RULES AND DIRECTIVES
BRANCH

PR 50 and 53
(71FR26267)

15 3:15

RECEIVED

AREVA

(1)

DOCKETED
USNRC

August 17, 2006 (11:42am)

OFFICE OF SECRETARY
RULEMAKINGS AND
ADJUDICATIONS STAFF

August 15, 2006
NRC:06:033

Secretary,
US Nuclear Regulatory Commission
Washington, DC 20555-0001

ATTN: Rulemaking and Adjudications Staff

Comments on 10 CFR 50 and 53 (RIN 3150-AH81) – Approaches to Risk-Informed and Performance Based Requirements for Nuclear Power Reactors.

AREVA NP, Inc. reviewed subject Advance Notice of Proposed Rulemaking (ANPR) and offers the attached responses to the NRC questions published in the Federal Register *Vol. 71, No. 86, pp. 26267-26275.* These responses represent AREVA NP's view on the proposed 67 exploratory questions in the ANPR and the new regulatory framework structure as delineated in the working draft NUREG-1860 and discussed in the June 15, 2006 NRC-sponsored meeting on this subject.

AREVA NP continues to support the efforts of the NRC to develop a risk-informed and technology-neutral framework and subsequent Regulations for licensing new reactors and we look forward to future interactions on this subject in your September 2006 NRC Sponsored workshop.

Sincerely,

Ronnie L. Gardner

Ronnie L. Gardner, Manager
Site Operations and Regulatory Affairs
AREVA NP

Enclosures

cc:	Adrian Heymer	(NEI)
	J. F. Williams	(NRC-NRR)
	M. T. Drouin	(NRC-RES)
	J. L. Birmingham	(NRC-NRR)

Template=SECY-067

SECY-02

AREVA NP INC.
An AREVA and Siemens company

3315 Old Forest Road, P.O. Box 10935 Lynchburg, VA 24506-0935
Tel.: 434 832 3000 - Fax: 434 832 3840 - www.areva.com

Attachment 1

AREVA NP Responses to NRC Questions on Technology-Neutral Framework and the Associated ANPR

Proposed Plan

The NRC has developed a proposed plan to develop an integrated risk-informed and performance-based revision to 10 CFR Part 50 that would cover power reactor applications including non-LWR reactor designs. To accomplish this goal, safety, security, and preparedness will be integrated into one cohesive structure. This structure will ensure that the reactor regulations, and staff processes and programs are built on a unified safety concept and are properly integrated so that they complement one another. Based on the above, the overall objectives of a risk-informed and performance-based revision to 10 CFR Part 50 are to:

(1) Enhance safety and security by focusing NRC and licensee resources in areas commensurate with their importance to health and safety,
(2) provide NRC with the framework to use risk information in an integrated manner to take action in reactor regulatory matters,
(3) use risk information to provide flexibility in plant design and operation, which can result in burden reduction without compromising safety and security,
(4) ensure that risk-informed activities are coherently and properly integrated such that they complement one another and continue to meet the 1995 Commission's PRA Policy Statement, and
(5) allow for different reactor technologies in a manner that will promote stability and predictability in the long term.

The approach addresses risk-informed power reactor activities and the associated guidance documents. Risk-informed activities addressing non-power reactors, nuclear materials and waste are not addressed.

The NRC's proposed approach is to create an entire new Part in 10 CFR (referred to as "10 CFR Part 53") that can be applied to any reactor technology and that is an alternative to 10 CFR Part 50. Two major tasks are proposed: (1) develop the technical basis for rulemaking for 10 CFR Part 53, and (2) develop the regulations and associated guidance for 10 CFR Part 53.

Task 1: Development of Technical Basis
The objective of this task is to develop the technical basis for a risk-informed and performance-based 10 CFR Part 53. The technical basis provides the criteria and guidelines for development and implementation of the regulations to be included in Part 53. Current activities associated with developing the technical basis are described in SECY-05-0006 (ADAMS accession number ML043560093).

As the technical basis is developed and completed, it is anticipated that additional issues will be identified for which stakeholder input is desired. Therefore, it is envisioned that supplemental issues will be added to this ANPR over time.

At the end of the ANPR phase, the Commission will decide whether to proceed to formal rulemaking.

Task 2: Rule Development

The objective of this task is to develop and issue the actual regulations for 10 CFR Part 53. If upon completion of the technical basis the Commission directs the NRC staff to proceed to rulemaking, the NRC will follow its normal rule development process. The NRC staff will develop proposed rule text, interact with stakeholders in an appropriate forum (e.g. posting on web, public workshops), and provide a proposed rule package to the Commission for consideration.

In development of the rulemaking, the necessary guidance documents to meet the regulatiohs in 10 CFR Part 53 will also be developed.

Specific Considerations

Before determining whether to develop a proposed rule, the NRC is seeking comments on this matter from all interested persons. Specific areas on which the Commission is requesting comments are discussed in the following sections. Comments, accompanied by supporting reasons, are particularly requested on the questions contained in each section.

A. Plan

The NRC is seeking comments on the plan described above:

Question 1: *Is the proposed plan to make a risk-informed and performance-based revision to 10 CFR Part 50 reasonable? That is, is there a better approach than to create an entire new Part 53 to achieve a risk-informed and performance-based 10 CFR Part 50? If yes, what is a better and different way?*

Response 1: AREVA supports the development of a risk-informed and performance based revision to 10 CFR Part 50. AREVA also supports the NRC's continued development of a Technology-Neutral Framework (TNF) to guide the development of regulatory requirements for new reactors. However, AREVA also believes that it is premature to write a new rule such a new Part 53 until more experience is available in the licensing of new reactors, especially new non-LWRs. Rather than commencing with a new Part 53, it would be preferable to first gain experience with a design certification of new non-LWRs under Part 52 in which case the TNF could be used as guidance for deciding which parts of Part 50 to apply and which parts need exemptions.

For licensing new reactors, especially non-LWRs such as the AREVA-HTR it is better to license one or more reactors under the current regulations and under the guidance of the TNF before developing a new rule. It is premature to start to write technology neutral rules until the staff and industry has more experience in licensing new reactors, especially that with a new non-LWR.

Question 2: *Are the objectives, as articulated, understandable and achievable? If not, why not? Should there be additional objectives? If so, please describe the additional objectives and explain the reasons for including them.*

Response 2: The objectives are reasonable.

Question 3: *Would 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?*

Response 3: The approach would accomplish the objectives if Task 1 included the licensing of at least one new reactor that is not based on existing LWR technology because until then, the generic vs. reactor specific requirements cannot be effectively sorted out. LWR characteristics are so ingrained in the current regulations that it will take time and case experience to sort out what is generic to all reactors and what is technology specific. The TNF can play an important role in helping to sort out these issues by guiding the way.

Question 4: *Would existing licensees be interested in using risk-informed and performance-based alternative regulations to 10 CFR Part 50 as their licensing basis? If not why not? If so, please discuss the main reasons for doing so.*

Response 4: AREVA is interested in using a risk informed and performance based licensing approach for the AREVA-HTR. We would prefer to gain NRC acceptance of a licensing approach topical report that would use elements of the existing Part 50 requirements, elements of the NRC technology neutral framework (TNF), and elements of licensing approaches that have been previously proposed for modular HTGRs to establish the licensing requirements for the AREVA-HTR. Part of this approach would be the application for a design certification under Part 52 followed by a COL for a specific site. After experience is gained in licensing a non-LWR we will have sufficient technical basis to write a new Part 53 that is technology-neutral.

Question 5: *Should the alternative regulations be technology-neutral (i.e., applicable to all reactor technologies, e.g., light water reactor or gas cooled reactor), or be technology-specific? Please discuss the reason for your answer. If technology-specific, which technologies should receive priority for development of alternative regulations?*

Response 5: We agree that technology neutral regulations should be developed but not until there is experience in licensing some non-LWR designs. Rulemaking should be preceded by an appropriate sequencing of the four part plan outlined in SECY 05-0006, namely:

1) a technology-neutral framework
2) a set of (candidate) technology-neutral requirements
3) a technology-specific framework
4) technology-specific regulatory guides

The qualification in parentheses in Item 2 was added by AREVA. After experience is gained with testing all four parts on a specific non-LWR design, rulemaking for Part 53 should proceed.

Question 6: *When would alternative regulations and supporting documents need to be in place to be of most benefit? Is it premature to initiate rulemaking for non-LWR technologies? If so, when should such an effort be undertaken? Could supporting guidance be developed later than the alternative regulations, e.g. phased in during plant licensing and construction?*

Response 6: Please see our response to Question 5.

Question 7: *The NRC encourages active stakeholder participation through development of proposed supporting documents, standards, and guidance. In such process, the proposed documents, standards, and guidance would be submitted to and reviewed by NRC staff, and the NRC staff could endorse them, if appropriate. Is there any interest by stakeholders to develop proposed supporting documents, standards, or guidance? If so, please identify your organization and the specific documents, standards, or guidance you are interested in taking the lead to develop?*

Response 7: As part of AREVA-HTR design certification, we plan to participate but first the Technology Neutral "Framework" and "Requirements' must be developed by the NRC before supporting technology specific documents, standards, and guidance can be formulated. In any case AREVA is currently involved with ASME and ANS standards committees and we are positioned to support the NRC.

B. Integration of Safety, Security and Emergency Preparedness

The Commission believes that safety, security and emergency preparedness should be integrated in developing a risk-informed and performance-based set of requirements for nuclear power reactors (i.e., in this context, Part 53). The NRC has proposed to establish security performance standards for new reactors (see SECY-05-0120, ADAMS Accession Number ML051100233). Under the proposed approach, nuclear plant designers would analyze and establish, at an earlier stage of design, security design aspects such that there would be a more robust and effective security posture and less reliance on operational (extrinsic) security programs (guns, guards and gates). This approach takes advantage of making plants more secure by design rather than security components being added on after design.

As part of this approach, the NRC is seeking comment on the following issues:

Question 8: *In developing the requirements for this alternative regulatory framework, how should safety, security, and emergency preparedness be integrated? Does the overall approach described in the technology-neutral framework clearly express the appropriate integration of safety, security, and preparedness? If not, how could it better do so?*

Response 8: AREVA has concerns with the idea of integrating safety, security, and emergency preparedness. The concern is that exchange of information on the safety design philosophy will compromise the protection against physical security threats and vice versa. Strategies to protect against physical security threats should be discussed very carefully in the public domain so as not compromise the security that is inherent in a given plant design. Full integration would seem to inevitably lead to the need for limiting public access to information related to the public security protection strategy. AREVA will work with the NRC to find solutions to these problems.

Question 9: *What specific principles, concepts, features or performance standards for security would best achieve an integrated safety and security approach? How should they be expressed? How should they be measured?*

Response 9: Please see our response to Question 8. The Commission should define security requirements before design measures that will meet the requirement can be determined.

Question 10: *The NRC is considering rulemaking to require that safety and security be integrated so as to allow an easier and more thorough understanding of the effects that changes in one area would have on the other and to ensure that changes with unacceptable impacts are not implemented. How can the safety-security interface be better integrated in design and operational requirements?*

Response 10: Please see response to Question 8. If security is designed into the plant then the boundaries of safety – security measure are integrated. The NRC should define key top level security requirements that should be included in the design.

Question 11: *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 Probabilistic Risk Assessment (PRA) and how?*

Response 11: See response to Question 8. To date the technology of PRA has been limited to the identification and quantification of randomly occurring accident sequences. Although some elements of PRA have been used to evaluate the protection against deliberate acts of sabotage and terrorism, the task of quantifying the probability of such acts is well beyond the state of the art of PRA. Hence the extent to which security requirements can be risk informed is very limited in relation to requirements to protect against accidental releases of radioactive material. We also should not require commercial entities to protect themselves from extreme natural disasters or domestic / foreign acts of war which are after all the function, duty, and responsibility of any sovereign government. Based on this, security requirements should be developed on a deterministic basis.

Question 12: *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?*

Response 12: Yes – any available risk information should be used and incorporated into the emergency preparedness measures and features.

C. Level of Safety

The staff, in SECY-05-0130 (ADAMS Accession Number ML051670388), proposed options for specifying a minimum level of safety from the standpoint of risk which would implement the Commission's expectation of enhanced safety for new plants (as expressed in the Commission's policy statement for Regulation of Advanced Nuclear Power Plants). Four options were evaluated which included: (1) perform a case-by-case review, (2) use the Quantitative Health Objectives (QHOs) in the Commission's policy statement on "Safety Goals for the Operation of Nuclear Power Plants" (ADAMS Accession Number ML051580401), (3) develop other risk objectives, and (4) develop new QHOs. The NRC is soliciting stakeholder views on these options.

Subsidiary risk objectives could also be developed to implement the Commission's expectation regarding enhanced safety for new plants. Such subsidiary risk objectives could be a useful way to:

- Focus more on plant design,
- Provide quantitative criteria for accident prevention and mitigation, and
- Provide top level goals to assist in establishing system and hardware reliability and availability targets.

Currently, subsidiary risk objectives of 10^{-5}/plant year and 10^{-6}/plant year that could be applicable to all reactor designs are being considered for accident prevention and accident mitigation, respectively, where:

- Accident prevention refers to preventing major fuel damage, and
- Accident mitigation refers to preventing releases of radioactive material offsite such that no early fatalities occur (*i.e.*, from acute radiation doses).

Feedback is sought specifically on the following:

Question 13: *Which of the options in SECY-05-0130 with respect to level of safety should be pursued and why? Are there alternative options? If so, please discuss the alternative options and their benefits.*

Response 13: We agree with the staff recommendation for Option 2 (QHOs are the minimum level of safety), and disagree with the idea of attempting to come up with technology neutral subsidiary risk criteria. Subsidiary risk criteria similar to CDF and LERF for LWRs are inherently technology specific. We also agree with the idea of using Frequency-consequence curves such as those proposed in the TNF because they are technology neutral and address the full frequency-consequence spectrum of events rather than being limited to severe accidents.

In AREVA's response to SECY 05-0006 the same issue was addressed and the following response was given:

"AREVA finds the staff position (Option 2) on this issue to be reasonable."

Question 14: *Should the staff pursue developing subsidiary risk objectives? Why or why not? Are there other uses of subsidiary risk objectives that are not specified above? If so, what are they?*

Response 14: No, these are just CDF and LERF in disguise and only are meaningful to reactors with safety design characteristics similar to LWRs (e.g. use of active engineered safety features to protect the fuel integrity, risk dominated by accidents involving large releases from the fuel in severe core damage accidents, and safety dependent on the capability of a containment building to mitigate a severe accident in which large releases from the fuel occur). The subsidiary objectives defined by the staff are not meaningful for modular HTGRs such as the AREVA-HTR.

Question 15: *Are the subsidiary risk objectives specified above reasonable surrogates for the QHOs for all reactor designs?*

Response 15: No, see response to Question 13.

Question 16: *Should the latent fatality QHO be met by preventive measures alone without credit for mitigative measures, or is this too restrictive?*

Response 16: The question is difficult to answer because the classification of a measure as being preventitative or mitigative is not well defined and has not been defined in a technology neutral manner. The concept of whether a measure is classified as preventative or mitigative has been advanced by the NRC in the context of preventing or mitigating core damage in an LWR. How this concept is applied to reactors with fundamentally different safety characteristics is not clear. AREVA believes that both preventative and mitigative measures need to be taken into account in evaluating a reactor's capability to meet the QHOs.

A key open question is how one defines prevention and mitigation because a given SSC may either be viewed as prevention or mitigation depending on the event sequence. For example the ECCS in an LWR can be viewed as providing measures to mitigate the consequences of a loss of coolant accident for sequences with successful ECCS response, as well as for preventing a core damage event involving sequences in which the ECCS fails.

Question 17: *Are there other subsidiary risk objectives applicable to all reactor designs that should be considered? What are they and what would be their basis?*

Response 17: No, subsidiary risk objectives are inherently technology-specific.

Question 18: *Should a mitigation goal be associated with the early fatality QHO or should it be set without credit for preventive measures (i.e. assuming major fuel damage has occurred)?*

Response 18: No, see response to Questions 13 through 17.

Question 19: *Should other factors be considered in accident mitigation besides early fatalities, such as latent fatalities, late containment failure, land contamination, and property damage? If so, what should be the acceptance criteria and why?*

Response 19: No, see response to Questions 13 through 17.

Question 20: *Would a level 3 Probabilistic Risk Assessment (PRA) analysis (i.e. one that includes calculation of offsite health and economic effects) still be needed if subsidiary risk objectives can be developed? For a specific technology, can practical subsidiary risk objectives be developed without the insights provided by level 3 PRAs?*

Response 20: A PRA that includes mechanistic source terms and offsite dose calculations will be needed to demonstrate that QHOs and other criteria incorporated into a frequency – consequence curve will be met. The technical basis for using subsidiary metrics such as CDF and LERF for LWRs is the information in the body of work of Level 3 PRAs. Unless that body of work can be shown to be applicable to any new reactor, it will have to be replaced with Level 3 PRAs for the new reactor. Please see our response to questions on Integrated Risk below.

D. Integrated Risk

For new plant licensing, some licensees have indicated their interest in locating new plants at existing sites. In addition, potential applicants have indicated interest in locating multiple (or modular) reactor units at new sites and existing sites. The NRC is evaluating the issue of integrated risk. The staff, in SECY-05-0130, evaluated three options which included: (1) No consideration of integrated risk, (2) quantification of integrated risk at the site 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. Another aspect of this issue is the level of safety associated with the integrated risk. The NRC is presently considering whether the integrated risk should be restricted to the same level that would be applied to a single reactor. If this approach were adopted, for an entity who proposed to add multiple reactors to an existing site, the integrated risk would not be allowed to exceed the level of safety expressed by the QHOs in the Commission's Safety Goal Policy Statement.

The NRC is soliciting stakeholder views on these or other options.

Feedback is sought specifically on the following:

Question 21: *Which of the options in SECY-05-0130 with respect to integrated risk should be pursued and why? Are there alternative options? If so what are they?*

Response 21: Our position is the same as that stated for the same issue when it came up in SECY 05-0006, and that is Option 3, integrated risk from all the reactors on the site. Our response as stated then is copied below:

Response:

1. AREVA agrees that among the options considered by the staff Option 3 is the most appropriate policy for the consideration of integrated risk. AREVA offers the following additional comments regarding the discussion of the other two options because some conclusions that were reached do not seem to be well supported.

2. As noted in the above discussion, for any multi-reactor site, there is a potential for accidents involving two or more reactors on the same site as well as accidents on individual reactors. This discussion has touched on an aspect of the Commission's safety goal policy statement and QHOs that is subject to interpretation but is essential to clarify not only for the licensing of new modular reactors but also for existing multi-reactor sites. This is the question of whether the safety goals should be applied to the entire site, to each reactor independently or to each plant entity where the plant could be a modular reactor plant, a single existing reactor plant, or an existing multiple reactor unit reactor plant. We believe that the most reasonable approach from the public's perspective is that the safety goals should be applied to the entire site, whether sites with one or multiple existing reactor, with one or multiple new reactors, or any mixture.

3. The surrogate risk metrics of CDF and LERF and the justification for how these surrogate metrics are adequate for demonstrating conformance to the NRC safety goal QHOs appears to be only valid for single reactor sites. The existing arguments that the integrated risk from multi-unit sites would still be within the safety goals due to margins between the risks and the QHOs has only considered the combination of independent reactor accidents. The risk from multi-reactor accidents has not been addressed in the PRAs used to establish the validity of these surrogate risk metrics to QHO correlations. It is not clear that the staff's arguments for excluding integrated risk for current multi-reactor sites are valid because it appears that these arguments have only considered the combination of independent single reactor accidents. Because the supporting PRAs have almost exclusively considered only single

reactor accidents, the impact of multiple reactor accidents on these conclusions is unknown.

4. AREVA's position is that frequency consequence criteria of the type that are proposed by the staff for the technology neutral framework should be based on frequencies calculated on a per site year basis where a site may be comprised of several modular reactors. This approach is appropriate for any multiple reactor facility especially when there are shared systems and interdependencies that could influence a PRA that considers both independent and dependent multiple reactor accidents. Further, the PRA that is done to demonstrate compliance with these criteria should include both single reactor and multi-reactor accidents to the extent that such event sequences meet the PRA screening criterion. The frequency-consequence criteria need to be revised to take these issues into account. This also impacts the criteria for deciding whether an event sequence is classified as frequent, infrequent, or rare as proposed in the TNF.

Question 22: *Should the integrated risk from multiple reactors be considered? Why or Why not?*

Response 22: See our response to Question 18. Risk needs to be considered on a site year basis for that is the risk that individuals surrounding the site and referred to in the definition of the QHOs are exposed to.

Question 23: *If integrated risk should be considered, should the risk meet a minimum threshold specified in the regulations? Why or why not?*

Response 23: Integrated risk should be considered and the level of integrated risk should meet the QHOs for the entire site.

E. ACRS Views on Level of Safety and Integrated Risk

In a letter dated September 21, 2005, the Advisory Committee on Reactor Safeguards (ACRS) raised a number of questions related to new plant licensing. The ACRS discussed issues of requiring new plants to meet a minimum level of enhanced safety and how the risk from multiple reactors at a single site should be accounted for. The details of the ACRS discussion are in the September 21, 2005, letter which is attached to this ANPR. The Commission, in a September 14, 2005, SRM, directed the staff to consider ACRS comments in developing a subsequent notation vote paper addressing these policy issues.

Feedback is sought specifically on the following:

Questions 24: *Should the views raised in the ACRS letter and by various members of the Committee be factored into the resolution of the issues of level of safety and integrated risk? Why and why not?*

Response 24: In this letter [1] ACRS members expressed their views on several policy issues regarding new plant licensing including the minimum level of safety for new reactors and how to address the risk of multiple reactors on the same site. Their letter was in response to NRC SECY-05-130 [2]. AREVA has differing perspectives on the formulation of an appropriate set of risk metrics for new reactors, such as the modular AREVA-HTR, on multi-reactor sites as outlined below.

In addressing possible solutions to the policy issues in Reference [2], the ACRS proposed to employ the risk metrics of core damage frequency (CDF) and large release frequency (LRF), which have been derived from LWR PRAs, as fundamental measures of the safety of new reactors. Specifically, the letter states:

"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."

AREVA takes objection to this attempt to generalize these risk metrics which have been derived from LWR PRAs to other types of reactors. We disagree that CDF and LERF, as defined by the ACRS and the NRC staff, are fundamental metrics and that they can be meaningfully applied to reactors that employ a safety design approach that is fundamentally different than those of LWRs. The attempt to use the metrics of CDF and LERF for new reactors fails to assess risk in a technology-neutral way and is more likely to result in misleading conclusions than technically sound licensing decisions for new reactors.

The currently accepted definitions of CDF and LERF, such as those presented in the ASME PRA standard [3], are linked to the definitions of core damage and core damage with large early release that are used in LWR PRAs. These definitions refer to the intrinsic properties of LWRs such as oxidation temperature of Zircalloy in the presence of superheated steam, coolant levels in the reactor vessel, melting point of uranium dioxide fuel and other factors that are specific to LWRs and their behavior during a severe accident.

The logic for why CDF and LERF make sense as risk metrics for LWRs is linked to the safety design philosophy of LWRs that is well correlated to these metrics. In an LWR, the core damage frequency is highly correlated to the capability of active engineered safety features to protect the fuel in response to initiating event challenges and the LERF is highly correlated to the capability of the containment barrier to protect against the challenges posed by core damage events and containment bypasses. This in turn supports the use of such metrics in evaluating the defense in depth strategies of prevention and mitigation.

As the safety design philosophy of new reactors may employ a fundamentally different safety design approach such as the greater reliance on intrinsic reactor characteristics and passive safety features in protecting the fuel barrier integrity as well as different approaches to arranging and protecting the containment barriers, it is necessary to rethink this logic as it relates to risk metrics. For a reactor that is designed to protect the fuel barrier based on inherent and passive design features, calculating the frequency that the design capabilities of these passive features would be exceeded would be a logical way to define a risk metric. In fact this seems to be logic followed by the ACRS in proposing a definition of core damage for the PBMR. However, such calculations would invariably be reactor design-specific and not fundamental to all reactor designs.

If risk metrics are to be regarded as fundamental, they should be technology-neutral. The ACRS has not offered any technology-neutral definitions of CDF but rather has proposed a new definition for one new reactor, the PBMR. The PBMR is one of a class of modular HTGRs that includes the AREVA-HTR, all of which use ~1600°C temperature as a rule-of-thumb for the long-term, local peak fuel temperature during rare, off-normal events in which active cooling systems are unavailable indefinitely. However, it is not the fundamental safety acceptance criterion; rather the offsite dose corresponding to the time-integrated release from the plant is the fundamental criterion. For the PBMR and presumably for other modular HTGRs such as the

AREVA-HTR, the ACRS proposes to modify the definition of CDF to the frequency of exceeding 1600°C fuel temperatures. This proposed definition is not technology-neutral and does not account for the published body of work on HTGR PRAs which does not use such ill-defined risk metrics ([4], [5], [6], and [7]).

A more fundamental risk metric that would apply to any existing or new reactor is to compute the frequency of a given level of radiological consequence such as site boundary dose, number of early or latent health effects, or extent of offsite property damage, i.e. the parameters of a Level 3 PRA. For each specific reactor this would entail the development of a PRA based on the fundamental elements of a PRA and not by applying metrics derived from another reactor, such as CDF and LERF. These fundamental elements include the definition of reactor-specific safety functions and success criteria; reactor-specific event sequences exposing challenges to these safety functions and resulting in reactor-specific end states, reactor-specific and - mechanistic source terms and consequences, and detailed analysis of the uncertainties. This more fundamental PRA approach is consistent with that which is used in the NRC Technology Neutral Framework and the risk-informed licensing approaches that were applied by DOE to the MHTGR [8] [9] and by Exelon for the PBMR [10].

We agree with the ACRS position on how to address the risk of multiple reactor sites as articulated in the following statement:

"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."

While both the NRC staff and the ACRS acknowledge that a full description of the risk from a multi-reactor site should address not only the risk from each reactor due to accidents that impact the reactors independently, but also risks from accidents involving two or more reactor units. However, neither the staff nor the ACRS address how the risk metrics CDF and LERF can be manipulated to express this component of the risk. The reason for this omission, we submit, is that CDF and LERF and the numerical limits that have been attached to these metrics have only been applied to single reactor accidents. This limitation of the surrogate risk metrics simply provides another reason why these metrics are not only not fundamental, but not fully adequate to express the risk of currently licensed LWRs when arranged on multi-unit sites. A more complete discussion of our views on the integrated risk issue is found in Reference [11].

In summary, AREVA has significant concerns with the ideas advanced in the ACRS letter regarding the extrapolation of the LWR risk metric of CDF and LRF to new reactors. The path we propose as an alternative is to more carefully retrace the PRA approach that was followed for LWRs beginning with the Reactor Safety Study to produce a body of work based on the technology neutral parameters of a Level 3 PRA. That path has a much greater opportunity to yield the risk insights that will be meaningful in the formulation of requirements for new reactors such as the AREVA-HTR which has safety characteristics that are fundamentally different than LWRs.

References:

[1] Letter from Graham B. Wallis, Chairman ACRS to The Honorable Nils J. Diaz, Chairman U.S. Nuclear Regulatory Commission, " Report On Two Policy Issues Related To New Plant Licensing", September 21, 2005
[2] 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
[3] ASME, "Standard for Probabilistic Risk Assessment for Nuclear Power Plant Applications", ASME-RA-S-2002
[4] K.N. FLEMING, et al, , "HTGR Accident Initiation and Progression Analysis Status Report, General Atomics Report No. GA-A15000, April 1978
[5] U.S. Department of Energy, DOE-HTGR-86011, "Probabilistic Risk Assessment for the Standard Modular High Temperature Gas-Cooled Reactor, Revision 5, April 1988
[6] Henriette van Graan, et al, "Use of PRA for the PBMR Demonstration Power Plant in South Africa", Proceedings of PSA '05, San Francisco CA September 2005
[7] Karl N. Fleming, "Challenges and Opportunities in the Performance of PRAs on New Reactors", Proceedings of PSA '05, San Francisco CA September 2005
[8] Department of Energy, "Preliminary Safety Information Document for the Standard MHTGR,", DOE-HTGR-86-024, September 1988
[9] U.S. NRC, "Draft Pre-application Safety Evaluation Report for the Modular High Temperature Gas-Cooled Reactor", NUREG-1338, March 1989
[10] Exelon Generation Company, "Proposed Licensing Approach for the Pebble Bed Modular Reactor in the United States", Available in NRC Public Reading Room, January 31, 2002.
[11] Fleming, K.N., "On The Issue of Integrated Risk – A PRA Practitioners Perspective", Proceedings of PSA '05, San Francisco CA September 2005

F. Containment Functional Performance Standards

The Commission has directed the staff to develop options for containment functional performance requirements and criteria which take into account such features as core, fuel, and cooling system design. In developing these options, the NRC is seeking stakeholder views on the following aspects:

Question 25: *How should containment be defined and what are its safety functions? Are the safety functions different for different designs? If so, how?*

Response 25: Previous staff papers on the topic of containment performance requirements such as SECY 05-0006 have included two distinct meanings of the term containment, one that is associated with a safety function and another that is associated with a building or structure. In answering this question it is important to distinguish the concept of a containment function and that of a structure or building.

One can define high level safety functions that are applicable to all reactors. One such high level safety function is the containment of radioactive material. Other generic high level safety functions would include control of core heat generation and control of core heat removal. Given a reactor specific safety design approach, these high level safety functions can be developed into reactor specific safety functions so that the roles of specific SSCs in the performance of safety functions can be clearly defined. So for example the AREVA-HTR does not have the "coolant inventory control" safety function that an LWR has, but it has a "control chemical attack" safety function to facilitate the protection of the core and graphite against air or water ingress challenges.

The method of deriving a set of reactor specific safety functions is not unique and is part of the safety design philosophy selected by the designer and approved by the regulator. Safety functions include required safety functions that are necessary to fulfill in order to meet minimum safety requirements, and supportive safety functions that are added as an element of defense-in-depth and to meet user requirements for reliability and investment protection.

The required reactor specific safety functions for AREVA-HTR, which are sufficient to ensure that radioactive material is adequately contained, include the following listed with appropriate hierarchical relationships:

- Retain Radionuclides within fuel particles (passive)
- Control heat generation
  - Control via reactivity feedback (passive)
  - Control via control rods (active)
    - Only needed following a long delay after the onset of an accident
  - Control core geometry (passive)
- Control core heat removal
  - Transfer heat from core to vessel (passive)
  - Transfer heat from vessel to environment (passive)
  - Control core geometry (passive)
- Control chemical attack
  - Maintain HPB integrity (passive)

The containment of radioactive material in the AREVA-HTR is accomplished using the "mechanistic containment system" concept that includes the following elements:

- highly reliable and robust fuel particle coatings
- demonstrated acceptable fuel performance during normal operation and accidents
- retention of the vast proportion of fission products within the fuel
- provisions for a highly reliable Helium Pressure Boundary
  - whose performance is not dependent on the performance of the fuel,
  - which retains fission products which may be released from the fuel
  - is designed to prevent excessive air ingress
- provisions for a reactor building
  - whose performance is not dependent on the performance of the fuel or HPB,
  - provides a concentric transport barrier to fission products released from HPB
  - prevents excessive air ingress, and
  - Most importantly maintains core geometry for heat removal and control of heat generation

Hence the reactor building has the following required and supportive safety functions for the AREVA-HTR, which differ from those of a containment building in a typical LWR:

- Required safety functions – necessary to meet minimum regulatory requirements
    - Structural support for reactor vessel, reactor cavity cooling system and major helium pressure boundary components; for maintenance of core geometry and passive heat removal
    - Structural protection of reactor vessel, helium pressure boundary, and all safety-related SSCs from loads from internal and external hazards during design basis accidents
    - Provide shielding for workers and prevent direct shine offsite doses from exceeding regulatory requirements
- Supportive safety function – provides margin for offsite requirements and element of defense-in-depth
    - Limit air ingress to control chemical attack
    - Protection of all SSCs from loads for internal and external hazards
    - Provide additional retention thru deposition and other natural phenomena released for any fission products released from helium pressure boundary

Question 26: *Should the containment functional performance standards be design and technology specific? Why or why not?*

Response 26: As discussed more fully in response to Question 25 above the functional requirement to contain radioactive material is generic to all reactors, the manner in which various SSCs are deployed to fulfill required and supportive safety functions is reactor specific.

Question 27: *What approach should be taken to develop technology-neutral containment performance standards that would be applicable to all reactor designs and technologies? Should containment performance be defined in terms of the integrated performance capability of all mechanistic barriers to radiological release or in terms of the performance capability of a means of limiting or controlling radiological releases separate from the fuel and reactor pressure boundary barriers?*

Response 27: Containment performance should be defined in terms of the integrated performance capability of all mechanistic barriers to radiological release. Since such mechanistic barriers are invariably reactor specific, the design criteria also need to be reactor specific. In attempting to develop technology neutral containment performance standards, the reactor specific safety design philosophy needs to be taken into account. This influences the way in which safety functions are defined and that makes the safety functions reactor specific. The way in which SSCs are deployed to fulfill safety functions is also reactor specific. In the LWR safety design philosophy the performance capabilities of the containment building have been developed, in part, to protect the public from core damage scenarios in which there are large releases from the fuel and reactor coolant system pressure boundary barriers. In the AREVA-HTR safety design philosophy, the reactor building must fulfill important structural integrity safety functions to protect the integrity of the fuel barrier and the reactor coolant pressure boundary barrier and the capability to control core heat generation, heat removal and chemical attack via passive means. Hence, both LWRs and modular HTGRs must address reactor specific safety functions and the reactor specific disposition of how events may challenge the capabilities of SSCs to support the safety functions.

Question 28: *What plant physical security functions should be associated with containment and what should be the related functional performance standards?*

Response 28: Physical security requirements need to consider all the required reactor specific safety functions that must be fulfilled to protect the public. These requirements should include physical barriers to protect all the SSCs that support these required safety functions. Plant security must be developed in an integrated fashion that would include operational security measures and design security measure that keeps the plant safety systems protected against external and internal aggressions and malfeasant acts.

Question 29: *How should the PRA information and insights be combined with traditional deterministic approaches and defense-in-depth in establishing the proposed containment functional performance requirements and criteria for controlling radiological releases?*

Response 29: Please see response to Question 25 on the approach to formulating the functional performance requirements. One element of demonstrating that such requirements are adequately met is to demonstrate that the frequencies and consequences of licensing basis events fall within the frequency-consequence criteria for reviewing the PRA results. Another element is to ensure that the offsite boundary dose requirements for the deterministic safety analysis of design basis events are satisfied. Evaluation of defense-in-deth should include a systematic review of the roles that SSCs play in the prevention and mitigation of accidents. AREVA plans to use the approach defined in a recent paper for this purpose [Reference: Fleming, Karl N. and Fred A. Silady, " A Risk Informed Framework for Defense-in-Depth for Advanced and Existing Reactors", Reliability Engineering and System Safety, Elsevier Publishing Company, 78 (2002) pp. 205–225]

Question 30: *How should the rare events in the range of 10^{-4} to 10^{-7} per year be considered in developing the containment functional performance requirements and criteria? Should events less than 10^{-7} per year in frequency be considered in developing the containment functional performance requirements and criteria?*

Response 30: Events in all frequency ranges should be considered in the formation of requirements for all plant SSCs in an integrated manner. Specific functions of specific SSCs that are necessary to contain radiological material and to perform other required safety functions need to be considered. When it can be demonstrated that the frequency of an event or event sequence is less than some "de minimus" value, such as 5×10^{-7} per plant year, such events should not be considered.

G. Technology-Neutral Framework

In support of determining the requirements for these alternative regulations, the NRC is developing a technology-neutral framework. This framework provides one approach in the form of criteria and guidelines that could serve as the technical basis for 10 CFR Part 53 that is technology-neutral, risk-informed, and performance-based. A working draft of this framework was issued for public review and comment in SECY–05–0006, dated January 7, 2005 (ML043560093). The latest working draft of the framework document is on the Ruleforum website. An updated version with additional information will be placed on the Ruleforum website in July 2006. The framework provides the criteria and guidelines for the following:

- Safety, security, and emergency preparedness expectations.
- Defense-in-depth and treatment of uncertainties.
- Licensing basis events (LBEs) identification and selection.
- Safety classification of structures, systems, and components.
- PRA technical acceptability.

The NRC is seeking stakeholder views of the following aspects:

Question 31: *Is the overall top-down organization of the framework, as illustrated in Figure 2–6 a suitable approach to organize the approach for licensing new reactors? Does it meet the objectives and principles of Chapter 1? Can you describe a better way to organize a new licensing process?*

Response 31: Figure 2-6 describes a reasonable approach for organizing the framework.

Question 32: *Do you agree that the framework should now be applied to a specific reactor design? If not, why not? Which reactor design concept would you recommend?*

Response 32: We agree that in order to demonstrate the capability of the approach to fulfill its objectives, the approach should be applied to a specific reactor. Ideally the reactor should be one that uses a safety design approach that is different than that of an LWR should be selected. A good example would be one of the available modular HTGR designs such as the PBMR. Selecting an existing LWR or ALWR will not be capable of a convincing demonstration.

Question 33: *The unified safety concept used in the framework is meant to derive regulations from the Safety Goals and other safety principles (e.g., defense-in-depth). Does this approach result in the proper integration of reactor regulations and staff processes and programs such that regulatory coherence is achieved? If not, why not?*

Response 33: The unified safety concept used in the framework has the potential of developing technology neutral regulations that can deliver a uniform level of safety. The one area of the framework that is very difficult to apply in a technology neutral and balanced way and needs to be developed further and that is selection of the deterministic LBE for assessing the challenge to final radionuclide barrier.

Question 34: *The framework is proposing an approach for the technical basis for an alternative risk-informed and performance-based 10 CFR Part 50. The scope of 10 CFR Part 50 includes sources of radioactive material from reactor and spent fuel pool operations.*

Similarly, the framework is intended to apply to this same scope. Is it clear that the framework is intended to apply to all of these sources? If not, how should the framework be revised to make this intention clear?

The Commission believes that safety, security, and emergency preparedness should be integrated. The approach in the framework to achieve this integration is to define the safety, security, and preparedness expectations that are needed and to define protective strategies and defense-in-depth principles for each area in an integrated manner.

Response 34: It is clear that the TNF applies to all sources of radioactive material that need to be included in the design certification or license. The approach to integrating the safety, security, and preparedness strategies and defense-in-depth principles is reasonable.

Question 35: *What role should the following factors play in integrating emergency preparedness requirements (as contained in 10 CFR 50.47) in the overall framework for future plants:*

- *The range of accidents that should be considered?*
- *The extent of defense-in-depth?*
- *Operating experience?*
- *Federal, state, and local authority input and acceptance?*
- *Public acceptance?*
- *Security-related events?*

Response 35: 1) Both infrequent and rare events as defined in the TNF should be considered in formulating emergency preparedness requirements. Frequent events need not be included because keeping the doses within the frequency consequence curve for these events would preclude the need for any offsite protective actions. 2) Emergency preparedness is recognized as an important element of defense-in-depth. Safety margins associated with emergency preparedness can be demonstrated by performing offsite dose calculations with and without credit for emergency planning protective actions. 3) Although operating experience with events and emergency plan drills may yield insights about the effectiveness of certain aspects of emergency planning, this is expected to be of limited value in demonstrating that emergency preparedness requirements are met. 4) Considerations of input and acceptance of federal, state, and local authorities should be no different than that reflected in 10 CFR 50.47. 5) public acceptance considerations should be adequately addressed through the intervener participation in the design certification and licensing process. 6) See earlier input on security related events.

Question 36: *What should the emergency preparedness requirements for future plants be? Should they be technology specific or generic regardless of the reactor type?*

Response 36: It is reasonable to have a high level requirement that future plants shall have an acceptable approach to emergency preparedness that is sufficient to meet the frequency consequence criteria for licensing basis events and addresses factors such as defense-in-depth and other factors such as those listed in Question 35. However the detailed requirements such as those for having provisions for large scale evacuations and emergency drills and emergency planning zones should be developed on a technology specific basis. Factors such as the timing of accident progression, capability to provide for ad hoc emergency measures, nature and magnitude of the mechanistic source terms, existence of emergency plans for co-located reactors, and other technology specific factors need to be taken into account in deriving specific emergency preparedness requirements.

The core of the NRC's safety philosophy has always been the concept of defense-in-depth, and defense-in-depth remains basic to the safety, security, and preparedness expectations of the technology-neutral framework.

Defense-in-depth is the mechanism used to compensate for uncertainty. This includes uncertainty in the type and magnitude of challenges to safety, as well as in the measures taken to assure safety.

Question 37: *Is the approach used in the framework for how defense-in-depth treats uncertainties well described and reasonable? If not, how should it be improved?*

Response 37: AREVA supports the philosophy of defense-in-depth and has adopted an approach to defense-in-depth that was presented at several of the NRC workshops and that is being used to support the design of the AREVA-HTR. Our approach defines defense-in-depth with respect to design aspects, process aspects, and scenario aspects as suggested in the paper by Fleming and Silady [Reference: Fleming, Karl N. and Fred A. Silady, " A Risk Informed Framework for Defense-in-Depth for Advanced and Existing Reactors", Reliability Engineering and System Safety, Elsevier Publishing Company, 78 (2002) pp. 205–225]. This approach includes an objective means of defining what is meant by prevention and mitigation and a method for evaluating the roles that specific SSCs play in the prevention and mitigation of accidents. This approach is not tied to reactor specific risk metrics such as CDF and LERF, in contrast with other approaches that have been proposed.

The nature of defense-in-depth according to all published definitions including the discussion on the topic in the draft NUREG for the TNF is that it does not lend itself to predictable and reproducible criteria for establishing whether a given design has an adequate degree of defense-in-depth. Part of the reason for this is the inherent difficulty in deriving criteria for comparing the degrees of defense-in-depth in reactor designs with fundamentally different inherent characteristics.

The presentation of defense-in-depth in the TNF is to be complemented for its excellent discussion of how it relates to uncertainty in the performance of a PRA. The strategy of using defense-in-depth to address uncertainty is well described, and the logic decision chart adds clarity to the process of how to apply it. The staff has improved the discussion of defense-in-depth, but not to the point where the issues identified above are resolved. The current version of the TNF is regarded as a major improvement over previously released versions of the framework in relation to how the defense-in-depth issue is addressed. It is clear that the NRC staff has made a serious effort to address concerns raised about this treatment in some of the previous workshops held on the TNF.

Two important aspects of the TNF that are impacted by the treatment of defense-in-depth are the derivation of deterministic technical requirements that are intended to address uncertainties in the capability to perform the protective strategies of physical protection, stable operation, protective systems, barrier integrity, protective actions, and the derivation of a deterministic LBE that addresses the limiting challenge to the final radiological barrier to address uncertainties that are not fully known. It is not clear what criteria will be applied to ensure that each reactor that is licensed according to this TNF will end up with a consistent and reproducible set of technical requirements. We are also concerned that the process for defining the deterministic LBE for evaluating the capabilities of the final barrier be tied to an objective review of the event sequences in the PRA and that the process is not arbitrarily used to produce a predetermined end result.

Although it is recognized that much of the motivation for defense-in-depth is to address uncertainties, it is AREVA's position that there are other motivations for having multiple lines of defense that are within the knowledge base of safety design philosophy. In other words, even without the uncertainties we would still have reasons to apply the basic principles associated with defense-in-depth. In some cases multiple barriers and redundancy and diversity are required just to develop the necessary reliability and capability of safety systems without regard to uncertainty in this performance. A final problematic aspect of the TNF approach to defense-in-depth is the lack of clear criteria to establish how a given requirement or design feature that is provided to address defense-in-depth actually protects against the uncertainty. It is important that the uncertainties are well characterized before decisions are made to add or modify a requirement or design feature in order to address defense-in-depth concerns.

Question 38: *Are the defense-in-depth principles discussed in the framework clearly stated? If not, how could they be better stated? Are additional principles needed? If so, what would they be? Are one or more of the stated principles unnecessary? If so, which principles are unnecessary and why are they unnecessary?*

Response 38: The principles are clearly stated. However, one area that could be improved is the development of a clear definition of what is meant be prevention and mitigation. Because a given SSC may provide prevention roles along some sequences and mitigation roles along others, it is not clear how these defense-in-depth strategies are evaluated under the TNF. The Fleming-Silady paper referred to earlier provides such a definition and an approach for quantifying the impacts of SSC prevention and mitigation on risk. Without such definitions and approach, judgments that are made as to how well the prevention and mitigation are balanced are difficult to predict and reproduce among different reviewers and analysts.

Question 39: *The framework emphasizes that sufficient margins are an essential part of defense-in-depth measures. The framework also provides some quantitative margin guidance with respect to LBEs in Chapter 6. Should the framework provide more quantitative guidance on margins in general in a technology-neutral way? What would be the nature of this guidance?*

Response 39: AREVA agrees that safety margins are an element of defense-in-depth and finds that the approach to margins in Chapter 6 is reasonable. A more quantitative approach to margins should only be applied to the technology specific aspects of the framework.

Question 40: *The framework stresses that all of the Protective Strategies must be included in the design of a new reactor but it does not discuss the relative emphasis placed on each strategy compared to the others. Are there any conditions under which any of these protective strategies would not be necessary? Should the framework contain guidelines as to the relative importance of each strategy to the whole defense-in-depth application?*

Response 40: AREVA agrees that all the protective strategies must be provided in some way. The fact that the relative emphasis is left open should provide each specific reactor designer with sufficient flexibility to implement a given safety design approach.

Question 41: *Are the protective strategies well enough defined in terms of the challenges they defend against? If not, why not? Are there challenges not protected by these five protective strategies? If so, what would they be?*

Response 41: AREVA is not aware of any challenges that the protective strategies do not defend against.

In the framework, risk information is used in two basic parts of the licensing process: (1) Identification and selection of those events that are used in the design to establish the licensing basis, and (2) the safety classification of selected systems, structures, and components.

Question 42: *Is the approach to and the basis for the selection LBEs reasonable? If not, why not? Is the cut-off for the rare event frequency at 1E–7 per year acceptable? If not, why not? Should the cut-off be extended to a lower frequency?*

Response 42: The approach to and basis for selecting the LBEs is reasonable and is similar to the approach that AREVA will propose for the AREVA-HTR.

Question 43: *Is the approach used to select and to safety classify structures, systems, and components reasonable? If not, what would be a better approach?*

Response 43: AREVA finds the approach to defining safety significant SSCs to be reasonable and this approach is similar to that which AREVA intends to apply when it applies for design certification for the AREVA-HTR. The AREVA approach draws from the approaches developed previously for the MHTGR [Reference U.S. Department of Energy, 'Preliminary Safety Information Document for the Standard MHTGR', DOE-HTGR-86-024, September 1988.], by Exelon for the PBMR [Reference Exelon Generation Company Letter, Subject: Proposed Licensing Approach for the Pebble Bed Modular Reactor in the United States, January 31, 2002.] , and as described more recently in a paper by Silady [Reference: Fred A. Silady, "Risk Informed Licensing Approach for Modular HTGRs", Proceedings of PSA '05, San Francisco, Sep 2005].

Question 44: *Is the approach and basis to the construction of the proposed frequency consequence (F–C) curve reasonable? If not, why not?*

Response 44: The approach could be improved by changing the frequency basis from events per reactor year to events per plant year, which would facilitate the treatment of modular reactor plants. The approach in the TNF makes it difficult to distinguish between event sequences that impact a single module vs. sequences that may involve two or more reactor modules, for example.

Another practical issue is that the frequency-consequence curve has too many steps to be practical. The frequency of a given LBE may have an uncertainty distribution that spans several orders of magnitude, in which case several different dose criteria could be applied depending on whether the mean or upper percentile is used.

The cutoff of $1x10^{-7}$ per (plant) year is reasonable and appears to be generally consistent with current PRA standards and practices. While a case could be made to increase it to, say, $5x10^{-7}$ per year, since sequences at this frequency would not exceed the safety goal QHOs for early health effects, it would be unreasonable to lower at as concerns about completeness and uncertainties in the PRA results would be dominant at such lower frequencies.

Question 45: *Are the deterministic criteria proposed for the LBEs in the various frequency categories reasonable from the standpoint of assuring an adequate safety margin? In particular, are the deterministic dose criteria for the LBEs in the infrequent and rare categories reasonable? If not, why not?*

Response 45: The deterministic criteria for the LBEs appear to be reasonable.

Question 46: *Is it reasonable to use a 95% confidence value for the mechanistic source term for both the PRA sequences and the sequences designated as LBEs to provide margin for uncertainty? If not, why not? Is it reasonable to use a conservative approach for dispersion to calculate doses? If not, why not?*

Response 46: It is reasonable to use a 95%confidence value for the mechanistic source term for the PRA sequences and the LBE sequences. The conservative approach to calculating doses is reasonable; however the applicant should be given the option of using an alternative approach if it can be justified.

The approach proposed in the framework requires a full-scope "living" PRA that would incorporate operating experience and performance-based requirements in the periodic reexamination of events designated as LBEs that were originally selected based on the design, and structures, systems, and components that were characterized as safety-significant.

Question 47: *The approach proposed in the framework does not predefine a set of LBEs to be addressed in the design. The LBEs are plant specific and identified and selected from the risk-significant events based on the plant-specific PRA. Because the plant design and operation may change over time, the risk significant events may change over time. The licensee would be required to periodically reassess the risk of the plant and, as a result, the LBEs may change. This reassessment could be performed under a process similar to the process under 10 CFR 50.59. Is this approach reasonable? If not, why not?*

Response 47: It is agreed that the requirement for a living PRA is reasonable in light of the important inputs that PRA will have to the licensing basis. When PRA updates are done, the frequencies and consequences of LBEs can be reasonably be expected to change. Under rare circumstances, the definition of the LBEs may also change, i.e. new LBEs may appear and others disappear. It is reasonable to require that all decisions derived from the LBE frequencies and consequence that have changed be reviewed to ensure that the basis for the decisions is still valid. It is important that the deterministic elements of the approach be applied in a manner that the licensing basis is not sensitive to the kind of PRA update changes that can be expected. In applying the process of 10 CFR 50.69 it is noted that the nature of the PRA change evaluation process might be different than simply calculating changes in CDF and LERF as these risk metrics may not be used for a given new reactor. If that is accounted for by the process under 50.59, then AREVA agrees that this approach is reasonable.

Question 48: *The framework provides guidance for a technically acceptable full-scope PRA. Is the scope and level of detail reasonable? If not, why not? Should it be expanded and if so, in what way?*

Response 48: The guidance for preparing technically acceptable full-scope PRAs presented in the TNF is reasonable but is only developed to define high level requirements, which are reasonable. A major concern is that the available consensus standards from which to draw supporting technical requirements are LWR specific and are highly focused on operating plants for which the as-built and as-operated design and operational characteristics are well known. Major needs to be filled to implement this framework are technology neutral requirements for PRAs on new reactors done at the design stage and technology specific requirements that are needed for specific reactor designs.

Question 49: *Because a PRA (including the supporting analyses) will be used in the licensing process, should it be subject to a 10 CFR Part 50 Appendix B approach to quality assurance? If not, why not?*

Response 49: The approach to PRA quality assurance that is described in the TNF appears to be reasonable. It is not clear that this approach is the same as 10 CFR 50 Appendix B, which was not written for PRA.

Chapter 8 describes and applies a process to identify the topics which the requirements must address to ensure the success of the protective strategies and administrative controls. This process is based upon:

• Developing and applying a logic diagram for each protective strategy to identify the pathways that can lead to failure of the strategy and then, through a series of questions, identify what needs to be done to prevent the failure;

• Applying the defense-in-depth principles from Chapter 4 to each protective strategy;

• Developing and applying a logic diagram to identify the needed administrative controls; and

• Providing guidance on how to write the requirements.

Question 50: *Is this process clear, understandable, and adequate? If not, why not? What should be done differently?*

Response 50: The approach is logical and understandable and well structured. The tie in to the protective strategies and the defense-in-depth approach is considered an improvement over what might be described as the ad hoc approach that was followed to develop the current requirements in Part 50. Before this process is codified in the regulations it is important that it be tested through a case study on an actual non-LWR design.

Question 51: *Is the use of logic diagrams to identify the topics that need to be addressed in the requirements reasonable? If not, what should be used?*

Response 51: The use of logic diagrams is reasonable.

Question 52: *Is the list of topics identified for the requirements adequate? Is the list complete? If not, what should be changed (added, deleted, modified) and why?*

Response 52: The list of topics appears to be complete but this question is better answered by application of the approach with specific examples on an actual non-LWR design.

Question 53: *A completeness check was made on the topics for which requirements need to be developed for the new 10 CFR Part 53 (identified in Chapter 8) by comparing them to 10 CFR Part 50, NEI 02–02, and the International Atomic Energy Agency (IAEA) safety standards for design and operation. Are there other completeness checks that should be made? If so, what should they be?*

Response 53: The process that was used for completeness checks is reasonable. However, the credibility of approach with respect to completeness would be enhanced if an independent group of analysts and licensing experts were to attempt to use the process to come up their own lists. The reproducibility of the process that was used to create this list is not clear.

Question 54: *The results of the completeness check comparison are provided in Appendix G. The comparison identified a number of areas that are not addressed by the topics but that are covered in the IAEA standards. Should these areas be included in the framework? If so, why should they be included? If not, why not?*

Response 54: These elements should not necessarily be included but a justification for excluding them should be developed and documented; otherwise the value of doing the comparison to check for completeness is reduced.

H. Defense-in-Depth

In SECY-03-0047 (ML030160002), the staff recommended that the Commission approve the development of a policy statement or description (e.g., white paper) on defense-in-depth for nuclear power plants to describe: the objectives of defense-in-depth (philosophy); the scope of defense-in-depth (design, operation, etc.); and the elements of defense-in-depth (high level principles and guidelines). The policy statement or description would be technology neutral and risk-informed and would be useful in providing consistency in other regulatory programs (e.g., Regulatory Analysis Guidelines). In the SRM to SECY-03-0047, the Commission directed the staff to consider whether it can accomplish the same goals in a more efficient and effective manner by updating the Commission Policy Statement on Use of Probabilistic Risk Assessment Methods in Nuclear Regulatory Activities to include a more explicit discussion of defense-in-depth, risk-informed regulation, and performance-based regulation.

The NRC is interested in stakeholder comment on a policy statement on defense-in-depth.

Question 55: *Would development of a better description of defense-in-depth be of any benefit to current operating plants, near term designs, or future designs? Why or why not? If so, please discuss any specific benefit.*

Response 55: A policy statement that defines Defense-in-depth in a technology neutral way would be of benefit to future designs to help determine whether the new designs have provided an adequate approach to defense-in-depth that is comparable to that applied in the existing plants.

Question 56: *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?*

Response 56: The DiD policy statement must be holistic to be effective and useful to the future designs. It must recognize all DiD measure and not rely only on Barrier measures, i.e. scenario, programmatic, and design choices. It must recognize the roles of inherent and passive approaches to fulfilling safety functions as well as the traditional use of redundant and diverse active systems. The time element is another feature of DiD that needs to be recognized. The policy statement would benefit the future designs as it provides a clear and holistic view of defense-in-depth. Rather than incorporating this into a revised PRA policy statement, a separate policy is needed because defense in depth is much broader than PRA, although we agree that PRA tools can be used to evaluate a design against defense-in-depth requirements.

Question 57: *RG 1.174 assumes that adequate defense-in-depth exists and provides guidance for ensuring it is not significantly degraded by a change to the licensing basis. Should RG 1.174 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?*

Response 57: It does not make sense to modify RG 1.174 to address this issue for new plants because RG 1.174 is framed to make decisions incrementally on the basis of an existing established set of deterministic requirements and because it focuses the question of risk on LWR specific risk metrics such as CDF and LERF. However, given the improvements in defining what is meant by defense-in-depth for new plants, there would be merit in considering whether RG.1.174 should be modified for existing plants.

Question 58: *How should defense-in-depth be addressed for new plants?*

Response 58: AREVA's response to this question was provided in response to Question 37.

Question 59: *Should development of a better description of defense-in-depth (whether as a new policy statement, a revision to the PRA policy statement, or as an update to RG 1.174) be completed on the same schedule as 10 CFR Part 53? Why or why not?*

Response 59: As noted earlier, AREVA's position is that 10 CFR 53 should be timed after a pilot study in which the TNF is tested on an actual non-LWR design certification application. Given that a new defense-in-depth policy statement should be done on an earlier schedule as this will be needed to complete the pilot study.

I. Single Failure Criterion

In SECY-05-0138 (ML051950619), the staff forwarded to the Commission a draft report entitled "Technical Report to Support Evaluation of a Broader Change to the Single Failure Criterion" and recommended to the Commission that any followup activities to risk-inform the Single Failure Criterion (SFC) should be included in the activities to risk-inform the requirements of 10 CFR Part 50. The Commission directed the staff to seek additional stakeholder involvement. The report provides alternatives to the SFC: (1) maintain the SFC as is, (2) risk-inform the SFC for design bases analyses, (3) risk-inform SFC based on safety significance, and (4) replace SFC with risk and safety function reliability guidelines.

The NRC is soliciting stakeholder feedback with regard to the proposed alternatives.

Question 60: *Are the proposed alternatives reasonable? If not, why not?*

Response 60: The proposed alternatives (Alternatives 1, 2, and 3 defined in SECY 05-0138 are reasonable, however Alternatives 2 and 3 are based on an LWR specific approach to safety significance classification as this approach uses CDF and LERF based risk importance metrics. AREVA is in favor of Alternative 4 for new reactors as in this alternative the SFC is effectively eliminated and replaced by a more general approach in which the frequency and consequences of each licensing basis event are taken into account and there are no arbitrary redundancy requirements.

Question 61: *Are there other options for risk-informing the SFC? If so, please discuss these options.*

Response 61: Under the TNF or something comparable, there is no need for an arbitrary redundancy requirement which is the design implication of the SFC. If this is what is meant by risk informing the SFC, then the TNF represents the best option.

Question 62: *Which option, if any, should be considered?*

Response 62: AREVA will propose a risk informed licensing approach similar to that proposed by Exelon for the PBMR in which there is no SFC and the requirements to achieve the necessary reliability of a safety related SSC are developed on a case by case basis. These requirements may include specific requirements for redundancy for specific SSCs but no arbitrary redundancy requirements just because the SSC is safety related. The approach proposed by AREVA is many points in common with the TNF.

Question 63: *Should changes to SFC in 10 CFR Part 50 be pursued separate from or as part of the effort to create a new 10 CFR Part 53? Why or why not?*

Response 63: Risk informed changes to the SFC will need to be made for the design certification of the AREVA-HTR.

J. Continue Individual Rulemakings to Risk-Inform 10 CFR Part 50

The NRC has for some time been revising certain provisions of 10 CFR Part 50 to make them more risk informed and performance-based. Examples are: (1) A revision to 10 CFR 50.65, "Requirements for Monitoring the Effectiveness of Maintenance at Nuclear Power Plants;" (2) a revision of 10 CFR 50.48 to allow licensees to voluntarily adopt National Fire Protection Association (NFPA) Standard 805, "Performance-Based Standard for Fire Protection for Light Water Reactor Electric Generating Plants, 2001 Edition," (NFPA 805); and (3) issuance of 10 CFR 50.69, "Risk-Informed Categorization and Treatment of Structures, Systems, and Components for Nuclear Power Reactors," as a voluntary alternative set of requirements. These actions have been effective but required extensive NRC and industry efforts to develop and implement.

The NRC plans to continue the current risk-informed rulemaking actions, *e.g.*, 10 CFR 50.61 on pressurized thermal shock and 10 CFR 50.46 on redefinition of the emergency core cooling system break size, that are ongoing, and would undertake new risk informed rulemaking only on an as needed basis.

The NRC is seeking comment on the following issues:

Question 64: *Should the NRC only continue with the ongoing current rulemaking efforts and not undertake any effort to risk-inform other regulations in 10 CFR Part 50, or should the NRC undertake new risk-informed rulemaking on a case-by-case priority basis? Why?*

Response 64: With one exception, discussed below, there are no obvious additional candidates among current regulations for risk-informed modifications. Most of the "easy" ones have been done, and it is not clear that any significant benefit would be gained in moving further into Part 50 (technical requirements, which still apply to new reactors). Experience has shown that determining how to modify these regulations takes a long time and a lot of effort, which may not be cost-effective.

The one rule that AREVA has identified relates to the use of digital instrumentation and control systems in both existing and new reactors. 10 CFR 50.55a(h) is out of date and as constructed, it will always be out of date with rapidly developing technology. It references IEEE 603-1991. Advances in guidance most relevant to digital I&C are in later versions IEEE 603; however, the compliance is link back to 1991. As a contrast, the ASME code rule has a mechanism to move to later versions as they are endorsed outside of the rulemaking process

Question 65: *If the NRC were to undertake new risk-informed rulemakings, which regulations would be the most beneficial to revise? What would be the anticipated safety benefits?*

Response 65: Aside from 10 CFR 50.55a(h), there might be some small value in looking at risk-informing Appendixes A and B (GDCs and QA), but even there, AREVA doesn't see a lot of value. AREVA suggest that the staff should wait until after the first few reviews of new COL applications, when we'll have a chance to see how the staff interprets the current regulations for the new generation of LWRs.

Question 66: *In addition to revising specific regulations, are there any particular regulations that do not need to be revised, but whose associated regulatory guidance documents, could be revised to be more risk-informed and performance-based? What are the safety benefits associated with revising these guides? Which ones in particular are stakeholders interested in having revised and why?*

Response 66: Since the staff is in the midst of updating relevant guidance for new plants--the SRP and Regulatory Guides—the staff should ensure, in general, that the updating process includes consideration of risk-informed and performance-based approaches to meeting regulatory requirements.

Specifically AREVA suggests the following:

- Diversity and Defense-in-Depth" guidance in BTP HICB-19 should be revised to clearly allow either of two risk-informed options to address LOCA analyses: leak-before-break to limit break size or the revised acceptance criteria being contemplated for larger than transition size breaks for 10 CFR 50.46a, and

- There should be a way to address the assumption of a complete software common mode failure for protection cut-sets. There has to be some logical cutoff where the probability of occurrence is too low to be considered credible for the D3 analysis assumption.

Question 67: *If additional regulations and associated regulatory guidance documents were to be revised, when should the NRC initiate these effort, e.g. immediately or after having started implementation of current risk-informed 10 CFR Part 50 regulations?*

At the end of the ANPR phase, the NRC will assess whether to adjust its approach to risk-inform the requirements for nuclear power reactors including existing and new plants.

Response 67: These efforts should begin immediately.