INTEGRATED RISK

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

BACKGROUND:

In SECY-03-0047, the staff recommended that the Commission 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).

In implementing the above, the staff also recommended:

- 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 should 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 new reactors.

In the June 26, 2003, staff requirements memorandum (SRM), the Commission approved the staff's recommendation on implementation of the Commission's expectation for enhanced safety in future non-light-water reactors, with the exception of accounting for the integrated risk posed by multiple reactors. The Commission requested that the staff provide further details on options for, and associated impacts of, requiring that modular reactor designs account for the integrated risk (i.e., cumulative effect on risk to the population around a site) posed by the use of multiple small reactors to equal the power output of one large reactor. These reactor modules generally would be located in close proximity to one another on a single site. The use of modular reactor designs is considered by some in the industry to be an attractive alternative to large single units because of potential inherent safety characteristics that are associated with some modular designs (e.g. passive decay heat removal) and potential economic advantages (e.g., increased use of factory fabrication and stepwise construction and operation bringing modules online as needed). Accordingly, the use of modular designs could result in a large number of reactors on a single site.

DISCUSSION:

Traditionally, it has been the staff's practice in making risk-informed decisions to consider risk on a per plant basis. This has been considered reasonable because of the limited number of plants on a site (maximum 3) and because of the low risk generally posed by currently operating plants, as indicated by staff and industry studies (e.g., NUREG-1150, Individual Plant Examination Program). However, it is recognized that the population around a site is exposed to the hazard of everything that is on that site. In promulgating the Safety Goal Policy in 1986 both the term "plant" and "site" were used. Whether this was intended to address integrated risk or not is not clear, but is a consideration with respect to how to treat integrated risk. Nevertheless, with the potential for modular reactors in the future it is appropriate to consider when and how (if at all) integrated risk should be addressed, since the number of reactors on a site could be significantly more than three.

The issue of how to treat modular reactors has also been raised in the context of certain legal and financial issues associated with new nuclear power plants. (In SECY-02-0180, "Legal and Financial Policy Issues Associated with Licensing New Nuclear Power Plants," dated October 7, 2002). In SECY-02-0180 the staff recognized that modular reactors may need to be treated differently in certain areas (e.g., number of licenses, financial protection) and indicated that the proposed Energy Bill Legislation, if approved by Congress, would amend the Atomic Energy Act to allow a combination of two or more reactor modules (each rated 100-300 Mwe) with a combined rated capacity of not more than 1300 Mwe to be considered one facility for the purposes of financial protection.

In SECY-03-0047, "Policy Issues Related to Licensing Non-Light Water Reactor Designs," the staff recommended and the Commission approved (in a June 26, 2003 SRM) a process for licensing future plants that parallels that used in the design certification of the evolutionary and advanced LWRs. This process is based upon the Commission's expectation that future reactor designs will be substantially safer than currently operating LWRs, will meet the Commission's Safety Goal Policy, and that the need for additional features to address uncertainties will be determined on a plant specific basis, with Commission approval. Accordingly, the addition of a single new reactor to a site with currently operating reactors would not add substantially to the overall risk. However, in making the recommendation in SECY-03-0047, the staff recognized that the addition of a modular reactor design to a site could add a large number of reactors to the site and thus recommended they be treated differently in that their integrated risk be considered. In its June 26, 2003, SRM, the Commission requested that the staff provide further details and options for this recommendation.

In response to the Commission's June 26, 2003 SRM, the staff has also reviewed previous dockets for sites where multiple reactors were approved to see if and how the issue of integrated risk was addressed. NRC has issued operating licenses to sites for three reactors (e.g., Palo Verde) and granted construction permits for four reactors at several sites (Shearon Harris, North Anna, Surry, Hartsville, and Vogtle). These construction permits were granted on the basis of preliminary safety evaluations and environmental impact statements. However, these preliminary safety evaluations and environmental impact statements did not consider the risk (individually or integrated) from accidents and, therefore, are not considered potential precedents. In all cases, the integrated affect of plant impacts on the environment from normal operation (e.g., thermal discharges, radiological releases from routine operation) were considered, but not the integrated risk from reactor accidents. In addition, in assessing the environmental impact of license renewal the staff developed a generic environmental impact statement (NUREG-1437) where the risk from reactor accidents was considered. However, the risk was considered on an individual reactor basis, not on an integrated site basis.

<u>OPTIONS</u>:

The staff indicated three options for considering integrated risk in licensing decisions for future modular reactors. Each option is evaluated with respect to its advantages, disadvantages, and impacts. In addressing integrated risk, risk associated with both accident prevention (e.g., core damage frequency[1]) and accident mitigation (e.g., large early release frequency[1]) were considered. A key factor in this consideration is reactor power level. Specifically, risk measures for accident prevention are considered to be independent of reactor power level (i.e., it is equally important to prevent core damage accidents in small reactors as it is in large reactors) whereas risk measures for accident mitigation may be dependent on reactor power level (i.e., the source term will vary).

It should also be noted that in assessing the risk from plants consisting of multiple reactor modules, the event sequences that contribute to risk will generally fall into two basic categories (1) those that affect each reactor module individually and (2) those that can affect two or more modules simultaneously (e.g., seismic events). Accordingly, the overall risk from a plant comprised of multiple reactor modules consists of the sum of the risk from both categories, and may be lower then the sum of the risk from all modules if they were treated separately, particularly if some systems are shared among reactor modules. This would be due to the fact that the risk from event sequences that affect all reactor modules simultaneously may not be equal among the reactor modules.

OPTION 1: No Consideration of Integrated Risk.

This option maintains the status quo. The risk information used in regulatory decisions on reactors (licensing, license amendments, or oversight) is developed and evaluated on a per reactor basis, not a per site basis. This approach has been judged acceptable for currently operating plants given that current sites in the U.S. have a relatively small number of reactors (up to 3) and many currently operating reactors achieve a level of safety comparable to that expressed in the Commission's Safety Goal Policy, thus ensuring their integrated risk is small. In the future, new reactor designs are expected to have significantly less risk (at least an order of magnitude based upon insights from reviews completed to date) than current operating reactors. If this expectation is realized, neither modular designs or large designs, would individually contribute significant additional risk to public health and safety. This option would not distinguish between large and small size reactors and would be reasonable if the number of modular reactors added to a site is limited, since this would serve to limit integrated risk. Also, it can be argued that uncertainties in risk assessments could be larger than the cumulative risk obtained by combining the risk from all reactor modules. However, since uncertainties are to be considered in risk-informed decisions this should not be a reason to ignore cumulative effects.

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[1]It should be noted that as part of work on a risk-informed process for future plant licensing, the staff is currently developing technology neutral risk metrics for accident prevention and mitigation, recognizing that core damage frequency and large early release frequency may not be appropriate for non-LWRs. In this regard, the use of Level 3 risk assessment is also being evaluated.

This option is consistent with the interpretation of the Commission's Safety Goal Policy that risk should be evaluated on an individual reactor basis. This option would also have minimal impact on current practices for risk-informing reactor regulatory requirements and activities (which assess risk on an individual reactor basis).

OPTION 2: Consideration of Integrated Risk (Frequency Only)

This option would require integrated risk to be considered in assessing all risk measures (prevention and mitigation) for future reactor licensing decisions regardless of reactor module size. In effect, it would require that the frequency associated with the risk criteria applied to large reactor designs be reduced for modular designs in proportion to the number of reactor modules needed to equal the output of a large reactor. This option would ensure that the integrated risk associated with accident prevention (e.g., core damage frequency) from modular reactors is no greater than the risk associated with accident prevention for a large reactor on a per Mw basis. It would not, however, recognize the effect of reactor power level on risk criteria associated with accident mitigation and would likely result in a de facto more stringent goal than intended by the Commission's Safety Goal Policy by not giving proper credit for reactor power level (i.e., source term) when assessing accident mitigation risk.

This option would broaden the frequency range of initiating events and event sequences which would have to be considered in a modular reactor risk assessment (as compared to a risk assessment for a large reactor). The reason for considering a broader frequency range would occur since lower frequency events and event sequences would need to be considered to ensure the lower frequency accident prevention and mitigation measures needed for each reactor module are adequately assessed. This option is consistent with an interpretation of the Commission's Safety Goal Policy that risk should be evaluated on a per site basis. This option would also require some change in current practices for risk informed activities when applied to modular reactors to account for integrated risk.

OPTION 3: Consideration of Integrated Risk (Reactor Power Level and Frequency)

This option recognizes that accident prevention is important regardless of reactor power level, whereas, in many cases accident mitigation has a relation to reactor power level (i.e., the lower the reactor power the fewer fission products available for release to the environment and thus the more difficult it is to have a large release). Given the non-linear response of early fatality health effects to dose, accounting for reactor power level, can make a large difference in the early fatality results. Accordingly, under this option the integrated risk associated with accident prevention risk criteria would need to be taken into account for modular reactor designs (similar to Option 2). However, the integrated risk associated with accident mitigation risk criteria could take into consideration reactor module size. This option would recognize the dependence of risk metrics associated with accident mitigation on reactor power level and would result in the integrated risk from multiple reactor modules being at least as low as the risk from an equivalent large reactor design. Therefore, this option would most realistically address integrated risk.

Like Option 2, this option would require that in assessing accident prevention, the risk assessment consider events and event sequences of low enough frequency to ensure that accident prevention measures can be adequately assessed. This option would also require that whatever accident mitigation risk measures are applied to modular reactors, they include consideration of reactor power and that some practices for risk informed activities would need to be modified to address integrated risk for modular reactors. In addition, this option represents an interpretation of the Commission's Safety Goal Policy that risk metrics associated with accident prevention and mitigation be assessed on a per site basis.

PROPOSED POSITION:

In evaluating the options, the staff primarily considered the two attributes: (1) which option is most consistent with treating risk in a realistic fashion and (2) which option best represents the level of safety intended in the Commission's Safety Goal Policy.

Regarding the first attribute, Option 1 is considered realistic and consistent with the Safety Goal Policy provided the Commission intends the safety expectations expressed by the safety goals to represent that associated with an individual reactor, not a site. It would also have the least impact on existing risk-informed practices. However, to be most realistic in assessing risk to the public, the integrated risk from all reactor modules on a site should be considered. Therefore, if the Commission intends the safety expectations expressed by the safety goals to represent that associated with a site, then Option 3 is most consistent with the Safety Goal Policy, since it treats risk in a realistic fashion by explicitly allowing reactor power to be considered in the assessment of risk measures related to accident mitigation while maintaining independence from reactor power in the assessment of accident prevention risk measures. Option 2 would result in an unrealistic assessment of risk related to accident mitigation measures, since it does not allow for consideration of reactor power (although it would ensure accident prevention is assessed in a realistic fashion).

Regarding the second attribute, either Option 1 or Option 3 could represent the level of safety intended in the Safety Goal Policy depending upon whether or not it is applied on an individual reactor or per site basis. Option 2 would likely result in a more stringent goal than intended by the Safety Goal Policy.

On this basis, the staff has developed a proposed position endorsing Option 3. Option 3 realistically accounts for modular reactor characteristics by treating accident prevention independent of reactor power, while allowing the assessment of accident mitigation risk measures to consider reactor power, thus not imposing a de facto more stringent goal than implied by the Safety Goal Policy. In addition, Option 3 would be most consistent with the proposed Energy Bill language that would allow a set of reactor modules to be treated as a single unit for the purposes of financial protection (i.e., the risk from the set of reactor modules should not exceed that from a single large reactor). Option 3 would result in staff treatment of the risk associated with modular reactors as follows:

- taking into consideration the integrated effect of risk when assessing accident prevention for modular reactor designs, independent of reactor power level, and

- taking into consideration the integrated effect of risk when assessing accident mitigation for modular reactor designs in a fashion that allows for consideration of the effect of reactor power level.

The staff is incorporating Option 3 into the framework and will solicit further comments on this option.

CONTAINMENT FUNCTIONAL PERFORMANCE REQUIREMENTS AND CRITERIA

ISSUE: Under what conditions can a plant be licensed without a pressure retaining containment?

BACKGROUND:

In SECY-03-0047, the staff recommended that the Commission approve the use of functional performance requirements to establish the acceptability of a containment (i.e., a non-pressure retaining building may be acceptable provided the performance requirements can be met). If approved by the Commission, the staff would 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 SECY-03-0047.

This recommendation is coupled to the recommendations on the issues regarding probabilistic approach 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 the issue on defense-in-depth above.

In the June 26, 2003, SRM, the Commission stated that there we insufficient information for the Commission to prejudge the best options and to make a decision on the viability of a confinement building (e.g., HTGRs). The Commission requested that the staff develop containment functional performance requirements and criteria working closely with industry experts (e.g., designers, Electric Power Research Institute, etc.) and other stakeholders regarding options in this area, taking into account such features as core, fuel, and cooling systems design for new plants. The staff was requested to pursue the development of containment functional performance standards and then submit options and recommendations to the Commission on this important policy decision.

DISCUSSION:

The functional performance requirements and criteria for containment[2] in protecting public health and safety vary significantly among new plant designs (e.g., high-temperature gas-cooled, liquid metal, molten salt, light water reactor). The functions of the containment are derived from the basic reactor-specific safety functions, such as controlling heat generation, removing heat, preventing chemical attack, and containing fission products. Differences in the

[2] There was no consensus among stakeholders on a single descriptive term such as "containment," "confinement," "vented low pressure containment," "reactor building" or "containment structure." Stakeholders indicated that each term implied a specific reactor technology with specific functions and specific functional performance requirements and criteria that were not necessarily applicable to every new reactor technology. However, regardless of the term, all "containment" designs provide or support accident prevention functions and accident mitigation functions. These functions are provided by a combination of civil structures (e.g., buildings) and systems. This paper uses the term "containment" the technology-neutral working term for all applicable functions. The paper identifies technology-neutral functions and develops technology-neutral functional performance requirements and criteria for the containment.

containment functional performance requirements and criteria also reflect differences in the integrated approach that designers take to optimize plant designs to meet risk objectives and safety requirements. For some reactor technologies, the fission product barrier function is not viewed by designers as among the most important safety functions of the containment.

The specific performance requirements and criteria that designers have developed for containment functions are also derived from and integrated with the requirements for other safety-related structures, systems, and components (SSCs) such as fuel, heat removal and coolant purification systems. Containment functional performance requirements and criteria for new plants are selected by designers with the intent of meeting NRC regulatory requirements and designer objectives. A containment may be described directly in terms of its derived safety functions, or indirectly in terms of the SSCs which carry out these functions. In general, the SSCs which perform the containment safety functions are physically located between the reactor pressure boundary and the environment.

The staff has developed options for the functional performance requirements and criteria for containment utilizing applicable Commission technical policies, informed by NRC and industry documents, foreign and domestic technical information, and stakeholder input. Stakeholder input includes feedback and comments from industry experts and other stakeholders received at public meetings conducted on November 19, 2003, January 14, 2004, and July 28, 2004. In addition, public input was received via letters:

- Nuclear Energy Institute, dated January 30 2004 and August 27, 2004
- Westinghouse, dated February 3, 2004,
- PBMR (Pty) Ltd., dated February 4, 2004 and,
- Framatome, dated August 20,2004.

These comments have been considered in developing and assessing containment functions, the options for containment functional performance requirements and criteria, the evaluation of the pros and cons for these options and the identification of the recommended option (i.e., criterion). The staff also met with the ACRS on April 15, October 13, and December 3, 2004 on this issue and their views have been used in developing and finalizing the options for containment performance requirements and criteria and assessing the impacts of the pros and cons of each option.

Applicable Commission policy guidance includes:

- 1986 Policy Statement on Safety Goals for the Operation of Nuclear Power Plants,

- 1985 Policy Statement on Severe Reactor Accidents Regarding Future Designs and Existing Plants,

- July 30, 1993, SRM (ADAMS Accession No. ML003760774) for SECY-93-0092, “Issues Pertaining to the Advanced Reactor (PRISM-Power Reactor Innovative Small Module, MHTGR- Modular HTGR, and PIUS-Process Inherent Ultimate Safety) and CANDU 3 Designs and Their Relationship to Current Regulatory Requirements” (ADAMS Accession No. ML040210725),

- 1994 Policy Statement on the Regulation of Advanced Nuclear Power Plants,

- 1995 Policy Statement on the Use of Probabilistic Risk Assessment Methods in Nuclear Regulatory Activities,

- March 11, 1999, White Paper on Risk-informed and Performance-Based Regulation,

- June 26, 2003, SRM for SECY-03-0047, “Policy Issues Related to Licensing Non-Light Water Reactor Designs, and

- June 26, 1990, SRM for SECY-90-016, “Evolutionary Light Water Reactor (LWR) Certification Issues and Their Relationship to Current Regulatory Requirements.”

In SECY-93-0092, the staff addressed the confinement concept for modular HTGRs by recommending that the acceptability of proposed containment designs be evaluated against a functional performance standard rather than a prescriptive criterion. Specifically, the staff proposed that containment designs must be adequate to meet the specified onsite and offsite radionuclide release limits for the event categories within their design envelope. The Commission’s July 30, 1993, SRM in response to SECY-93-0092 approved the staff’s recommendation. In addition, the Commission stated that for the MHTGR, the staff should also address the loss of primary coolant pressure boundary integrity whereby air ingress could occur (from the “chimney effect”) resulting in core graphite oxidation and loss of fuel particle integrity.

NRC and industry documents used in developing and assessing the options include NRC regulations for light-water reactors, non-LWR pre-application design and safety analysis information and related documents, associated staff preliminary safety assessments, and U.S. Department of Energy and national laboratory information related to Generation IV plant design concepts. Design, safety analysis and safety review documents related to the vented low pressure containment systems for the Savannah River Plant and the N-Reactor Plant were also reviewed as were selected documents and information pertaining to the containment design and safety basis for foreign HTGRs (e.g., the Japan Atomic Energy Research Institute High Temperature Engineering Test Reactor and Gas Turbine High Temperature Reactor 300, the Peoples Republic of China, Institute of Nuclear Engineering and New Energy Technology HTR-10 reactor) and the Toshiba liquid metal reactor.

The results from these reviews have resulted in the staff identifying functions, requirements and criteria for these functions. Metrics were developed to evaluate the requirements and criteria. This assessment and the results of the assessment (i.e., staff recommendation) are described in detail below.

Containment Functions and Technology Neutral Performance Requirements

The staff has concluded that the function of containment designs, includes a direct or support functional role for the following accident prevention and accident mitigation safety functions:

1. Protect risk-significant SSCs from internal and external events
2. Physically support risk-significant SSCs
3. Protect onsite workers from radiation
4. Remove heat to prevent risk-significant SSCs from exceeding design or safety limits

5. Provide physical protection (i.e., security) for risk-significant SSCs
6. Reduce radionuclide releases to the environs (including limiting core damage)

The policy issue raised to Commission regarding containment performance is directly related to the function on reducing radionuclide releases to the environs (i.e., Function 6). The other functions (1 thru 5), while they need to be considered in the design and construction, are not associated with the policy issue raised to the Commission and are addressed in the framework. Functions 5 is to be addressed in a separate paper. Therefore, the staff evaluation only focuses on Function 6, "reduce radionuclide releases to the environs."

For Function 6, the following technology-neutral performance requirement is proposed:

- The containment must be adequate to reduce radionuclide releases to the environs to ensure that doses do not exceed the dose criteria for the selected events in the event categories.

Approach for Developing Options for Functional Performance Criteria

The containment functional requirement stated above and the criteria (i.e., the options) discussed below for reducing radioactive material release to the environs have been developed to be consistent with the proposed regulatory structure for new plant licensing. The approach used to ensure this consistency includes the following:

- The containment supports meeting the overall plant risk criteria, which includes accident prevention criteria and accident mitigation criteria.

- A probabilistic approach may be used to identify events which must be considered in the design. Frequency-based categories are established for: normal operation and anticipated operational occurrences; design-basis events; and events beyond the design-basis. Design-specific PRA information, including consideration of uncertainty, is used to categorize the event sequences. This approach requires that the probabilistic information that supports event categorization is adequate and acceptable. Additionally, in categorizing events, deterministic engineering judgement may be used to ensure that uncertainties associated with event probabilities are adequately treated. A set of events from the design-basis accident category is selected on a deterministic basis as scenarios that most severely challenge the containment to meet the dose criteria and are used for assessing site suitability. The actual events selected for the design-basis are determined at the time of the staff review of a particular plant design.

- An event frequency versus event dose consequence limit curve is used. For the events selected for the design-basis category, the dose consequence limit curve provides that the offsite dose does not exceed the limits specified in 10 CFR100 and 10 CFR50.34 (a) (1).

- For each of the selected events in each of the event categories, the source terms used to assess radionuclide releases into and out of the containment may be calculated on a mechanistic basis. That is, the radionuclides released into containment, and radionuclide release out of the containment to the environs, takes credit for the reactor, fuel, core and containment characteristics (i.e., accident response), including radionuclide retention and attenuation characteristics of each of the multiple mechanistic barriers and obstacles to radionuclide transport. The use of a mechanistic approach requires sufficient quantitative understanding and assurance of both design-specific plant system performance (including radionuclide transport behavior) and fuel system performance (including radionuclide transport behavior) to adequately model all pathways, barriers and obstacles to the environs. Adequate data is required to provide the quantitative basis for the performance of each of the mechanistic barriers and obstacles for the range of plant conditions associated with the selected events in each category. This quantitative basis must utilize either existing applicable data or a suitable technology development program. Deterministic engineering judgement is applied to ensure that the (technology-specific) calculated source term for each event selected is consistent with the guidance provided for the use of scenario specific source terms.

- Events selected for deterministic analysis from the containment design-basis category are analyzed using best estimate methods, including uncertainty analysis. The results of the best estimate analysis are compared with the dose acceptance criteria and must be shown to meet it at the 95% confidence level. Bounding calculations may also be performed. Events beyond the design-basis are analyzed in the PRA, including uncertainty analysis, and the mean value is compared with the overall plant risk acceptance criteria.

- Defense-in-depth is applied to ensure that compensatory measures are in place to prevent and mitigate accidents and to address both random (stochastic) uncertainties and state of knowledge (i.e., completeness) uncertainties. The application of defense-in-depth for developing the performance requirement and criteria of the containment for radioactive releases to the environs is based on the following principles and model:

  – The design should provide for the prevention and mitigation of accidents

  – Safety functions (e.g., control of fission product release, control of chemical attack on core components) should not depend on a single element of design, construction or operation

  – Uncertainties in the performance of risk-significant structures, systems and components and the performance of humans should be accounted for

  – Defense-on-depth should be a combination of: (1) a probabilistic element to account for model and parameter uncertainties; (2) a deterministic element to account for completeness uncertainties (unknowns).

Metrics for Evaluating the Options

To qualitatively evaluate the options for containment functional performance criteria for reducing radionuclide releases to the environs, the staff used the following metrics:

- Does the option (i.e., criteria) adequately accommodate all containment functions (i.e., are there potential adverse effects on plant safety, event consequences, or other functions of the containment)?

- Would the option be expected to substantially improve plant safety by

  - preventing certain types of accidents?
  - significantly reducing fission product release to the environs?
  - addressing known uncertainties?

- Does the option account for plant risk (e.g., is it risk-informed, does it consider unknowns due to lack of knowledge)?

- Does the option provide flexibility to the designer in meeting the event consequence acceptance criteria (e.g., could it discourage design innovation or accident prevention)?

In addition, the staff considered each option from the following perspectives:

- Is it technology-neutral and performance-based?

- How does it relate to the designer-proposed criteria for prospective new plants?

- Would the criteria involve significant incremental costs without commensurate safety benefits?

Options for Reducing Radionuclide Releases to the Environs

Four policy options (i.e., criteria) for containment performance for reducing radioactive releases to the environs have been developed for use in the proposed regulatory structure for new plant licensing:

Option 1: The containment must adequately reduce radionuclide releases to the environs to meet the onsite and offsite radionuclide dose acceptance criteria for the events selected for the event categories.

Option 2: The containment must adequately reduce radionuclide releases to the environs to meet the onsite and offsite radionuclide dose acceptance criteria for the events selected for the event categories (including within the design-basis category, selected credible events having the potential for high consequence source terms).

Option 3: The containment must adequately reduce radionuclide releases to the environs to meet the onsite and offsite radionuclide dose acceptance criteria for the events selected for the event categories (including within the design-basis category, selected credible events having the potential for high consequence source terms) and have the capability to establish controlled leakage and controlled release of delayed accident source term radionuclides.

Option 4: The containment must adequately reduce radionuclide releases to the environs to meet the onsite and offsite radionuclide dose acceptance criteria for the events selected for the event categories (including within the design-basis category, selected credible events having the potential for high consequence source terms) by being essentially leak tight against the release of prompt and delayed accident source term radionuclides.

Evaluation of Each Option

Option 1: The containment must adequately reduce radionuclide releases to the environs to meet the onsite and offsite radionuclide dose acceptance criteria for the events selected for the event categories.

This performance criterion represents a significant departure from the prescriptive standard of conventional LWR containment building system designs for independently reducing radioactive release to the environs. The performance required of the containment would be dependant on how effectively the other mechanistic barriers (e.g., fuel and core barriers) performed. This option would provide a very broad application of the Commission guidance in the SRM for SECY-03-0047 for the issue on the use of PRA and the issue on the use of a mechanistic source term and Commission guidance on the application of performance-based principles. It would provide a structured, consistent, technology-neutral and performance-based process to establish the acceptability of containment designs.

The staff would review each proposed containment design, including the events selected for treatment as design-basis accidents. Deterministic engineering judgement would be applied to include, as needed, additional core challenge scenarios that challenge the containment to meet the dose criteria. Additionally, deterministic engineering judgement would be applied to ensure that the calculation of the source term for each event selected is bounded. If available, technology-specific regulatory guides would be used to provide guidance to designers and staff on the selection of events and source term calculation, including the application of deterministic engineering judgement in both areas. Staff recommendations would be made, as needed, for additional events and the source term calculation and/or enhancements to the containment to address areas of high uncertainty. Enhancements would be subject to Commission endorsement. The level of defense-in-depth provided by the containment to address uncertainties, in the fuel, plant-system performance and event sequences, would be tied to the severity of the selected events and attendant source terms included in the licensing basis.

This option would not be expected to adversely affect safety and would provide the designer with significant flexibility in developing new reactor concepts and in meeting the acceptance criteria for event consequences. It is technology neutral and consistent with the basis for the containment performance proposed by new plant (e.g., modular HTGR) designers.

With this option, the Commission would provide the designer and the staff with discretion in applying deterministic engineering judgement to supplement probabilistic information for the selection of events to be included in the containment design-basis. It would also continue to encourage accident prevention and provide significant flexibility in allowing alternative mitigation approaches, including those designs that can take advantage of significant event response times for human actions. It would not explicitly require that the containment to have additional capability (i.e. margin) to reduce radionuclide releases for unexpected events.

Depending on their nature, additional events included in the design-basis (or the emergency planning basis), this option might require further technology development (i.e.,costs) to support the mechanistic source term calculations for these events. Depending on any needed design enhancements, this option might also involve incremental design-related costs.

Because some reactor designs (e.g., HTGRs) are expected to involve a much lower fission product release into the containment for frequency-based design-basis events, these designs could result in enhanced public confidence. However, because this option could allow a containment to have less capability to reduce fission product release to the environs compared to a conventional LWR containment design for some technologies, it might be perceived as providing less defense-in-depth to compensate for uncertainties, thereby potentially reducing public confidence overall.

Option 2: The containment must adequately reduce radionuclide releases to the environs to meet the onsite and offsite radionuclide dose acceptance criteria for the events selected for the event categories (including within the design-basis category, selected credible events having the potential for high consequence source terms).

This criterion is the same as Option 1 except that it specifically requires that selected low probability, but credible events, with the potential for a large source term and a significant radionuclide release to the environs, be included in the design-basis event category. Such events would be included in order to challenge the capability of the mechanistic barriers, including the capability of the containment, to reduce radionuclides sufficiently to meet dose criteria. Such bounding design-specific core challenge events would be selected even if their frequency (including uncertainties) fell below the lower cutoff for the design-basis event category, or potentially even below the lower cutoff frequency for the beyond the design-basis event category. The selected design-specific events, referred to as “cliff-edge” events because of their potential for a steeply increased source term, would be included as credible design-basis “core challenge” events. These core challenge events would be included to assess the adequacy of the mechanistic barriers, including the containment, in meeting the limits specified in 10 CFR100 and 10 CFR50.34 (a) (1). This option would demonstrate or require that significant additional margin is available to reduce radioactive releases to compensate for uncertainties, including completeness uncertainties, which might otherwise result in a significant increase in dose. If a reduction in radionuclide releases to the environs were necessary to meet dose limits for the core challenge events, cost-effective containment design improvements would be targeted, although some reductions, through enhancements in the performance of other mechanistic barriers or SSCs, or other mitigation strategies could be considered. This option would also provide a structured, consistent, technology-neutral and performance-based process to establish the acceptability of containment designs.

This option would ensure or require that significant margin is provided by the mechanistic barriers, including the containment, to address source term uncertainties, due to uncertainties in fuel or plant-system performance, event sequences and event frequencies.

This option is consistent with the event selection approach and conservative treatment of events (in the containment design-basis) approved by the Commission in the SRM for SECY-03-0047 for the issue on the use of PRA, but would add a requirement that higher consequence events of potentially very low probability be included in the design-basis. This option is consistent with traditional bounding approach to LWR siting source term analyses and comparable to the direction provided by the Commission in its SRM for SECY-93-0092 for the MHTGR. Special treatment requirements (e.g., quality assurance, maintenance, testing) for any additional required containment or other SSC enhancements that would be needed to meet the limits specified in 10 CFR100 and 10 CFR50.34 (a) (1) would follow the approach of the proposed regulatory structure for new plant licensing.

This option would not be expected to adversely affect safety or other containment functions and would give the designer flexibility in developing new reactor concepts. Targeting the containment for any needed improvements to meet the dose criteria is consistent with the defense-in-depth philosophy but could discourage the use of alternative prevention or mitigation strategies. It is technology neutral, but including credible bounding cliff-edge events in the design-basis may not be consistent with the approach taken for establishing containment performance proposed by all designers of all new plants (e.g., modular HTGRs). It may, or may not be considered risk-informed for new plant designs having relatively limited operational experience and PRA experience.

The inclusion of “cliff-edge” events in the design-basis could require additional technology development to support the source term calculations for these events. Also, depending on the analysis results, this option might require design-related enhancements involving incremental costs. Including more challenging and lower probability events in the containment design-basis would likely increase public confidence relative to Option 1.

Option 3: The containment must adequately reduce radionuclide releases to the environs to meet the onsite and offsite radionuclide dose acceptance criteria for the events selected for the event categories (including within the design-basis category, selected credible events having the potential for high consequence source terms) and have the capability to establish controlled leakage and controlled release of delayed accident source term radionuclides.

This option is the same as Option 1, but includes the prescriptive requirement that the containment have the capability to establish a controlled leakage and a controlled radionuclide release capability. This capability ensures that the containment provides a significant deterministic element of defense-in-depth to controlling radioactive releases, should the other mechanistic barriers and obstacles to fission product transport provided by the fuel, core materials and reactor coolant system not perform as expected or should unanticipated events involving a larger than expected accident source term occur. This element is independent of the performance of the other mechanistic barriers and, for some designs, also has the potential to prevent or mitigate certain kinds of accidents (e.g., HTGR air ingress) .

This option would reduce concerns related to maintaining fuel quality and fuel performance during normal operation and accidents over the life of the plant. However, by requiring the containment to have an additional capability to reduce releases, it might reduce the incentive to emphasize accident prevention in designs, thereby potentially having an adverse affect on plant safety.

Since this criterion involves a prescriptive element, it is not totally performance-based. This option also goes beyond the containment functional performance criteria that is being proposed for selected new plant designs (i.e., HTGRs)

Compared to Option 1, this option could add to the cost of the containment. It would also differ from the prior Commission decision on containment performance requirements documented in the SRM for SECY-93-0092 by requiring additional mitigation capability regardless of meeting onsite and offsite dose performance criteria. However, It would likely further increase public confidence compared to Options 1 or 2.

Option 4: The containment must adequately reduce radionuclide releases to the environs to meet the onsite and offsite radionuclide dose acceptance criteria for the events selected for the event categories (including within the design-basis category, selected credible events having the potential for high consequence source terms) by being essentially leak tight against the release of prompt and delayed accident source term radionuclides.

This option requires containment designs to have the same prescriptive functional performance criteria as conventional LWR containment designs (i.e., essentially leak-tight against the release of radionuclides to the environs). It provides a significant deterministic element of defense-in-depth to reduce radioactive releases to the environs should the other fission product transport mechanistic barriers and obstacles associated with fuel, core and reactor coolant system not perform as expected or should a severe cliff-edge event occur. This option would significantly limit the benefits of the Commission guidance in the SRM for SECY-03-0047 on the use of PRA and the issue on the use of a mechanistic source term and would be contrary to Commission guidance on the use of performance-based principles.

This deterministic element of defense-in-depth is independent of the other mechanistic barriers and, for some designs would prevent or mitigate certain kinds of accidents (e.g., HTGR air ingress). However, for certain plant designs (e.g., HTGRs), it would likely discourage accident prevention (e.g., fuel performance) and could adversely affect plant safety (e.g., degraded heat removal function, sustained motive force for delayed source term radionuclide transport). It would also generally impact designer flexibility in developing new reactor concepts and in meeting dose criteria. This option is not supported by industry as a standard requirement for containment for all new reactor concepts, and is not consistent with the containment performance proposed by certain new reactor designs (i.e., HTGRs).

This option is neither consistent with the position taken in the Commission’s July 30, 1993, SRM, nor the Commission’s advanced reactor policy, which states that regulatory guidance must be sufficiently general to avoid placing unnecessary constraints on the development of new design concepts. Compared to Options 1, 2 and 3, this option would add significantly to the cost of certain reactor designs which may not be commensurate with the safety benefits. HTGR designers state that this option would make HTGR plant designs uneconomical. For certain designs (i.e., HTGRs) this option would likely result in higher public confidence than Options 1, 2, or 3.

<u>PROPOSED POSITION</u>:

With respect to the containment functional performance requirement for reducing radionuclide releases to the environs, the staff proposes the following technology-neutral requirement:

- The containment be adequate to reduce radionuclide releases to the environs to ensure that doses do not exceed the dose criteria for the selected events in the event categories.

With respect to the containment functional performance criteria for reducing radionuclide releases to the environs, the staff proposes Option 3 as the technology-neutral criteria:

- The containment must adequately reduce radionuclide releases to the environs to meet the onsite and offsite radionuclide dose acceptance criteria for the events selected for the event categories (including within the design-basis category, selected credible events having the potential for high consequence source terms) and have the capability to establish controlled leakage and controlled release of delayed accident source term radionuclides.

Option 3 would require that the containment have an independent capability to reduce delayed radionuclide releases to the environment independent of other radionuclide transport barriers associated with the fuel, core and reactor coolant pressure boundary. This is consistent with the Commission's defense-in-depth philosophy which provides that safety functions (e.g., control of fission product release) should not depend on a single element of design, construction or operation. Resolution of this issue will also establish a key element of the policy description of defense-in-depth.

The staff is incorporating Option 3 into the framework and will solicit further comments on this option within the context of the framework.

LEVEL OF SAFETY

ISSUE: What level of safety should be the goal for the technology-neutral requirements to achieve?

BACKGROUND:

In SECY-03-0047, the staff recommended and the Commission approved a process to achieve enhanced safety on new reactors similar to that used in the evolutionary LWR and advanced light-water reactor (ALWR) design certification reviews (i.e., reactor designers were expected to propose designs with enhanced safety characteristics and address the requirements in SECY-93-087 ("Policy, Technical, and Licensing Issues Pertaining to Evolutionary and Advanced Light-Water Reactor Designs," dated July 21, 1993, ML003708021). Such enhancements could include additional design features, additional testing by the designed, or additional confirmatory testing and 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.

Such a process is appropriate if future designs are to be licensed using 10CFR50, where case-by-case determinations are made regarding the applicability of requirements of the design and the need for additional requirements to account for the unique aspects of the design, including uncertainties. However, in developing a new structure for new plant licensing, a safety target is needed to guide the development of the requirements (i.e., what level of safety is the goal to be achieved?).

DISCUSSION:

In the development of the framework associated with the structure for new plant licensing, the staff has chosen as a target level of safety the Quantitative Health Objectives (QHOs) as expressed in the Commission Safety Goal Policy. The staff considers this selection consistent with the Commission expectations, as expressed in the Advanced Reactor Policy Statement, where it was stated that the Commission "expects that advanced reactor designs will comply with the Commission's Safety Goal Policy/" Selecting the QHOs as the target level of safety provides a foundation for the development of risk-informed, technology-neutral requirements that can be applied to any new design.

Use of the QHOs as the safety target is not considered a more stringent requirement on the industry since this same target has been used by the industry as a target in their own design and regulatory initiatives (e.g., NEI-02-02, "A Risk-Informed, Performance-Based Framework for Power Reactors," dated May 2002). In addition, many of the currently operating LWRs are considered to meet the level of safety expressed by the QHOs. Use of the QHOs as the goal for the level of safety to be achieved also provides for margin above adequate protection to account for uncertainties and variations in plant performance.

At the present time, the staff plans to solicit stakeholder input on this issue and then develop a final recommendation for Commission consideration.

DEFINITION OF DEFENSE-IN-DEPTH

ISSUE: How to specify "defense-in-depth" for non-light-water reactors (non-LWRs), (Should a description be developed?)

BACKGROUND:

In SECY-03-0047, "Policy Issues Related to Licensing Non-Light-Water Reactor Designs," dated March 28, 2003 (ML030160002), with respect to defense-in-depth, the staff recommended 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.

In the June 26, 2003, staff requirements memorandum (SRM), the Commission approved development of a description of defense-in-depth for incorporation into the policy statement on the use of probabilistic risk assessment (PRA).

DISCUSSION:

The concept of defense-in-depth is fundamental to the NRC's safety philosophy that there must be adequate measures to deal with uncertainty. Regulatory Guide 1.174 ("An Approach for Using Probabilistic Risk Assessment in Risk-Informed Decisions on Plant-Specific Changes to the Licensing Basis," Revision 1, November 2002, ML020810773) states: "The defense in depth philosophy ......has been and continues to be an effective way to account for uncertainties in equipment and human performance." In the Commission's Strategic Plan for FY 2004-20 defense-in depth is described as "an element of the NRC's Safety Philosophy that employs successive compensatory measures to prevent accidents or lessen the effects of damage if a malfunction or accident occurs at a nuclear facility. The NRC's Safety Philosophy ensures that the public is adequately protected and that emergency plans surrounding a nuclear facility are well conceived and will work. Moreover, the philosophy ensures that safety will not be wholly dependent on any single element of the design, construction, maintenance, or operation of a nuclear facility." On a number of occasions, the Advisory Committee on Reactor Safeguards (ACRS) examined defense-in-depth as a means of dealing with uncertainty.

A summary of the objectives of defense-in-depth can be stated as the ability to:

- compensate for potential adverse human actions (this includes commission as well as omission) and component failures,

- maintain the effectiveness of barriers by averting damage to the plant and the barriers themselves, and

- protect the public and environment from harm in the event that these barriers are not fully effective.

The staff's current approach in the technology-neutral framework for specifying defense-in-depth has three key elements: (1) development of defense-in-depth principles, (2) development of a defense-in-depth model for application, and (3) guidance on the implementation of the defense-in-depth model.

Defense-in-Depth Principles

To achieve the defense-in-depth objectives, and therefore assure public safety despite uncertainties, the staff is proposing some fundamental principles. The first principle requires that *measures against intentional as well as inadvertent events are provided***.** This is intended to ensure that in the application of defense-in-depth human initiated (e.g., security), as well as random events and natural phenomena, are considered.

From the first principle of defense-in-depth, the staff is developing four more defense-in-depth principles.

The design should provide accident prevention and mitigation capability. Accident prevention and mitigation capability should be provided such that there is no undue emphasis on either' at the expense of the other, for maintaining the plant in a safe condition given various challenges. Specific measures are sometimes seen as either preventive or mitigative depending on the point in the event sequence and the point of view of the observer. Often prevention is emphasized relative to mitigation because preventive measures are usually more economical, prevention avoids having to deal with the phenomenological uncertainties that arise once an accident progresses, etc. From a defense-in-depth standpoint such an emphasis is acceptable as long as it does not result in an exclusive reliance on prevention with a neglect of mitigative features.

Accomplishment of key safety functions should not be dependent upon a single element of design, construction, maintenance or operation. Redundancy, diversity, and independence in structures, systems, and components (SSCs) and actions will ensure that no key safety functions will be dependent on a single element (i.e., SSC or action) of design, construction, maintenance or operation. The key safety functions include: control of reactivity, removal of decay heat, and the functionality of physical barriers to contain the release of radioactive materials. In addition, hazards such as fire, flooding, and seismic events which have the potential to defeat redundancy, diversity, and independence, need to be considered.

Uncertainties in SSCs and human performance should be accounted for such that reliability and risk goals can be met. Allocation of risk goals for a new design must include uncertainty. The setting of success criteria for the achievement of safety functions should be set, and the calculations that show they have been met should be performed, in such a way that uncertainties are accounted for with a high level of confidence. For future reactors this needs to be accomplished without the benefit of reviewing past performance. The role of safety margins is important here in achieving a robust design. Both physical and temporal margins should be incorporated in the plant equipment and procedures.

Plants should be sited in areas that meet the intent of Part 100 and are consistent with the principles for siting established in Regulatory Guide 4.7 ("General Site Suitability Criteria for Nuclear Power Plants). The location of regulated facilities should be chosen so as to serve the protection of public health and safety. Consideration of population densities and the proximity of natural and man-made hazards in the siting of plants can provide further assurance that hazards to the public are minimized. For reactors, this principle is also intended to ensure that accident management, including emergency preparedness, remains a fundamental element of defense-in-depth. However, the staff recognizes that the scope and nature of offsite emergency preparedness activities could be different for future reactors, due to factors such as reactor size (i.e., power level), location, level of safety (i.e., likelihood of release), magnitude and chemical form of the radionuclide release, and timing of releases (i.e., long-term response).

Defense-in-Depth Model

The model of defense-in-depth which the staff is recommending for application to new reactors incorporates both deterministic and probabilistic elements. The deterministic part of the model mainly addresses completeness uncertainties by asking the question, "What if this barrier or safety feature fails?" without relying on a quantitative estimate of the likelihood of such a failure. As a result, the deterministic element is defined by protective strategies that are successive measures designed to protect public health and safety even if some of the strategies fail. The protective strategies of the technology-neutral framework are to ensure Physical Protection, maintain Barrier Integrity, limit Initiating Event Frequencies, assure adequate reliability of Protective Systems, and provide Accident Management. In addition, the deterministic element imposes specific qualitative requirements to be included in the regulations to ensure that the accomplishment of key safety functions are not dependent upon a single element of plant design construction, maintenance or operation.

The probabilistic part of the model seeks to evaluate the uncertainties in the analysis and to determine what steps should be taken to compensate for those uncertainties. The probabilistic elements address primarily modeling and parameter uncertainties, and establish specific quantitative performance goals, such as equipment reliability goals, that compensate for the calculated uncertainty.

The staff's defense-in-depth model uses a deterministic approach at a high level by requiring that all the protective strategies are included. Within each protective strategy a probabilistic approach is used to determine how much defense-in-depth is needed to achieve the desired quantitative goals on initiating event frequency and safety system reliability, including uncertainty.

Implementation of the Defense-in-Depth

The staff's approach for implementation of the above model relies on the application of the defense-in-depth principles as qualitative criteria to be adhered to, and the use of a PRA for achieving quantitative risk goals. Inclusion of all the protective strategies assures some protection against completeness uncertainty. Within each strategy, a probabilistic defense-in-depth element is applied to ensure adequate performance in meeting the objective of the strategy. The systems, barriers and actions used in the performance of the safety functions associated with the protective strategy are examined in terms of deterministic and probabilistic elements of defense-in-depth. Quantitative risk information is be used, where possible, to assess the degree of conformance and the need for additional defense-in-depth measures (e.g., redundancy, diversity, safety margins).

Monitoring and feedback are essential aspects of this process, since the validity of initial design assumptions, and of design changes made as part of the outlined steps, will be established by the actual operation of the reactor. Additional hardware or procedural changes may result from this feedback. This is especially important for the new and innovative designs for which there is no operating experience.

The staff envisions whole process of applying defense-in-depth as an iterative process, a series of steps, that is expected to be used initially by the designer and ultimately by the designer and regulator to develop the emerging design. As the design evolves the PRA will also be able to be developed to greater detail.

PROBABILISTIC APPROACH FOR ESTABLISHING THE LICENSING BASIS

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

BACKGROUND:

In SECY-03-0047, the staff recommended that the Commission take the following actions with respect to using a probabilistic approach to establish the licensing basis

- 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 identifying 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.

- Allow a probabilistic approach for the safety classification of structures, systems, and components.

- Replace the single failure criterion with a probabilistic (reliability) criterion.

These recommendations are consistent with a risk-informed approach. The recommendation expands the use of PRA into forming part of the basis for licensing and thus put greater emphasis on PRA quality, completeness, and documentation.

In the June 26, 2003, SRM, the Commission approved the staff recommendation.

DISCUSSION:

As part of developing the technology-neutral framework for new plant licensing, draft guidance has been developed related to implementation of a probabilistic approach for establishing the licensing basis. This draft guidance is intended for staff use in developing technology-neutral requirements based upon the framework guidance. Summarized below are the key elements of the draft guidance developed for implementation of a probabilistic approach for establishing the licensing basis, which the staff proposes for use in developing the technology-neutral requirements:

Probabilistic Event Selection Criteria

The following criteria are proposed for the categorization of event scenarios (identified in a design specific PRA) which must be considered in the design

- frequent $\geq 10^{-2}$/plant year (mean value)
- infrequent $<10^{-2}$/plant year but $\geq 10^{-5}$/plant year (mean value)
- rare $<10^{-5}$/plant year but $\geq 10^{-7}$/plant year (mean value)

These proposed criteria are intended to ensure that a sufficiently broad spectrum of event scenarios are considered consistent with the safety expectations expressed in the

Commission's Safety Goal Policy Statement. It is proposed to use each of these event categories as follows:

- Frequent event scenarios represent the anticipated operational occurrence (AOOs) range from which AOOs will be selected and will have to meet a deterministic dose criteria of 100 mrem as described in Part 20.

- Infrequent event scenarios represent the design basis accident (DBAs) range from which DBAs will be selected and will have to meet deterministic dose criteria associated with siting (e.g., 25 rem total effectiveness dose equivalent of the EAB).

- Rare event scenarios will be used for assessing emergency preparedness, as well as be used (along with the frequent and infrequent events) in assessing overall plant risk.

- Event scenarios of lower frequency than the rare category will not have to be considered for licensing purposes; however, it will also be necessary for an applicant to show that catastrophic initiating events (e.g., reactor pressure vessel rupture) that can cause the breach of all barriers to radiation release must be kept below a frequency of 10^{-7}/plant year.

Probabilistic Safety Classification

The staff proposes the safety classification of SSCs be based upon their risk importance. PRA results would be analyzed using conventional risk importance measures (e.g., risk achievement worth-where the failure rate of the SSC is set to one to determine the change in risk) and criteria established to categorize the importance of the SSC. The risk importance measures and criteria are yet to be developed, but will build upon the work done in support of the 10 CFR 50.69 rulemaking.

Single-Failure Criterion:

The single failure criterion will be replaced with the event sequences from the design specific PRA. Whichever number of failures are contained in those event sequences, the design and safety analysis will also need to consider.

As a final consideration, it is expected that designs that are licensed using a probabilistic approach will need to feedback operating experience into their PRA and maintain it as a living document. As such, event sequences and SSC importance may change over time potentially affecting the event categorization, AOO and DBA selection and analysis and safety classification of SSCs. Accordingly, a process to incorporate such changes into the license (for both certified and non-certified designs) will need to be developed.

USE OF SCENARIO-SPECIFIC SOURCE TERMS FOR LICENSING DECISIONS

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

BACKGROUND:

In SECY-03-0047, with regard to using scenario-specific accident source term for licensing decisions, the staff recommended 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 and builds upon the recommendation under the issue on the use of PRA. 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.

In the June 26, 2003, SRM, the Commission approved the staff's recommendation.

DISCUSSION:

As part of developing the technology neutral framework for future plant licensing, draft guidance has been developed and included in the framework related to implementation of a scenario specific source term approach. This draft guidance is intended for staff use when developing technology-neutral requirements based upon the framework. Summarized below are the key elements of the draft guidance developed for implementation of scenario specific licensing source terms, which the staff intends to incorporate in the technology-neutral requirements:

- The scenarios to be used for the source term evaluation are to be selected from a design specific probabilistic risk assessment, with due consideration of uncertainties, as discussed under the issue addressing the use of a probabilistic approach for establishing the licensing basis.

- The source term calculation, using the selected scenarios, should be based upon analytical tools that have been verified with sufficient experimental data to cover the range of conditions expected and to determine uncertainties.

- The source terms used for assessing compliance with dose related siting requirements should be 95% confidence level values based upon best estimate calculations with quantified uncertainties. Where uncertainties cannot be quantified, engineering judgement shall be used.

- The source terms used in assessing emergency preparedness should be mean values based upon best estimate calculations with quantified uncertainties.

- The source terms used for licensing decisions should reflect the scenario specific timing, form and magnitude of radioactive material released from the fuel and coolant. Credit may be taken for natural and/or engineered attenuation mechanisms in estimating the release to the environment, provided there is adequate technical basis to support their use.

The guidance is intended to provide a flexible, performance-based, approach for establishing scenario specific licensing source terms. However, it also puts the burden on the applicant to develop the technical bases (including experimental data) to support their proposed source terms. Applicants could, however, propose to use a conservative source term for licensing purposes (in order to reduce research and development costs and schedule), provided the use of such a source term does not result in design features or operational limits that could detract from safety.

Finally, it should be noted that in parallel with developing technology-neutral regulations, the staff also plans to develop technology-specific Regulatory Guides that will provide guidance on one acceptable way to implement the technology-neutral regulations on a specific reactor technology (e.g., high temperature gas-cooled reactors). These Regulatory Guides could provide further guidance on the use of scenario specific source terms, such as credit for attenuation mechanisms. In this regard, it is expected that some future LWR designs will also propose to use scenario specific source terms. These requests could be reviewed on a case-by-case basis using the guidance.

POSSIBLE MODIFICATIONS OF EMERGENCY PREPAREDNESS REQUIREMENTS

ISSUE: Under what conditions can the emergency preparedness requirements be modified to give credit for reactor designs with enhanced safety characteristics?

BACKGROUND:

In SECY-03-0047, the staff recommended 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 (“Emergency Plans”) 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.

In the longer term, the staff also recommended that the role of emergency preparedness in defense-in-depth would be addressed as part of the staff’s work to develop a policy or description of defense-in-depth which is part of the framework development, as recommended under the defense-in-depth issue. 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. In the June 26, 2003, SRM, the Commission approved the staff recommendation in SECY-03-0047.

Current requirements associated with emergency preparedness (i.e., 10 CFR 50.47, and 10 CFR Part 50, Appendix E [“Emergency Planning and Preparedness for Production and Utilization Facilities”]) have been developed primarily in consideration of the risks from currently operating LWRs. However, 10 CFR 50.47 does recognize that for gas-cooled nuclear reactors and for reactors with authorized power level less than 250 Mwt, the size of the emergency planning zones (EPZs) may be determined on a case-by-case basis. This situation was the case for the Fort Saint Vrain reactor which had a 5-mile EPZ, instead of the 10-mile EPZ, that is applied to currently operating LWRs.

In the past, there have been proposals to modify current emergency preparedness requirements to give credit for designs with enhanced safety characteristics. Staff reviews and response to these proposals were provided. In general, these responses indicated that for new reactor designs, it is too early to identify specific conditions that would allow a reduction in the 10-mile plume exposure pathway EPZ. Until sufficient experience is gained on any prototype reactor, a case-by-case basis should be used to evaluate whether a requested reduction in the size of the EPZ can be allowed. This criterion would also apply to the 50-mile ingestion control pathway EPZ. Some conditions that would have particular importance would include, but would not be limited to, the following:

- consideration of the full range of accidents
- use of the defense-in-depth philosophy
- prototype operating experience is gained

- acceptance by federal, state, and local agencies
- acceptance by the public

Finally, all sixteen Planning Standards and Evaluation Criteria (A through P) in NUREG-0654/FEMA-REP-1, Rev. 1, should be addressed for any size EPZ. The specific requirements under each applicable standard could be scaled down, as appropriate, in order to account for any reduction in EPZ size. Modification of the rules or guidance documents should not occur until sufficient experience is gained in dealing with reduced EPZs.

DISCUSSION:

The staff plans to obtain stakeholder feedback on the above emergency preparedness considerations, as they relate to modifying emergency preparedness requirements to give credit for reactor designs with enhanced safety characteristics. Based upon feedback and further technical considerations, provide a recommendation to the Commission in late 2005.

ASSESSMENT OF CONTAINMENT OPTIONS FOR MODULAR-HTGRs

Each of the four alternative technology-neutral containment functional performance criteria (i.e., options) are discussed and evaluated on the basis of its specific application to modular HTGRs. In support of and in advance of these evaluations, the following additional modular HTGR-specific information is provided.

Modular HTGR Approach to Radionuclide "Containment"

Compared to operating LWRs, modular HTGR designers have proposed a very different design approach to prevent unacceptable releases of radionuclides to the environs. Modular HTGRs are designed to contain the vast majority of radionuclides at the source, within billions of small, high integrity, refractory coated fuel particles. To ensure dose acceptance criteria are met, the failure of a radiologically significant fraction of the fuel particles is not permitted during either normal operation, anticipated transients, design basis accidents or beyond design basis accidents. Thus, the safety philosophy of modular HTGRs is to assure the integrity of the coated particle fuel particle (i.e., the "containment barrier"). Modular HTGR designers propose to use high quality fuel, which has been qualified for the specified operating and accident conditions and then reliably limiting the fuel operating and accident conditions (e.g., maximum transient fuel temperature) to values within the qualification envelope. This objective is to be reliably accomplished with a reactor design having a relatively low core power density compared to operating LWRs (to limit accident decay heat input into the core) and inherent safety characteristics and simplified passive means, to shutdown and remove core decay heat in the event of an transient or accident. The safety philosophy is to assure the fuel containment barrier rather than to allow significant fuel failures and then have to rely extensively on either backup barriers (such as a containment) or other mechanistic barriers associated with the core graphite structures or reactor coolant pressure boundary. In this regard, preventing significant releases of fission products from the fuel is consistent with the ultimate objective of the Commission's advanced reactor policy which expects advanced reactor designs to minimize the potential for severe accidents.

Mechanistic Barriers

As allowed by Commission policy, in determining on-site and offsite dose, modular HTGR designers propose to take credit for all of the multiple "mechanistic barriers" to radionuclide transport associated with the fuel, core graphite structures, reactor coolant pressure boundary and containment. For modular HTGRs, designers propose that the containment be relied on to assist in protecting the fuel, core graphite, the reactor pressure boundary and in meeting dose criteria, but need not provide an essentially leak-tight barrier against the release of radionuclides to the environs. Modular HTGR designers have stated that given the effectiveness of the other mechanistic barriers, the containment provides only additional safety margin and margin to the dose criteria. However, it's contribution as a mechanistic barrier is not required to meet the radionuclide dose criteria, at least for the events that modular HTGR designers have selected for the event categories.

Vented Low Pressure Containment

Modular HTGR reactor coolant system (RCS) circulating activity and plateout activity are to be monitored and controlled in order to limit radionuclides within the RCS to relatively low levels during normal operation. For this reason, modular HTGR designers have proposed that the containment be what is referred to as a vented low pressure containment (VLPC). For moderate-to-large pipes breaks in the RCPB, a VLPC is designed to allow the hot pressurized helium and the limited contained and entrained radioactivity in the reactor coolant system to blowdown directly to the environs. During the blowdown, credit is usually taken for plateout of some of the condensible radionuclides on the cooler surfaces of the VLPC. Accordingly, radionuclides released to the environs during the RCS blowdown (i.e., the prompt radionuclide release) is expected to involve a relatively low radiological dose, even at the site boundary. Additionally, modular HTGR designers state that depressurizing the RCPB and VLPC down to atmospheric pressure through a reclosable ventilation duct removes the motive force, that might otherwise be available for radionuclide transport later in the accident, when additional fractional failure of fuel particles are expected to occur (i.e., delayed radionuclide release).

The Effect of Event Selection on Containment Functional Performance and Design

Modular HTGR design and safety analysis information reveals that the functional performance and the design of the VLPC depends on the RCPB break scenarios included in the design-basis. This is due to the potential for additional challenges to the core and increase in the delayed accident source term that can be caused by significant air ingress from a RCPB break. These additional challenges include the potential for degradation of the graphite core support structures due to oxidation (i.e., degradation of core cooling), the potential for additional fuel particle failures due to oxidation of the ceramic coatings and the potential for re-introducing a motive force to radionuclide transport from the core, the RCS and the VLPC with the onset of natural circulation gas mixture flow through the core. A severe air ingress event therefore has the potential to significantly degrade the mechanistic barriers, significantly increase the magnitude of the delayed source term and potentially significantly increase the dose consequences.

HTGR designers state that the VLPC has a functional role to prevent a large volume air ingress in order to prevent these consequences. Thus, a severe air ingress event can establish a maximum allowed (post-blow down) leakage rate for the VLPC. For example, failure of the large diameter cross-connect duct/vessel, or failure of smaller diameter vessel penetrations above and below the core can result in severe air ingress events. However, recent HTGR design and safety analysis information indicates that the more challenging RCPB breaks are not always selected for analyzing air ingress consequences since they are considered to be very low probability events. Less severe air ingress events may not be sufficiently challenging to require establishment of a specific VLPC leakage rate. Additionally, some modular HTGR designers have targeted break prevention and/or alternative mitigation strategies (that would seek to take advantage of the relatively long time available for human actions due to the expected very slow rate of core heat up) to address potential severe air ingress events as an alternative to reducing the allowed post-blowdown VLPC leakage acceptance criteria.

Evaluation of Technology-Neutral Containment Options for Modular HTGRs

The staff has evaluated each of the above four options for modular HTGRs based on the above metrics. These modular HTGR-specific assessments supplement (rather than replace) the technology-neutral assessments provided in Attachment 2.

Option 1: The containment must adequately reduce radionuclide releases to the environs to meet the onsite and offsite radionuclide dose acceptance criteria for the events selected for the event categories.

For advanced HTGRs this option would likely allow a VLPC, currently proposed by the HTGR designers. As acknowledged by HTGR designers, it would require a high of level assurance that fuel and other SSC performance and related uncertainties are well-understood for a wide range of conditions and that the fuel fabrication process maintains the requisite fuel quality over the life of the plant. Also, HTGR designers state that a function of the VLPC is to limit air ingress into the core to prevent excessive graphite oxidation and fuel degradation but some propose a large allowable VLPC leakage rate (e.g., 100%/day) and/or other mitigation approaches to limit the volume of air that might otherwise enter the core. Not all HTGR designers propose to include in the VLPC design-basis, the more severe air ingress events since they consider them to be very low probability events. This option would not explicitly require inclusion of these more severe air ingress events although the staff could recommend that they be included and enhancements applied, if needed, based on deterministic engineering judgement. However, if included, such enhancements could include strengthened prevention measures or alternative mitigation strategies. Accordingly, this option could result in allowing a VLPC with a relatively large allowed leakage rate.

It is not explicitly responsive to the July 30, 1993 SRM for SECY-93-092 which directed the staff to address in the development of the containment performance criteria, loss of primary coolant pressure boundary integrity events which can result in ingress of air leading to natural circulation through the core with the potential loss of fuel particle integrity. However, as noted above such breaks could still be included in the design-basis if deterministic engineering judgement found that they should be included in order to bound uncertainties.

Except for any additional required enhancements, this option would not involve incremental costs, which HTGR designers believe could make such designs less competitive.

Because modular HTGRs, are expected to involve a much lower release of radionuclides into the containment during normal operation and frequency-based design-basis events, public confidence could be enhanced. However, because this option would likely allow an HTGR containment with less capability to reduce radionuclide releases to the environs compared to LWR containment designs, it might be perceived as providing less protection, thereby potentially reducing public confidence overall.

Option 2: The containment must adequately reduce radionuclide releases to the environs to meet the onsite and offsite radionuclide dose acceptance criteria for the events selected for the event categories (including within the design-basis category, selected credible events having the potential for high consequence source terms).

This criterion is the same as Option 1 except that it specifically requires that credible very low probability, high consequence, source term events, such as, potentially, the failure of the cross-connect duct/vessel, be included in the design-basis event category. Such bounding source term events would be used to assess whether all mechanistic barriers, including the containment performance, provide sufficient defense-in-depth in reducing radionuclide transport, to meet dose criteria. For modular HTGRs, this option would likely allow a VLPC, but for some HTGR designs, it could necessitate limiting the volume of air (i.e., lower VLPC leakage rate) that would be available for core oxidation and delayed radiological source term release to the environs. HTGR designers may seek to pursue alternative mitigation strategies to limit the volume of air ingress into the core rather than limiting the post-blowdown leakage rate of the VLPC, which the staff may, or may not accept as sufficiently reliable. If alternative strategies are not accepted, limiting the VLPC post-blowdown leakage rate would likely be required.
This option is explicitly responsive to the July 30, 1993 SRM for SECY-93-092. The SRM directed that modular HTGR containment performance criteria include consideration of a loss of primary coolant pressure boundary integrity which results in ingress of air leading to natural circulation through the core and the potential loss of fuel particle integrity.

Currently, most (but not all) worldwide modular HTGR designers do not include in the design-basis event category, such bounding events of potentially very low probability. For designs that currently do not include such events, additional technology development would be required to support the source term calculation associated with air ingress and graphite and fuel oxidation. Additionally, for modular HTGR designs that currently do not include a cross-connect duct/vessel failure in the design-basis envelope, if included, the significantly higher thermal-dynamic loads on the VLPC structures could require structural changes to the VLPC in order for the higher stresses to meet structural stress (i.e., American Society for Mechanical Engineers, ASME) limits. This would add to the design and construction costs of the VLPC for these plants.

Because modular HTGRs, are expected to involve a relatively low release of radionuclides into the VLPC during normal operation and frequency-based design-basis events, public confidence could be enhanced. If more challenging and lower probability events were included in the design-basis category and air ingress and delayed source term were limited by the limiting the VLPC leakage rate, it would likely further increase public confidence relative to Option 1.

Option 3: The containment must adequately reduce radionuclide release to the environs to meet the onsite and offsite radionuclide dose acceptance criteria for the events selected for the event categories (including within the design-basis category, selected credible events having the potential for high consequence source terms) and have the capability to establish controlled leakage and release of delayed accident source term radionuclides.

This criterion is same as Option 2 that credible very low probability, high consequence, source term events, such as, potentially, the failure of the cross-connect duct/vessel, be included in the design-basis event category. For modular HTGRs, this option would still likely allow a VLPC. However, it would further prescriptively require that the VLPC have the capability to establish controlled leakage and release of delayed accident source term radionuclides following the depressurization event. This VLPC design has been referred to as a “hybrid containment” because it would allow the initial RCS depressurization to vent directly to the environs for loss of reactor coolant pressure boundary events, but would require that the VLPC have the capability to establish a controlled, low leakage, thereafter. This option would limit the volume of air in-leakage available for core oxidation and would limit the volume of air out-leakage available for radionuclide transport to the environs of the delayed radiological source term. Accordingly, this option is also responsive to the July 30, 1993 SRM for SECY-93-092.

Currently, not all HTGR designs require that the VLPC have the capability the limit the in-leakage rate and out-leakage rate from the VLPC to a controlled limited value after a severe depressurization event. Accordingly, for such plants, this option would likely involve some VLPC design changes. For such plants, this option would also likely require structural changes to the VLPC design in order to meet structural stress (i.e., ASME) limits and upgrades to the vent system in order to assure a reliable vent path reclosure capability. This would add to the design and construction costs of the VLPC.

Because modular HTGRs, are expected to involve a relatively low release of radionuclides into the VLPC during normal operation and frequency-based design-basis events, public confidence could be enhanced. Including the capability for controlled leakage and release of the delayed accident source term radionuclides would likely further increase public confidence relative to Options 1 or 2.

Option 4: The containment must adequately reduce radionuclide releases to the environs to meet the onsite and offsite radionuclide dose acceptance criteria for the events selected for the event categories (including within the design-basis category, selected credible events having the potential for high consequence source terms) by being essentially leak tight against the release of prompt and delayed accident source term radionuclides.

This option would prescriptively require that modular HTGRs have a conventional LWR containment design rather than a VLPC. It would prevent the release to the environs of both the initial and delayed source term. It would also effectively limit the volume of air that would be available for core graphite oxidation as a result of a bounding air ingress event. However, a conventional containment would have a negative impact on modular HTGR safety by reducing the effectiveness of the passive design approach to decay heat removal and by retaining the motive force for radionuclide transport for core heatup events. This option is not consistent with the position taken in the Commission's July 30, 1993, SRM, nor with the Commission's advanced reactor policy, which states that regulatory guidance must be sufficiently general to avoid placing unnecessary constraints on the development of new design concepts.

This option is inconsistent with the Commission white paper on risk-informed and performance-based regulation and the performance-based approach to containment functional performance criteria proposed by modular HTGR designers.

For modular HTGRs, this option would add substantially to the cost which is not considered commensurate with the safety benefits. HTGR designers state that this option would make plant designs uneconomical. This option is also inconsistent with the Commission's prior decision documented in the SRM for SECY-93-0092, but could result in higher public confidence than Options 1, 2, or 3.