

POLICY ISSUE NOTATION VOTE

March 28, 2003

SECY-03-0047

FOR: The Commission

FROM: William D. Travers
Executive Director for Operations

SUBJECT: POLICY ISSUES RELATED TO LICENSING NON-LIGHT-WATER REACTOR
DESIGNS

PURPOSE:

To provide for Commission consideration options and recommended positions for resolving the seven policy issues associated with the design and licensing of future non-light-water reactor (LWR) designs discussed in SECY-02-0139, "Plan for Resolving Policy Issues Related to Licensing Non-Light Water Reactor Designs," July 22, 2002 (ADAMS Accession No. ML021790610), and to highlight any implications for licensing future LWRs.

SUMMARY:

This paper contains recommendations for Commission consideration on seven technical policy issues identified in the pre-application reviews to date on non-LWR designs. The seven issues involve the approach to licensing on key aspects of reactor design and operation which relate to Commission policy and practice and which could impact the viability of future non-LWR designs. The recommendations in this paper are intended as first steps to set direction on these issues and provide the basis for further work.

BACKGROUND:

In SECY-02-0139, the staff identified seven technical issues with policy implications resulting from the pre-application activities to date on non-LWR designs (these designs included the Pebble Bed Modular Reactor [PBMR] and the Gas Turbine-Modular Helium Reactor [GT-MHR]). As described in SECY-02-0139, the seven technical issues with policy implications are:

- How should the Commission's expectations for enhanced safety be implemented for future non-LWRs?

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- Should specific defense-in-depth attributes be defined for non-LWRs?
- How should NRC requirements for future non-LWR plants relate to international codes and standards?
- To what extent should a probabilistic approach be used to establish the plant licensing basis?
- Under what conditions, if any, should scenario-specific accident source terms be used for licensing decisions regarding containment and site suitability?
- Under what conditions, if any, can a plant be licensed without a pressure-retaining containment building?
- Under what conditions, if any, can emergency planning zones be reduced, including a reduction to the site exclusion area boundary?

The purpose of SECY-02-0139 was to provide the Commission an early indication of the scope and nature of the technical policy issues, consistent with the Commission's April 1, 2002, staff requirements memorandum (SRM), and to describe the staff's plans for developing recommendations for Commission action.

As discussed in SECY-02-0139, the staff requests Commission action at this time on the policy issues so as to provide early feedback to reactor designers and the staff on issues key to the licensability of future designs. Such early feedback is consistent with the intent of the Commission's Statement of Policy on the Regulation of Advanced Nuclear Power Plants (51 FR 24643) dated July 8, 1986, and is intended to improve the effectiveness, efficiency and predictability of the review process. The guidance provided by the Commission on these seven issues can be used by designers and staff on a plant specific basis or can form the foundation for generic requirements for future plant licensing, if and when a decision is made to proceed in this fashion. It should be noted that four of the seven issues (probabilistic event selection, source term, containment, and emergency planning) were identified as policy issues in previous non-LWR pre-application reviews in the early 1990s, and Commission guidance was received on these issues in an SRM dated July 30, 1993. However, in light of the subsequent emphasis on risk-informed, performance-based regulation, the staff has revisited these issues and the guidance contained in the July 30, 1993, SRM. This is addressed as part of the background and discussion on each issue.

The issues covered by this paper pertain to the approach to licensing on key aspects of reactor design and operation. Although these issues were identified in the context of non-LWR pre-application reviews, it should be noted that the Commission's decisions on these issues have the potential to impact future LWRs. Where this is the case, the impacts are discussed. If any policy issues related to non-LWR fuel cycles or safeguards are identified, they will be addressed in separate papers. It is recognized that the resolution of security issues is still in progress and their resolution may impact future design requirements. Accordingly, the impact of security will be addressed consistent with direction provided by the Commission.

DISCUSSION:

The seven policy issues are discussed in detail in Attachments 1–7. The discussion in the attachment for each issue includes background, options for resolution (including pros and cons), stakeholder feedback, and a recommended position. Each attachment also describes previous Commission guidance on these issues. Summarized below are the guidelines used in evaluating the issues and the staff recommendation for each of the seven issues.

In assessing the options and developing the recommendations on the seven issues, the following general guidelines were employed:

- Keep the risk to the population around a nuclear power plant site consistent with the Commission's 1986 Reactor Safety Goal Policy (51 FR 28044).
- Choose a risk-informed and performance-based approach, wherever practical, consistent with the Commission's 1995 Policy Statement on the Use of Probabilistic Risk Assessment Methods in Nuclear Regulatory Activities (60 FR 42622) and the March 11, 1999, White Paper on Risk-Informed and Performance Based Regulation.
- Use a technology-neutral approach.
- Use the Commission's four performance goals to assess the advantages and disadvantages of the options and to develop recommendations.
- Consider previous Commission guidance on these issues.
- Consider the practicality of the options and recommendations.

Application of these guidelines resulted in the following recommendations for the seven policy issues.

Issue 1: How to implement the Commission's expectations for enhanced safety in future non-LWRs.

The staff recommends that the Commission take the following actions:

- Approve implementation of enhanced safety through a process similar to that used in the evolutionary LWR and advanced light-water reactor (ALWR) design certification reviews (i.e., reactor designers are expected to propose designs with enhanced safety characteristics and the staff reviews each design on its own merits and, on an as-needed basis, recommends additional enhancements in areas of high uncertainty subject to Commission endorsement). Such enhancements could include additional design features, additional testing by the designer, or additional confirmatory testing and/or oversight by NRC in areas of large uncertainty, and would be recommended with the intent to achieve a level of safety and confidence similar to that achieved in the evolutionary and ALWR design certifications.

- In implementing the above, apply the following considerations:
 - When using probabilistic or risk information, modular reactor designs should account for the integrated risk posed by multiple reactors necessary to achieve the overall electrical output desired.
 - The incremental risk to the surrounding population from adding additional units to an existing site is expected to be small due to the enhanced safety characteristics of new designs.

The above recommendations are intended to help ensure that the intent of the Commission's Safety Goal Policy is met.

In the longer term, the Commission may wish to consider a revision to the Policy Statement on the Regulation of Advanced Nuclear Power Plants to include the above recommendation (if approved by the Commission) as well as to expand the scope of the policy statement to include fuel cycle and security considerations for future reactors.

Issue 2: How to specify "defense-in-depth" for non-LWRs (i.e., should a description be developed?)

The staff recommends that the Commission take the following actions:

- 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.)
 - 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).

- Develop the policy statement/description through a process involving stakeholder review, input, and participation.

As part of developing a framework for future plant licensing, as discussed in the Advanced Reactor Research Plan,¹ defense-in-depth considerations will be included. However, given the fundamental nature of the defense-in-depth philosophy to reactor safety, it is recommended that the Commission articulate the elements of this philosophy in a fashion that receives wide distribution and visibility. This would then also serve as guidance for the staff to use and build upon in developing the framework.

¹A draft of the Advanced Reactor Research Plan was provided to the Commission in July 2002. A final version will be provided in April 2003.

The development of such a policy statement or description of defense-in-depth for reactor design and operation would take approximately 1 year, considering time for Advisory Committee on Reactor Safeguards (ACRS) review, public comment, and internal review.

Issue 3: How should NRC requirements for non-LWRs relate to international codes and standards?

The staff recommends that the Commission take the following action:

- Approve NRC proactively participating in development of and endorsing international codes and standards where such codes and standards have been identified by applicants or pre-applicants for use in their submittals or by staff as needed to fill gaps in the NRC's non-LWR infrastructure.²

The intended scope of the codes and standards under consideration would be those associated with the safety of design, manufacture, construction, and operation, and would most likely be in technical areas associated with non-LWRs. However, programmatic areas may also be included, where international harmonization could contribute to agency effectiveness and public confidence. In implementing this recommendation, the staff would use the International Atomic Energy Agency and the Nuclear Energy Agency as resources to assist in the identification of codes and standards needs, the understanding of international codes and standards, and their relation to NRC needs. Application of this recommendation could also be considered for future LWRs in light of the international nature of many of the future LWR designs. However, in either case, implementation of this recommendation will require a stable commitment of resources over the next several years. If approved by the Commission, the staff would proceed to develop a plan for proactive involvement in international codes and standards activities.

The next three issues (Issues 4, 5, and 6) are closely related and should be considered as a package, since the recommendation on each successive issue builds upon the previous recommendations. Accordingly, if the Commission desires to change any of the recommendations on these three issues, the implications for the remaining issues need to be considered.

Issue 4: To what extent can a probabilistic approach be used to establish the licensing basis?

The staff recommends that the Commission take the following actions:

- Modify the Commission's guidance, as described in the SRM of July 30, 1993, to put greater emphasis on the use of risk information by allowing the use of a probabilistic approach in the identification of events to be considered in the design, provided there is sufficient understanding of plant and fuel performance and deterministic engineering judgement is used to bound uncertainties.^{3 4}

²Note: The staff believes that this recommendation is consistent with Management Directive 6.5, "NRC Participation in the Development and Use of Consensus Standards," November 1999.

³Note: The actual design basis events for any particular design would be determined at the time of the staff review of that design. The criteria that would be used to guide this determination would be technology neutral, would include guidance on how to treat uncertainties, and would be determined as part of the development of a framework for future plant licensing consistent with the Commission's decisions on the issues discussed in this paper.

⁴Note: The staff believes that this recommendation is consistent with the Commission's 1995 PRA policy statement and the 1999 Risk-Informed and Performance-Based Regulation White Paper.

- Allow a probabilistic approach for the safety classification of structures, systems, and components.
- Replace the single failure criterion with a probabilistic (reliability) criterion.

This recommendation is consistent with a risk-informed approach. It should be noted that this recommendation expands the use of probabilistic risk assessment (PRA) into forming part of the basis for licensing and thus puts greater emphasis on PRA quality, completeness, and documentation.

Issue 5: Under what conditions should scenario-specific accident source terms be used for licensing decisions?

The staff recommends that the Commission take the following action:

- Retain the Commission's guidance contained in the July 30, 1993, SRM that allows the use of scenario-specific source terms, provided there is sufficient understanding and assurance of plant and fuel performance and deterministic engineering judgement is used to bound uncertainties.

This recommendation will allow credit to be given for the unique aspects of plant design (i.e., performance-based) and builds upon the recommendation under Issue 4. Furthermore, this approach is consistent with prior Commission and ACRS views. However, this approach is also dependent upon understanding fuel and fission product behavior under a wide range of scenarios and on ensuring fuel and plant performance is maintained over the life of the plant.

Issue 6: Under what conditions can a plant be licensed without a pressure retaining containment building (i.e., a confinement building instead of a containment)?

The staff recommends that the Commission take the following actions:

- Approve the use of functional performance requirements to establish the acceptability of a containment or confinement structure (i.e., a non-pressure retaining building may be acceptable provided the performance requirements can be met).
- If approved by the Commission, develop the functional performance requirements using as a starting point guidance contained in the Commission's July 30, 1993, SRM and the Commission's guidance on the other issues contained in this paper.

This recommendation is coupled to the recommendations on Issues 4 and 5 (event selection and source term) discussed above and, similar to those issues, would represent a risk-informed and performance-based method to account for the unique aspects of each reactor design. In addition, resolution of this issue will establish a key element for incorporation into any policy or description of defense-in-depth as recommended under Issue 2 above.

Issue 7: Under what conditions can the emergency planning zone be reduced, including a reduction to the site exclusion area boundary?

The staff recommends that no change to emergency preparedness requirements be made at this time. This recommendation is consistent with the guidance contained in the Commission's July 30, 1993, SRM and is based upon the following two considerations:

- Provision already exists in 10 CFR 50.47 for accommodating the unique aspects of high-temperature gas reactors.
- In the near term, new plants are likely to be built on an existing site which conforms to current requirements.

If approved by the Commission, the role of emergency preparedness in defense-in-depth would be addressed as part of the development of a policy or description of defense-in-depth as recommended under Issue 2 above. In the longer term, if and when a need for change in emergency preparedness requirements is identified, that policy or description would serve as guidance in assessing the proposed change.

The above issues address the approach to licensing on key aspects of plant design and operation and could affect the viability of future non-LWR designs. The staff's recommendations are intended as first steps to set direction on these issues, recognizing that actual implementation of these recommendations will require additional time, work, documentation, and stakeholder interaction to develop the details. However, it is important to consider that the Commission's decision on these issues will establish the approach to licensing in key areas and provide the basis for follow-on work.

RESOURCES:

If approved by the Commission, the recommendations contained in this paper will require resources for implementation. Some of the resources are already in the FY 2003 budget and future budget requests while others will need to be obtained through the PBPM process. In summary, the resource implications of the recommendations in this paper are as follows:

- Issue 1: no additional resource implications.
- Issue 2: will require one additional FTE in FY 2003.
- Issue 3: will require one additional FTE in FY 2003 and, depending upon the number of international codes and standards NRC reviews or participates in, could require an additional 1–3 FTE/FY beginning in FY 2004. However, it should be noted that in the long run, such an expenditure of resources should result in increased efficiency during the review of actual applications.
- Issue 4: no additional resources required (included in activities related to development of a framework for future plant licensing).

- Issue 5: no additional resources required to develop criteria; however, if scenario-specific source terms are used in an actual application, implementation would likely include confirmatory research on fuel performance and fission product transport. The cost of this research would be design specific and would need to be developed based upon plant specific details and the potential for cooperative research.
- Issue 6: no additional resources required (included in activities related to development of a framework for future plant licensing).
- Issue 7: no additional resource implications.

COORDINATION:

In developing the above recommendations, the staff had the benefit of ACRS and other stakeholder input. ACRS's views are contained in a letter to Chairman Meserve, dated December 13, 2002 (Attachment 8). A public workshop was held October 22–23, 2002, where the issues and options were discussed. A summary of the key points made at the workshop is included as Attachment 9 and the complete summary is available at ADAMS Accession No. ML023180764. The Office of the General Counsel has no legal objections. The Office of the Chief Financial Officer has reviewed this paper for resource implications and has no objections.

RECOMMENDATION:

That the Commission approve the recommendations summarized above and described in more detail in Attachments 1–7. Note that:

- The staff will proceed in accordance with the recommendations, unless otherwise directed.
- Security concerns will be addressed consistent with direction provided by the Commission.

/RA/

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Attachments: See attached list

SUBJECT: POLICY ISSUES RELATED TO LICENSING NON-LIGHT-WATER REACTOR
DESIGNS

Attachments:

1. Expectations for Safety
2. Defense-in-Depth
3. Relation to International Standards
4. Probabilistic Event Selection, Safety Classification, and Reliability Criteria
5. Source Term
6. Containment vs. Confinement
7. Emergency Preparedness
8. ACRS Letter to Chairman Meserve, December 13, 2002
9. Workshop Summary—Key Points

Expectations for Safety

ISSUE 1: How to implement the Commission's expectations for enhanced safety in future non-light-water reactors.

BACKGROUND:

The Commission's Policy Statement on Severe Accidents (50 FR 32138), August 8, 1985, stated that the Commission expects new plants to achieve a higher standard of severe accident safety performance than prior designs.

The Commission's "Policy Statement on the Regulation of Advanced Nuclear Power Plants" (51 FR 24643), July 8, 1986, stated that the Commission expects advanced reactors to provide enhanced margins of safety and/or utilize simplified, inherent, passive, or other innovative means to accomplish their safety functions. In addition, in that same policy statement the Commission stated that it expects, as a minimum, at least the same degree of protection of the public and the environment that is required for current generation light-water reactors (LWRs) and that advanced reactor designs will comply with the Commission's Safety Goal Policy Statement.

For the evolutionary and advanced light-water reactors (ALWRs) both the industry and the staff have taken steps to implement the Commission's expectations. In the late 1980s and early 1990s, the industry (led by the Electric Power Research Institute [EPRI]) developed a "Utility Requirement Document" (URD) for ALWRs which defined the requirements that utilities desired in ALWRs. These utility requirements included reduced core damage frequency (CDF) and radioactive material release objectives from those achieved by current generation LWRs as well as other plant features that EPRI considered would make future LWRs "substantially safer than existing plants." The staff reviewed the URD and documented its findings in NUREG-1242, "NRC Review of EPRI's ALWR-URD." In summary, the staff found that a plant designed and operated in accordance with the URD would meet NRC requirements and expectations for enhanced safety.

For non-LWRs the staff reviewed three conceptual designs sponsored by the U.S. Department of Energy (DOE) (one high-temperature gas-cooled reactor, the Modular High-Temperature Gas-Cooled Reactor (MHTGR), and two liquid metal reactors, Power Reactor Innovative Small Module [PRISM] and Sodium Advanced Fast Reactor [SAFR]). These designs were submitted in the late 1980s for the pre-application review in accordance with the "Commission's Policy Statement on the Regulation of Advanced Nuclear Power Plants." Each of these designs had as an objective enhanced safety through the use of simplified, passive safety systems, less reliance on human actions, and greater prevention of core damage. The staff issued the following pre-application safety evaluation reports on these designs.

- NUREG-1338 (Draft), "Preapplication Safety Evaluation Report for the Modular High-Temperature Gas-Cooled Reactor (MHTGR)," June 1995.
- NUREG-1368, "Preapplication Safety Evaluation Report for the Power Reactor Innovative Small Module (PRISM) Liquid-Metal Reactor," February 1994.

- NUREG-1369, “Preapplication Safety Evaluation Report for the Sodium Advanced Fast Reactor (SAFR) Liquid-Metal Reactor,” December 1991.

Although steps have been taken toward incorporating enhanced safety into future designs, the Commission has not required enhanced safety through the promulgation of generic requirements. In fact, in its staff requirements memorandum (SRM) of June 15, 1990, “SECY-89-102—Implementation of the Safety Goals,” the Commission explicitly stated that it will not use the industry’s design objectives for advanced plants (e.g., 10^{-5} /reactor year CDF) as the basis to establish new requirements. The Commission has, however, on a design-specific basis, required enhancements on the certified evolutionary LWR and ALWR designs (ABWR, System 80⁺ and AP-600) in areas of higher uncertainty to help ensure a higher standard of severe accident performance and has approved those features in the design certification rulemaking for these designs.

DISCUSSION:

In past activities related to future reactors (both LWRs and non-LWRs) the Commission has, for the most part, relied on industry to provide enhanced safety in new designs and has only proposed enhancements in specific areas of high uncertainty (e.g., SECY-93-087, “Policy, Technical, and Licensing Issues Pertaining to Evolutionary and Advanced Light-Water Reactor [ALWR] Designs, April 2, 1993). In addition, siting locations or the number of units on a site were not a part of the previous reviews except for the issue of modular reactors, where several smaller reactors were needed to achieve the electrical output of one large unit.

Currently, there is a possibility that one or more applications for new plants will be forthcoming in the near future and siting locations will likely be considered through the early site permit process. Three potential early site permit applications are currently under discussion, all involving sites with existing nuclear power plants, and possibly including multiple units.

Traditionally, risk calculations related to the Safety Goal quantitative health objectives (QHOs) have been done on a per plant basis and the guidelines developed and used in the risk-informed process (e.g., Regulatory Guide 1.174, Option 3 Framework) were based on risk from an individual plant. The Safety Goal Policy itself implies that the risk should be calculated on a per site basis. In defining the population at risk in applying the QHOs, the Safety Goal Policy refers to the “plant site.” To be properly risk-informed, the assessment should consider the integrated effect of multiple plants. This has implications for the level of safety for future LWRs as well as non-LWRs.

Accordingly, the possibility of additional nuclear power plants (some on sites with existing plants) raise the following fundamental question for Commission consideration:

“How should the integrated risk of multiple units on a site be accounted for?”

At the public workshop held October 22–23, 2002, the industry participants supported a process similar to that used in certifying the evolutionary LWR and ALWR designs (ABWR, System 80⁺, and AP600) where the designers proposed designs with enhanced safety characteristics and each design was reviewed on its own merits, with additional safety enhancements targeted only at areas of high uncertainty.

OPTIONS:

The options considered by the staff in addressing this issue are:

- (a) Do not generically require enhanced safety on future non-LWRs, but rather rely on industry to propose designs with enhanced safety characteristics and, through a process similar to that used in the evolutionary LWR and ALWR design certifications, impose any additional enhancements with Commission endorsement only to address areas of high uncertainty.

This option would, in effect, maintain the status quo and is consistent with the overall philosophy in the Nuclear Energy Institute's (NEI's) May 2002 white paper on "A Risk-Informed, Performance-Based Regulatory Framework for Power Reactors." It also would have the least impact on future LWRs (those under review as well as those already certified). All future designs currently under consideration have claimed enhanced safety. The staff would review each design on an individual plant basis and licensing criteria and risk metrics for non-LWRs would be directed toward the same level of accident and core damage prevention as current criteria as well as the same level of accident mitigation. Any safety enhancements would be applied on a plant-specific basis to address areas of high uncertainty. The incremental increase in risk to the surrounding population from additional reactors on a site would be expected to be small due to the enhanced safety characteristics of the new designs, which are likely to be an order of magnitude better than current designs.

- (b) Require an enhanced level of safety on future non-LWRs.

This option would generically impose criteria and risk metrics on future non-LWRs directed toward improved accident and core damage prevention as well as improved accident mitigation. Setting higher standards for the level of safety to be achieved could also help compensate for the greater uncertainties associated with new non-LWR designs. For example, enhanced accident and core damage prevention in a design could help compensate for uncertainties in the severe accident area, since the risk from severe accidents would be lower. This option would also help keep the incremental increase in risk to the population around a site small. This option is not consistent with NEI's May 2002 white paper and would apply equally to future LWRs as well as non-LWRs.

- (c) Require an enhanced level of confidence in the performance of plant systems, structures and components.

This option would utilize criteria and risk metrics directed toward the same level of safety as current criteria; however, to compensate for the reduced experience with non-LWRs, enhanced research and development, testing, and NRC oversight (e.g., fuel quality) could be required to increase confidence in the performance of plant systems, structures and components, and confirm plant safety. This option would be similar to Option a in other aspects.

- (d) Do not generically require enhanced safety on future non-LWRs, but rely on industry to propose designs with enhanced safety characteristics. Use a process similar to that used in the evolutionary LWR and ALWR design certification reviews to impose any additional enhancements, including enhancements to establish increased confidence in the design and/or performance of plant systems, structures or components with Commission endorsement.

This option is a combination of options (a) and (c) above and acknowledges that enhancements recommended by the staff can be related to the reactor design as well as to programs and processes (e.g., oversight) needed to ensure confidence in the performance of plant systems, structures, and components.

RECOMMENDATION:

The staff recommends that the Commission take the following actions:

- Approve implementation of enhanced safety through a process similar to that used in the evolutionary LWR and advanced light-water reactor (ALWR) design certification reviews (i.e., reactor designers are expected to propose designs with enhanced safety characteristics and the staff reviews each design on its own merits and, on as needed basis, recommends additional enhancements in areas of high uncertainty subject to Commission endorsement). Such enhancements could include additional design features, additional testing by the designer, or additional confirmatory testing and/or oversight by NRC in areas of large uncertainty, and would be recommended with the intent to achieve a level of safety and confidence similar to that achieved in the evolutionary and ALWR design certifications.
- In implementing the above, apply the following considerations:
 - When using probabilistic or risk information, modular reactor designs should account for the integrated risk posed by multiple reactors necessary to achieve the overall electrical output desired.
 - The incremental risk to the surrounding population from adding additional units to an existing site is expected to be small due to the enhanced safety characteristics of new designs.

This recommendation is consistent with Option d above. The recommendation is intended to help ensure that the intent of the Commission's Safety Goal Policy is met.

In the longer term, the Commission may wish to consider a revision to the Policy Statement on the Regulation of Advanced Nuclear Power Plants to include the above recommendation (if approved by the Commission) as well as to expand the scope of the policy statement to include fuel cycle and security considerations for future reactors.

Defense-in-Depth

ISSUE 2: How to specify defense-in-depth for non-light-water reactors (i.e., should a description be developed?).

BACKGROUND:

The philosophy of defense-in-depth (DID) has been a fundamental part of NRC's regulatory programs since NRC's inception. It is mentioned in numerous places, including the Safety Goal Policy Statement, the probabilistic risk assessment (PRA) Policy Statement and the Commission's 1999 White Paper on Risk-Informed, Performance-Based Regulation. However, the specific elements that constitute DID are not described. The current regulations are also based upon a philosophy of DID; however, the only places the term DID is used in the regulations are in 10 CFR Part 50, Appendix R, Fire Protection, and 10 CFR Part 100.1, Reactor Site Criteria.

It should be recognized that compliance with the regulations ensures DID for light-water reactors (LWRs). The goal of DID is best described by the definition in the Commission's 1999 white paper on risk-informed, performance-based regulation which states: "Defense-in-depth is an element of the NRC's Safety Philosophy that employs successive compensatory measures to prevent accidents or mitigate damage if a malfunction, accident, or naturally-caused event occurs at a nuclear facility. The defense-in-depth philosophy ensures that safety will not be wholly dependent on any single element of the design, construction, maintenance, or operation of a nuclear facility. The net effect of incorporating defense-in-depth into design, construction, maintenance, and operation is that the facility or system in question tends to be more tolerant of failures and external challenges." In addition, Regulatory Guide 1.174 contains a discussion of DID and those elements of DID that need consideration when proposing risk-informed changes to a plant's current licensing basis, however, the focus of the discussion is on assessing changes to DID, not defining it.

Others have attempted to describe the elements of DID. Examples include the International Atomic Energy Agency (IAEA) and the International Nuclear Safety Advisory Group (INSAG) in their documents:

- IAEA Safety Series US-R-1, "Safety Assessment and Verification for Nuclear Power Plants," 2001.
- IAEA-TECDOC-986, "Implementation of defense in depth for next generation light water reactors," December 1997.
- Safety Series No. 75-INSAG-3, "Basic Safety Principles for Nuclear Power Plants," 1988.
- Safety Series No. 75-INSAG-3, Rev. 1, INSAG-12, "Basic Safety Principles for Nuclear Power Plants," 1999.

These documents describe DID as having a series of levels, with each level building upon the previous one. The levels contain programmatic as well as physical elements. ACRS, in a letter dated May 19, 1999, discussed the role of DID in a risk-informed regulatory system. In that letter they discuss two fundamental approaches to DID, which they call structuralist and rationalist. The structuralist approach is mainly one of deterministic engineering judgement regarding what constitutes the elements of DID and could be developed generically or on a plant-specific basis, and the rationalist approach is mainly one utilizing a PRA whereby the elements of DID are those items necessary to compensate for uncertainties identified by a plant-specific PRA, such that design and performance goals can be met. Finally, the Nuclear Energy Institute (NEI), in its white paper on "A Risk-Informed, Performance-Based Regulatory Framework for Power Reactors," has proposed that DID be considered a process to account for uncertainties and applied on a design-specific basis.

DISCUSSION:

With the LWR orientation of the current regulations, application of the DID philosophy for non-LWRs has, in the past, been done on a case-by-case basis. With the potential for future plant applications, some of which could be non-LWRs or LWR designs very different from current LWR designs, it may be appropriate to consider developing more explicit guidance describing the DID philosophy as it pertains to reactor design and operation. This could help ensure a more uniform application of the DID philosophy in the future (either on a plant-specific basis or generically) and could also be of use in other areas where DID is important, such as the Regulatory Analysis Guidelines (NUREG/BR-0058, Rev. 3). If more explicit guidance is developed, a fundamental question then becomes what are considered the elements of DID? For example, do they include programmatic as well as physical elements? The development of more explicit guidance would also support development of a framework for future plant licensing and the dissemination of such guidance could be through a policy statement or white paper such that it receives broad visibility and application.

At the public workshop held October 22–23, 2002, there was broad support for developing a description of DID as long as the development was done through a process that included opportunity for public review and comment.

OPTIONS:

The options considered by the staff in addressing this issue are:

- (a) Assess DID on a case-by-case basis as part of the review of a specific design (i.e., do not develop a description).

This option would, in effect, maintain the status quo with no specific guidance on DID, other than what is necessary to include in the framework for future plant licensing. The need for design or programmatic features to compensate for uncertainties would be decided case-by-case based upon confidence in the design, including its supporting research and development, and worldwide experience. This option would not ensure uniform application of DID among designs nor would it provide guidance on DID for use in other activities, such as the Reactor Oversight Program or the Regulatory Analysis Guidelines.

- (b) Develop a policy statement or description (e.g., white paper) of the elements considered as DID as guidance to designers and the staff.

This option would, in effect, implement the Commission's definition of DID contained in the March 11, 1999, White Paper on RIPB Regulation. It would describe those elements (which could be a combination of structuralist and rationalist elements, and include programmatic as well as hardware-related items) necessary to ensure DID and could be useful in the design, review and oversight process. It could also be useful in other areas such as regulatory analysis. The documentation of the DID description could be in the form of a Commission Policy Statement, White Paper or other high-level document. The policy statement or description would be technology neutral and risk-informed and written to describe:

- the objectives of defense-in-depth (philosophy)
- the scope of defense-in-depth (design, operation, etc.)
- the elements of defense-in-depth (high level principles and guidelines)

The advantage of this option is that it would help ensure uniform application of DID by designers and the staff and would establish a set of attributes that the plant would have to have no matter what the design or calculated risk. This could contribute to public confidence. As part of developing a framework for future plant licensing, as discussed in the Advanced Reactor Research Plan,¹ DID considerations will be included. A comprehensive description of DID could form a structure from which to develop the framework for future plant licensing. This framework could then be used to implement the DID description and to guide future plant reviews, either on a case-by-case basis or through a generic action to codify the framework, if it is decided to take such a generic action.

- (c) Develop a programmatic process to ensure DID is implemented in reactor designs.

This option would be similar to that proposed by NEI in their May 2002 white paper on "A Risk-Informed, Performance Based Regulatory Framework for Power Reactors." It would not specify any specific DID plant features but rather would set up a process to be followed by designers and the staff whereby a design could be evaluated against a set of criteria and, depending upon uncertainties in the analysis, additional features or actions would be added to reduce the uncertainty. These additional plant features or actions would be considered DID, and the DID process would, in effect, be a way to treat uncertainties. This process could be documented in various ways, similar to Option (b) above. This option would provide flexibility in the application of DID to different designs and could be a process applicable to non reactor activities as well. Its disadvantages are that it could be subject to non-uniform application and it does not specify any specific attributes that must be included as part of DID (e.g., two ways to accomplish reactor shutdown).

¹A draft of the Advanced Reactor Research Plan was provided to the Commission in July 2002. A final version will be provided in April 2003.

- (d) Develop a policy statement or description (e.g., white paper) of DID that could include specific technical elements as well as process elements as guidance to designers and their staff.

This option is a combination of options b and c above and is put forth in recognition of the fact that in developing the policy statement or description of DID, input will be received from stakeholders that could influence the scope and content of the DID description. Accordingly, the elements of DID could be technical and/or process and will be determined as part of developing the DID policy statement or description.

Nevertheless, the policy statement or description would be written to be technology neutral and risk-informed and address:

- the objectives of defense-in-depth (philosophy)
- the scope of defense-in-depth (design, operation, etc.)
- the elements of defense-in-depth (high level principles and guidelines)

RECOMMENDATION:

The staff recommends the Commission take the following actions:

- 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.)
 - the elements of defense-in-depth (high level principles and guidelines)

The policy statement/description would be technology neutral and risk-informed and would be useful in providing consistency in other regulatory programs (e.g., Regulatory Analysis Guidelines).

- Develop the policy statement/description through a process involving stakeholder review, input and participation.

This recommendation is consistent with Option d above. Given the fundamental nature of the defense-in-depth philosophy to reactor safety, it is recommended that the Commission articulate the elements of this philosophy in a fashion that receives wide distribution and visibility. A description of DID would help provide consistency to the application of DID and coherence with other regulatory activities that include consideration of DID (e.g., Regulatory Analysis Guidelines). Clearly, such a description would need to be assessed for its implications for future LWRs. The schedule for developing such a description would likely be 1 year considering the need for stakeholder input, ACRS review, and internal review and comment.

Relation to International Standards

ISSUE 3: How should NRC requirements for non-light-water reactors (non-LWRs) relate to international codes and standards?

BACKGROUND:

Other countries have had experience with non-LWRs and continue to perform research and development on these technologies. The United Kingdom operates 14 advanced gas reactors, Japan recently began operation of a 30-megawatt high-temperature, gas-cooled research reactor, and China operates the HTR-10, a 10-MWt high-temperature, gas-cooled reactor (HTGR) prototype. Both the International Atomic Energy Agency (IAEA) and the European Union have active programs on HTGR safety and development. Accordingly, other countries and organizations have developed, to varying degrees, an infrastructure supporting non-LWR technologies, including the development of standards and requirements. In addition, many future reactor design and development efforts are being conducted via international partnerships and are intended to be marketed internationally. This has been the case with the pebble bed modular reactor (PBMR) and gas-turbine modular helium-cooled reactor (GT-MHR) and will likely be the case on other future efforts, including advanced LWRs.

NRC Management Directive 6.5, "NRC Participation in the Development and Use of Consensus Standards," provides guidance on and encourages the use of consensus standards, where practical. One of the objectives of this directive is to "Promote the efficient and effective use of NRC resources by focusing staff participation on the development of standards that address a defined current or anticipated regulatory need." This directive advises that staff should seek out existing consensus standards to address a need for new or revised technical standards rather than writing a Government-unique standard. In order to improve agency understanding of consensus standards, this management directive encourages staff participation in the development of consensus standards. In addition, this directive implements OMB Circular A-119 and states that "OMB Circular A-119 does not establish a preference between domestic and international consensus standards, but in the interests of promoting trade and implementing the provisions of international treaty agreements, international standards, such as those from the International Standards Organization and the International Electrotechnical Commission, are considered for agency regulatory and procurement applications."

The Commission has previously given staff direction which could be relevant to the level of involvement it expects with international safety standards and requirements. In an SRM from the August 14, 2001, briefing on NRC International Activities, the staff was directed as follows:

- "The staff should continue to look at all program areas where we can benefit from cooperation with the international community, including opportunities in the U.S."

The scope of this issue addresses the extent to which NRC should be proactive in developing and endorsing international codes and standards related to the safety of design, manufacturing,

construction, and operation of non-LWRs. This could include programmatic as well as technical codes and standards.

DISCUSSION:

Many future reactor design and development efforts are being conducted via international partnerships. These international efforts include design, manufacturing, research and development and marketing. With these international efforts comes the desire to harmonize the licensing requirements as much as possible. For NRC purposes, the international efforts represent an opportunity to build upon work done by others and to benefit from their experience. This would be of great value considering that much of the NRC's current regulatory infrastructure, technical knowledge, and experience is LWR oriented. However, any such effort to participate in or review international standards would require a stable commitment of resources.

The international codes and standards that could be of interest to NRC include both technical and programmatic areas. The IAEA has produced a number of standards pertaining to reactor safety. Additionally, the IAEA has produced documents dealing with regulatory issues such as the application of the defense-in-depth concept. The International Standards Organization (ISO) has also generated standards dealing with subjects such as quality assurance. International collaboration with other regulatory agencies and international standards organizations could be a useful way to identify the need for codes and standards and facilitate their development.

At the public workshop held October 22–23, 2002, participants agreed that the NRC must be involved at some level with international codes and standards because of the international nature of new reactor designs. There was general agreement that the NRC should review any international codes and standards referenced in applications and pre-applications. Additionally, stakeholders recommended that the NRC utilize the experience of other regulatory bodies to facilitate the review of international codes and standards. This recommendation came with the cautionary note that the NRC should not “rubberstamp” a standard simply because another regulatory body endorsed it. The workshop participants agreed that the NRC should be proactive in selected areas by participating in developing and endorsing international codes and standards that could fill critical gaps in NRC's infrastructure for non-LWRs.

OPTIONS:

The options considered by the staff in addressing this issue are:

- (a) Review international codes and standards only as part of an application or pre-application review.

This option would have NRC review international codes and standards only as necessary to review a licensing or pre-application submittal which incorporates such standards.

While this option is consistent with MD 6.5 and would provide some benefit from international experience by the fact that international codes and standards could be endorsed, it does not include NRC participation in their development (including exposure to the technical basis and experience supporting the standard) and thus cannot influence their outcome. Accordingly, the

NRC staff understanding will not be as great as it would be if staff was involved in developing the codes or standards. In addition, the codes and standards review itself as well as the resolution of any problems with the standards would need to be addressed as part of the review of an actual application or preapplication, thus potentially delaying the review process. This option would involve the least amount of resources.

- (b) Proactively participate in the identification, development, and endorsement of international codes and standards where such standards have been identified by the staff as being needed by applicants or pre-applicants for use in their submittals and can fill gaps in the NRC's non-LWR infrastructure.

This option would have NRC help identify, participate in, and endorse international codes and standards where such codes and standards are expected to be part of an application or are needed to fill gaps in our infrastructure. It allows NRC greater benefit from international experience than Option a by including involvement in the creation of the codes and standards. It also would contribute to the efficiency of the review of an actual application by reviewing and resolving in advance issues related to international codes and standards use. Additionally, it helps to fill gaps in the NRC infrastructure related to non-LWR reactors without requiring NRC endorsement of all existing international codes and standards, and it is consistent with Management Directive 6.5. If approved by the Commission, work in this regard would most likely be in technical areas associated with non-LWRs; however, programmatic areas may also be included, where international harmonization could contribute to agency effectiveness and public confidence. In implementing this recommendation, the staff would use the IAEA and the Nuclear Energy Agency (NEA) as resources to assist in the identification of codes and standards needs and in the understanding of international codes and standards. Application of this recommendation could also be considered for future LWRs in light of the international nature of many of the future LWR designs. However, in either case, implementation of this recommendation will require a stable commitment of resources over the next several years.

RECOMMENDATION:

The staff recommends that the Commission take the following action:

- Approve proactive participation by NRC in development of and endorsing international codes and standards where such codes and standards have been identified by applicants or pre-applicants for use in their submittals or by staff as needed to fill gaps in the NRC's non-LWR infrastructure.¹

This recommendation is consistent with Option b above. If approved by the Commission, the staff would proceed to develop a plan for proactive involvement in international codes standards activities consistent with the above.

¹Note: The staff believes that this recommendation is consistent with Management Directive 6.5, "NRC Participation in the Development and Use of Consensus Standards," November 1999.

Probabilistic Event Selection, Safety Classification and Reliability Criteria

This issue, along with the issues of source term and containment discussed in Attachments 5 and 6, respectively, are closely related and should be considered together since the recommendation on each successive issue builds upon the recommendations in the previous issues.

ISSUE 4: To what extent can a probabilistic approach be used to establish the licensing basis for:

- selection of events to be considered in the design and for emergency planning?
- safety classification of systems, structures, and components?
- replacement of the single-failure criterion?

BACKGROUND:

In SECY-93-092, "Issues Pertaining to the Advanced Reactor (PRISM, MHTGR, and Process Inherent Ultimately Safe [PIUS]) and CANDU 3 Designs and Their Relationship to Current Regulatory Requirements," April 8, 1993, the staff described the approaches proposed by non-light-water reactor (LWR) designers for the selection of events to be considered in the design and for safety classification. Except as noted below, these approaches are similar to what was proposed for the pebble bed modular reactor (PBMR) during its pre-application review.

As a result of reviewing the proposed approaches the staff, in SECY-93-092, proposed to develop a single approach for accident evaluation to be applied to all advanced reactor designs and the Canadian Deuterium Uranium (CANDU 3) design during the pre-application review. The approach proposed in SECY-93-092 had the following characteristics:

- Events and sequences would be selected deterministically and would be supplemented with insights from probabilistic risk assessments (PRAs) of the specific designs.
- Categories of events would be established according to expected frequency of occurrence. One category of events that would be examined is accident sequences of a lower likelihood than traditional LWR design basis accidents. These accident sequences would be analyzed without applying the conservatisms used for design basis accidents. Events within a category equivalent to the current design basis accident category would require conservative analyses, as is presently done for LWRs.
- Consequence acceptance limits for core damage and onsite or offsite releases would be established for each category to be consistent with Commission guidance.
- Methodologies and evaluation assumptions would be developed for analyzing each category of events consistent with existing LWR practices.
- Source terms would be determined as approved by the Commission.

- A set of events would be selected deterministically to assess the safety margins of the proposed designs, to determine scenarios to mechanistically determine a source term, and to identify a containment challenge scenario.
- External events would be chosen deterministically on a basis consistent with that used for LWRs.
- Evaluations of multi module reactor designs would be considered as to whether specific events apply to some or all reactors on site for the given scenario for all operations permitted by proposed operating practices.

In response to SECY-93-092 the Commission issued a staff requirements memorandum (SRM) on July 30, 1993, which approved the staff proposal.

The approach described in SECY-93-092 has differences from what was proposed for the PBMR during its pre-application activities. The PBMR approach consisted of defining event categories by the frequency of their occurrence, identifying acceptance criteria for each event category and designing the plant for events down to a frequency of 10^{-5} /plant-year. Events less frequent than this would be considered for emergency planning purposes down to 5×10^{-7} /plant-year. Structures, systems, and components (SSCs) would be classified "safety related" if they were necessary to enable the plant to meet acceptance criteria.

DISCUSSION:

As a result of the recent non-LWR pre-application activities, the staff has reviewed the previous Commission approved approach to see if any changes should be considered. Specifically, this review focused on what has happened since 1993 where greater emphasis has been placed on using risk information in the regulatory process. The Commission's expectations in this regard were expressed in the 1995 PRA Policy Statement on the uses of PRA which states: "The use of PRA technology should be increased in all regulatory matters to the extent supported by the state-of-the-art in PRA methods and data and in a manner that complements the NRC's deterministic approach and supports the NRC's traditional defense-in-depth philosophy." The issuance of this policy statement initiated a number of activities to increase the use of PRA in regulatory activities. These included the development of Regulatory Guides 1.174, 1.175, 1.176, 1.177 and 1.178, initiating work to risk-inform 10 CFR Part 50 and implementing the revised reactor oversight process.

In risk-informing the technical requirements of 10 CFR Part 50 (Option 3) the staff has developed a "framework" document to guide the staff which contains an event category approach similar to what has been proposed for the PBMR. In addition, in risk-informing 10 CFR 50.46, "Acceptance Criteria for Emergency Core Cooling Systems for Light-Water Nuclear Power Reactors," the staff has proposed to allow the use of risk information to replace the single-failure criterion and, in SECY-02-0176, has proposed to use risk information to modify the special treatment requirements on safety-related SSCs in 10 CFR Part 50. Accordingly, much of what has been proposed for the PBMR has, in one way or another, received Commission attention in other programs. Therefore, it may be appropriate to reconsider the approach approved in 1993 to allow more reliance on PRA.

At a public workshop on October 22–23, 2002, the industry participants supported a probabilistic approach for event selection, safety classification, and replacing the single-failure criterion. The public interest group representative did not support such a change due to a lack of confidence in PRAs.

OPTIONS:

The options considered by the staff in addressing this issue are:

- (a) Retain the approach approved by the Commission in its July 30, 1993, SRM where engineering judgement, based upon experience, is used to deterministically select events to be considered in the design with PRA information being used to supplement the selection.

This option uses experience and qualitative judgements regarding likelihood and risk to select events to be considered in the design and to classify SSCs as safety related. In general, bounding events are identified to envelop categories of events. This option would bring PRA into the licensing basis only as a supplement to the deterministic approach.

- (b) Use a probabilistic approach to select events to be considered in the design, to classify SSCs as safety related and to replace the single-failure criterion.

This option would be similar to that proposed for the PBMR. It is, in effect, a risk-based approach. Event categories for anticipated operational occurrences (AOOs), design basis events (DBE) and emergency planning (EP) would be defined probabilistically and PRA would be used to identify those events which fall into each category. Acceptance criteria for each event category would be established. An example of such event categories and their associated acceptance and frequency criteria are shown in the following table. Conservative analysis would be used for AOOs and DBEs (using, for example, a specified level of confidence) and best-estimate analysis for EP. The safety classification of SSCs would be determined based upon the importance of the SSC to meeting the acceptance criteria or staying within the event category (as an example, importance measures could be used for this determination). The single-failure criterion would be replaced with a reliability criterion and the event scenarios identified in the PRA would be examined against this criterion. This could lead to having to consider multiple failures in AOO, DBE, and EP accident scenarios.

The key advantage to this option is that it provides a technology-neutral, structured and consistent way to establish the design basis that realistically considers equipment and human performance. It does, however, require that PRA become part of the licensing basis for the plant with appropriate controls on PRA completeness, quality and documentation, including change control for the PRA.

Example Event Selection Criteria

Event Category	Frequency Range *	Acceptance Criteria
AOO**	$> 10^{-2}/\text{plant-year}$	10 CFR Part 20
DBEs**	$10^{-2}/\text{plant-year} - 10^{-6}/\text{plant-year}$	10 CFR 50.34 or a fraction thereof
EP***	$10^{-6}/\text{plant-year} - 10^{-8}/\text{plant-year}$	N/A

-
- * Frequency range applies to initiating event or event scenario
 ** Conservative analysis
 *** Best estimate analysis

(c) Use an approach where probabilistic information is supplemented by deterministic engineering judgement.

This option differs from Option a in that in this option the deterministic engineering judgement is used to complement the PRA, whereas in option a the PRA is used to complement the deterministic engineering judgement. This change in orientation would be appropriate if it is the Commission's view that the use of PRA information has matured to the point that it can play a more prominent role in safety analysis and licensing decisions. Similar to option b, this option would define event categories for AOOs, DBEs, and EP events probabilistically and would use a probabilistic approach for safety classification of SSCs and a reliability approach to replace the single failure criteria. Even though this option places greater reliance on the PRA, it is still considered risk-informed, since it does not place sole reliance on the PRA. It could, however, be considered to go beyond the PRA Policy Statement in the intended use of PRA information. Similar to Option b, this option requires that PRA become part of the licensing basis of the plant, with appropriate controls on PRA completeness, quality and documentation.

RECOMMENDATION:

The staff recommends the Commission take the following actions:

- Modify the Commission's guidance, as described in the SRM of July 30, 1993, to put greater emphasis on the use of risk information by allowing the use of a probabilistic approach in the identification of events to be considered in the design, provided there is sufficient

understanding of plant and fuel performance and deterministic engineering judgement is used to bound uncertainties.^{1 2}

- Allow a probabilistic approach for the safety classification of structures, systems, and components.
- Replace the single-failure criterion with a probabilistic (reliability) criterion.

This recommendation is consistent with Option c above and, if approved by the Commission, would lead to a technology neutral, systematic and consistent approach for establishing key aspects of the licensing basis for non-LWRs, while accommodating their unique aspects. The actual probabilistic criteria for each event category would be developed as a follow-on activity (i.e., as part of the development of a framework for future plant licensing) and would be consistent with the level of safety for future plants approved by the Commission under Issue 1. It is envisioned that this approach would still result in a set of design basis accidents for each plant design (i.e., consisting of key accident scenarios from each event category).

¹Note: The actual design basis events for any particular design would be determined at the time of the staff review of that design. The criteria that would be used to guide this determination would be technology neutral, would include guidance on how to treat uncertainties and would be determined as part of the development of a framework for future plant licensing consistent with the Commission's decisions on the issues discussed in this paper.

²Note: The staff believes that this recommendation is consistent with the Commission's 1995 PRA policy statement and the 1999 Risk-Informed and Performance-Based Regulation White Paper.

Source Term

ISSUE 5: Under what conditions, if any, should scenario-specific accident source terms be used for licensing decisions regarding containment and site suitability?

BACKGROUND:

Current light-water reactors (LWRs) use site-specific parameters (e.g., exclusion area boundary) and a deterministic predetermined source term into containment to analyze the effectiveness of the containment and site suitability for licensing purposes. The LWR deterministic source terms are described in Atomic Energy Commission (AEC) Technical Issue Document 14844 (TID-14844) and Nuclear Regulatory Commission (NRC) NUREG-1465, and represent a bounding fission product release associated with an in-vessel core melt accident in an LWR.

The Fort St. Vrain Nuclear Generating Station (FSV), a large high-temperature gas-cooled reactor (HTGR) which operated from 1974 to 1989, used a deterministic source term based upon TID-14844, which was made conservative by assuming unrestricted core heat up, total loss of forced circulation, and a fission product release rate from the fuel that was much faster than experimental results had shown. At the time FSV was licensed, the trend of the radioactive source term for HTGR siting purposes was to use more conservative releases as the plant size increased.¹

In the late 1980s, the U.S. Department of Energy (DOE) sponsored work on an HTGR design known as the Modular High-Temperature Gas-Cooled Reactor (MHTGR). For this design, DOE had proposed a mechanistic (scenario-specific) source term that was based upon the characteristics of the fuel and plant to determine the magnitude, timing and nature of fission product release from the core. The proposed use of such a source term represented a major departure from both LWRs and earlier high-temperature gas-cooled reactor HTGR designs with respect to containment and siting evaluations. NRC staff stated in NUREG-1338, a 1989 draft safety evaluation report (SER) for the MHTGR, that final acceptance and use of a mechanistic source term was contingent on the satisfactory resolution of technical and policy considerations, and noted that extensive research and testing was needed to address the technical issues.

In SECY-93-092, the staff addressed the source term issue for the PRISM, the MHTGR, the PIUS, and the Canadian Deuterium-Uranium (CANDU 3) reactor designs and recommended to the Commission that mechanistic source terms should be allowed provided that:

¹Fisher, Orvis. *Licensing of HTGRs in the United States*. International Atomic Energy Agency, Vienna (Austria). International Working Group on Gas-Cooled Reactors. Specialists meeting on gas-cooled reactor safety and licensing aspects, Lausanne, Switzerland, 1-3 September 1980. Summary Report Jan 1981. IWGGCR-1, pp: 184-190.

- The reactor and fuel performance under normal and off-normal operating conditions is sufficiently well understood to permit a mechanistic analysis. Sufficient data should exist on the reactor and fuel performance through the research, development, and testing programs to provide the adequate confidence in the mechanistic approach.
- The transport of fission products can be adequately modeled for all barriers and pathways to the environs, including specific consideration of containment design. The calculations should be as realistic as possible so that the values and limitations of any mechanism or barrier are not obscured.
- The events considered in the analyses to develop the set of source terms for each design are selected to bound severe accidents and design-dependent uncertainties.

The Advisory Committee on Reactor Safeguards (ACRS) agreed with the staff, and stated in a letter dated February 19, 1993, to NRC Chairman Selin:

The staff proposal to base the source terms on mechanistic analyses appears reasonable, although it is clear that the present data base will need to be expanded. We note that the staff is now developing for LWRs a revision to the TID-14844 source term. It will be appropriate for the staff to consider using the newer approach when it develops source terms, and to take specific account of the unique features of each of the reactor types.

In a July 30, 1993, SRM, the Commission approved the staff's recommendations, stating: "The Commission approves the staff's recommendations including its agreement with the ACRS."

DISCUSSION:

The source term used in LWR siting evaluations is based on an in-vessel core melt accident in which a large portion of the radionuclide inventory is released early into containment. This may very well not be applicable to non-LWR designs. In the recent non-LWR pre-application activities, the designers propose an approach to source term selection similar to that proposed for the MHTGR for assessing the effectiveness of plant mitigation features or site suitability. This has resulted in proposals to use mechanistic scenario-specific source terms derived from anticipated operating occurrences (AOOs) and design basis events (DBEs) defined for the plant, using phenomenological models of fission product transport. Such mechanistic source terms that would take into account the predicted fuel and plant performance over a wide range of conditions would need to account for uncertainties. Accordingly, it should be emphasized that such an approach will require much research and testing to fully understand and develop technically sound mechanistic source terms. Alternatively, some may consider the use of a licensing source term representative of severe core damage as an element of defense-in-depth and, as such, it should be applied to all reactors.

Using mechanistic source terms based on a selection of design basis events and a good foundation of plant-specific knowledge, such as fuel behavior and core response under off-normal conditions, has been the trend for advanced non-LWRs worldwide. The HTR-10 in China used a source term based on a mechanistic approach in which severe core damage was

not arbitrarily postulated for the siting evaluation; instead the radiation release was calculated specifically for the individual accidents leading to the largest release of radionuclides from the fuel elements. Both Exelon and ESKOM proposed mechanistic source terms based on design basis events and predictions of fuel behavior for the pebble bed modular reactor (PBMR).

At a public workshop held on October 22–23, 2002, participants stated that although a bounding source term is used in the licensing of current LWRs, the regulations do allow for either approach. Workshop participants felt that the regulations should retain the flexibility of allowing the use of either a bounding or scenario-specific source terms. Additionally, some stated that there was no need to modify the Commission decision of 1993.

OPTIONS:

The options considered by the staff in addressing this issue are:

- (a) Use a deterministic bounding source term which is based on the conservative assumption of core damage and fission product release.

This option represents the current practice used on LWRs and is consistent with the view (i.e., defense-in-depth) that siting and containment decisions should be made in consideration of a severe core damage accident. In fact, some may consider the use of a source term representative of severe core damage as an element of defense-in-depth and, as such, it should be applied to all reactors. For non-LWRs this may be highly conservative, not accurately reflecting the design attributes of non-LWRs. Using such a source term could provide a disincentive for designers to develop high quality fuel and emphasize accident prevention.

- (b) Retain prior Commission guidance that allows scenario-specific (mechanistic) source terms based upon understanding fission product behavior and plant conditions and performance.

This option has the benefit of recognizing the unique features of the non-LWR designs. In addition, this option is consistent with previous Commission and ACRS decisions. However, in order to properly determine a mechanistic source term, a large amount of resources will need to be devoted to research and testing to understand plant performance, fuel behavior, and fission product release characteristics over a wide range of conditions.

RECOMMENDATION:

The staff recommends that the Commission take the following action:

- Retain the Commission's guidance contained in the July 30, 1993, SRM that allows the use of scenario-specific source terms, provided there is sufficient understanding and assurance of plant and fuel performance and deterministic engineering judgement is used to bound uncertainties.²

²Note: This represents a fundamental change in practice from that used on LWRs, in that the source term used for siting considerations may not be that associated with a core melt accident.

This recommendation is consistent with Option b above and, if approved by the Commission, would allow credit to be given for the unique aspects of plant design and build upon the scenario specific recommendation under Issue 4. Furthermore, this approach is consistent with prior Commission and ACRS views. However, this approach is also dependent upon understanding fuel and fission product behavior under a wide range of scenarios and on ensuring fuel and plant performance is maintained over the life of the plant. This approach is also very dependent on the event selection process.

For the purpose of siting and containment/confinement decisions, the staff recommends that conservative source terms for AOOs and DBEs be used. For emergency planning purposes a best estimate source term would be acceptable.

Containment vs. Confinement

ISSUE 6: Under what conditions can a plant be licensed without a pressure-retaining containment building (i.e., a confinement building instead of a containment)?

BACKGROUND:

In SECY-93-092, "Issues Pertaining to the Advanced Reactor (PRISM, MHTGR and PIUS) and CANDU 3 Designs and Their Relationship to Current Regulatory Requirements," April 8, 1993, the staff described the containment and confinement concepts for the reactor designs under consideration. In the case of the Modular High-Temperature Gas-Cooled Reactor (MHTGR), a confinement concept (negative pressure building with filtered exhaust, but no pressure-retaining capability) was proposed. This same concept has been incorporated into the pebble bed modular reactor (PBMR) and gas-turbine modular helium reactor (GT-MHR) designs, thus again raising this issue for Commission consideration. With a confinement building, large leaks from the primary coolant system would cause the confinement building to initially vent (unfiltered) to the atmosphere until the pressure was relieved and then either continue to relieve any additional pressure buildup (unfiltered) or resume negative pressure filtered operation for the duration of the accident, unless electric power was lost, in which case slow unfiltered leakage from the building would occur. In addition to filtering, fission product attenuation in the reactor or by other building structures can also retain fission products. A confinement building was used on Fort St. Vrain and many gas-cooled reactors in other countries. For HTGRs, the long core heatup time and the fission product retention capability of the fuel are considered by the designers to be sufficiently robust to make a conventional pressure-retaining containment unnecessary and possibly detrimental to the passive decay heat removal systems proposed.

In SECY-93-092 the staff addressed the MHTGR confinement concept and proposed functional performance criteria to evaluate the acceptability of proposed designs, rather than to rely exclusively on prescriptive containment design criteria. The proposed criteria were based upon ensuring that building performance is consistent with accident evaluation criteria. Specifically, the staff proposed that:

- Designs were to be adequate to meet the specified onsite and offsite radionuclide release limits for the event categories within their design envelope.
- For a period of approximately 24 hours following the onset of core damage,¹ the specified containment challenge event was to result in no greater than the limiting leak rate used in evaluation of the event categories, and structural stresses were to be maintained within acceptable limits (i.e., ASME level C requirements or equivalent). After this period, the uncontrolled release of radioactivity must be prevented.

In response to SECY-93-092, the Commission issued a staff requirements memorandum (SRM) on July 30, 1993, which approved the staff's recommendation. In addition, the Commission went on to state that "for the MHTGR, the staff should also address the following type of event. The loss of primary coolant pressure boundary integrity whereby air ingress

¹A core damage accident was to be postulated as a containment challenge event.

could occur (from the “chimney effect”) resulting in a graphite fire and the subsequent loss of integrity of the fuel particle coatings.”

DISCUSSION:

The use of a pressure-retaining containment building has been considered a key element of the basic design as well as a significant element of defense-in-depth for LWRs. It has been a traditional way to protect against unknowns or uncertainties that could lead to core damage and the uncontrolled release of large quantities of radioactive material. It is an especially important feature if extensive core damage can occur and result in the release of large quantities of radioactive material. It should be emphasized that the presence of a traditional containment building has been useful in protecting public health and safety (e.g., TMI-2) and has been emphasized as an important plant feature which can enhance public confidence. However, certain reactor designs may have a long delay before the release of large amounts of radioactive material or may preclude extensive core damage altogether. In such cases it is reasonable to consider whether or not a traditional pressure-retaining containment building is necessary or is the most effective way to protect public health and safety. In addition, physical protection of the reactor and critical systems, structures and components from external threats can be accomplished with non-pressure-retaining structures. Key considerations regarding the need for a pressure-retaining containment building include:

- Would the addition of such a feature substantially improve safety?
- Does the reactor design preclude or significantly delay core damage?
- Is there sufficient confidence in the reactor design such that a pressure-retaining containment is not needed to cover uncertainties?
- Is there sufficient confidence in the reactor design to eliminate from consideration severe core damage and the release of large quantities of fission products?
- Can criteria be developed that, if met, would provide sufficient confidence that a pressure-retaining containment is not needed?

At the public workshop held on October 22–23, 2002, there was no general consensus among participants regarding the need for a pressure-retaining containment building or the use of the criteria defined in SECY-93-092. Some participants felt a conventional approach to containment was necessary to account for uncertainties and to ensure public confidence, while others felt it could detract from safety, particularly for HTGRs, as well as impose a large unnecessary capital and operational cost.

OPTIONS :

The options considered by the staff in addressing this issue are:

- (a) Require all future reactors include in their design a low leakage containment building capable of retaining pressure.

This option would clearly establish traditional containment as a key element of defense-in-depth and provide a degree of protection against uncertainties in a plant's ability to prevent severe core damage, for example through the entrance of air into a hot HTGR core. This option would also reduce concern over uncertainties in fuel performance and fission product release over a wide range of accident scenarios as well as reduce any concern about a loss in fuel quality due to fabrication problems over the life of the plant. In addition, it would likely tend to have a positive effect on public confidence. However, this option could reduce the incentive for designers to emphasize accident and fuel damage prevention in their designs. Also, the addition of a traditional containment would add cost that may not be commensurate with the safety benefit. In fact, some have argued that the addition of a traditional containment building will have a negative impact on safety by reducing the reliability of decay heat removal systems (which in many designs are to be passive) and by causing the retention of hot, pressurized non-condensable gas in the building following a loss of coolant accident, thus providing a driving force for any fission products that might ultimately be released from the core during the course of the accident. Finally, this option is not consistent with the position taken in the Commission's July 30, 1993, SRM.

- (b) Allow a plant to be licensed without a containment building capable of retaining pressure, provided certain performance criteria are met.

This option would build upon the Commission's July 30, 1993, SRM, which approved the licensing of a design without a building capable of retaining pressure provided certain plant performance criteria could be met. These criteria were defined in SECY-93-092 (as discussed above). However, this option is limited to the fundamental issue of whether a plant can be licensed without a pressure retaining containment building. In view of the progress made since 1993 on risk-informed and performance-based regulation and the recommendations on the other issues in this paper, it may be appropriate to revisit the 1993 criteria after Commission guidance is received on the issues in this paper. Revisiting the 1993 criteria could be done as part of developing a policy or description of defense-in-depth, if approved by the Commission, or as part of developing a framework for future plant licensing.

This option would provide flexibility and incentive for designers to emphasize accident prevention and the prevention of core damage. As such, this option could improve safety if it results in designs with enhanced prevention characteristics. However, this option places emphasis on ensuring there is sufficient understanding of fuel and plant performance and uncertainties over a wide range of conditions and that the fuel fabrication process maintains fuel quality over the life of the plant. Therefore, in the absence of operating experience for new designs, this option would require robust research and development to confirm plant and fuel performance and to characterize uncertainties. This option may not, however, have a positive effect on public confidence.

RECOMMENDATION:

The staff recommends the Commission take the following action:

- Approve the use of functional performance requirements to establish the acceptability of a containment or confinement structure (i.e., a non-pressure retaining building may be acceptable provided the performance requirements can be met).
- If approved by the Commission, develop the functional performance requirements using as a starting point the guidance contained in the Commission's July 30, 1993, SRM and the Commission's guidance on the other issues contained in this paper.

This recommendation is consistent with Option b above and is coupled to the recommendations on Issues 4 and 5 discussed in Attachments 4 and 5, respectively. Similar to those recommendations it would represent a risk-informed and performance-based method to account for the unique aspects of each reactor design. In addition, resolution of this issue will establish a key element for incorporation into any policy or description of defense-in-depth.

If approved by the Commission, this recommendation could also permit a hybrid containment design, whereby pressure is vented early in the accident sequence and then a low leakage pressure-retaining configuration is established for the long term.

Emergency Preparedness

ISSUE 7: Under what conditions can the emergency planning zone (EPZ) be reduced, including a reduction to the site exclusion area boundary?

BACKGROUND:

Currently operating light-water reactors (LWRs) have a 10-mile plume exposure pathway EPZ and a 50-mile ingestion pathway EPZ as required by 10 CFR 50.47, "Emergency Plans." There have been instances in the past where smaller EPZs were approved (e.g., Ft. St. Vrain had a 5-mile plume exposure pathway EPZ) and the regulation (10 CFR 50.47) includes a provision allowing EPZs for High Temperature Gas-Cooled Reactors (HTGRs) to be determined on a case-by-case basis. In SECY-93-092, "Issues Pertaining to the Advanced Reactor (PRISM, MHGTR, and PIUS) and CANDU Designs and Their Relationship to Current Regulatory Requirements," dated April 8, 1993, the staff, at that time, proposed no changes to the existing regulations governing emergency preparedness (EP) for advanced reactor licensees. The staff proposed to provide regulatory direction at or before the start of the design certification phase so that any EP implications on design could be addressed. The staff viewed the inclusion of emergency preparedness by advanced reactor licensees as an essential element in NRC's "defense-in-depth" philosophy so that, even in the unlikely event of an offsite fission product release, there is reasonable assurance that emergency protective actions can be taken to protect the population around nuclear power plants.

The Commission, in its staff requirements memorandum (SRM) of July 30, 1993, stated that it was premature to reach a conclusion on emergency planning for advanced reactors and that for ongoing review purposes, the staff should use existing regulatory requirements. The SRM went on to say that the staff should remain open to suggestions to simplify the emergency planning requirements for reactors that are designed with greater safety margins, and that the work on EP should be closely correlated with work on accident evaluation and source term, in order to avoid unnecessary conservatism.

DISCUSSION:

Emergency preparedness is considered by many to be the last line of defense in the defense-in-depth philosophy. Its requirements have been established in consideration of the potential for accidents that could lead to severe core damage and the subsequent release of large amounts of radioactive material. For LWRs this release could occur in a matter of hours after the initiating event and a 10-mile plume exposure pathway EPZ has been chosen to envelope the distance beyond which it is very unlikely doses large enough to cause early fatalities would occur. In considering whether or not to modify the EPZ, similar considerations would need to be taken into account. These could include:

- What is the potential for a severe core damage accident?
- What is the potential for a large offsite release of radioactive material?

- Should the assumption of a large offsite release be a fundamental part of defense-in-depth?
- How should the characteristics of the release be used to set the EPZ (e.g., potential for early fatalities, timing of release)?
- How should uncertainties and experience with the design and technology be taken into account?

At the public workshop, held October 22–23, 2002, most participants felt that in the near term there was no urgency to change EP requirements for future plants. The reasons cited were that for HTGRs, 10 CFR 50.47 already contains a provision allowing the EPZs for HTGRs to be determined on a case-by-case basis and, in the near term, it is likely that new plants will be proposed using existing sites, which already have EP provisions in accordance with 10 CFR 50.47

OPTIONS:

The options considered by the staff in addressing this issue are:

- (a) No change from current requirements.

This option reflects the position of the Commission in its July 30, 1993, SRM. It recognizes the lack of experience with non-LWRs and, given the possible changes in the area of event selection, source term and containment discussed in Issues 4, 5, and 6, represents a step-by-step approach wherein changes to the final line of defense (EP) will not be made until there is experience with the new designs. In addition, 10 CFR 50.47 already contains a provision for HTGRs (the most likely non-LWRs in the near future) which allows a case-by-case determination for their EPZs. Thus, it can be argued that for HTGRs, no changes to current requirements are needed. Finally, in the near term it is likely that any new plants would be built on an existing site with an emergency plan, thus removing any urgency for changes in this area.

- (b) Allow a reduction in the EPZ based upon the extent and timing of predicted core damage and fission product release.

This option could be considered a follow-on to the position stated in the Commission's July 30, 1993, SRM directing the staff to remain open to suggestions to simplify the emergency planning requirements for reactors that are designed with greater safety margins. Under this option criteria would be developed that, if met, would allow reductions in the EPZ. Such criteria could be developed as part of development of a framework for future plant licensing. Such criteria could also provide guidance to implement the case-by-case provision for HTGRs in 10 CFR 50.47.

This option would be consistent with a risk-informed regulatory approach and with the risk-informed approaches recommended for event selection, source term and containment and could provide incentive for reactor designers to stress accident and core damage prevention. This option could lead to a reduction in the EPZ and, in the extreme, to the site exclusion area boundary.

RECOMMENDATION:

The staff recommends the Commission not modify EP requirements at this time. This recommendation is consistent with Option a above and is based upon the following two considerations::

- Provision already exists in 10 CFR 50.47 for accommodating the unique aspects of high-temperature gas reactors.
- In the near term, new plants are likely to be built on an existing site which conforms to current requirements.

If approved by the Commission, the role of emergency preparedness in defense-in-depth would be addressed as part of the development of a policy or description of defense-in-depth as recommended under Issue 2 above. In the longer term, if and when a need for change in emergency preparedness requirements is identified, that policy or description would serve as guidance in assessing the proposed change.

December 13, 2002

The Honorable Richard A. Meserve
Chairman
U. S. Nuclear Regulatory Commission
Washington, DC 20555-0001

Dear Chairman Meserve:

**SUBJECT: DRAFT COMMISSION PAPER ON POLICY ISSUES RELATED TO NON-LIGHT-
WATER REACTOR DESIGNS**

During the 498th meeting of the Advisory Committee on Reactor Safeguards, December 5-7, 2002, we met with representatives of the NRC's Office of Nuclear Regulatory Research (RES) to discuss the subject draft Commission paper. We also had the benefit of the referenced documents.

The RES staff has identified seven technical issues with policy implications that need resolution prior to certification reviews of advanced non-light-water reactor designs. The staff has also provided options for resolving those issues and has recommended the preferred options. We agree with the staff's recommended preferred options for each of the seven issues.

We commend the staff on its thoughtful study and look forward to further interactions on this subject.

Sincerely,

/RA/

George E. Apostolakis
Chairman

References:

- S Draft Predecisional SECY paper, undated, from William D. Travers, Executive Director for Operations, NRC, to the Commission, Subject: Policy Issues Related to Licensing Non-Light-Water Reactor Designs.
- S U. S. Nuclear Regulatory Commission, NUREG-1226, "Development and Utilization of the NRC Policy Statement on the Regulation of Advanced Nuclear Power Plants."

Workshop Summary—Key Points

A public workshop was held on October 22–23, 2002, to solicit stakeholders views on key issues that may have policy implications on licensing non-light water reactors. These issues are listed below along with the key points made by the workshop participants.

Expectation of Enhanced Safety

- Do not require enhanced safety.
- Follow ALWR certification model—applicant proposes enhanced safety features, staff reviews and supplements, if necessary, on a design-specific basis.
- Focus on developing the right metrics and criteria for non-LWRs.
- Reasonable to require additional oversight/testing on key areas where experience is less or uncertainty high.

Defense-in-Depth (DID)

- NRC should define DID.
- DID should include programmatic and process elements.
- Development of DID description should follow a process that includes stakeholder input.

Use of International Codes and Standards

- NRC should review codes and standards submitted in an application.
- NRC should utilize the experience of other regulatory bodies to facilitate the review of international codes and standards.
- NRC should be proactive in selected areas by participating in and endorsing international codes and standards that could fill a critical gap in the infrastructure for non-LWRs.
- NRC needs to be cognizant of State laws that affect codes and standards use.

Event Selection

- Define event categories probabilistically.
- Use PRA to identify events and select certain limiting events as design basis, considering uncertainties.
- Use risk to classify systems, structures, and components as safety related and replace the single-failure criterion with a reliability criterion.
- Develop a PRA standard to address how to perform a PRA for non-LWRs.

Source Term

- Severe core damage is the wrong metric for HTGR.
- Current regulations allow use of scenario-specific source terms.
- Develop an equivalent to the NUREG-1465 source term for HTGRs.

Containment vs. Confinement

- Focus on functional requirements rather than a prescriptive requirement that imposes a design solution.
- NEI's "Risk-Informed, Performance-Based White Paper" provides some functional requirements.
- Designs with limited operating experience should have containment.
- Requiring containment for GT-MHR may be adverse to safety.

Emergency Preparedness

- Since all planned Early Site Permits are for existing sites, this issue is not one that needs resolution in the near term.
- 10 CFR 50.47 includes provision for determining the EPZ for HTGRs on a case-by-case basis.