UNITED STATES
NUCLEAR REGULATORY COMMISSION
ADVISORY COMMITTEE ON REACTOR SAFEGUARDS
WASHINGTON, DC 20555 - 0001

April 5, 2007

MEMORANDUM TO: ACRS Members

FROM: David C. Fischer, Senior Staff Engineer
Technical Support Staff
ACRS/ACNW

David C. Fischer

SUBJECT: CERTIFICATION OF THE MINUTES OF THE ACRS FUTURE PLANT DESIGNS SUBCOMMITTEE MEETING ON THE TECHNOLOGY NEUTRAL LICENSING FRAMEWORK (WORKING DRAFT NUREG-1860) MARCH 7, 2007, ROCKVILLE, MARYLAND

The minutes of the subject meeting were certified on April 5, 2007, as the official record of the proceedings of that meeting. A copy of the certified minutes is attached.

Attachment: As stated

cc w/o Attachment:
F. Gillespie
S. Duraiswamy
C. Santos

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MEMORANDUM TO: David C. Fischer, Senior Staff Engineer
Technical Support Staff, ACRS

FROM: Thomas S. Kress, Chairman
Future Plant Designs Subcommittee

SUBJECT: CERTIFICATION OF THE MINUTES OF THE ACRS FUTURE PLANT DESIGNS SUBCOMMITTEE MEETING ON THE TECHNOLOGY NEUTRAL LICENSING FRAMEWORK (WORKING DRAFT NUREG-1860) MARCH 7, 2007, ROCKVILLE, MARYLAND

I hereby certify, to the best of my knowledge and belief, that the minutes of the subject meeting on March 7, 2007, are an accurate record of the proceedings for that meeting.

T. S. Kress 4/5/07

Thomas S. Kress, Date
Future Plant Designs Subcommittee, Chairman

UNITED STATES
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Certified

Issued: April 5, 2007

ADVISORY COMMITTEE ON REACTOR SAFEGUARDS FUTURE PLANT DESIGNS SUBCOMMITTEE MEETING MINUTES MARCH 7, 2007 ROCKVILLE, MARYLAND

INTRODUCTION

The ACRS Subcommittee on Future Plant Designs met on March 7, 2007, at 11545 Rockville Pike, Rockville, Maryland, in Room T-2B3. The purpose of this meeting was to review the NRC staff's work on technology neutral licensing framework (i.e., Working Draft NUREG-1860) with a focus on ensuring the value of such an approach versus the development of a licensing framework for specific designs, such as a high temperature gas cooled reactor or a liquid metal cooled reactor (reference the Commission's November 8, 2006, Staff Requirements Memorandum to Dr. Larkins). During the briefing, the Committee also explored with the NRC staff the pros and cons of developing a licensing framework for specific designs. The Subcommittee gathered information, analyzed relevant issues and facts, and formulated proposed positions and actions, as appropriate, for deliberation by the full Committee. The entire meeting was open to public attendance. Mr. David C. Fischer was the cognizant staff engineer and the Designated Federal Official for this meeting. The Subcommittee received no written comments, or requests for time to make oral statements from any members of the public regarding this meeting. The meeting was convened at 10:00 am and adjourned at 4:21 pm.

ATTENDEES

ACRS

Thomas S. Kress, Chairman
Said Abdel-Khalik, Member
George E. Apostolakis, Member
Mario V. Bonaca, Member
Michael Corradini, Member
Otto L. Maynard, Member
Dana A. Powers, Member
William J. Shack, Member
Graham B. Wallis, Member
David C. Fischer, ACRS Staff

NRC

Sud Basu, RES/DRASP/NRCA
Ben Beasley, RES/DRASP/PRA
David Bessette, RES/DRASP/NRCA
Joe Birmingham, NRR/DPR
Mary Drouin, RES/DRASP/PRA
Don Dube, NRR/DRA
Lauren Killian, RES/DRASP/PRA
Eileen McKenna, NRR/DPR
Lynn Mrowca, NRO/DSRA
Yuri Orecheva, NRR/DSS
Stuart Rubin, RES/DRASP/NRCA
Rajendra Solanki, NRR/DRA/APLA

Ronaldo Jenkins, RES/DRASP/PRA
Thomas Kenyon, NRO/DNRL/MWB
Martin Stutzke, NRR/DRA/APLA

OTHERS

Biff Bradley, NEI
Edward Burns, PBMR
Shawnie Harris, CH2M Hill
Tom King, ISL, Inc.
Alan Levin, AREVA
John Lehner, BNL
Bruce Mrowca, ISL
Vinod Mubayi, BNL
Patrick O'Regan, EPRI

A complete list of attendees is in the ACRS Office file and will be made available upon request. The presentation slides and handouts used during the meeting are attached to the Office Copy of these minutes.

OPENING REMARKS BY THE SUBCOMMITTEE CHAIRMAN

Dr. Thomas S. Kress, Chairman of the Future Plant Designs Subcommittee, stated that the purpose of this meeting was to review the NRC staff's work on technology neutral licensing framework (i.e., Working Draft NUREG-1860) with a focus on ensuring the value of such an approach versus the development of a licensing framework for specific designs, such as a high temperature gas cooled reactor or a liquid metal cooled reactor.

STAFF PRESENTATIONS

Introductory Remarks:

Ms. Mary Drouin (RES), Mr. Tom King (ISL), Mr. Marty Stutzke (NRR/NRO), and Mr. John Lehner (BNL) were seated at the head table with the subcommittee.

Mr. John Monninger (RES) introduced the staff's "Framework for Future Plant Licensing," oftentimes referred to as the "technology-neutral framework." He said that the framework has been under development for the past three years and that it benefitted from numerous meetings with stakeholders and workshops. He said that the framework has gained importance with the passage of the Energy Policy Act of 2005 and that the staff planned to use the lessons learned during the development of the framework to formulate the Next Generation Nuclear Plant (NGNP) licensing strategy.

Ms. Mary Drouin said that in January 2003 the RES Advanced Reactor Research Plan recognized the need for a licensing framework for advanced reactors. She said that the current regulatory structure focused on light-water reactors (LWRs) and had limited applicability to non-LWR technologies. She noted that some of the specific requirements in current regulations are not applicable to advanced reactor designs and added that advanced reactors will likely have design and operational issues different than LWRs. She said that the current regulatory structure has evolved with only limited insights from PRAs and severe accident research and suggested that PRA and PRA insights should be an integral part of licensing advanced reactors. In 2003 a program was initiated to develop a risk-informed, performance-based regulatory structure to support future reactor licensing.

The initial technology-neutral licensing "framework" is documented in draft NUREG1860 which, Ms Drouin said is ready to be published. Ms. Drouin said the framework provides guidance and criteria for creating a "risk-derived" and performance-based regulatory structure that can be implemented on either a technology-neutral or a technology-specific basis. She said the framework integrates Commission expectations as addressed in various policy statements, such as the policy statements on severe accidents, advanced reactors, PRA, and safety goals. Ms. Drouin said the framework was attached to an Advanced Notice of Proposed Rulemaking (ANPR) published in May 2006 for public review and comment. Public workshops were held on the framework in March 2005 and in September 2006. She said that two different versions of the framework were put on the NRC's public website which resulted in receiving two sets of comments from some organizations. Ms. Drouin identified the stakeholders who submitted comments on the ANPR and highlighted three general areas where comments were received (i.e., overall views, technology-neutral versus technology-specific, and how to proceed forward). She said that most comments received were generally supportive of moving forward with the development of a risk-informed performance-based regulatory framework for new reactors. However, she said that one comment was received that the framework "departs too far from the approximately 3000 reactor years experience gained using the deterministic approach," particularly as it relates to addressing common cause failure. Ms. Drouin said that some comments supported technology-neutral regulations with technology-specific implementing guidance, some supported technology-specific regulations, and others (e.g., NEI) indicated that it was too premature to decide. She said that there was not a consensus on this issue. With regard to how to proceed with rulemaking, Ms. Drouin said there seemed to be a consensus <u>not</u> to go to rulemaking in the near term. She identified three options for proceeding forward. The first was to gain experience with design certification of a non-LWR design using the framework approach. The second was a multi-year phased approach to rulemaking. And the third was a stepped approach in which the staff would develop a preliminary draft rule to be published for information upon receipt of a non-LWR application. The staff would then review and approve the non-LWR design using the current Part 50 and Part 52 regulations. Then the staff would evaluate the draft rule against the non-LWR design before publishing the proposed rule for public comment.

Ms. Drouin said that the staff planned to publish the framework (i.e., NUREG-1860) in the early summer (June) 2007. She said the staff is preparing a SECY paper to respond to Commission direction to "provide its [staff] recommendation on whether and, if so, how to proceed with rulemaking." She said that all activities related to the framework were going to be terminated. Dr. Kress expressed concern that work on the framework was going to be stopped. He said that some work still needed to be done to fine tune the framework. Ms. Drouin said that the staff is evaluating the need to defer rulemaking until experience is gained with NGNP and GNEP. She said that the staff is planning on briefing the ACRS on its plan for moving forward with rulemaking at the May full Committee meeting.

Dr. Wallis asked Ms. Drouin if the output from the PRA is core damage frequency (CDF) or is it a more comprehensive assessment of the effects on the public. Ms. Drouin said that when she uses the word PRA, she is using it in its entirety, not just CDF. Dr. Abdel-Khalik expressed concern that using the framework approach could lead to getting around certain Part 50 requirements, resulting in less stringent requirements for a plant. Ms. Drouin said that using the framework might lead to a different set of requirement but added that the requirements generated by using the framework should result in a safer plant (than by using existing deterministic requirements).

Framework Overview:

Ms. Drouin explained that the "framework" is a technical report (NUREG) that provides a structured and systematic approach in the form of guidelines and criteria for developing new requirements. The framework itself is not regulation but could serve as the basis for rulemaking (e.g., a new Part 53, exemptions or additions to Part 50). She said the framework uses a "risk-derived" approach and then explained the difference between a risk-derived and a risk-informed approach. A risk-derived approach starts with PRA results and integrates deterministic and defense-in-depth criteria (to compensate for uncertainties) as an integral part in the development of requirements. A risk-informed approach uses deterministic criteria to develop the requirements and then supplements them with risk insights. Ms. Drouin said that the framework can be applied or implemented (i.e., the development of requirements) on either a technology-neutral or on a technology-specific basis. The framework can be used to generate a risk-derived set of design, maintenance, and operating regulations or regulatory guidance. Dr. Kress questioned how long it would take to go from the framework to final rulemaking. Ms. Drouin said that she thought it could be completed in 5 years. Next, Ms. Drouin showed examples of regulations that might be generated by using the framework. Dr. Powers observed that the example regulation on Energetic Reaction Control (i.e., Reactor designs that have the potential for energetic reactions between the fuel, coolant or other material shall include provisions to prevent or mitigate the effects of such reactions.) did not tell the licensee what it is supposed to achieve. He said that without telling a licensee what it is supposed to achieve, almost anything could satisfy the requirement. Dr. Powers called such a requirement (unlike 10 CFR 50.46 which places specific limits on maximum cladding oxidation and maximum hydrogen generation) "non-functional."

Ms. Drouin reiterated that the framework describes a top-down process for developing a risk-derived, performance-based set of regulations that can be applied to any reactor technology. The process should define a goal (i.e., a defined measure of performance such as a quantitative measure of public health and safety) and the guidelines and criteria for achieving the goal. She said the process must also deal with completeness. The framework identifies high-level goals to protect public health and safety. The framework uses the protective strategies needed to ensure public health and safety as the structure to identify the requirements. Then the framework defines the guidelines and criteria to meet the overall goal within this structure (based on PRA and defense-in-depth guidance in the framework). By implementing these guidelines and criteria one can identify design, maintenance and operational requirements (on either a technology-neutral or technology-specific basis). Ms. Drouin said that the framework document contained guidance and criteria on the probabilistic approach, defense-in-depth, PRA technical acceptability, and how to identify requirements.

Round Table Discussion of Key Issues:

F-C Curve and Licensing Basis Event Selection

The subcommittee spent a considerable amount of time discussing the frequency-consequence (F-C) curve contained in draft NUREG-1860 (i.e., Figure 6-3).

Stakeholder Comments
- Different anchor points
- Add a CCDF curve

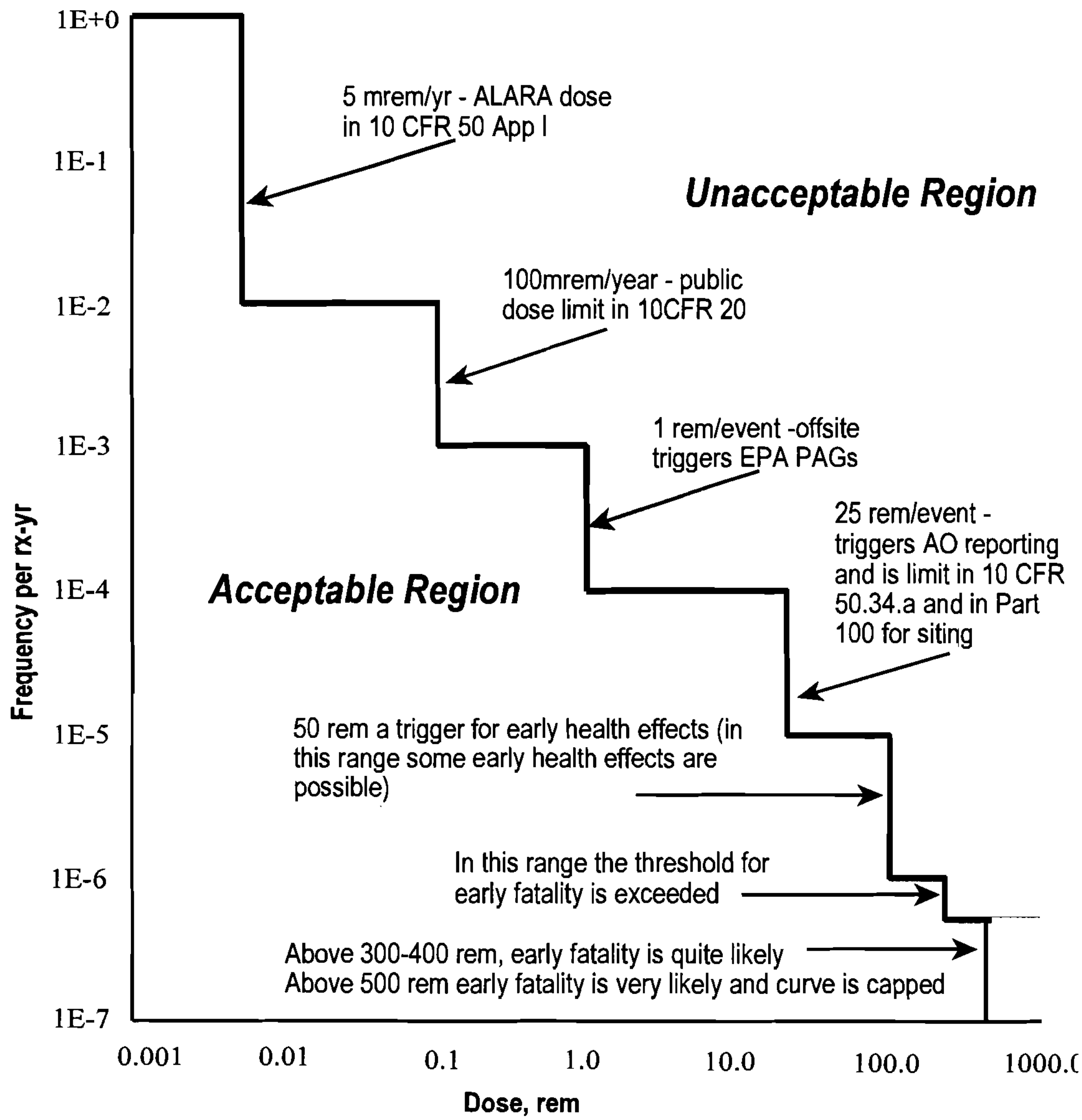

Figure 6-3 Frequency-consequence curve.

Dr. Kress commented that there was no reason why the curve should be stair-stepped. He said it could be a non-risk averse straight line. Second, he said that he would call it an F-C curve for identifying licensing basis events as opposed to a risk acceptance curve. Dr. Kress agreed that the curve should address different accident types and frequency ranges but said he would select the sequences with the maximum F-C product, not maximum frequency or maximum consequence. Dr. Kress observed that the curve does not summate the risk from all sequences. He said the framework needed a complementary cumulative distribution function curve. Finally, Dr. Kress said he thought the consequences should be measured in curies as opposed to dose (i.e., as calculating dose would require some knowledge of site characteristics and use of a Level 3 PRA whereas calculating curies would only require a Level 2 PRA). Dr. Wallis agreed with Dr. Kress that the proposed curve did not measure the cumulative impact on the public from all possible events. He also questioned the need for establishing licensing basis events. Dr. Powers agreed that the proposed curve should be called a licensing basis event curve, or something other than an F-C curve. Dr. Wallis also agreed with Dr. Kress that the curve should be a straight line. Dr. Bonaca asked if the prosed curve was developed by leveraging existing regulations and criteria. Dr. Kress agreed with Dr. Bonaca that the proposed curve seemed to be derived from existing regulation and therefore, he said, was not responsive to the Commission's expectation that new reactor designs have a higher level of safety than existing nuclear power plants. Dr. Bonaca questioned the need for defining licensing basis events, given the fact that the applicant will have a PRA, but acknowledged the benefits of having some limiting events to help anchor plant operations, technical specifications, identify SSCs for special treatment, etc. Mr. Lehner (BNL) said they tried to base the curve on current regulations which are generally in terms of consequence, but not frequency. He said they used the dose limits in the regulations, and qualitative estimates of the frequency where these dose limits would apply, to develop the frequency ranges. He said they had thought about constructing a straight line but thought they would have difficulty justifying intermediate points. Dr. Wallis questioned why a continuous release of 1mrem would be allowable. Mr. Lehner said that the doses up to 100 mrem are cumulative. Above 100 mrem the doses are individual events or sequences. Mr. King (ISL) noted that in addition to meeting the proposed curve the framework also requires an analysis to show that the design also meets the QHOs. Mr. Lehner noted that designers and reviewers can add design basis accidents (DBAs). The staff and subcommittee discussed the need for caution when aggregating event sequences into classes to avoid slicing it so thin that each class has a very low frequency of occurrence and therefore an acceptable level of risk. Mr Lehner said that one reason why you want licensing basis events is that you don't want it to be totally risk-based and acknowledged the benefits previously cited by Dr. Bonaca (e.g., stability in the parameters to which the operators control the plant). Dr. Kress said that having licensing basis events provides some defense-in-depth in that it forces the designer to consider a wide range of events including those that are not very risk significance. Mr. King said that considering licensing basis events is conservative and is not risk-based. He noted that identifying licensing basis events is necessary to implement certain other regulations like Part 20 and Part 100 and is used for safety classification, and to test the PRA, and to put margin in the design for defense-in-depth. Dr. Powers said that he did not think that a F-C curve needed to be constructed to reflect the Commission's desire that new plants be safer than existing plants. He also agreed with Dr. Wallis that DBAs are a dangerous concept because, he explained, plants are then designed to the DBAs rather than being designed to the risk. He clarified that he was, however, okay with identifying types of accidents to look at. Dr. Powers said that new plants are coming in with CDFs or events that are exceptionally low. Therefore, the risk is going to be dominated by those things that the PRA treats very poorly, such as aging, defects in construction, and external events. He added that he did not think the regulatory system needed to set lines for the operators. Ms. Drouin emphasized that using a risk-derived approach, as described in the framework, would require a PRA fundamentally better than those that are in use today. But she said that decisions would not be risk-based. Dr. Powers criticized the staff for over reliance on the PRA end points, like CDF and LERF, rather than gaining insights from the PRA. Mr King said that the framework uses defense-in-depth to address these completeness issues. Mr Lehner said that for these new technologies, the PRA is a tool for trying to discover new threats and combinations that you would not have otherwise thought of, a systematic way of looking for unique accident situations. Mr. Maynard said he preferred the stair-step F-C curve because it ties things together in a way that makes sense, but indicated that a straight line would be fine by him as well. He did not like the idea of relying totally on the PRA and favored defining DBEs to facilitate the development of processes and procedures to be used in the plant. Dr. Wallis suggested that DBAs would not be necessary if the PRA adequately modeled thermal-hydraulic phenomena (as opposed to just using deterministic success criteria). Dr. Apostolakis indicated that it was impractical to do thermal-hydraulic analyses for as many sequences as are in a PRA and, therefore, he thought it would be useful to identify bounding event sequences. Dr. Abdel-Khalik viewed the staff's proposed F-C curve as a way of identifying limiting event sequences which would be analyzed in greater detail. He said that other than meeting some cumulative risk criteria, the framework should not accept a plant design in which events of relatively high consequence would have probabilities of exceeding a certain value. Dr. Abdel-Khalik indicated that he liked the proposed F-C curve as a design tool because it is tied to current regulation but criticized it because it did not separate design acceptance from site acceptance. He also did not favor changing the unit for consequence from dose to curies because, he said, a curie of tritium is different than a curie of polonium. Dr. Kress suggested that this could be accounted for by converting various fission products to a standard using weighting factors. Dr. Apostolakis the three region approach (i.e., desirable, tolerable, and unacceptable regions) was used in the staff's proposed F-C curve. Ms. Drouin responded that the three region approach is not used in LBE selection. Mr. Maynard explained that the three region approach was part of the defense-in-depth philosophy for the safety, security, and preparedness expectation. Dr. Apostolakis argued against having hard and fast lines between the acceptable and unacceptable regions on the staff's proposed F-C curve. He said that having such lines of demarcation is impractical. He did not advocate establishing firm requirements (e.g., on reliability) or relying solely on the results of the PRA. He favored an approach similar to that used in RG 1.174 where there is increased management attention the closer you get to a particular value. Dr. Apostolakis suggested that the framework be pilot tested. Drs. Kress, Wallis, Abdel-Khalik, and Mr. Maynard all seemed to favor having "bright lines" separate the acceptable and unacceptable regions on the F-C curve. Dr. Corradini asked if the licensing basis events derived by using the framework would differ from the design basis accidents developed using a deterministic approach (e.g., for Clinch River or Fort St. Vrain). Mr. Mrowca (ISL) said that while the two different approaches would produce generally consistent results, there would also be differences (i.e., some additional sequences and some that would be out of the scope of the framework) because of the F-C curve or defense-in-depth criteria. Ms. Drouin clarified that the framework outlines a process for determining linensing basis events and that the results could vary design to design and plant to plant. Dr. Wallis asked where the dose on the staff's proposed F-C curve is measured. Mr. Lehner explained that the dose is measured at the exclusion area boundary up to 100 rem. Above 100 rem, the dose is measured one mile from the exclusion area boundary. Dr. Wallis asked if an energetic release, that threw the radioactivity well beyond 1 mile from the exclusion area boundary, would be acceptable? Mr. King said that the implementing regulatory guide directs licensees to assume a ground level plume. Dr. Apostolakis noted that the frequency on the staff's F-C curve not the frequency of the dose, it is the frequency of events that do not meet the acceptance criteria. He also noted that the frequencies on the curve reflect redundancy but they don't reflect the margin from exceeding the acceptance criteria. Dr. Apostolakis explained that the NSSS vendor establishes the T-H success criteria an then the PRA assesses the frequency of not meeting the success criteria (i.e., not the frequency of exceeding a T-H regulatory limit like 2200°F). Dr. Wallis expressed his opinion that the PRA could be expanded to consider the margin to regulatory T-H regulatory limits. Dr. Apostolakis agreed that it would be beneficial to define the PRA success criteria in terms of the regulatory limit(s) as opposed to some intermediate state like two out of three trains available. Dr. Kress reiterated his opinion that the consequences should be in terms of radioactive material released and not some intermediate point like a regulatory T-H limit or core damage. Mr. Maynard said that he thinks it is much easier to deal with issues that unexpectedly come up e.g., construction deficiency) when margins, such as operational margin, design margin, and regulatory margin, are segregated out rather than trying to quantify them in a PRA. Ms. Drouin suggested that the subcommittee focus on the overall framework, and not get bogged down in the details of the proposed F-C curve and licensing basis event selection.

Process for Development of Technology-Neutral or Design-Specific Requirements

Ms. Drouin described the top-down process for developing technology-neutral or design-specific requirements. It begins with the five protective strategies (i.e., Physical Protection, Stable Operation, Protective Systems, Barrier Integrity, and Protective Actions). Logic diagrams or fault trees are then developed based on questions related to identification of what needs to be done to accomplish each strategy. What thing could challenge each protective strategy? Ms. Drouin explained that the framework does not dictate how each protective strategy is to be satisfied. The framework uses this deductive logic to generate requirements for design, construction, and operation related to each protective strategy, requirements Ms. Drouin said should be performance-based as much as practicable. Dr. Wallis and Dr. Apostolakis both expressed concern that the guidance for developing performance-based requirements was too prescriptive. Ms. Drouin said that the F-C curve may help one figure out how to write particular requirements but stressed that it is the protective strategies and logic diagrams that identify the need for requirements in certain areas.

Mr. Lehner explained [referring to Section 6.4.2.2] that there are aspects of the framework, other than PRA, to preserve defense-in-depth. For example, he said that for frequent events (greater than or equal to 1E-2) there should be no barrier failure. For the infrequent event range there should be at least one barrier remaining. Mr. King explained that these additional defense-in-depth requirements apply to all sequences with a frequency of say 1E-2 or greater i.e., in addition to meeting the proposed F-C dose curve. Mr. Mrowca said that an example would be a steam generator tube failure sequence that is greater than 1E-2/yr and which meets the F-C curve. However, it essentially has a barrier failure. Therefore, it would not meet the additional (deterministic) defense-in-depth criteria. So that sequence would have to be modified such that it would have zero barrier failures or it would have to be reduced in frequency to a range where it is allowed to sustain a barrier failure. Dr. Wallis questioned why additional defense-in-depth was being required for the more frequent events. He suggested that you should want more defense-in-depth against the big events that are harder to predict. Ms. Drouin said that defense-in-depth is designed into the plant to protect against uncertainties. She referred to Table 8-2 (page 8-7) of the framework document to explain how each of the six defense-in-depth principles is addressed or implemented for each protective strategy. She stressed that defense-in-depth is implemented independent of the proposed F-C curve. She said that the framework required containment functional capability and not necessarily an actual containment structure. Dr. Kress observed that the defense-in-depth criteria described in the framework document are independent deterministic design criteria, independent of the PRA accident sequence frequencies and the proposed F-C curve.

Dr. Shack asked the staff why they didn't incorporate a CCDF curve into the framework. He said that the framework appeared to be built on criteria that has already been accepted in regulatory space (i.e., to define the proposed F-C curve break points). He also said he thought the staff may have wanted to duck the issue because the NRC doesn't regulate to a cumulative risk limit today. Mr. King said the original reason was that the framework calculates the QHOs, and that takes care of the cumulative effect.

Dr. Kress questioned why the prosed F-C curve was not replaced with a straight line. With a slope of approximately negative one, the curve shows a risk-aversion towards the end of the curve. The subcommittee and staff discussed the use of a CCDF F-C curve versus use of the QHOs. Dr. Kress expressed concern over using the QHOs because he said they are site-related characteristic. He also said that a plant can be designed to meet the QHOs, but then questioned how one would address a multiple-unit site. Dr. Kress suggested using a CCDF F-C curve, equivalent to a CDF of 10^{-5} and LERF of 10^{-6}, as a design requirement for fission product release. He added that then you could put 10 plants on a site and still meet the QHOs. Dr. Apostolakis expressed concern over making the design curve overly conservative. He favored having separate siting requirements (expanded QHOs that perhaps address societal things). Dr. Apostolakis said that for the Committee to make an informed decision on the framework it needs to know, in detail, what other organizations have proposed and how they differ with the staff. The subcommittee discussed the need for writing a Committee letter on the framework (i.e., to respond to the November 8, 2006 SRM, to provide the staff with its comments on technical/policy aspects of the framework). Ms. Drouin said that public comments on the framework were generally favorable, at the conceptual level, but most indicated that they wanted to see it tested before the staff proceeded with rulemaking. She said that the staff did a very limited test of the framework on an existing LWR and that the July version of the framework reflects that test. She said that a more complete test of how the framework translates into actual requirements will be published in the version of the framework they plan on issuing early this summer. The version to be published early this summer would also contain an appendix that summarized all the stakeholder comments. The subcommittee discussed the possibility of testing the framework on an actual plant (e.g., PBMR, NGNP). Mr. Ruben, part of an inter-office team looking at the licensing strategy for the NGNP, described several licensing approaches being considered by the staff, each with varying degrees of the use of PRA information. The spectrum of options includes 1) using a completely deterministic approach as was done for Fort St. Vrain, 2) using a deterministic approach to establish DBEs and related requirements supplemented by PRA insights, 3) using a risk-derived approach that uses the guidance in the framework document to adapt existing requirements, and 4) an approach that is based on new regulation derived from the framework. The first three approaches would involve adapting existing Parts 50 and 52 requirements and issuance of exemptions as needed to license an HTGR. Dr. Abdel-Khalik suggested that the framework could be published as a Standard Review Plan as opposed to as a NUREG report. Dr. Apostolakis noted that if the staff wanted a decision for moving forward in a reasonable amount of time, the staff would need to go with the second option i.e., use existing criteria/regulations supplemented by risk insights. Mr. Ruben responded that the industry may not be enthusiastic about the staff taking that option, at least for HTGRs. He said he thought the HTGR, VHTGR, and the NGNP would probably prefer a risk-derived approach, that is, an approach that relied more heavily on PRA and risk insights to select events, identify safety-related equipment, establish defense-in-depth requirements, etc. The Subcommittee and staff discussed options for moving forward with the framework, how to test the framework, what plant design to test it on, and what regulatory mechanism one might use to implement the framework concepts.

Dr. Powers questioned why some ACRS members felt like the Committee should design the CCDF F-C curve for ten plants on a site. Dr. Kress explained that if he had a design that was required to meet an F-C requirement of a CDF of 10^{-4} and a LERF of 10^{-5} (which are derived from the QHOs), then they could only put one plant of that design on the site and still meet the QHOs. However, if they required the design to meet a lower F-C requirement, for example a CDF of 10^{-5} and a LERF of 10^{-6}, they might be able to put more plants of that design on the site and still meet the QHOs. Dr. Powers argued that the NRC should not be concerned with how may plants an applicant proposes to put on a plant site. He said that that is an economic decision. Mr. King noted that the subject of an acceptable level of integrated risk from a plant site is a policy issue that the Commission has yet to decide. The subcommittee and staff discussed various alternatives for dealing with integrated risk at sites with existing nuclear power plants. Dr. Kress said that there were several sites that he thought should not be candidates for additional nuclear power plants, even though a new plant may add an insignificant amount of additional risk. He said that to identify those sites would require the consideration of societal risk.

Dr. Powers asked the staff why it thinks it needs defense-in-depth, not so much at the containment level but at the lower levels. He asked if it was to address uncertainties or is it to cover unforseen issues or things that are modeled incorrectly. He also asked the staff how much defense-in-depth is enough. Ms. Drouin explained that defense-in-depth is there because of uncertainties, not so much the PRA data uncertainties as the state-of-knowledge uncertainties, things you don't know or don't model well. She said that basing the proposed requirements on successive protective strategies provides for some defense-in-depth. In addition, deterministic defense-in-depth criteria are imposed on LBEs dependant on their frequency of occurrence (reference Section 6.4.2.2 of draft NUREG-1860).

Dr. Powers asked Ms. Drouin how she saw Appendix B QA/QC in the mix of defense-in-depth and risk information. Mr. King said that QA is a good engineering practice that applies across the board and was not part of the framework's defense-in-depth strategy.

Dr. Wallis said that if he knew where the design was on a CCDF F-C curve he would understand how safe it is. He then said that saying they have put a lot of defense-in-depth into the design doesn't tell him anything about how safe the design is. Ms. Drouin noted that in addition to including defense-in-depth in the design, the framework requires that the design meets the QHOs. Dr. Apostolakis said that a plant isn't safe because it meets some number that comes out of the PRA. He said that it is presumptively safe (or provides adequate public protection) because it has gone through the licensing process, based on a number of factors (e.g., analyses, regulations, defense-in-depth, safety margins, etc.).

Dr. Apostolakis asked if the framework addressed the treatment of structures, systems, and components (SSCs). Ms. Drouin explained that any SSC relied on to meet the F-C curve, or whatever quantitative goal that is ultimately established, would be classified as safety significant by the framework. She added that the treatment applied to the safety significant SSCs would be a function how safety or risk significant they were. Mr. Lehner clarified that the special treatment is supposed to reflect the fact that the SSC is reliable under the conditions that you took credit for when you did the PRA. Dr. Wallis question why the staff felt compelled to classify SSCs at all, given the fact that they would be required to meet some F-C curve. He said that plant designers and managers will know that certain things are significant for safety and presumably they will take care of them. Dr. Apostolakis said that we need to classify the SSCs in part because we don't trust the licensees to take adequate care of them. Dr. Bonaca and Mr. Maynard said they did not think it was a matter of trust. Rather, they thought it was the regulator's responsibility to define the appropriate level of treatment. Dr. Kress suggested that importance measures, as opposed to LBE, be used to classify SSCs. Mr. Maynard suggested that the treatment requirements be more performance-based rather than being prescriptive based on their classification. He added however that if you credit an SSC in your PRA, you should do something to provide assurance that it has the reliability you assumed in the PRA. Ms. Drouin said that the framework calls for the use of risk importance measures to determine the treatment that should be applied to SSCs. Dr. Wallis suggested that having a living PRA should identify components whose reliability have degraded. Ms. Drouin challenged Dr. Wallis' supposition.

The subcommittee discussed with the staff plans for moving forward with the technology-neutral framework. Dr. Wallis said that he can't imagine licensing new reactors without some framework or structure from which to start. Ms. Drouin indicated that it was clear from the public received on draft NUREG-1860 and the associated ANPR, that stakeholders did not want the NRC to proceed with rulemaking at this time. She said the main question at this juncture is whether the staff approaches new reactor licensing from a deterministic footing or from a PRA perspective. Dr. Powers and Dr. Apostolakis indicated that new reactor licensing would necessarily start from a deterministic footing and that the question is how to best incorporate PRA insights (i.e., a risk-informed as opposed to a risk-derived approach).

Dr. Corradini asked if the technology-specific licensing strategy for the NGNP could be done in parallel with further development and testing of the technology-neutral framework. Mr. Monninger acknowledged the progress the staff has made on the technology-neutral framework and said that the staff had not reached any conclusions with regard to how best to proceed with the framework. He questioned whether it would be better to pursue resolution of some of the key policy issues on a technology-neutral basis, without any specific application in mind, or would testing the framework on a specific design be more productive? He said the staff is working with DOE to develop an NGNP licensing strategy and at the same time having interactions with PBMR on some high-level policy white papers. Mr. Monninger also mentioned the Global Nuclear Energy Project (GNEP). He asked whether the staff should pursue these other programs and see the extent to which the framework can contribute to them? Ms. Drouin said the framework was developed as the technical basis for a rule that would resolve several key policy issues and provide a efficient and effective regulatory structure for advanced non-LWR reactor designs. She said that the staff planned to make the rule technology-neutral and planned to put design-specific guidance in regulatory guides. She said that integrating risk into the framework from the ground up was a bigger challenge than developing the technoloy-neutral regulatory framework. Ms. Drouin said the framework document was ready to be published. Dr. Kress recommended against publishing the framework just yet, because he said a few items needed to be straightened out and it needed polishing. Ms. Drouin suggested that the next step is to develop an implementing guidance document, perhaps for the PBMR, which she thought could be done in a year. Both Ms. Drouin and Dr. Kress agreed with the stakeholder comments that it would be premature to proceed with rulemaking at this time. Dr. Wallis suggested comparing the results (regulatory requirements) derived from applying the framework to the PBMR design with the results obtained without the use of this framework. Mr. Maynard said the technology neutral framework is a useful process, whether you use the existing regulations and use this process for where you take exemptions, or whether you develop different rules for each technology. He said it is the framework by which you start making the decision. Dr. Corradini suggested that the Committee recommend that the staff continue with the development of the technology-neutral framework, because this is the fundamental underpinning, but also apply the framework in a pragmatic way using the current rules to a non-LWR design (i.e., having no choice but to do the latter given time constraints). Dr. Kress said he agreed with this approach and suggested that the design-specific implementing guidance be developed for the PBMR since the NRC has the relevant inputs for that right now in the form of white papers on PRA, Licensing Basis Events, SSC Classification, etc. Dr. Kress said that in his mind the key thing missing from the framework document is the CCDF F-C curve. He said that such a curve needs to be included in the framework for coherence in the regulatory process. Dr. Bonaca agreed that the document should be complete before it is published, even if there is a plan to issue more detailed implementation guidance. Dr. Powers said that the Fort St. Vrain and the Fast Flux Test Facility reviews were done on an ad hoc (but not entirely capricious) basis. He doubted the Commission would want to face the public confusion that would come about from an ad hoc review of new reactor designs. He acknowledged that time schedules are pressing but added "there is always time to do things over, there is never time to do it right." Dr. Powers said he would publish the framework document with just the minor editing as proposed by the staff. He said that completing the framework would be an iterative process, learning as the framework is implemented for various reactor designs. Dr. Apostolakis agreed that the framework should be polished as much as practicable and then published. Dr. Wallis said polishing was fine but agreed with Dr. Powers that something other than an ad hoc approach for the review of new reactor designs is needed. He called the framework document a good first draft but admitted there are ways to improve it. He expressed concern that the staff might stop working on it. The subcommittee and staff discussed the benefits that might be derived from pilot testing the framework on, for example, PBMR. Dr. Abdel-Khalik suggested that the framework be piloted in the development of a regulatory guide to support licensing under Part 50. Mr. Ruben suggested that the staff could take the technology-neutral framework approach and exercise it with the PBMR design and PRA to see what they come up with in terms of design-basis accidents, safety related systems, and defense-in-depth. Then they could compare those results to the results PBMR (Pty) Ltd. obtained in their white papers on LBE Selection, SSC Classification, and Defense-in-Depth Approach. This effort would be separate from the review of the PBMR design certification application.

Dr. Apostolakis acknowledged that the staff did not intend to write SERs on the PBMR white papers but asked if the Committee was going to be made aware of the staff's comments back to PBMR (Pty) Ltd. on them? Mr. Ruben indicated that he thought the staff would be bring its assessment of the PBMR white papers to the Committee for its review and possible comments.

The subcommittee had some preliminary discussion regarding whether or not the Committee would prepare a letter on the technology-neutral framework during the March ACRS full Committee meeting.

Mr. Ed Burns, Licensing Manager for PBMR, briefly described each of the four PBMR white papers that have been submitted to the staff as part of the PBMR pre-application process. He said that PBMR would be licensed under Parts 50 and 52. He said the white papers deal with technical issues that are new to the PBMR design. While the papers are not focused on the technology-neutral framework, he said that the elements of what is inside the framework will be useful.

Dr. Abdel-Khalik complimented the staff on the technology-neutral framework document and expressed his concern that work on the framework might stop. In order to make the document a truly worthwhile document, he suggested that the issues and concerns raised during the subcommittee meeting should be resolved before it is published. He also said that he thought the framework should be pilot tested, comparing whatever you are going to get with the process that is being contemplated for the gas cooled reactors.

Dr. Wallis also complimented the staff on the technology-neutral framework document and said it was a good first step. However, he acknowledged that there were ways to improve it and said the ACRS needs to mull this over probably for a year or more. He said that this is not a one meeting, one letter type of an issue and indicated that the framework will evolve over time.

Similarly, Dr. Apostolakis commented favorably on the document and expressed his concern that work on the framework might stop. He supported testing the framework on the PBMR design, since there seems to be some time without the pressure of an actual application. Dr. Kress said he thought that would be a good idea too and complimented the staff on job well done.

Ms. Drouin said she appreciated the subcommittee's comments and thought the meeting was productive. She said the framework deals with some very complex issues and acknowledged that it is evolving. She also thought it should be published, tested, and then adjusted accordingly. Finally, she recognized the staff and contractors who helped her develop the framework. Mr Monninger asked for as much early feedback as possible from the ACRS on the framework. He said the staff appreciated the ACRS's views on the need for further development of the framework and look forward to further interactions with the Committee.

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SUMMARY / PLANS FOR FULL COMMITTEE PRESENTATION

The Subcommittee suggested that the staff address the following topics during the March Full Committee meeting:

- A high-level discussion of the stakeholder comments
- The staff's plans for moving forward with the technology-neutral framework

Agreements

None.

Staff/Industry Follow-up Actions

The staff plans to provide a briefing on the technology-neutral licensing framework to the full Committee during the March 8-10, 2007, ACRS meeting.

Subcommittee's Action

The subcommittee plans to provide the full Committee with proposed comments on the technology-neutral framework during the March 8-10, 2007, ACRS meeting.

Documents Provided to the Subcommittee

None

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NOTE : Additional details of this meeting can be obtained from a transcript of this meeting available for downloading or viewing on the Internet at http://www.nrc.gov/reading-rm/adams.html or http://www.nrc.gov/reading-rm/doc-collections/ can be purchased from Neal R. Gross and Co., 1323 Rhode Island Ave., N.W., Washington, DC 20005 (202) 234-4433.

UNITED STATES
NUCLEAR REGULATORY COMMISSION
ADVISORY COMMITTEE ON REACTOR SAFEGUARDS
WASHINGTON, DC 20555 - 0001

March 28, 2007

MEMORANDUM TO: Thomas Kress, Chairman
Future Plant Designs Subcommittee

FROM: David C. Fischer, Senior Staff Engineer
Technical Support Staff
ACRS/ACNW

SUBJECT: WORKING COPY OF THE MINUTES OF THE ACRS FUTURE PLANT DESIGNS SUBCOMMITTEE MEETING ON THE TECHNOLOGY NEUTRAL LICENSING FRAMEWORK (WORKING DRAFT NUREG-1860) MARCH 7, 2007, ROCKVILLE, MARYLAND

A working copy of the minutes of the subject meeting is attached for your review.

Please review and comment on them at your earliest convenience. If you are satisfied with these minutes please sign, date and return the attached certification letter.

Attachment: Minutes (Working Draft)
Certification Letter

cc: ACRS Members (attendees)

cc w/o Attachment:
F. Gillespie
C. Santos
S. Duraiswamy

UNITED STATES
NUCLEAR REGULATORY COMMISSION
ADVISORY COMMITTEE ON REACTOR SAFEGUARDS
WASHINGTON, DC 20555 - 0001

Working Draft

Issued: March 28, 2007

ADVISORY COMMITTEE ON REACTOR SAFEGUARDS FUTURE PLANT DESIGNS SUBCOMMITTEE MEETING MINUTES MARCH 7, 2007 ROCKVILLE, MARYLAND

INTRODUCTION

The ACRS Subcommittee on Future Plant Designs met on March 7, 2007, at 11545 Rockville Pike, Rockville, Maryland, in Room T-2B3. The purpose of this meeting was to review the NRC staff's work on technology neutral licensing framework (i.e., Working Draft NUREG-1860) with a focus on ensuring the value of such an approach versus the development of a licensing framework for specific designs, such as a high temperature gas cooled reactor or a liquid metal cooled reactor (reference the Commission's November 8, 2006, Staff Requirements Memorandum to Dr. Larkins). During the briefing, the Committee also explored with the NRC staff the pros and cons of developing a licensing framework for specific designs. The Subcommittee gathered information, analyzed relevant issues and facts, and formulated proposed positions and actions, as appropriate, for deliberation by the full Committee. The entire meeting was open to public attendance. Mr. David C. Fischer was the cognizant staff engineer and the Designated Federal Official for this meeting. The Subcommittee received no written comments, or requests for time to make oral statements from any members of the public regarding this meeting. The meeting was convened at 10:00 am and adjourned at 4:21 pm.

ATTENDEES

ACRS

Thomas S. Kress, Chairman
Said Abdel-Khalik, Member
George E. Apostolakis, Member
Mario V. Bonaca, Member
Michael Corradini, Member
Otto L. Maynard, Member
Dana A. Powers, Member
William J. Shack, Member
Graham B. Wallis, Member
David C. Fischer, ACRS Staff

NRC

Sud Basu, RES/DRASP/NRCA
Ben Beasley, RES/DRASP/PRA
David Bessette, RES/DRASP/NRCA
Joe Birmingham, NRR/DPR
Mary Drouin, RES/DRASP/PRA
Don Dube, NRR/DRA
Lauren Killian, RES/DRASP/PRA
Eileen McKenna, NRR/DPR
Lynn Mrowca, NRO/DSRA
Yuri Orecheva, NRR/DSS
Stuart Rubin, RES/DRASP/NRCA
Rajendra Solanki, NRR/DRA/APLA

Ronaldo Jenkins, RES/DRASP/PRA
Thomas Kenyon, NRO/DNRL/MWB
Martin Stutzke, NRR/DRA/APLA

OTHERS

Biff Bradley, NEI
Edward Burns, PBMR
Shawnie Harris, CH2M Hill
Tom King, ISL, Inc.
Alan Levin, AREVA
John Lehner, BNL
Bruce Mrowca, ISL
Vinod Mubayi, BNL
Patrick O'Regan, EPRI

A complete list of attendees is in the ACRS Office file and will be made available upon request. The presentation slides and handouts used during the meeting are attached to the Office Copy of these minutes.

OPENING REMARKS BY THE SUBCOMMITTEE CHAIRMAN

Dr. Thomas S. Kress, Chairman of the Future Plant Designs Subcommittee, stated that the purpose of this meeting was to review the NRC staff's work on technology neutral licensing framework (i.e., Working Draft NUREG-1860) with a focus on ensuring the value of such an approach versus the development of a licensing framework for specific designs, such as a high temperature gas cooled reactor or a liquid metal cooled reactor.

STAFF PRESENTATIONS

Introductory Remarks:

Ms. Mary Drouin (RES), Mr. Tom King (ISL), Mr. Marty Stutzke (NRR/NRO), and Mr. John Lehner (BNL) were seated at the head table with the subcommittee.

Mr. John Monninger (RES) introduced the staff's "Framework for Future Plant Licensing," oftentimes referred to as the "technology-neutral framework." He said that the framework has been under development for the past three years and that it benefitted from numerous meetings with stakeholders and workshops. He said that the framework has gained importance with the passage of the Energy Policy Act of 2005 and that the staff planned to use the lessons learned during the development of the framework to formulate the Next Generation Nuclear Plant (NGNP) licensing strategy.

Ms. Mary Drouin said that in January 2003 the RES Advanced Reactor Research Plan recognized the need for a licensing framework for advanced reactors. She said that the current regulatory structure focused on light-water reactors (LWRs) and had limited applicability to non-LWR technologies. She noted that some of the specific requirements in current regulations are not applicable to advanced reactor designs and added that advanced reactors will likely have design and operational issues different than LWRs. She said that the current regulatory structure has evolved with only limited insights from PRAs and severe accident research and suggested that PRA and PRA insights should be an integral part of licensing advanced reactors. In 2003 a program was initiated to develop a risk-informed, performance-based regulatory structure to support future reactor licensing.

The initial technology-neutral licensing "framework" is documented in draft NUREG1860 which, Ms Drouin said is ready to be published. Ms. Drouin said the framework provides guidance and criteria for creating a "risk-derived" and performance-based regulatory structure that can be implemented on either a technology-neutral or a technology-specific basis. She said the framework integrates Commission expectations as addressed in various policy statements, such as the policy statements on severe accidents advanced reactors, PRA, and safety goals. Ms. Drouin said the framework was attached to an Advanced Notice of Proposed Rulemaking (ANPR) published in May 2006 for public review and comment. Public workshops were held on the framework in March 2005 and in September 2006. She said that two different versions of the framework were put on the NRC's public website which resulted in receiving two sets of comments from some organizations. Ms. Drouin identified the stakeholders who submitted comments on the ANPR and highlighted three general areas where comments were received (i.e., overall views, technology-neutral versus technology-specific, and how to proceed forward). She said that most comments received were generally supportive of moving forward with the development of a risk-informed performance-based regulatory framework for new reactors. However, she said that one comment was received that the framework "departs too far from the approximately 3000 reactor years experience gained using the deterministic approach," particularly as it relates to addressing common cause failure. Ms. Drouin said that some comments supported technology-neutral regulations with technology-specific implementing guidance, some supported technology-specific regulations, and others (e.g., NEI) indicated that it was too premature to decide. She said that there was not a consensus on this issue. With regard to how to proceed with rulemaking, Ms. Drouin said there seemed to be a consensus <u>not</u> to go to rulemaking in the near term. She identified three options for proceeding forward. The first was to gain experience with design certification of a non-LWR design using the framework approach. The second was a multi-year phased approach to rulemaking. And the third was a stepped approach in which the staff would develop a preliminary draft rule to be published for information upon receipt of a non-LWR application. The staff would then review and approve the non-LWR design using the current Part 50 and Part 52 regulations. Then the staff would evaluate the draft rule against the non-LWR design before publishing the proposed rule for public comment.

Ms. Drouin said that the staff planned to publish the framework (i.e., NUREG-1860) in the early summer (June) 2007. She said the staff is preparing a SECY paper to respond to Commission direction to "provide its [staff] recommendation on whether and, if so, how to proceed with rulemaking." She said that all activities related to the framework were going to be terminated. Dr. Kress express concern that work on the framework was going to be stopped. He said that some work still needed to be done to fine tune the framework. Ms. Drouin said that the staff is evaluating the need to defer rulemaking until experience is gained with NGNP and GNEP. She said that the staff is planning on briefing the ACRS on its plan for moving forward with rulemaking at the May full Committee meeting.

Dr. Wallis asked Ms. Drouin if the output from the PRA is core damage frequency (CDF) or is it a more comprehensive assessment of the effects on the public. Ms. Drouin said that when she uses the word PRA, she is using it in its entirety, not just CDF. Dr. Abdel-Khalik expressed concern that using the framework approach could lead to getting around certain Part 50 requirements, resulting in less stringent requirements for a plant. Ms. Drouin said that using the framework might lead to a different set of requirement but added that the requirements generated by using the framework should result in a safer plant (than by using existing deterministic requirements).

Framework Overview:

Ms. Drouin explained that the "framework" is a technical report (NUREG) that provides a structured and systematic approach in the form of guidelines and criteria for developing new requirements. The framework itself is not regulation but could serve as the basis for rulemaking (e.g., a new Part 53, exemptions or additions to Part 50). She said the framework uses a "risk-derived approach and then explained the difference between a risk-derived and a risk-informed approach. A risk-derived approach starts with PRA results and integrates deterministic and defense-in-depth criteria (to compensate for uncertainties) as an integral part in the development of requirements. A risk-informed approach uses deterministic criteria to develop the requirements and then supplements them with risk insights. Ms. Drouin said that the framework can be applied or implemented (i.e., the development of requirements) on either a technology-neutral or on a technology-specific basis. The framework can be used to generate a risk-derived set of design, maintenance, and operating regulations or regulatory guidance. Dr. Kress questioned how long it would take to go from the framework to final rulemaking. Ms. Drouin said that she thought it could be completed in 5 years. Next, Ms. Drouin showed examples of regulations that might be generated by using the framework. Dr. Powers observed that the example regulation on Energetic Reaction Control (i.e., Reactor designs thathave the potential for energetic reactions between the fuel, coolant or other material shall include provisions to prevent or mitigate the effects of suc reactions.) did not tell the licensee what it is supposed to achieve. He said that without telling a licensee what it is supposed to achieve, almost anything could satisfy the requirement. Dr. Powers called such a requirement (unlike 10 CFR 50.46 which places specific limits on maximum cladding oxidation and maximum hydrogen generation) "non-functional."

Ms. Drouin reiterated that the framework describes a top-down process for developing a risk-derived, performance-based set of regulations that can be applied to any reactor technology. The process should define a goal (i.e., a defined measure of performance such as a quantitative measure of public health and safety) and the guidelines and criteria for achieving the goal. She said the process must also deal with completeness. The framework identifies high-level goals to protect public health and safety. The framework uses the protective strategies needed to ensure public health and safety as the structure to identify the requirements. Then the framework defines the guidelines and criteria to meet the overall goal within this structure (based on PRA and defense-in-depth guidance in the framework). By implementing these guidelines and criteria one can identify design, maintenance and operational requirements (on either a technology-neutral or technology-specific basis). Ms. Drouin said that the framework document contained guidance and criteria on the probabilistic approach, defense-in-depth, PRA technical acceptability, and how to identify requirements.

Round Table Discussion of Key Issues:

F-C Curve and Licensing Basis Event Selection

The subcommittee spent a considerable amount of time discussing the frequency-consequence (F-C) curve contained in draft NUREG-1860 (i.e., Figure 6-3).

Stakeholder Comments
- Different anchor points
- Add a CCDF curve

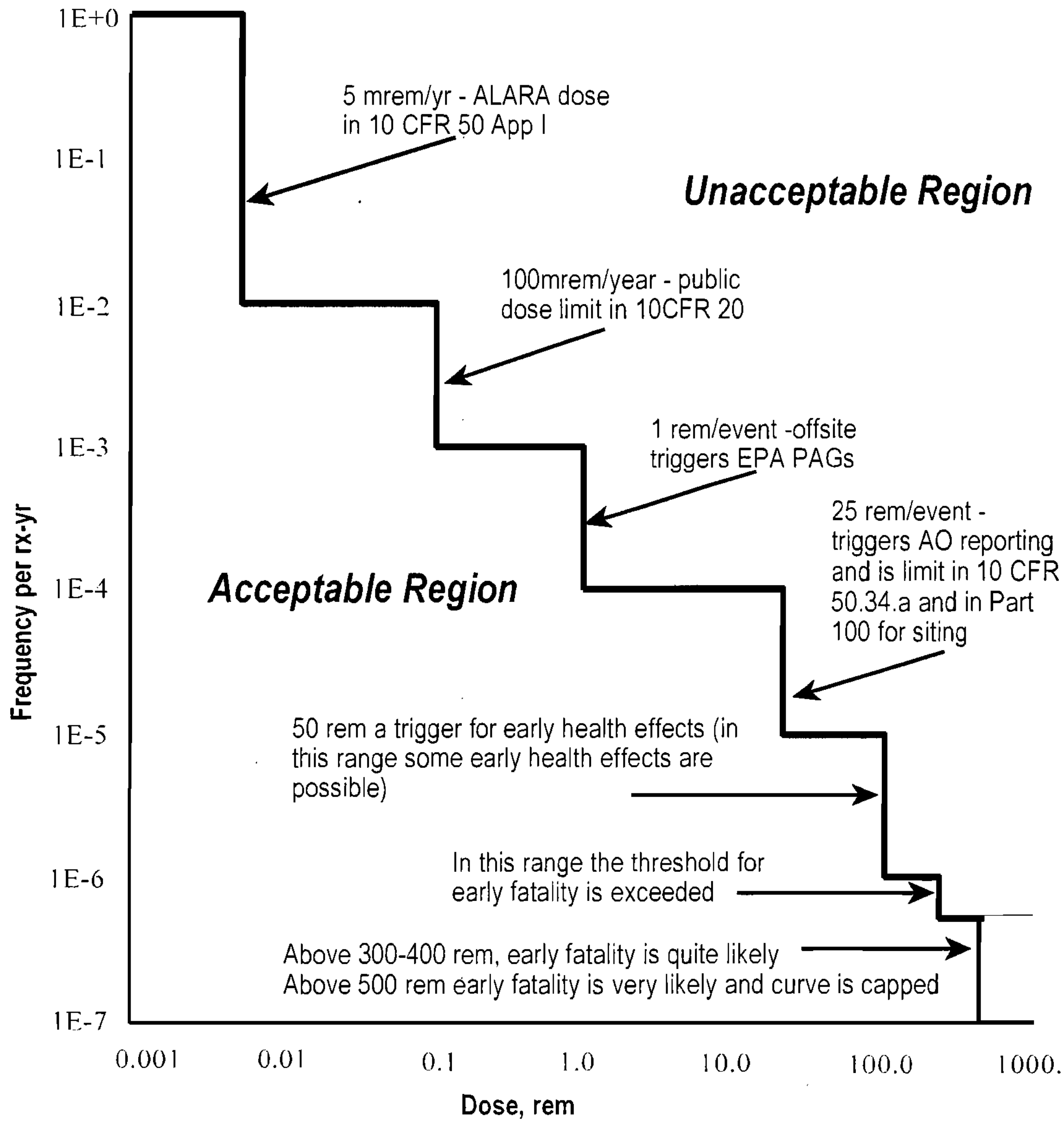

Figure 6-3 Frequency-consequence curve.

Dr. Kress commented that there was no reason why the curve should be stair-stepped. He said it could be a non-risk averse straight line. Second, he said that he would call it an F-C curve for identifying licensing basis events as opposed to a risk acceptance curve. Dr. Kress agreed that the curve should address different accident types and frequency ranges but said he would select the sequences with the maximum F-C product, not maximum frequency or maximum consequence. Dr. Kress observed that the curve does not summate the risk from all sequences. He said the framework needed a complimentary cumulative distribution function curve. Finally, Dr. Kress said he thought the consequences should be measured in curies as opposed to dose (i.e., as calculating dose would require some knowledge of site characteristics and use of a Level 3 PRA whereas calculating curies would only require a Level 2 PRA). Dr. Wallis agreed with Dr. Kress that the proposed curve did not measure the cumulative impact on the public from all possible events. He also questioned the need for establishing licensing basis events. Dr. Powers agreed that the proposed curve should be called a licensing basis event curve, or something other than an F-C curve. Dr. Wallis also agreed with Dr. Kress that the curve should be a straight line. Dr. Bonaca asked if the prosed curve was developed by leveraging existing regulations and criteria. Dr. Kress agreed with Dr. Bonaca that the proposed curve seemed to be derived from existing regulation and therefore, he said, was not responsive to the Commission's expectation that new reactor designs have a higher level of safety than existing nuclear power plants. Dr. Bonaca questioned the need for defining licensing basis events, given the fact that the applicant will have a PRA, but acknowledged the benefits of having some limiting events to help anchor plant operations, technical specifications, identify SSCs for special treatment, etc. Mr. Lehner (BNL) said they tried to base the curve on current regulations which are generally in terms of consequence, but not frequency. He said they used the dose limits in the regulations, and qualitative estimates of the frequency where these dose limits would apply, to develop the frequency ranges. He said they had thought about constructing a straight line but thought they would have difficulty justifying intermediate points. Dr. Wallis questioned why a continuous release of 1mrem would be allowable. Mr. Lehner said that the doses up to 100 mrem are cumulative. Above 100 mrem the doses are individual events or sequences. Mr. King (ISL) noted that in addition to meeting the proposed curve the framework also requires an analysis to show that the design also meets the QHOs. Mr. Lehner noted that designers and reviewers can add design basis accidents (DBAs). The staff and subcommittee discussed the need for caution when aggregating event sequences into classes to avoid slicing it so thin that each class has a very low frequency of occurrence and therefore an acceptable level of risk. Mr Lehner said that one reason why you want licensing basis events is that you don't want it to be totally risk-based and acknowledged the benefits previously cited by Dr. Bonaca (e.g., stability in the parameters to which the operators control the plant). Dr. Kress said that having licensing basis events provides some defense-in-depth in that it forces the designer to consider a wide range of events including those that are not very risk significance. Mr. King said that considering licensing basis events is conservative and is not risk-based. He noted that identifying licensing basis events is necessary to implement certain other regulations like Part 20 and Part 100 and is used for safety classification, and to test the PRA, and to put margin in the design for defense-in-depth. Dr. Powers said that he did not think that a F-C curve needed to be constructed to reflect the Commission's desire that new plants be safer than existing plants. He also agreed with Dr. Wallis that DBAs are a dangerous concept because, he explained, plants are then designed to the DBAs rather than being designed to the risk. He clarified that he was, however, okay with identifying types of accidents to look at. Dr. Powers said that new plants are coming in with CDFs or events that are exceptionally low. Therefore, the risk is going to be dominated by those things that the PRA treats very poorly, such as aging, defects in construction, and external events. He added that he did not think the regulatory system needed to set lines for the operators. Ms. Drouin emphasized that using a risk-derived approach, as described in the framework, would require a PRA fundamentally better than those that are in use today. But she said that decisions would not be risk-based. Dr. Powers criticized the staff for over reliance on the PRA end points, like CDF and LERF, rather than gaining insights from the PRA. Mr King said that the framework uses defense-in-depth to address these completeness issues. Mr Lehner said that for these new technologies, the PRA is a tool for trying to discover new threats and combinations that you would not have otherwise thought of, a systematic way of looking for unique accident situations. Mr. Maynard said he preferred the stair-step F-C curve because it ties things together in a way that makes sense, but indicated that a straight line would be fine by him as well. He did not like the idea of relying totally on the PRA and favored defining DBEs to facilitate the development of processes and procedures to be used in the plant. Dr. Wallis suggested that DBAs would not be necessary if the PRA adequately modeled thermal-hydraulic phenomena (as opposed to just using deterministic success criteria). Dr. Apostolakis indicated that it was impractical to do thermal-hydraulic analyses for as many sequences as are in a PRA and, therefore, he thought it would be useful to identify bounding event sequences. Dr. Abdel-Khalik viewed the staff's proposed F-C curve as a way of identifying limiting event sequences which would be analyzed in greater detail. He said that other than meeting some cumulative risk criteria, the framework should not accept a plant design in which events of relatively high consequence would have probabilities of exceeding a certain value. Dr. Abdel-Khalik indicated that he liked the proposed F-C curve as a design tool because it is tied to current regulation but criticized it because it did not separate design acceptance from site acceptance. He also did not favor changing the unit for consequence from dose to curies because, he said, a curie of tritium is different than a curie of polonium. Dr. Kress suggested that this could be accounted for by converting various fission products to a standard using weighting factors. Dr. Apostolakis the three region approach (i.e., desirable, tolerable, and unacceptable regions) was used in the staff's proposed F-C curve. Ms. Drouin responded that the three region approach is not used in LBE selection. Mr. Maynard explained that the three region approach was part of the defense-in-depth philosophy for the safety, security, and preparedness expectation. Dr. Apostolakis argued against having hard and fast lines between the acceptable and unacceptable regions on the staff's proposed F-C curve. He said that having such lines of demarcation is impractical. He did not advocate establishing firm requirements (e.g., on reliability) or relying solely on the results of the PRA. He favored an approach similar to that used in RG 1.174 where there is increased management attention the closer you get to a particular value. Dr. Apostolakis suggested that the framework be pilot tested. Drs. Kress, Wallis, Abdel-Khalik, and Mr. Maynard all seemed to favor having "bright lines" separate the acceptable and unacceptable regions on the F-C curve. Dr. Corradini asked if the licensing basis events derived by using the framework would differ from the design basis accidents developed using a deterministic approach (e.g., for Clinch River or Fort St. Vrain). Mr. Mrowca (ISL) said that while the two different approaches would produce generally consistent results, there would also be differences (i.e., some additional sequences and some that would be out of the scope of the framework) because of the F-C curve or defense-in-depth criteria. Ms. Drouin clarified that the framework outlines a process for determining linensing basis events and that the results could vary design to design and plant to plant. Dr. Wallis asked where the dose on the staff's proposed F-C curve is measured. Mr. Lehner explained that the dose is measured at the exclusion area boundary up to 100 rem. Above 100 rem, the dose is measured one mile from the exclusion area boundary. Dr. Wallis asked if an energetic release, that threw the radioactivity well beyond 1 mile from the exclusion area boundary, would be acceptable? Mr. King said that the implementing regulatory guide directs licensees to assume a ground level plume. Dr. Apostolakis noted that the frequency on the staff's F-C curve not the frequency of the dose, it is the frequency of events that do not meet the acceptance criteria. He also noted that the frequencies on the curve reflect redundancy but they don't reflect the margin from exceeding the acceptance criteria. Dr. Apostolakis explained that the NSSS vendor establishes the T-H success criteria an then the PRA assesses the frequency of not meeting the success criteria (i.e., not the frequency of exceeding a T-H regulatory limit like 2200°F). Dr. Wallis expressed his opinion that the PRA could be expanded to consider the margin to regulatory T-H regulatory limits. Dr. Apostolakis agreed that it would be beneficial to define the PRA success criteria in terms of the regulatory limit(s) as opposed to some intermediate state like two out of three trains available. Dr. Kress reiterated his opinion that the consequences should be in terms of radioactive material released and not some intermediate point like a regulatory T-H limit or core damage. Mr. Maynard said that he thinks it is much easier to deal with issues that unexpectedly come up e.g., construction deficiency) when margins, such as operational margin, design margin, and regulatory margin, are segregated out rather than trying to quantify them in a PRA. Ms. Drouin suggested that the subcommittee focus on the overall framework, and not get bogged down in the details of the proposed F-C curve and licensing basis event selection.

Process for Development of Technology-Neutral or Design-Specific Requirements

Ms. Drouin described the top-down process for developing technology-neutral or design-specific requirements. It begins with the five protective strategies (i.e., Physical Protection, Stable Operation, Protective Systems, Barrier Integrity, and Protective Actions). Logic diagrams or fault trees are then developed based on questions related to identification of what needs to be done to accomplish each strategy. What thing could challenge each protective strategy? Ms. Drouin explained that the framework does not dictate how each protective strategy is to be satisfied. The framework uses this deductive logic to generate requirements for design, construction, and operation related to each protective strategy, requirements Ms. Drouin said should be performance-based as much as practicable. Dr. Wallis and Dr. Apostolakis both expressed concern that the guidance for developing performance-based requirements was too prescriptive. Ms. Drouin said that the F-C curve may help one figure out how to write particular requirements but stressed that it is the protective strategies and logic diagrams that identify the need for requirements in certain areas.

Mr. Lehner explained [referring to Section 6.4.2.2] that there are aspects of the framework, other than PRA, to preserve defense-in-depth. For example, he said that for frequent events (greater than or equal to 1E-2) there should be no barrier failure. For the infrequent event range there should be at least one barrier remaining. Mr. King explained that these additional defense-in-depth requirements apply to all sequences with a frequency of say 1E-2 or greater i.e., in addition to meeting the proposed F-C dose curve. Mr. Mrowca said that an example would be a steam generator tube failure sequence that is greater than 1E-2/yr and which meets the F-C curve. However, it essentially has a barrier failure. Therefore, it would not meet the additional (deterministic) defense-in-depth criteria. So that sequence would have to be modified such that it would have zero barrier failures or it would have to be reduced in frequency to a range where it is allowed to sustain a barrier failure. Dr. Wallis questioned why additional defense-in-depth was being required for the more frequent events. He suggested that you should want more defense-in-depth against the big events that are harder to predict. Ms. Drouin said that defense-in-depth is designed into the plant to protect against uncertainties. She referred to Table 8-2 (page 8-7) of the framework document to explain how each of the six defense-in-depth principles is addressed or implemented for each protective strategy. She stressed that defense-in-depth is implemented independent of the proposed F-C curve. She said that the framework required containment functional capability and not necessarily an actual containment structure. Dr. Kress observed that the defense-in-depth criteria described in the framework document are independent deterministic design criteria, independent of the PRA accident sequence frequencies and the proposed F-C curve.

Dr. Shack asked the staff why they didn't incorporate a CCDF curve into the framework. He said that the framework appeared to be built on criteria that has already been accepted in regulatory space (i.e., to define the proposed F-C curve break points). He also said he thought the staff may have wanted to duck the issue because the NRC doesn't regulate to a cumulative risk limit today. Mr. King said the original reason was that the framework calculates the QHOs, and that takes care of the cumulative effect.

Dr. Kress questioned why the prosed F-C curve was not replaced with a straight line. With a slope of approximately negative one, the curve shows a risk-aversion towards the end of the curve. The subcommittee and staff discussed the use of a CCDF F-C curve versus use of the QHOs. Dr. Kress expressed concern over using the QHOs because he said they are site-related characteristic. He also said that a plant can be designed to meet the QHOs, but then questioned how one would address a multiple-unit site. Dr. Kress suggested using a CCDF F-C curve, equivalent to a CDF of 10^{-5} and LERF of 10^{-6}, as a design requirement for fission product release. He added that then you could put 10 plants on a site and still meet the QHOs. Dr. Apostolakis expressed concern over making the design curve overly conservative. He favored having separate siting requirements (expanded QHOs that perhaps address societal things). Dr. Apostolakis said that for the Committee to make an informed decision on the framework it needs to know, in detail, what other organizations have proposed and how they differ with the staff. The subcommittee discussed the need for writing a Committee letter on the framework (i.e., to respond to the November 8, 2006 SRM, to provide the staff with its comments on technical/policy aspects of the framework). Ms. Drouin said that public comments on the framework were generally favorable, at the conceptual level, but most indicated that they wanted to see it tested before the staff proceeded with rulemaking. She said that the staff did a very limited test of the framework on an existing LWR and that the July version of the framework reflects that test. She said that a more complete test of how the framework translates into actual requirements will be published in the version of the framework they plan on issuing early this summer. The version to be published early this summer would also contain an appendix that summarized all the stakeholder comments. The subcommittee discussed the possibility of testing the framework on an actual plant (e.g., PBMR, NGNP). Mr. Ruben, part of an inter-office team looking at the licensing strategy for the NGNP, described several licensing approaches being considered by the staff, each with varying degrees of the use of PRA information. The spectrum of options includes 1) using a completely deterministic approach as was done for Fort St. Vrain, 2) using a deterministic approach to establish DBEs and related requirements supplemented by PRA insights, 3) using a risk-derived approach that uses the guidance in the framework document to adapt existing requirements, and 4) an approach that is based on new regulation derived from the framework. The first three approaches would involve adapting existing Parts 50 and 52 requirements and issuance of exemptions as needed to license an HTGR. Dr. Abdel-Khalik suggested that the framework could be published as a Standard Review Plan as opposed to as a NUREG report. Dr. Apostolakis noted that if the staff wanted a decision for moving forward in a reasonable amount of time, the staff would need to go with the second option i.e., use existing criteria/regulations supplemented by risk insights. Mr. Ruben responded that the industry may not be enthusiastic about the staff taking that option, at least for HTGRs. He said he thought the HTGR, VHTGR, and the NGNP would probably prefer a risk-derived approach, that is, an approach that relied more heavily on PRA and risk insights to select events, identify safety-related equipment, establish defense-in-depth requirements, etc. The Subcommittee and staff discussed options for moving forward with the framework, how to test the framework, what plant design to test it on, and what regulatory mechanism one might use to implement the framework concepts.

Dr. Powers questioned why some ACRS members felt like the Committee should design the CCDF F-C curve for ten plants on a site. Dr. Kress explained that if he had a design that was required to meet an F-C requirement of a CDF of 10^{-4} and a LERF of 10^{-5} (which are derived from the QHOs), then they could only put one plant of that design on the site and still meet the QHOs. However, if they required the design to meet a lower F-C requirement, for example a CDF of 10^{-5} and a LERF of 10^{-6}, they might be able to put more plants of that design on the site and still meet the QHOs. Dr. Powers argued that the NRC should not be concerned with how may plants an applicant proposes to put on a plant site. He said that that is an economic decision. Mr. King noted that the subject of an acceptable level of integrated risk from a plant site is a policy issue that the Commission has yet to decide. The subcommittee and staff discussed various alternatives for dealing with integrated risk at sites with existing nuclear power plants. Dr. Kress said that there were several sites that he thought should not be candidates for additional nuclear power plants, even though a new plant may add an insignificant amount of additional risk. He said that to identify those sites would require the consideration of societal risk.

Dr. Powers asked the staff why it thinks it needs defense-in-depth, not so much at the containment level but at the lower levels. He asked if it was to address uncertainties or is it to cover unforseen issues or things that are modeled incorrectly. He also asked the staff how much defense-in-depth is enough. Ms. Drouin explained that defense-in-depth is there because of uncertainties, not so much the PRA data uncertainties as the state-of-knowledge uncertainties, things you don't know or don't model well. She said that basing the proposed requirements on successive protective strategies provides for some defense-in-depth. In addition, deterministic defense-in-depth criteria are imposed on LBEs dependant on their frequency of occurrence (reference Section 6.4.2.2 of draft NUREG-1860).

Dr. Powers asked Ms. Drouin how she saw Appendix B QA/QC in the mix of defense-in-depth and risk information. Mr. King said that QA is a good engineering practice that applies across the board and was not part of the framework's defense-in-depth strategy.

Dr. Wallis said that if he knew where the design was on an F-C curve he would understand how safe it is. He then said that saying they have put a lot of defense-in-depth into the design doesn't tell him anything about how safe the design is. Ms. Drouin noted that in addition to including defense-in-depth in the design, the framework requires that the design meets the QHOs. Dr. Apostolakis said that a plant isn't safe because it meets some number that comes out of the PRA. He said that it is presumptively safe (or provides adequate public protection) because it has gone through the licensing process, based on a number of factors (e.g., analyses, regulations, defense-in-depth, safety margins, etc.).

Dr. Apostolakis asked if the framework addressed the treatment of structures, systems, and components (SSCs). Ms. Drouin explained that any SSC relied on to meet the F-C curve, or whatever quantitative goal that is ultimately established, would be classified as safety significant by the framework. She added that the treatment applied to the safety significant SSCs would be a function how safety or risk significant they were. Mr. Lehner clarified that the special treatment is supposed to reflect the fact that the SSC is reliable under the conditions that you took credit for when you did the PRA. Dr. Wallis question why the staff felt compelled to classify SSCs at all, give the fact that they would be required to meet some F-C curve. He said that plant designers and managers will know that certain things are significant for safety and presumably they will take care of them. Dr. Apostolakis said that we need to classify the SSCs in part because we don't trust the licensees to take adequate care of them. Dr. Bonaca and Mr. Maynard said they did not think it was a matter of trust. Rather, they thought it was the regulator's responsibility to define the appropriate level of treatment. Dr. Kress suggested that importance measures, as opposed to LBE, be used to classify SSCs. Mr. Maynard suggested that the treatment requirements be more performance-based rather than being prescriptive based on their classification. He added however that if you credit an SSC in your PRA, you should do something to provide assurance that it has the reliability you assumed in the PRA. Ms. Drouin said that the framework calls for the use of risk importance measures to determine the treatment that should be applied to SSCs. Dr. Wallis suggested that having a living PRA should identify components whose reliability have degraded. Ms. Drouin challenged Dr. Wallis' supposition.

The subcommittee discussed with the staff plans for moving forward with the technology-neutral framework. Dr. Wallis said that he can't imagine licensing new reactors without some framework or structure from which to start. Ms. Drouin indicated that it was clear from the public received on draft NUREG-1860 and the associated ANPR, that stakeholders did not want the NRC to proceed with rulemaking at this time. She said the main question at this juncture is whether the staff approaches new reactor licensing from a deterministic footing or from a PRA perspective. Dr. Powers and Dr. Apostolakis indicated that new reactor licensing would necessarily start from a deterministic footing and that the question is how to best incorporate PRA insights (i.e., a risk-informed as opposed to a risk-derived approach).

Dr. Corradini asked if the technology-specific licensing strategy for the NGNP could be done in parallel with further development and testing of the technology-neutral framework. Mr. Monninger acknowledged the progress the staff has made on the technology-neutral framework and said that the staff had not reached any conclusions with regard to how best to proceed with the framework. He questioned whether it would be better to pursue resolution of some of the key policy issues on a technology-neutral basis, without any specific application in mind, or would testing the framework on a specific design be more productive? He said the staff is working with DOE to develop an NGNP licensing strategy and at the same time having interactions with PBMR on some high-level policy white papers. Mr. Monninger also mentioned the Global Nuclear Energy Project (GNEP). He asked whether the staff should pursue these other programs and see the extent to which the framework can contribute to them? Ms. Drouin said the framework was developed as the technical basis for a rule that would resolve several key policy issues and provide a efficient and effective regulatory structure for advanced non-LWR reactor designs. She said that the staff planned to make the rule technology-neutral and planned to put design-specific guidance in regulatory guides. She said that integrating risk into the framework from the ground up was a bigger challenge than developing the technoloy-neutral regulatory framework. Ms. Drouin said the framework document was ready to be published. Dr. Kress recommended against publishing the framework just yet, because he said a few items needed to be straightened out and it needed polishing. Ms. Drouin suggested that the next step is to develop an implementing guidance document, perhaps for the PBMR, which she thought could be done in a year. Both Ms. Drouin and Dr. Kress agreed with the stakeholder comments that it would be premature to proceed with rulemaking at this time. Dr. Wallis suggested comparing the results (regulatory requirements) derived from applying the framework to the PBMR design with the results obtained without the use of this framework. Mr. Maynard said the technology neutral framework is a useful process, whether you use the existing regulations and use this process for where you take exemptions, or whether you develop different rules for each technology. He said it is the framework by which you start making the decision. Dr. Corradini suggested that the Committee recommend that the staff continue with the development of the technology-neutral framework, because this is the fundamental underpinning, but also apply the framework in a pragmatic way using the current rules to a non-LWR design (i.e., having no choice but to do the latter given time constraints). Dr. Kress said he agreed with this approach and suggested that the design-specific implementing guidance be developed for the PBMR since the NRC has the relevant inputs for that right now in the form of white papers on PRA, Licensing Basis Events, SSC Classification, etc. Dr. Kress said that in his mind the key thing missing from the framework document is the CCDF F-C curve. He said that such a curve needs to be included in the framework for coherence in the regulatory process. Dr. Bonaca agreed that the document should be complete before it is published, even if there is a plan to issue more detailed implementation guidance. Dr. Powers said that the Fort St. Vrain and the Fast Flux Test Facility reviews were done on an ad hoc (but not entirely capricious) basis. He doubted the Commission would want to face the public confusion that would come about from an ad hoc review of new reactor designs. He acknowledged that time schedules are pressing but added "there is always time to do things over, there is never time to do it right." Dr. Powers said he would publish the framework document with just the minor editing as proposed by the staff. He said that completing the framework would be an iterative process, learning as the framework is implemented for various reactor designs. Dr. Apostolakis agreed that the framework should be polished as much as practicable and then published. Dr. Wallis said polishing was fine but agreed with Dr. Powers that something other than an ad hoc approach for the review of new reactor designs is needed. He called the framework document a good first draft but admitted there are ways to improve it. He expressed concern that the staff might stop working on it. The subcommittee and staff discussed the benefits that might be derived from pilot testing the framework on, for example, PBMR. Dr. Abdel-Khalik suggested that the framework be piloted in the development of a regulatory guide to support licensing under Part 50. Mr. Ruben suggested that the staff could take the technology-neutral framework approach and exercise it with the PBMR design and PRA to see what they come up with in terms of design-basis accidents, safety related systems, and defense-in-depth. Then they could compare those results to the results PBMR (Pty) Ltd. obtained in their white papers on LBE Selection, SSC Classification, and Defense-in-Depth Approach. This effort would be separate from the review of the PBMR design certification application.

Dr. Apostolakis acknowledged that the staff did not intend to write SERs on the PBMR white papers but asked if the Committee was going to be made aware of the staff's comments back to PBMR (Pty) Ltd. on them? Mr. Ruben indicated that he thought the staff would be bring its assessment of the PBMR white papers to the Committee for its review and possible comments.

The subcommittee had some preliminary discussion regarding whether or not the Committee would prepare a letter on the technology-neutral framework during the March ACRS full Committee meeting.

Mr. Ed Burns, Licensing Manager for PBMR, briefly described each of the four PBMR white papers that have been submitted to the staff as part of the PBMR pre-application process. He said that PBMR would be licensed under Parts 50 and 52. He said the white papers deal with technical issues that are new to the PBMR design. While the papers are not focused on the technology-neutral framework, he said that the elements of what is inside the framework will be useful.

Dr. Abdel-Khalik complimented the staff on the technology-neutral framework document and expressed his concern that work on the framework might stop. In order to make the document a truly worthwhile document, he suggested that the issues and concerns raised during the subcommittee meeting should be resolved before it is published. He also said that he thought the framework should be pilot tested, comparing whatever you are going to get with the process that is being contemplated for the gas cooled reactors.

Dr. Wallis also complimented the staff on the technology-neutral framework document and said it was a good first step. However, he acknowledged that there were ways to improve it and said the ACRS needs to mull this over probably for a year or more. He said that this is not a one meeting, one letter type of an issue and indicated that the framework will evolve over time.

Similarly, Dr. Apostolakis commented favorably on the document and expressed his concern that work on the framework might stop. He supported testing the framework on the PBMR design, since there seems to be some time without the pressure of an actual application. Dr. Kress said he thought that would be a good idea too and complimented the staff on job well done.

Ms. Drouin said she appreciated the subcommittee's comments and thought the meeting was productive. She said the framework deals with some very complex issues and acknowledged that it is evolving. She also thought it should be published, tested, and then adjusted accordingly. Finally, she recognized the staff and contractors who helped her develop the framework. Mr Monninger asked for as much early feedback as possible from the ACRS on the framework. He said the staff appreciated the ACRS's views on the need for further development of the framework and look forward to further interactions with the Committee.

SUMMARY / PLANS FOR FULL COMMITTEE PRESENTATION

The Subcommittee suggested that the staff address the following topics during the March Full Committee meeting:

- A high-level discussion of the stakeholder comments
- The staff's plans for moving forward with the technology-neutral framework

Agreements

None.

Staff/Industry Follow-up Actions

The staff plans to provide a briefing on the technology-neutral licensing framework to the full Committee during the March 8-10, 2007, ACRS meeting.

Subcommittee's Action

The subcommittee plans to provide the full Committee with proposed comments on the technology-neutral framework during the March 8-10, 2007, ACRS meeting.

Documents Provided to the Subcommittee

None

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NOTE : Additional details of this meeting can be obtained from a transcript of this meeting available for downloading or viewing on the Internet at http://www.nrc.gov/reading-rm/adams.html or http://www.nrc.gov/reading-rm/doc-collections/ can be purchased from Neal R. Gross and Co., 1323 Rhode Island Ave., N.W., Washington, DC 20005 (202) 234-4433.

UNITED STATES
NUCLEAR REGULATORY COMMISSION
ADVISORY COMMITTEE ON REACTOR SAFEGUARDS
WASHINGTON, DC 20555 - 0001

April , 2007

MEMORANDUM TO: David C. Fischer, Senior Staff Engineer
Technical Support Staff, ACRS

FROM: Thomas S. Kress, Chairman
Future Plant Designs Subcommittee

SUBJECT: CERTIFICATION OF THE MINUTES OF THE ACRS FUTURE PLANT DESIGNS SUBCOMMITTEE MEETING ON THE TECHNOLOGY NEUTRAL LICENSING FRAMEWORK (WORKING DRAFT NUREG-1860) MARCH 7, 2007, ROCKVILLE, MARYLAND

I hereby certify, to the best of my knowledge and belief, that the minutes of the subject meeting on March 7, 2007, are an accurate record of the proceedings for that meeting.

Thomas S. Kress, Date
Future Plant Designs Subcommittee, Chairman

DcF

NUCLEAR REGULATORY COMMISSION

Advisory Committee on Reactor Safeguards Subcommittee Meeting on Future Plant Designs; Notice of Meeting

The ACRS Subcommittee on Future Plant Designs will hold a meeting on March 7, 2007, Room T–2B3, 11545 Rockville Pike, Rockville, Maryland.

The entire meeting will be open to public attendance.

The agenda for the subject meeting shall be as follows:

Wednesday, March 7, 2007—10 a.m. Until the Conclusion of Business

The Subcommittee will review the NRC staff's work on technology neutral licensing framework (*i.e.*, Working Draft NUREG–1860) with a focus on ensuring the value of such an approach versus the development of a licensing framework for specific designs, such as a high temperature gas cooled reactor or a liquid metal cooled reactor (reference the Commission's November 8, 2006, Staff Requirements Memorandum to Dr. Larkins). During the briefing, the Committee will also explore with the NRC staff the pros and cons of developing a licensing framework for specific designs. The Subcommittee will gather information, analyze relevant issues and facts, and formulate proposed positions and actions, as appropriate, for deliberation by the full Committee.

Members of the public desiring to provide oral statements and/or written comments should notify the Designated Federal Official, Mr. David C. Fischer (telephone 301–415–6889) between 7:30 a.m. and 5 p.m. (ET) five days prior to the meeting, if possible, so that appropriate arrangements can be made. Electronic recordings will be permitted.

Further information regarding this meeting can be obtained by contacting the Designated Federal Official between 7:30 a.m. and 5 p.m. (ET). Persons planning to attend this meeting are urged to contact the above named individual at least two working days prior to the meeting to be advised of any potential changes to the agenda.

Dated: February 12, 2007.

Cayetano Santos,

Acting Branch Chief, ACRS.

[FR Doc. E7–3035 Filed 2–21–07; 8:45 am]

BILLING CODE 7590–01–P

POSTAL REGULATORY COMMISSION

Sunshine Act Meetings

NAME OF AGENCY: Postal Regulatory Commission.

TIME AND DATE: Thursday, February 22, 2007 at 2 p.m.

PLACE: Commission conference room, 901 New York Avenue, NW., Suite 200, Washington, DC 20268–0001.

STATUS: Closed.

MATTERS TO BE CONSIDERED: Personnel matters—selection of director of Public Affairs and Congressional Relations.

CONTACT PERSON FOR MORE INFORMATION: Stephen L. Sharfman, General Counsel, Postal Regulatory Commission, 901 New York Avenue, NW., Suite 200, Washington, DC 20268–0001, 202–789–6818.

Dated: February 16, 2007

Steven W. Williams

Secretary.

[FR Doc. 07–810 Filed 2–16–07; 4:21 pm]

BILLING CODE 7710–FW–M

SECURITIES AND EXCHANGE COMMISSION

[Release No. 34–55293; File No. SR–NYSE–2006–120]

Self-Regulatory Organizations; New York Stock Exchange LLC; Order Granting Approval of Proposed Rule Change and Notice of Filing and Order Granting Accelerated Approval to Amendment No. 1 Regarding the Proposed Combination Between NYSE Group, Inc. and Euronext N.V.

February 14, 2007.

I. Introduction

On December 29, 2006, New York Stock Exchange LLC ("NYSE" or "Exchange") filed with the Securities and Exchange Commission ("Commission"), pursuant to Section 19(b)(1) of the Securities Exchange Act of 1934, as amended, ("Exchange Act") [1] and Rule 19b–4 thereunder,[2] a proposed rule change regarding the proposed business combination ("Combination") between NYSE Group, Inc. ("NYSE Group") and Euronext N.V. ("Euronext"). The proposed rule change was published for comment in the **Federal Register** on January 8, 2007.[3] The Commission has received two comments on the proposal.[4] The Exchange filed a response to comments on February 14, 2007.[5]

On February 13, 2007, the Exchange filed Amendment No. 1 to the proposed rule change.[6] This order approves the proposed rule change, grants accelerated approval to Amendment No. 1, and solicits comments from interested persons on Amendment No. 1.

The Commission has reviewed carefully the proposed rule change, the comment letters, and the NYSE Response to Comments, and finds that the proposed rule change is consistent with the requirements of the Exchange Act and the rules and regulations thereunder applicable to a national securities exchange.[7] In particular, the Commission finds that the proposed rule change is consistent with Section 6(b) of the Exchange Act,[8] which, among other things, requires a national securities exchange to be so organized and have the capacity to be able to carry out the purposes of the Exchange Act and to enforce compliance by its members and persons associated with its members with the provisions of the Exchange Act, the rules and regulations thereunder, and the rules of the exchange, and assure the fair representation of its members in the selection of its directors and administration of its affairs, and provide that one or more directors shall be representative of issuers and investors and not be associated with a member of the exchange, broker, or dealer. Section 6(b) of the Exchange Act [9] also requires that the rules of the exchange be designed to promote just and equitable principles of trade, to remove impediments to and perfect the mechanism of a free and open market and a national market system, and, in general, to protect investors and the public interest.

[1] 15 U.S.C. 78s(b)(1).

[2] 17 CFR 240.19b–4.

[3] *See* Securities Exchange Act Release No. 55026 (December 29, 2006), 72 FR 814 ("Notice").

[4] *See* letter from Andrew Rothlein, to Nancy Morris, Secretary, Commission, dated January 17, 2007 ("OTR Investors Letter"); and letter from Professor J. Robert Brown, Jr., University of Denver Sturm College of Law, to Nancy Morris, Secretary, Commission, received by the Commission, February 13, 2007 ("Brown Letter").

[5] *See* letter from Mary Yeager, Assistant Secretary, NYSE, to Nancy M. Morris, Secretary, Commission, dated February 14, 2007 ("NYSE Response to Comments").

[6] *See* Partial Amendment dated February 13, 2007 ("Amendment No. 1"). The text of Amendment No. 1 and Exhibits 5C, 5D, 5F, 5G, 5H, 5I, 5J, and 5M, which set forth certain governing documents as proposed to be amended, are available on the Commission's Web site (*http://www.sec.gov/rules/sro.shtml*), at the Commission's Public Reference Room, at the NYSE, and on the NYSE's Web site (*http://www.nyse.com*).

[7] In approving the proposed rule change, the Commission has considered its impact on efficiency, competition, and capital formation. *See* 15 U.S.C. 78c(f).

[8] 15 U.S.C. 78f(b).

[9] *Id.*

ADVISORY COMMITTEE ON REACTOR SAFEGUARDS

SUBCOMMITTEE MEETING ON FUTURE PLANT DESIGNS

March 7, 2007
Date

NRC STAFF SIGN IN FOR ACRS MEETING

PLEASE PRINT

	NAME	NRC ORGANIZATION
1	MARTIN STUTZKE	NRR/DRA/APLA
2	Mary Drouin	RES/DRASP/PRA
3	THOMAS KENYON	NRO/DNRL/NWE/
4	Eileen McKenna	NRR/DPR
5	Ben Beasley	RES/DRASP/PRA
6	DON DUBE	NRR/DRA
7	Joe Birmingham	NRR/DPR
8	Ronaldo V. Jenkins	RES/DRASP/PRA
9	Yuri Orechwa	NRR/DSS
10	Sud Basu	RES/DRASP/NRCA
11	Lynn Mrowca	NRO/DSRA
12	Lauren Killian	RES/DRASP/PRA
13	Stuart Rubin	RES/DRASP/NRCA
14	David Bessette	RES/DRASP/NRCA
15	Rajendra Solanki	NRR/DRA/APLA
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ADVISORY COMMITTEE ON REACTOR SAFEGUARDS

SUBCOMMITTEE MEETING ON FUTURE PLANT DESIGNS

March 7, 2007
Date

PLEASE PRINT

	NAME	AFFLIATION
1	John Lehner	BNL
2	Vinod Mubayi	BNL
3	Tom King	ISL, Inc.
4	Alan Levin	AREVA
5	Edward Burns	PBMR
6	Patrick O'Regan	EPRI
7	Biff Bradley	NEI
8	Bruce Mrowca	ISL
9	SHAWNIE HARRIS	CH2M HILL
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Advisory Committee on Reactor Safeguards
Future Plant Designs Subcommittee
Technology Neutral Licensing Framework
March 7, 2007
Rockville, MD

Annotated with actual times

-PROPOSED AGENDA-

Cognizant Staff Engineer: David C. Fischer DCF@NRC.GOV (301) 415-6889

	Topics	Presenters	Presentation Time
I	Opening Remarks	T. Kress, ACRS	10:00 am - 10:10 am
II	Staff Introductory Remarks • Framework for Development of a Risk-Informed, Performance-Based Alternative to Part 50 • Staff Plan for Moving Forward, SECY Paper • Relationship to the Development of NGNP Licensing Options and Activities	~~F. Eltawila~~, RES J. Monniger	10:10 am - ~~10:20~~ am 10:12 am 12:05 pm
III	Framework Overview	M. Drouin, RES	10:12 am ~~10:20~~ am - ~~10:30~~ am
IV	Framework Technical Issues*: • Probabilistic Approach -- Risk-informed vs risk-derived -- Level of safety -- integrated risk -- Frequency-consequence curve -- LBE selection and sequence classes -- safety classification -- safety margins *Includes Views raised by ACRS letter, T. Kress, stakeholder comments from ANPR	NRC Staff	10:30 am - 12:00 pm ~~10:45 am - 11:00 am~~ DF ~~- 12:05 pm~~ DF 12:05
	LUNCH		~~12:00~~ pm - 1:00 pm
V	Framework Technical Issues (cont'd)*: • Probabilistic Approach -- LBE selection and sequence classes -- safety classification -- safety margins • Defense-in-depth -- definition -- principles -- implementation *Includes Views raised by ACRS letter, T. Kress, stakeholder comments from ANPR	NRC staff	1:00 pm - 3:00 pm 1:05
	BREAK		3:00 pm - 3:15 pm
VI	Framework Technical Issues (cont'd)*: • PRA -- scope and level of detail -- design stage vs operational stage -- living *Includes Views raised by ACRS letter, T. Kress, stakeholder comments from ANPR	NRC staff	3:15 pm - 3:45 pm
VII	Response to the November 8, 2006, SRM - Pros and Cons of the Alternative Approaches	NRC Staff/ ACRS Members	~~3:45 pm - 4:45 pm~~ DF
VIII	Summary / Plans for Full Committee	T. Kress, ACRS	~~4:45~~ pm - ~~5:00~~ pm 4:21 pm

NOTE:

- Presentation time should not exceed 50 percent of the total time allocated for a specific item. The remaining 50 percent of the time is reserved for discussion.
- 35 copies of the presentation materials to be provided to the Subcommittee.

March 22, 2006

MEMORANDUM TO: Luis A. Reyes
Executive Director for Operations

FROM: Annette L. Vietti-Cook, Secretary **/RA/**

SUBJECT: STAFF REQUIREMENTS - SECY-06-0007 - STAFF PLAN TO MAKE A RISK-INFORMED AND PERFORMANCE-BASED REVISION TO 10 CFR PART 50

The Commission has approved the staff's recommendation to issue an Advanced Notice of Proposed Rulemaking (ANPR) on approaches for making technical requirements for power reactors risk-informed, performance-based, and technology neutral, subject to the comments below and edits provided below. The Commission has approved the staff's recommendation to supplement the ANPR with new information, as needed. The staff should provide advance notice to the Commission offices of any significant changes to the ANPR. The staff should place the latest working draft of the technology neutral framework on the RuleForum website no later than the date of publication of the ANPR.

The staff should complete the ANPR stage by December 2006 and provide its recommendation on whether and, if so, how to proceed with rulemaking by May 2007 having considered ACRS views. At the end of the ANPR stage, the staff should provide, with its recommendation, a detailed summary of any differing stakeholder views to ensure that the Commission has the benefit of these views when deliberating on the recommendation. The staff's recommendations may need to consider a broader range of options than just whether to proceed to rulemaking. The staff's recommendation should include a proposed schedule to complete the effort.

The staff should include an appropriate list of questions in the section on the technology neutral framework prior to publication of the ANPR. Stakeholder input should be sought in areas such as whether this effort is premature; the definition of a "unified safety concept"; whether NRC should be focusing on developing technology-specific frameworks for non-light water (LWR) reactors; and what priority should be given for various non-LWR technologies.

To facilitate stakeholder participation, the staff should hold public meetings and workshops starting soon after the ANPR is issued. In addition, the staff should keep stakeholders informed of progress throughout the public comment period.

The staff should inform the Commission on the additional resources needed to accelerate the schedule to meet the December 2006 expiration date for the ANPR.

<u>Changes to the *Federal Register* Notice</u>

1. Page 2, paragraph 1, revise line 4 to read ' ... regulations to make them ~~be~~ risk-informed and ....'

2. Page 3, paragraph 2, revise line 1 to read ' ... December 29, 2006 ~~31, 2007~~.'

3. Page 5, insert the following at the beginning of the Background section:

   The NRC is considering developing a comprehensive set of risk-informed, performance-based, and technology neutral requirements for licensing nuclear power reactors. These requirements would be included in NRC regulations as a new 10 CFR Part 53 and could be used as an alternative to the existing requirements in 10 CFR Part 50.

4. Page 5, paragraph 2, revise lines 1 and 2 to read ' ... NRC staff to ~~: (1)~~ develop an ANPR to facilitate early stakeholder participation in this effort. The Commission also directed the NRC staff to: (1) ~~(2)~~ incorporate in the ANPR a formal program plan for ~~a~~ risk-informing 10 CFR ....' Revise line 3 to read ' ... efforts, (2) ~~and (3)~~ integrate ....' Revise lines 4 and 5 to read ' ... ~~(ADAMS Accession Numbers ML051290351 and ML052570437). The Commission also directed the staff~~ and (3) include the ....' Revise line 6 to read ' ... Accession Numbers ML051290351, ML052570437, and ML052640492).'

5. Page 5, last paragraph, revise line 4 to read ' ... development of a technology- ....'

6. Page 6, 1st full paragraph, delete the last 2 sentences (However, the NRC ... Part 50.)

7. Page 7, paragraph 1, revise line 2 to read ' ... -based alternative ~~revision~~ to 10 ....' Revise line 3 to read ' ... designs. ~~To accomplish this goal, s~~ Safety, security ....' Revise line 4 to read ' ... integrated into this effort to provide one ....' Revise line 7 to read ' ... -based alternative ~~revision~~ to 10 ....' Revise line 9 to read ' ... importance to public health and safety, (2) provide NRC with a ~~the~~ framework that uses ~~to use~~ risk ....' Revise line 10 to read ' ... manner ~~to take action in reactor regulatory matters~~, (3) ....' Revise lines 11 and 12 to read ' ... operation ~~, which can result in burden reduction without compromising~~ while maintaining or enhancing safety ....'

8. Page 7, last paragraph, revise lines 1 and 2 to read ' ... approach ~~to develop a risk-informed and performance-based revision to 10 CFR Part 50~~ is to create ....'

9. Page 8, 1st full paragraph, revise line 4 to read ' ... in SECY-05-0006 ....'

10. Page 8, 2nd full paragraph, revise line 1 to read ' ... technical basis is being developed ~~and completed~~, it is ....' Delete the last sentence (Consequently , the time ... is complete.)

11. Page 8, last paragraph, revise line 1 to read ' ... and issue the ~~actual~~ regulations for Part 53. If upon completion of the technical basis the Commission directs the NRC staff to proceed to rulemaking, t~~T~~he ....' Revise lines 2 and 3 to read ' ... NRC staff will follow its normal rule development process ~~upon completion of the technical basis.~~ The ~~Commission will direct the~~ NRC staff will ~~to~~ develop proposed ....' Revise line 4 to read ' ... on web, public workshopd), and provide ~~send~~ a proposed rule ....' Revise line 5 to

read ' ... for consideration ~~if rulemaking is undertaken~~.'

12. Page 9, paragraph A., revise line 1 to read ' ... on the proposed plan ....'

13. Page 9, paragraph A.1., revise line 1 to read ' ... -based alternative ~~revision~~ to 10 ....' Revise line 2 to read ' ...reasonable? ~~That is, i~~Is there a ....' Revise lines 3 and 4 to read ' ... -based regulatory framework for nuclear power reactors ~~10 CFR Part 50~~? If yes, please describe the better approach ~~what is a better and different way~~.

14. Page 9, paragraph A.2., revise line 1 to read ' ... articulated above in the proposed plan section, understandable ....' Revise line 2 to read ' ... so, please describe the additional objectives and explain the reasons for including them ~~why and what are they~~?'

15. Page 10, paragraph 3., revise line 1 to read 'Would ~~Does~~ the approach described above in the proposed plan section accomplish the objectives? If not, why not and what changes to the approach would allow for accomplishing the objectives?'

16. Page 10, paragraph 4., add the following to the end: If not, why not? If so, please discuss the main reasons for doing so.

17. Page 10, paragraph 5., add the following at the end of line 2: Please discuss the reasons for your answer.

18. Page 10, paragraph 7., revise lines 1 through 5 to read 'The NRC encourages active stakeholder participation through ~~If industry wishes to participate in the~~ development of ~~an alternative process, the NRC envisions the process could involve the following:~~ proposed supporting documents, ~~and~~ standards, and guidance ~~could be developed by industry, and provided in writing to NRC staff for consideration~~. In such a process, the ~~The~~ proposed documents, standards, and guidance would be submitted to and reviewed by ....' Delete the sentence in lines 5 through 9 (To the extent ... the subject.) Revise lines 9 and 10 to read '~~What~~ Is there any interest by stakeholders to develop proposed supporting documents, standards, or guidance? If so, please identify your organization and the specific documents, ~~and~~ standards, or guidance you are interested in taking ~~would industry be willing to take~~ the lead ....'

19. Page 11, paragraph B., revise line 1 to read ' ... security, and ....' Revise line 7 to read ' ... and effective (intrinsic) security posture ....'

20. Page 11, paragraph 8., revise line 1 to read ' ... alternative regulatory framework ~~licensing basis~~, how ....'

21. Page 11, paragraph 10., revise lines 1 and 2 to read ' ... security be ~~better~~ integrated so as to allow an easier and more thorough understanding of the effects that changes in one area would have on ~~not adversely affect~~ the other and to ensure that changes with unacceptable impacts are not implemented. How ....'

22. Page 12, after line 2 from the top, insert a new paragraph number 11. which reads as follows: Should security requirements be risk-informed? Why or why not? If so, what specific security requirements or analysis types would most benefit from the use of PRA and how? Renumber the original paragraph 11. to be paragraph 12.

23. Page 12, paragraph 11., replace the text as follows: Should emergency preparedness requirements be risk-informed? Why or why not? How should emergency preparedness requirements be modified to be better integrated with safety and security?

24. Page 12, paragraph C., revise line 2 through 4 to read ' ... options for establishing a regulatory standard that would be applied during licensing to ~~specifying a minimum level of safety from the standpoint of risk which would implement the Commission's expectation of~~ enhance~~d~~ safety for new plants consistent with ~~(as expressed in~~ the Commission's policy statement for Regulation of Advanced Nuclear Power Plants~~)~~.' Revise line 8 to read ' ... risk objectives for the acceptable level of safety, and ....'

25. Page 12, last paragraph, revise lines 1 and 2 to read '~~With regard to specifying the minimum level of safety from the standpoint of risk, s~~Subsidiary risk objectives could also be developed to implement the Commission's expectation regarding enhanced safety for new plants. Such ....'

26. Page 13, revise line 1 from the top to read 'provide high ~~top~~ level goals to assist in establishing plant system and equipment ~~hardware~~ ....'

27. Page 13, revise line 8 from the top to read ' ... offsite such that no ~~suuficient to cause one or more~~ early fatalities occur (i.e., from acute radiation doses).'

28. Page 13, renumber paragraphs as necessary to conform to changes throughout the document.

29. Page 13, paragraph 12., add the following to the end: If so, please discuss the alternative options and their benefits.

30. Page 13, paragraph 13., revise to read '~~Are subsidiary risk objectives useful, and~~ Should the staff pursue developing subsidiary risk objectives? Why or why not? Are ~~are~~ there other uses of ~~the~~ subsidiary risk objectives that are not specified above? If so, what are they?'

31. Page 13, paragraph 14., delete the 2nd question up through ' ... QHO, i.e.,' and move the remainder of the question to a new numbered paragraph starting with 'S~~s~~hould the latent ...?' In line 4, after the question mark, move the last 2 questions to a new numbered paragraph. Revise the last line to read 'What are they and what would ...?'

32. Page 14, paragraph 16., revise line 3 to read ' ... criteria and why ~~its basis~~?'

33. Page 14, paragraph 17., revise line 1 to read ' ... analysis (i.e., one that includes calculation of offsite health and economic effects) still be needed ....'

34. Page 14, paragraph D., revise line 1 to read ' ... licensing, potential applicants ~~some licensees~~ have indicated ~~their~~ interest ....' Revise line 2 to read ' ... at new and existing sites. In addition, potential applicants have indicated interest in locating ~~or~~ multiple (or modular) reactor units at new and existing sites. The ....' Revise line 5 to read ' ... site only from new reactors (i.e., the integrated risk would not consider existing reactors), and (3) quantification of integrated site risk ~~(~~ for all reactors (new and existing) at that site~~)~~.' Revise lines 7 and 8 to read ' ... integrated risk should be restricted to the same level that would be applied to a single reactor. ~~From the new plants should meet the~~

~~level of safety that the NRC has proposed for new plants.~~ If this ~~new~~ approach ....' Revise the last line to read ' ... integrated risk ~~of these new~~ ....'

35. Page 15, revise line 1 from the top to read ' ... ~~plants should not~~ would not be allowed to exceed the ....'

36. Page 15, paragraph 18., add the following to the end: If so, what are they?

37. Page 15, paragraph 19., revise line 1 to read ' ... considered? Why or why not? ~~and if so, should the~~ ....' Move the next question after the 2nd question mark in line 1 to a new numbered paragraph and revise it to read 'If integrated risk should be considered, should the risk meet a minimum ...? Why or why not?' Delete the remainder of this item (If not, why not? Or should ... yes, why?)

38. Page 15, paragraph E., revise line 3 to read ' ... issues related to requiring ~~of requiring new plants to meet a minimum level of~~ enhanced ....' In lines 5 and 6, delete the semicolon after "2005".

39. Page 16, delete paragraph 20.

40. Page 16, paragraph 21., revise line 1 to read '~~How s~~Should the views raised in the ACRS letter and by various ....' Add the following to the end: Why or why not?

41. Page 16, paragraph F., revise line 1 to read ' ... Commission has directed ~~asked~~ the staff ....'

42. Page 16, after paragraph 22., insert a new numbered paragraph which reads: Should the containment functional performance standards be design and technology specific? Why or why not?

43. Page 17, delete the questions in lines 2 and 3 from the top (Should the ... so, how?)

44. Page 17, delete paragraph 25.

45. Page 17, paragraph 26., revise line 2 to read ' ... approaches and ~~to~~ defense ....'

46. Page 17, paragraph 27., revise line 1 to read ' ... should the "rare" events in the range 10^{-4} to 10^{-7} per year be considered ....' Revise line 2 to read ' ... events less than ~~below~~ 10^{-7} per year in frequency be ....' Delete the last question (Should postulated ... criteria?)

47. Page 18, delete paragraph 28.

48. Page 18, delete the last paragraph (The latest working draft ... comment.) Replace it with the following: The NRC is seeking stakeholder views of the following aspects: [The staff should include specific questions on this area.]

49. Page 19, paragraph H., revise line 7 to read ' ... the SRM on ~~to~~ SECY-03-0047, ...'

50. Page 19, paragraph 29., Revise lines 1 through 3 to read ' ... development of a better description of ~~policy statement or~~ defense-in-depth ~~for incorporation into the~~

~~Commission's Policy Statement on PRA as described above,~~ be of any benefit to current operating plants, near-term designs, or future designs? Why or why not? If so, please discuss any specific benefits.

51. Page 19, delete paragraph 30. and replace it with the following: If the NRC undertakes developing a better description of defense-in-depth, would it be more effective and efficient to incorporate it into the Commission's Policy Statement on PRA or should it be provided in a separate policy statement? Why?

52. Page 20, delete paragraph 31.

53. Page 20, paragraph 32., revise lines 2 and 3 to read ' ... basis. Should ~~If~~ RG 1.174 ~~were to~~ be revised to include a better description of defense-in-depth? Why or why not? If so, would a change to RG 1.174 be sufficient instead of a policy statement? Why or why not? Move the remainder of this item to a new numbered paragraph and revise as follows: '~~h~~How should defense-in-depth be addressed for new plants ~~where defense-in-depth is being incorporated into the de sign~~?'

54. Page 20, paragraph 33., delete the 1st question (For both ... statement?) Revise lines 2 through 5 to read '~~Is it reasonable to link~~ Should development of a better description of ~~policy statement on~~ defense-in-depth (whether as a new policy statement, ~~or~~ a revision to the PRA policy statement, or as an update to RG 1.174) be completed on the same schedule as ~~to the development of~~ Part 53? Why or ....?' Delete the last question in lines 5 and 6 (That is, if ... statement?)

55. Page 21, revise line 1 from the top to read ' ... provides the following options ~~alternatives to the SFC~~: (1) maintain ....'

56. Page 21, paragraph 34., revise line 1 to read ' ... proposed options ~~alternatives~~ reasonable? If ...?'

57. Page 21, insert a new numbered paragraph after paragraph 34. which reads: Are there other options for risk-informing the SFC? If so, please discuss these options.

58. Page 21, paragraph 35., revise lines 1 through 3 to read 'Which option ~~alternative~~, if any should be considered?' Move the remainder of this item to a new numbered paragraph and revise as follows: '~~That is, s~~Should ~~any~~ changes to the SFC in 10 CFR Part 50 be pursued separate from or as a part of the effort to create ~~or should it be considered in the context of creating~~ a new Part 53? Why or why not?'

59. Page 21, paragraph J., delete the 1st sentence (Currently, 10 CFR ... requirements.)

60. Page 22, 1st full paragraph, delete the last sentence (In the longer term, the ... requirements.)

61. Page 22, paragraph 36., revise line 1 to read ' ... NRC ~~only~~ continue with ....' Revise lines 2 and 3 to read ' ... Part 50, or should the NRC ~~only~~ undertake ....' After the question mark in line 3, insert "Why?" Move the last 2 questions to a new numbered paragraph revised to read 'If the NRC were to undertake new risk-informed rulemakings, w~~W~~ich regulations would be the most beneficial to revise? What would be the anticipated safety benefits?'

62. Page 22, paragraph 37., revise line 2 to read ' ... but whose ~~their~~ associated ...?' Revise the last line to read ' ... having revised and why?'

63. Page 23, paragraph 38., revise line 1 to read ' ... regulations and /or associated ...?' Revise line 2 to read ' ... when should the NRC ~~does it make sense to~~ initiate ...?'

cc: Chairman Diaz
Commissioner McGaffigan
Commissioner Merrifield
Commissioner Jaczko
Commissioner Lyons
OGC
CFO
OCA
OPA
Office Directors, Regions, ACRS, ACNW, ASLBP (via E-Mail)
PDR

UNITED STATES
NUCLEAR REGULATORY COMMISSION
ADVISORY COMMITTEE ON REACTOR SAFEGUARDS
WASHINGTON, DC 20555 - 0001

ACRSR-2149

September 21, 2005

The Honorable Nils J. Diaz
Chairman
U.S. Nuclear Regulatory Commission
Washington, DC 20555

SUBJECT: REPORT ON TWO POLICY ISSUES RELATED TO NEW PLANT LICENSING

Dear Chairman Diaz:

During the 523rd meeting of the Advisory Committee on Reactor Safeguards, June 1-3, 2005, we met with the NRC staff and discussed two policy issues related to new plant licensing. We also discussed this matter during our 524th, July 6-8, 2005, and 525th, September 8-10, 2005 meetings. We had the benefit of the documents referenced. These policy issues were:

- What shall be the minimum level of safety that new plants need to meet to achieve enhanced safety?
- How shall the risk from multiple reactors at a single site be accounted for?

In SECY-05-0130, the staff recommends that the expectation for enhanced safety be met by requiring that new plants meet the Quantitative Health Objectives (QHOs), i.e., by applying the QHOs to individual plants. The staff maintains that this would represent an enhancement in safety over current plants, which are now required to meet adequate protection, but may not meet the QHOs. The staff argues that this position is consistent with the Commission's Policy Statement on Regulation of Advanced Nuclear Power Plants.

The staff proposes to address the risk of multiple reactors at a single site by requiring that the integrated risk associated with only new reactors (i.e., modular or multiple reactors) at a site not exceed the risk expressed by the QHOs. The risk from existing plants, which may already exceed the QHOs, is not considered.

We discussed these issues and concluded that use of the existing QHOs is not sufficient to resolve either of these issues. In considering the overall scope of the issues raised by the staff, we found it more apt and effective to reframe the two issues into the following questions:

1. What are the appropriate measures of safety to use in the consideration of the certification of a new reactor design?

2. Should quantitative criteria for these measures be imposed to define the minimum level of safety?

3. How should these measures be applied to modular designs?

4. How should risk from multiple reactors at a site be combined for evaluation by suitable criteria?

5. How should the combination of new and old reactors at a site be evaluated by these criteria?

6. What should these criteria be?

7. How should compliance with these criteria be demonstrated?

DISCUSSION

Question 1. What are the appropriate measures of safety to use in the consideration of the certification of a new reactor design?

The QHOs are criteria for the risk at a site and thus involve not only the design and operation of the reactor(s), but also the site characteristics, the number and power level of plants on the site, meteorological conditions, population distribution, and emergency planning measures. By themselves, the QHOs do not express the defense-in-depth philosophy that the Commission seeks to limit not only the risk from accidents, but also the frequency of accidents.

Although core damage frequency (CDF) and large, early release frequency (LERF) have been viewed by the NRC as light water reactor (LWR)-specific surrogates for the QHOs, they have come to be accepted as metrics to gauge the acceptable level of safety of certified designs and the acceptability of proposed changes in the licensing basis. They are measures of reactor design safety that incorporate a defense-in-depth balance between prevention and mitigation. Currently used values of these metrics have been derived from the QHOs. If they were no longer to be viewed as surrogates, acceptance values for these metrics could be independently specified and need not be derived from the QHOs. Thus, they would be fundamental characteristics of reactor design independent of siting and emergency planning requirements.

If these measures are no longer viewed as surrogates for the QHOs, the appropriate measure of a large release need not be restricted to "early" but could be a "large release frequency" (LRF) which would apply to the summation of all large release frequencies regardless of the time of occurrence. The LRF would thus have broader applicability to designs in which the release is likely to occur over an extended period.

A majority of the Committee members favors the use of CDF and LRF as fundamental measures of the enhanced safety of new reactor designs and not simply as surrogates for the QHOs.

In SECY-05-0130, the staff argues that it will be difficult to derive such measures for different technologies, although the staff proposes to include them as subsidiary goals in their technology-neutral framework document. Although the processes and mechanisms for failure and release will differ greatly for different reactor technologies, technology-neutral definitions in terms of a release from the fuel (the accident prevention/CDF goal) and from the containment/ confinement (the large release goal) seem feasible to us. For example, the CDF of a Pebble Bed Modular Reactor (PBMR), would be an indicator of the success criteria for the design measures intended to prevent release from the fuel of that module. It could be defined in terms of the frequency of exceeding a fuel temperature of 1600 °C.

Question 2. Should quantitative criteria for these measures be imposed to define the minimum level of safety?

In the current Policy Statement on the Regulation of Advanced Nuclear Power Plants, the Commission decided not to set numerical criteria for enhanced safety but rather focused on aspects which might make designs more robust. In addition, the Safety Goal Policy Statement was intended to provide a definition of "how safe is safe enough." If a plant would meet the QHOs at a proposed site, then the additional risk it imposes is already very low compared to other risk in society. It now seems possible to build economically competitive reactors with risks at most sites that would be much lower than implied by the QHOs. The Electric Power Research Institute (EPRI) and European Utility Requirements Documents specify CDF and LERF values that would provide large margins to the QHOs for virtually all sites. An explicit commitment to lower values of CDF and LRF would be responsive to the Commission's desire for enhanced safety and may have significant impact on public perceptions and confidence.

We considered the following alternatives, identifying arguments in favor of each. Since such a decision has broad practical implementation and policy implications, we recommend that the staff further explore the consequences of these (and possibly other) choices as a basis for an eventual Commission decision.

a. Set maximum values for CDF and LRF at 10^{-5}/yr and 10^{-6}/yr for new reactor designs. This would make more explicit the Commission's stated expectation that future reactors provide enhanced safety. This could also provide a basis for establishing multinational design approval (as these would now be independent of U.S. QHOs). The suggested values are consistent with those in the EPRI and the European Utility Requirements Documents, the EPR Safety Document, and those used in the certification of advanced reactors (the ABWR, AP600 and CE-System 80+). These values are also consistent with the generic values for an accident prevention frequency and a LRF in the staff's draft technology-neutral framework document.

b. Leave the values unspecified. CDF and LRF would be considered along with other aspects of the design, such as defense-in-depth and passive safety features, in reaching a decision about design certification. This would give the staff more flexibility to respond to technology-specific features.

On a preliminary basis, the majority of the Committee members favor Alternative (a), but is not ready to make a recommendation until more is understood about the likely consequences and policy implications of the decision.

Question 3. How should these measures be applied to modular designs?

The staff's considerations of integrated risk do not distinguish between criteria for modular reactor designs and criteria for the risk due to multiple plants on a site. Thus, the staff treats CDF and LRF (or LERF) for modular designs and/or multiple plants on a site as still being QHO risk surrogates. In our view, the CDF and LRF metrics are design criteria that are to be "imposed" at the plant design certification stage independent of any site considerations.

New reactors could include PBMR, AP600, AP1000, Economic and Simplified Boiling Water Reactor (ESBWR), and EPR, and the number of new reactors at a site could vary by an order of magnitude.

Some Committee members believe that to get consistency in expectations of enhanced safety in all cases, the integrated risk from all new reactors on a site is the appropriate measure. This is true both for the risk metric LRF and the defense-in-depth accident prevention metric CDF. Thus, for the PBMR, which is proposed in terms of an eight-module package, the CDF and LRF goals (e.g., 10^{-5}/ry and 10^{-6}/ry) would be applied to the package. In effect each module would have to have a somewhat lower CDF and LRF. Because of the potential for interactions, analysis of individual modules may not be meaningful and the analysis should focus on the "eight pack."

Other Committee members prefer CDF and LRF design specifications that are independent of the number of modules. These members believe the specified acceptable CDF for enhanced safety (e.g. 10^{-5}/yr) should be applied to each module at the design stage and would be an indicator of the success criteria for the design measures provided for each module intended to prevent release from the fuel of that module. Similarly, LRF would be on a modular basis. As it may be possible to restrict the total power of a given module to a level that the quantity of fission products releasable cannot exceed the acceptance LRF value (e.g. 10^{-6}/yr), a modular design implicitly represents a kind of defense-in-depth (given appropriate consideration of common-mode failures and module interactions).

Question 4. How should risk from multiple reactors at a site be combined for evaluation by suitable criteria?

The QHOs address the risk to individuals that live in the vicinity of a site. Logically, the risk to these individuals should be determined by integrating the risk from all the units at the site. The manner by which the risks of different units at a site are to be integrated must address the treatment of modular designs, units with differing power levels, and accidents involving multiple units.

Question 5. How should the combination of new and old reactors at a site be evaluated by these criteria?

Any new plant that meets the independent safety criteria discussed in Questions 1 through 3 would be expected to add substantially less risk to an existing site than that already provided by existing plants on the site. If a proposed site already exceeds the QHOs, it should not be approved for new plants. For existing sites not being proposed for the addition of new plants, there would be no need to assess their risk status because they provide adequate protection. These sites would, thus, be grandfathered in the new framework.

Question 6. What should these criteria be?

Use of the QHOs for evaluating the site suitability for new reactors is attractive because the QHOs represent a fundamental statement about risk independent of any particular technology. The current QHOs (prompt and latent fatalities), however, only address individual risk and do not directly address societal risks such as total deaths, injuries, non-fatal cancers, and land contamination. These societal impacts are addressed somewhat in the current regulations by the siting criteria on population.

Some ACRS members believe that measures of societal risk need to be an explicit part of any new technology-neutral framework. The staff argues in the technology-neutral framework document that the limits proposed there for CDF and LRF limit societal risks such as land contamination and dose to the total population. However, these members recognize that CDF and LRF are not equivalent to risk and disagree with the staff's position.

Other ACRS members believe that the current siting criteria have served to limit societal risks. In addition, societal risks are considered in the environmental impact assessments of license renewal. The estimates presented in NUREG-1437 Vol. 1 indicate that the risk of early and latent fatalities from current nuclear power plants is small. The predicted early and latent fatalities from all plants (that is, the risk to the population of the United States from all nuclear power plants) is approximately one additional early fatality per year and approximately 90 additional latent fatalities per year, which is a small fraction of the approximately 100,000 accidental and 500,000 cancer fatalities per year from other sources. The evaluation of Severe Accident Mitigation Alternatives (SAMAs) as part of the license renewal process also considers societal risk measures and monetizes them to perform cost benefit studies. Based on current NRC regulatory analysis guidance, very few of these SAMAs appear cost beneficial.

Environmental impact statements (EISs) also assess the societal costs of probabilistic accidents at the current sites. The results, although very approximate, indicate that the societal costs at many current reactor sites would likely exceed a reasonable societal cost risk acceptance criterion. For example, these would exceed the cost associated with 0.1% of the above noted 100,000 early fatalities due to all accidents.

Thus, the inclusion of a quantitative societal risk acceptance measure appears important and could add to greater public confidence and understanding of the risks of nuclear power. It may be worthwhile for the staff to consider supplementing the current QHOs with additional risk acceptance measures that relate directly to societal risks.

7. *How should compliance with these criteria be demonstrated?*

The establishment of goals or criteria of various kinds cannot be divorced from the ability to demonstrate compliance. Considerable improvement in PRA practice will be needed to provide confidence that the goals on CDF and LRF for future plants will be met in a meaningful way. Operating experience has been crucial for the analysts to appreciate the significance of potential errors/faults. For example, before TMI, it was assumed that operators would not have problems diagnosing what is going on under certain conditions.

Some of the challenges that new plants will create for PRA analysts are:

I. Operating experience on component failure rate distributions and frequencies developed for light-water reactors has limited applicability to other reactor types.
ii. Some designs are considering components, e.g., microturbines and fuel cells, for which reliability data are nearly non-existent.
iii. Digital Instrumentation and Control systems are expected to be an integral part of future reactor designs. The risk consequences of such practice are difficult to quantify at this time.

Thus, in addition to the imposition of design goals for low CDF and LRF, it will be important to maintain sufficient defense-in-depth in the technology-neutral framework.

We look forward to additional discussion with the staff on these issues.

Sincerely,

/RA/

Graham B. Wallis
Chairman

Additional comments from ACRS Members Dana A. Powers and John D. Sieber

We disagree with our colleagues on the matter of this letter. The Commission has indicated a laudable expectation that future reactors will be safer than current reactors. The question that our colleagues should have addressed first is whether a quantitative metric is needed to substantiate this expectation. It is by no means obvious that such a metric is essential. We can well imagine future plants designed in conjunction with far more comprehensive probabilistic safety analyses that realistically address all known accident hazards during all modes of operation to a depth far greater than is attempted now for elements of the fleet of operating reactors. Our experience has been that whenever improvements are made in quantitative risk analysis methods, unforeseen, hazardous, plant configurations, systems interactions and operations become apparent. Hidden, these configurations, interactions and operations may arise unexpectedly with undesirable consequences. Revealed, they can be avoided often with modest efforts. This is exploitation of the full potential of quantitative risk analysis to achieve greater safety in nuclear power plants. It contrasts with the more effete pursuit of the "bottomline" results of PRA to compare with arbitrarily proliferated safety metrics.

Our objective should be to foster the voluntary development of quantitative risk analysis methods both in scope and depth in order to improve the safety of nuclear power plants. Fostering voluntary development of methods by nuclear community is especially important now when methods developments have stagnated at NRC relative to the situation a decade ago.

Our colleagues seem to presume it essential that future reactors meet the Quantitative Health Objectives (QHOs). These QHOs define a very stringent safety level that has always been viewed as an "aiming point" or a benchmark and not as some minimum standard that cannot be exceeded. Indeed, the definition of the QHOs was undertaken to define "how safe is safe enough" so that no additional regulatory requirements for greater safety would be needed. Requiring such a stringent standard as the QHOs as a minimum level of safety for advanced reactors appears to go well beyond the authority granted by the Atomic Energy Act that requires adequate protection of the public health and safety. We are unaware that the Commission has made such a demand for advanced reactors. Were the Commission to make such a demand, we would question the wisdom of doing so. By demanding such a stringent level of safety, our colleagues appear to be willing to forego great strides in safety that can be achieved with advanced plants if these plants fail to live up to what can only be viewed as an extreme safety standard.

I think I agree with this

The demands our colleagues appear to make on the safety of advanced reactors lack a critical dimension of practicality since we do not believe the technology now exists to do the calculations needed to compare a plant's safety profile to the QHOs. By the very definitions of the QHOs, such calculations would entail analyses of modes of operation only very crudely addressed today by most (fire risk, shutdown risk and natural phenomena risk) and the conduct of uncertainty analyses dealing with both parameters and models that to our knowledge have been done by no one.

Because of the limitations of risk assessment technology available today for the evaluation of the current fleet of nuclear power plants, surrogate metrics such as core damage frequency (CDF) and large early release frequency (LERF) have been introduced and widely used. Our colleagues seem to believe that there are known critical values of these surrogate metrics that mark the point at which a plant meets the QHOs. We know of no defensible analysis that establishes such critical values of these surrogate metrics. We are, of course, quite aware of very limited analyses considering only risk during normal operations that purport to show existing reactors meet the QHOs. Such limited analyses are simply not pertinent. They do not meet the exacting standards required by the definitions of the QHOs. Should defensible analyses ever be done, we are sure that they will show the critical values of the surrogate metrics are technology dependent. Indeed, more defensible analyses will show in all likelihood that better surrogate measures can be defined for advanced reactor technologies.

Our colleagues are sufficiently enamored with the existing surrogate metrics that they recommend these surrogates be enshrined on a level equivalent to QHOs. More remarkable, our colleagues want to establish critical values of the metrics that are a factor of ten less than the values they assert mark a plant meeting the rather stringent level of safety defined by the QHOs. They do this, apparently, for no other reason than the fact that clever engineers can design plants meeting these smaller values at least for a limited number of operational states. While we are willing to congratulate the engineers on their designs, we can see no reason why such stringent safety requirements should be made regulatory requirements to be imposed on the designers' efforts. Again, we worry that doing so may create unnecessary burdens that cause our society to sacrifice for practical reasons great improvements in power reactor safety simply because these improvements fall short of our colleagues unreasonably high safety expectations.

Though surrogate metrics have been useful, it is important to remember that they are only expedients. The full promise of risk-informed safety assessment will not be realized until it is possible to do routinely risk assessments of sufficient scope and depth so it is possible to dispense with surrogate metrics. Enshrining these surrogates along with the QHOs will only delay efforts to reach this preferred status.

The potential of our colleagues recommendations have to stifle new technology and forego improved safety reaches a crisis when they speak to the location of modern, safer plants on sites with older but still adequately safe plants. Our colleagues have no tolerance for a single older plant if a newer, safer plant is to be collocated on the site. They are willing to tolerate any number of similarly old plants on a site if a new, safer plant is not added to this site. We find this remarkable. Our colleagues' recommendations give no credit for experience with a site. They fail to recognize the finite life of older plants even when licenses have been renewed. We fear that our colleagues have failed to assess the integral safety consequences of their stringent demands on this matter. A very great concern is that our colleagues pursuit of ideals in risk avoidance may well arrest the current, healthy quest for improved safety among those exploring advanced reactor designs.

<u>References:</u>

1. U.S. Nuclear Regulatory Commission, SECY-05-130," Policy Issues Related to New Plant Licensing and Status of the Technology Neutral Framework for New Plant Licensing," dated July 21, 2005

2. U.S. Nuclear Regulatory Commission, "Safety Goals for the Operations of Nuclear Power Plants, Policy Statement," Federal Register, Vol. 51, (51 FR 30028), August 4, 1986

3. U.S. Nuclear Regulatory Commission, "Commission's Policy Statement on the Regulation of Advanced Nuclear Power Plants," 59 FR 35461, July 12, 1994

4. U.S. Nuclear Regulatory Commission, NUREG-1437, Volume 1, "Generic Environmental Impact Statement for License Renewal of Nuclear Plants," May 1996

Thus, in addition to the imposition of design goals for low CDF and LRF, it will be important to maintain sufficient defense in depth in the technology neutral framework.

We look forward to additional discussion with the staff on these issues.

Sincerely,

Graham B. Wallis
Chairman

Additional comments from ACRS Members Dana A. Powers and John D. Sieber

We disagree with our colleagues on the matter of this letter. The Commission has indicated a laudable expectation that future reactors will be safer than current reactors. The question that our colleagues should have addressed first is whether a quantitative metric is needed to substantiate this expectation. It is by no means obvious that such a metric is essential. We can well imagine future plants designed in conjunction with far more comprehensive probabilistic safety analyses that realistically address all known accident hazards during all modes of operation to a depth far greater than is attempted now for elements of the fleet of operating reactors. Our experience has been that whenever improvements are made in quantitative risk analysis methods, unforeseen, hazardous, plant configurations, systems interactions and operations become apparent. Hidden, these configurations, interactions and operations may arise unexpectedly with undesirable consequences. Revealed, they can be avoided often with modest efforts. This is exploitation of the full potential of quantitative risk analysis to achieve greater safety in nuclear power plants. It contrasts with the more effete pursuit of the "bottomline" results of PRA to compare with arbitrarily proliferated safety metrics.

Our objective should be to foster the voluntary development of quantitative risk analysis methods both in scope and depth in order to improve the safety of nuclear power plants. Fostering voluntary development of methods by nuclear community is especially important now when methods developments have stagnated at NRC relative to the situation a decade ago.

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IN RESPONSE, PLEASE
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November 8, 2006

MEMORANDUM TO: John T. Larkins
Executive Director, ACRS

FROM: Annette L. Vietti-Cook, Secretary /RA/

SUBJECT: STAFF REQUIREMENTS - MEETING WITH ADVISORY COMMITTEE ON REACTOR SAFEGUARDS, 2:30 P.M., FRIDAY, OCTOBER 20, 2006, COMMISSIONERS' CONFERENCE ROOM, ONE WHITE FLINT NORTH, ROCKVILLE, MARYLAND (OPEN TO PUBLIC ATTENDANCE)

The Commission met with the Advisory Committee on Reactor Safeguards (ACRS) to discuss the Committee's activities and current focus.

In ACTS ESP Lessons learned

As licensing under Part 52 continues the Committee should advise the Commission on effectiveness and efficiency of staff's implementation of lessons learned in areas it has reviewed, for example, the development of guidance documents for early site permits.

The Committee should provide its views to the Commission on staff's effort related to digital instrumentation and controls. The Committee should consider potential means for providing reasonable backup, if appropriate.

In ACTS Future Plant Designs

The ACRS should provide its views to the Commission with respect to staff's work on technology neutral licensing framework with a focus on ensuring the value of such an approach versus the development of a licensing framework for specific designs, such as a high temperature gas cooled reactor or a liquid metal cooled reactor.

The ACRS should provide the Commission with its recommendations and basis for areas in which NRC should perform additional long term research.

The Committee should work with the staff and external stakeholders to evaluate the different Human Reliability models in an effort to propose either a single model for the agency to use or guidance on which model(s) should to be used in specific circumstances.

cc: Chairman Klein
Commissioner McGaffigan
Commissioner Merrifield
Commissioner Jaczko
Commissioner Lyons
OGC

CFO
OCA
OIG
OPA
Office Directors, Regions, ACRS, ACNW, ASLBP (via E-Mail)
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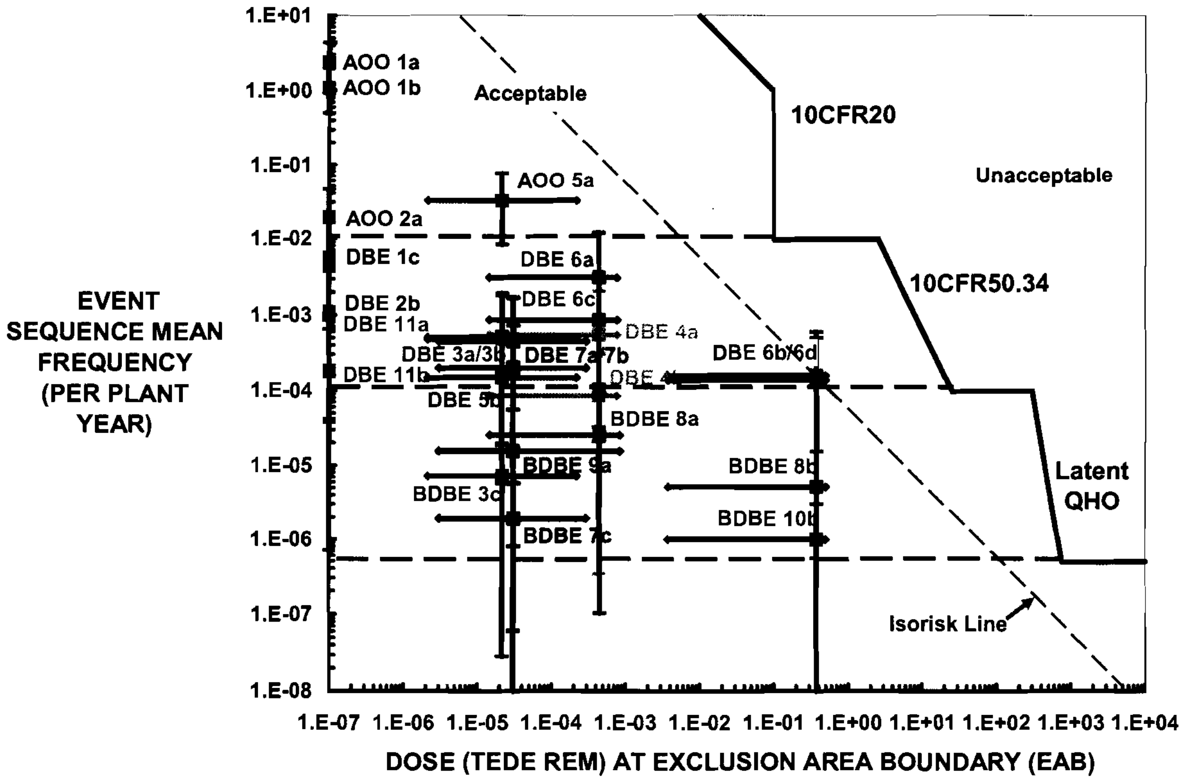
Technical Elements of a Risk-Informed, Technology-Neutral Design and Licensing Framework for New Nuclear Plants

1013582



Technical Elements of a Risk-Informed, Technology-Neutral Design and Licensing Framework for New Nuclear Plants

1013582

Technical Update, December 2006

EPRI Project Manager

S. Hess

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This is an EPRI Technical Update report. A Technical Update report is intended as an informal report of continuing research, a meeting, or a topical study. It is not a final EPRI technical report.

NOTE

CITATIONS

This document was prepared by

Scientech LLC
1740 Massachusetts Avenue
Boxboro, MA 01719

Principal Investigator
J. Chapman

This document describes research sponsored by the Electric Power Research Institute (EPRI).

This publication is a corporate document that should be cited in the literature in the following manner:

Technical Elements of a Risk-Informed, Technology-Neutral Design and Licensing Framework for New Nuclear Plants. EPRI, Palo Alto, CA: 2006. 1013582.

CONTENTS

REPORT SUMMARY

A critical component of long-term energy policy is the deployment of advanced nuclear plant designs. Many of the proposed advanced reactor concepts employ technologies that are significantly different from the light water reactor technologies used in the current operating fleet of reactors and from those which will be deployed in the near future. Because the current licensing framework in the United States has centered on the regulation of the existing light water reactor plants, a new framework that accounts for the differences inherent in the advanced reactor designs will be necessary. To support the attainment of future policy goals, the licensing framework developed for application to advanced reactor designs should possess the characteristics of providing maximum flexibility while incorporating the lessons obtained from the current generation of operating plants. Hence, the framework should possess the characteristics of being technology-neutral, risk-informed and performance-based.

Background

In the United States, the licensing framework for the current generation of operating nuclear reactors has evolved over several decades. This framework also is developed to permit effective regulation of the light water reactor designs of the current operating fleet. The advent of advanced reactor designs, some of which are not based on light water reactor technology, provides incentive and need for an improved and updated regulatory framework. A key industry initiative is the development of a framework that possesses the attributes of being technology-neutral, risk-informed and performance-based with corresponding processes (i.e. regulations and guidance) that support its implementation. Because the economics that will determine which of these reactor technologies provides the best choice for deployment is not clear, this new framework will need to be sufficiently flexible to encompass varying alternatives while incorporating the insights obtained from several decades of reactor operating experience. The framework structure also will need to account for advances in analytical techniques and methods across a broad spectrum of technologies.

Objectives

The objectives of this research are:

1. Identify and assess specific elements of the proposed technology neutral, risk-informed, performance based frameworks.
2. Develop a preliminary, revised framework based on the results of the reviews and evaluation conducted to meet objective 1.
3. Support the industry in developing responses to the NRC Advanced Notice of Proposed Rulemaking applicable to advanced reactor designs.

4. Provide recommendations in areas where additional development and testing would be most beneficial.

The scope for this research is limited to risk-informed elements aimed at supporting the development of a new approach to developing the safety case and support the licensing for advanced nuclear reactor designs.

Approach

To meet project objectives the following tasks were performed.

- Review and summarize existing licensing practices applicable to current light water reactor based plants and compare this to the frameworks currently proposed for implementation by domestic and international organizations, in particular the proposed frameworks developed by the United States Nuclear Regulatory Commission and by vendors developing advanced high temperature gas reactor technology.
- Identify and critically evaluate key technical elements of the proposed frameworks individually and as part of an integrated design and licensing process.
- Based on this review, develop a proposed integrated framework to address the issues identified.
- Recommend areas where additional development and testing would be most beneficial.

Results

The assessment conducted for this project suggests that the proposed frameworks being developed are promising, and provide a solid foundation for further enhancement. Use of PRA and other risk evaluations in the design process could improve upon the processes which were used for currently operating plants; and possibly on the process used for the certified advanced light water reactor designs. The proposed integrated framework developed as part of this project, is intended to provide a structure from which further progress can be made and a licensing basis developed.

EPRI Perspective

The deployment of advanced nuclear technology will serve as a critical element in an increasingly energy intensive and environmentally constrained global marketplace. The proposed integrated licensing framework developed during this research can serve as a catalyst to develop and implement a comprehensive licensing framework for these advanced nuclear reactor concepts.

Keywords
Risk-informed regulation
Technology neutral framework
Generation IV advanced nuclear plants

- Acknowledges that rules and guidance could be either technology neutral or specific to a technology.
- Establishes desired principles of the overall framework, which are (1) to provide assurance of adequate safety and security, (2) to assure regulatory openness and effectiveness, (3) to be TN, RI, and PB, (4) to address uncertainty and (5) to maintain adequate Defense-in-Depth (D-in-D).

Key differences between the proposed RI, TN, PB framework and the current approaches specified in 10 CFR Part 50 and 10CFR Part 52 are the selection process of licensing basis events (LBEs) and the selection process for determining the safety significance of systems, structures and components (SSCs) that prevent or mitigate these events. The proposed framework is characterized as being more risk informed than existing practice since it links the PRA analysis with the Licensing Basis (LB), LB acceptance criteria, and safety significant SSC selection and treatment. In Part 50 and Part 52 there is no direct link between LBEs and PRA.

The structure of the framework provided in Reference [1] is based on the following:

- A set of safety/security/preparedness expectations, which are ensured
- By Defense-in-Depth expectations, which are fulfilled
- By a set of protective strategies and certain design criteria and guidance, which are used
- Specification of a process for the development of licensing requirements.

Although Reference [1] addresses worker safety, security and preparedness, an assessment of these areas was not within the scope of this project. For Safety, the expectation is based on the Safety Goals (SGs) [13] and the report refers to the Policy Statement on "Regulation of Advanced Nuclear Power Plants" [21]. The framework is intended to provide assurance of enhanced margins of safety; and uses a combination of QHOs, a Frequency-Consequence approach, and deterministic practices to establish acceptance criteria to provide this assurance. An example Frequency - Consequence function, with bases, is provided, which is used with results obtained from a PRA. To be acceptable, PRA results must be below the function limits, meet QHOs in aggregate, and address uncertainties. PRA results are then used to select Licensing Basis Events (LBEs) and safety significant SSCs. In addition, Defense-in-Depth (D-in-D) principles must also be met and LBEs must be analyzed with appropriate conservatism.

In the proposed NRC framework the Defense-in-Depth principles established are the following:

1. Consideration of intentional (e.g. security-related) as well as inadvertent (e.g., random failure of SSCs and human error) events.
2. Providing both accident prevention and mitigation capability.
3. Ensuring key safety functions (KSFs) are not dependent upon a single element of design, construction, maintenance or operation.
4. Consideration of uncertainties in equipment and human performance.
5. Providing for alternative capability to prevent unacceptable releases.
6. Siting considerations.

3
OVERVIEW OF FRAMEWORK ELEMENTS

3.1 Selection of Integrated Reference Framework

There are different visions for a revised design and licensing framework applicable to advanced (i.e. Generation III "Plus" and IV) reactors. The dominant visions are represented by activities underway at NRC, at two gas cooled reactor vendors (PBMR (Pty) Ltd. and Areva), and by ANS. Reference [1] is the most recent work published NRC; References [3, 4, 5 and 20] are the most recent, available work by the gas cooled reactor vendors. Additionally, work is continuing on an ANS Standard; however, this is not complete and the proposed standard has not been published.

In this project, the key features of these various proposed frameworks were reviewed and combined into an integrated reference framework. This integrated reference framework includes elements that are intended to provide the roadmap for successfully designing and licensing an advanced reactor design using a TN, RI, PB approach. We note that this discussion of the framework does not address all design and licensing aspects; for example design codes (such as are currently available from ASME), relevant codes of federal regulations (CFRs), new CFRs which would be desired, the specific form and content required for licensing submittals, and other licensing topics such as submittal and approvals of topical reports (e.g., PRA and safety analysis methods) are not addressed. Each of these areas would need to be addressed during the development and application of a consensus framework.

3.1.1 NRC Draft Framework

The NRC draft framework is described in Reference [1], which currently is issued as a draft report. This proposed framework is intended to support development of a new 10 CFR Part 53 that would be an alternative to 10 CFR Part 50 or 10 CFR Part 52. Even if a new 10 CFR Part 53 is not published, the content of Reference [1] would be expected to exert significant influence during the review of reactor licensing applications, for example, the pre-application submittals from PBMR (Pty) Ltd (References [3, 4 and 20]). We incorporated the elements of this framework into the integrated reference framework used in this report.

Key characteristics provided in Reference [1] include:

- Requires the use of PRA in the licensing process. PRA analysis is required and would become part of the licensing bases.

Task 2: Identify and critically evaluate key technical elements of the proposed frameworks individually and as part of an integrated design and licensing process.

Each identified element should result in some benefit to the design and licensing process, and each will most likely have some cost and drawbacks. For example, some elements might favor one reactor type over another, some might overly complicate the design process or result in requirements that might be hard to sustain during operation and some might be technically efficient but be too different from existing reactor regulations to be practical.

The evaluation conducted for this study was limited in scope to the impact on nuclear safety; the potential impacts on cost and reliable plant operation were not addressed. The evaluation considered the bases provided for each of the key technical elements and a comparison was made between the frameworks and comparable elements in existing practices (where possible). This evaluation focused on the framework and did not attempt to assess all of the processes which would be needed to implement it. For example, the quality of the PRA needed, or the need for new PRA methods or data, were not examined.

This task was supported by table top exercises when deemed appropriate to support the assessment and provide a validation of conclusions. For example, fundamental safety principles and F-C functions were evaluated using table top exercises.

Task 3: Develop preliminary, new "integrated reference framework."

Based on the results of Task 2, proposed changes to the conceptual frameworks to address the issues identified were developed. These changes are preliminary, in that they have not been fully developed, tested or subjected to extensive review.

The changes identified serve as an initial step to support work necessary to complete the development of a framework which would have a full technical basis, is stable and repeatable, and meets or exceeds the intent of existing regulations, policies and practices.

Task 4: Recommend areas where additional development and testing would be most beneficial.

Recommendations were identified in several areas. The recommendations can be grouped into the following categories: (1) further development of the framework necessary to address the assessment findings; (2) testing and validation necessary to support further development and refinement of the framework; (3) reaching consensus on the fundamental aspects of the framework and its elements; and (4) reaching agreement on key, fundamental terms.

2
APPROACH

To meet project objectives the four tasks described below were performed.

Task 1: Review and summarize existing, mostly deterministic, licensing practices applicable to current LWR based plants and compare this to the proposed TN, RI, PB frameworks that are under consideration for implementation by domestic and international organizations.

This review was comprised of the following specific reviews:

Subtask 1.1: Current U.S design and licensing practices;
Subtask 1.2: Practices employed on the advanced, certified designs;
Subtask 1.3: IAEA guidance; and
Subtask 1.4: Risk-Informed, Performance-Based, Technology-Neutral and Technology-Specific concepts under development by NRC, NEI, NSSS vendors and ANS.

Subtasks 1.1 and 1.2 were conducted in order to develop a baseline for assessing the frameworks under development. A regulatory expectation for future reactors is that their level of safety will at least be equal to, or exceed, that of LWR plants currently in operation. A summary of existing practices for addressing the level of safety and the results of these practices is provided in Section 4. The structure of this summary was aligned to the frameworks under development, where possible, to support a comparison to them. This summary was developed on the basis of experience, simple table top exercises (scoping evaluations) and a review of the design control documentation (DCD) for advanced LWR designs currently certified by the NRC.

For subtasks 1.3 and 1.4, references [1-11, 14 and 20] were reviewed. On the basis of this review, the basic structure of a framework and elements common to NRC [1], the gas cooled reactor nuclear steam supply (NSSS) vendors (references [3, 20]), and ANS, with modifications identified during the conduct of Task 2 and implemented in Task 3, was adopted as the standard approach for this project. The frameworks reviewed are relatively mature, have existing constituencies (NRC and the NSSS vendors) and encompass, at a fundamental level, similar (although not identical), relevant features. Although alternative ideas on structuring the framework have been proposed, the proposed approaches reviewed for this report have benefited from considerable input by many stakeholders and constitute a workable approach. With further development, as addressed in Sections 3 – 5 below, agreement on a common framework and criteria is viewed to be achievable.

1.2 Purpose and Objectives

This project was initiated to develop the technical and licensing knowledge base needed to support the following purposes: (1) dialog between industry stakeholders and NRC on the development of a TN, RI and PB framework for use in developing new regulations and accompanying implementation guidance (including responding to the ANPR noted above) and (2) implementation of the Next Generation Nuclear Plant (NGNP) project underway at the Department of Energy (DOE).

To meet these purposes, the following objectives were established for this project:

1. Identify and assess specific elements of the proposed frameworks.

2. Develop a preliminary, revised framework based on the results of the reviews and evaluation conducted to meet objective 1.

3. Support NEI in developing responses to the ANPR.

4. Provide recommendations in areas where additional development and testing would be most beneficial.

The scope for this project is limited to risk-informed elements aimed at supporting the development of a new approach to developing the safety case for advanced nuclear reactor designs. The results of this effort are provided in this report. Section 2 describes the project approach including the specific tasks performed, Section 3 provides an overview of the proposed framework elements, Section 4 presents the results obtained, and Section 5 discusses conclusions and recommendations.

The potential alternative frameworks currently under development have different features, and in some technical elements possess differences that are significant. These frameworks, however, have two common features which differ significantly from existing approaches. These features are:

1. Extensive use of Probabilistic Risk Assessment (PRA) in the design and licensing process (for example, to establish licensing basis events (LBEs) and to identify safety significant systems, structures and components (SSCs).)
2. Explicit Use of Quantitative Health Objectives (QHOs) and Frequency-Consequence (F-C) acceptance criteria (these have also been referred to as Top Level Review Criteria (TLRC)).

In contrast to the existing, mostly deterministic licensing structure used for the current generation of LWRs, PRA technology and results obtained from its application are an integral part of the proposed frameworks under development. One example of this use of PRA is in the identification of licensing basis events which address the full spectrum of possible sequences from anticipated operational occurrences (AOOs) to beyond design basis events (BDBEs). In the proposed frameworks, QHOs (established in the Safety Goal Policy Statement [13]), frequency – consequence functions, and other considerations (such as defense-in-depth) are proposed for use as acceptance criteria in demonstrating the safety case.

In these new frameworks the allowable consequences are permitted to increase as the frequency of a sequence decreases. Appropriate PRA calculations are used to demonstrate the F-C limits are not exceeded. The approach to define the design basis accident (DBA) for reactor siting (reference 10 CFR Part 100 - Reactor Site Criteria) used for the licensing of the current generation of plants (which is equivalent to a core melt event with an intact containment), and its impact on containment systems design and plant site location approval, is being reconsidered. Primarily, in the new framework, the maximum credible challenge to containment is being reexamined on the basis of the safety characteristics of the non-LWR based reactor technology that currently is under development. For example, for certain advanced reactor designs the maximum proposed challenge does not involve severe core damage.

The policy and technical topics and decisions associated with completing these frameworks are challenging as the approach to developing the safety case differs considerably from existing practices. It is recognized that the frameworks under development, and their technical bases, are not yet complete or agreed upon between NRC and industry stakeholders. Further development, testing, refinement based on testing, and concurrence are needed. As a means to engage the NRC on a specific design, PBMR (Pty) Ltd. has submitted pre-application papers (for example, see References [3, 20]) which use such frameworks as a part of their safety case. The objective of these pre-application submittals is to develop technical and policy positions and bases with the intention of reaching agreement on the criteria and content that would be acceptable for construction and operation of an advanced reactor design.

1 INTRODUCTION

1.1 Background

In the United States, the licensing framework for the current generation of operating nuclear reactors has evolved over several decades. Since reactors currently in commercial operation in the US are all light water reactor (LWR) based technology, the current framework of regulations and associated implementation guidance documents are structured to support effective regulation of this technology. In addition, because commercial nuclear power plants are complex engineered systems, the majority of the existing regulations are based on deterministic engineering analyses. As experience with the operation of these facilities has accumulated, the regulatory structure has begun to migrate to one that is risk-informed and performance-based. Examples of these types of regulations and processes include the Maintenance Rule and the reactor oversight process (ROP).

The advent of new reactor designs, some of which are not based on LWR technology, provides incentive and need for an improved and updated regulatory framework. A key industry initiative is development of a framework that possesses the attributes of being technology-neutral (TN), risk-informed (RI) and performance-based (PB) with corresponding processes (i.e. regulations and guidance) that support its implementation. Because the economics that will determine which of these reactor technologies provides the best choice for deployment is not clear, this new framework will need to be sufficiently flexible to encompass varying alternatives while incorporating the insights obtained from several decades of reactor operating experience. The framework structure also will need to account for advances in analytical techniques and methods across a broad spectrum of technologies.

In recognition of this situation, the Nuclear Regulatory Commission (NRC) and several industry stakeholders have ongoing activities to develop a new framework from which these objectives can be accomplished. NRC has been conducting research in this area for several years, and in July 2006 published a working draft report [1] and an Advanced Notice of Proposed Rulemaking (ANPR) for 10 CFR Part 53. Industry activities have been led by the Nuclear Energy Institute (NEI), EPRI (in support of NEI) and the nuclear steam supply system (NSSS) vendors. In addition, the American Nuclear Society (ANS) has a related standard activity for gas-cooled reactors.

LIST OF TABLES

LIST OF FIGURES

The report notes that Defense-in-Depth would be applied as a means to address uncertainties regardless of the level of safety determined using a PRA. Since an underlying principle of the framework is that accident prevention alone cannot be relied upon to reach an acceptable level of safety, capabilities to mitigate accidents (item 5 above) are also required in this framework.

In this framework, protective strategies also are established. The following were identified in [1]:

1. Physical protection from hazards (e.g., radiolpgical and chemical) for workers and the public,
2. Stable operation (by limiting the frequency of events that can upset plant stability and challenge safety functions),
3. Protective systems (by providing sufficiently available, reliable and capable SSCs, including human actions, on the basis of the frequency and significance of the challenge),
4. Barrier integrity (by providing adequate barriers for workers and public),
5. Protective actions (by providing severe accident management capability and emergency planning).

The two principle deterministic D-in-D elements of the framework are implementation of the protective strategies and the D-in-D principles. The probabilistic defense in depth element of the framework is the use of PRA techniques and other logical processes to search for and identify unexpected scenarios, to address uncertainty, and to further assure adequate D-in-D, including adequate safety margin.

Design criteria and guidance are then established using the expectations, D-in-D principles and protective strategies summarized above. The design criteria and guidance can be represented by the elements shown in the first column of Table 3-1 below. The second column represents the corresponding elements from the framework proposed by the gas reactor NSSS vendors [3, 4, 5, 20] which are discussed next.

3.1.2 Gas Cooled Reactor Framework

Based on References [3, 4, 5, 20], and knowledge of the ongoing standards development activities by ANS, key elements of this framework are provided in Table 3-1. In this table, key differences between these proposed frameworks and that provided in Reference [1] are noted in **bold font.** The elements contained in the different proposed frameworks generally are comparable, but implementation varies, and in some elements the differences are significant. A brief summary comparison to the NRC proposed framework described in Reference [1] is provided below. Section 4 addresses key differences in more detail.

<u>Top Level Review Criteria:</u> Both NRC and the gas cooled reactor frameworks use the QHOs and an F-C approach. However, the approaches to applying the QHOs, the approaches used to develop and apply the F-C function, to assessing aggregated risk, and to addressing siting criteria are significantly different. An example of a key difference is development and use of the F-C function. In the framework proposed by the NSSS vendors [3, 20], the F-C function permits acceptance of much higher consequences for most of the frequency ranges than in the NRC proposed framework [1]. Additionally, the function is "populated" differently in the two

frameworks with the NSSS framework using "families" of sequences as compared to evaluation for specific sequences in the NRC proposed framework.

Safety functions are addressed in a comparable manner in both frameworks.

PRA use is comparable in both frameworks.

Comparison of the PRA results to the F-C function is different in that "families of sequences" rather sequences at the system level is used.

SSC Classification is similar but not identical; and, as the approach in Reference 20 is based on a specific design, additional detail, including classification criteria (i.e., safety class and non-safety class), is provided.

LBEs are initially established in a manner similar to Reference [1] (other than using families of sequences rather than system level sequences.) However, whereas Reference [1] applies a grouping process to LBEs, and then addresses uncertainty, in Reference [3], the grouping results in anticipated operational occurrences (AOOs), design basis accidents (DBAs), and beyond design basis events (BDBEs) being established, which are addressed much differently. In Reference 1, DBAs are referred to as LBEs, and the F-C function limits must be met for these events. For the gas reactor approaches under development, consequence criteria must be met for AOOs and DBAs, independent of event frequency. Because the consequence criteria used are selected as the upper limit from the F-C function, this approach is less restrictive than that proposed in Reference [1].

Defense-in-Depth is addressed in both the ongoing ANS standards activity and in Reference [14]. Documentation comparable to that specified in References [3] and [20] had not been submitted to NRC when this report was prepared. Therefore, a direct comparison to the proposed framework requirements specified in Reference [1] was not completed. This area can be considered in the future if appropriate.

Deterministic Events resulting in a severe challenge to confinement integrity do not appear to be included in the NSSS vendor proposed framework. Inclusion of such an event in a TN, RI, and PB framework is the subject of the ANPR and a position is not provided in this report. The approach provided in Reference [1] is summarized in Section 4.

TABLE 3-1: DESIGN CRITERIA AND GUIDANCE (ELEMENTS) -COMPARISON-	
Reference [1] Criteria and Guidance (NRC)	**References [3 ,20] Criteria and Guidance (GCR NSSS Vendor)**
Acceptance criteria are established (QHOs, F-C function, **aggregated risk profile,** stable operation, D-in-D (e.g., barriers and prevention and mitigation), and **siting criteria.**)	Top Level Review Criteria are established. (QHOs, F-C function, and establishment of AOO, Design Basis Event (DBE), BDBE and DBA acceptance criteria.)
Safety functions and associated design criteria and operating limits are established.	Safety functions and associated design criteria and operating limits are established.
Full scope PRA, including consideration of uncertainty, is conducted.	Full scope PRA, including consideration of uncertainty, is conducted.
PRA results are confirmed to meet F-C function limits, on a **sequence** specific basis. QHOs are verified to be met.	PRA results are confirmed to meet F-C function limits, on a **family** basis. QHOs are verified to be met.
Safety significant SSCs, which are those SSCs needed to meet F-C limits and other deterministic criteria, are established.	**Safety significant SSCs are established using a RI approach which results in several classification categories.**
LBEs, which are representative of all PRA sequences and which only credit safety significant SSCs, are established.	**AOOs and DBAs,** which are representative of all PRA sequences and which only credit safety significant SSCs, are established.
LBEs are demonstrated to meet F-C function and aggregated risk at high confidence (e.g., 95% confidence levels for both frequency and consequences.)	**AOOs and DBAs are demonstrated to meet deterministic consequence acceptance criteria, rather than F-C limits. BDBEs are demonstrated to meet QHOs only.**
D-in-D, which includes safety margins, is assessed. Here specific deterministic measures are applied which vary depending on the frequency of an LBE (e.g. no barrier failure for frequent events and availability of at least one intact barrier for infrequent events.)	D-in-D, which includes safety margins, is assessed. However, the specific approach was not available at the time this evaluation was conducted.
An event, on a deterministic basis, is established to demonstrate that 10 CFR Part 100 siting criteria are met. (E.g., an event involving fuel failure and RCS integrity failure which could result in exceeding 10 CFR Part 100 if not for confinement by the containment.)	**Appears not to be included**
Treatment for SSCs on the basis of significance and reliability assurance practices is established.	Treatment for SSCs on the basis of significance and reliability assurance practices is established.
Reliability Assurance Program and Monitoring and Feedback Capability and Processes are established.	Reliability Assurance Program and Monitoring and Feedback Capability and Processes are established.

3.2 Integrated Reference Framework and Elements

The "integrated reference" framework developed as a result of the research conducted for this project is depicted in Figure 3-1. As shown in Table 3-2, this framework represents the key elements included in Reference [1] and in the various gas cooled reactor references (e.g., References [3, 5, 20]). (Note that information for References [1, 3, 20] depicted in Table 3-2 is based on the summary provided in Table 3-1, but with minor reordering.) A summary of each element is provided below. In the next section selected elements are discussed in more detail and reviewed to identify issues where additional development and testing are warranted. Candidate changes are provided, and in Section 5 conclusions and recommendations are presented.

TABLE 3-2: FRAMEWORK ELEMENTS

Framework Element in Figure 3-1	Reference [1] (NRC)	References [3, 20] (GCR NSSS Vendor)
Fundamental Safety Principles (FSPs) and Fundamental Design Principles (FDPs)	Acceptance Criteria	Top Level Review Criteria
F-C Function and QHOs	Consistent, but developed and used differently	Consistent, but developed and used differently
Identification and Classification of Safety Functions	Consistent	Consistent
Performance of Probabilistic Risk Assessment (PRA) and Other Risk Evaluations and Population of Frequency-Consequence Function	Consistent, but does not address "other risk evaluations."	Consistent, but does not address "other risk evaluations," and uses families of sequences.
Comparison to Relevant Quantitative Criteria	Comparable	Comparable
Uncertainty Analysis and Comparison to Relevant Quantitative Criteria	Comparable	Comparable
Selection of Safety Significant Systems, Structures and Components (SSCs)	Comparable	Comparable
Licensing Basis Events (LBEs) and Design Basis Accidents (DBAs)	Consistent	Developed and used differently
Defense-in-Depth, Safety Margins and Other Acceptance Criteria	Similar	The specific approach was not available for review during this project.
Cumulative Distribution of Risk as Measured by Complementary Cumulative Distribution Function (CCDF)	Aggregate risk is addressed using a different approach.	Aggregate risk is addressed using a different approach.
Selection of Treatment for SSCs	Similar	Similar
Development of a Reliability Assurance Program (RAP)	Similar	Similar
Monitoring and Feedback	Similar	Similar

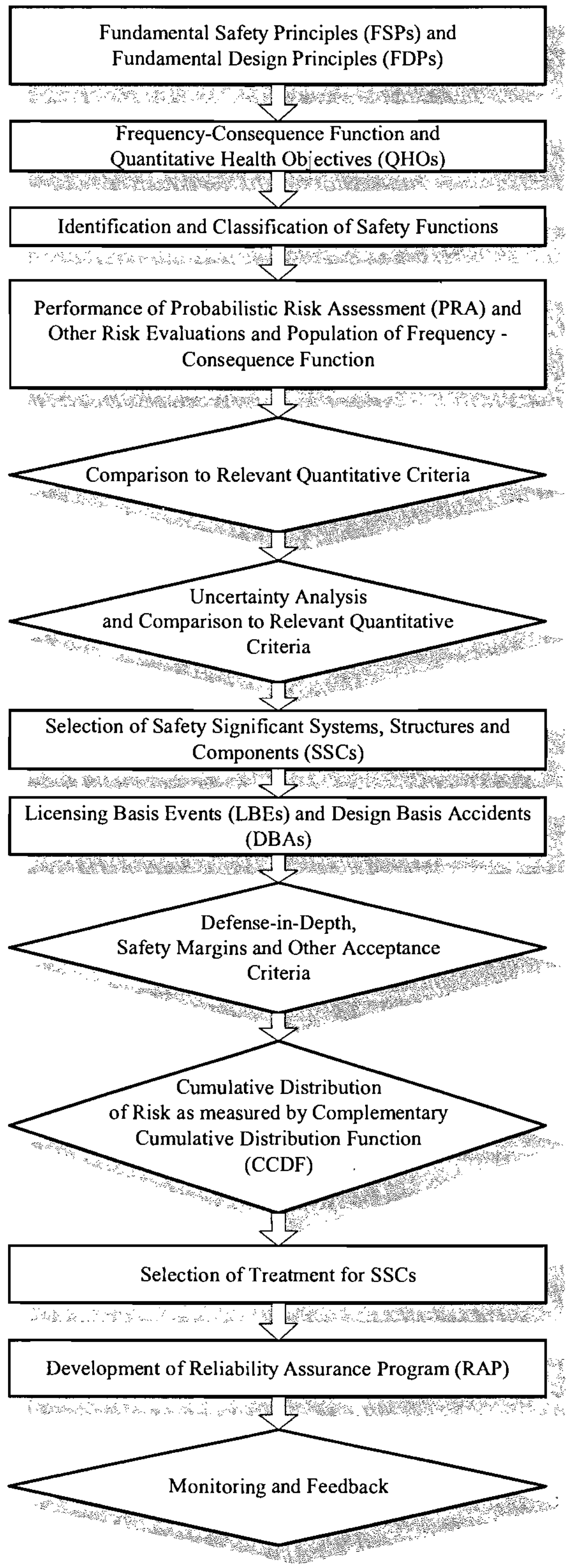

Figure 3-1: Framework Elements for Assessment

3.2.1 Fundamental Safety and Design Principles

In this element Fundamental Safety Principles (FSPs) and Fundamental Design Principles (FDPs) are developed (see below for discussion of these terms). In the proposed integrated reference framework developed in this report, these principles are intended to replace the criteria summarized in the first row in Table 3-1.

Reference [1] and the frameworks proposed for the gas cooled reactor designs use comparable concepts. For example, as discussed above, the Reference [1] framework is based on development of a set of safety expectations (e.g. QHOs and an F-C function); which are ensured by application of D-in-D principles using a set of protective strategies and corresponding design criteria. The proposed gas cooled reactor framework uses Top Level Review Criteria, Top Level Design Criteria, Defense-in-Depth, and other considerations which are similar but are implemented differently.

In this project the principles and corresponding criteria were combined into fundamental safety and associated design principles. This distinction is intended to establish and apply common terminology. Fundamental Safety Principles (FSPs) are intended to provide, at a high level, the safety expectations of a design and the bases for these expectations. That is, what shall be achieved and why. Fundamental Design Principles (FDPs) are based on the FSPs, and, when combined with the other elements of the framework, are used to describe how the design will be demonstrated to have achieved the FSPs. That is, the FDPs are intended to demonstrate how the FSPs are to be achieved. Note, the FSPs and FDPs are limited to application to public safety; worker protection and security are not addressed here.

Examples of FSPs are:

- Meeting the QHOs with margin, in order to meet Commission policy expectations.
- Meeting an acceptable frequency-consequence function, to assure allowable consequences versus frequency are equal to or better than current plants.
- Having sufficient barriers, to prevent and mitigate the release of radioactive and other hazardous materials. (Please note in this report, however, we focus on nuclear safety.)
- Achieving stable operation, so as to minimize challenges to mitigation systems.
- Having a balance between prevention and mitigation (on the basis of inherent safety characteristics), to assure safety is not based solely on either prevention or mitigation.
- Including safety margins to account for uncertainties.

An example of an FDP for the QHO FSP is as follows: A PRA and other risk evaluations, of sufficient scope and quality, and considering uncertainties and all hazards and operating modes, for the spectrum of possible consequences, shall be developed to demonstrate the QHO FSP is achieved.

Section 4 provides a discussion of the candidate FSPs and FDPs developed as part of this research, and provides a comparison to Reference [1], the gas cooled reactor approach, and to IAEA guidance provided in References [9, 10, 11].

3.2.2 Frequency-Consequence Function and QHOs

In this element an F-C function, the process for using the F-C function, and the approach for using QHOs are developed. This element of the integrated reference framework developed in this project is consistent with Reference [1] and the gas cooled reactor framework, except that candidate changes are included to address the issues identified.

Quantitative Health Objectives (QHOs) and Frequency-Consequence (F-C) functions (dose vs. frequency of an event sequence or event sequence family) are key quantitative criteria for determining safety adequacy in the frameworks under development. In these frameworks QHOs and an F-C function shall be demonstrated to be met. This is achieved by specific PRA evaluation of event sequences (or event sequence families), for the entire spectrum of postulated events (from those that are anticipated to occur to those that are believed to be extremely unlikely). These evaluations are conducted to demonstrate that the results are, in all cases, below the specific F-C function limits. The PRA results in aggregate are used to demonstrate that the QHOs are met.

In the current licensing framework, QHOs have been used to support decision making for existing plants. To facilitate the development and implementation of RI, PB regulation applicable to the current generation of LWRs, surrogate measures such as core damage frequency (CDF) and large early release frequency (LERF), and corresponding values, have been developed and serve as the metrics against which RI decisions are made. However, neither QHOs nor their surrogates are intended to provide the sole basis for demonstrating the safety case; and meeting the QHOs is not sufficient to make a safety case. Thus, an assessment of the use of QHOs is included in Section 4.

The F-C functions under development are based on the principle that the frequency of an event sequence should decrease as the consequences of the event sequence increases. Currently operating LWR plants meet this principle in a general sense for event sequences within the design basis. For example, an event sequence involving a loss of feedwater initiating event combined with the limiting single active failure is expected to result in insignificant radiological consequences; whereas a small break LOCA (SBLOCA) initiating event combined with a limiting single active failure is permitted to result in small, but not significant, radiological consequences.

F-C functions have not been applied as a part of the design and licensing process to currently operating plants and their development would pose a significant challenge. Figure 3-2 provides an example of a sample F-C function for a proposed advanced gas reactor design, which was extracted from Reference [3].

The approach to developing and using an F-C function and the results of the PRA varies significantly between that described in Reference [1] and the framework proposed for the gas cooled reactors (specifically the approach documented in Reference [3], which is similar to that described in Reference [5] and used here for the purposes of comparison). Section 4 summarizes the two approaches, compares them, and based on the evaluation, proposes significant but achievable changes for further evaluation and testing.

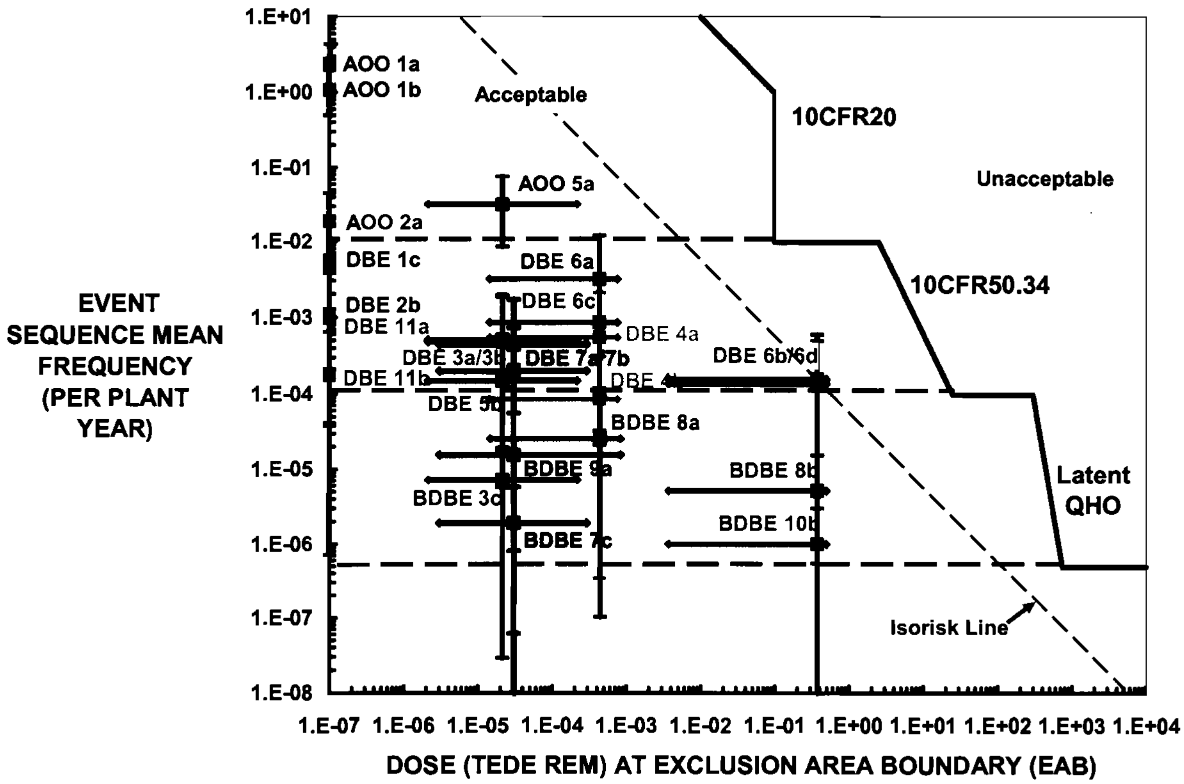

Figure 3-2: Example F-C Function for AOOs, DBEs and BDBEs (Figure 9 of Reference [3])

3.2.3 Identification and Classification of Safety Functions

In this element safety functions (SFs), success criteria (SC), and operating limits (OLs) are developed. This element of the integrated reference framework is consistent with Reference [1] and the proposed gas cooled reactor frameworks. No changes are proposed. In this report example SFs and SC are provided in Section 4 based on the gas cooled reactor technology literature. An assessment is not conducted since this element is straightforward and is a fundamental aspect of the plant design.

SFs and SC are needed to design the SSCs necessary to achieve them. Once the design is selected, they are needed to conduct the PRA, to determine approporiate design basis accidents (DBAs) and to establish acceptance criteria for safety analyses. Typical safety functions include reactivity control, reactor pressure control, reactor coolant inventory control, heat removal, and barrier integrity. To be more useful to implement the framework on a particular reactor design, it is envisioned that the generic characterization would be modified to address design-specific features. Finally, we note that the success criteria described here are referred to as Top Level Design Criteria (TLDC) in some of the references reviewed, and these terms are intended to have the same meaning.

In addition to developing SFs and SC, operating limits (OLs) need to be established. These would include Technical Specifications on Limiting Conditions for Operation (LCOs) and other administrative limits. Examples include licensed plant power level and associated uncertainty, fuel integrity limits, Helium Pressure Boundary (HPB) integrity (gas cooled reactor design), radioactivity levels in the primary coolant, and heat exchanger integrity. These limits are necessary to support both PRA and DBA analyses.

3.2.4 Performance of PRA and Other Risk Evaluations and Population of Frequency-Consequence Function

In this element a PRA and other risk evaluations are conducted, and results are provided in a form for comparison to the F-C function and other quantitative criteria. The comparison of risk assessment results to the criteria is conducted in the next element. The risk assessment element of the integrated reference framework also is consistent with the proposed approach described in Reference [1] and the gas cooled reactor frameworks, with the addition of proposed features to address the identified issues. A review of PRA quality and methods was not included in this project.

PRA is proposed to be used extensively in the frameworks under development that were included in this review. These frameworks, if not changed, would require a full scope, level 3 PRA (for varying dose levels from benign to severe – if possible) for all hazards. As the PRA is used to establish LBEs, these analyses would most likely require documentation and review in consonance with 10 CFR Part 50 Appendix B quality assurance (QA) requirements. Additionally, the risk assessments also are proposed to be subjected to an independent peer review. Key applications during design include generating a complete set of accident sequences, developing a rigorous accounting of uncertainties, evaluating conformance with the QHOs, evaluating conformance with the F-C function, identifying and characterizing LBEs, and identifying and characterizing safety significant SSCs.

Other applications noted in Reference [1] include supporting Environmental Impact Statement (EIS) and Severe Accident Mitigation Design Alternatives (SAMDA) analyses, risk informing inspections during fabrication and construction, supporting the determination of staffing requirements, supporting the development of the Technical Specifications (or equivalent), supporting the development of inspection, testing and preventative maintenance, supporting the development of procedures and training, supporting the development of emergency preparedness, assessing and managing operational risk, assessing and managing plant changes, and monitoring SSC performance. Several of these other applications would be expected to be optional and not a requirement. These applications are not addressed in this report..

As result of this review, several issues were identified and candidate changes are provided for inclusion in this integrated reference framework.

3.2.5 Comparison to Relevant Quantitative Criteria

In this element the results of the PRA are compared to the quantitative criteria described in Section 3.2.2. This comparison would evaluate the results of the risk assessment against the quantitative criteria developed from the FSPs and FDPs. If the criteria are not achieved, plant design and/or operational characteristics would be changed as necessary to ensure the criteria are met.

This element of the integrated reference framework developed in this project is consistent with the frameworks described in Reference [1] and the proposed gas cooled reactor frameworks, with the addition of proposed features to specifically address FSPs and "other risk evaluations."

3.2.6 Uncertainty Analyses and Comparison to Relevant Quantitative Criteria

In this element uncertainties are addressed so that the corresponding results can be compared to relevant quantitative criteria used in the above element. This element of the integrated reference framework also is consistent with the frameworks described in Reference [1] and the gas cooled reactor work, with a few small, but important changes.

Uncertainties would be addressed in the calculation of both frequencies and consequences. Parameter, model and completeness uncertainties would be addressed. Uncertainties have been addressed in establishing the licensing basis of current plants. However, with limited exceptions (e.g., determining the frequency of external hazards and addressing pressurized thermal shock (PTS) in PWRs), most uncertainty analyses for currently operating plants involve the consequences of sequences. The frequency of most sequences, and the associated uncertainties, is not addressed in detail. Instead, for the licensing basis applicable to current generation plants, the frequency has been addressed using good design practices, monitoring and feedback, and, recently by the application of specific performance indicators (e.g. ROP and INPO performance indicators.)

3.2.7 Selection of Safety Significant SSCs

In this element safety significant SSCs are identified. In Reference [1], safety significant SSCs are defined as those SSCs which are needed in order to meet the F-C function limits and other deterministic requirements. In Reference [20], a similar, but not identical, approach is used. Since this approach is based on a specific design, additional detail, including specific classification criteria, is provided. For both frameworks, the objective is to identify those SSCs which should be subject to special treatment due to their safety significance.

A few issues were identified during this review. This element of the integrated reference framework is consistent with Reference [1] and the gas cooled reactor frameworks, with the addition of proposed features to address the issues identified.

3.2.8 Licensing Basis Events (LBEs) and Design Basis Events (DBAs)

In Reference [1], LBEs are identified and analyzed conservatively for comparison to F-C function limits. In References [3] and [20], LBEs are also identified and analyzed, but the approach is significantly different. In the approach proposed for the gas reactor applications, LBEs are subdivided into AOO, DBA and BDBE categories. For AOO and DBA categories deterministic analyses are conducted to demonstrate acceptability against specific, frequency independent, consequence criteria; while the BDBEs are integrated for comparison to the QHOs.

This element of the integrated reference framework described here is consistent with the approach in Reference [1], with the addition of proposed features to specifically address issues identified in the review.

In Reference [1], LBEs consider only safety significant SSCs to be available to mitigate an initiating event. Representative sequences are selected based on a grouping of similar sequences from the PRA. The groups so defined must meet the F-C function limits at the 95% confidence level, where the highest frequency and consequence level of the sequences within the group are used to define a representative sequence. In addition, each representative sequence must meet deterministic criteria, which will be discussed in Section 3.2.9. In Reference [1], LBEs are characterized as being similar to DBAs in the current licensing framework (such as are documented in Section 15 and other sections of a Final Safety Analysis Report (FSAR)).

The proposed approach developed by the gas cooled reactor NSSS vendors is different. In this framework, all sequences from the PRA are grouped into families and are referred to as LBEs. These LBEs are then subdivided on the basis of the frequency of a family of sequences into AOO (frequency >1E-2/yr.), DBE (frequency between 1E-2/yr. and 1E-4/yr.) and BDBE (frequency <1E-4/yr) categories. Deterministic DBAs are selected from the DBEs, using a grouping process, by assuming that only SSCs classified as safety-related are available to perform the safety functions required to meet 10 CFR 50.34. The DBEs are reanalyzed deterministically with only the safety-related SSCs responding in a mechanistically conservative manner to demonstrate that the mean consequence of each DBA is less than 25 rem total effective dose equivalent (TEDE) limit at the exclusion area boundary (EAB). AOOs are treated in a similar manner but use a lower consequence limit of 100 mrem TEDE.

The impact of the differences between Reference [1] and the approach being developed for the gas cooled reactors is potentially significant and is examined in Section 4.

3.2.9 Defense-in-Depth, Safety Margins and Other Acceptance Criteria

In this element of the framework an evaluation of Defense-in-Depth, including safety margins, and other acceptance criteria, is conducted. Defense-in-Depth and safety margins are key elements of the existing approach to nuclear plant design and operational safety, and are explicitly addressed in the frameworks being developed. In this report, only the framework discussed in Reference [1] is reviewed, as specific, current information on the gas cooled reactor approach was not available.

This element is addressed in detail in Section 4 and is consistent with Reference [1], with the addition of proposed features to specifically address issues identified in the review.

3.2.10 Cumulative Distribution of Risk as Measured by a Complementary Cumulative Distribution Function (CCDF)

In this element the cumulative distribution of risk is determined in the form of a risk profile represented by a CCDF and other measures. In this report a sample CCDF is provided which is based on a table top review of existing acceptance criteria or metrics (e.g. Reference metrics used in LWR PRAs (e.g., CDF and LERF)), operating experience, and the results of traditional safety analyses. In contrast to current RI approaches, the F-C functions used in both Reference [1] and the approaches under development by the gas cooled reactor NSSS vendors are not "risk curves," and therefore the acceptability of the aggregate (i.e. cumulative) result of the sequences which populate the F-C function is indeterminate. Reference [1] and the approaches under development for the gas cooled reactors (e.g., Reference [3]) recognize this feature of an F-C function, when this function is applied on a sequence-specific or "family" basis. Approaches to address this characteristic of the F-C function are provided in these references.

Based on the evaluation conducted for this report, these approaches were determined to be potentially incomplete. The approach described in Reference [1] includes steps to develop integrated risk measures in addition to a comparison to the QHOs. These measures may be sufficient, but need additional bases and testing before they can be adopted. The approach described in Reference [3] appears to have more significant limitations, especially for event sequences which have a reasonable likelihood of occurring during the operating lifetime of a plant or fleet of plants.

In this report we propose an alternative approach which is believed to have a more transparent and justified basis. We propose use of a cumulative distribution for comparison to acceptance criteria, where a complementary cumulative distribution function (CCDF) is used for frequency versus consequences, and the total frequency of Initiating Events (IEs) by category is used as a measure for stable operation.

3.2.11 Selection of Treatment for SSCs

In this element the treatment for safety significant SSCs is determined. In both frameworks the term "special treatment" is used to distinguish requirements on SSCs that exceed requirements for "commercial grade" SSCs. The type of treatment will vary depending on the SSC. At a minimum a Reliability Assurance Program (RAP) would be applied to safety significant SSCs, where availability and reliability goals are established and monitored, and action is taken if performance is deficient. In addition the functional capability of such SSCs must be demonstrated. Additional requirements, such as procurement and safety class designation, may also apply.

This element is not reviewed here because significant activity is underway within the industry to address this element of the framework. Thus, no changes to this element are proposed.

3.2.12 Development of Reliability Assurance Program

In this element a reliability assurance program is defined to support achieving the reliability and availability assumed in the analyses. In the design and operational phases, a RAP is an important element of the Industry Requirements Document for 10 CFR Part 52 plants, they are requirements in DG-1145 as part of Quality Assurance, and they are part of the AP-1000 Design Certification Document. References to a Reliability Assurance Program are also discussed throughout the NRC framework [1]. RAP also is planned to be addressed in the ongoing ANS standard activity, but specific details have not yet been defined.

In the documents that describe 10 CFR Part 52, the emphasis is on reliability for safety significant SSCs as represented through failure rates in the PRA and other risk determinations. The goals of the RAP are that 1) reliability be designed into the plant, 2) reliability be sustained through plant construction and operation, and 3) plant challenges be minimized through SSC reliability. The operational program is the embodiment of the RAP into Maintenance Rule, testing and inspection programs, quality assurance activities, and station Technical Specifications.

In Reference [1], the emphasis is on establishing and monitoring reliability and availability goals for safety significant SSCs. Reference to these reliability and availability goals appear throughout the document, but they occur most often in support of three requirements:

- Maintaining Defense-in-Depth – RAP and reliability goals compensate for the absence of single failure criteria, for various uncertainties, and assure safety margins.
- Implementing the Protective Strategy: "Stable Operations" – RAP is implemented via existing operational programs.
- Maintaining a Living PRA – Reliability from actual plant operating data is used to verify and/or update PRA data and results.

The design process would be expected to include reliability assurance practices to provide confidence that the design, when implemented, would operate in accordance with the design expectations and assumptions.

The characteristics of a RAP have been addressed in detail as part of the design certification process and will be provided in associated regulatory guidance in the near term. As such, this element is not addressed further in this report.

3.2.13 Monitoring and Feedback

In this element, the plant design is reviewed to assure that SSC performance during operation can be monitored so that the results of this monitoring can feed back to operational programs. Example operational programs include in-service testing (IST) and in-service inspections (ISI). Both the NRC and gas cooled reactor NSSS vendor frameworks address this element.

Thus, this element is not addressed further in this report, as this element is a fundamental aspect of the design process.

4
RESULTS

4.1 Traditional and Advanced LWR Safety Level Demonstration

A regulatory expectation for future reactors is for their level of safety to be equal to, or exceed, that of plants currently in operation. A summary of existing practices for addressing safety and the results of these practices is provided below. The structure of this summary was aligned to the frameworks under development, where possible, to support a comparison to them. This summary was developed on the basis of experience, table top exercises (scoping evaluations) and a review of a certified design's Design Control Documentation (DCD).

4.1.1 Approach for Currently Operating Plants

The licensing approach for commercial nuclear power plants has evolved over several decades, based on operating experience, testing and analyses. For example, fuel designs have changed, Station Blackout (SBO) and anticipated transients without scram (ATWS) have been considered, and operating and configuration risk management practices have changed significantly. Fundamentally, however, the following elements have been considered in developing an acceptable design and licensing basis:

1. **Barriers and Basic Design:** A design is developed which has several barriers and means to minimize the potential for release of radioactive material. (This includes the fuel pellet, fuel clad, reactor coolant system (RCS), containment and associated SSCs aimed at protecting these barriers.)
2. **Operating Regimes:** Operating regimes (such as power level, fuel performance, and RCS and Containment integrity) are identified.
3. **Stable Operation:** A design, selection of SSCs, and operating practices are established which are intended to result in a facility which will operate reliably throughout its lifetime and have an acceptable frequency of initiating events which would affect reliable operation and challenges to safety.
4. **Initiating Events:** Challenges to plant systems which could cause an initiating event (including internal and external challenges) are identified. This is generally addressed by initiating event categories (such as a spectrum of loss of coolant accidents (LOCAs), loss of feedwater, loss of offsite power (LOOP) and seismic events).
5. **Frequency of IEs:** The potential frequency of the initiating events (IEs) is established and each IE is assigned to a frequency category (such as AOOs and DBEs)) on a qualitative basis.

6. **Success Criteria:** Acceptance criteria (e.g., fuel performance, barrier integrity, radiation dose, SSC temperature and pressure limits) are established. This is generally based on the frequency and type of initiating event, and not on the frequency of a sequence.
7. **SSCs to Mitigate IEs:** Based on the IEs, General Design Criteria (GDC), standards and other guidance, SSCs are designed to have the capacity, availability and reliability to respond to IEs.
8. **Event Sequence Analyses:** Analyses are conducted which consider potential sequences (e.g., stuck control rod, fuel performance, RCS integrity, steam generator and containment integrity, offsite power availability, single failures, and response of non-nuclear safety (NNS) systems, structures and components (SSCs)).
9. **Acceptability of Results and Margins:** Acceptance criteria are demonstrated to be met using conservative analysis approaches and assumptions so as to assure design and operating margins are maintained.
10. **Capability, Availability and Reliability of SSCs and Human Actions:** Performance capability, and redundancy and diversity, of SSCs are increased beyond the minimum required by the Code of Federal Regulations (CFRs) and GDC (and associated implementation guidance) based on operating experience, designer judgment, and regulator input. Time available for operator action is also considered.
11. **Siting Criteria:** A "DBA" is considered for siting and containment evaluations (e.g., a source term comparable to a core melt with an intact containment) for comparison to 10 CFR Part 100 site criteria.
12. **Operating Procedures and Limits:** Operating procedures (e.g., emergency procedures) and operating limits (e.g., Technical Specifications and other safety and configuration control practices) are established to support operation within the design basis assumptions.
13. **Monitoring and Feedback:** Monitoring and feedback programs are established to provide reasonable assurance of operation within the licensing basis and to use operating experience as input to considerations to design, maintenance and operational practices.

In addition to the design and operating approach discussed above, PRAs, and other risk evaluations such as seismic margins evaluations, for internal events and external events, have been completed by all operating plants in the U.S. These PRAs were intended to identify "vulnerabilities" and to support improvements in design and operation to address these vulnerabilities. These PRAs and other risk management practices are used extensively at each plant and by NRC to support the management and oversight of the residual risk.

4.1.2 Performance Achieved and Expected

Design and PRA analyses, combined with plant operating experience, show the following has been achieved (for currently operating LWR plants), or is expected (for advanced LWR construction), by using the above approach:

1. The consequences, as measured in dose to the public, for an event sequence which remains within the assumptions made in the analysis of AOO IEs and DBE IEs, are quite

limited, due to inherent features of LWR fuel and the number of barriers to fission product release (e.g. fuel pellet, fuel clad, RCS, and containment).

2. With almost 10,000 years of U.S. and international LWR operating experience, there have been no releases from a LWR approaching the current framework licensing limits (e.g., 25 rem total effective dose equivalent (TEDE) whole body at the exclusion area boundary (EAB)). This criterion has been used for unlikely DBE IEs combined with the worst single active failure and conservative analysis assumptions). Actually, no release exceeding ~1 rem TEDE whole body at the EAB is documented, to the best of our knowledge, to have occurred.
3. To obtain a severe release (e.g., > 25 rem TEDE whole body at the EAB), a BDBE involving severe core damage and failure of the reactor coolant system and containment (or bypass of the containment) must occur. No events of this magnitude have occurred in the U.S. or internationally for any LWR plant licensed to U.S. or European standards.
4. The calculated core damage frequency (CDF) for U.S. LWRs is <~1E-4 per year, and such an event, should it occur, would be expected to result in <~ 25 rem TEDE whole body at the EAB if containment integrity is maintained.
5. The calculated large early release frequency (LERF) for U.S. LWRs is <~ 1E-5 per year.

In addition to these results, the following characteristics for AOO IEs and DBE IEs are applicable to both currently operating plants and certified LWR designs:

AOO IEs: The conditional probability that an AOO initiating event will result in consequences greater than "very small/negligible" is low (in the range of <1E-5 to 1E-3 for LWRs and lower for Advanced Light Water Reactors (ALWRs)). Typically, several failures are needed before there is a meaningful escalation in the severity of the consequences of an event.

We conclude, therefore, as a means to address Defense-in-Depth when developing or applying an F-C function, the consequence limits established for AOO sequences should consider this existing practice and operating performance, rather than developing an F-C function which allows consequences to increase in proportion to the frequency of event occurrence. Note that this also can be addressed in the D-in-D element in the approach applied in Reference [1].

DBE IEs: DBE initiating events are less likely than AOO initiating events, but can have more severe (but still low to moderate) consequences even if all mitigation SSCs operate as intended. However, for most DBE IEs, the consequences do not change significantly absent failures beyond those postulated in the design basis.

Thus, we also conclude for DBEs that, as a means to address Defense-in-Depth, when developing an F-C function, the consequence limits established for DBE sequences should consider this existing practice and operating performance rather than developing an F-C function which allows consequences to increase in proportion to frequency of event occurrence. Note that this can also be addressed in the D-in-D element in the approach applied in Reference 1.

4.1.3 Certified Designs

The approach used for the advanced LWR certified designs is modeled on the practices used for operating LWR plants. The approach has benefited from the ability to address both design and operational issues at the design stage rather than after final design and construction are complete. This approach permits the use of PRA in the design process and feedback of insights obtained into the final design.

The certified design approach can be summarized as consisting of the following four elements:

1. The "traditional" approach described above in Section 4.1.1 was used (e.g., identification of a comprehensive spectrum of initiating events (from AOOs to DBEs), consideration of single failures, credit/consideration for NNS SSCs, and establishing acceptance criteria (e.g., fuel thermal limits, process temperature and pressure limits, and radiation dose limits)).
2. Relevant "beyond design basis" issues and generic safety issues (such as severe accident phenomena, Station Blackout (SBO), decay heat removal reliability, and ATWS) were addressed.
3. SSC redundancy and diversity principles, including physical separation, were applied using engineering assessments based on the relative frequency and consequences of postulated initiating events and sequence types supported by input from PRA analyses.
4. PRAs and severe accident analyses were conducted during the design process to demonstrate nuclear safety goals were met, to analyze alternative designs, to identify risk significant SSCs, and to support other areas, such as Regulatory Treatment of Non-safety Systems (RTNSS).

This fourth element is the key difference between the design and licensing practices applied to currently operating plants and that used in the certification process for advanced LWRs. In addition, the second and third elements described above were addressed with the benefit of several decades of operating experience, the consideration of explicit safety goals (and corresponding surrogate goals such as CDF and LERF) and the consideration of generic issues identified during the many years of operation of existing LWR technology.

The safety analyses for the certified LWR designs have obtained results for AOO IEs and DBE IEs which are comparable to those obtained for currently operating plants. For BDBEs, calculated values of both CDF and LERF have been significantly reduced compared to some existing LWRs.

4.1.4 Use to Support Development of Integrated Reference Framework

Each of the frameworks reviewed in this research consider the spectrum of possible event sequences, their frequency of occurrence and the elements discussed in Section 4.1.1 above. In comparison to the licensing approach for existing plants, the proposed frameworks have the potential to improve on the identification of sequences because they use a PRA to support this identification during the design stage.

As will be discussed in subsequent sections of this report, there can be important differences in the manner in which the spectrum of postulated events is addressed. The key differences that define the proposed frameworks are the establishment of event consequences and respective frequency estimates for the F-C function and the use of the F-C function as the basis for integrated risk management, including the risk profile. In the next section the following fundamental safety principle (FSP) will be discussed, as this principle is common to all of the frameworks reviewed.

> **FSP for Frequency-Consequence (F-C) Relationship:** Allowable consequences shall decrease as the frequency of events increases. Risk, as determined using consequences compared to the anticipated event frequency shall be demonstrated to be equal to or reduced compared to the current generation of operating plants. This analysis shall evaluate the complete spectrum of possible hazards, sequences and consequences. In particular, the analysis shall not be limited to sequences beyond the design basis.

To support the assessment of the NRC and gas cooled reactor (GCR) frameworks reviewed against this FSP, two table top exercises were conducted using the insights gained from analyses and operating experience of currently operating LWRs. In the table top exercises a sample F-C function for use on a sequence specific basis (at the system or system train level) and a sample complementary cumulative distribution function (CCDF) for frequency versus consequences were developed. A summary of the development of these samples is provided below. First, however, we reviewed available references applicable in support of this table top exercise; from this review, Reference [15] was identified as providing an assessment of current generation LWR technology which could support the table top analysis. Second we considered alternative approaches that could be used to develop a CCDF and F-C function for existing LWRs.

Reference [15] was published in 1979 and evaluated the integrated risk and CCDF for LWR sequences that do not result in core melt. Other than this reference, a CCDF for the spectrum of possible consequences applicable to current generation LWRs is not believed to have been previously developed, or at least could not be readily identified via a standard literature search. Reference [15] examined the risk for event sequences less severe than core melt, and compared these risks to those reported in WASH-1400 [23]. Such sequences were referred to as Class 3-8 accidents, whereas core melt accidents were referred to as Class 9 accidents. Although the evaluation in Reference [15] was conducted several decades ago, the results provide insight into the characteristics of a CCDF for currently operating plants. Table 4-1 provides a summary of the results. In addition, integrated risk, in terms of released radionuclide activity, was determined to be:

- Normal Operation: ~2 Ci/year
- AOOs and DBEs: ~0.1 Ci/year
- Core Melt Sequences: ~500 Ci/year

Although the consequence level is addressed using activity measured in curies (Ci), the results are consistent with the analysis and operating experience summarized above. That is, public health consequences are very small if significant core damage does not occur.

TABLE 4-1: SUMMARY OF RESULTS FOR CCDF FROM NUREG/CR-0603

Frequency of Exceeding Consequence (I-131 Equivalent in Ci) (per year) per plant	I-131 Equivalent Release (Ci)
1	1E-4
1E-2	<1
1E-3	~1
1E-4 to 5E-5	>1 to 1E+3 (Higher value dominated by core melt sequences)
~5E-5 to ~1E-5	1E+3 to 1E+8 (Dominated by core melt sequences)

CCDF and F-C Table Top Evaluations: Several alternatives exist for developing a baseline CCDF and an F-C Function. These alternatives include:

- Conduct of a detailed risk assessment on currently operating plants for the spectrum of possible releases.
- Collecting and combining existing PRA results and design analysis results (which would need to be adapted to include an estimate of frequency for event sequences, the approach used in Reference [15]).
- Using operating experience.
- A combination of the above alternatives.

In this project we did not attempt to develop fully defendable functions. Instead we used table top exercises to develop a sample F-C function and a sample CCDF to illustrate the key features. The sample F-C and CCDF functions were then used as benchmarks against which the F-C functions and processes included in Reference [1] and the gas cooled reactor frameworks could be assessed.

CCDF: For the CCDF, the sample we developed addresses the following Key Points:

Key Point #1: Based on design analyses and operating experience the consequences of potential AOOs and DBEs are negligible to small if sequence conditions remain within the licensing basis assumptions.

Key Point #2: Based on experience and analyses, the frequency of exceeding a release on the order of 25 rem TEDE whole body at the EAB is in the range of 1E-3 per year to 1E-5 per year (and possibly even lower). Given the characteristics of LWR fuel and plant design, a similar frequency range appears reasonable for a release exceeding a value as low as 1 rem.

Key Point #3: Although the surrogate goals of CDF and LERF can not be directly correlated to radiation dose level at the EAB, they do suggest the following.

a. The frequency of exceeding a dose level on the order of 25 rem TEDE whole body at the EAB should be less than 1E-4 per year; and
b. The frequency of exceeding a dose level exceeding ~100 rem TEDE whole body at the EAB should be less than 1E-5 per year.

These key points were used to establish fundamental principles from which a CCDF which is consistent with current practices, analysis results and operating experience for currently operating LWRs could be specified. These fundamental principles are:

1. Normal operation should meet standard ALARA radiation projection principles.
2. During the lifetime of an individual plant no more than a small release (e.g., 5 to 50 mrem at the EAB) would be expected (corresponds to an annual exceedance frequency of ~1E-2/year per plant).
3. During the lifetime of a plant, the likelihood of having a release exceeding a small fraction of 10 CFR Part 100 release limits (e.g., 10% of 25 rem at the EAB) is low (1% to 10%, and thus corresponds to an annual exceedance frequency in the range of 1E-3 to 1E-4 per year per plant).
4. During the lifetime of a plant the likelihood of having a release exceeding the 10 CFR Part 100 release limits (25 rem whole body at the EAB) is very low (0.1% to 1%, and thus corresponds to an annual exceedance frequency in the range of 1E-4 to 1E-5 per year per plant).
5. During the lifetime of a plant the likelihood of having a release with the potential for significant early health effects (e.g. greater than ~300 rem) is exceedingly low (<0.1%, and thus corresponds to an annual exceedance frequency of less than 1E-5 per year per plant; in this report we assumed a range of 1E-5 to 1E-6 per year per plant for this type of event).

Table 4-2 summarizes these principles using a complementary cumulative distribution (CCDF) approach providing sample frequency and consequence values. Note that Table 4-2 also provides cumulative values for operation of a fleet consisting of 100 similar plants. The fleet values provide additional perspective on the industry impact of using these principles for an individual plant.

The values used above and in Table 4-2 are approximate, but representative and reasonable based on operating experience and analysis results. Development of a fully justified CCDF is beyond the scope of this project. As will be shown in subsequent sections, this sample CCDF is adequate for assessing actual or potential issues associated with the proposed frameworks being developed.

F-C Function: For developing an F-C function, the following guidelines, which are consistent with a typical, currently operating LWR, were used.

1. An AOO initiating event plus a single failure should have negligible consequences. A frequency consequence point of 3E-3/year and 5 mrem was selected for this condition.

2. AOO initiating events and sequences initiated by an AOO IE more frequent than 3E-3 should have consequences less than 5 mrem.
3. A single sequence initiated by an AOO which could result in a radiological dose of 25 rem TEDE at the EAB should have a frequency less than 1E-5 per year. Thus, if 10 sequences were possible for this frequency and consequence level, the total frequency would be 1E-4 per year, which is consistent with a CDF goal of 1E-4 per year.
4. Sequences initiated by an AOO which have frequencies between 1E-5 per year and 3E-3 per year should have consequence criteria between 25 rem and 5 mrem.
5. The guidelines used to address DBE IEs are comparable.

As with the development of the above CCDF, approximate values are used. At the level of review performed for this evaluation, such an approach is sufficient for assessing the proposed frameworks reviewed in this project. Table 4-3 summarizes this F-C function, including the results and the basis. This F-C function would be applied on a system or system train level, consistent with current practices.

TABLE 4-2: SAMPLE FUNDAMENTAL CCDF FREQUENCY-CONSEQUENCE VALUES (1)		
Frequency of Exceeding Consequence (per year) per plant	**Consequence (in rem)**	**Discussion** (Assume 100 plants in the fleet, each with a lifetime of 60 years)
In the range of 1E-5 to 1E-6	>300	1. Consistent with LWR PRA results. 2. Potential for significant early health effects and ~consistent with LERF of 1E-5/y. 3. .006% to .06% likelihood per plant during its lifetime. 4. During the lifetime of a fleet of plants the likelihood of having a release with the potential for significant health effects is .6% to 6%.
In the range of 1E-4 to 1E-5	~25	1. ~ Consistent with intent of 10 CFR Part 100 and currently operating plants. 2. .06% to .6% likelihood per plant during its lifetime. 3. During the lifetime of a fleet of plants the likelihood of having a release exceeding 10 CFR Part 100 release limits is 6% to 60%.
In the range of 1E-3 to 1E-4	~1 to 2.5	1. ~ Consistent with current practices (small fraction of 10 CFR Part 100 limits). 2. .6% to 6% likelihood per plant during its lifetime. 3. During the lifetime of a fleet of plants the likelihood of having a release exceeding a small fraction of 10 CFR Part 100 release limits is 0.6 to ~6 events are to be expected.
1E-2	~ 0.0050 to 0.050	1. ~ Consistent with current practices. 2. During lifetime of plant no more than a small release would be expected (60%).
1E-1	Negligible [2]	Consistent with current practices.
1	Negligible	Consistent with current practices.
>1	Negligible	Consistent with current practices.
Normal Operation (No frequency assigned)	ALARA	Consistent with current practices.
Notes: (1) Many Existing plants and ALWRs meet or improve on these values. (2) Negligible, i.e., meets ALARA principles		

TABLE 4-3: F-C FUNCTION FROM TABLE TOP EXERCISE		
IE/Sequence	**Frequency (per year)**	**Consequence Criteria**
AOO IE w/ or w/o failures which have no impact on consequences	10	.1 mrem (i.e., ALARA)
AOO IE w/ or w/o failures which have no impact on consequences	1	.5 mrem (i.e., ALARA)
AOO IE w/ or w/o failures which have no impact on consequences	1E-1	.5 mrem (i.e., ALARA)
AOO IE w/ or w/o failures which have no impact on consequences	2.5E-2	.5 mrem (i.e., ALARA)
AOO IE + 1F which impacts consequences	3E-3	5 mrem (i.e., ALARA)
AOO IE + 2F which impacts consequences	1E-4	3% Part 100 limits (.75 rem)
AOO IE + 3F which impacts consequences	3E-5	10% Part 100 limits (2.5 rem)
AOO IE + 4F which impacts consequences	1E-5	100% Part 100 limits (25 rem)
AOO IE + Severe	1E-6	1000% Part 100 limits (250 rem)
Limit on Frequency and Consequences	5E-7	1000 rem
DBE IE	1E-3	1% Part 100 limits (.25 rem)
DBE IE + 1F which impacts consequences	3E-5	10% Part 100 limits (2.5 rem)
DBE IE + 2F which impacts consequences	1E-6	100 % Part 100 limits (25 rem)
DBE IE + Severe	3E-7	1000% Part 100 limits (250 rem)
Limit on Frequency and Consequences	3E-7	1000 rem
Legend		
1F = 1 Failure; 2F = 2 Failures; Severe = Failures sufficient to result in release much greater than 10 CFR Part 100 limits. Note: This analysis assumes that a DBE with a frequency >1E-3 per year must meet corresponding AOO criteria based on frequency.		
F-C Description		
Two Initiating Event categories are used (AOOs and DBEs.) A BDBE IE category could be used but would not change the conclusions obtained from this exercise.		
Sample Sequences assigned to an AOO, DBE, and BDBE are provided in Columns 1 for illustrative purposes. Number of failures is only used to distinguish among the sequence types and does not impact the assessment.		
To be consistent with current, deterministic practices, an AOO with a single failure should meet AOO consequence criteria. A "~Negligible" consequence level of 5 mrem has been established as representative for an AOO plus SF. Consequence values for frequency values >3E-3 per year should not be interpreted as proposed values. They are included simply to establish a function. Their actual values should be based on ALARA principles.		
To be consistent with current practices a DBE with a single failure should meet DBE consequence criteria.		
Consequence Criteria were selected based on judgment to: • Incorporate, in an approximate manner, existing practices (when viewed from a Risk Informed perspective), and • Provide a manageable number of consequence points.		
The F-C curve is developed for both AOOs and DBEs simply to illustrate the approach. A single F-C curve could be developed and not change the insights or conclusions.		
Note: Purposely, a gap has been introduced in the IE frequency between the least likely AOO (2.5E-2 per year) and the beginning of the DBE category (1E-3 per year.) This gap recognizes that there is not a well defined definition of AOOs and DBEs. In more recent safety analyses, an intermediate category of infrequent has been addressed, but even in this case there in overlap between the infrequent and DBE categories.		

4.2 Fundamental Safety and Design Principles

In this framework element FSPs and FDPs are developed. These fundamental principles replace, and are for the most part consistent with, comparable criteria provided in Reference [1] and the proposed gas cooled reactor frameworks.

FSPs and FDPs provide the underlying basis on which the proposed licensing framework ensures adequate plant safety. In this structure, FSPs provide the nuclear safety objective which is to be achieved. Associated with each FSP is a corresponding FDP that provides mechanisms by which achievement of the FSP can be demonstrated. In this section, the FSP / FDP combinations which were developed during this review are provided. A review of these FSPs / FDPs demonstrates that they provide a structured and comprehensive set of candidate principles from which the proposed licensing process can ensure adequate nuclear safety levels.

In Section 5, we recommend that stakeholders reach agreement on a common set of such principles. These candidate principles are intended to assist in reaching this agreement, and to support development and implementation of the remaining elements of this reference framework.

Eleven FSPs with their corresponding FDP and relationship to similar elements contained in the other frameworks reviewed are described below.

FSP-1: Quantitative Health Objectives (QHOs)

> **FSP:** Quantitative Health Objectives (QHOs) and the intent of the Safety Goal Policy Statement shall be met with margin. In this framework, the QHOs provide one measure of nuclear safety risk. The objective is that plants licensed under this framework shall have demonstrated equal or reduced levels of risk compared to the current generation of operating plants. In addition, the spectrum of possible hazards, sequences and consequences, shall be addressed, and not be limited to sequences that are beyond those specified in the design basis.
>
> **FDP:** A PRA and other risk evaluations, of sufficient scope and quality that appropriately consider uncertainties shall be performed. This analysis shall evaluate a complete spectrum of hazards over all plant operating modes. The resulting spectrum of possible consequences shall be compared to the QHOs to demonstrate this FSP is achieved.
>
> **Relationship to Frameworks Reviewed:** This FSP/FDP is equivalent to similar criteria in the proposed frameworks reviewed, with one exception. Here, we allow for the use of "other risk evaluations," rather than a full PRA. This change is intended to allow alternatives, such as a PRA-based margins approach for seismic events that are appropriate to assess hazards that are characterized by considerable uncertainty and for which a blended approach rather than a full PRA, might be preferable and adequate. This is consistent with the regulatory guidance which is anticipated to be issued in early 2007 for applying for a combined operating license (COL) using 10 CFR Part 52.

FSP-2: Frequency-Consequence (F-C) Relationship

FSP: Allowable consequences shall decrease as the frequency of events increases. Risk, as determined using consequences compared to the anticipated event frequency shall be demonstrated to be equal to or reduced compared to the current generation of operating plants. This analysis shall evaluate the complete spectrum of possible hazards, sequences and consequences. In particular, the analysis shall not be limited to sequences that are beyond those specified in the design basis.

FDP: Frequency consequence relationship(s) and processes shall be developed. The results of the PRA and other risk evaluations shall be used to demonstrate this FSP is achieved on an aggregate basis, e.g., by using a complementary cumulative distribution function (CCDF) for frequency versus consequences.

Relationship to Frameworks Reviewed: This FSP/FDP also is equivalent to similar criteria in the proposed frameworks reviewed, with the following exceptions. We added "equal to or reduced compared to current generation of operating plants" and "..CCDF.." This language is included to provide assurance that a plant designed and licensed using a new framework has a defendable technical basis for concluding that the safety level is equal to or better than the current generation of operating plants. The frameworks reviewed do not provide a clear basis for supporting this desired conclusion.

FSP-3: Operational Stability

FSP: Stable plant operation shall be achieved; the frequency of events that can upset plant stability and challenge safety functions shall be limited.

FDP: The design shall meet a total initiating event frequency goal that is determined to provide an adequate level of public safety. (At this time, a goal of this type is not yet developed and would constitute a future research task) An initiating event which could result in releases exceeding ALARA principles shall not be expected during a plant's lifetime. Additionally, challenges to safety systems shall be minimized consistent with the potential safety significance of the challenge.

Relationship to Frameworks Reviewed: This FSP/FDP is equivalent to similar criteria in the proposed frameworks reviewed, but has been expanded to include an ALARA criterion.

FSP-4: Barrier Defense in Depth (D-in-D)

FSP: To assure sufficient D-in-D, the number of barriers and the integrity of each specific barrier shall be sufficient to meet the QHO, F-C and operational stability requirements specified in FSPs 1, 2 and 3, respectively. The number, and integrity, of barriers designed to prevent radioactive material release shall increase as the frequency of events which could challenge the barriers increases.

FDP: Barrier failure shall not be expected during the lifetime of a plant. At least one barrier shall be available to mitigate a potentially significant release for events that have a reasonable probability of occurring during the lifetime of a fleet of plants.

Relationship to Frameworks Reviewed: This FSP/FDP is equivalent to similar criteria in the proposed frameworks reviewed.

FSP-5: Accident Prevention and Mitigation Defense-in-Depth (D-in-D)

FSP: Accident prevention and mitigation, consistent with inherent safety characteristics, shall be addressed to assure a sufficient level of defense-in-depth (D-in-D) is achieved. Reliance on accident prevention alone is not sufficient; strategies and systems shall be put in place such that effective mitigation actions can be performed to maintain nuclear safety during abnormal plant conditions to a high degree of confidence. In the design of protective systems, plant response, including human actions, shall be demonstrated to be sufficiently available, reliable, and capable of ensuring adequate safety margins are maintained.

FDP: The plant design shall be demonstrated to have sufficient mitigation capability, consistent with inherent safety characteristics, with the assumption that the SSCs intended to prevent a BDBE have failed to perform their intended functions. The availability, reliability and capability of the SSCs intended to prevent and mitigate accidents shall be demonstrated to be acceptable with respect to meeting FSPs/FDPs 1–4. SSCs designed to provide accident prevention and mitigation capabilities shall receive treatment and monitoring appropriate to their safety significance. In selecting SSCs and developing monitoring and feedback processes, the potential causes of degraded reliability or capability (such as design errors, human errors and common cause failures) shall be addressed.

Relationship to Frameworks Reviewed: This FSP/FDP is equivalent to, and basically a summary of similar requirements provided in, Reference [1].

FSP-6: Key Safety Function (KSF) Defense in Depth (D-in-D)

FSP: Key Safety Functions (KSFs), consistent with inherent safety characteristics, shall not be dependent on a single element of design, construction, maintenance or operation.

FDP: To ensure sufficient D-in-D is achieved, no KSF, consistent with inherent safety characteristics, shall be dependent on a single element (either physical SSC or associated human action). Additionally, hazards, such as fire, flooding, and seismic events, shall not prevent KSFs from achieving their intended objectives of ensuring FSPs 1-5.

Relationship to Frameworks Reviewed: This FSP/FDP is equivalent to, and basically a summary of similar requirements provided in, Reference [1].

FSP- 7: Siting

FSP: Site selection shall consider all hazards, emergency response capability and intentional acts.

FDP: Natural and man made hazards shall be considered and demonstrated to be acceptable with respect to meeting FSPs/FDPs 1-6. Siting decisions shall address emergency response and protection against both inadvertent and intentional acts. In addition, siting decisions should consider the acceptability of routine operations, in addition to off-normal occurrences such as AOOs, DBEs and BDBEs.

Relationship to Frameworks Reviewed: This FSP/FDP is equivalent to, and basically a summary of similar requirements provided in, Reference [1].

FSP-8: Consideration of Uncertainties

FSP: Uncertainties in analyses, in equipment and human performance, and in plant response (e.g., lack of operational experience, type and magnitude of challenges, and physical, chemical and aging phenomena) shall be considered in the assessments that demonstrate the FSPs / FDPs are achieved.

FDP: To ensure sufficient safety margins, appropriate uncertainties shall be identified and explicitly addressed or bounded, so as to provide assurance that FSPs 1-6 are achieved. The margin appropriate to address applicable uncertainties shall be identified and included within the design and operating requirements.

Relationship to Frameworks Reviewed: This FSP/FDP is equivalent to, and basically a summary of similar requirements provided in, Reference [1].

FSP 9: Operating Limits and Practices

FSP: Operating limits and practices shall be established to provide assurance that operating conditions are within the bounds of the plant design and corresponding analysis requirements and assumptions, including allowances for analysis and monitoring uncertainties and response to conditions which could place the plant outside these bounds.

FDP: Operating limits (e.g., Limiting Conditions for Operation (LCOs)) shall be established which are based on plant design and corresponding analysis requirements and assumptions. Limits shall consider uncertainties in analyses and monitoring capability. Operating practices shall be established to implement these limits and to respond to conditions which could result in operating outside of them. These practices include operating procedures, abnormal response procedures, and emergency procedures.

Relationship to Frameworks Reviewed: This FSP/FDP is equivalent to similar requirements provided in the frameworks reviewed, and is consistent with current practice.

FSP-10: Emergency Preparedness

FSP: Emergency plans and procedures shall be developed and demonstrated to be effective in mitigating the potential impacts of events.

FDP: Emergency plans and emergency procedures (such as severe accident management guidelines) shall be developed, which consider the comprehensive use of installed SSCs and human intervention, coordination with regulatory and government agencies, and access to resources (human and SSCs) outside the plant.

Relationship to Frameworks Reviewed: This FSP/FDP is equivalent to, and basically a summary of similar requirements provided in, Reference [1].

FSP-11: Monitoring and Feedback

FSP: Operating experience shall be used to confirm design and analysis assumptions and modify plant design and/or operating practices as appropriate.

FDP: The design shall provide for monitoring of SSCs at a level commensurate to their safety importance. Operating practices, including a reliability assurance program, shall be established so as to support effective monitoring and to provide feedback into decision making.

Relationship to Frameworks Reviewed: This FSP/FDP is equivalent to, and basically a summary of similar requirements provided in, Reference [1].

Comparison to an Alternative View: Figure 4-1, based on IAEA References [9, 10, 11] and extracted from the ongoing ANS standard development activity (and which is also provided in Reference [14]), is based on the scenario levels of a hierarchical Defense-in-Depth approach. This construct provides a hierarchical structure that is generic to any reactor design. Thus, as an alternative to comparing the FSPs/FDPs to the frameworks reviewed, one could consider the comparison of the FSPs to the hierarchy provided by IAEA. In Table 4-4 we provide an initial mapping of the IAEA elements provided in Figure 4-1 to the proposed FSPs.

TABLE 4-4: FSP COMPARISON TO IAEA LEVELS OF D-IN-D	
Figure 4-1 Element	**Corresponding FSP**
Prevent Initiating Events	FSP-3: Operational Stability FSP-9: Operating Limits and Practices FSP-11: Monitoring and Feedback
Control Events and Prevent Accidents	FSP-3: Operational Stability FSP-5: Accident Prevention and Mitigation FSP-9: Operating Limits and Practices FSP-11: Monitoring and Feedback
Control Accidents within the Design Basis	FSP-4: Barrier Defense in Depth FSP-5: Accident Prevention and Mitigation FSP-6: KSF D-in-D FSP-7: Siting FSP-8: Consideration of Uncertainties FSP-9: Operating Limits and Practices
Mitigate the Consequences of Severe Accidents	FSP-4: Barrier Defense in Depth FSP-5: Accident Prevention and Mitigation FSP-6: KSF D-in-D FSP-7: Siting FSP-8: Consideration of Uncertainties FSP-9: Operating Limits and Practices
Provide Emergency Protective Actions	FSP-7: Siting FSP-10: Emergency Preparedness
No Undue Risk to Public Health and Safety	FSP-1: QHOs FSP-2: Frequency-Consequence FSP-11: Monitoring and Feedback In addition FSPs 3 through 10 also apply.

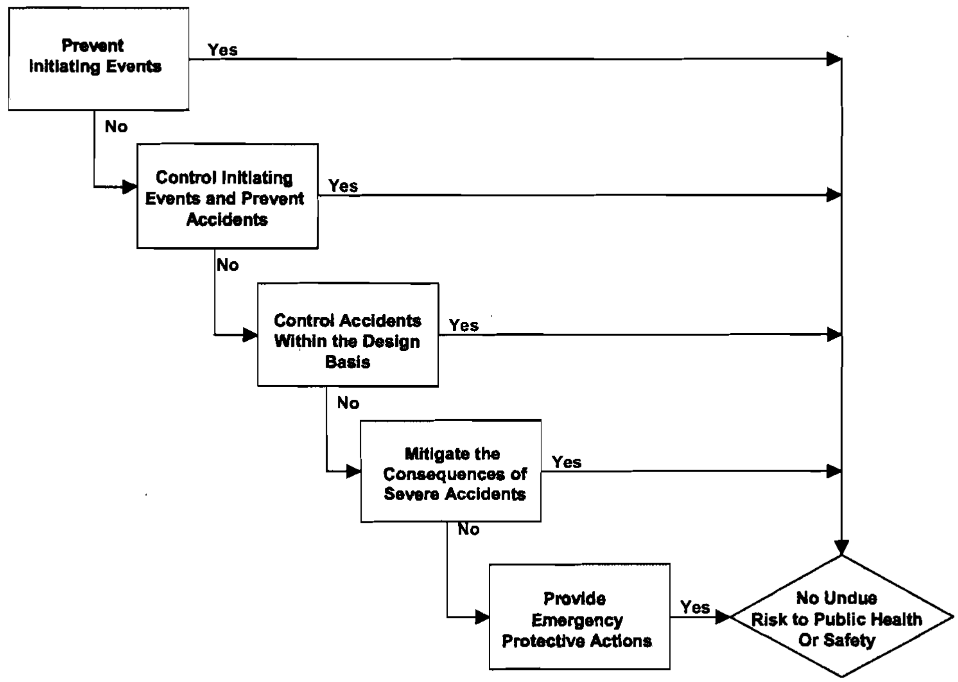

Figure 4-1: Scenario Defense-in-Depth Framework of IAEA [9, 10, 11]

4.3 Frequency Consequence Function and QHOs

These topics address FSPs 1 and 2 and are used in subsequent elements of the integrated reference framework. In this element an F-C function, the process for using the F-C function and the approach for using QHOs are developed.

Quantitative Health Objectives (QHOs) and Frequency-Consequence (F-C) functions (radiological dose vs. frequency of an event sequence or event sequence family) provide key quantitative criteria for determining safety adequacy in the frameworks under development. In these frameworks, QHOs shall be demonstrated to be met; and the spectrum of event sequences or event sequence families, from anticipated to extremely unlikely, shall be demonstrated to be below the F-C function limits.

4.3.1 QHOs

The QHOs have been used to support decision making for existing plants; and surrogates, such as core damage frequency (CDF) and large early release frequency (LERF), have been developed and used for LWRs. QHOs, however, are not intended as the sole basis for demonstrating the safety case; and meeting the QHOs is not sufficient to demonstrate an adequate safety case.

For the purposes of this review, an evaluation of the application of the QHOs to advanced reactor licensing was conducted as follows:

- First, a summary of the QHOs is provided. This summary addresses key features including their intended use, past and current uses, and limitations. The basis and specific quantitative values are not included.
- Second, the use of the QHOs in the frameworks that were reviewed is discussed.
- Third, potential issues are identified.
- Fourth, the approach adopted for this integrated reference framework is provided.

Background

The QHOs (Reference [13]) were established about 20 years ago, and have not fundamentally changed in their bases, values or intent. Key features are:

- The risk measures addressed are individual latent fatality risk, individual acute fatality risk and cost-benefit relationship for risk reduction opportunities.
- The use of the various risk measures is not independent. For example, the latent fatality risk (determined on an average individual basis for a radius of 10 miles) is expected to be bounded by the acute fatality risk (determined on an average individual basis for a radius of ~ 1 mile.)
- The QHOs are intended to address the risks associated with all operating modes and plant challenges (routine operations, AOOs, DBEs, and BDBEs.) For LWRs, they have mostly been used in assessments related to BDBEs involving severe core damage (e.g., cost-benefit evaluations, backfit determinations, and to support assessments of the safety of existing plants and ALWR plant designs which have been certified using 10 CFR Part 52.).
- The focus on BDBEs involving severe core damage (for the current generation of LWRs) is based on two factors: (1) that less severe events pose less risk (due to the limited dose levels possible for routine occurrences, AOOs and DBEs for LWRs); and (2) that less severe events are either within or would not be significantly outside the licensing basis, and thus would have been the subject of extensive evaluations and safety deliberations.
- Surrogate goals for LWRs of Core Damage Frequency (CDF) and Large Early Release Frequency (LERF) have been established with values of 1E-4 per year and 1E-5 per year, respectively. CDF is a surrogate for the latent fatality risk QHO; LERF is the corresponding surrogate for the acute fatality risk QHO.
- QHOs are not intended to be, and have not been, used as a sole basis for determining safety adequacy. Use of QHOs as the sole criteria for demonstrating acceptable risk and an acceptable risk profile is inappropriate, and not consistent with Safety Goal Policy statements or past and current practices. Meeting the QHOs does not provide assurance that a design meets other key design principles and has no, or limited, regulatory history for AOOs and DBEs.
- Key design principles and past practices, such as Defense-in-Depth and the expectation that the total frequency of varying levels of release with the potential for latent consequences or

impact on public activities is sufficiently low, are not addressed by comparing total calculated risk to the QHOs.

- The QHOs have been used to supplement decision processes. Traditional design basis events, combined with expectations on design basis response capability, have been the primary means to demonstrate an adequate safety level.

Use of QHOs in the NRC Framework described in Reference [1]

In this framework, QHOs are included as a FSP (referred to in reference [1] as a safety expectation) and as explicit acceptance criteria. In recognition of the intent and limitations of the QHOs, Reference [1] uses several acceptance criteria in addition to the QHOs, all of which must be met. These acceptance criteria include meeting F-C limits at high confidence levels (95%) with only safety significant SSCs responding, meeting integrated risk limits by event sequence category, and meeting Defense-in-Depth principles. Each of these requirements is addressed in subsequent sections of this report.

The role of the QHOs in Reference [1] is straightforward and appears to be a reasonable extension of current accepted licensing practices. In the proposed NRC framework, the QHOs must be demonstrated to be met using mean values by integrating all event sequences.

Use in Proposed Gas Cooled Reactor Approaches

References [3] and [20] are the most recent, available work. Like Reference [1], these frameworks do not rely solely on the QHOs as the acceptance criteria for demonstrating the safety case. However, in these frameworks, the proposed use of the QHOs is quite different. The development of the F-C function uses the QHOs as a consideration in establishing frequency and consequences values. Also, in addition to demonstrating that AOOs, DBEs and BDBEs meet the F-C function limits, they are integrated and compared to the QHOs. Finally, the AOOs and DBEs also must be demonstrated to meet deterministic radiological dose limits with only safety significant SSCs responding to the event.

In these frameworks, the dependence on QHOs is stronger compared to Reference [1]. An evaluation of the use of QHOs in the gas reactor framework to that proposed in the NRC framework indicates the following:

- Stronger for the gas reactor proposed framework in that the development of the F-C function considers the QHOs.
- Stronger for the gas reactor proposed framework in that the BDBEs, when combined with DBEs and AOOs, in total must only be demonstrated to meet the QHOs. This framework does not fully address integrated risk measures, such as development of a risk profile.

Assessment

The use of QHOs in Reference [1] appears to be reasonable and appropriate. The key issue for this framework is the absence of the development of a risk profile, possibly in the form of a CCDF. This has been discussed previously and will be discussed in more detail in the next section of this report. Instead of a CCDF other measures (such as risk as measured in annual dose), which may be sufficient, are used in this reference.

The use of QHOs in References [3] and [20] may be overly ambitious. This approach has the following potential issues, which should be addressed:

- A risk profile, or other means of controlling the allowable radiological dose versus total frequency, is not included. This is the case for all event categories (AOOs, DBEs, and BDBEs.) As discussed above, there is no obvious technical basis for concluding that solely meeting the QHOs and deterministic criteria for AOOs and DBEs will result in an acceptable risk profile.
- The approach to including AOOs and DBEs in the comparison to the QHOs is unclear. In Reference [3] they appear to be excluded. (Note that one proposed advanced gas reactor concept (Reference [20]) does include AOOs and DBEs in determining acute and latent fatality risk for comparison to the QHOs.)

A summary of these issues and recommendations is provided in Section 5.

Approach Adopted for the Proposed Integrated Reference Framework

In the framework being proposed as a result of this research, a FSP is that the QHOs shall be met with an adequate degree of margin. In recognition of the intent and limitations of the QHOs, several acceptance criteria, in addition to QHOs, all of which must be met, are included. These criteria include meeting F-C limits at high confidence with only safety significant SSCs responding, meeting integrated risk limits via demonstration of an acceptable risk profile (e.g. CCDF), and meeting Defense-in-Depth principles.

All hazards, operating modes and potential sequences shall be included in this determination. For certain hazards and sequences, a full quantification using PRA methods is not required. Other risk evaluations or bounding assessments can be used.

In this report we have not defined the term "with margin." This topic is being addressed by several industry stakeholders and constitutes an area for further research.

4.3.2 F-C Functions

The F-C functions under development for the frameworks proposed by the gas cooled reactor vendors and NRC are intended to provide a limit on allowable consequences versus the frequency of a sequence or group of sequences. The spectrum of possible sequences from those events that are anticipated to occur to BDBEs is addressed. The F-C functions under development are based on the principle that the frequency of an event sequence should decrease as the consequences of the event sequence increases.

Currently operating plants meet this principle, in a general sense on a qualitative basis, for event sequences within the design basis. For example, an event sequence involving a loss of feedwater initiating event combined with the limiting single active failure is expected to result in insignificant consequences to public health and safety; whereas a small break LOCA (SBLOCA) initiating event combined with a limiting single active failure is expected to result in negligible to small, but not significant, consequences.

Fully quantitative F-C functions are not used in the licensing basis for existing plants, nor are they in the 10 CFR Part 52 licensing basis applicable to ALWRs. Since the development of an F-C function is a relatively new concept and its implementation is not straightforward, the approach is not seen as necessary to support the licensing of this reactor technology.

However, for advanced reactors that are not based on LWR technologies, a different approach to achieving the goals for evaluating and licensing these technologies (in particular, development of a framework that is risk-informed and technology neutral) is seen as an improvement. The concept of use of an F-C function was developed specifically for this purpose. Due to the relatively recent development of this concept, different stakeholders have attempted to define and apply the F-C function concept. Thus, the approach to developing and using an F-C function, and the results of the PRA, currently varies significantly between the framework proposed by the regulatory authority (Reference [1]) and the work conducted by the gas cooled reactor NSSS vendors (where, for the purposes of this review the work documented in Reference [3] was chosen as the representative illustration).

For this investigation, the evaluation of the proposed F-C functions was conducted as follows:

- First, a review of F-C functions in general is provided. This review addresses the strengths and limitations of these functions.
- Second, the development of F-C functions in the frameworks reviewed is discussed. This includes the basis for the functions and limiting values selected.
- Third, potential issues with these functions and processes were identified. This was supported by conduct of table top exercises.
- Fourth, issues common to both approaches and actions to address these issues are provided.

A new, reference F-C function and process was not developed during this project. This development was not within the scope of the project.

4.3.2.1 F-C Functions: Basic Uses, Strengths and Limitations

Basic uses of an F-C function and its associated application include:

1. Support the identification of sequences for consideration in plant design.
2. Comparison of sequence frequencies and consequences to the F-C function limits to demonstrate acceptable plant safety levels.
3. Support SSC classification for purposes of implementing appropriate risk management strategies.

Reference [5] also comments that TLRC, which includes the F-C function, are defined as the "standards for judging the licensability of the plant with respect to the retention of radionuclides.

As such, the TLRC answer the question of what must be achieved." As will be discussed below, the results of this review indicate that this statement could not be fully justified.

The benefits of the F-C function concept are potentially significant. The approach establishes quantitative guidance in the requirements necessary for "safe" operation that can be used during the design stage. The function and process can support designers in determining availability and reliability requirements for SSCs intended to mitigate initiating events by explicitly identifying and considering the frequency of those events and evaluating the impact of the response of plant SSCs on the calculated consequence levels and corresponding sequence frequencies. This approach to identification of initiating events and sequences, without the constraint of certain deterministic practices (such as single failure criterion), may result in a more complete identification and effective management of potential risk contributors. Also, an approach which explicitly considers frequency and consequences in a quantitative manner may improve the classification of SSCs and lead to risk-management strategies that provide assurance that the safety objectives are achieved while being cost-effective to implement.

However, the limitations of the F-C function also can be significant. These limitations do not appear to have not been fully addressed in the references reviewed. If an F-C function is used as a decision criterion in the licensing process, these limitations will need to be addressed. During this review, the following issues and limitations were identified.

- Developing a basis for the frequency for a specific consequence level: The two approaches noted above vary by several orders of magnitude in the level of frequencies that are proposed as "acceptable" at several consequence levels. It is unclear how an F-C function can be developed independent of the process for using the function and without reliance on an acceptable risk profile. It is also unclear if proposed methods of applying the F-C functions would support the principle that future plants should be at least as safe as currently operating plants.

- Defining an event sequence: For a specific initiating event there are several means to display a sequence which results in a comparable consequence level. A sequence displayed at the basic event level (e.g. component unavailable due to preventive maintenance) would have a lower frequency than if aggregated at the system or function level, where all causes of unavailability or failure would be aggregated. Thus the development of an F-C function must consider the process for using the function, and in particular, to ensure an adequate level of plant safety is achieved, some evaluation of aggregate results must be performed.

- Defining an initiating event: This activity has comparable challenges to that associated with the definition of event sequences. In particular, the question of to what level of detail should any IE be defined needs to be addressed. As an explicit example, the process should provide a method of specifying how many fire initiating events should be used (e.g. one representing the total plant fire frequency, frequency by room, etc.). As another example, the process should provide guidance on specifying how many transient IEs should be used (e.g. define at system level, train level, etc.)

- Establishing a Risk Profile: Since the F-C functions under development treat individual or families of sequences and not the aggregate risk profile resulting from them, a fundamental question is how is a determination of the acceptability of the risk profile performed, what criteria are to be used in this determination and, in particular, are the QHOs and proposed deterministic measures sufficient?

- Establishing Requirements for Each Sequence: Requiring every sequence to meet the F-C function limits appears to be inappropriate. In the proposed frameworks, if any one sequence is above the F-C limit (regardless of its postulated frequency), the design is considered unacceptable, whereas, if the same event is just below the limiting value of the F-C function, the design is acceptable. This application of a "hard acceptance threshold" provides limited flexibility to both the regulatory authority and reactor designers. Within the proposed framework, there does not appear to be provision for evaluating the risk / reward tradeoffs.

As an example of the challenges identified see Table 4-5, which provides a comparison of F-C functions from References [1] and [3], and the table top CCDF and F-C functions provided in Tables 4 and 5. This tabulation demonstrates significant differences in the F-C functions between the frameworks specified in References [1] and [3]. In Table 4-5, the radiological dose results for the frameworks proposed in references [1] and [3] are displayed next to each other to facilitate direct comparison, with the differences noted in ***bold italic font.***

The F-C results obtained from the table top evaluations conducted are similar to those provided in Reference [1]. These table top results for CCDF are higher than Reference [1], which is to be expected, as the CCDF is applied on a cumulative rather than on a sequence-specific basis. The table top results are significantly different than those provided in Reference [3] with the CCDF developed in the table top exercises being more restrictive than the Reference [3] F-C values which are applied on a "family" basis.

Next the F-C functions proposed in Reference [1] and [3] are reviewed. This review provides insight into the reasons for the differences described above.

TABLE 4-5: FREQUENCY-CONSEQUENCE FUNCTION EXAMPLE							
Reference [3] (PBMR)		Reference [1] (NRC)		Table Top			
				F-C		CCDF	
Frequency (per year)	Dose Range	Dose Range	Frequency (per year)	Frequency (per year)	Dose	Frequency (per year)	Dose
1E+1	10 mrem	NA	NA	10	0.1 mrem	>1	Negligible
1E+0	***100 mrem***	***1 mrem – 5mrem***	1E+0	1E+0	0.5 mrem	1E+0	Negligible
1E-1	***100 mrem***	***5 mrem***	1E-1	1E-1	0.5 mrem	1E-1	Negligible
1E-2	***100 mrem - 2.5 rem***	***5 mrem – 100 mrem***	1E-2	1E-2	5 mrem	1E-2	5 mrem-50 mrem
1E-3	***~ 10 rem***	***100 mrem – 1 rem***	1E-3	1E-3	250 mrem	1E-3	1 rem- 2.5 rem
1E-4	***25 rem – 300 rem***	***1 rem – 25 rem***	1E-4	1E-4	~ 1 rem	1E-4	2.5 rem- 25 rem
1E-5	***~400 rem***	***25 rem – 100 rem***	1E-5	1E-5	~10 rem	1E-5	25 rem- 300 rem
1E-6	~500 rem	100 rem – 300 rem	1E-6	1E-6	250 rem	1E-6	>300 rem
5E-7	~700 rem – 10000 rem	300 rem – 500 rem	5E-7	5E-7	1000 rem	Not included	Not included
1E-7	NA	>500 rem	1E-7	1E-7	>1000 rem	Not included	Not included

4.3.2.2 NRC Framework F-C Function Development

The principle underlying the F-C curve provided in Reference [1] is that event frequency and radiological dose to the public are inversely related, i.e., the higher the postulated dose the lower is the acceptable frequency of the event. As discussed in Reference [1], "This principle, and the whole F-C curve, is broadly consistent with the approach of ICRP 64 which provides recommendations on the annual frequencies and doses to individual members of the public from accidental exposures (Reference [16]).

Reference [1] developed an F-C function that was derived from current regulatory requirements in 10 CFR Parts 20, 50 and 100. Table 4-6 provides a summary of the basis for this F-C function.

TABLE 4-6: PROPOSED DOSE/FREQUENCY RANGES FOR PUBLIC EXPOSURES (Based on Table 6-1 from Reference [1])

Dose Range	Frequency (per year)	Comment (all doses are TEDE)
1 mrem – 5mrem	1E+0	5mrem/year is ALARA dose in 10 CFR 50 Appendix I.
5 mrem – 100 mrem	1E-2	100 mrem/year is the public dose limit from licensed operation provided in 10 CFR 20.
100 mrem – 1 rem	1E-3	1 rem/event offsite triggers Environmental Protection Agency (EPA) Protective Action Guidelines (PAGs) (Reference [17]).
1 rem – 25 rem	1E-4	25 rem/event triggers Abnormal Occurrence (AO) (Reference [18]) reporting and is limit in 10 CFR 50.34a and in 10 CFR 100 for siting.
25 rem – 100 rem	1E-5	50 rem is a trigger for deterministic effects (i.e, some early health effects are possible). (Reference [19])
100 rem – 300 rem	1E-6	In this range the threshold for early fatality is exceeded.
300 rem – 500 rem	5E-7	Above 300-400 rem, early fatalities are quite likely.
>500 rem	1E-7	Above 500 rem early fatalities are very likely and curve is capped.

Reference [1] provides an insightful and fairly comprehensive discussion of the development of an F-C function. The consequence levels selected appear reasonable, are based on existing regulatory practice and are consistent with results obtained in the table top exercise. The function and the frequency values assigned to most of the consequence values are based on "informed judgment." We use the term "informed judgment" because, although there is no technical or historical basis provided for frequency values and the F-C function, the results are consistent with those obtained from our table top exercise. As has been discussed, the table top exercise did consider operating experience, design basis practices and results, and PRA results in an approximate manner.

4.3.2.3 Gas Reactor Framework F-C Function Development

The proposed framework described in Reference [3] uses an approach for developing the F-C function that is different from that presented in the NRC framework. In this report we have extracted key language from References [3] and [5] to provide a summary of the basis for how the F-C function was constructed.

- The AOO region is characterized as "those conditions of plant operation which are expected to occur one or more times during the life of the plant" with a lower bound frequency of 1E-2 per year specified.
- The DBE region is characterized as encompassing "releases that are not expected to occur during the lifetime of a single nuclear power plant but may be encountered during the lifetimes of a population of nuclear power plants" and an event frequency range of 1E-2 per year to 1E-4 per year is specified.
- The BDBE region is characterized as considering "improbable events that are not expected to occur during the lifetime of a large fleet of nuclear power plants, but that should be considered to assure that the risk to the public from low probability events is acceptable." "The frequency cutoff implicit in the acute fatality risk goal in NUREG-0880 (Reference [24]) is taken as the lower frequency boundary of the BDBE Region."
- "The points to be plotted against the TLRC represent the frequencies and consequences of accident families, which are groups of event sequences having a common initiating event category, common functional response of the SSCs that support key safety functions, and a common end state." "In addition to this frequency-consequence evaluation of the event sequence families, a cumulative risk evaluation of all events is performed in the conventional form of a complementary cumulative distribution function to demonstrate that the safety goal QHOs are met." These event sequences are to consider all operating modes and potential challenges, including external and internal hazards, such as seismic and fire events.
- The bases for the consequence levels selected are similar to those in Reference [1] and results obtained from the table top exercises conducted during this project.

4.3.2.4 Assessment on Adequacy of the F-C Functions

In this section, consequence and frequency values are assessed for the two frameworks and a table top exercise is documented. The specific characteristics (e.g. the specific functions linking the reference points - shapes) of the respective F-C functions are not addressed.

Consequence Values: The consequence levels selected in References [1] and [3] are generally consistent with criteria and dose values used in analyses for the current generation of operating and proposed advanced LWRs. (Note: The specific definitions and assumptions may vary, e.g., CAB or EAB compared to site boundary, duration of exposure, and release characteristics. These differences are not believed to impact this assessment and can be addressed in future assessments if appropriate.)

NRC Framework Frequency Values: The frequency values from Reference [1] appear reasonable and are consistent with the table top results. Additional table top exercises are provided below to further examine the reasonableness of the Reference [1] and [3] F-C functions.

Gas Reactor Frequency Values: For Reference [3], the frequencies are consistent with current practices, with one critical difference. The frequencies provided in Reference [3] are assigned to sequences, whereas for currently operating plants the frequency, for the associated consequence level, only includes the initiating event frequency (although on a qualitative basis). In addition, consider that when demonstrating that an AOO IE or a DBE IE meets the corresponding acceptance criterion on dose, current practice is to assume failures in response and mitigation systems (generally the worst active single failure) and to make conservative assumptions in potentially important areas such as offsite power availability, reactor scram, fuel performance, and integrity of other barriers (such as steam generator tube and containment leakage rates.) In current practice, these sequences must meet dose criteria associated with a specific initiating event category. For example, a SB LOCA plus limiting single active failure must meet DBE dose criteria. In the Reference [3] process, this sequence would be assigned to the BDBE category, which has higher dose criteria. From this difference, one can conclude that the proposed licensing framework presented in Reference [3] could result in regulatory levels of plant safety, for certain frequency ranges, that are less restrictive than standards that are applied to plants that currently are in operation.

If consequence criteria are to be consistent with levels of safety considered acceptable in current practice for AOOs and DBEs (which appears to be the intent of the framework described in Reference [3]), to achieve this consistency, frequency should also consider past practices. As a minimum, the following, sample changes should be considered:

1. An AOO IE with the equivalent of a single active failure should meet AOO consequence criteria, and be placed in the AOO category.
2. A DBE IE with the equivalent of a single active failure should meet DBE consequence criteria, and be placed in the DBE category.

The specific approach to address this issue needs to be established. Rather than assuming single failures, an approach that explicitly considers the reliability of SSCs included to mitigate an IE could serve as an alternative to the single failure criterion. Absent these considerations, the consequences for AOOs and DBEs allowed by the Reference [3] framework would exceed that intended for (and achieved by) the current generation of plants and advanced, certified designs. The analyses that support this conclusion will be demonstrated quantitatively in exercises 1 and 2 below. Consider, for example, the following benign event sequence at an operating PWR: a loss of feedwater with failure of 2 (of 3) auxiliary feedwater pumps. The frequency of such a sequence is in the range of 1E-3 to 1E-4 per year; Application of the approach provided in Reference [3] would allow a consequence level in the range of 10 rem to 300 rem, whereas the consequence level for this sequence for an operating PWR is negligible.

For the BDBE region, Reference [3] allows greater than 300 rem at the EAB for event sequences with mean frequencies less than 1E-4 per year. Contrast this with the expected total (not just the frequency of a "family") mean frequency for consequences exceeding 300 rem at the EAB for current LWR and certified ALWR designs. These frequencies are on the order of <1E-5 per year and <1E-6 per year, respectively.

In summary, the F-C function and licensing framework proposed in Reference [3] appears to be less restrictive on a "family" basis than has been determined on a cumulative basis for existing and advanced LWRs.

Table Top Exercises: Simple, approximate table top exercises were conducted using the proposed Reference [3] F-C function and the F-C function developed as part of this project. As the F-C function developed as part of this project is consistent with the F-C function in Reference [1], the results using the table top F-C function also are representative of the F-C function in Reference [1].

- Exercise 1 assumed that one sequence was possible for each of the points which define the F-C function. Therefore one sequence was assumed to exactly match the F-C curve for each point. This experiment was intended to represent an optimistic outcome.
- Exercise 2 assumed ten sequences were possible for each of the points which define the function. Therefore ten sequences were assumed to exactly match the F-C curve for each point. This was intended to be more representative of the "family of sequences" approach described in Reference [3].

Table 4-7 provides the results of these evaluations. Values of interest are provided in ***bold italic font***. As is evident from the data, the results using the Reference [3] F-C function do not meet the CCDFs of existing LWR plants. The results using the table top F-C function do meet the CCDF for 1 sequence per point but exceed the CCDF at a few frequency values when 10 sequences are used.

As a result of this characteristic, it can be concluded that the F-C function proposed in Reference [3] needs further development. To ensure the objective that advanced nuclear reactor concepts achieve at least the same level of safety as current generation plants, F-C functions need to consider a limit on the number of sequences or use a CCDF as verification.

4.3.2.5 Common Issues and Actions

As noted previously, the use of F-C functions could be used to improve upon the practices employed for currently operating plants, and perhaps on the practices used for the ALWRs. The limitations of F-C functions, however, are significant, and appear to have not been fully addressed in either proposed framework or this project. If an F-C function is used as a criterion for licensing decisions, these limitations must be addressed. Specific issues and limitations identified in this review include the following:

- Developing a basis for the acceptable frequency associated with a specific consequence level.

The two approaches noted above vary by several orders of magnitude at several consequence levels. In the proposed frameworks, it is unclear how an F-C function can be developed independent of the process for using the function and without reliance on an acceptable risk profile.

To address this issue, it is proposed that the methods and results obtained from current generation LWRs be considered in the development of the F-C function. Since the F-C function provided in Reference [1] appears reasonable, this presents the most likely fruitful starting point for this development.

- Defining an event sequence.

  For a specific initiating event there are several means to display a sequence which results in a comparable consequence level. A sequence displayed at the basic event level would have a lower frequency than if aggregated at the system, function or "family" level.

  To address this issue, the development of an F-C function must consider the process for using the function. Event sequences should be defined consistent with their use in the F-C process and on the basis of the F-C function.

- Defining an initiating event.

  The definition of initiating events has comparable challenges. Similar actions to those proposed for defining an event sequence should be taken to reach consensus on defining an IE.

- Evaluation of aggregate risk.

  Since the F-C functions under development treat sequences and not the aggregate risk, it is not clear how the acceptability of the risk profile will be determined in these frameworks.

  To address this issue, we note that solely relying on the QHOs as a risk metric does not provide a sufficient basis for plant licensing under the existing framework. The framework proposed in Reference [1] recognizes this feature of the QHOs and applies other integrated risk acceptance criteria; we propose use of a CCDF as an approach to achieve this objective.

- Requiring every sequence to meet the F-C function limits appears to be inappropriate.

  In the proposed frameworks, if any one sequence is above the F-C limit (regardless of its postulated frequency), the design is considered unacceptable, whereas, if the same event is just below the limiting value of the F-C function, the design is acceptable. This application of a "hard acceptance threshold" provides limited flexibility to both the regulatory authority and reactor designers.

  To address this issue, it is recommended that a CCDF be applied rather than treating individual sequences, or families, as needing to always meet the F-C limits. In this manner the benefits of a F-C function and the sequences which populate the function are preserved, but an unnecessary limitation is removed.

TABLE 4-7: COMPARISON OF EXERCISES 1 AND 2 TO CCDF					
CCDF from Table 4-5		**Reference [3] F-C**		**Table Top F-C**	
		1 Sequence per Point	**10 Sequences per Point**	**1 Sequence per Point**	**10 Sequences per Point**
Dose (TEDE at EAB)	**Frequency of incurring Consequence Level or higher does not exceed this Value (per year)**				
.001/ Negligible	>1	>10	>100	~1	~10
.05	.01	***~1***	***>10***	>1E-3 to <1E-2	>1E-2 to <1E-1
2.5	1E-3 to 1E-4	***>1E-2***	***>1E-1***	>1E-5 to <1E-4	>1E-4 to <1E-3
25	1E-4 to 1E-5	***>2E-4***	***>2E-3***	>1E-6 to <1E-5	>1E-5 to <1E-4
300 or higher	1E-5 to 1E-6	***>1E-4***	***>1E-3***	~1E-6	<1E-5
700	1E-6	~1E-6	***~1E-5***	NA	NA
1000	1E-6	>5E-7	***~5E-6***	NA	NA

Notes:

1. Values are approximate.
2. 750 rem and 1000 rem consequence levels were included for comparison to Reference [3] F-C curve values. Lower end of 1E-5 to 1E-6 range was used for 750 rem and 1000 rem consequence levels.
3. Entries for the .001 rem consequence level are not important. They are included to anchor the curve.
4. Frequency values for the .001 rem consequence level used for these exercises would not be expected to be realized. They are simply a result of the F-C function's frequency value at a consequence level of .001 rem.)

4.4 Identification and Classification of Safety Functions

In this element, safety functions, success criteria and operating limits are established. No changes from the frameworks reviewed are proposed to this element.

At the highest level, safety functions (SFs) can be viewed as generic; for example reactivity control, reactor pressure control, reactor coolant inventory control, heat removal, and barrier integrity. To be useful on a design-specific basis, the generic characterization is modified to address design-specific features. For example, the proposed High Temperature Gas Reactor (HTGR) concepts rely on a passive design to ensure radionuclide retention within the fuel and passive heat removal to maintain fuel integrity. Therefore, certain LWR-specific safety functions may not be directly applicable for use in these (and other) designs. For example, some of the references reviewed state that the nature of HTGRs eliminates the need for some of the LWR-specific safety functions (such as the need for inventory control) but creates new ones, (such as the need to control chemical attack of the graphite reactor core and fuel).

Independent of the applicability of past practices for establishing safety functions, complete and useable design-specific functions, and associated success criteria, must be established to develop a licensable reactor design and to support the evaluations needed to demonstrate the FSPs and FDPs are met. Provided below are design-specific safety functions and success criteria that serve as proposed TLDCs for HTGR designs. These FSPs/FDPs are based on information generated as part of the ongoing ANS standards activity. For each SF, there is a corresponding TLDC.

SF I: Control Radionuclides in Fuel Particles

This function refers to designing, fabricating, and operating fuel such that normal operational releases are limited to the extent that only the radionuclide inventory within the fuel itself presents a potential challenge to meeting consequence limits.

TLDC I: The coated fuel particles shall provide the primary barrier to the release of radionuclides and demonstrate performance capability that, in concert with other plant barriers, satisfies the approved acceptance criteria.

SF II: Control Chemical Attack

This function refers to preventing fuel or graphite degradation which could be caused by the intrusion of compounds (e.g., air or water) other than helium into the primary coolant.

TLDC II: Capability shall be provided to prevent chemical attack that would lead to fuel or core structural degradation resulting in exceeding the acceptance criteria. The plant components that provide this capability shall be demonstrated to meet approved design acceptance criteria.

SF III: Control Heat Generation

This function refers to controlling the heat generated in the reactor core so that fuel temperatures do not exceed acceptable limits. Since TLDC II limits exothermic chemical reactions, the principle requirement of this function is to assure reliable reactor shutdown.

TLDC III: The reactor shall be designed, fabricated, and operated such that inherent nuclear feedback characteristics ensure that the reactor thermal power will not exceed acceptable limits. Additionally, the reactivity control system shall be designed, fabricated, and operated such that during insertion of reactivity, the reactor power will not exceed acceptable limits.

SF IV: Remove Core Heat

This function refers to removing reactor heat so that fuel temperatures are not excessive. Since the design selections needed to meet TLDC II and TLDC III limit chemical attack and fission heat generation, the principle requirement of this SF is to assure reliable decay heat removal.

TLDC IV: 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 fuel conditions would not exceed acceptance criteria or violation of other TLDC.

SF V: Maintain Core Geometry

This function refers to maintaining a core geometry that is capable of ensuring adequate core cooling. This function is complementary to controlling heat generation and removing core heat by maintaining, for example, a fixed geometry for the insertion of movable poisons and assurance of adequate heat transfer. The core geometry is maintained by limiting the excessive displacement of, and maintaining the structural integrity of, core elements.

TLDC-V: The intrinsic dimensions of the reactor core, reactor internals, and reactor vessel shall be maintained such that these structures sustain the completion of all protective and mitigation actions assumed in the safety analysis.

SF VI: Maintain Reactor Building Geometry

This function refers to maintaining cavity and building geometry. This function is complementary to controlling heat generation and removing core heat by providing structural support for the reactor vessel, passive cooling pathway SSCs, and other major reactor components; for maintenance of core geometry and passive heat removal; and providing structural protection of the reactor vessel, helium coolant pressure boundary, and safety-related SSCs from loads generated by internal and external hazards.

TLDC-VI: The reactor building shall be designed, constructed, and operated such that the structural integrity of the building is maintained and sustains the completion of all protective and mitigation actions assumed in the safety analysis.

4.5 Performance of PRA and Other Risk Evaluations and Use in Frequency-Consequence Function

In this element a PRA and other risk evaluations are conducted, and results are provided in a form that supports comparison to the F-C function and other quantitative criteria, such as the QHOs. This comparison is conducted in the next element.

This element of the integrated reference framework developed in this project is consistent with both the proposed NRC and the gas cooled reactor frameworks, with the addition of proposed features to address the issues discussed below. A review of PRA quality and methods necessary to support this application was not included in this project.

PRA Use

PRA is used extensively in the frameworks under development. As currently proposed, these frameworks would require a full scope, level 3 PRA (for varying dose levels from benign to severe – if possible) for all hazards. As PRA is used to establish LBEs, use of the proposed frameworks most likely would require the PRAs to be documented and reviewed in consonance with 10 CFR Part 50 Appendix B quality assurance (QA) requirements and be subjected to an independent peer review. Key applications during design include generating a complete set of accident sequences, developing a rigorous accounting of uncertainties, evaluating conformance with the QHOs, evaluating conformance with the F-C function, identifying and characterizing LBEs, and identifying and characterizing safety significant SSCs.

Other potential applications noted in Reference [1] include supporting Environmental Impact Statement (EIS) and severe accident mitigation design alternatives (SAMDA) analyses, risk informing inspections during fabrication and construction, supporting the determination of staffing requirements, supporting the development of the Technical Specifications (or equivalent), supporting the development of inspection, testing and preventative maintenance programs, supporting the development of procedures and training, supporting the development of emergency preparedness policies, assessing and managing operational risk, assessing and managing plant changes, and monitoring SSC performance. Several of these other applications would be expected to be optional and thus they are not addressed in this report.

Issues identified in this project and addressed by candidate changes included in the integrated reference framework are discussed below.

- **"Other Risk Evaluations":** In the framework proposed in this project, we added the phrase "and other risk evaluations." This change would permit application of other techniques, either in addition to or instead of PRA evaluations to address the FSPs/FDPs when appropriate. This change would provide more flexibility while not compromising achievement of the plant licensing objectives.

  The frameworks under development use a full scope PRA as noted above. "Other risk evaluations" refers to risk evaluations which are not fully quantitative, such as the PRA-based seismic margins approach used for the certified, ALWR designs. In this project we acknowledge that certain hazards and operating modes might be better, or sufficiently, addressed using approaches and methods other than PRA.

- **PRA for AOOs:** In the proposed frameworks, use of PRA analysis for all sequences, including anticipated operational occurrences (AOOs) is required. The use of a more deterministic approach for AOOs may be merited and is acknowledged in the references reviewed.

  AOOs are, by definition, events that are anticipated to occur. All hazards at a plant site are to be addressed in the frameworks under development. However, some hazards may have such minor potential consequences that the quantitative frequency of an AOO or DBE is not important. That is, the hazard can be treated deterministically.

4.6 Comparison to Relevant Quantitative Criteria

In this element the results of the PRA are compared to quantitative criteria.

This element of the integrated reference framework developed in this project is consistent with that proposed in Reference [1] and the gas cooled reactor frameworks, with the addition of proposed features to specifically address FSPs and "other risk evaluations." Point estimates would be used in the conduct of this element. Uncertainty, including margin, is addressed in the next element.

This comparison is straightforward, and would address the quantitative criteria developed from the FSPs and FDPs. These criteria include the following:

- Quantitative Health Objectives (QHOs) shall be met with margin.
- Risk, as determined using the QHOs shall be demonstrated to be no greater than or reduced compared to currently operating plants.
- F-C function limits shall be met.

- Stable operation criteria shall be met.

If the criteria are not achieved, design and/or operational characteristics would be changed until the criteria are met.

4.7 Uncertainty Analyses and Comparison to Relevant Criteria

In this element uncertainties are addressed so that the corresponding results can be compared to the relevant quantitative criteria used in the above element. This element of the integrated reference framework developed in this project is consistent with Reference [1] and the gas cooled reactor frameworks, except an alternative approach to allow selected conservative analysis is included.

In both Reference [1] and Reference [3], the mean results of the PRA are used for comparison to the QHOs and F-C limits. LBEs (see Section 4.8) are addressed differently. In this element the use of mean results appears to be appropriate. However, conservative analyses should be allowed as an alternative approach when their application would not change the conclusions obtained. This alternative could support more efficiency in both the analysis and regulatory review.

4.8 Selection of Safety Significant Systems, Structures and Components

In this element safety significant SSCs are identified. This element of the framework developed in this project is consistent with the approach proposed in the NRC framework, with the addition of proposed features to address initiating events.

In Reference [1], safety significant SSCs are those SSCs which are needed in order to meet the F-C function limits and other deterministic requirements. In Reference [20], a similar, but not identical approach is used; and, as this approach is based on a specific design, additional detail, including specific classification criteria, is provided. For both frameworks, the objective is to identify those SSCs which should be subject to special treatment due to their safety significance.

The evaluation of this element was conducted as follows:

- First, the approaches in References [1] and [20] are summarized.
- Second, an assessment of these references is provided.
- Third, the changes deemed appropriate to develop this integrated reference approach are discussed.

4.8.1 NRC Framework Approach

In Reference [1], safety significant SSCs are characterized as summarized below. This summary was developed by extracting text from this reference.

- The term 'safety significant' is assigned to those SSCs whose functionality plays a role in meeting the acceptance criteria imposed on the LBEs. Thus, SSCs required to meet the LBE acceptance criteria (both frequency and consequences) are classified as safety significant. Other SSCs are characterized as non-safety significant.
- The term 'special treatment' is used to designate requirements imposed on SSCs that go beyond industry established requirements for equipment classified as 'commercial grade,' These requirements provide additional confidence that the equipment is capable of meeting its functional requirements. The type of special treatment varies depending on the function and importance of the SSC. The application of special treatment helps to ensure that the SSCs will perform reliably (as postulated in the PRA) under the conditions assumed to prevail in the event sequences for which the SSCs successful function is credited in the risk analysis.
- A basic special treatment requirement for all safety-significant SSCs will be the establishment and monitoring of reliability and availability goals. All safety-significant SSCs will have reliability and availability consistent with the values assumed in the PRA. During operation, a process similar to the required monitoring of the performance and condition of structures, systems, or components, against licensee-established goals of 10 CFR 50.65, the Maintenance Rule, is expected to be an integral part in the monitoring program for this special treatment requirement.

4.8.2 Gas Reactor Framework Approach

In contrast to the NRC framework, in the proposed gas reactor framework discussed in Reference [20], SSCs are divided into multiple classes. This classification is used to assist in defining special treatment to assure that SSCs are sufficiently capable and reliable to prevent and/or mitigate LBEs. According to Reference [20], "Special treatment is applied to safety-classified SSCs to provide assurance that the reliability and capability of the SSCs relied on to perform required safety-functions during LBEs meet the TLRC. The special treatment to be applied is graded commensurate with the risk-importance of the LBEs."

Four steps are employed in the process, as summarized in Table 4-8.

4.8.3 Assessment of Proposed Frameworks

The approach used in Reference [20] is similar to that in Reference [1]. However, the approach discussed in Reference 1 would assign a safety significant classification to all SSCs required to meet the F-C function at a 95% confidence level on both frequency and consequence. The basis for the approach in Reference [20] not addressing the complete F-C function is uncertain. Regardless, this identification using either approach is reasonably straightforward for standby SSCs.

The approach to identifying safety significant SSCs which could cause an initiating event is not fully developed, and is not specifically addressed, at least not clearly, in either framework. Instead, a living PRA is used in both frameworks and target frequencies for initiating event categories are considered in Reference [1].

4.8.4 Approach for Integrated Reference Framework

In this integrated reference framework, we have adopted the Reference 1 approach, which as discussed above is quite similar to Reference [20]. In addition, the need to consider initiating events explicitly is added. The approach to addressing initiating events was not within the scope of this project. However, we recommend that approaches applied to currently operating reactors be reviewed and considered.

TABLE 4-8: REFERENCE [20] APPROACH TO SSC CLASSIFICATION (Extracted from Reference [20])

Intent	Description
Mitigation of DBE Consequences	SSCs relied on to perform the safety functions required for DBEs to meet 10 CFR 50.34 dose limits to the public are classified as *safety-related.* Assures that SSCs are available for mitigation of the consequences of DBEs.
Prevention of High Consequence BDBEs	SSCs relied on to perform safety functions required to prevent the frequency of BDBEs with consequences greater than the 10 CFR 50.34 dose limits from increasing into the DBE region are classified as *safety-related.* Assures that SSCs are available for prevention of events with unacceptable consequences for DBEs.
Mitigation of AOO Consequences	SSCs relied on to perform the safety functions required for AOOs to meet 10 CFR Part 20 dose limits to the public are classified as *non-safety-related with special treatment.* Assures that SSCs are available for mitigation of the consequences of AOOs.
Prevention of High Consequence DBEs	SSCs relied on to perform safety functions required to prevent the frequency of DBEs with consequences greater than the 10 CFR Part 20 offsite dose limits from increasing into the AOO region are classified as *non-safety-related with special treatment.* Assures that SSCs are available for prevention of events with unacceptable consequences for AOOs. It also assures limited challenges to safety-related SSCs.

4.9 Licensing Basis Events and Design Basis Accidents

In the integrated framework proposed in this report, LBEs are identified and analyzed conservatively for comparison to F-C function limits. This is comparable to the approach developed by NRC, but differs significantly from that proposed by the gas reactor NSSS vendors. The primary modifications we propose to the approach presented in Reference [1] are the basis for grouping and the use of a 95% confidence level for determining the frequency of the representative sequences.

In Reference [1] LBEs consider only safety significant SSCs to be available to mitigate an initiating event. Representative sequences are selected based on a grouping of similar sequences from the PRA. Each representative sequence must meet the F-C function limits at the 95% confidence level, where the highest frequency and consequence level of the sequences within the group are used to define a representative sequence. In addition, each representative sequence must meet deterministic criteria, which will be discussed in the next element. In Reference [1], LBEs are characterized as being similar to DBAs in the current licensing framework (such as are documented in Section 15 (and other sections) of a Final Safety Analysis Report (FSAR)).

The proposed gas cooled reactor approach is different. As discussed in Reference [3], all sequences from the PRA, grouped into families, are referred to as LBEs. These LBEs are then subdivided on the basis of the frequency of a family of sequences into AOO (frequency >1E-2/yr.), DBE (frequency between 1E-2/yr. and 1E-4/yr.) and BDBE (frequency <1E-4/yr) categories. Deterministic DBAs are selected from the DBEs, using a grouping process, by assuming that only SSCs classified as safety-related are available to perform the safety functions required to meet 10 CFR 50.34. The DBEs then are reanalyzed deterministically with only the safety-related SSCs responding in a mechanistically conservative manner to demonstrate that the mean consequence of each DBA is less than 25 rem TEDE limit at the EAB. AOOs are treated in a similar manner but use a lower consequence limit of 100 mrem TEDE. The BDBEs are integrated for comparison to the QHOs.

The differences between these two proposed frameworks are significant. The evaluation of this element was conducted using a process similar to that employed for the evaluation of safety significant SSCs (i.e. Section 4.8).

4.9.1 NRC Framework Approach

In the proposed NRC framework, the LBEs selected from the PRA event sequence results provide a means to ensure that potentially risk significant challenges meet design criteria with adequate Defense-in-Depth (including safety margin). These LBEs must also meet some deterministic criteria in addition to meeting the F-C curve. A deterministically selected LBE is also used to evaluate the design from the standpoint of the dose guidelines in the site criteria of

10 CFR Part 100. (Note: D-in-D and other deterministic criteria, including site criteria, are addressed in Section 4.10 of this report.)

The following provides a summary of the steps used in Reference [1] to select LBEs using the PRA results:

- The point estimate frequency for each event sequence is determined by only crediting those SSCs that are to be considered safety significant.
- All event sequences with 95th percentile frequencies > 1E-7 per year are determined for sequences with point estimate frequencies equal to or greater than 1E-8 per year.
- Event sequences with a 95th percentile frequency > 1E-7 per year, having similar initiating events and similar accident behavior in terms of system failures and/or phenomena and similar source terms, are grouped into event classes. The goal of the grouping is to strike a reasonable balance between the number of event classes and the degree of conservatism used in the grouping process. As a result of the grouping process, all PRA sequences are covered by an LBE.
- For each event class, an event sequence from the event class that represents the bounding consequence is selected. The LBE frequency for a given event class is determined by setting the LBE's mean frequency to the highest mean frequency of the event sequences that constitute the event class and its 95th percentile frequency to the highest 95th percentile of the event sequences in the event class.
- For each event class, the frequency and consequences must meet the F-C curve plus the Defense-in-Depth requirements that are a function of the LBE frequency range. If these criteria are not met, either the event class is refined or modifications are made to the design.

Reference [1] provides the following discussion for selection of LBEs: "For the LBE selection the question remains at what 'level' are the selected sequences defined: cutset, systematic, or functional? In the framework approach the LBEs are sequences selected from the PRA at the 'systematic' level in terms of front-line systems that provide the needed safety functions. The specific level of detail for these 'front-line' systems are different technologies (that) will be determined in the technology specific Regulatory Guides."

Reference [1] has one additional criterion applicable to initiating events as follows: "To ensure that the assumptions in the PRA on initiating events (IEs) are preserved, each applicant proposes cumulative limits on IE frequency for each of the LBE event frequency categories. These limits will be monitored during the plant operation as part of the living PRA program."

4.9.2 Gas Reactor Framework Approach

The proposed LBE process for the gas reactor designs is significantly different from the approach proposed in Reference [1]. Table 4-9 tabulates the approach, as discussed below.

AOOs: All sequences with frequencies within the AOO frequency range must meet the corresponding F-C limits. A sequence is then referred to as a LBE within the AOO category if the consequences of the AOO could exceed dose criteria if not for prevention/mitigation accomplished by plant design features. Mean values for frequency and dose are used. In addition, DBE sequences with an upper bound frequency exceeding the lower threshold frequency for AOOs (1E-2/yr.) are evaluated using AOO dose criteria. A weighted summation (on the basis of frequency and consequences) of all AOOs for comparison to the 10 CFR Part 20 annual limit of 100 mrem appears to be the acceptance criterion for integrated dose.

DBEs: All sequences with frequencies within the DBE frequency range must meet the corresponding F-C limits. A sequence is referred to as a LBE within the DBE category if the consequences of the DBE could exceed dose criteria if not for prevention/mitigation accomplished by plant design features. Mean values for frequency and dose are used.

BDBEs: As these are considered to be beyond the design basis they must meet the F-C limits and when integrated together with all AOO and DBE sequences, meet the QHOs using mean value estimates.

DBAs: Deterministic DBAs from the DBEs are then identified by assuming only SSCs classified as safety-related are available for prevention/mitigation. Expected consequences are compared to 10 CFR Part 50.34 limits (25 rem TEDE). Then the DBEs are reanalyzed using conservative assumptions. Reference [3] characterizes the deterministic DBAs as the analog to traditional LWR DBAs in Chapter 15 of a FSAR.

TABLE 4-9: LBE IDENTIFICATION PROCESS FOR PBMR (REFERENCE [3])			
Region	**Frequency Range for Region (per year)**	**Acceptance Criteria**	**Comments**
AOO	1E+1 to 1E-2	Frequency at 10/yr: 10 mrem TEDE at controlled area boundary (CAB) Frequency from 1E+0 to 1E-2/yr: 100 mrem TEDE at CAB	Mean values are used All SSCs and Operator Actions are considered AOO becomes a LBE if design features are needed to meet dose criteria.
DBE	1E-2 to 1E-4	Frequency at 1E-2/yr: 2.5 rem TEDE at EAB Frequency at 1E-4/yr: 25 rem TEDE at EAB	Mean values are used. All SSCs and Operator Actions are considered. DBE becomes a LBE if design features are needed to meet dose criteria. If DBE upper bound (UB) frequency is above 1E-2, AOO acceptance criteria are used.
BDBE	1E-4 to 5E-7	QHOs	Mean values are used. All SSCs and Operator Actions are considered.
DBAs	1E-2 to 1E-4	25 rem TEDE at EAB	Only safety-related SSCs are assumed to respond. (Similar to Chapter 15 in current Safety Analysis Report (SAR).) Conservative consequence analysis approach is used.

4.9.3 Assessment

The assessment of each reference is addressed separately. A comparison is then provided, and conclusions are reached. The previously identified issues associated with limitations of an F-C function and the definition of a sequence (discussed in Section 4.4) are not repeated here. Reference [1] clearly notes that event sequence definition will need to be addressed in technology-specific guidance; Reference [3] proposes the use of families to address this issue.

NRC Framework (Reference [1])

The approach described in this reference is straightforward and generally reasonable. The key uncertainty in using the proposed approach is expected to be how the event sequence groups will be determined. Presumably the intent of this grouping is related to focusing and managing the level of effort in conducting the uncertainty analyses.

Use of a 95% confidence level for consequence analysis appears reasonable for frequent (i.e. event sequences with frequency >1E-2/yr) and infrequent (event sequences with frequency between 1E-2/yr. and 1E-5/yr.) event categories. This approach is consistent with current practices. The use of a 95% confidence level for consequence analysis of rare events (i.e. event sequences with frequency <1E-5/yr.) appears to be overly restrictive. A mean value should be considered for this event category.

Use of the 95% confidence level for frequency also appears reasonable for the frequent and infrequent categories, but overly restrictive for the rare category.

Gas Reactor framework (Reference [3])

The approach proposed in Reference [3] differs from the NRC framework and is dependent on the SSC classification approach discussed in Reference [20] (refer to Table 4-8). The major differences between the gas reactor framework and that proposed by NRC are the following:

- Mean values are used in Reference [3], except for the DBA category.
- Consequences of BDBEs are not addressed in Reference [3].
- Explicit consideration of meeting both frequency and consequence limits for the spectrum of the F-C function is not addressed in Reference [3]. Instead, preventing a movement to a lower consequence LBE category is considered.
- The approach to analyzing DBAs in Reference [3] is unclear. The discussion appears to not consider failures of safety-related SSCs (i.e., assumes such SSCs respond as designed with 100% reliability) and appears to use 25 rem TEDE at the EAB regardless of the frequency of the DBA.

If the above characterization is correct, this approach appears to be not only significantly less restrictive than the approach in Reference [1], but considerably less restrictive than current practices.

4.9.4 Proposed Revised Framework

In the integrated reference framework we propose the approach in Reference [1] be adopted with the additional requirement to apply mean values for the rare event categories.

4.10 Defense-in-Depth, Safety Margins and Other Acceptance Criteria

In this element of the framework an evaluation of Defense-in-Depth, including safety margins, and other acceptance criteria, is conducted. Reference [1], with some modifications is used here, as the gas cooled reactor approach was not available to support a review during the schedule of this project.

The evaluation was conducted as follows:

- First, a summary of the overall D-in-D approach discussed in Reference [1] is provided.
- Second, some preliminary changes to the Reference [1] approach are proposed. The intent of these changes is to improve the communication of D-in-D, so that this important practice is implemented consistently.
- Third, the specific approach used in the design guidance in Reference [1] is summarized.
- Fourth, an assessment is provided.
- Fifth, proposed changes are discussed.

4.10.1 Summary of Defense-in-Depth

Summary of Proposed NRC Framework

Recall from Section 3, that the structure of the framework provided in Reference 1 is based on the following:

- A set of safety/security/preparedness expectations, which are ensured
- By Defense-in-Depth expectations, which are fulfilled
- By a set of protective strategies and certain design criteria and guidance, which are used
- Specification of a process for the development of licensing requirements.

In the proposed NRC framework, the following Defense-in-Depth principles were established:

1. Consideration of intentional (e.g. security-related) as well as inadvertent (e.g., random failure of SSCs and human error) events,
2. Providing both accident prevention and mitigation capability,
3. Ensuring key safety functions (KSFs) are not dependent upon a single element of design, construction, maintenance or operation,
4. Consideration of uncertainties in equipment and human performance,
5. Providing for alternative capability to prevent unacceptable releases, and
6. Siting considerations.

The report notes that D-in-D principles would be applied regardless of the level of safety determined using a PRA. This approach provides a (qualitative) means to address uncertainties and corresponds to good engineering practice developed over several decades of LWR operation. Since an underlying principle of the NRC framework is that accident prevention alone cannot be relied upon to reach an acceptable level of safety, capabilities to mitigate accidents (item 5 above) are also required in this framework. This resulted in the identification of the following protective strategies:

1. Provide physical protection from hazards (e.g., radiological and chemical) for workers and public,
2. Ensure stable operation by limiting the frequency of events that can upset plant stability and challenge safety functions,
3. Provide adequate protective mitigation systems (by providing sufficiently available, reliable and capable SSCs, including human actions, on the basis of the frequency of challenge and the significance of the challenge),
4. Ensure barrier integrity (by providing adequate barriers for workers and public),
5. Develop effective protective actions (by providing severe accident management capability and emergency planning)

The two principle deterministic D-in-D elements of the framework are implementation of the protective strategies and the D-in-D principles. The probabilistic Defense-in-Depth element of the framework is the use of PRA techniques and other logical processes to search for and identify unexpected scenarios, to address uncertainty, and to further assure adequate D-in-D, including adequate safety margin.

Preliminary Proposed Changes to NRC Approach

The description of D-in-D in Reference [1] is understandable, but appears to refer to all aspects of design and operation as D-in-D. Here we propose modest changes which retain all of the relevant features but reorganize the description of D-in-D.

First, we tabulated protective strategies, D-in-D elements and discussions provided in Regulatory Guide 1.174 (Reference [22]) that address D-in-D. Table 4-10 provides this tabulation, as discussed below:

- Column 1 provides the Protective Strategies in Reference [1].
- Column 2 Provides the "Most Directly Related D-in-D Element" from Reference [1].
- Column 3 provides all D-in-D elements from Reference [1] and characterizes their applicability to the protective strategy (with Y = Yes; N = No). Clarifying comments are provided as needed.
- Column 4 provides the D-in-D characterization in RG 1.174.

Note: A numbering scheme is applied, and PS = Protective Strategy and DID = D-in-D

Next, by grouping protective strategies and D-in-D elements, an alternative view to Design Principles and D-in-D Principles was established. Table 4-11 provides this alternative view. In this view the Protective Strategies and D-in-D elements have been grouped into nine Design and Defense-in-Depth Principles.

We will use this grouping in the assessment discussed in Section 4.10.3

TABLE 4-10: COMPARISON OF PROTECTIVE STRATEGIES, D-IN-D ELEMENTS AND RG 1.174

NRC Protective Strategy	Most Directly Related D-in-D element	NRC D-in-D Element per Protective Strategy (Y = Yes; N = No)	RG 1.174 D-in-D Approach
PS1: Physical protection (workers and public)	DID1: Consideration of Intentional as well as Inadvertent events	Y DID1: Consideration of Intentional as well as Inadvertent events Y DID2: Accident prevention and mitigation capability Y DID3: Ensuring key safety functions (KSFs) are not dependent upon a single element of design, construction, maintenance or operation Y DID4: Consideration of uncertainties in equipment and human performance Y DID5: Alternative capability to prevent unacceptable releases Y DID6: Siting considerations	Addressed within the Licensing Basis and other D-in-D elements.
PS 2: Stable operation (limit frequency of events that can upset plant stability and challenge safety functions)	DID2: Accident prevention and mitigation capability	Y DID1: Consideration of Intentional as well as Inadvertent events Y DID2: Accident prevention and mitigation capability Y DID3: Ensuring key safety functions (KSFs) are not dependent upon a single element of design, construction, maintenance or operation Y DID4: Consideration of uncertainties in equipment and human performance N DID5: Alternative capability to prevent unacceptable releases Y DID6: Siting considerations Principle 1, consideration of intentional and inadvertent events, certainly will play a role in the design. Principle 2, prevention and mitigation, can lead to limits on initiating events. Principle 3, no dependence on a single element of design, construction, or operation, applies, as does Principle 4, accounting for uncertainties. Principle 5, preventing uncontrolled release, does not apply, but Principle 6, siting can play a role for stable operation in terms of seismicity, grid reliability, etc.	Defense-in-depth is preserved (for example, system redundancy, diversity, and independence are maintained commensurate with the expected frequency and consequence of challenges to the system; defenses against potential common cause failures are maintained and the introduction of new common cause failure mechanisms is assessed; and defenses against human errors are maintained). Sufficient safety margins are maintained (for example, NRC-approved codes and standards are met or deviations justified; and safety analysis acceptance criteria in the LB are met, or proposed revisions provide sufficient margin to account for analysis and data uncertainty).

TABLE 4-10: COMPARISON OF PROTECTIVE STRATEGIES, D-IN-D ELEMENTS AND RG 1.174			
NRC Protective Strategy	**Most Directly Related D-in-D element**	**NRC D-in-D Element per Protective Strategy (Y = Yes; N = No)**	**RG 1.174 D-in-D Approach**
PS 3: Protective systems (Ensure systems that mitigate initiating events have sufficient reliability and capability, including consideration of Human Actions)	DID2: Accident prevention and mitigation capability DID3: Ensuring key safety functions (KSFs) are not dependent upon a single element of design, construction, maintenance or operation DID4: Consideration of uncertainties in equipment and human performance	Y DID1: Consideration of Intentional as well as Inadvertent events Y DID2: Accident prevention and mitigation capability Y DID3: Ensuring key safety functions (KSFs) are not dependent upon a single element of design, construction, maintenance or operation Y DID4: Consideration of uncertainties in equipment and human performance Y DID5: Alternative capability to prevent unacceptable releases Y DID6: Siting considerations	Defense-in-depth is preserved (for example, system redundancy, diversity, and independence are maintained commensurate with the expected frequency and consequence of challenges to the system; defenses against potential common cause failures are maintained and the introduction of new common cause failure mechanisms is assessed; and defenses against human errors are maintained). Sufficient safety margins are maintained (for example, NRC-approved codes and standards are met or deviations justified; and safety analysis acceptance criteria in the LB are met, or proposed revisions provide sufficient margin to account for analysis and data uncertainty).
PS 4: Barrier integrity (adequate barriers for workers and public-physical and chemical)	Not explicitly addressed	Y DID1: Consideration of Intentional as well as Inadvertent events Y DID2: Accident prevention and mitigation capability Y DID3: Ensuring key safety functions (KSFs) are not dependent upon a single element of design, construction, maintenance or operation Y DID4: Consideration of uncertainties in equipment and human performance Y DID5 Alternative capability to prevent unacceptable releases Y DID6: Siting considerations	Defense-in-depth is preserved (for example, system redundancy, diversity, and independence are maintained commensurate with the expected frequency and consequence of challenges to the system; defenses against potential common cause failures are maintained and the introduction of new common cause failure mechanisms is assessed; and defenses against human errors are maintained).

TABLE 4-10: COMPARISON OF PROTECTIVE STRATEGIES, D-IN-D ELEMENTS AND RG 1.174			
NRC Protective Strategy	**Most Directly Related D-in-D element**	**NRC D-in-D Element per Protective Strategy (Y = Yes; N = No)**	**RG 1.174 D-in-D Approach**
			Sufficient safety margins are maintained (for example, NRC-approved codes and standards are met or deviations justified; and safety analysis acceptance criteria in the LB are met, or proposed revisions provide sufficient margin to account for analysis and data uncertainty).
PS 5: Protective Actions (includes emergency procedures, accident management and emergency preparedness	Not explicitly addressed DID5 Alternative capability to prevent unacceptable releases	Y DID1: Consideration of Intentional as well as Inadvertent events Y DID2: Accident prevention and mitigation capability Y DID3: Ensuring key safety functions (KSFs) are not dependent upon a single element of design, construction, maintenance or operation Y DID4: Consideration of uncertainties in equipment and human performance Y DID5: Alternative capability to prevent unacceptable releases Y DID6: Siting considerations	Defense-in-depth is preserved (for example, system redundancy, diversity, and independence are maintained commensurate with the expected frequency and consequence of challenges to the system; defenses against potential common cause failures are maintained and the introduction of new common cause failure mechanisms is assessed; and defenses against human errors are maintained).
Performance Monitoring is a fundamental aspect of all protective strategies.	Included as a fundamental aspect of design and operation.	Included as a fundamental aspect of design and operation.	Included as a fundamental aspect of design and operation.

TABLE 4-11: ALTERNATIVE DESIGN PRINCIPLES AND D-IN-D APPROACH			
Alternative DID Elements & Design Principles	**Protective Strategies Addressed**	**NRC DID Elements Addressed**	**RG 1.1.74 DID Element Addressed**
PS1, Physical protection (workers and public) is a Design Principle.	PS1: Physical protection (workers and public)	DID1: Consideration of Intentional as well as Inadvertent events	Implicit in meeting GDCs
PS4, Barrier integrity (adequate barriers for workers and public - physical and chemical), is both a Design Principle and DID.	PS4: Barrier integrity (adequate barriers for workers and public - physical and chemical)	DID2: Accident prevention and mitigation capability	Implicit in meeting GDCs
PS2, Stable operation (limit frequency of events that can upset plant stability and challenge safety functions), is a Design Principle.	PS2: Stable operation (limit frequency of events that can upset plant stability and challenge safety functions)	DID2: Accident prevention and mitigation capability	Considered
DID2/PS5, Accident prevention and mitigation capability (Includes Protective Actions - emergency procedures, accident management and emergency preparedness), is a Design Principle.	PS3: Protective systems (Ensure systems that mitigate initiating events have sufficient reliability and capability, including consideration of Human Actions) PS 5: Protective Actions (includes emergency procedures, accident management and emergency preparedness	DID2: Accident prevention and mitigation capability	Considered
DID3, Ensuring key safety functions (KSFs) are not dependent upon a single element of design, construction, maintenance or operation, is DID.	PS3: Protective systems (Ensure systems that mitigate initiating events have sufficient reliability and capability, including consideration of Human Actions)	DID3: Ensuring key safety functions (KSFs) are not dependent upon a single element of design, construction, maintenance or operation	Considered
DID4, Consideration of uncertainties in equipment and human performance, is a Design Principle.	PS3: Protective systems (Ensure systems that mitigate initiating events have sufficient reliability and capability, including consideration of Human Actions)	DID4: Consideration of uncertainties in equipment and human performance	Considered
DID5, Alternative capability to prevent	PS5: Protective Actions (includes emergency	DID5: Alternative capability to prevent	Implicit in meeting GDCs

TABLE 4-11: ALTERNATIVE DESIGN PRINCIPLES AND D-IN-D APPROACH			
Alternative DID Elements & Design Principles	**Protective Strategies Addressed**	**NRC DID Elements Addressed**	**RG 1.1.74 DID Element Addressed**
unacceptable releases, is DID.	procedures, accident management and emergency preparedness	unacceptable releases	
DID6, Siting considerations, is a Design Principle.	PS5: Protective Actions (includes emergency procedures, accident management and emergency preparedness	DID6: Siting considerations	Implicit in meeting GDCs or considered explicitly
Monitoring and Feedback is a Design Principle.	Included as a fundamental aspect of design and operation.	Included as a fundamental aspect of design and operation.	Included as a fundamental aspect of design and operation.

4.10.2 Discussion of NRC Framework

In Reference [1] additional D-in-D criteria are applied to the LBEs selected in Section 4.9 and a deterministically selected LBE is established. Each area is summarized below.

Additional D-in-D Criteria

The additional criteria applied to the LBEs selected by the process described in Section 4.9 depend on the frequency of the particular event sequences. In Reference [1] the region under the F-C curve is divided into frequency categories for the purposes of specifying frequency related deterministic criteria. Table 4-12 lists the categories and their basis (Note: in Tables 4-12 and 4-13, the time basis is defined to be ry = reactor year of operation).

The additional deterministic criteria imposed on the LBEs selected in Section 4.9 are provided in Table 4-13. In addition to these criteria, provided in column 3, Table 4-13 provides additional dose criteria in the third column, which will be discussed in Section 4.11. These additional dose criteria are intended to address total annual dose limits. The middle column provides a summary of the statistic for meeting the F-C curve which was discussed in Sections 4.7 and 4.9, and further below, where safety margin is discussed.

In the NRC framework, the use of high confidence levels for both frequency and consequences is intended to assure adequate safety margin. Since the framework is performance-based, specific margins are not prescribed. However, for certain key variables (such as those which could affect success criteria determination), Reference [1] states that margin would be specified on a design-specific basis; in particular, "In all cases, the success criteria are to be fully defensible and biased such that issues of manufacturer or construction variability, code limitations and other uncertainties are unlikely to result in a failure path being considered a success path."

Deterministically Selected LBE

In Reference [1], a deterministic Defense-in-Depth provision is proposed whereby each design would need to have the capacity to establish a controlled low leakage barrier in the event plant conditions result in the release of radioactive material from the fuel and reactor coolant system that is in excess of anticipated conditions. The specific features would be design-specific, and would be based on a postulated event which would represent a serious challenge to fission product retention in the fuel and coolant system. Under the proposed framework, this event would be agreed upon between the applicant and the NRC based on the technology and safety characteristics of the design.

The deterministic LBE event would to be analyzed mechanistically such that the worst two-hour dose at the EAB and the dose at the outer edge of the low population zone (LPZ) for the duration of the event would not exceed 25 rem TEDE. This is consistent with 10 CFR Part 100 Reactor Site Criteria limits.

4.10.3 Assessment

The assessment of this element was conducted as follows:

- First the frequency categories in Table 4-12 were reviewed.
- Second, the sample acceptance criteria in Table 4-13 were reviewed

The deterministically selected LBE was not reviewed as this topic is the subject of an ongoing review being conducted by the industry and is subject to policy making decisions.

Frequency Categories

Appendix E of Reference [1] discusses the bases for the proposed frequency categories and compares this categorization to existing practices. Subject to the limitations of F-C functions and the process for using them discussed previously, the frequency categories are judged to be reasonable. They are also reasonably consistent with existing practices, when viewed from a sequence frequency versus initiating event frequency (current practice) perspective.

Additional Acceptance Criteria

As noted, we view these criteria as sample acceptance criteria. Additional effort is needed to formally develop a comprehensive set of acceptance criteria and fully document the technical basis for them. Subject to this perspective, our review concluded the sample criteria are reasonably consistent with current practices. In Table 4-11 we propose that criteria applicable to D-in-D include:

- barrier integrity,

- ensuring key safety functions (KSFs) are not dependent upon a single element of design, construction, maintenance or operation, and
- alternative capability to prevent unacceptable releases.

Each of these is addressed in the examples in Reference [1] evaluated for this project.

However, there is uncertainty in the approach to applying the "reactor shutdown and decay heat removal" criteria. For example,

- For the frequent category, does "redundant" apply in addition to the failures that define the event sequence?
- For the infrequent category, does "at least one means..." apply independent of the event sequence frequency? Does "at least one barrier remains" apply to the entire frequency range?

To develop a licensing basis that can be implemented, these issues will need to be addressed and consensus reached between the various stakeholders.

4.10.4 Summary

The sample from Reference [1] discussed above provides a reasonable initial approach. Additional development and testing is warranted to:

- Compare the results which would be achieved using this approach to those obtained from application of existing practices.
- Specifically address how other D-in-D principles in Reference [1] would be addressed.
- Address the broad range in the infrequent category where allowable consequences very significantly.

This may benefit from additional review and consensus on the definition of D-in-D which was proposed above.

We have concluded that inclusion of deterministic D-in-D principles to support the use of results obtained from a PRA and other risk evaluations is appropriate. In order to develop a consistent approach the recommendations provided above should be addressed. In addition, the development of specific D-in-D acceptance criteria, consistent with the consensus approach, may be better addressed on a technology-specific basis. Additional work can confirm or alter this speculation. However, we did not attempt to provide a proposed solution to these issues in this project.

TABLE 4-12: LBE FREQUENCY CATEGORIES (based on Table 6-2 from Reference [1])		
Category	**Frequency**	**Basis**
Frequent	≥ 10-2/ry	Capture all event sequences expected to occur at least once in lifetime of a plant, assumed lifetime is 80 years
Infrequent	< 10-2/ry to >10-5/ry	Capture all event sequences expected to occur at least once in lifetime of population of plants, assumed population of 1000 reactors
Rare	<10-5/ry to ≥ 10-7/ry	Capture all event sequences not expected to occur in the lifetime of the plant population, but needed to assess Commission's safety goals

4.11 Cumulative Distribution of Risk as Measured by Complementary Cumulative Distribution Function

In this element the cumulative distribution of risk is determined in the form of a risk profile represented by a CCDF and/or other means. This feature is not fully included in either the NRC or the gas cooled reactor frameworks. If the results do not meet the specified acceptance criteria, design or operational changes would need to be identified and assessed.

Intent of CCDF: In Section 4.3, we discussed the basis for and use of a CCDF. As noted previously, the use of F-C functions can improve upon the practices used for currently operating plants, and perhaps on the practices used for the ALWRs. The limitations of F-C functions, however, are significant, and appear to have not been fully addressed in either framework. If an F-C function is used as a basis for licensing decisions and requirements, these limitations must be addressed, both with respect to the development of the F-C function and the process for its application. A specific recommendation provided in this report is to augment the F-C function with an analysis of the integrated risk impact by applying a CCDF. The issues which can be addressed by developing and using a CCDF include:

1. Uncertainty in the definition of an event sequence.
2. Uncertainty in the definition of an initiating event.
3. Establishing a common acceptable risk profile.
4. Development of a risk profile for comparison to acceptance criteria.
5. Treating sequences which do not meet the F-C function limits, when the total risk profile is acceptable.

TABLE 4-13: PRA AND LBE CRITERIA (based on Table 6-3 from Reference [1])			
Category **(Mean Event Frequency per ry)**	**Statistic for Meeting F-C curve**		**Additional acceptance criteria for LBEs (demonstrated with calculations at the 95% probability value with success criteria that meet adequate regulatory margin)**
	PRA	**LBE**	
Frequent ($\geq 10^{-2}$)	Mean	95% probability value*	- no barrier failure - no impact on safety analysis assumptions - redundant means for reactor shutdown and decay heat removal remain functional -annual dose to a receptor at the EAB $\leq$ 5mrem TEDE
Infrequent ($< 10^{-2}$ to $>10^{-5}$)	Mean	95% probability value*	- at least one barrier remains - a coolable geometry is maintained - at least one means of reactor shutdown and decay heat removal remains functional - for LBEs with frequency >1E-3 annual dose to a receptor at EAB $\leq$100mrem TEDE - for LBEs with frequency <1E-3 the worst two-hour dose at the EAB meets the F-C curve
Rare ($<10^{-5}$ to $\geq 10^{-7}$)	Mean	95% probability value*	- 24 hour dose at 1 mile from EAB meets the F-C curve
Note: With the exception of the source term, realistic calculations are carried out to obtain the mean value and uncertainty distribution of the important parameters for estimating frequency and consequences. Source Term calculations use the 95% probably value* of the amount of radionuclides released, obtained from a mechanistic calculation, and use Regulatory Guide 1.145 [25] or the equivalent for calculating atmospheric dispersion			
EAB – exclusion area boundary TEDE – total effective dose equivalent * The upper value of the 95% Bayesian probability interval			

NRC and GCR Approach: The approach used to address these issues in the gas cooled reactor work is unclear. The NRC framework provides criteria to partially address issues 3 and 4 identified above, and which is summarized in Table 4-13. As discussed in Reference [1],

- For the frequent event range (>1E-2 per year), "the cumulative dose meets the 5 mrem dose specification of Appendix I of 10 CFR 50."
- For the infrequent event range (between 1E-2 per year and 1E-3 per year), the cumulative dose of LBEs must meet the 100 mrem specification of 10 CFR Part 20.
- For the infrequent event range (between 1E-3 per year and 1E-5 per year), "the worst (maximum based on meteorological conditions) two hour dose at the EAB meets the F-C curve.
- For the rare event range (<1E-5 per year), "the 24 hour dose at one mile from the EAB meets the F-C curve."

The approach proposed in Reference [1] is intended to control the risk profile, and may be a reasonable approximation. A detailed review and comparison to the CCDF approach proposed here was not conducted in this project.

Conclusion: A CCDF is one means to address to address the above issues. We recommend this approach be further investigated.

5 CONCLUSIONS

The assessment conducted in this project suggests that the concepts being developed are promising, and provide a solid foundation for further development. Use of PRA and other risk evaluations in the design process could improve upon the processes which were used for currently operating plants; and possibly on the process used for the certified ALWR designs.

The conclusions obtained from this review are summarized below. Additional details can be obtained in Section 4.

Consensus Framework: The frameworks proposed by NRC (Reference [1]) and the gas reactor NSSS vendors (as for example in Reference [3]) differ. A framework which includes the best features of these frameworks was developed during this research and is discussed in Section 3 of this report.

> Recommendation 1: Further development of a consensus framework to improve interactions with all stakeholders and to assist in reaching agreement on the most challenging elements in these frameworks. The consensus framework developed as part of this research can serve as a starting point to achieve this objective.

Fundamental Safety and Design Principles: In this project the safety expectations and design criteria discussed in the frameworks proposed by the NRC and gas reactor vendors were combined into safety and associated design principles, so as to establish and use common terminology.

> Recommendation 2: Develop consensus Fundamental Safety and Design Principles to improve interactions with all stakeholders and to assist in reaching agreement on the most challenging elements in these frameworks. The FSPs and FDPs developed as part of this research can serve as a starting point to achieve this objective.

QHOs: The QHOs are used as one of the acceptance criteria in both frameworks. The use of QHOs in the NRC developed framework appears reasonable and appropriate. The key issue is the lack of the development and use of a risk profile, possibly in the form of a CCDF. The use of QHOs in the framework proposed by the gas reactor vendors appears to be overly ambitious. Additionally, a risk profile, or other means of controlling the allowable dose versus total frequency, does not appear to be included in this framework. This is the case for all event categories (AOOs, DBEs, and BDBEs.) We were unable, during this project, to develop a technical basis for concluding that solely meeting the QHOs and the proposed deterministic criteria for AOOs and DBEs would result in an acceptable risk profile.

Recommendation 3: The work under way to support the pre-licensing activity summarized in Reference [3] should consider the approach proposed in Reference [1] and the assessment results and insights documented in this report. The basis for the decision criteria should be enhanced or the criteria changed as needed to address this report's findings.

Frequency Consequence Function and Process: The use of F-C functions can improve upon the practices used for currently operating plants, and perhaps on the practices used for the certified ALWR designs. The limitations of F-C functions, however, are significant, and appear to have not been fully addressed in either the NRC or gas reactor vendor proposed frameworks (or even in this project). If an F-C function will be employed to serve as a licensing basis, these limitations associated with the development of the F-C function and the process for using it must be addressed. Issues and limitations identified include the following:

- Developing a basis for the frequency for a specific consequence level.

  The two approaches reviewed result in F-C functions which vary by several orders of magnitude at several consequences levels. It is unclear how an F-C function can be developed independent of the process for using the function and without reliance on an acceptable risk profile.

  Recommendation 4: The F-C function proposed in Reference [1] appears reasonable, and can most likely be improved by considering the capabilities of currently operating plants. The F-C function proposed in Reference [3] is much less restrictive than has been determined for currently operating plants. Since an objective of advanced nuclear power plants will be to demonstrate they will operate at a level of safety that meets or exceeds that of current LWR technology, future work should consider the alternatives discussed in Section 4 to develop a revised F-C function that possesses these characteristics.

- Event sequence definition.

  In the proposed frameworks, the requirements for event sequence definition are not entirely clear. For example, for a specific initiating event there are several means to display a sequence which results in a comparable consequence level. A sequence displayed at the basic event level would have a lower frequency than if aggregated at the system or function level. For the proposed framework to serve as an acceptable licensing basis, this issue needs to be addressed.

  Recommendation 5: The development of an F-C function must consider the process for using the function. Event sequences should be defined consistent with their use in the F-C process and on the basis of the F-C function.

- Initiating event definition.

  Defining an initiating event has challenges similar to the definition of event sequences. Similar to this issue, if the proposed framework is to serve as an acceptable licensing basis, this issue also needs to be addressed.

Recommendation 6: Similar to defining an event sequence, research needs to be performed to develop a consensus process to define IEs.

- Aggregate risk.

  Since the F-C functions under development treat sequences and not the aggregate risk profile resulting from the sequences, a method of evaluating the aggregate risk is required. As discussed in the body of the report, sole reliance on the QHOs is not sufficient to achieve this objective.

  Recommendation 7: The NRC framework proposed in Reference 1 recognizes this feature of the QHOs and applies other integrated risk acceptance criteria. A CCDF, or other means, should be developed and integrated into the licensing basis.

- Requiring every sequence to meet the F-C function limits does not appear to be a reasonable approach to providing a licensing basis that assures adequate plant safety.

  In the proposed frameworks, if any one sequence is above the F-C limit (regardless of its postulated frequency), the design is considered unacceptable, whereas, if the same event is just below the limiting value of the F-C function, the design is acceptable. This application of a "hard acceptance threshold" provides limited flexibility to both the regulatory authority and reactor designers.

  Recommendation 8: Rather than treating individual sequences, or families, as needing to always meet the F-C limits, a CCDF should be used. In this manner the benefits of F-C functions and the sequences which populate the function are preserved, but an unnecessary limitation is removed.

  Recommendation 9: A new, reference F-C function and process should be developed that integrates and improves on the F-C functions proposed by the NRC and gas cooled reactor vendor frameworks. The F-C function developed should consider the issues and actions discussed above, and in detail in Section 4, to reach a consensus framework that addresses the interests of all stakeholders.

PRA and Other Risk Evaluations: PRA is used extensively in the frameworks under development. These frameworks, if not changed, would require a full scope, level 3 PRA (for varying dose levels from benign to severe) for all hazards. This PRA would most likely require documentation and reviews in consonance with 10 CFR Part 50 Appendix B quality assurance requirements, and would be subjected to independent peer review.

Issues identified in this project and addressed by candidate changes included in the integrated reference framework are as follows:

- "Other Risk Evaluations": The frameworks under development use a full scope PRA as noted above. "Other risk evaluations" refers to risk evaluations which are not fully quantitative, such as the PRA-based seismic margins approach used for the certified, advanced designs.

  Recommendation 10: In this project we acknowledge that certain hazards and operating modes might be better, or sufficiently, addressed using approaches and methods other than PRA that are not fully quantitative. We recommend establishing approaches, similar to those used for ALWR designs certified using 10 CFR Part 52, for addressing hazards such as seismic and other external hazards where PRA is not needed to demonstrate an adequate safety case.

- PRA for AOOs: AOOs are anticipated events. All hazards at a plant site are to be addressed in the frameworks under development. Some hazards may have such minor potential consequences that the quantitative frequency of an AOO or DBE is not important. That is, the hazard can be treatment deterministically.

  Recommendation 11: Certain AOOs may have such small potential consequences that modeling with PRA techniques will be unnecessary. The framework should be enhanced to allow for such screening.

Uncertainty Analyses for Comparison to QHOs and F-C Limits: In both the proposed NRC and gas reactor vendor developed frameworks, the mean results of the PRA are proposed for comparison to the QHOs and F-C limits. LBEs are addressed differently.

Recommendation 12: Mean results appear to be appropriate for comparison to the QHOs and F-C limits. However, conservative analyses should be permitted as an alternative when their application would not impact the conclusions obtained. This alternative could support more efficiency in both the analysis and regulatory review process.

Selection of Safety Significant SSCs: The approach developed by the gas reactor vendors and described in Reference [20] is similar to the NRC proposed framework (Reference [1]); however the NRC developed approach would assign a safety significant classification to all SSCs required to meet the F-C function at a 95% confidence level on both frequency and consequence. The basis for the approach in Reference 20 not addressing the complete F-C function is uncertain. Regardless, this identification using either approach is reasonably straightforward for standby SSCs. The approach to identifying safety significant SSCs which could cause an initiating event is not fully developed, and is not specifically addressed, in either framework. Instead, a living PRA is used in both frameworks and target frequencies for initiating event categories are considered in the NRC developed framework.

Recommendation 13: Neither the NRC nor gas reactor vendor frameworks explicitly address the need to consider initiating events in the evaluation of safety significant SSCs. It is recommended that an approach to evaluating initiating events for this purpose be developed. The approach should consider practices that currently are applied to operating LWRs (e.g. in SSC classification for the Maintenance Rule 10CFR 50.65).

Licensing Basis Events and Design Basis Events: The NRC framework (described in Reference [1]) is straightforward and generally reasonable. The key uncertainty in using the approach is expected to be the decisions which will be required to determine event sequence groups. Presumably the intent of this grouping is related to focusing and managing the level of effort in conducting the uncertainty analyses. Use of a 95% confidence level for consequence appears reasonable for the frequent (event sequences with frequency >1E-2/yr.) and infrequent (event sequences with frequency between 1E-2/yr. and 1E-5/yr.) categories, as this approach is consistent with current practices. The use of a 95% confidence level for consequence for the rare category (event sequences with frequency <1E-5/yr.) appears to be overly restrictive. A mean value should be considered for the rare category. The approach proposed by the gas reactor vendors differs from this with the following significant differences:

- Mean values are used, except for the DBA category.
- Consequences of BDBEs are not addressed.
- Explicit consideration of meeting both frequency and consequence limits for the spectrum of the F-C function is not addressed. Instead, preventing a movement to a lower consequence LBE category is considered.
- The approach to analyzing DBAs is unclear. The discussion appears to not consider failures of safety-related SSCs (i.e., the proposed framework appears to assume such SSCs respond as designed with 100% reliability) and appears to use 25 rem TEDE at the EAB regardless of the frequency of the DBA.

> Recommendation 14: We propose the NRC approach be used as a starting point to address this issue, but that mean values be used for the rare event categories. We recommend the differences between the two frameworks be reviewed and consensus obtained.

Defense in Depth and Safety Margins: We have concluded that inclusion of deterministic D-in-D principles to support the use of results obtained from a PRA and other risk evaluations is appropriate. The approach developed by NRC appears to be a reasonable starting point for reaching consensus. In order to develop a consistent approach the recommendations provided below should be addressed.

> Recommendation 15: Compare the results which would be achieved using the proposed framework to results achieved using existing practices.

> Recommendation 16: Specifically address how other D-in-D principles discussed in Reference 1 would be addressed.

> Recommendation 17: Address the broad range in the infrequent category where allowable consequences can vary significantly.

> Recommendation 18: In addition, the development of specific D-in-D acceptance criteria, consistent with the consensus approach, may be better addressed on a technology-specific basis. Additional work can confirm or alter this speculation.

Terms and Definitions: Many terms, such as event and event sequence, are not fully defined in the references reviewed. This is especially important when using an F-C function on a sequence-specific or family basis. Otherwise, two experts working separately could develop significantly different results and conclusions. Because the frameworks are ultimately intended to be used as a licensing basis for advanced reactors, explicit agreement on terms and definitions will be required.

> Recommendation 19: Develop explicit definitions of all significant terms applicable to the licensing framework, and provide examples (including multiple examples) where needed. Specifically address all key terms, which include "event" and "event sequence."

6
REFERENCES

1. U.S. Nuclear Regulatory Commission, Office of Nuclear Regulatory Research, NUREG-1860, Working Draft, "Framework for Development of a Risk-Informed, Performance-Based Alternative to 10CFR Part 50," July 2006.

2. Nuclear Energy Institute, NEI 02-02, "A Risk-Informed, Performance-Based Regulatory Framework for Power Reactors," May 2002.

3. Pebble Bed Modular Reactor (Pty) Ltd., White Paper, "Licensing Basis Event Selection for the Pebble Bed Modular Reactor, Revision 1," submitted to U.S. NRC July 3, 2006.

4. Pebble Bed Modular Reactor (Pty) Ltd., White Paper, "Probabilistic Risk Assessment Approach for the Pebble Bed Modular Reactor, Revision 1," submitted to U.S. NRC June 13, 2006.

5. Fred A. Silady, "Licensing Approach for Modular HTGRs," Proceedings of PSA 2005, American Nuclear Society, September 2005.

6. U.S. DOE, "Preliminary Safety Information Document for the Standard Modular High Temperature Gas-Cooled reactor," DOE-HTGR-86-024, Amendment 13, August, 1992.

7. EXELON GENERATION COMPANY, "Proposed Licensing Approach for the Pebble Bed Modular Reactor in the United States," submitted to U.S. NRC, March 2002.

8. U.S. DOE, "Top-Level Regulatory Criteria for the Standard Modular High Temperature Gas-Cooled Reactor," Department of energy, DOE-HTGR-85002, Revision 3, September 1989.

9. IAEA, Safety Standards Series No. NS-R-1, "Safety of Nuclear Power Plants: Design" September 2000.

10. IAEA, International Nuclear Safety Advisory Group, "Basis Safety Principles for Nuclear Power Plants," 75-INSAG-3 Revision 1, INSAG-12, October 1999.

11. IAEA, International Nuclear Safety Advisory Group, "Defence in Depth in Nuclear Safety," INSAG-10, June 1996.

12. American National Standards Institute, Inc., "Nuclear Safety Criteria for Design of Stationary Pressurized Water Reactors," ANSI N18.2-1973, Approved August 6, 1973.

13. U.S. Nuclear Regulatory Commission, "Safety Goals for the Operation of Nuclear Power Plants; Policy Statement," Federal Register, Vol. 51, p. 30028, August 21, 1986.

14. Karl Fleming and Fred Silady, Technology Insights, "A Risk Informed Defense-in-Depth Framework for Existing and Advanced Reactors, Reliability Engineering and System Safety, 78 (2002).

15. U.S. Nuclear Regulatory Commission, prepared by Brookhaven National Laboratory, NUREG/CR-0603, "A Risk Assessment of a Pressurized Water Reactor for Class 3-8 Accidents, October 1979.

16. International Commission on Radiation Protection, "ICRP Publication 64: Protection from Potential Exposure: A conceptual Framework," Annals of the ICRP Volume 23/1, Elsevier, May 1993.

17. U.S. Environmental Protection Agency, Office of Radiation and Indoor Air Radiation Protection Division, "Protective Action Guides (PAG) Manual."

18. U.S. Nuclear Regulatory Commission, "Abnormal Occurrence Reports: Implementation of Section 208 Energy Reorganization Act of 1974," 62 FR 18820, 1997.

19. International Commission on Radiation Protection, "ICRP: Non-stochastic effects in ionizing radiation" 1984.

20. Pebble Bed Modular Reactor (Pty) Ltd., White Paper, "Safety Classification of Structures, Systems, and Components for the Pebble Bed Modular Reactor," Revision 1," submitted to U.S. NRC August 28, 2006.

21. "Commission Policy Statement on the Regulation of Advanced Nuclear Plants," 59 FR 35461, 1994.

22. U.S. Nuclear Regulatory Commission, Office of Nuclear Regulatory Research, Regulatory Guide 1.174, "An approach for Using Probabilistic Risk Assessment in Risk-Informed Decisions on Plant-Specific Changes to the Licensing Basis."

23. N. Rasmussen, et. al., "Reactor Safety Study – An Assessment of Accident Risks in U. S. Commercial Nuclear Power Plants", U.S. Nuclear Regulatory Commission Report WASH-1400, NUREG-75/014; October 1975.

24. U.S. Nuclear Regulatory Commission, Office of Policy Evaluation, NUREG-0880, "Safety Goals for Nuclear Power Plant Operation, Revision 1," May 1983.

25. U.S. Nuclear Regulatory Commission, Regulatory Guide 1.145, "Atmospheric Dispersion Models for Potential Accident Consequence Assessments at Nuclear Power Plants."

7 ACRONYMS AND DEFINITIONS

Acronym	Definition
ALARA	As Low as Reasonably Achievable
ALWR	Advanced Light Water Reactor
ANPR	Advanced Notice of Proposed Rulemaking
ANS	American Nuclear Society
AO	Abnormal Occurrence
AOOs	Anticipated Operational Occurrences (e.g., a loss of feedwater initiating event using existing practice or a loss of feedwater initiating event plus failure of an emergency feedwater pump using the approaches in the frameworks under development.) Note: Existing practice defines an AOO as an initiating event which would be expected to occur during a calendar year or during the lifetime of a particular plant (This would generally include incidents of moderate frequency and infrequent incidents - e.g., see Reference [12].) The frameworks under development would define an AOO as a sequence which is either planned or anticipated (one or more times) during a particular plant's lifetime
ASME	American Society of Mechanical Engineers
ATWS	Anticipated Transient without Scram
BDBE	Beyond Design Basis Event. Using existing practice, these are rare events, or event sequences, which are not within the design basis and which may have lower frequencies than events within the design basis (e.g., severe core damage and failure of the containment.) For the approaches under development these are defined as event sequences with frequencies lower than the lower bound frequency for DBEs.
CAB	Controlled Area Boundary

CCDF	Complementary Cumulative Distribution Function
CDF	Core Damage Frequency
COL	Combined Operating License
CFRs	Code of Federal Regulations
DBA	Design Basis Accident - Note: The definition of this term varies in different documents. For example, for existing practices, a DBA is generally referred to use of an assumed sequence which results in a source term used for reactor siting, containment and containment systems. In contrast, for the gas cooled reactor framework under development, a DBA is referred to as a deterministic event sequence in which only SSCs classified as safety-related are available.
DBEs	Design Basis Events (e.g., a large LOCA using existing practice or a loss of feedwater initiating event plus failure of 2 emergency feedwater pumps using the approaches in the frameworks under development.) Note: Existing practice would characterize a DBE as a limiting fault initiating event that would not be expected to occur, but which is postulated because its consequences would include the potential for a release of significant amounts of radioactive material. Although the frameworks under development do not have consistent definitions, they each characterize a DBE, or its equivalent, as a sequence which is not expected in the plant's lifetime, but which might occur in the lifetime of a fleet of plants.
DCD	Design Control Document
D-in-D	Defense-in-Depth
DOE	United States Department of Energy
EAB	Exclusion Area Boundary
EIS	Environmental Impact Statement
EPA	United States Environmental Protection Agency
EPRI	Electric Power Research Institute
F-C	Frequency-Consequence
FDPs	Fundamental Design Principles

FSAR	Final Safety Analysis Report
FSPs	Fundamental Safety Principles
GCR	Gas Cooled Reactor
GDC	General Design Criteria
HPB	Helium Pressure Boundary
HTGR	High Temperature Gas Reactor
IAEA	International Atomic Energy Agency
IE	Initiating Event
ISI	In-Service Inspection
IST	In-Service Testing
KSFs	Key Safety Functions
LB	Licensing Basis
LB LOCA	Large Break LOCA Initiating Event
LBEs	Licensing Basis Events
LCOs	Limiting Conditions for Operation
LERF	Large Early Release frequency
LOCA	Loss of Coolant Accident
LOOP	Loss of Offsite Power Initiating Event
LPZ	Low Population Zone
LWR	Light Water Reactor
MSLB	Main Steam Line Break Initiating Event
NEI	Nuclear Energy Institute

NGNP	Next Generation Nuclear Plant
NNS	Non-nuclear safety
NOPR	Notice of Proposed Rulemaking
NRC	United States Nuclear Regulatory Commission
NSSS	Nuclear Steam Supply System
QA	Quality Assurance
OLs	Operating Limits
PAG	Protective Action Guidelines
PB	Performance-based, which, from Reference 1, is a "descriptor of processes that can be monitored by quantitative measures of performance."
PBMR	Pebble Bed Modular Reactor
PRA	Probabilistic Risk Assessment
PTS	Pressurized Thermal Shock
PWR	Pressurized Water Reactor
QHOs	Quantitative Health Objectives
RAP	Reliability Assurance Program
RCS	Reactor Coolant System
RI	Risk-informed, which, from Reference 1, is "a characteristic of regulatory decision-making that includes results and findings that derive from risk assessments and other factors that are designed to better focus licensee and regulatory attention on design and operation issues commensurate with their importance to health and safety."
ROP	Regulatory Oversight Process
RTNSS	Regulatory Treatment of Non-safety Systems

SAMDA	Severe Accident Mitigation Design Alternatives
SAR	Safety Analysis Report
SB LOCA	Small Break LOCA Initiating Event
SBO	Station Blackout
SC	Success Criteria
SFs	Safety Functions
SGs	Safety Goals
SSCs	Systems, Structures and Components
TEDE	Total Effective Dose Equivalent
TLDC	Top Level Design Criteria
TLRC	Top Level Review Criteria
TN	Technology-neutral

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Framework for Future Plant Licensing

Presented to Advisory Committee on Reactor Safeguards

Future Plant Designs Sub-Committee

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March 7, 2007

Purpose of Today's Briefing

- Technical exchange with ACRS on the technical issues addressed in the "Framework for Future Plant Licensing"
  - Referred to as the "Technology-Neutral Framework"

Outline

- Introductory remarks
  - History
  - Status
  - Stakeholder comments
  - Next steps
- Framework overview
- Round table discussion

History

- January 2003: RES Advanced Reactor Research Plan recognized the need for a licensing framework for advanced reactors
- Current regulatory structure
  - Focused on LWRs with limited application to non-LWRs
    - Advanced reactors will have design and operational issues different from LWRs
    - Contain specific requirements not applicable to advanced reactor designs
  - Evolved with limited insights from PRAs and severe accident research
    - PRA and PRA insights will be an integral part of licensing advanced reactors
- Program initiated to develop a risk-informed, performance-based regulatory structure that could be technology-neutral to support future licensing

Status

- Initial work (development of the “Framework”) complete
  - Documented in NUREG-1860, to be published
- Framework provides guidance and criteria for creating a “risk-derived” and performance-based regulatory structure that can be implemented on either a technology-neutral or a technology-specific basis
- Framework integrates Commission’s expectations as addressed in various policy statements
  - Severe Accident, Advanced Reactors, PRA, Safety Goals

Stakeholder Comments

- Framework attached to ANPR (May 2006) requesting public review and comment
- Public workshops held March 2005 and September 2006
- Stakeholder comments received from:
  - AREVA
  - ASME (two sets)
  - Strategic Teaming and Resource Sharing (STARS)
  - Nuclear Energy Institute (two sets)
  - American Nuclear Society (two sets)
  - Pebble Bed Modular Reactor (Pty) Ltd.
  - Westinghouse
  - IEEE Power Engineering Society
  - GE Energy Nuclear
  - Nuclear Equipment Quorum
- High level comments with regard to rulemaking
  - Overall views
  - Technology-neutral versus technology-specific
  - How to proceed forward

Stakeholder Comments

Risk-Informed and Performance-Based "Part 53"

- Example comments
  - "should move forward with developing a risk-informed and performance based…"
  - "supports the NRC's efforts to improve … its regulations by establishing … a comprehensive set of risk-informed and performance-based…"
  - "support a regulatory framework that would establish a comprehensive set of risk-informed and performance-based…"
  - "departs too far from the approximately 3000 reactor years experience gained using the deterministic approach … the significant area of departure … in addressing common cause failure…"

Stakeholder Comments

Technology-Neutral vs Technology-Specific

- Mixture of views
  - Some supported technology-neutral regulations with technology-specific implementing guidance
  - Some supported technology-specific regulations
  - Some indicated too premature to decide

Stakeholder Comments

How to proceed forward with regard to rulemaking

- Gain experience first with design certification of a non-LWR using Framework approach
- Use a multi-year phased approach to rulemaking
- Use a Step approach
  - Develop a preliminary draft rule
  - Upon receipt of non-LWR application, publish the draft rule for information
  - Review and approve non-LWR design using Part 50/52
  - Evaluate draft rule against non-LWR design
  - Publish draft rule for comment

Next Steps

- Framework, NUREG-1860, to be published early summer 2007
- Staff preparing SECY paper to respond to Commission direction to "provide its [staff] recommendation on whether and, if so, how to proceed with rulemaking"
  - All activities related to Framework to be terminated
  - Evaluating the need to defer rulemaking until experience is gained with NGNP and GNEP
- Staff will brief ACRS at May full-committee meeting

Framework Overview

What is this Framework?

- It is a "NUREG" – a technical report that provides a structured and systematic approach in the form of guidelines and criteria for developing new requirements

⇒ ***The "Framework" is a set of technical guidelines and criteria***

- The Framework itself is not regulations

The Framework

- Could serve as the technical basis for rulemaking (e.g., “Part 53”), exemptions or additions to Part 50
- Uses a “risk-derived” approach
- Can be applied or implemented (i.e., development of requirements) on either a technology-neutral basis or a technology-specific basis

Risk-Derived versus Risk-Informed

- Risk-derived approach starts with PRA results and integrates deterministic and defense-in-depth criteria (to compensate for uncertainties) as an integral part in development of the requirement

- Risk-informed approach uses deterministic criteria to develop the requirements and then supplements with risk insights

The Framework

The Framework

Risk-Derived Set of Design, Maintenance and Operating Regulations or Regulatory Guidance

Guidelines, criteria, ground-rules for determining regulations for risk-derived technical licensing structure

Example Application of Framework

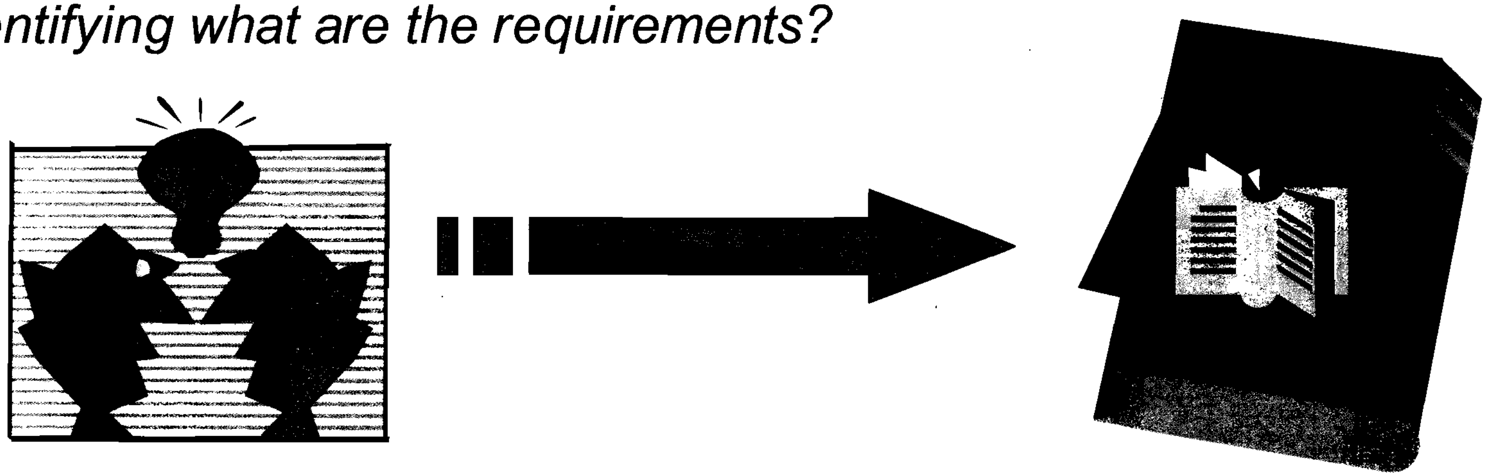
Example Regulations	Source	Tech-N/S
• **Aging Management Program:** Each applicant to construct and operate a NPP under this Part shall develop, implement and maintain an aging management program to detect and control aging of safety significant SSCs so as to maintain the plant within the assumptions used in the licensing analysis. A description of the aging management program shall be submitted to the NRC for review.	FW	TN
• **Requirements for monitoring the effectiveness of maintenance at nuclear power plants:** 10 CFR §50.65 language plus A maintenance program shall be developed, implemented and maintained to ensure that the reliability, availability and performance of safety significant SSCs remain consistent with assumptions in the licensing analysis. The SSC reliability, availability and performance shall be monitored and fed back into the licensing analysis.	Part 50 + FW	TN
• **Energetic Reaction Control:** Reactor designs that have the potential for energetic reactions between the fuel, coolant or other material shall include provisions to prevent or mitigate the effects of such reactions.	FW	TN/TS

Example Application of Framework

Example Regulations	Source	Tech-N/S
• **PRA Scope and Technical Acceptability**: each application to construct and operate a NPP shall include a design specific probabilistic risk-assessment (PRA) that (a) analyzes the risk from full power and low power operation, shutdown, refueling, and spent fuel storage (except dry cask storage); (b) includes assessment of internal and external events and quantifies uncertainties; (c) includes assessment of all event sequences down to 10-8/yr; and (d) is conducted in accordance with accepted standards appropriate for the reactor technology.	FW	TN
• **Living PRA:** Each licensee to operate a NPP shall maintain its licensing analysis up to date. The plant specific PRA shall be updated to reflect actual operating experience at least once every xx years, or sooner if major unanalyzed situations are discovered. The information from the updated PRA shall be used to update the plant's licensing basis including: LBE selection and analysis, safety classification of SSCs, procedures, NDE, ISI, and IST programs, plant aging program, emergency preparedness. Major changes resulting from these updates will require NRC approval.	FW	TN

How do we get there?

How do we take this idea for a risk-derived, performance-based set of regulations they may apply to any reactor technology and actually start identifying what are the requirements?



- Need a "process"
- Process should define a goal and the guidelines and criteria for achieving the goal
- Process should address completeness

Framework Process

Define the Goal

Define the Structure to Identify the Requirements

Define the Guidelines and Criteria to Meet the Overall Goal

"Turn the Crank"

Implement the Guidelines and Criteria to Identify the Requirements

Protect the public health and safety

Protective strategies needed to ensure public health and safety

- Probabilistic criteria
- Defense-in-depth
- PRA technical acceptability

- Design, maintenance, Operational
- Technology-neutral
- Technology-specific

Guidelines/Criteria Contained in Framework

- Risk-derived/probabilistic approach
  - Level of safety and integrated risk
  - Frequency-consequence curve
  - Licensing basis event identification and selection
  - Safety Classification
- Defense-in-depth
  - Definition
  - Principles
  - Implementation
  - Safety margins

Guidelines/Criteria Contained in Framework (cont'd)

- PRA Technical Acceptability
  - Scope and level of detail
  - Design stage vs operational stage
  - Living
- Identification of Design, Maintenance and Operation Requirements
  - Scope and depth of requirement similar to current GDCs
  - Keep applicable Part 50 regulations

“Round Table Discussion”

For each topic:

- Key issues
- ACRS views
- Stakeholders views

Probabilistic approach: Frequency- consequence curve

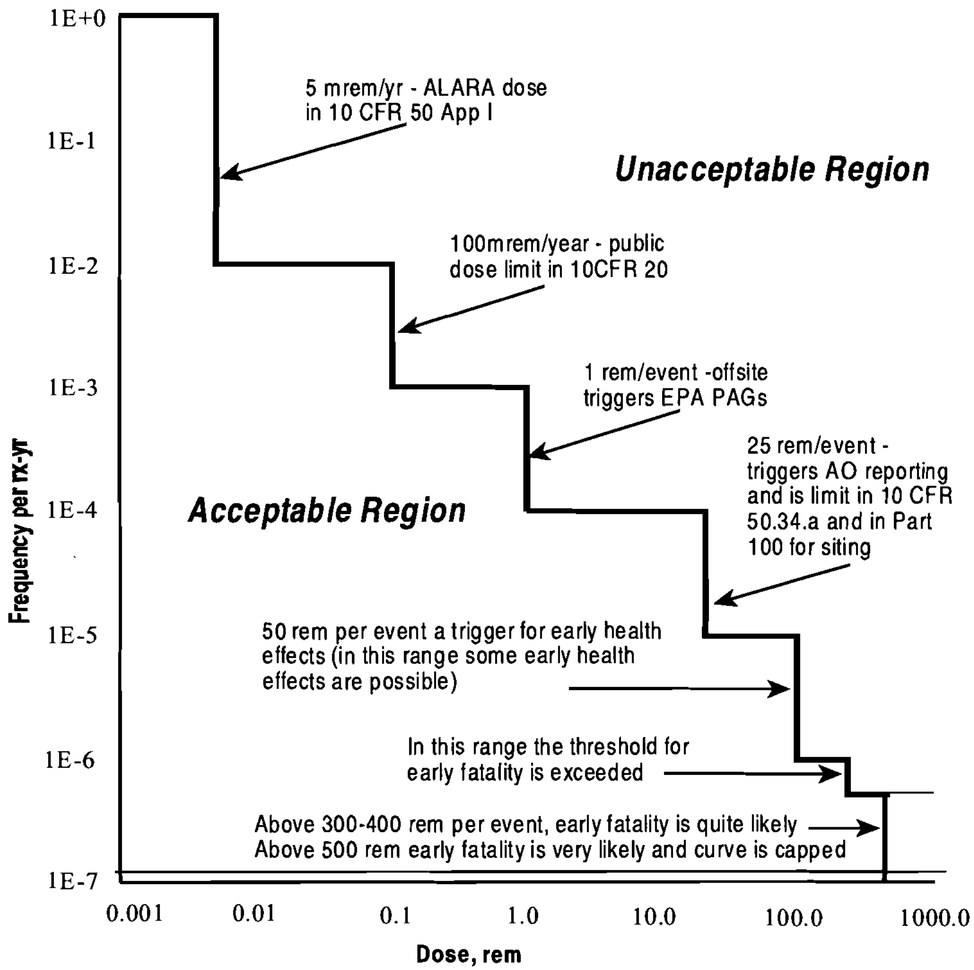


Figure 6-3 Frequency -consequence curve

Stakeholder comments:

- Different anchor points
- Add a CCDF curve

Probabilistic approach: LBE (Licensing Basis Event) Selection

- Identification and selection based on PRA sequences
  - Similar accident sequences are grouped into 'event classes'
  - LBE represents the event class scenario and is assigned the 'bounding' frequency and consequences of that class - has to meet f-c curve
  - LBEs have to meet additional deterministic criteria depending on frequency range they fall in: frequent, infrequent, or rare
- Stakeholder comments generally supportive of approach
  - Some differences in the detail; e.g., different "cut-off" values defining the event categories

Probabilistic approach: SSC selection / special treatment

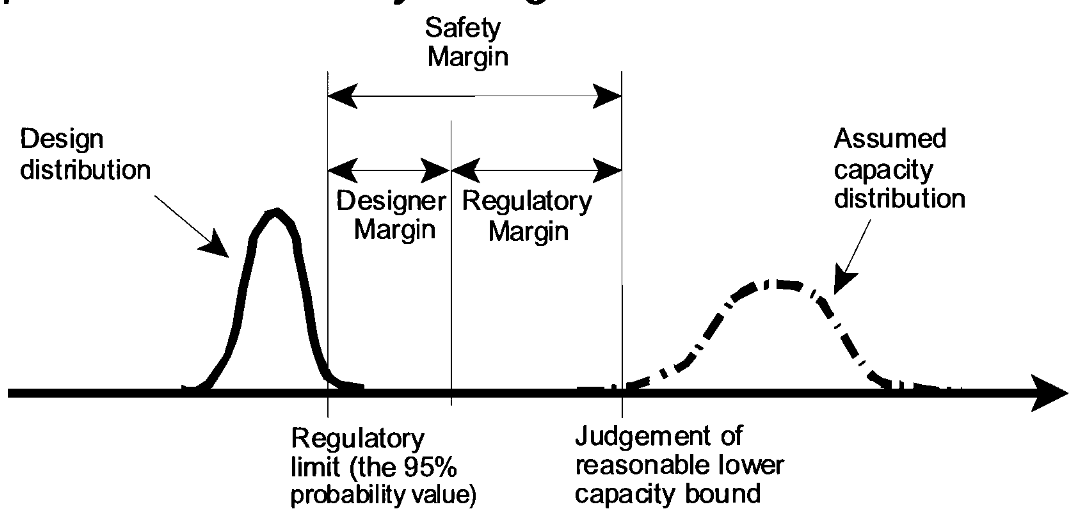
- Only two categories of SSCs
  - Safety significant
  - Non-safety significant
- Safety significant SSCs are all those whose functionality is credited in meeting the LBE acceptance criteria
- Special treatment varies depending on the function and importance of the SSC

Defense-in-depth

- *Definition:* An element of NRC's safety philosophy that is used to address uncertainty by employing successive measures including safety margins to prevent and mitigate damage if a malfunction, accident or naturally caused event occurs at a nuclear facility.
- Six principles of defense-in-depth are presented in framework, related to:
  - Intentional as well as inadvertent events
  - Prevention and mitigation capability
  - Diversity for key safety functions
  - Uncertainty in SSCs and human performance
  - Containment functional capability
  - Plant siting
- Stakeholder comments:
  - Defense-in-depth should be a separate policy statement
  - Additional clarity is needed

Defense-in-depth (continued)

- *Defense-in-depth implementation*
  - *Use of Protective strategies*
  - *Application of Defense-in-depth principles*
  - *Incorporation of Safety margins*



- Stakeholder comments
  - ASME approach to defining safety margin generally uses the mean while the Framework uses the 95%

PRA Requirements

Technical Acceptability	High level requirements provided. Increased requirements are included for completeness, defensibility and transparency
Scope	Encompasses the whole spectrum of off-normal events including sequences that address conditions less than the core damage sequences of the current reactors and those similar to current reactor core damage sequences
Quality Assurance	More rigorous requirements than those that are typical for current PRAs. Recommended requirements are similar to Appendix B.
Independent Peer Review	Performed by qualified personnel using an established process similar to current peer review requirements.

PRA Stages

- Design Stage
  - Integrated with design process and expected to evolve as the design evolves
  - Supports
    - Evaluation of QHOs
    - Identification and characterization of LBEs
    - Identification and characterization of special treatment SSCs
- Construction Stage
  - Supports risk-informing the construction inspection program
- Startup (Pre-operational) Stage
  - Supports risk-informing initial staffing, training and other programs (technical specifications, testing, maintenance, procedures, etc.)
- Operational Stage (Living PRA)
  - Supports the assessment and management of operational risk and of plant changes

Living PRA

Input Monitoring	Process similar to the monitoring of the performance and condition of SSCs against licensee-established goals of 10 CFR 50.65. PRA related SSCs are monitored and compared with the framework's reliability and availability goals to verify that the goals are being met.
Planned Plant Changes	Process similar to 10 CFR 50.59. Plant changes are assessed using the PRA prior to implementation.
Unplanned Changes	Can result in changes to the frequency or consequence of identified LBEs, in the identification of new LBEs or in changes to safety significant SSCs. Changes that reduce margin but do not impact the framework's design criteria will not require reassessment.
Update Frequency	Primarily dependent on the scope and nature of pending changes. A maximum update interval of 5 years is proposed.

PRA Stakeholder Comments

- Full-scope PRA not necessary for such areas as seismic, anticipated operational events
- Determination of PRA quality does not need to be based on an 10 CFR Part 50 Appendix B approach for Quality Assurance

Requirements Development

- **Process Described in Chapter 8 (Figure 8-1)**
  - For each protective strategy, a logic diagram is used to identify what could go wrong to cause the strategy to not be met.
  - Requirements are intended to address the items that could go wrong.
  - DID principles applied to each protective strategy to identify where additional requirements are necessary to address uncertainties.

- **Example Requirements are Contained in Appendix J**
  - Technology neutral
  - Level of detail similar to GDCs

- **Completeness Check**
  - IAEA, NEI02-02, 10 CFR 50

- **Stakeholder Comments**
  - General agreement process seems reasonable.

Level of safety / Integrated risk

- Regulations for new reactors should achieve the safety goal level of safety
- Safety goals should apply to the collective risk from the entire fleet of new reactors on a site
- Stakeholder comments:
  - General agreement with staff recommendation on both issues

Summary

- Framework, NUREG-1860,to be published in early summer 2007
- Stakeholder comments will be summarized into five categories in an appendix
  - Comment is more of an observation and no change to the Framework is needed.
  - Comment is associated with implementation of the Framework. The issue(s) raised will be addressed dependent on if, how, and when the Framework is implemented.
  - Comment is significant enough that the framework needs to be revised before it is released.
  - Comment is more of a clarification and does not change the technical basis in the framework; will be addressed dependent on if, how, and when the Framework is implemented.
  - The staff disagrees with the comment and no change is made to the Framework.