. These members had more than 180 years of total nuclear industry experience. Backgrounds included experience in LWR and gas reactor design, operations, maintenance, construction, licensing, reactor regulation and risk assessment.

Each of the individuals on the expert panel was separately provided with an advanced sample of approximately 160 regulations (largely consisting of the regulations in Part 50 plus selected other regulations). Each individual was directed to review each regulation and designate the regulation as applicable, partially applicable or not applicable to the PBMR design. Each panel member was to independently assess and "vote" on which category a given regulation falls into. The vote was recorded on the individual ballot sheets provided in the pre-meeting package. Additionally, each panelist made notes on key considerations or personal questions that shaped their decision-making process. This package was completed before the meeting without collaboration with other members of the panel.

The expectation was that the experience of the panel was greater than typical reviewers and would provide an accurate initial reaction to the question of applicability. With the diverse backgrounds of the panel, some variability was expected in the results for some of the regulations and those differences provided needed insights for deriving a consistent future process. For that reason, it was important to capture the logic for each reviewer's "votes." For example, if the legal interpretation of the regulation drove the decision, "legal" was to be noted as the basis (and so forth for partial technical, intent or purpose of

the regulations, etc.). If there were multiple considerations, they were to be so noted. Extensive commentary was not required to explain or defend the answer.

To guide the initial effort, for the large list of regulations provided, reviewers were directed to not spend more than eight hours total on the pre-meeting preparation of ballots.

The panel captured the answers on each regulation and discussed individual significant points of departure. At the end of the session, time was devoted to discussing the common elements of the decision process that emerged from the review and a draft decision logic prepared. A small sample of additional regulations was reviewed with the new composite logic to test its utility for others to use. The panel was also polled for other ideas that could make the screening process more effective. Finally, the panel discussed PBMR design-specific topics that could potentially be needed in a new set of application or review guidance documents because they are not addressed at all in the current NRC regulatory set.

Finally, the panel discussed whether the process appears suitable as a means to screen the current set regulations and lower tier documents in to a more focused PBMR-oriented population.

The PBMR project team compiled all the results from each panelist and from the common discussions. The results were summarized for presentation to the NRC on what the findings were from a process point of view and what insights were gained on the potential exemptions, changes, etc. that could be required to support a PBMR application.

The process yielded several products that will help shape future activities. First, the process provided a preliminary view on the applicability of each of the regulations in 10CFR50 plus a partial set of other rules that could be used to shape the application and review requirements for the PBMR design. The preliminary results are provided in Appendix A. For the 163 total regulations and appendices screened, 115 were viewed as "applicable", 22 "partially applicable" and 26 "not applicable". Five of the "not applicable" topics were also identified as needing a PBMR-specific replacement[2]. As these results demonstrate, the expert panel determined that a substantial fraction of the regulations are applicable in whole or part to the PBMR.

In determining the applicability of the regulations to the PBMR, the expert panel considered two questions: 1) does the regulation literally apply to gas-cooled reactors; and 2) if not (e.g., if the regulation on its face applies only to LWRs), is the regulation useful as guidance for the PMBR. If either of these questions was answered yes, the regulation was designated as applicable to the PBMR (or partially applicable in cases in which the regulations has multiple parts, part of which is applicable and part which is

[2] Subsequent to discussion with the NRC on August 9, 2001 Appendix B was generated. This appendix further breaks down the results to differentiate the focus of the regulations, i.e., Technical Design, Administrative, Operations/Maintenance, or non-reactor, and whether the degree of applicability was considered guidance.

not). In those cases in which a regulation was applicable or partially applicable, the expert panel also determined whether the regulation was applicable as a legal requirement or as guidance (i.e., a LWR regulation that will be applied as guidance to the PBMR).

Given that the effort was a pilot for future more detailed examinations of the complete regulatory set for utilization license applicants, the development of a standardized logic chart was one of the deliverables. The resulting preliminary chart is shown as Figure 3-2. The final chart will be used to assess each regulation, regulatory guide, standard review plan or other required regulatory reference to determine how it applies to the PBMR design.

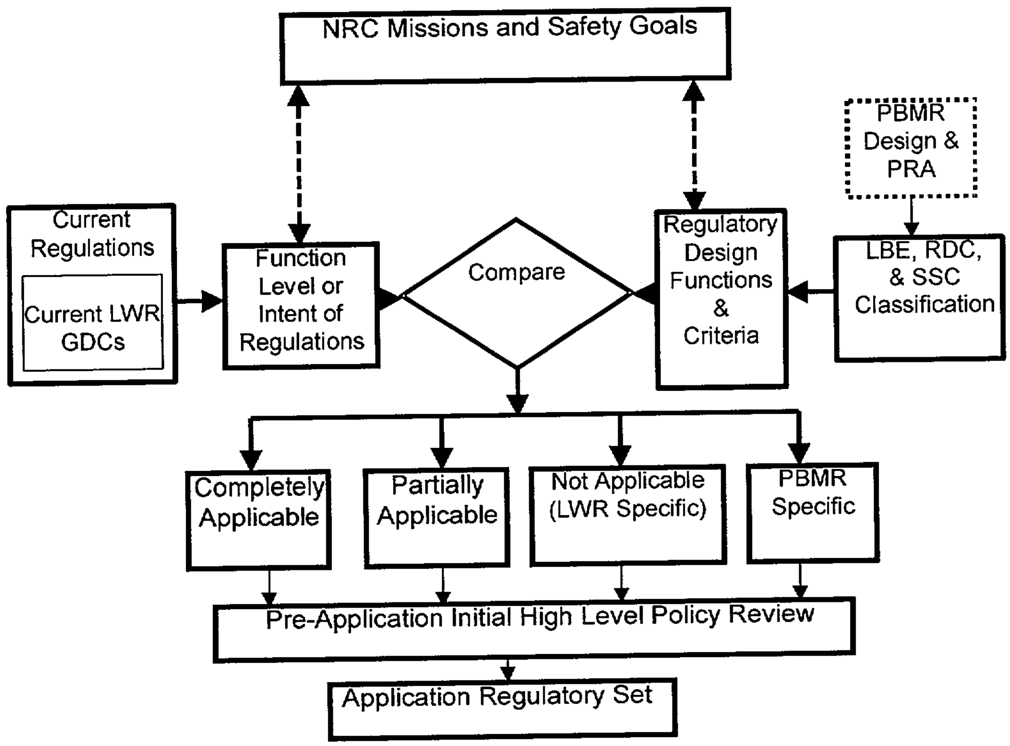

Figure 3-1

COMPARISON OF RISK-INFORMED LICENSING BASES WITH CURRENT REGULATIONS

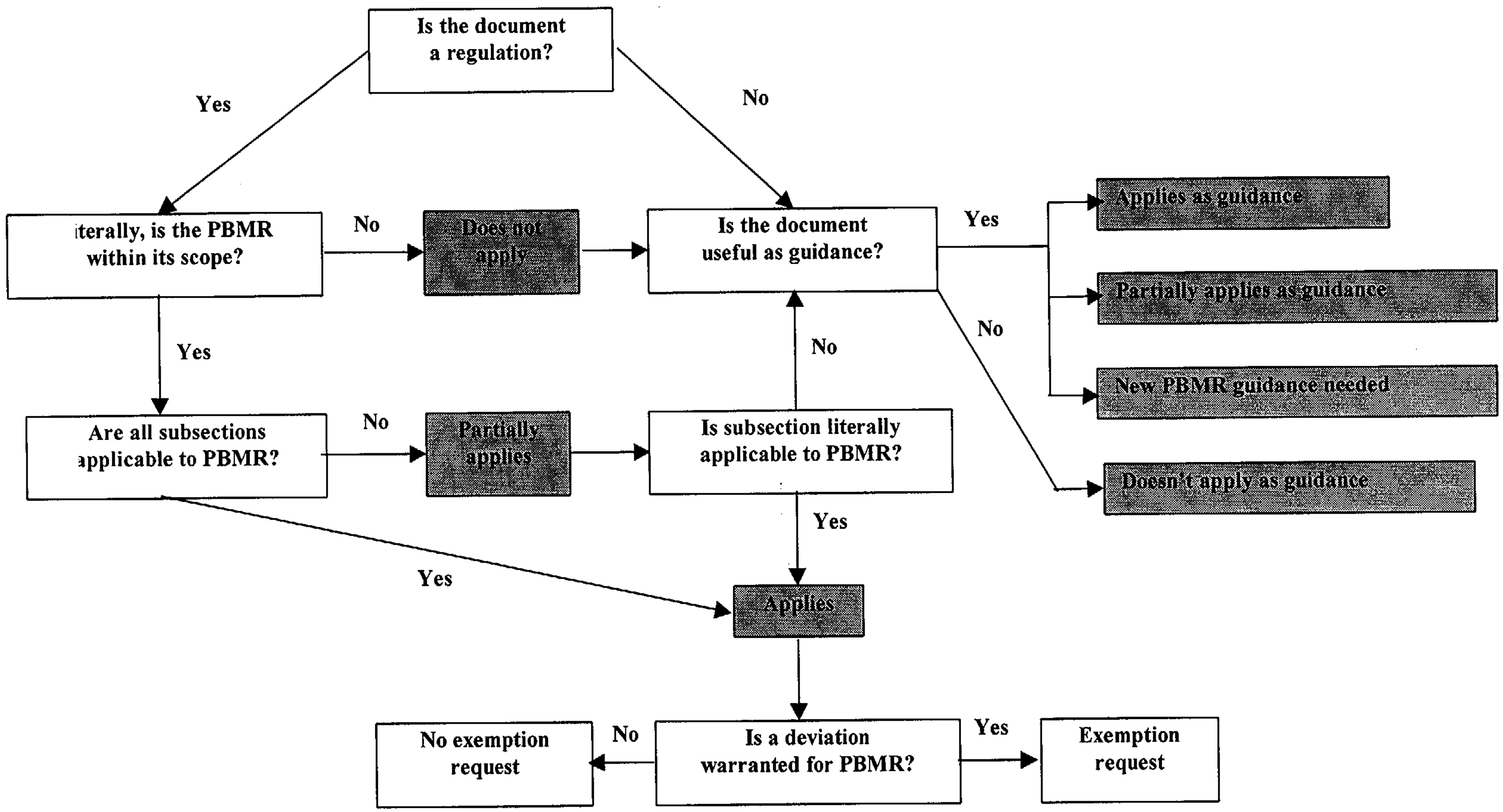

Figure 3-2

PBMR REGULATORY SCREENING PROCESS

3.3 Examples of Pilot Screening

Several examples are provided to explain how the logic chart in Appendix B is used to reach the classifications shown in Appendix A. Three examples will be shown that are considered straight forward applications of the process:

- Applies [50.59]
- Partially applies [50.54(o)]
- Does not apply [50.44]

Example 1 - SECTION 50.59 - CHANGES, TESTS AND EXPERIMENTS

The first consideration is whether the regulatory document is a regulation. For this example, the answer is yes, since Section 50.59 is a regulation. The second consideration is whether the PBMR design is within the literal scope of the regulation. In this example, the answer is yes since the regulation says it applies to all licensees with production or utilization licenses. The next consideration is whether the entire regulation applies to the PBMR in all subparts or some subparts specifically provide design-dependent requirements. In this example, all subparts apply to the PBMR, therefore the requirement is considered fully applicable. Finally, the last step is to consider whether the PBMR approach will include deviations from the applicable document. In this example the answer is no. Thus there will not be any need for exemption to this regulation.

Example 2 - SECTION 50.54(o) - CONDITIONS OF LICENSES
Testing of primary reactor containments for water-cooled power reactors

The first consideration is whether the regulatory document is a regulation. For this example, the answer is yes. The second consideration is whether the PBMR design is within the literal scope of the regulation. In this example, the answer is no since the regulation says it applies to water cooled reactors. The next step is to determine whether the regulation is useful as guidance for the PBMR. The answer to that question is yes, since the PBMR will have a containment. The next consideration is whether the entire regulation applies to the PBMR in all subparts or some subparts specifically provide design-dependent requirements. In this example, all subparts do not apply to the PBMR, therefore this regulation is considered to partially apply and the requirement requires further examination as to whether it provides guidance that should be applied to the PBMR. In this example, using general knowledge of the PBMR design, although the PBMR is not a water-cooled reactor, it does have containment functions included in the design, therefore the stated purpose of the regulation is useful as guidance, i.e., the requirement to conduct appropriate leakage testing for the containment functions consistent with the requirements of the design. The next step is to consider whether as guidance the document is fully applicable or partially applicable. In this example, the document is partially applicable as guidance since the reference to Appendix J - PRIMARY REACTOR CONTAINMENT LEAKAGE TESTING FOR WATER-COOLED POWER REACTORS is not fully applicable.

Example 3 - SECTION 50.44 - STANDARDS FOR COMBUSTIBLE GAS CONTROL SYSTEM IN LIGHT-WATER-COOLED POWER REACTORS

The first consideration is whether the regulatory document is a regulation. For this example, the answer is yes. The second consideration is whether the PBMR design is within the scope of the regulation. In this example, the answer is no since the regulation says it applies only to LWRs. The next consideration is whether the regulation is nevertheless useful as guidance. In this example, the regulation provides no useful guidance since it specifically relates to metal-clad fuel in water-cooled reactors.

There are other regulations that require greater judgement and design insight to make the applicability determination. Examples of these regulations are:

- Applies - SECTION 50.75 - REPORTING AND RECORDKEEPING FOR DECOMMISSIONING PLANNING
- Partially applies - SECTION 50.49 - ENVIRONMENTAL QUALIFICATION OF ELECTRIC EQUIPMENT IMPORTANT TO SAFETY FOR NUCLEAR POWER PLANTS
- Does not apply - SECTION 50.46 - ACCEPTANCE CRITERIA FOR EMERGENCY CORE COOLING SYSTEMS FOR LIGHT-WATER NUCLEAR POWER REACTORS

In some cases, there may be some safety objective or purpose of regulations that, while the legal requirement or guidance may be not literally applicable, nevertheless lead to a conclusion that a PBMR application should contain certain information for staff review. For example, the staff review of accident evaluation models is an appropriate consideration for any reactor type. However, since there is no direct existing guidance for the PMBR design, new guidance may have to be developed with the staff in order to complete the application. This process should identify areas where this is appropriate to reduce uncertainty on the content of the application. The preapplication period should be utilized to define these new application requirements.

3.4 Process Considerations

In order to achieve consistent answers, the panel found it necessary to standardize PBMR design-specific meanings for some common terms in the regulations. The following understandings were needed:

> Reactor Coolant Pressure Boundary – The regulations and General Design Criteria (GDC) frequently refer to the Reactor Coolant Pressure Boundary (RCPB). 10CFR50.2 defines RCPB as certain pressure containing components for an LWR. Therefore, the term RCPB is not literally applicable to the PBMR. However, the role played by the RCPB in light water reactor safety has some applicability but not complete applicability to the PBMR. For the PMBR, the safety functions of the system are different. The primary system serves as one barrier to fission product release; assures core geometry is retained; protects against chemical attack; and removes decay heat via conduction and radiation. The primary system does not have to retain its normal helium content for convective cooling. Therefore when considering regulations referring to the RCPB, the function was considered to be a Reactor Pressure Boundary (RPB).

Containment – The PBMR design does include a containment function. However, the functional requirements are fundamentally different than low leakage containments found on LWRs. For the PBMR, the containment design was considered to be a high leakage, vented containment comprised of two concentric boundaries. The inner boundary is the citadel. It surrounds the reactor pressure boundary and is intended to provide protection against external events and certain internal events. The second barrier is the outer module building confinement structure. Both enclosures contain separate ventilation systems to filter the contained atmosphere.

Loss of coolant accidents – The PBMR does not have loss of coolant accidents similar to LWRs. Breaches in the RPB resulting in depressurized loss of forced cooling release the helium from the circuit and stop the convective flow through the reactor. For the larger breaks, the entire helium inventory can be vented to atmosphere, but decay heat can still be removed passively by conduction and radiation heat losses to the heat sink or the surrounding building and structures.

Merchant Plants – The advent of merchant generators in a deregulated utility market create conditions not contemplated in some existing regulatory requirements. These requirements are process-related requirements, rather than design requirements. Exemptions from the requirements for merchant owners may be appropriate to specifically address these conditions.

Modular Plants – The advent of small, modular reactor designs create conditions not contemplated in some existing regulatory requirements. These requirements are process-related requirements or operational requirements (e.g., number of licensed operators), rather than design requirements. Exemptions from the requirements for modular designs may be appropriate to specifically address these conditions.

Additionally, it was necessary to have knowledge of the PBMR design and its fundamental differences from LWR in order to consider the guidance value of many of the LWR based regulations. Detailed design knowledge was not considered necessary. Similarly, it was necessary to have a rudimentary understanding of normal and transient event sequences including potential licensing basis events for the PBMR design. Based on knowledge of LBE, it could then be concluded what functional capabilities are necessary in the design to satisfy safety missions. Finally, comparisons were made between the functional capabilities and the current regulatory set to determine level of applicability of each regulation.

Finally, in order to distinguish how the regulations applied, it was necessary to differentiate between literal applicability and applicability for guidance or insight. This provided a necessary step between legal interpretations of applicability and consideration of the underlying purpose of the regulation. It was thus necessary as example to evaluate the guidance contained in the GDC rather than simply exclude them based on their reference to LWR designs. This also gave rise to an additional result with respect to the need for exemptions to regulations. It is not necessary to request exemptions for deviations from regulations that do not apply on their face to a given reactor. For example, pressurized water reactors do not have to address requirements unique to

boiling water reactors nor do they have to be granted an exemption from those requirements clearly stated as applying to boiling water reactors. This same practice has been carried forward in this process. As a result, Exelon will not need to request an exemption for deviations that apply on their face to LWRs or to LWR component such as zircaloy clad fuel.

3.5 Follow-on Steps

It in important to obtain agreement during the pre-application period on the use and utility of the resulting logic diagram and needed definitions in order to prepare a high quality application for the PBMR. This will enable a properly focused application that can be efficiently and effectively reviewed and address the necessary and sufficient requirements to provide reasonable assurance of public safety and security for the PBMR design.

Once agreement is reached with the NRC on the process and logic to be used, Exelon will screen the entire set of NRC regulations to determine applicability, partial applicability and inapplicability. During the expanded screening effort, Exelon will continue to validate the logic chart. Also, actual PBMR design information will be used as it becomes available as well as risk insights from PRA work as it becomes available to refine if needed the decisions on specific partial or not applicable documents. At the completion of the regulation screening, the results will be reviewed with NRC. The final logic chart will be confirmed for use to examine sub-tier regulatory documents that stem from the regulations such as regulatory guides and the standard review plan.

It is the intent of the process that NRC and Exelon will use the pre-application period to assure that the full set of regulatory documents that must be considered in whole or in part for the PBMR design are identified in advance of completing the application.

The process described above is an iterative process. For example, the determination of whether a LWR regulation is useful in whole or part as guidance for the PBMR depends upon the design of the PBMR. However, the design of the PBMR will be affected by the guidance that is applied during the design process. Thus, it may be that the both the design and the identification of applicable regulations will evolve over time, with finality being achieved once the license is issued. At the pre-application stage, Exelon is only seeking a tentative agreement with the NRC staff on what regulations are applicable or might be applicable, realizing that the NRC (and Exelon) cannot make a final determination until both have had the opportunity to review the design and the design itself is final.

4 SELECTION OF LICENSING BASIS EVENTS

4.1 USE OF PROBABILISTIC RISK ASSESSMENT (PRA)

The purpose of this section is to define the objectives, scope, level of detail, treatment of uncertainties, and conformance with relevant industry standards for the PBMR PRA that will be needed to support the proposed risk informed licensing approach for U.S. sited PBMR plants.

4.1.1 Rationale for Use of PRA

Probabilistic Risk Assessment provides a logical and structured method to evaluate the overall safety characteristics of the PBMR plant. This is accomplished by systematically enumerating a sufficiently complete set of accident scenarios and by assessing the frequencies and consequences of the scenarios individually and in the aggregate to predict the overall risk profile. It is the only available safety analysis method that captures the dependencies and interactions among systems, structures, components (SSC), human operators and the internal and external plant hazards that may perturb the operation of the plant that could produce an accident. The quantification of both frequencies and consequences must address uncertainties because it is understood that the calculation of risk is affected by uncertainties associated with the potential occurrence of rare events. These quantifications provide an objective means of comparing the likelihood and consequences of different scenarios and of comparing the assessed level of safety against the TLRC.

PRA is selected for the following objectives:

- Provide a systematic examination of dependencies and interactions and the role that each SSC and operator action plays in the development of each accident scenario; this is referred to in the PRA community as the capability to display the cause and effect relationships between the plant characteristics and the resulting risk levels.
- Provide quantitative estimates of accident frequencies and consequences under the most realistic set of assumptions that can be supported by available evidence.
- Address uncertainties through full quantification of the impact of identifiable sources of uncertainty on the results and by appropriate structured sensitivity studies to understand the risk significance of key issues.
- Apply conservatism only through the examination of explicit percentiles of uncertainty distributions and not by inappropriate combinations of non-physical conservative assumptions.
- Provide a reasonable degree of completeness in treatment of appropriate combinations of failure modes, including multiple failures necessary to determine risk levels

It is important that all key assumptions that are used to develop success criteria, to develop and apply probability and consequence models, and to select elements for incorporation into the models are clearly documented and are scrutable.

4.1.2 Objectives of PBMR PRA

In order to determine the scope and necessary characteristics of the PRA that will be required for the development of licensing bases for the PBMR it is important to list the objectives of the evaluation. The objectives include:

- To confirm that the Top Level Regulatory Criteria, including that the safety goal Quantitative Health Objectives for individual and societal risks are met at a U.S. site or sites
- To support the identification of licensing basis events
- To provide a primary technical basis for the development of regulatory design criteria for the plant
- To support the determination of safety classification and special treatment requirements of systems, structures, and components (SSCs)
- To support the identification of emergency planning specifications including the location of the site boundary
- To support the development of technical specifications
- To provide insight on the available defense in depth in the design

4.1.3 Elements of the PBMR PRA

In the case of LWR PRA, the scope of a PRA is defined in two dimensions, with one dimension used to define the scope of the accident sequence end state and the other for the scope of initiating events and plant initial states to consider. The different treatment of end states is expressed in terms of three PRA Levels. The Level 1 PRA is used to describe the part of the PRA needed to characterize the core damage frequency (CDF); Level 2 is used to describe the aspects of the scenarios involving releases of radioactive material from the containment including the frequencies of different release states and estimates of the source terms for the releases; and Level 3 is used to characterize the aspects of the scenarios involving transport of radioactive material from the site to the ultimate determination of consequences to public safety, health, and the environment so that the frequency of different consequence magnitudes is quantified.

LWR accident initiating events are normally placed into two major categories, one for internal events and the other to capture external events such as seismic events and transportation accidents. (Internal plant flooding events are normally included as part of the internal events scope, but internal plant fires are normally included within the external events scope.) Due to the combination of inherent LWR characteristics and the fact that major changes to thermal hydraulic configuration occur during shutdown, the expansion of scope to include shutdown and low power conditions usually requires a completely different set of initiating events and event sequence models compared with the PRA models for full power initial conditions.

The scope of the PBMR PRA needed to support this risk-informed approach to PBMR licensing will be as comprehensive and sufficiently complete as would be covered in a full scope, all modes, Level 3 PRA covering a full set of LWR internal and external events. However, the inherent features of the PBMR tend to simplify the number of different elements that need to be assembled to accomplish a comparably scoped PRA in relation to an LWR.

The first observation in defining the PBMR PRA elements is that the traditional Level 1- 2- 3 model of an LWR PRA that was originally defined in NUREG/CR-2300 and still used today does not fit the unique characteristics of the PBMR. Since there is no counterpart for the LWR core damage end-state, the splitting up of event sequences involving releases into Level 1 and Level 2 segments does not apply to the PBMR. The elements of the PBMR PRA are integrated around a single, event sequence model framework that starts with initiating events and ends in PBMR specific end states for which radionuclide source terms and offsite consequences are calculated. The integral PBMR PRA encompasses the functions of a full scope Level 1-2- 3 PRA.

Another distinction in the definition of PBMR PRA elements is in the treatment of initial operating states such as full power, low power and shutdown modes. In the LWR case, the early PRA work was focused on the full power-state as intuitively representing the most limiting potential for producing risk significant sequences. In the late 1980's to early 1990's it was realized that accidents initiated during shutdown were even more risk significant until controls were applied to better manage safety functions during plant activities at shutdown. Importantly, PRA for shutdown conditions in LWR were much more complex than for full power as there were many plant configurations to deal with and many different time frames during an outage that created a need to develop separate PRA models for each unique configuration. By contrast, the different configurations of the PBMR do not have so many different applications of the safety functions and therefore lend themselves to a single integrated PRA that accounts for all operating and shutdown states. Furthermore, the on-line refueling aspect and specifications for maintenance on the large rotating machinery (i.e., the turbo units and power turbine generator) mean that the fraction of time the plant is shutdown is expected to be an order of magnitude less than current LWR. Hence for each PBMR PRA element, it is necessary to address applicable sequences in all modes of operation and this can be accomplished without the need for separate models for each mode of operation.

The modular aspect of the PBMR creates the potential for anywhere from one to as many as 10 reactors located at the same site. The PRA needs to account for the risk of multiple modules, which is comparable to the LWR PRA case of a multi-unit site. The existence of multiple modules increases the likelihood of scenarios that impact a single module independently, and creates the potential for scenarios that may dependently involve two or more modules.

The elements of the PBMR PRA, which comprise a full scope treatment of initiating events and end states, include:

1. Initiating Events Analysis
2. Event Sequence Development
3. Success Criteria Development
4. Thermal Hydraulics Analysis
5. Systems Analysis
6. Data Analysis
7. Human Reliability Analysis
8. Internal Flooding Analysis

9. Internal Fire Analysis
10. Seismic Risk Analysis
11. Other External Events Analysis
12. Event Sequence Quantification (includes full uncertainty quantification)
13. Source Term Analysis
14. Consequence Analysis (includes full uncertainty quantification)
15. Risk Integration and Interpretation of Results
16. Peer Review

As emphasized in the current LWR PRA standards, the PBMR PRA must be capable of a thorough treatment of dependent failures including the comprehensive treatment of common cause initiating events, functional dependencies, human dependencies, physical dependencies, and common cause failures impacting redundant and diverse components and systems.

The ASME PRA standard includes both High Level and Supporting Criteria for dependency treatment that arises in essentially all of the above elements. In general, the applicability of the PBMR PRA will be consistent with the ASME PRA standard (Reference 14) for PRA Capability Category III, a full quantification of uncertainties is required that must reflect the iterative nature of the PRA as the PBMR evolves from conceptual design, completion of construction, and eventual commissioning. Quantification of uncertainties provides the capability to determine the mean frequencies and consequences of each accident family to be compared against the TLRC, to compare specific percentiles of the uncertainty distributions against the criteria, and to compute the probability that specific criteria are met.

In order to support the evaluation of regulatory design criteria, the PRA will be capable of evaluating the cause and effect relationships between design characteristics and risk as well as be able to support a structured evaluation of sensitivities to examine the risk impact of adding and removing selected design characteristics.

4.1.4 Applicability of LWR PRA Practices and Standards

The increased use of PRA in the risk-informed regulatory process has led to a number of initiatives to address and improve PRA quality. These initiatives include an industry PRA peer review program (Reference 13) and efforts to develop PRA standards by the ASME (Reference 14), and ANS (References 15, 16, and 17). The concepts and principles that are being developed in these initiatives address both fundamental aspects of PRA technology and certain aspects that are rooted in characteristics of LWR that are not shared by the PBMR. While the fundamental aspects are applicable, the following aspects of these quality initiatives will be modified to apply to a PBMR PRA.

- The current quality initiatives are focused on PRA that are used to calculate CDF and LERF. If one replaces CDF and LERF with the PBMR task of providing estimates of each characteristic PBMR accident family, which is defined by appropriate combinations of PBMR specific initiating events and end-states, then the associated high level and supporting requirements can be viewed as directly applicable to the PBMR.

- As noted in the previous section it is not appropriate to fit a PBMR PRA into the mold of the Level 1 -2 -3 framework. Instead an integrated PRA that develops sequences from initiating events all the way to source terms and consequences is developed.
- As noted in the previous section, it is not necessary to perform a completely different set of PRA models for full power vs. low power and shutdown, such that the PBMR lends itself to an integrated treatment of accident sequences that cover all operating and shutdown modes.
- Unlike the current LWR applications in which it is rarely necessary to extend the PRA to Level 3, the initial PBMR applications will need to include off-site dose consequences to demonstrate the safety case and to meet licensing framework objectives.
- In view of the applications envisioned for the PBMR PRA, a full scope treatment of internal and external events is anticipated.

With these adjustments, it is reasonable to apply the applicable LWR PRA standards and peer review process to assessing PBMR PRA quality until such time as PBMR specific standards and peer review processes are developed. A proposal for application of these standards to each PBMR PRA element is provided in Table 4-1. Note that the ASME standard proposes three Capability Categories to address PRA requirements for different applications. The applications envisioned for the PBMR are assumed in this PRA plan to use ASME PRA Capability Category III. This is a reasonable assumption because of the expectation that the PRA will be integral to the licensing basis of the reactor. These are the standards assumed for defining the scope, level of detail, and capability levels needed to support the risk informed approach to licensing the PBMR.

Table 4-1
COMPARISON OF PBMR PRA TECHNICAL ELEMENTS AND APPLICABLE PRA STANDARDS

PBMR PRA Technical Elements	Applicable PRA Standards	Comments
1. Initiating Events Analysis	• ASME PRA Standard Initiating Events Analysis • ANS shutdown PRA standard for low power and shutdown states	• PBMR and LWR PRA essentially equivalent for this element; separate shutdown PRA not needed for PBMR
2. Accident Sequence Definition	• ASME PRA Standard Accident Sequence Analysis • ANS shutdown PRA standard for accident sequence analysis in low power and shutdown states	• Replace LWR focus on CDF and LERF with focus on major PBMR accident classes; separate shutdown PRA not needed for PBMR
3. Success Criteria Development	• ASME PRA Standard Success Criteria and Supporting Engineering Analysis	• Use of PRA to support licensing basis will make it easier to delineate realistic vs. conservative success criteria relative to LWR
4. Thermal Hydraulics Analysis	• ASME PRA Standard Success Criteria and Supporting Engineering Analysis	• Computer codes to support this developed in Germany and being installed at PBMR; Existing LWR codes are not applicable to PBMR conditions
5. Systems Analysis	• ASME PRA Standard Systems Analysis	• PBMR and LWR PRA essentially equivalent for this element except that PBMR has fewer systems to analyze

COMPARISON OF PBMR PRA TECHNICAL ELEMENTS AND APPLICABLE PRA STANDARDS

PBMR PRA Technical Elements	Applicable PRA Standards	Comments
6. Human Reliability Analysis	• ASME PRA Standard Human Reliability Analysis	• PBMR and LWR PRA essentially equivalent for this element
7. Data Analysis	• ASME PRA Standard Data Analysis	• PBMR and LWR PRA essentially equivalent for this element
8. Internal Flooding Analysis	• ASME PRA Standard Internal Flooding Analysis	• PBMR and LWR PRA essentially equivalent for this element
9. Internal Fires Analysis	• ANS Standard for Internal Fires Analysis	• PBMR and LWR PRA essentially equivalent for this element
10. Seismic Analysis	• ANS PRA Standard External Events Analysis	• PBMR and LWR PRA essentially equivalent for this element
11. Other External Events Analysis	• ANS PRA Standard External Events Analysis	• PBMR and LWR PRA essentially equivalent for this element
12. Accident Sequence Quantification	• ASME PRA Standard Quantification	• LWR separation of accident sequences into Level 1-2-3 not appropriate for PBMR; scope of accident sequences includes doses at the site boundary; risk importance measures to be developed and analyzed for each major PBMR accident class
13. Source Term Analysis	• No corresponding standard	• This task is similar to the T/H and source terms analysis in an LWR Level 2 PRA which is not currently covered in LWR PRA standards
14. Accident Consequence Analysis	• No corresponding standard	• This task is similar to the consequence analysis in an LWR PRA which is not currently covered in LWR PRA standards
15. Risk Integration and Interpretation	• No corresponding standard	• This task is needed to integrate the frequency and consequence information into a frequency-consequence format and to interpret the results compared to TLRC
Not applicable	• ASME PRA Standard Level 2/LERF Analysis	• The treatment of physical and chemical processes that impact source terms are reflected as an integral process into the PBMR accident event trees and fault trees; there is no segregation into Level 1-2-3 as in LWR PRA
16. Peer Review	• ASME PRA Standard for full power internal events, ANS external events and low power and shutdown sections on peer review; NEI guide for industry PRA Certification Peer Review process	• A peer review can be performed for each site specific PBMR PRA that reflects the PRA scope and uses applicable aspects of the NEI PRA Certification Peer Process.

4.2 SELECTION METHOD FOR LICENSING BASIS EVENTS

With a PBMR PRA as outlined above, the selection of LBE proceeds by comparing the risk results with the three frequency-consequence regions defined on the PBMR risk criteria chart of Figure 2-1. This section describes the selection process for each of the three subsets of LBE, namely for the Anticipated Operational Occurrences, the Design Basis Events, and the Emergency Planning Basis Events. The process is utilized as the design detail and technology development proceed so that the PRA certainty advances and a final set of LBE are selected.

The following sub-sections provide noted preliminary PBMR LBE results based on an initial PBMR PRA. At this early stage, the PBMR PRA is limited to at-power events occurring with the reactor operating at full power, and doesn't yet include coverage of external events except for a brief examination of seismic events. The PRA will be updated and expanded as described in section 4.1 to include additional operating states and external events as further details of design, operation, and maintenance are completed. Therefore, the list of LBEs may change as necessary to reflect new findings from the PRA.

4.2.1 Anticipated Operational Occurrences

Anticipated Operational Occurrences (AOO) are selected from those families of events whose mean frequency falls within the AOO region, as shown on the risk criteria chart, and that would exceed the 10CFR50 Appendix I criteria on a mean value basis were it not for design selections that control radionuclide release. Those that meet this condition, or a bounding set of these, are designated AOO.

Families of events may have significant uncertainties in the estimate of their frequencies. The consideration of these uncertainties is necessary to ensure that all events will be assessed against the appropriate criteria. The mean value of frequency, which involves an integral over the complete uncertainty spectrum, is the selected parameter for accounting for frequency uncertainties. An additional factor (2 at the early stage of the design) is placed on the mean frequency to assure that event families falling just above or below a region are evaluated in the most stringent manner.

AOO typically have associated with them relatively small consequences. Furthermore, the uncertainties in the consequences of AOO are relatively small, and are monitored and reduced during the life of the plant. Therefore, although the PRA assessment provides the entire consequence distribution, including the mean, and upper and lower bound doses, it is appropriate that the consequences of AOO meet 10CFR50 Appendix I criteria on a mean-value basis. The mean-value represents a first order consideration of uncertainty. This consideration of uncertainty is consistent with LWR precedent for AOO.

An example of the AOO selection process utilizing comparison of a PRA with TLRC is taken from the initial PBMR PRA results as shown in Figure 4-1. This initial PBMR PRA is for the demo module planned for a South African site. Thus, the effect of multiple modules for the PBMR plant is not yet considered.

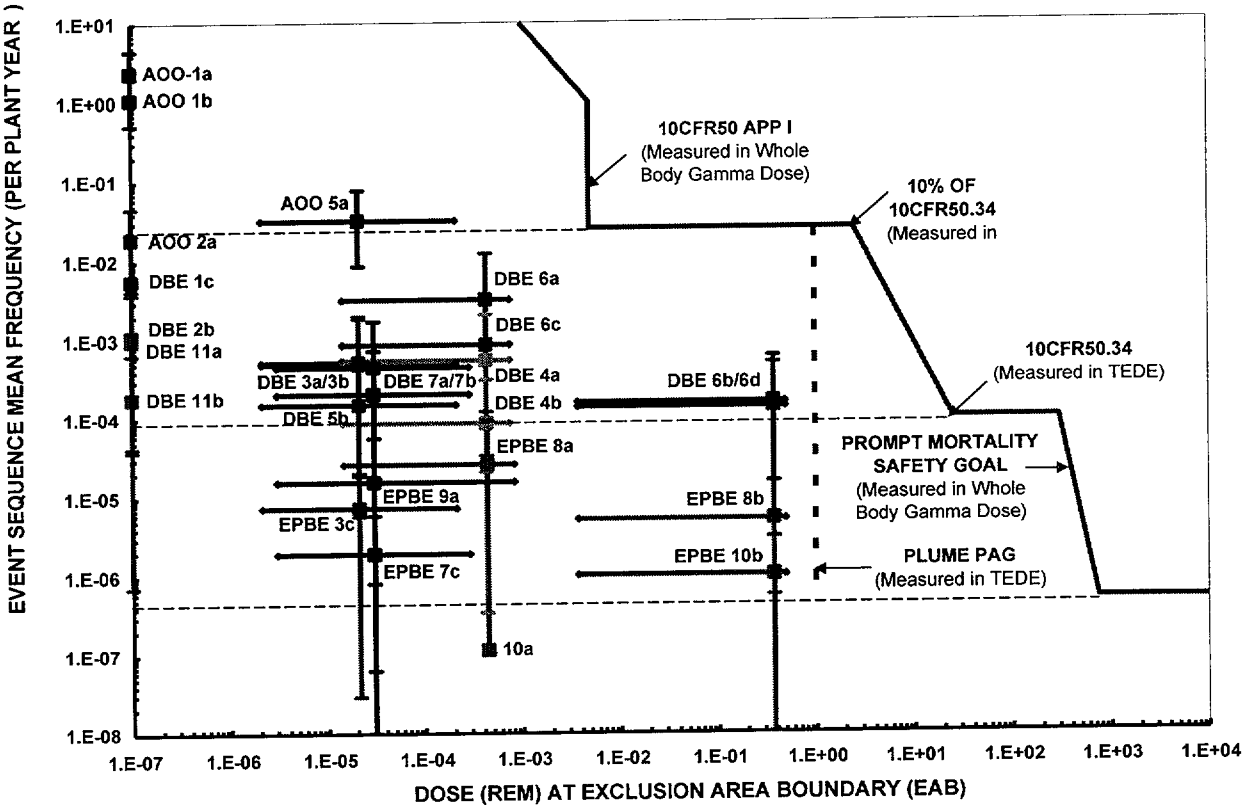

Figure 4-1

INITIAL COMPARISON OF PBMR PRA EVENT FAMILIES WITH THE PBMR TOP LEVEL REGULATORY CRITERIA

Table 4-2 provides the list of the four families of events designated as AOO in the figure. AOO-5a, a small primary coolant leak in one of the PBMR PCU helium-to-water heat exchangers, is the only event with an offsite consequence. Both frequency and consequence uncertainty bands are explicitly shown in the figure.

While the other three AOO do not involve an offsite release, each involves a source sufficiently large that could exceed 10CFR50 Appendix I limits if it were not for a design feature, e.g., protection of the helium pressure boundary.

4.2.2 Design Basis Events

Design Basis Events are selected from those families of events whose mean frequency falls within (or within a factor of) the DBE region as shown on the risk criteria chart and that would exceed the 10CFR50.34 criteria on a mean value basis were it not for design selections that control radionuclide release. Those that meet this condition are designated DBE.

Figure 4-1 again provides the results from the initial PBMR PRA. In this initial PRA for purposes of selecting LBE, the dose consequences have been grouped and bounded. No holdup or retention of the delayed radionuclide release from the fuel has been assumed in the reactor, the helium pressure boundary, or in the reactor building (other than in the HVAC filter). Later revisions will provide more realistic assessments of the radionuclide releases for each of the event families. Thus, as shown in Figure 4-1, there is presently more detail in the frequency assessment than in the consequence analyses.

Table 4-3 provides the list of the fifteen families of events designated as DBE in the figure for the PBMR. These are lower frequency events that oftentimes involve multiple failures, both dependent and independent. By definition, these events would exceed the acceptance criteria if it were not for a selected design feature. Two external events are included as DBE-11a and DBE-11b, which is a 0.27g SSE for the South African demo site. One of the events, DBE-4b has a mean frequency outside the DBE region, but is included because of its large uncertainty.

Eleven of the DBE have offsite doses and the mean and upper and lower bound doses are shown. Within the DBE region both the mean values and upper bound (95% confidence) doses are compared to the criteria. The mean values provide a more consistent comparison of the doses in this region to those in the other two regions, and the upper bounds are used to be consistent with the traditional use of conservative assumptions in performance of design basis accident safety analyses for LWR.

4.2.3 Emergency Planning Basis Events

Emergency Planning Basis Events are selected from those families of events with offsite doses whose mean frequency falls within (or within a factor of) the EPBE region as shown on the risk criteria chart.

Figure 4-1 again provides the results from the initial PBMR PRA. Table 4-4 provides the list of the six families of events designated as EPBE in the figure for the PBMR. These are lower frequency events that involve multiple failures, both dependent and independent.

For all EPBE the mean and upper and lower bound doses are shown. The EPBE and DBE mean doses are compared to the PAG and the EPBE mean doses together with those of the DBE and the AOO are summed over their entire frequency distribution and compared to the safety goal QHO.

Events below the EPBE region are examined to assure that the residual risk is negligible with respect to the latent mortality safety goal and to provide general assurance that there is no "cliff" in which a high consequence event goes unnoticed. The event labeled 10a is a large, unisolated helium pressure boundary break in which the HVAC filter operates. It is not as likely as its counterpart EPBE-10b in which the filter is unavailable. It is an example of an event from the initial PBMR PRA that is checked to assure that the residual risk is low.

Table 4-2
PRELIMINARY IDENTIFICATION OF PBMR ANTICIPATED OPERATIONAL OCCURRENCES

AOO Designation	Anticipated Operational Occurrence
AOO-1a	Loss of Power Conversion Unit with SBS forced cooling
AOO-1b	Loss of Power Conversion Unit with CCS forced cooling
AOO-2a	Control rod group withdrawal with SBS forced cooling
AOO-5a	Heat Exchanger tube break, manually isolated with CCS forced cooling

Table 4-3

PRELIMINARY IDENTIFICATION OF PBMR DESIGN BASIS EVENTS

DBE Designation	Design Basis Event
DBE-1c	Loss of PCU w/ core conduction cooling to RCCS
DBE-2b	Control rod withdrawal w/ CCS forced cooling
DBE-3a	Small, auto isolated HPB break w/ SBS forced cooling
DBE-3b	Small, manually isolated HPB break w/ CCS cooling
DBE-4a	Small, unisolated HPB break w/ pumpdown w/ RCCS cooling
DBE-4b	Small, unisolated HPB break w/o pumpdown w/ RCCS cooling
DBE-5b	HX tube break, manually isolated w/ RCCS cooling
DBE-6a	HX tube break unisolated w/ pumpdown w/ RCCS cooling w/ filtererd release
DBE-6b	HX tube break unisolated w/ pumpdown w/ RCCS cooling w/ unfiltered release
DBE-6c	HX tube break unisolated w/o pumpdown w/ RCCS cooling w/ filtered release
DBE-6d	HX tube break unisolated w/o pumpdown w/ RCCS cooling w/ unfiltered release

DBE-7a	Medium, auto isolated HPB break w/ SBS cooling
DBE-7b	Medium, isolated HPB break w/ CCS cooling
DBE-11a	Safe shutdown earthquake w/ SBS cooling
DBE-11b	Safe shutdown earthquake w/ CCS cooling

Table 4-4

PRELIMINARY IDENTIFICATION OF PBMR EMERGENCY PLANNING BASIS EVENTS

EPBE Designation	Emergency Planning Basis Events
EPBE-3c	Small, isolated HPB break w/ core conduction cooling to RCCS
EPBE-7c	Medium, isolated HPB break w/ RCCS cooling
EPBE-8a	Medium, unisolated HPB break w/ RCCS cooling w/ filtered release
EPBE-8b	Medium, unisolated HPB break w/ RCCS cooling w/ unfiltered release
EPBE-9a	Large, isolated HPB break w/ forced cooling
EPBE-10b	Large, unisolated HPB break w/ RCCS cooling w/ unfiltered release

5 METHOD FOR SELECTION OF EQUIPMENT SAFETY CLASSIFICATION AND DEVELOPMENT OF REGULATORY DESIGN CRITERIA

5.1 REQUIRED SAFETY FUNCTIONS

The selection of the Licensing Basis Events (LBEs) requires that the radionuclide retention functions that keep the events in the Anticipated Operational Occurrence (AOO) and Design Basis Event (DBE) regions are identified from the Probabilistic Risk Assessment (PRA). Even if the event does not have a release, it becomes a basis for regulatory review to show compliance with the associated Top Level Regulatory Criteria (TLRC). Identification of the required safety functions is the first step in equipment classification and the corresponding regulatory design criteria.

The safety functions to meet the TLRC have been identified for the Pebble Bed Modular Reactor (PBMR) as shown in Figure 5-1. The figure includes functions needed for both public and personnel TLRC. As shown, the design included functions for radionuclide retention within the fuel particles, fuel spheres, Helium Pressure Boundary (HPB), reactor building, and site. Not all the functions in Figure 5-1 are required for each TLRC. Safety analyses have been performed to show that the required safety functions to keep the DBEs within the dose limits of 10CFR50.34 are the subset that is shaded.

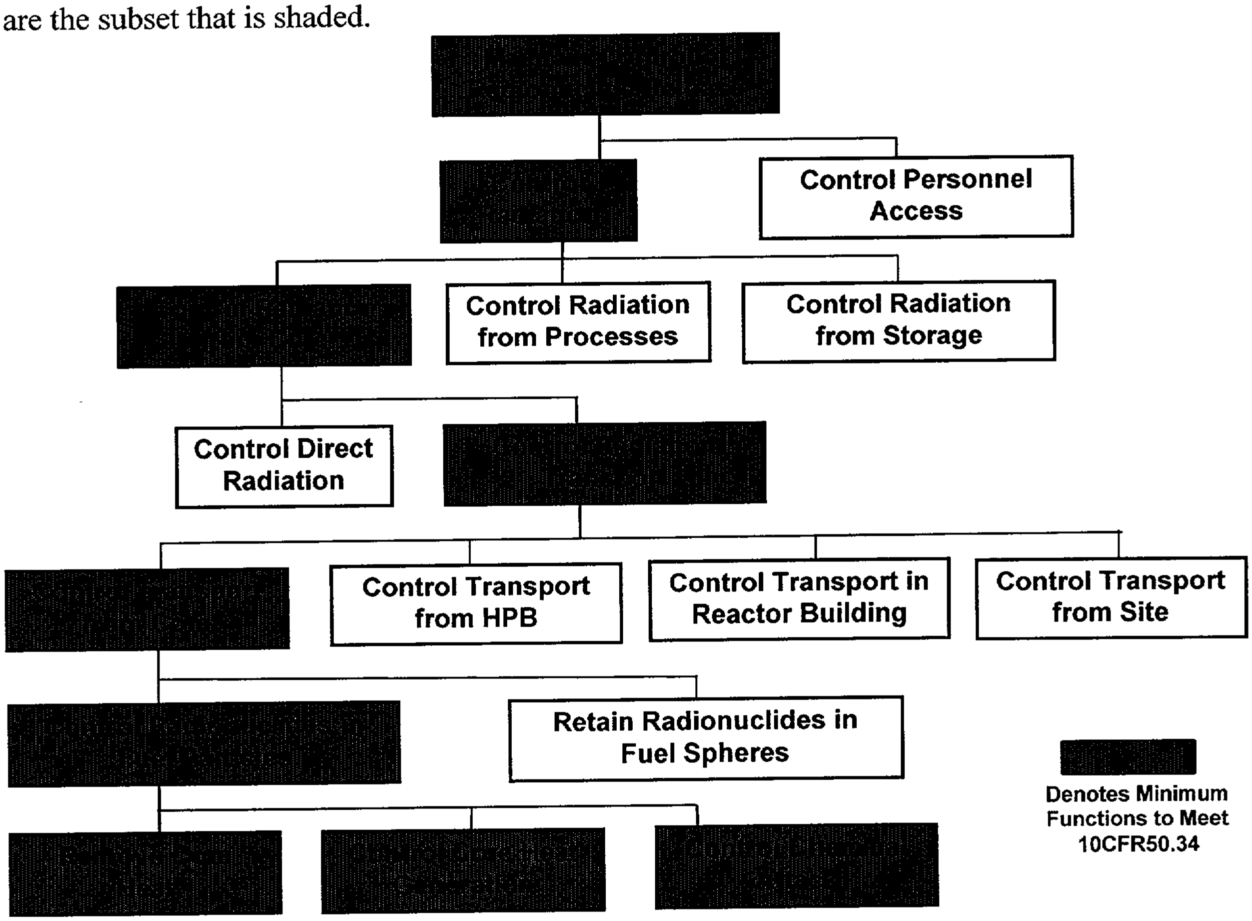

Figure 5-1

5.2 METHOD FOR EQUIPMENT SAFETY CLASSIFICATION

The method for selecting safety-related equipment to be relied on for performing the required safety functions consists of two steps: one to assure that DBE consequences meet 10CFR50.34 doses and one to assure that the frequencies of high consequence EPBEs are kept in the acceptable range.

Consequence Mitigation The first step is to classify one or more structures, systems, or components (SSC) that are available and sufficient to perform the required safety functions to assure that all DBEs meet the DBE dose criteria. On a risk chart, the first step keeps events to the left of the DBE dose criteria line.

The process and results of the first step of the safety classification method for the PBMR are illustrated by examples for two of the required safety functions.

The first required safety function is to *remove core heat.* Table 5-1 presents the evaluation of which sets of SSC are available and sufficient to perform this function during the DBE. As shown, there are a number of sets of SSC that are available and sufficient to remove the core heat to the degree required for radionuclide retention within the fuel. The first three alternatives involve forced convection cooling systems. In each of these an active systems circulates helium from the core to a heat exchanger: the Power Conversion System (PCS) for normal operation power generation, the Startup Blower System (SBS) for startup and shutdown heat removal during outages, and the Core Conditioning System (CCS) for shutdown heat removal during planned and unplanned outages. Each of these three forced cooling systems reject their heat to a secondary water system, the Active Cooling System (ACS), which in turn rejects the heat to the environment. The last three alternatives involve conduction and radiation cooling radially from the fuel to outside the reactor vessel. The Reactor Cavity Cooling System (RCCS) has an active mode for normal operation and unplanned outages and a passive mode for rare off-normal events. The active mode involves the circulation of water to a heat exchanger that transfers the heat to a separate loop of the ACS used for the forced cooling systems. The passive mode boils off the water directly to the environment. The last alternative transfers the heat from the reactor vessel directly in the buildings, ground and surroundings.

The table shows for illustrative purposes the response for the limiting DBEs, those that have the greatest challenge to the core heat removal function. For each DBE column, the question is asked, one-by-one, if the alternative set of SSCs is available and sufficient to remove core heat. . For example, DBE 1c in Table 5-1 is the Loss of the PCU with Core Conduction to the RCCS. Thus, the forced cooling systems involving the PCU, the SBS, and CCS are shown as "No," while the active mode of the RCCS is shown as "Yes." Note, however, that the last two alternatives, while not called upon in this DBE, are available and sufficient, as indicated by "Yes." (These last two alternatives provide defense-in-depth for this particular DBE; the overall PBMR approach to defense-in-depth is discussed in Section 6.2.1.)

Table 5-1

PBMR SELECTION OF SETS OF SAFETY-RELATED SSC FOR CORE HEAT REMOVAL FUNCTION

	SSC Available and Sufficient to Remove Core Heat in the DBE?						
Sets of Alternative SSC	**DBE 1c**	**DBE 2b**	**DBE 6c**	**DBE 7a**	**DBE 7b**	**DBE 11b**	**Safety Related?**
Reactor PCU ACS	**No**	**No**	**No**	**No**	**No**	**No**	
Reactor SBS ACS	**No**	**No**	**No**	Yes	**No**	**No**	
Reactor CCS ACS	**No**	Yes	**No**	Yes	Yes	Yes	
Reactor Reactor vessel Active RCCS ACS	Yes	Yes	Yes	Yes	Yes	Yes	
Reactor Reactor vessel Passive RCCS	Yes	Yes	Yes	Yes	Yes	Yes	Yes
Reactor Reactor vessel Building & ground	Yes	Yes	Yes	Yes	Yes	Yes	

Note: **Bold** indicates actual response during DBE

DBE 1c involves conduction cooling with the helium coolant pressurized within the HPB. Other DBEs, such as DBE 6c, involve conduction cooling with the helium coolant depressurized. DBE 1c involves heat removal with the reactor immediately shutdown; DBE 2b involves heat removal with the initiating event of a group of control rods withdrawn. DBE 1c includes contributors from internal events that cause the loss of the PCU as well as external events such as the loss of the offsite power. DBE 11b is for an external event of the Safe Shutdown Earthquake. Thus, by examining the spectrum of internal and external events, the full set of conditions and requirements applicable to the alternative sets of SSC are explicitly considered.

After filling in the table column-by-column, the results are reviewed. If there is no alternative set of SSC that is sufficient and available for each DBE (i.e., there is not at least one row with all "Yes"), then the design is changed or alternatives are grouped together as required and the resultant alternative set of SSC are classified as safety-related. If there is one alternative set that is available and sufficient for all DBEs, it is classified as safety-related. If there is more than one alternative set of SSC that is determined to be available and sufficient as illustrated in Table 5-1, the set that is selected as the safety-related set is the one that reflects the highest level of confidence that it will perform its required safety function. For example, for core heat removal, the three alternatives sets of SSC involving the forced cooling systems are not available and sufficient for all DBEs, but all three of the alternative sets involving conduction cooling are. The alternative set with the passive mode of the RCCS is selected as safety-related since it is the one

that is the simplest and most reliable, and, therefore, is the alternative that is judged to perform its required safety function with the highest level of confidence.

A second required safety function to *control heat generation* is shown in Table 5-2. As indicated, the reactor fuel's and core's negative temperature coefficient are available in all DBEs. In addition, other control and protection systems are available, depending on the event, to insert the Reactivity Control System (RCS) control rods or small boronated spheres. In most DBEs, the Equipment Protection System (EPS) successfully inserts the control rods. For DBE 2b, the inadvertent withdrawal of a group of control rods, the Reactor Protection System (RPS) will respond by causing the control rods to insert. In all DBEs the operator has time (i.e.,on the order of tens of hours) to manually insert either the control rods or the boronated spheres or to defuel the reactor. And finally, no action could be taken with complete reliance on the negative temperature coefficient, as demonstrated at the German test reactor AVR, to initially shut the reactor down and, in the longer term, after tens of hours of Xenon decay, to maintain the reactor at an acceptable fuel temperature, albeit with the generation of a small amount of power. (Note that these options again reflect defense-in-depth.) Of these alternatives, the negative temperature coefficient and the manual insertion of the control rods is selected as safety-related, since this is the most fundamental and reliable choice for immediate and long term control of heat generation, and, therefore, is the alternative that is judged to perform its required safety function with the highest level of confidence.

Table 5-2

PBMR SELECTION OF SETS OF SAFETY-RELATED SSC FOR CONTROL OF HEAT GENERATION FUNCTION

SSC Available and Sufficient to Control Heat Generation in the DBE?								
Alternative Sets of SSC	**DBE 1c**	**DBE 2b**	**DBE 3a**	**DBE 4a**	**DBE 5a**	**DBE 6a**	**DBE 11a**	**Safety Related?**
Reactor neg temp coeff EPS RCS control rods	Yes	No	Yes	Yes	Yes	Yes	Yes	
Reactor neg temp coeff RPS RCS control rods	No	Yes	No	No	No	No	No	
Reactor neg temp coeff Operator action RCS control rods	Yes	Yes	Yes	Yes	Yes	Yes	Yes	Yes
Reactor neg temp coeff Operator action RCS Boronated spheres	Yes	Yes	Yes	Yes	Yes	Yes	Yes	
Reactor neg temp coeff Operator action defueling	Yes	Yes	Yes	Yes	Yes	Yes	Yes	
Reactor neg temp coeff	Yes	Yes	Yes	Yes	Yes	Yes	Yes	

Note: **Bold** indicates actual response during DBE

This process has been carried out for the other required safety functions. The preliminary list of PBMR safety-related SSC and their corresponding safety functions are shown in Table 5-3.

Table 5-3

PRELIMINARY LISTING OF PBMR SAFETY-RELATED SYSTEMS, STRUCTURES, AND COMPONENTS

Specific SSC Selected as Safety-Related	Required Safety Function Performed by SSC
Fuel	Retain radionuclides Control heat generation (Reactor negative temperature coefficient in Table 5-2)
Fuel and Graphite Spheres	Remove core heat (Reactor in Table 5-1) Control heat generation (Reactor negative temperature coefficient in Table 5-2)
Reactivity Control System (RCS)	Control heat generation
Reactor Core Structure	Remove core heat (Reactor in Table 5-1) Maintain core geometry
Core Barrel	Remove core heat (Reactor in Table 5-1) Maintain core geometry
Core Reflector	Remove core heat (Reactor in Table 5-1) Control heat generation (Reactor negative temperature coefficient in Table 5-2) Maintain core geometry
Reactor Vessel	Remove core heat Maintain core geometry Control chemical attack
Reactor Cavity Cooling System (RCCS)	Remove core heat
Reactor Building	Maintain core geometry Control chemical attack

High Consequence Prevention The second step is to classify one or more sets of SSC that are available and sufficient to perform the required safety functions to assure that all EPBEs with doses greater than the dose limits in 10CFR50.34 remain below the DBE region lower frequency. This step has the effect on a risk chart of preventing high consequence events, that is, those with consequences greater than the dose limits of 10CFR50.34, from moving up into the DBE region.

Since there are no EPBEs with doses greater than the dose limits of 10CFR50.34, the second step of the process has not led to any additional sets of SSC being classified as safety-related. Nevertheless, this step will continue to be applied as the design progresses to provide assurance that the list of safety-related SSC is complete.

This section has summarized and illustrated with PBMR examples the proposed method for assigning safety classification to sets of SSC. The objective was the same as in conventional practice: to select a subset of equipment for safety classification that is available and sufficient during DBEs. Since this approach is based on the DBEs selected with risk assessment as discussed in Section 4, there is added assurance that this objective is met. Thus, the method does not examine only initiating events and assume an active or passive failure. The DBEs selected with PRA may be initiated by internal or external events and involve single or multiple failures and/or common cause failures of active or passive components. The approach has the additional advantage that there is an explicit link between the classified SSCs, not only with the DBEs, but with the required safety function(s) for each. This provides an opportunity, as discussed in the next sections, to focus the SSC requirements and special treatment to assure with a high level of confidence that these sets of SSC perform their intended functions.

5.3 DEVELOPMENT OF REGULATORY DESIGN CRITERIA

Regulatory Design Criteria (RDC) are statements written at a functional level to describe the requirements for SSC needed during DBE to assure compliance with 10CFR50.34. The RDC are similar in nature and purpose to the GDC in Appendix A to Part 50, and will address PBMR safety functions that are not addressed in the GDC. The RDC have a one-to-one correspondence to the required safety functions.

Examples of RDC from the MHTGR licensing process are provided in Reference 18 (designated there as 10CFR100 design criteria). The two that correspond to the two examples in the previous section are:

> **Remove Core Heat:** The intrinsic dimensions and power densities of the reactor core, internals, and vessel, and the passive cooling pathways from the core to the environment shall be designed, fabricated, and operated such that the fuel temperatures will not exceed acceptable values.
>
> **Control Heat Generation:** The reactor shall be designed, fabricated, and operated such that the inherent nuclear feedback characteristics ensure that the reactor thermal power will not exceed acceptable values. Additionally, the reactivity control system(s) shall be designed, fabricated, and operated such that during insertion of reactivity the reactor thermal power will not exceed acceptable values.

Regulatory design criteria were also written for lower level functions providing more specificity. For example, Reference 18 includes design criteria for sub-functions to the function remove core heat: conduct heat from core to vessel wall, radiate heat from vessel wall, maintain geometry for conduction and radiation, and transfer heat to ultimate heat sink.

5.4 REQUIREMENTS FOR SAFETY-RELATED EQUIPMENT

The RDC are qualitative, functional statements for the SSC classified as safety-related. Quantitative requirements are developed by requiring that the safety-related SSC by themselves be sufficient for each of the DBE to meet the DBE dose criteria. Reevaluating the DBE non-mechanistically with only the safety-related SSC available leads to the Safety-Related Design Conditions (SRDC). The SRDC are used to develop the temperatures, stresses, heat loads, etc. that the SSC must meet for each of the DBE. The design, fabrication, and operational requirements for the safety-related SSC are directly linked to the DBE on a case-by-case basis.

An example of the process from the MHTGR is considered for the reactor pressure vessel. The reactor pressure vessel was classified as safety-related based on consideration of the following required safety functions:

1) radiate core heat from vessel wall,
2) maintain core geometry, and
3) limit air ingress to core.

With regard to the function to maintain core geometry, the reactor vessel must maintain its strength during off-normal events including conduction cooldown events (loss of forced cooling

from both the main heat transport and shutdown cooling systems). These include a pressurized conduction cooldown (DBE 1 and 4 in Table 4-3) and a depressurized conduction cooldown (DBE 10 and 11in Table 4-3). An ASME Code Case was submitted and approved to allow elevated temperature operation for the stainless steel 508/533 material (up to 1000 hours over 700F but not to exceed 1000F). Additionally, the other DBE lead to requirements for this function that include seismic loads for the .3g SSE (DBE 5 in Table 4-1) and for vessel pressure relief capability for steam generator leaks into the primary system (DBE 6-9 in Table 4-3).

This process will be followed for the PBMR, but, at this level, the requirements even for similar components to the MHTGR may vary.

6 CONSISTENCY WITH CURRENT REGULATORY PRACTICE

The purpose of this section is to describe how the PBMR licensing approach is consistent with current regulatory practice. This includes the degree of consistency with NRC policies on the licensing of advanced reactors, the use of risk-informed approaches to regulate nuclear reactor safety, and with regulatory practice for special treatment of safety-related equipment.

The Advanced Reactor Policy Statement (Reference 1) sets forth NRC expectations for design features that qualify for consideration as an "advanced reactor" and solicits early dialogue on innovative approaches that may be appropriate for reactors that possess one or more of these advanced reactor attributes. A summary of how the PBMR aligns with the policy statement on advanced reactors is provided in Section 6.1.

The principles of risk-informed regulation that are being used for selected applications with LWR are discussed in Regulatory Guide 1.174 (Reference 20). A discussion of how each of these principles is addressed in the design and licensing of the PBMR is described in Section 6.2.

Section 6.3 addresses the PBMR licensing approach's consistency with regulatory practice on the special treatment of safety-related equipment.

6.1 NRC ADVANCED REACTOR POLICY

A primary goal of the PBMR licensing approach is to be responsive to the Advanced Reactor Policy, namely to foster early dialogue and agreement on the approach in a manner that enhances the stability and predictability of the process. The Advanced Reactor Policy statement sets forth NRC expectations for characteristics in order to be regarded as an advance reactor. The policy statement acknowledges that the NRC regards an earlier version of a gas cooled reactor concept as qualifying for the advanced reactor designation. There are additional enhancements and innovations in the PBMR that provide additional reasons to support the PBMR designation as an advanced reactor concept.

A summary of the advanced reactor characteristics identified in the Advanced Reactor Policy Statement and how they are addressed in the preliminary PBMR design is provided in Table 6-1. There are significant PBMR preliminary design features to address each characteristic identified in the NRC Advanced Reactor Policy.

In addition to the characteristics identified in Table 6-1, the Advanced Reactor Policy concludes that advanced reactors should have enhanced margins of safety and meet or exceed the safety of existing reactors. The PBMR will meet this policy. As evidenced in Section 2 on the Top Level Regulatory Criteria and in Section 4 in the spectrum of events to be evaluated, the TLRC will ensure that the PBMR meets existing requirements with margin including the NRC Safety Goals. Additionally, as discussed in Section 6.2.3, the PBMR will have enhanced safety margins.

Table 6-1

COMPARISON OF PBMR PRELIMINARY DESIGN FEATURES AND NRC DEFINITION OF ADVANCED REACTOR CHARACTERISTICS

NRC's Definition of Advanced Reactor Characteristics	Corresponding PBMR Preliminary Design Features
Highly reliable and less complex shutdown and decay heat removal systems; The use of inherent or passive means to accomplish this objective....(negative temperature coefficient, natural circulation)	• Low excess reactivity and negative temperature coefficient provide passive shutdown capability • Two diverse active systems provided to insert negative reactivity to assure long term sub-criticality • Redundant, diverse and independent active forced cooling systems to remove core decay heat • Conduction/radiation cool-down capability without forced or natural convection of the primary coolant • No requirement for maintaining an inventory of primary coolant inside the reactor vessel.
Longer time constants and sufficient instrumentation to allow for more diagnosis and management prior to reaching safety systems challenge and/or exposure of vital equipment to adverse conditions.	• Low power density and large heat capacity of core fuel and graphite provides long time constants for power/temperature transients over full range of accident conditions • Low stored energy and single phase of primary coolant prevents rapid thermal and mechanical energy transfer to primary boundary and to containment structures; eliminates fuel coolant interactions that could challenge barrier integrity. • Capability to monitor circulating primary system radioactivity to confirm integrity of the fuel is within design limits
Simplified safety systems which, where possible, reduce required operator actions, equipment subjected to severe environmental conditions, and components needed for maintaining safe shutdown conditions.	• Capability to limit consequences of event sequences independent of any prompt operator actions; and reliant on passive safety features. • Safety systems are few, simple, and have few components needed to operate
Designs that minimize the potential for severe accidents and their consequences by providing sufficient inherent safety, reliability, redundancy, diversity and independence in safety systems	• The inherent capabilities of the fuel particles to retain their structural integrity over the range of normal and event sequence conditions with margins limit the source terms to very small levels; operation of active systems not required to support this capability • Long time constants of any releases and absence of any adverse physical and chemical processes • Any sequence with the primary system boundary intact results in no release of radioactivity • Design features that limit the potential for air or water ingress.
Designs that provide reliable equipment in the balance of plant, (or safety system independence from balance of plant) to reduce the number of challenges to safety systems	• The entire plant is very simple with a small number of components and support systems;

Table 6-1

COMPARISON OF PBMR PRELIMINARY DESIGN FEATURES AND NRC DEFINITION OF ADVANCED REACTOR CHARACTERISTICS

NRC's Definition of Advanced Reactor Characteristics	Corresponding PBMR Preliminary Design Features
Designs that provide easily maintainable equipment and components	• Fuel elements are continuously monitored via on-line refueling and monitoring of circulating activity; broken and spent fuel elements replaced • Power conversion equipment (turbo-generator, turbo-units, etc.) can be maintained without compromising ability to support key safety functions
Designs that reduce the potential radiation exposures to plant personnel	• Performance of the fuel greatly reduces level of circulating primary coolant activity • Inert helium provides no impurities for activation products
Designs that incorporate defense-in-depth philosophy by maintaining multiple barriers against radiation release and by reducing potential for consequences of severe accidents	• Fuel particles, fuel spheres, primary pressure boundary, citadel structure, containment envelope serve as concentric, independent barriers (See more detailed discussion in Section 6.2.1) • Design features provide accident prevention and mitigation (See more detailed discussion in Section 6.2.2)
Design features that can be proven by citation of existing technology or which can be satisfactorily established by commitment to suitable technology development program	• Innovation of earlier designs: extensive experience with gas cooled reactors, HTGRs, and significant experience with pebble bed reactors to provide confidence in performance of fuel and major components. • New and unique PBMR features important for power production but not needed to support key safety functions • experimental evidence to support confidence in the integrity of the fuel under normal and adverse conditions • Formula for proven fuel manufacturing process and quality assurance testing that ensure manufacturing reliability • Plan to feedback operating experience from early PBMR to refine technology

6.2 IMPLEMENTATION OF RISK-INFORMED REGULATION PRINCIPLES

Regulatory Guide 1.174 provides the foundation for NRC LWR risk-informed activities and provides guidance for the Option 1 activities. In Reg. Guide 1.174, the NRC outlined five principles of risk-informed regulation for changes to existing facilities:

1. The proposed change meets the current regulations unless it is explicitly related to a requested exemption or rule change, i.e., a "specific exemption" under 10 CFR 50.12 or a "petition for rulemaking" under 10 CFR 2.802.
2. The proposed change is consistent with the defense-in-depth philosophy.
3. The proposed change maintains sufficient safety margins.
4. When proposed changes result in an increase in core damage frequency or risk, the increases should be small and consistent with the intent of the Commission's Safety Goal Policy Statement
5. The impact of the proposed change should be monitored using performance measurement strategies.

While the above principles are expressed to evaluate changes to an existing licensing basis, they provide a useful way to evaluate the risk-informed aspects of the PBMR licensing approach. The licensing approach that is proposed for the PBMR addresses each of these risk-informed principles. Item 1 is being addressed by examining each of the current regulations and determining their applicability to the PBMR. An analysis of how the PBMR design employs the defense-in-depth philosophy by use of inherent and active and passive engineered safety features is discussed in Section 6.2.1. An evaluation of prevention and mitigation in achieving defense-in-depth is discussed in Section 6.2.2. The incorporation of safety margins into the design specification is discussed in Sections 6.2.3. As noted in the process for selecting Licensing Basis Events in Section 4 and their derivation from Top Level Regulatory Criteria in Section 2, the top down licensing process proposed for the PBMR is rooted in basic requirement that the NRC Safety Goal Policy will be met. Since RG 1.174 was written for changes to existing LWR, the fifth principle does not literally apply, however, key plant parameters will be monitored during operation as discussed in Section 6.2.4.

6.2.1 Defense-in-Depth

The risk-informed defense-in-depth framework described by the Staff in SECY 00-198 (see Figure 2-1 in Reference 21) is comprised of the goal of protecting public health and safety; reactor safety cornerstones expressed in terms of initiating events, mitigation systems, barrier integrity, and emergency planning; and strategies and tactics to assure reactor safety. The strategies include accident prevention and mitigation, whereas the tactics cover the areas of design, construction and operation using the principles of safety margins, redundancy, diversity, independence, general design criteria and special treatment requirements.

Regulatory Guide 1.174 offers several considerations for ensuring that defense-in-depth is maintained in risk-informed changes proposed to the current LWR licensing basis, including those to ensure:

- A balance between accident prevention and mitigation,
- No over-reliance on programmatic activities to compensate for weaknesses in plant design,
- System redundancy, independence, and diversity are employed,
- Potential common cause failures are minimize through the use of passive, and diverse active systems to support key safety functions,
- Barriers to radionuclide release are independent, and
- The potential for human errors is minimized.

A discussion of how the PBMR will address prevention and mitigation is provided in the next section that includes an illustration for the MHTGR of the degree of independence for each radionuclide barrier during several DBE. With regard to the other items, it is expected that the process for selecting LBEs and developing the associated regulatory design criteria will lead to design decisions to employ an appropriate level of system redundancy, independence, diversity, and appropriate defenses against common cause failures and human errors. For example, the preliminary design of the PBMR includes two diverse shutdown systems and several diverse decay heat removal systems. Hence applying the above considerations for ensuring defense-in-depth is consistent with the proposed licensing approach and anticipated PBMR design requirements.

One physical connotation of defense-in-depth is to provide multiple independent barriers to the transport of radionuclides to the environment. To apply the defense in depth concepts to the PBMR it is helpful to outline the key safety functions that protect the barriers to radionuclide transport. The barriers to radionuclide transport in the PBMR include:

- The fuel including the coated fuel particles and the pebble bed spherical fuel elements
- The primary pressure boundary (PPB) which comprises the reactor vessel and connecting vessels and piping that contain the helium coolant and interfacing helium inventory control and purification systems.
- The containment and including the citadel, which provides a structural barrier, and the confinement boundary and HVAC systems, which control and filter any releases from the PPB.

Using the above preliminary design barriers and the key safety functions that support them, a representative description of the design features that support the defense-in-depth concept in the PBMR is presented in Table 6-2. Specifics of this representative table may change as the design progresses.

The inherent features in this table, as well as the engineered passive and active features are intended to achieve a high degree of independence among each of the above barriers. Dependent interactions between these barriers are intentionally minimized in the design selections. In addition to providing prevention and mitigation, these features strive to strike a balance between inherent and engineered passive features that exhibit robust safety margins when challenged, and highly reliable active features that reduce the likelihood of these challenges. A model for examining these important components of defense-in-depth is presented in the next section.

Table 6-2 is a composite of the features providing defense-in-depth organized by the required safety features discussed in Section 4. However, as discussed in Section 5, the safety classification approach examines the spectrum of DBE to select a set of SSC that is available and sufficient during each event. Therefore, the physical concept of defense-in-depth varies by event sequence family. In addition, the process concept of defense-in-depth is employed for the SSC selected as safety-related. Namely, the design, procurement, fabrication, construction, operation, monitoring, in-service-inspection and testing are each defined to provide a high level of assurance that the functional requirements are met for each DBE

Table 6-2

REPRESENTATIVE PBMR PRELIMINARY DESIGN FEATURES SUPPORTING DEFENSE-IN-DEPTH CONCEPT

Barrier to Radio-nuclide Release	Safety Functions Supporting Barrier Integrity	Elements of Defense-in-Depth in Supporting Safety Functions	
		Inherent features and attributes	**Engineered active and passive features**
Retain Radio-nuclides in Fuel	Maintain fuel integrity	• Multiple layer ceramic TRISO fuel particles encapsulated by graphite sphere • Low levels of circulating and plateout activity within pressure boundary via fuel reliability and performance	• Robust fuel quality assurance program to assure high fuel particle reliability • On-line fuel measurements to monitor burn-up • On-line monitoring of circulating activity • On-line sphere physical defect monitoring
	Control heat generation	• Negative temperature coefficient • Small excess reactivity • Slow reactivity response	• Gravity-driven control rods via EPS and RPS • Diverse, gravity-driven boron pellets via RSS • High shutdown margins achieved by either system
	Control heat removal	• Low power density • High thermal heat capacity • Core geometry and power level allows passive cool-down capability independent of coolant convection	• Forced cooling via MPS (Brayton cycle) operation • Forced cooling via SBS and normal MPS heat sinks • Forced cooling via Reactor Unit Conditioning System • Passive conduction/radiation cool-down via the active/passive RCCS • Redundant and diverse heat removal paths • Several day passive capability independent of active components or operator actions
	Control chemical attack	• Ceramic fuel particles and graphite matrix • Design limitations on extent of air or water ingress • Self limiting aspect of reactions • Inert gas coolant	• Low pressure cooling water sources limits potential for water ingress • Design interfaces with high pressure primary system • High quality primary vessels & piping with robust seismic capability
	Maintain core geometry	• Use of refractory ceramics for structural materials exposed to high temperatures • Use of high temperature alloys for reactor vessel and core barrel components	• High quality reactor vessel and PPB designed to large seismic margins • Robust citadel structure provides strength and protection from external missiles • Forced convection of Helium maintains vessel and core barrel metal components at relatively low temperatures

REPRESENTATIVE PBMR PRELIMINARY DESIGN FEATURES SUPPORTING DEFENSE-IN-DEPTH CONCEPT

<table>
<tr><th rowspan="2">Barrier to Radio-nuclide Release</th><th rowspan="2">Safety Functions Supporting Barrier Integrity</th><th colspan="2">Elements of Defense-in-Depth in Supporting Safety Functions</th></tr>
<tr><th>Inherent features and attributes</th><th>Engineered active and passive features</th></tr>
<tr><td>Retain Radionuclides in Primary Pressure Boundary</td><td>Maintain PPB integrity</td><td>• Use of chemically inert single phase coolant
• Inherently low over-pressurization potential due to low stored energy, large pressure gradients during normal operation, and limited pressurization capacity of HICS
• Immediate reduction in PPB pressure on cessation of Brayton Cycle</td><td>• Maintenance of high chemical purity by operation of helium purification system
• Use of high quality reactor vessel and PPB components
• PPB capability to retain PTG missiles
• Use of a Citadel structure to protect the PPB
• Forced convection flow paths maintain PPB components at relatively low temperatures.
• Capability of HICS to pump down inventory to reduce driving head for releases from PPB.</td></tr>
<tr><td>Retain Radio-nuclides in Containment</td><td>Maintain integrity of containment</td><td>• Low stored energy and inert primary coolant
• Completely envelopes PPB boundary
• Events evolve slowly allowing for manual compensating measures</td><td>• HVAC filtration system to reduce exposures
• Blowout panels for large depressurization events prevent pressure loads
• Re-closable vent for an elevated release
• Robust construction to protect equipment from external hazards
• Partially below grade</td></tr>
<tr><td>Provide Emergency Planning</td><td>Provide ample warning time for effective protective actions</td><td>• Large thermal capacity provides ample time for implementing emergency plans
• Low source terms reduce EP contingencies</td><td>• Features relied upon in conservative EP strategy are solely sufficient to limit radiological doses at the site boundary below Protective Action Guidelines levels for all LBEs.</td></tr>
</table>

Legend:

MPS	Main Power System	RSS	Reserve Shutdown System
PPB	Primary Pressure Boundary	PTG	Power Turbine Generator
EPS	Equipment Protection System	EP	Emergency Planning
RPS	Reactor Protection System	SBS	Startup Blower System
HICS	Helium Inventory Control System		

6.2.2 Prevention and Mitigation

The risk-informed defense-in-depth framework in SECY 00-198 describes the roles of accident prevention and mitigation to ensure defense-in-depth. To address the degree of prevention and mitigation for the PBMR, it is necessary to generalize the concept of prevention and mitigation so its application to the PBMR is not obscured by its fundamental differences with LWR.

Consistent with the notions of defense-in-depth described in various NRC documents (e.g., NUREG 1150 (Reference 22)) is the insight that prevention and mitigation be discussed in the context of an event sequence or family of event sequences with similar characteristics. An event sequence is examined in terms of the following generic elements

1. An initiating event that constitutes a challenge to the plant systems responsible for control of transients and protection of the plant SSCs including the radionuclide transport barriers.
2. The response (successes and failures) of plant active systems that support key safety functions responsible for protection of barriers, retention of radioactive material, and protection of the public health and safety, as defined by the event sequence.
3. The response of passive design features responsible for supporting key safety functions.
4. The response of each barrier to radionuclide transport to the environment; these barriers typically include the fuel elements, the primary pressure boundary (PPB), and the containment or containment structure.
5. The implementation of emergency plan protective actions to mitigate the radiological consequences of a given release from the plant.

The development of a generic model for discussing event sequence prevention and mitigation makes use of two key PRA insights:

1. A given design feature exhibits varying degrees of importance on different event sequences. Hence it is necessary to examine a spectrum of sequences to understand the safety significance of a particular feature.
2. A design feature may be postulated to fail along one sequence, but operate successfully on another so it may prevent an accident in some cases and mitigate an accident in others. Hence the extent to which risk is managed by prevention or mitigation varies across the event sequence spectrum.

An example of the analysis of prevention and mitigation aspects for a given event sequence in the MHTGR is illustrated in Table 6-3 for DBE 10, a moderate primary coolant leak with forced cooling (References 9 and 23). This same type of sequence is expected to be a candidate for selection as an LBE for the PBMR. The description of the sequence in this table has been constructed to identify the key design features responsible for each of the prevention and mitigation elements of the generalized model.

To provide a quantitative assessment of the preventive and mitigative aspects of design features along this sequence, components of the event sequence frequency and the role of each barrier in the retention of one important radionuclide, Iodine-131, are identified. The values of the initiating event frequencies, failure probabilities, and release fractions are each proportional to the risk of release of I-131.

DBE 10 is initiated by a moderate size failure in the primary pressure boundary up to size of about 13 in^2, which results in a depressurization of the primary system into the containment. For this sequence, there is successful insertion of the control rods, and continued forced circulation cooling of the core using the Heat Transport System (HTS) circulators and heat removal paths. The releases into the containment include a large fraction of the initial circulating activity and a small fraction of the initial plateout activity from shear forces during the rapid depressurization. These source terms are significantly influenced by the reliability of the fuel particles as manufactured and the performance of the fuel during normal operation.

The design feature identified as contributing to prevention for this sequence include:
- the reliability of the PPB pressure boundary itself which helps to reduce the initiating event frequency to its indicated value.

The factors that contribute to mitigation include:
- the successful response of the reactor trip system and the forced cooling systems that prevent any increases in reactor temperatures relative to normal operating temperatures during the event sequence transient,
- the performance of the fuel during normal operation which limits the circulating activity and plateout activity available for release and its continued performance during the sequence in which the forced cooling system prevent any temperature increases,
- the performance of the PPB as a radionuclide barrier which retains most of the plateout activity and some of the circulating activity, and
- the performance of the containment which retains part of the source term released from the PPB during the event sequence.

Additional perspective can be gained from examining the sequence in which additional systems fail, as exemplified by DBE 11, a small primary coolant leak without forced cooling. DBE 11 is initiated by a small leak in the primary pressure boundary (PPB) up to 1in^2 in size and involves failure of both MHTGR systems normally available to provide forced circulation of helium, successful insertion of the control rods, and successful conduction cooldown of the core using the passive Reactor Cavity Cooling System (RCCS). The RCCS is successful in cooling the core in a very slowly evolving temperature transient in which the spatial temperature profile shifts as core heat is removed by conduction and radiation from the reactor vessel to the passive RCCS located in the containment envelope. As parts of the core temporarily increase in temperature before cooling down there is a contribution to the source term that is released from part of the inventory associated with initially failed fuel particles and from external fuel particle contamination, in addition to the circulating activity. Lift-off of plateout is negligible, as the depressurization rates are slow for this size of PPB leak.

As shown in Table 6-4 factors identified as contributing to prevention for this sequence include:
- the reliability of the PPB boundary and
- the reliability of the forced cooling systems that are postulated to fail along this sequence.

The key factors that are identified as mitigation features along this sequence include:
- the successful operation of the reactor trip system,
- successful pump-down of the helium inventory which reduces the driving head for transport of radionuclides, and
- successful operation of the passive RCCS to effect cool-down of the core.

Upon comparing the roles of common design features for these two event sequences, it is seen that the forced cooling systems provide a mitigation role in DBE 10 and a prevention role in DBE 11. As with DBE-10, the radionuclide barriers of the fuel, PPB, and containment each provide important mitigation roles.

The above discussion of prevention and mitigation considered the integrated response of the entire plant as assessed in a PRA on the MHTGR. It is expected that application of this approach to the PBMR will exhibit some similarities and differences with respect to these two examples due to similarities and differences between the PBMR and the MHTGR

Table 6-3

PREVENTION/MITIGATION ANALYSIS OF MHTGR DBE-10

Standard Form of Event Sequence	Prevention Aspects	Mitigation Aspects
Initiating event	• Moderate PPB failure (1 in^2 to 13 in^2 leak assessed to have a frequency of about $8x10^{-3}$ per year)	
Response of active systems supporting key safety functions; combinations of successes and failures of specific safety functions that are appropriate for a specific event sequence.	• No system failures for this sequence	• Successful forced cooling via the HTS with reliability of .83 • Successful insertion of control rods via reactor trip system • Pump-down of Helium inventory ineffective due to large size of PPB failure and rate of depressurization into containment
Response of passive features supporting key safety functions; combinations of successes and failures of passive features supporting specific safety functions that are appropriate for a specific event sequence.	• No failures of passive features along this sequence except for the initiating event and the fuel particle performance during normal operation which results in a relatively small circulating and plateout radioactivity source term	• Forced cooldown transient resulting in cooldown of all parts of the core from normal operation • Initially intact fuel particles remain intact • Initial circulating and plateout primary coolant activity is very low due to fuel performance during normal operation • Small fraction of initially plated out radionuclides lifted off due to shear forces from depressurization
Fraction of source term released from fuel into primary reactor coolant system PPB		• $< 2x10^{-6}$ of I-131 inventory available for release from circulating and plateout activity due to fuel performance during normal operation .
Fraction of source term released from PPB into containment		• About $1x10^{-3}$ of the I-131 in the plateout and all of circulating activity escapes the PPB into the containment
Fraction of source term released from containment		• About 1/3 of the I-131 released into the containment from the PPB is released from the plant
Time available to implement emergency plan protective actions.		• Less than .01 Ci of I-131 is released from the plant mostly during the depressurization event

Table 6-4

PREVENTION/MITIGATION ANALYSIS OF MHTGR DBE-11

Standard Form of Event Sequence	Prevention Aspects	Mitigation Aspects
Initiating event	• Small PPB failure (.03 to 1 in^2 leak assessed to have a frequency of about $3x10^{-2}$ per year)	
Response of active systems supporting key safety functions; combinations of successes and failures of specific safety functions that are appropriate for a specific event sequence.	• Reliability of the assumed failed HTS of .83 reduces the event sequence frequency by factor of 0.17. • Reliability of the assumed failed SCS of .97 reduces the event sequence frequency by a factor of 0.03	• Successful insertion of control rods via reactor trip system • Successful pump-down of Helium inventory to reduce radionuclide transport potential
Response of passive features supporting key safety functions; combinations of successes and failures of passive features supporting specific safety functions that are appropriate for a specific event sequence.	• No failures of passive features along this sequence except for the initiating event and the initially failed fuel particles	• Successful operation of RCCS to remove heat conducted and radiated from core and reactor vessel • Conduction cooldown transient resulting in elevated core temperatures • Initially intact fuel particles remain intact • Initial circulating primary coolant activity is very low due to fuel performance during normal operation
Fraction of source term released from fuel into primary reactor coolant system contained by PPB		• $< 2x10^{-5}$ of I-131 inventory released from fuel over a period of several days; release limited to part of the inventory of failed and contaminated fuel particles and part of circulating primary coolant activity.
Fraction of source term released from PPB into containment		• About ½ of the I-131 released from the fuel escapes the PPB into the containment
Fraction of source term released from containment		• About 4% of the I-131 released into the containment is released from the plant.
Time available to implement emergency plan protective actions.		• Less than 3 Ci of I-131 is released from the plant over a period of 50 to 150 hours after the initiating event.

6.2.3 Safety Margins

The need to maintain adequate safety margins is one of the principles of risk-informed regulations set forth in Regulatory Guide 1.174. In general, this term connotes the relative conservatism employed in the design, the selection of design requirements, and the design evaluation process to achieve a level of confidence that the TLRC are met in light of uncertainties in performance of SSCs and in plant behavior during accident conditions. There are several applications of safety margins in the PBMR design and several opportunities to evaluate adequacy of these margins in the proposed licensing approach.

The PBMR incorporates safety margins in the design of the core and fuel, selection of core power and geometry, system design and selection of operating conditions to ensure that over a spectrum of operating conditions including low frequency event sequence conditions, a set of stringent regulatory design requirements are met. One of these thresholds is that the maximum fuel temperatures will not approach unacceptable (nominally1600°C) over the full range of LBEs. The selection of this temperature limit is well below the levels that would challenge the fuel barrier integrity. The concept of safety margins is applied in the selection of this limit and in decisions to apply it to the full spectrum of LBEs including those whose frequency is well below the design basis region. Even though the plant is capable of achieving safe shutdown independent of any active forced circulation of helium, there are three independent and diverse systems for heat removal via forced circulation of helium to gas to water heat exchangers. Hence the existence of these forced cooling systems in the design provides margins by reducing the frequency that the conduction cooldown capability will be needed.

In addition, the concept of safety margins is applied in the safety analyses to support the licensing basis. In these analyses conservative estimates are used for the reliability of the primary barrier that is achieved in the fuel particle manufacturing process. In addition, the proposed PBMR licensing approach utilizes a comprehensive risk assessment, which includes quantitative consideration of uncertainties to demonstrate that the TLRC are met. This allows the examination of contributors to uncertainty and a quantitative approach to the setting of specific safety margin in the development of regulatory design criteria. In addition, the uncertainties are accounted for in the identification of licensing basis events (LBE) and safety significant structures, systems and components. Hence both the design and the safety analysis framework apply the principle of safety margins to assure that safety requirements are met. Thus, the proposed PBMR licensing approach is consistent with the risk-informed principle of safety margins.

6.2.4 Monitoring Performance of SSCs

The need to monitor SSCs to identify unexpected developments in performance and to ensure that regulatory design requirements are being met is another important principle of risk-informed regulation as noted in Regulatory Guide 1.174. The PBMR design incorporates several different monitoring strategies to apply this principle. There is an on-line system to monitor the circulating radioactivity and chemical purity of the primary coolant to provide an immediate indication of unexpected adverse fuel performance

and/or intrusion of impurities into the primary coolant. The on-line refueling system continuously monitors fuel elements to ensure that safety margins on fuel burn-up are not approached and provides an opportunity to identify damaged fuel elements. Temperature and flow measurements throughout the primary cooling circuit and the various cooling water circuits ensure that the reactor vessel and PPB components are operating within temperature limits and that cooling water system integrity is being maintained. There is a capability to defuel the reactor if needed to perform inspections and maintenance activities at shutdown. In addition, radioactivity monitors in the containment and management of helium inventories provide indications of any leakage from the primary barrier. On balance, the use of monitoring strategies is consistent with the PBMR with the selection of equipment for safety classification.

6.3 SPECIAL TREATMENT OF SAFETY-RELATED EQUIPMENT

Special treatment requirements have been developed and incorporated into NRC regulations to assure reliability and effectiveness of safety-related SSCs during design basis accidents. These requirements address quality assurance, maintenance, in-service testing, in-service inspection, equipment qualification, and other treatments to assure adequate reliability of SSCs during design basis accidents. More recently, the industry and the NRC are working to incorporate risk-informed insights into attempts to reformulate these special treatment requirements.

Beyond Reg. Guide 1.174, the NRC has risk-informed regulatory activities underway to support changes to the special treatment rules in Part 50 to modify their scope to be risk-informed (Option 2). These activities are consistent with the philosophy of Reg. Guide 1.174, but the specific applications are still evolving.

The Option 2 effort involves the categorization of SSCs into risk-informed safety classifications. The process includes consideration of the existing safety classification and plant-specific risk insights. The industry has been developing implementation guidelines. These guidelines developed for existing LWR, rely upon the CDF and LERF risk metrics, and account for the variety of the risk tools employed across the industry. Consequently, the details of the categorization process are different than the proposed PBMR licensing framework. However, the philosophy of the approach is similar in both documents. In the PBMR rather than using CDF and LERF, the frequencies and dose-consequences of the Licensing Basis Events derived from the PRA are utilized. Fundamentally, this is equivalent to the Option 2 concepts applied to relevant PBMR event sequence classes.

Given the DBE, the PBMR approach to safety classification outlined in Section 6 follows the conventional regulatory practice: a set of SSC are selected that are shown to be sufficient in the event sequence analyses. Furthermore, since the SSC are explicitly linked to the spectrum of DBE, all regulatory requirements including the so called special treatment requirements are expected to be directly developed on a case-by-case basis. In essence, risk-informing a deterministic licensing basis as in the Option 2 and 3 efforts for LWR is not required for the PBMR as risk-informed principles will be used to derive the

special treatment requirements within the initial license. Indeed, the risk-significance is built-in with the probabilistic foundation of the PBMR DBE.

Rather than impose additional arbitrary, blanket special treatment requirements for all safety-related SSC, and other artifacts of the pre-risk-informed licensing era, it is proposed that an appropriate set of regulatory design requirements be developed for each DBE on a case by case basis and that risk-informed special treatment then applied to the corresponding SSCs.

Currently, it is not expected that there will be a need for special treatment for SSCs solely for the purpose of preventing or mitigating EPBEs. For example, for the MHTGR, the design functions that ensured that EPBEs remained within acceptable limits were the same functions that were needed for the DBEs. Since an appropriate level of special treatment is applied to the ensure the reliability and availability of these design functions for purposes of protecting against DBEs, additional treatment ss not needed for these functions with respect to EPBEs. A similar result is expected for the PBMR.

Additionally, it is expected that some non-safety-related SSCs will perform a defense-in-depth function or provide safety margin. These SSCs will be evaluated on a case-by-case basis to determine whether enhanced treatment (i.e., treatment in excess of normal industrial practices) is warranted. In some cases such as fire protection systems and radwaste systems, some enhanced treatment may be warranted. For active systems that are normally operating, no additional treatment may be warranted.

7 CONCLUSIONS

The development of the PBMR licensing approach is fully consistent with the NRC's regulations and Policy on Advanced Reactors. The early agreement on the processes and tools that are described in this paper is important to the development of a more certain and stable regulatory environment within which the PBMR design and application can mature. Equally important, the development of the regulatory set needed will identify work needed by the NRC to expand, modify or develop regulatory guidance that currently does not exist for gas-cooled reactors like the PBMR. Finally, the complete implementation of these processes will provide a firm foundation for the NRC staff to prepare for and conduct an efficient and effective review of the PBMR application.

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20. "An Approach for Using Probabilistic Risk Assessment In Risk-Informed Decisions On Plant-Specific Changes to the Licensing Basis," USNRC, Regulatory Guide 1.174, July 1998.

21. "Framework for Risk-Informed Changes to the Technical Requirements of 10 CFR 50," USNRC, SECY 00-198, August 2000.

22. "Reactor Risk Reference Document," USNRC, NUREG-1150, Volume 1, February 1987.

23. "Probabilistic Risk Assessment for the Standard High Temperature Gas-Cooled Reactor," Department of Energy, DOE-HTGR-86011, Revision 5, April 1988.

APPENDIX A

SCREENING SAMPLE AND PRELIMINARY CLASSIFICATION					
Regulation / GDC / Appendix	**Applies**	**Partially Applies**	**Not Applicable**	**PBMR Specifics Needed**	**Comments**
§50.1	X				
§50.2		X			
§50.3	X				
§50.4	X				
§50.5	X				
§50.7	X				
§50.8	X				
§50.9	X				
§50.10	X				
§50.11			X		
§50.12	X				
§50.13	X				
§50.20	X				
§50.21			X		
§50.22	X				
§50.23	X				
§50.30	X				
§50.31	X				
§50.32	X				
§50.33	X				
§50.33f	X				
§50.33g	X				

SCREENING SAMPLE AND PRELIMINARY CLASSIFICATION					
Regulation / GDC / Appendix	**Applies**	**Partially Applies**	**Not Applicable**	**PBMR Specifics Needed**	**Comments**
§50.33k	X				
§50.33A	X				
§50.34		X			
§50.34A	X				See below with respect to the references to Appendix I
§50.35	X				
§50.36	X				
§50.36A	X				See below with respect to the references to Appendix I
§50.36B	X				
§50.37	X				
§50.38	X				
§50.39	X				
§50.40	X				
§50.41			X		
§50.42	X				
§50.43	X				
§50.44			X		
§50.45	X				

SCREENING SAMPLE AND PRELIMINARY CLASSIFICATION					
Regulation / GDC / Appendix	**Applies**	**Partially Applies**	**Not Applicable**	**PBMR Specifics Needed**	**Comments**
§50.46			X	X	
§50.47	X				
§50.48	X				See below with respect to the references to Appendix R
§50.49		X			
§50.50	X				
§50.51	X				
§50.52	X				
§50.53	X				
§50.54		X			
§50.55	X				
§50.55a	XX				
§50.56	X				
§50.57	X				
§50.58	X				
§50.59	X				
§50.60			X		
§50.61			X		
§50.62			X		
§50.63			X	X	
§50.64			X		
§50.65	X				

SCREENING SAMPLE AND PRELIMINARY CLASSIFICATION					
Regulation / GDC / Appendix	**Applies**	**Partially Applies**	**Not Applicable**	**PBMR Specifics Needed**	**Comments**
§50.66			X		
§50.67			X		
§50.68		X			
§50.70	X				
§50.71	X				
§50.72	X				
§50.73	X				
§50.74	X				
§50.75	X				
§50.78	X				
§50.80	X				
§50.81	X				
§50.82	X				
§50.90	X				
§50.91	X				
§50.92	X				
§50.100	X				
§50.101	X				
§50.102	X				
§50.103	X				
§50.109	X				
§50.110	X				
§50.111	X				
§50.120	X				

SCREENING SAMPLE AND PRELIMINARY CLASSIFICATION					
Regulation / GDC / Appendix	**Applies**	**Partially Applies**	**Not Applicable**	**PBMR Specifics Needed**	**Comments**
App. A	X				
GDC 1 – Quality	XX				
GDC 2 – Protection against Natural Phenomena	XX				
GDC 3 – Fire Protection	XX				
GDC 4 – Environmental and Dynamic Qualification	XX				
GDC 5 – Sharing Systems between Units	XX				
GDC 10 – Reactor Design	XX				
GDC 11 – Reactor Inherent Protection	XX				
GDC 12 – Power Oscillations	XX				
GDC 13 – Instrumentation and Control	XX				
GDC 14 – RCS Boundary		XX			
GDC 15 – RCS Design	XX				
GDC 16 – Containment		XX			
GDC 17 – Electric Power Systems		XX			
GDC 18 – Inspection and Testing of Electric Power Systems	XX				
GDC 19 – Control Room		XX			
GDC 20 – Protection System Functions	XX				
GDC 21 – Protection System Reliability	XX				
GDC 22 - Protection System Independence	XX				
GDC 23 - Protection System Failure Mode	XX				
GDC 24 - Separation of Protection and Control Systems	XX				

SCREENING SAMPLE AND PRELIMINARY CLASSIFICATION					
Regulation / GDC / Appendix	Applies	Partially Applies	Not Applicable	PBMR Specifics Needed	Comments
GDC 25 – Requirements for Reactivity Control Malfunctions	XX				
GDC 26 – Reactivity Control System Redundancy		XX			
GDC 27 – Reactivity Control System Capability		XX			
GDC 28 – Reactivity Limits		XX			
GDC 29 – Protection against Anticipated Operational Occurrences	XX				
GDC 30 – Quality of RCS Boundary		XX			
GDC 31 – Fracture Prevention of RCS	XX				
GDC 32 – Inspection of RCS Boundary	XX				
GDC 33 – Reactor Coolant Makeup			X		
GDC 34 - Residual Heat Removal	XX				
GDC 35 - Emergency Core Cooling			X		
GDC 36 - Inspection of ECCS			X		
GDC 37 - Testing of ECCS			X		
GDC 38 - Containment Heat Removal			X		
GDC 39 – Inspection of Containment Heat Removal			X		
GDC 40 - Testing of Containment Heat Removal			X		
GDC 41 - Containment Atmospheric Cleanup		XX			
GDC 42 - Inspection of Containment Atmospheric Cleanup	XX				
GDC 43 - Testing of Containment Atmospheric Cleanup	XX				
GDC 44 - Cooling Water,	XX				
GDC 45 - Inspection of Cooling Water System	XX				

SCREENING SAMPLE AND PRELIMINARY CLASSIFICATION					
Regulation / GDC / Appendix	**Applies**	**Partially Applies**	**Not Applicable**	**PBMR Specifics Needed**	**Comments**
GDC 46 - Testing of Cooling Water System	XX				
GDC 50 - Containment Design Basis		XX			
GDC 51 - Fracture Prevention of Containment		XX			
GDC 52 - Containment Leak Rate Testing		XX			
GDC 53 - Containment Testing and Inspection		XX			
GDC 54 - Systems Penetrating Containment		XX			
GDC 55 - RCS Penetrating Containment			X		
GDC 56 - Primary Containment Isolation		XX			
GDC 57 - Closed System Isolation Valves		XX			
GDC 60 – Control of Radioactive Releases	XX				
GDC 61 - Fuel Storage and Handling	XX				
GDC 62 – Prevention of Criticality	XX				
GDC 63 – Monitoring Fuel and Waste	XX				
GDC 64 - Monitoring Releases	XX				
App. B	X				
App. C	X				
App. D					
App. E	X				
App. G			X	X	
App. H			X	X	
App. I		XX			
App J			X		
App. K			X	X	
App. L	X				

SCREENING SAMPLE AND PRELIMINARY CLASSIFICATION					
Regulation / GDC / Appendix	**Applies**	**Partially Applies**	**Not Applicable**	**PBMR Specifics Needed**	**Comments**
App. M	X				
App. N	X				
App. O	X				
App. Q	X				
App. R			X		
App. S	X				
Part 55	X				
Part 60			X		
Part 70	X				
Part73	X				
Part 75	X				
Part 100	X				
Part 110	X				
Part 140	X				
Totals	115	22	26	5	

Note: "XX" indicates that the regulation in question is applicable (or partially applicable) as guidance.

APPENDIX B

PBMR EXPERT PANEL REGULATORY SCREENING RESULTS July 10-11, 2001							
	Legend	A=Admin T=Technical Design	N=Non Reactor O=O&M				
GDC/Reg/App.	Applies	Partially Applies	Applies as Guidance	Partially Applies as Guidance	Not Applicable	PBMR Specifics Needed	Comments
§50.1 - BASIS, PURPOSE, AND PROCEDURES APPLICABLE	A						
§50.2 - DEFINITIONS		T					
§50.3 - INTERPRETATIONS	A						
§50.4 - WRITTEN COMMUNICATIONS	A						
§50.5 - DELIBERATE MISCONDUCT	A						
§50.7 - EMPLOYEE PROTECTION	A						
§50.8 - INFORMATION COLLECTION REQUIREMENTS: OMB APPROVAL	A						
§50.9 - COMPLETENESS AND ACCURACY OF INFORMATION	A						
§50.10 - LICENSE REQUIRED	A						
§50.11 - EXCEPTIONS AND EXEMPTIONS FROM LICENSING REQUIREMENTS					N		
§50.12- SPECIFIC EXEMPTIONS	A						

PBMR EXPERT PANEL REGULATORY SCREENING RESULTS July 10-11, 2001							
	Legend	A=Admin T=Technical Design	N=Non Reactor O=O&M				
GDC/Reg/App.	**Applies**	**Partially Applies**	**Applies as Guidance**	**Partially Applies as Guidance**	**Not Applicable**	**PBMR Specifics Needed**	**Comments**
§50.13 - ATTACKS AND DESTRUCTIVE ACTS BY ENEMIES OF THE UNITED STATES; AND DEFENSE ACTIVITIES	T						
§50.20- TWO CLASSES OF LICENSES	A						
§50.21 - CLASS 104 LICENSES; FOR MEDICAL THERAPY AND RESEARCH AND DEVELOPMENT FACILITIES					N		
§50.22- CLASS 103 LICENSES; FOR COMMERCIAL AND INDUSTRIAL FACILITIES	A						
§50.23 - CONSTRUCTION PERMITS	A						
§50.30 - FILING OF APPLICATIONS FOR LICENSES; OATH OR AFFIRMATION	A						
§50.31- COMBINING APPLICATIONS	A						
§50.32 - ELIMINATION OF REPETITION	A						
§50.33 - CONTENTS OF APPLICATIONS; GENERAL INFORMATION	A						
§50.33f... financial qualification of the applicant	A						

PBMR EXPERT PANEL REGULATORY SCREENING RESULTS

July 10-11, 2001

	Legend	A=Admin T=Technical Design	N=Non Reactor O=O&M				
GDC/Reg/App.	**Applies**	**Partially Applies**	**Applies as Guidance**	**Partially Applies as Guidance**	**Not Applicable**	**PBMR Specifics Needed**	**Comments**
§50.33g... radiological emergency response plans	T						
§50.33k... funds will be available to decommission the facility.	A						
§50.33A - INFORMATION REQUESTED BY THE ATTORNEY GENERAL FOR ANTITRUST REVIEW	A						
§50.34 - CONTENTS OF APPLICATIONS; TECHNICAL INFORMATION				T			Parts apply as written, parts apply as guidance, parts do not apply
§50.34A - DESIGN OBJECTIVES FOR EQUIPMENT TO CONTROL RELEASES OF RADIOACTIVE MATERIAL IN EFFLUENTS-NUCLEAR POWER REACTORS	T						
§50.35 - ISSUANCE OF CONSTRUCTION PERMITS	A						
§50.36 - TECHNICAL SPECIFICATIONS	A						

PBMR EXPERT PANEL REGULATORY SCREENING RESULTS July 10-11, 2001							
	Legend	A=Admin T=Technical Design	N=Non Reactor O=O&M				
GDC/Reg/App.	**Applies**	**Partially Applies**	**Applies as Guidance**	**Partially Applies as Guidance**	**Not Applicable**	**PBMR Specifics Needed**	**Comments**
§50.36A- TECHNICAL SPECIFICATIONS ON EFFLUENTS FROM NUCLEAR POWER REACTORS	A						
§50.36B- ENVIRONMENTAL CONDITIONS	A						
§50.37 - AGREEMENT LIMITING ACCESS TO CLASSIFIED INFORMATION	A						
§50.38- INELIGIBILITY OF CERTAIN APPLICANTS	A						
§50.39- PUBLIC INSPECTION OF APPLICATIONS	A						
§50.40- COMMON STANDARDS	A						
§50.41 - ADDITIONAL STANDARDS FOR CLASS 104 LICENSES					N		
§50.42- ADDITIONAL STANDARDS FOR CLASS 103 LICENSES	A						
§50.43 - ADDITIONAL STANDARDS AND PROVISIONS AFFECTING CLASS 103 LICENSES FOR COMMERCIAL POWER	A						

PBMR EXPERT PANEL REGULATORY SCREENING RESULTS

July 10-11, 2001

Legend: A=Admin, T=Technical Design, N=Non Reactor, O=O&M

GDC/Reg/App.	Applies	Partially Applies	Applies as Guidance	Partially Applies as Guidance	Not Applicable	PBMR Specifics Needed	Comments
§50.44 - STANDARDS FOR COMBUSTIBLE GAS CONTROL SYSTEM IN LIGHT-WATER-COOLED POWER REACTORS					T		
§50.45 - STANDARDS FOR CONSTRUCTION PERMITS	A						
§50.46 - ACCEPTANCE CRITERIA FOR EMERGENCY CORE COOLING SYSTEMS FOR LIGHT-WATER NUCLEAR POWER REACTORS					T	T	
§50.47 - EMERGENCY PLANS	T						
§50.48 - FIRE PROTECTION	T						
§50.49- ENVIRONMENTAL QUALIFICATION OF ELECTRIC EQUIPMENT IMPORTANT TO SAFETY FOR NUCLEAR POWER PLANTS				T			
§50.50- ISSUANCE OF LICENSES AND CONSTRUCTION PERMITS	A						
§50.51 - CONTINUATION OF LICENSE	A						
§50.52 - COMBINING LICENSES	A						
§50.53- JURISDICTIONAL LIMITATIONS	A						
§50.54 - CONDITIONS OF LICENSES				T			

PBMR EXPERT PANEL REGULATORY SCREENING RESULTS

July 10-11, 2001

	Legend	A=Admin T=Technical Design	N=Non Reactor O=O&M				
GDC/Reg/App.	**Applies**	**Partially Applies**	**Applies as Guidance**	**Partially Applies as Guidance**	**Not Applicable**	**PBMR Specifics Needed**	**Comments**
§50.55 - CONDITIONS OF CONSTRUCTION PERMITS	A						
§50.55a - CODES AND STANDARDS	T						
§50.56 - CONVERSION OF CONSTRUCTION PERMIT TO LICENSE; OR AMENDMENT OF LICENSE	A						
§50.57 - ISSUANCE OF OPERATING LICENSE	A						
§50.58 - HEARINGS AND REPORT OF THE ADVISORY COMMITTEE ON REACTOR SAFEGUARDS	A						
§50.59- CHANGES, TESTS AND EXPERIMENTS	A						
§50.60 - ACCEPTANCE CRITERIA FOR FRACTURE PREVENTION MEASURES FOR LIGHT-WATER NUCLEAR POWER REACTORS FOR NORMAL OPERATION					T		
§50.61 - FRACTURE TOUGHNESS REQUIREMENTS FOR PROTECTION AGAINST PRESSURIZED THERMAL SHOCK EVENTS					T		

PBMR EXPERT PANEL REGULATORY SCREENING RESULTS

July 10-11, 2001

Legend: A=Admin, T=Technical Design, N=Non Reactor, O=O&M

GDC/Reg/App.	Applies	Partially Applies	Applies as Guidance	Partially Applies as Guidance	Not Applicable	PBMR Specifics Needed	Comments
§50.62 - REQUIREMENTS FOR REDUCTION OF RISK FROM ANTICIPATED TRANSIENTS WITHOUT SCRAM (ATWS) EVENTS FOR LIGHT-WATER-COOLED NUCLEAR POWER PLANTS					T		
§50.63 - LOSS OF ALL ALTERNATING CURRENT POWER					T	T	
§50.64 - LIMITATIONS ON THE USE OF HIGHLY ENRICHED URANIUM (HEU) IN DOMESTIC NON-POWER REACTORS					T		
§50.65 - REQUIREMENTS FOR MONITORING THE EFFECTIVENESS OF MAINTENANCE AT NUCLEAR POWER PLANTS	O						
§50.66 - REQUIREMENTS FOR THERMAL ANNEALING OF THE REACTOR PRESSURE VESSEL					T		
§50.67 - ACCIDENT SOURCE TERM					A		
§50.68 - CRITICALITY ACCIDENT REQUIREMENTS				T			
§50.70 - INSPECTIONS	A						

PBMR EXPERT PANEL REGULATORY SCREENING RESULTS July 10-11, 2001							
	Legend	A=Admin T=Technical Design	N=Non Reactor O=O&M				
GDC/Reg/App.	**Applies**	**Partially Applies**	**Applies as Guidance**	**Partially Applies as Guidance**	**Not Applicable**	**PBMR Specifics Needed**	**Comments**
§50.71- MAINTENANCE OF RECORDS, MAKING OF REPORTS	A						
§50.72 - IMMEDIATE NOTIFICATION REQUIREMENTS FOR OPERATING NUCLEAR POWER REACTORS	A						
§50.73 - LICENSE EVENT REPORT SYSTEM	A						Requires identification of all PBMR-specific structures, systems, components
§50.74 - NOTIFICATION OF CHANGE IN OPERATOR OR SENIOR OPERATOR STATUS	A						
§50.75 - REPORTING AND RECORDKEEPING FOR DECOMMISSIONING PLANNING	A						Requires PBMR specific decommissioning information
§50.78 - INSTALLATION INFORMATION AND VERIFICATION	A						
§50.80 - TRANSFER OF LICENSES	A						
§50.81 - CREDITOR REGULATIONS	A						
§50.82 - TERMINATION OF LICENSE	A						
§50.90 - APPLICATION FOR AMENDMENT OF LICENSE OR CONSTRUCTION PERMIT	A						

PBMR EXPERT PANEL REGULATORY SCREENING RESULTS

July 10-11, 2001

	Legend	A=Admin T=Technical Design	N=Non Reactor O=O&M				
GDC/Reg/App.	**Applies**	**Partially Applies**	**Applies as Guidance**	**Partially Applies as Guidance**	**Not Applicable**	**PBMR Specifics Needed**	**Comments**
§50.91 - NOTICE FOR PUBLIC COMMENT; STATE CONSULTATION	A						
§50.92 - ISSUANCE OF AMENDMENT	A						
§50.100 - REVOCATION, SUSPENSION, MODIFICATION OF LICENSES AND CONSTRUCTION PERMITS FOR CAUSE	A						
§50.101 - RETAKING POSSESSION OF SPECIAL NUCLEAR MATERIAL	A						
§50.102 - COMMISSION ORDER FOR OPERATION AFTER REVOCATION	A						
§50.103 - SUSPENSION AND OPERATION IN WAR OR NATIONAL EMERGENCY	A						
§50.109 - BACKFITTING	A						
§50.110 - VIOLATIONS	A						
§50.111 - CRIMINAL PENALTIES	A						
§50.120 - TRAINING AND QUALIFICATION OF NUCLEAR POWER PLANT PERSONNEL	O						

PBMR EXPERT PANEL REGULATORY SCREENING RESULTS July 10-11, 2001							
	Legend	A=Admin T=Technical Design	N=Non Reactor O=O&M				
GDC/Reg/App.	**Applies**	**Partially Applies**	**Applies as Guidance**	**Partially Applies as Guidance**	**Not Applicable**	**PBMR Specifics Needed**	**Comments**
App. A - GENERAL DESIGN CRITERIA FOR NUCLEAR POWER PLANTS	T						Applies as guidance for non-light water reactors as written in the introduction
GDC 1 – Quality			A				
GDC 2 – Protection against Natural Phenomena			T				
GDC 3 – Fire Protection			T				
GDC 4 – Environmental and Dynamic Qualification			T				
GDC 5 – Sharing Systems between Units			T				
GDC 10 – Reactor Design			T				
GDC 11 – Reactor Inherent Protection			T				
GDC 12 – Power Oscillations			T				
GDC 13 – Instrumentation and Control			T				
GDC 14 – RCS Boundary				T			
GDC 15 – RCS Design			T				Linked to the treatment on GDC 14
GDC 16 – Containment				T			

PBMR EXPERT PANEL REGULATORY SCREENING RESULTS

July 10-11, 2001

Legend: A=Admin, T=Technical Design, N=Non Reactor, O=O&M

GDC/Reg/App.	Applies	Partially Applies	Applies as Guidance	Partially Applies as Guidance	Not Applicable	PBMR Specifics Needed	Comments
GDC 17 – Electric Power Systems				T			
GDC 18 – Inspection and Testing of Electric Power Systems			T				
GDC 19 – Control Room				T			
GDC 20 – Protection System Functions			T				
GDC 21 – Protection System Reliability			T				
GDC 22 - Protection System Independence			T				
GDC 23 - Protection System Failure Mode			T				
GDC 24 - Separation of Protection and Control Systems			T				
GDC 25 – Requirements for Reactivity Control Malfunctions			T				
GDC 26 – Reactivity Control System Redundancy				T			
GDC 27 – Reactivity Control System Capability				T			
GDC 28 – Reactivity Limits				T			

PBMR EXPERT PANEL REGULATORY SCREENING RESULTS

July 10-11, 2001

Legend: A=Admin, T=Technical Design, N=Non Reactor, O=O&M

GDC/Reg/App.	Applies	Partially Applies	Applies as Guidance	Partially Applies as Guidance	Not Applicable	PBMR Specifics Needed	Comments
GDC 29 – Protection against Anticipated Operational Occurrences			T				
GDC 30 – Quality of RCS Boundary				T			
GDC 31 – Fracture Prevention of RCS			T				
GDC 32 – Inspection of RCS Boundary			T				
GDC 33 – Reactor Coolant Makeup					T		
GDC 34 - Residual Heat Removal			T				
GDC 35 - Emergency Core Cooling					T		
GDC 36 - Inspection of ECCS					T		
GDC 37 - Testing of ECCS					T		
GDC 38 - Containment Heat Removal					T		
GDC 39 – Inspection of Containment Heat Removal					T		
GDC 40 - Testing of Containment Heat Removal					T		
GDC 41 - Containment Atmospheric Cleanup				T			
GDC 42 - Inspection of Containment Atmospheric Cleanup			T				
GDC 43 - Testing of Containment Atmospheric Cleanup			T				

PBMR EXPERT PANEL REGULATORY SCREENING RESULTS July 10-11, 2001							
	Legend	A=Admin T=Technical Design	N=Non Reactor O=O&M				
GDC/Reg/App.	**Applies**	**Partially Applies**	**Applies as Guidance**	**Partially Applies as Guidance**	**Not Applicable**	**PBMR Specifics Needed**	**Comments**
GDC 44 - Cooling Water,			T				
GDC 45 - Inspection of Cooling Water System			T				
GDC 46 - Testing of Cooling Water System			T				
GDC 50 - Containment Design Basis				T			
GDC 51 - Fracture Prevention of Containment				T			
GDC 52 - Containment Leak Rate Testing				T			
GDC 53 - Containment Testing and Inspection				T			
GDC 54 - Systems Penetrating Containment				T			
GDC 55 - RCS Penetrating Containment					T		
GDC 56 - Primary Containment Isolation				T			
GDC 57 - Closed System Isolation Valves				T			
GDC 60 – Control of Radioactive Releases			T				

PBMR EXPERT PANEL REGULATORY SCREENING RESULTS July 10-11, 2001							
	Legend	A=Admin T=Technical Design	N=Non Reactor O=O&M				
GDC/Reg/App.	**Applies**	**Partially Applies**	**Applies as Guidance**	**Partially Applies as Guidance**	**Not Applicable**	**PBMR Specifics Needed**	**Comments**
GDC 61 - Fuel Storage and Handling			T				
GDC 62 – Prevention of Criticality			T				
GDC 63 – Monitoring Fuel and Waste			T				
GDC 64 - Monitoring Releases			T				
App. B- QUALITY ASSURANCE CRITERIA FOR NUCLEAR POWER PLANTS AND FUEL REPROCESSING PLANTS	T						
App. C - A GUIDE FOR THE FINANCIAL DATA AND RELATED INFORMATION REQUIRED TO ESTABLISH FINANCIAL QUALIFICATIONS FOR FACILITY CONSTRUCTION PERMITS	T						
App. E - EMERGENCY PLANNING AND PREPAREDNESS FOR PRODUCTION AND UTILIZATION FACILITIES	T						
App.G - FRACTURE TOUGHNESS REQUIREMENTS					T	T	
App. H - REACTOR VESSEL MATERIAL SURVEILLANCE PROGRAM REQUIREMENTS					T	T	

PBMR EXPERT PANEL REGULATORY SCREENING RESULTS

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Legend: A=Admin; T=Technical Design; N=Non Reactor; O=O&M

GDC/Reg/App.	Applies	Partially Applies	Applies as Guidance	Partially Applies as Guidance	Not Applicable	PBMR Specifics Needed	Comments
App. I - NUMERICAL GUIDES FOR DESIGN OBJECTIVES AND LIMITING CONDITIONS OF OPERATION TO MEET THE CRITERION "AS LOW AS IS REASONABLY ACHIEVABLE" FOR RADIOACTIVE MATERIAL IN LIGHT-WATER-COOLED NUCLEAR POWER REACTOR EFFLUENTS		T					
App J - PRIMARY REACTOR CONTAINMENT LEAKAGE TESTING FOR WATER-COOLED POWER REACTORS					T		
App. K - ECCS EVALUATION MODELS					T	T	
App. L- Information Requested by the Attorney General for Antitrust Review of Facility Construction Permits and Initial Operating Licenses	A						

PBMR EXPERT PANEL REGULATORY SCREENING RESULTS July 10-11, 2001							
	Legend	A=Admin T=Technical Design	N=Non Reactor O=O&M				
GDC/Reg/App.	**Applies**	**Partially Applies**	**Applies as Guidance**	**Partially Applies as Guidance**	**Not Applicable**	**PBMR Specifics Needed**	**Comments**
App. M - STANDARDIZATION OF DESIGN; MANUFACTURE OF NUCLEAR POWER REACTORS; CONSTRUCTION AND OPERATION OF NUCLEAR POWER REACTORS MANUFACTURED PURSUANT TO COMMISSION LICENSE	A						
App. N - STANDARDIZATION OF NUCLEAR POWER PLANT DESIGNS: LICENSES TO CONSTRUCT AND OPERATE NUCLEAR POWER REACTORS OF DUPLICATE DESIGN AT MULTIPLE SITES	A						
App. O - STANDARDIZATION OF DESIGN: STAFF REVIEW OF STANDARD DESIGNS	A						
App. Q - PRE-APPLICATION EARLY REVIEW OF SITE SUITABILITY ISSUES	A						
App. R - FIRE PROTECTION PROGRAM FOR NUCLEAR POWER FACILITIES OPERATING PRIOR TO JANUARY 1, 1979					T		

PBMR EXPERT PANEL REGULATORY SCREENING RESULTS

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Legend	A=Admin T=Technical Design	N=Non Reactor O=O&M	

GDC/Reg/App.	Applies	Partially Applies	Applies as Guidance	Partially Applies as Guidance	Not Applicable	PBMR Specifics Needed	Comments
App. S - EARTHQUAKE ENGINEERING CRITERIA FOR NUCLEAR POWER PLANTS	T						
§55 - OPERATORS' LICENSES	O						
§60 - DISPOSAL OF HIGH-LEVEL RADIOACTIVE WASTES IN GEOLOGIC REPOSITORIES					N		
§70 - DOMESTIC LICENSING OF SPECIAL NUCLEAR MATERIAL	A						
§73 - PHYSICAL PROTECTION OF PLANTS AND MATERIALS	T						
§75 - SAFEGUARDS ON NUCLEAR MATERIAL-IMPLEMENTATION OF US/IAEA AGREEMENT	A						
§100 - REACTOR SITE CRITERIA	T						
§110 - EXPORT AND IMPORT OF NUCLEAR EQUIPMENT AND MATERIAL	A						
§140- FINANCIAL PROTECTION REQUIREMENTS AND INDEMNITY AGREEMENTS	A						
Totals	84	2	31	20	26	5	