UNITED STATES NUCLEAR REGULATORY COMMISSION

POLICY ISSUE
(Information)

June 30, 1995 SECY-95-172

FOR: The Commissioners

FROM: James M. Taylor
Executive Director for Operations

SUBJECT: KEY TECHNICAL ISSUES PERTAINING TO THE WESTINGHOUSE AP600 STANDARDIZED PASSIVE REACTOR DESIGN

PURPOSE:

To inform the Commission of proposed resolutions, and, where appropriate, staff positions on key technical issues pertaining to the Westinghouse AP600 standardized passive reactor design certification. The staff is not identifying any new policy issues in this paper. However, should a new policy issue be identified upon resolution of an issue, it will be promptly identified to the Commission.

BACKGROUND:

In June 1992, Westinghouse submitted its application for design certification of the Westinghouse AP600 passive reactor design. In November 1994, the staff issued its draft safety evaluation report (DSER) for the AP600 and is in the process of resolving open issues identified in the DSER and issues identified during the review of recent Westinghouse submittals. The staff has identified key technical issues for which it is developing or has reached a position regarding acceptable resolution. A draft of this technical issues paper was provided to the Commission and the Advisory Committee on Reactor Safeguards (ACRS) on May 15, 1995. The draft was also made available to the public on May 18, 1995.

DISCUSSION:

In Attachment 1, the staff describes proposed resolutions, and, where appropriate, gives its current positions regarding resolution of select technical

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issues pertaining to the Westinghouse AP600 passive reactor design. The description also includes, where appropriate, a detailed discussion of the basis for the staff's position. Some of the issues were identified to the Commission as policy issues in SECY-90-016, "Evolutionary Light-Water Reactor (LWR) Certification Issues and Their Relationship to Current Regulatory Requirements," SECY-93-087, "Policy, Technical, and Licensing Issues Pertaining to Evolutionary and Advanced Light-Water Reactor (ALWR) Designs," SECY-94-084, "Policy and Technical Issues Associated with the Regulatory Treatment of Non-Safety Systems (RTNSS) in Passive Plant Designs," and SECY-95-132, "Policy and Technical Issues Associated with the Regulatory Treatment of Non-Safety Systems (RTNSS) in Passive Plant Designs (SECY-94-084)." The Commission provided guidance on these issues in its staff requirements memoranda dated June 26, 1990, July 21, 1993, and June 30, 1994. This paper provides the status of the implementation of that guidance for those issues.

In Attachment 1, the staff also identifies new issues resulting from the staff's review of the AP600 design which, because of the stage of the review process, the staff considers to be technical issues. The staff is still discussing the resolution of these issues with Westinghouse. However, where appropriate, the staff also discusses resolution approaches and its current positions on these matters.

WESTINGHOUSE COMMENTS

In its letter dated June 9, 1995, Westinghouse provided its comments on the draft Commission paper. Attachment 2 provides a copy of that letter. The staff has factored these comments into the paper, where appropriate.

COORDINATION:

The Office of the General Counsel (OGC) has reviewed this paper and has no legal objection.

The ACRS was briefed during meetings on May 31 and June 9, 1995. The Committee has provided its comments in a letter dated June 15, 1995.

SUMMARY:

The staff is not identifying any new policy issues in this paper. Therefore, the staff intends to proceed with its review of the AP600 design in accordance with the positions identified in the attachment.

Should a new policy issue be identified upon resolution of an issue, it will be promptly identified to the Commission.

James M. Taylor
Executive Director
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Attachments:
1. Key Technical Issues on the AP600 Design
2. Letter from Westinghouse Electric Corporation dated June 9, 1995

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Key Technical Issues
on the AP600 Design

Attachment 1

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KEY TECHNICAL ISSUES ON THE AP600 DESIGN

I. Leak-Before-Break Approach

In General Design Criterion (GDC) 4 of Appendix A to Part 50 of Title 10 of the Code of Federal Regulations (10 CFR), it specifies that structures, systems, and components important to safety shall be appropriately protected against dynamic effects that may result from equipment failures and from events and conditions outside the nuclear power unit. However, it also specifies that dynamic effects associated with postulated pipe ruptures may be excluded from the design basis when analyses reviewed and approved by the Commission demonstrate that the probability of fluid system piping rupture is extremely low under conditions consistent with the design basis for the piping.

In SECY-93-087, the Nuclear Regulatory Commission (NRC) staff recommended to the Commission that the leak-before-break (LBB) approach be approved for both evolutionary and passive advanced light water reactors (ALWRs) seeking design certification under 10 CFR Part 52, in lieu of postulating pipe breaks as required by GDC 4. This approval was limited to instances in which appropriate bounding limits are established using preliminary analysis results during the design certification phase and verified during the combined operating license (COL) phase by implementing appropriate inspections, tests, analyses, and acceptance criteria (ITAAC). The Commission approved the staff's recommendation in its memorandum dated July 21, 1993. The staff also noted in SECY-93-087 the need to develop specific details as the process is implemented. In this position, the staff discusses the need for an appropriate bounding limit for the LBB leakage rate.

The leakage rate limit is an important parameter in the LBB evaluation because it (1) initiates actions for the plant operator in the plant Technical Specifications if the limit is exceeded and (2) establishes the initial size of the leakage flaw and, consequently, the magnitude of the design-basis loads that may be imposed on the piping before the flaw becomes unstable. For the AP600 plant design, Westinghouse is proposing to use a leakage rate limit of 0.5 gpm for a Technical Specification limit and for determining the initial flaw size. Westinghouse has stated that its proposed leak detection methods are in conformance with Regulatory Guide (RG) 1.45 and will result in the detection of a 0.5 gpm leak rate[1].

Operating experience has shown that leakage can be detected to a point where it may be possible to detect leakage rates as low as 0.5 gpm. Therefore, the leakage detection element is not of concern.

In establishing the initial flaw size, it is essential to ensure that the margins available in the LBB approach are adequate to bound the uncertainties

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[1] To date, the NRC has only approved a leakage rate limit of 1.0 gpm to determine the initial flaw size for all operating and evolutionary plants with one exception. For the Beaver Valley Unit 2 facility, the NRC staff approved the use of 0.5 gpm leakage rate when a margin of 1.4 is used on the normal plus SSE loads.

and do not allow the overprediction of the loads at which the flaw becomes unstable. In NUREG-1061, Volume 3, "Report of the U.S. Nuclear Regulatory Commission Piping Review Committee, Evaluation of Potential Pipe Breaks," dated November 1984, the staff discusses its recommended margins for establishing the critical crack size, loads, and leak detection capability as well as overall acceptance criteria for LBB. For the AP600, Westinghouse has committed to meet the NUREG-1061, Volume 3, margins for critical crack size and leak detection capability, and these margins are thus acceptable. However, in NUREG-1061, Volume 3, the staff recommended a margin of $\sqrt{2}$ (1.4) between the maximum combined load calculated for the piping (which is the algebraic sum of normal operating plus accident (e.g., SSE) loads) and the load at which the leakage size flaw becomes unstable. In its letter dated June 9, 1995, Westinghouse proposed instead to use a factor of 1.0 on loads as recommended by the staff in draft Standard Review Plan Section 3.6.3 (52 FR 32628, August 28, 1987) when the loads are combined by the absolute sum method. This is also acceptable to the staff because the combination of loads by the absolute sum method provides more margin on loads in the LBB analyses than if the loads were combined algebraically. This is important when LBB is applied to smaller diameter piping.

It should be noted that the staff has learned from recent testing that the capability to predict leakage rates is more difficult in smaller diameter piping (i.e., less than 10-inch nominal pipe size) than in larger diameter piping because of the complexities involved in predicting the crack opening angle and flaw roughness as a result of the more pronounced effect of weld residual stresses in smaller diameter piping. The staff is continuing its review of these uncertainties in the application of LBB to smaller diameter piping on a generic basis.

On the basis of the above discussion, the staff concludes that an LBB leakage rate limit of 0.5 gpm may be used to establish the initial leakage flaw size for the AP600 plant provided the margins for critical crack size and leak detection capability are the same as those delineated in NUREG-1061, Volume 3, and the margin on loads is 1.0. The deadweight, thermal expansion, pressure, SSE (inertial), seismic anchor motion, and other applicable loads are to be combined by the absolute sum method as delineated in draft SRP Section 3.6.3.

In its letter dated June 15, 1995, the ACRS stated that the staff is hard pressed to justify adding conservatism on all the piping loads above that which has been applied to other plants. However, since the ACRS letter was issued, the staff re-examined its position on the margin for loads and finds that using a factor of 1.0 on loads and combining the loads by the absolute sum method provides a conservative approach for the AP600 LBB analysis. Thus, the staff is applying a level of conservatism on the piping loads equivalent to that which has been applied to operating plants.

II. Security Design

As specified in 10 CFR 73.55(a), to achieve the general performance objective for physical protection of licensed activities in nuclear power reactors against radiological sabotage, the onsite physical protection system and security organization must include, but not necessarily be limited to, the capabilities to meet the specific requirements in 10 CFR 73.55(b) through (h). As further specified, the Commission may authorize an applicant or licensee to provide measures for protection against radiological sabotage other than those required in 10 CFR 73.55(b) through (h) if the applicant or licensee demonstrates that the measures have the same high assurance objective as specified in 10 CFR 73.55(a) and the overall level of system performance provides protection against radiological sabotage equivalent to that which would be provided by 10 CFR 73.55(b) through (h) and meets the general performance requirements of 10 CFR 73.55.

Westinghouse is proposing a conceptual design for the AP600 security program that differs significantly from current site security programs. As required by 10 CFR 73.55(c), vital equipment must be contained in a vital area, access to which is controlled by a physical barrier. The vital areas, in turn, must be located inside a protected area with access controlled by a physical barrier. Access to vital equipment, therefore, requires passage through at least two physical barriers. Typically, vital equipment must be located inside a building that constitutes the second barrier. The first barrier is usually a fence at the protected area perimeter. The detection of intrusion at the protected area boundary now provides licensees a certain amount of response time before vital equipment is affected by an external threat. Vital access portals, as currently constructed, generally do not provide a significant delay of access vital areas due to an external threat. A licensee must rely on its ability to interdict with proper security forces.

The Westinghouse AP600 design locates all vital equipment in the contiguous containment and auxiliary building complex referred to as the "nuclear island." Security, as proposed by Westinghouse, will rely primarily on enhanced defensive capabilities at access points or openings in the vital area barriers and the design of the reinforced concrete walls to provide for the delay and detection of an attempted penetration through the walls. For substantial portions of the nuclear island, protected area physical barriers and an intrusion detection and assessment system are not included in the design. A vehicular barrier will still exist, such that the AP600 design discussed above could protect against the malevolent use of a land vehicle.

The AP600 design divides the nuclear island into two major areas, the clean side and the dirty side, and limits the number of normal personnel access points to two. Security personnel with high-powered rifles will be stationed at these access points. For design certification, the AP600 security design includes hardened defensive positions for security officers at these key access routes. Most other access points, such as maintenance access points, are close to the location of security personnel or require the adversary to pass through hardened defensive points to gain access to vital equipment. Access points that do not channel the adversary through hardened defensive points will be hardened to the point that adequate time is provided for security response.

Because of the design strength of vital area walls, delay time would be sufficient to allow security personnel to respond to an external attack. The pattern and spacing of reinforcing bars in the walls are such that a single bomb detonation would not result in access to vital areas. The detonation of an explosive large enough to create a man-size hole in the concrete would be sufficiently loud and distinctive to assure detection of the penetration attempt. A second detonation would be needed to remove the reinforcing bars and allow intruders access through the breach in the wall.

The staff considers that Westinghouse's proposal has merit. The proposed design would reduce the number of security personnel and the amount of intrusion detection and alarm assessment equipment needed. The conceptual design locates security personnel with the necessary weaponry in hardened defensive structures at the potentially vulnerable penetration points for external attacks. The design, however, is likely to require the development of security ITAACs.

The staff is still reviewing Westinghouse's submittals. The staff's evaluation of this issue has not yet reached a stage where a final conclusion regarding the acceptability of this proposal can be reached. The results of its review will be documented in the AP600 final safety evaluation report (FSER).

III. Technical Specifications (TS)

For the AP600 design, Westinghouse is proposing that Limiting Condition for Operation (LCO) 3.0.3 specify MODE 4 (hot shutdown) instead of MODE 5 (cold shutdown) as the safe shutdown end state. Westinghouse offered the following justification, to be included in the TS bases:

- The passive design safe shutdown is defined at 420 °F.
- There is an increase in risk during equipment lineup changes necessary to enter MODE 5. (Probabilistic risk assessment (PRA) basis to be provided.)
- Repair of unisolable or single barrier isolable fluid systems would be the main reason to go to MODE 5, that is, personnel safety (i.e., radiological or other personnel hazards) as opposed to nuclear safety.
- Releasing steam to the containment from the in-containment refueling water storage tank would not be a significant problem under most conditions.
- Operating plants would use normal non-safety-related systems available to remove decay heat.

To the extent the staff fully accepts the passive design along with its defined safe shutdown, as discussed in Section III.D of SECY-93-087, it will consider the Westinghouse-proposed LCO 3.0.3. However, the staff's evaluation of this issue has not yet reached the stage where a final position regarding its resolution can be taken. The staff will document the results of its review regarding this issue in the AP600 FSER.

IV. Initial Test Program

Chapter 14, "Initial Test Program (ITP)," of the AP600 standard safety evaluation report (SSAR) is generally comprehensive and encompasses the major phases of a test program as described by the staff's review guidance. In the AP600 design certification draft safety evaluation report (DSER), the staff identified two major issues related to the proposed ITP test abstracts that describe the test details for the AP600 systems. First, the staff identified concerns with regard to the capability of proposed testing methods to subject the AP600 passive systems, components, and/or design features to representative (actual or simulated) design-basis operating conditions to demonstrate the capability of the passive systems to perform their design functions. The staff is continuing its discussions with Westinghouse to reach a consensus on a viable ITP framework to resolve this issue.

Second, the staff identified a concern with regard to Westinghouse's proposal to perform certain preoperational tests on the first AP600 plant only. Westinghouse proposed that these tests not be conducted on subsequent plants because of the standardization of the AP600 design and the experience gained during the startup of the first plant. To resolve this issue, Westinghouse is developing new criteria to be used for determining if an ITP test should be performed at the first AP600 plant only or at all AP600 plants. The staff will review the criteria and redesignation of lead-plant-only abstracts in Chapter 14 of the SSAR when they become available. Westinghouse has also proposed an approach to the testing of AP600 safety-related and non-safety-related systems that the staff is reviewing.

The staff is still reviewing Westinghouse's submittals. The staff's evaluation of these issues has not yet reached a stage where final conclusions regarding the acceptability of these proposals can be reached. The results of the staff's review of both issues will be given in the AP600 FSER.

Chapter 14 indicates that some construction, installation, and preoperational testing may be part of the ITAAC program. The staff agrees that passive system testing for ITAAC will be confirmed in large measure during the preoperational test program.

V. Passive System Thermal-Hydraulic Performance Reliability

The Westinghouse AP600 passive reactor design has a number of unique features that distinguish it from both operating and advanced evolutionary light water reactor designs. Although active and "passive" systems for accident prevention and mitigation are used in all designs, only the "passive" systems are safety-related in the AP600 design.

"Passive" safety systems rely on natural forces such as gravity and stored energy to perform their functions. The net driving forces are relatively small compared to those of pumped systems, their magnitude depends on the accident scenario, and they are subject to large uncertainties. These uncertainties affect the thermal-hydraulic (T/H) performance reliability (T/H reliability) of "passive" systems and must be assessed and accounted for in the PRA. However, quantification of this reliability involves a prohibitively large number of computations. For this reason, a conservative risk-based "margins" approach was developed to eliminate the need to quantify T/H reliability for most, if not all, accident sequences. The staff is planning to perform a case study to quantify thermal-hydraulic uncertainties for a selected AP600 PRA sequence. Lessons learned from the margins approach and this case study will shape the staff's review strategy for future advanced passive reactor designs.

The staff believes that the use of a risk-based margins approach in lieu of a quantitative assessment of system reliability can adequately address the issue of T/H performance reliability of passive systems if implemented appropriately. The staff is still reviewing Westinghouse's submittals. The staff's evaluation of this issue has not yet reached a stage where a final position regarding the resolution of this issue can be taken. The results of the staff's review of this issue will be given in the AP600 FSER.

VI. Regulatory Treatment of Non-Safety Systems (RTNSS)

The staff previously discussed the RTNSS issue in SECY-93-087, SECY-94-084, and SECY-95-132. The staff and Westinghouse are meeting regularly to identify and resolve key aspects of RTNSS issues. One key issue is the acceptability of the baseline PRA, particularly the question of passive system reliability and the treatment of thermal-hydraulic uncertainty (which is discussed in Section V of this paper). Other issues include evaluation of adverse systems interactions, application of the Technical Specifications for short-term availability control of RTNSS systems, and post-72 hour supporting action strategies.

Westinghouse is proposing a systematic approach to identify potential systems interaction between safety- and non-safety-related systems for the AP600. The staff is evaluating this effort. The staff is also discussing the need for Technical Specifications to ensure short-term availability control of RTNSS systems that are identified to be important during a limited period of plant operation. Westinghouse believes that other administrative controls (plant procedures) are adequate to ensure that these systems are operational during the period of time that they are important to the safe operation of the plant. The key issue with regard to post-72 hour actions is that Westinghouse is proposing that the licensing basis for reliable, long-term cooling be the use of equipment which is out of the control of the licensee and offsite. At a meeting on April 27, 1995, Westinghouse presented a discussion of its long-term cooling strategy. The staff believes that reliance on offsite support for post-72 hour actions raises significant concerns regarding long-term cooling after an event. The staff has expressed those concerns to Westinghouse and is actively reviewing its proposal.

The staff is still reviewing Westinghouse's proposals. The staff's evaluation of these issues has not yet reached a stage where final positions regarding the resolution of these issues can be taken. The results of the staff's review of these issues will be given in the AP600 FSER.

VII. Containment Performance

The containment performance goal (CPG) is intended to ensure that the containment structure has a high probability of withstanding the loads associated with severe accident phenomena and that the potential for significant radioactive releases from the containment is small. The CPG includes both a deterministic goal that containment integrity be maintained for approximately 24 hours after the onset of core damage for the more likely severe accident challenges, and a probabilistic goal that the conditional containment failure probability (CCFP) be less than approximately 0.1 for the composite of all core damage sequences.

The CPG was previously discussed in SECY-90-016 and SECY-93-087. In SECY-93-087, the staff proposed an interim CPG for passive reactor designs and stated that it would determine the need to revise the interim approach as a result of the review of the passive reactor designs and evaluation of comments on the advance notice of proposed rulemaking concerning "Severe Accident Plant Performance Criteria for Future LWRs." In the SRM of July 21, 1993, the Commission approved the staff's interim approach of using a deterministic CPG in the evaluation of the passive reactor designs as a complement to the CCFP approach.

In SECY-93-226, "Public Comments on 57 FR 44513 - Proposed Rule on ALWR Severe Accident Performance," the staff provided the Commission with a summary of public comments received regarding the advance notice of proposed rulemaking, and the recommendations regarding policy issues raised in these comments. On the basis of a review of these comments and experience gained from the evaluation of the evolutionary reactor designs, the staff continues to believe that use of both a deterministic and a probabilistic CPG should be pursued for the passive reactor designs. The results of the staff's review of this issue will be given in the AP600 FSER.

VIII. External Reactor Vessel Cooling

External reactor vessel cooling (ERVC) is a severe accident management strategy and design feature that involves cooling the outside of the reactor vessel by flooding the reactor cavity to prevent reactor vessel melt-through following core melt. This strategy appears to offer significant potential for mitigating severe accidents by preventing ex-vessel severe accident phenomena, such as core-concrete interaction, high-pressure melt ejection, containment liner melt-through, and ex-vessel steam explosions. Because of the international interest in this strategy, experimental programs are being conducted worldwide to assess its feasibility and effectiveness.

Westinghouse is pursuing staff endorsement of ERVC for the AP600 design. The AP600 appears to be conducive to ERVC as the design has no bottom head penetrations, a low core power density, and the capability to flood the reactor cavity to the primary system piping. The staff is evaluating the AP600 design and particular core melt scenarios that could be affected by ERVC. If successful, ERVC represents a significant improvement in the mitigation of severe accidents.

However, several technical issues associated with ERVC need to be resolved before any regulatory endorsement of this accident management strategy. The key issues are (1) applicability and scaling of experimental data to the AP600 design, (2) impact of insulation surrounding the reactor vessel on water ingression and steam venting, (3) uncertainties in heat transfer coefficients both within the molten debris pool and from the reactor vessel lower head to the surrounding water, (4) reactor vessel material properties and strength at elevated temperatures, and (5) the potential for the strategy to increase the loadings from any ex-vessel steam explosions in the event ERVC fails. Westinghouse is aware of the technical issues associated with ERVC and is attempting to resolve them.

The staff believes that the ERVC strategy is consistent with the guidance in the SRM pertaining to SECY-93-087. In particular, under the topic of core debris coolability, the Commission stated that the staff should not limit vendors to only one method for addressing containment responses to severe accident events but permit other technically justified means for demonstrating adequate containment response. The staff is still reviewing Westinghouse's submittals. The staff's evaluation of this issue has not yet reached a stage where a final conclusion regarding the acceptability of this proposal can be reached. The results of the staff's review of this issue will be given in the AP600 FSER.

IX. Passive Hydrogen Control Measures

Control of hydrogen generated after a design-basis accident (DBA) or a severe accident is required of advanced reactor designs. The hydrogen generated can have a significant impact on the containment's performance through mechanisms including pressurization, hydrogen burns, hydrogen detonations, and degradation of containment heat removal systems.

For DBAs, NRC requirements for hydrogen control are specified in 10 CFR 50.44 and 10 CFR Part 50, Appendix A, GDC 41. These requirements specify the hydrogen source term and systems to control the concentration of hydrogen and oxygen that may be released into the reactor containment after postulated accidents. For large, dry containments of operating reactors, safety-grade electrically powered hydrogen recombiners have been used for controlling the hydrogen concentration. Westinghouse has recently informed the staff that it will pursue approval of the use of passive autocatalytic recombiners instead of electrically powered hydrogen recombiners for DBAs.

For severe accidents, the staff recommended in SECY-93-087 that the Commission approve the position that passive reactors should be designed to (1) accommodate hydrogen generation equivalent to a 100-percent metal-water reaction of the fuel cladding, (2) limit containment hydrogen concentration to no more than 10 percent by volume, and (3) provide containment-wide hydrogen control (such as igniters or inerting) for severe accidents. In a subsequent Commission SRM of July 21, 1993, the Commission approved the staff's position with the clarification that the possible use of passive autocatalytic hydrogen recombiners should not be precluded from consideration a priori. Westinghouse plans to use hydrogen igniters for severe accidents.

Several technical issues associated with the use of passive hydrogen control measures need to be resolved before any regulatory endorsement for DBAs or severe accidents. Some of these issues are (1) adequacy and applicability of experimental tests and facilities, (2) potential for degraded performance as a result of catalytic poisons, (3) number and location of hydrogen control measures, (4) need for operability testing or surveillances, (5) impact of recombiner discharge on circulation of the containment atmosphere and on surrounding equipment, and (6) the response time of passive measures to mitigate hydrogen. Staff and industry experience with the design of electrically powered hydrogen recombiners and igniters is well established, whereas experience with passive hydrogen control measures is limited. Nevertheless, the staff believes that benefits are to be gained from the use of passive hydrogen control measures and will evaluate the specific design when it is provided by Westinghouse. The staff will assess the above issues as they relate to the control of hydrogen following a DBA for the AP600.

The staff believes that the use of passive hydrogen control measures for DBA or severe accidents is in concert with the Commission-approved position stated in SECY-93-087. If all technical issues associated with the use of passive hydrogen control measures are resolved, the staff will approve their use for the particular purpose proposed. The results of the staff's review of this issue will be given in the AP600 FSER.

X. Design Basis Accident and Long-Term Severe Accident Radiological Consequences

Although a number of technical issues remain to be resolved, the AP600 containment design could represent an enhancement in safety over current designs, insofar as it is expected to maintain design-basis accident (DBA) pressures and temperatures below containment design values for 72 hours without operator action and without the use of active systems. In addition, the failure modes of the containment heat removal system are independent of the scenarios that could lead to containment challenges which may result in a higher level of reliability of containment cooling than for currently operating plants. The passive containment cooling system cools the containment externally through natural convection and evaporation on the containment shell and does not rely on active systems internal to the containment. The number of containment penetrations is significantly less than in containments for operating reactors.

The staff, however, is concerned about the capability of the containment to mitigate offsite dose consequences in accidents that progress beyond the design basis, including accidents involving significant core damage. In particular, during severe accident conditions, the AP600 containment may remain at elevated pressures for extended periods. Consequently, any leakage path or containment bypass would result in higher leakage rates than those in the current generation of containment designs. Additionally, in the absence of active removal systems internal to the containment, the concentration of airborne radionuclides would be higher and would remain higher for extended periods. The range of uncertainty in staff estimates of the concentration of airborne radionuclides would be greater than for currently operating plants because of the reliance of the AP600 design solely on natural processes for fission product removal, in contrast to operating plants with an internal containment spray system.

The combined effect of elevated containment pressure and higher concentrations of airborne radionuclides for longer periods could lead to higher offsite dose consequences after a core damage accident. Accordingly, the staff believes that Westinghouse should consider improvements internal to the containment that would mitigate the elevated containment pressure and the higher concentration of airborne radionuclides following a severe accident. Because such improvements to the AP600 design would be intended to address severe accidents rather than the DBA, the staff anticipates that single-train, active, non-safety-related systems with non-safety-grade support systems would be acceptable. Nevertheless, such systems would be designed, operated, maintained (in conformance with 10 CFR 50.65), and tested to ensure availability and reliability, and would be supported by reliable power sources and equipment cooling systems. Finally, procedures for use of the systems and appropriate personnel training would be provided.

In light of the enhanced safety that is expected from the existing design features discussed above and contingent upon inclusion of additional features for decreasing pressure and removing airborne fission products following a severe accident, the staff will allow flexibility in some aspects of the DBA dose assessment.

The staff believes that this position on appropriate consideration for enhanced safety features is consistent with the framework previously outlined to the Commission with the proposed rule change to 10 CFR Parts 50, 52, and 100, "Reactor Site Criteria..." (59 FR 52255). In particular, among the reactor design characteristics and proposed operation that will be taken into consideration by the Commission, the proposed rule includes "[T]he extent to which the reactor incorporates unique, unusual, or enhanced safety features having a significant bearing on the probability or consequences of accidental release of radioactive materials" (emphasis added). This addition to the phrase that currently exists in Part 100 is intended to establish the framework for consideration of certain safety enhancements in the assessment of the consequences of accidental releases.

In performing DBA dose assessments for the AP600 design, the staff plans to use the accident source terms in NUREG-1465, "Accident Source Terms for Light-Water Nuclear Power Plants" (see SECY-94-300, "Proposed Issuance of Final NUREG-1465"). The staff also plans to use the framework presented in the proposed changes to 10 CFR Parts 50 and 100 (SECY-94-194, "Proposed Revisions to 10 CFR Part 100 and 10 CFR Part 50, and New Appendix S to 10 CFR Part 50"). The acceptance criterion in the proposed rule is in terms of total effective dose equivalent (TEDE) for the 2-hour period giving the highest dose. The DBA is also considered for the assessment of control room habitability issues. GDC 19, "Control Room," of Appendix A to 10 CFR Part 50 requires that "[A]dequate radiation protection shall be provided to permit access and occupancy of the control room under accident conditions without personnel receiving radiation exposures in excess of 5 rem whole body, or its equivalent to any part of the body, for the duration of the accident." Consistent with the proposed staff position to use the TEDE criterion for the offsite radiological consequence assessments for DBAs, the staff also plans to use 5 rem TEDE (instead of 5 rem whole body) as an acceptance criterion for the AP600 control room habitability (GDC-19) design review. In addition to the Commission papers identified above, the source term issue was also discussed in SECY-90-016 and SECY-93-087.

Westinghouse's proposal for performing the DBA dose assessments agrees with the NRC's approach in most ways. There are three departures. First, Westinghouse proposes that core release fractions identified in NUREG-1465 for low-volatile elements be reduced (reduction factors of five for strontium, barium, and the cerium group and a reduction factor of two for the lanthanide group). Secondly, Westinghouse contends that the duration for the exclusion area boundary dose assessment should be the first 2-hours of the DBA or, "alternatively, the first portion of the accident including the time up to the initiation of the core damage sequence plus the first two hours after onset of core damage." Lastly, Westinghouse has suggested that the 25 rem TEDE dose guideline value identified in the proposed revisions to 10 CFR Part 50 and Part 100 be increased. The second and third departures were included in Westinghouse's comments of June 2, 1995, to the NRC on the proposed changes to 10 CFR Parts 50 and 100.

Westinghouse proposes to use lower release fractions of low-volatile fission products than those presented in NUREG-1465. In NUREG-1465, the magnitude of low-volatile fission product release fractions into the containment was based on (1) the results of the expert panel elicitation for NUREG-1150, "Severe Accident Risks: An Assessment for Five U.S. Nuclear Power Plants"; (2) additional research results obtained since the issuance of NUREG-1150; (3) the results of the in-pile severe fuel damage experiments at the Power Burst Facility; and (4) further examination of the Three Mile Island accident. In NUREG-1465, the staff selected the 75th percentile value for the low-volatile nuclides on the basis that it bounds most of the range of research data values. Therefore, the staff plans to use the low-volatile fission product release fractions in NUREG-1465 in its evaluation of the AP600 design.

The timing proposed by Westinghouse for release of fission products into the containment following a reactor accident differs from that in NUREG-1465. The staff indicated in SECY-94-302, "Source Term-Related Technical and Licensing Issues Pertaining to Evolutionary and Passive Light-Water Reactor Designs," that fission product timing is dependent upon reactor type and design and upon the bounding reactor DBA sequences chosen for source term analysis. It was further stated that the staff will evaluate each proposed ALWR design on a case-specific basis to determine appropriate fission product release timing. Using AP600 design-specific analyses, Westinghouse proposed that the release of fission products from the fuel to the containment would begin in about 1 hour. The staff is reviewing this proposal and believes this estimate to be in the correct time range.

The important timing issue is the selection of the appropriate time period for the DBA dose calculations at the exclusion area boundary (EAB). The performance measure used by the NRC to assess the short-term efficacy of engineered safety features (ESFs) is the 2-hour projected dose from a DBA to a hypothetical individual at any location at or beyond the exclusion area boundary. Consistent with past accepted practice, Westinghouse proposes to perform its dose assessment based on the source terms for the first 2-hours of the DBA. An alternate approach proposed by Westinghouse is to perform the assessment based on the first portion of the accident including the time up to the initiation of the core damage sequence plus the first 2-hours after the onset of core damage.

Under the existing 10 CFR Part 100 framework, the 2-hour period reflects the regulatory position that was implemented in 1962, that is, "immediately following onset of the postulated fission product release." That position was consistent with the underlying regulatory guidance outlined in TID-14844, "Calculation of Distance Factors for Power and Test Reactor Sites" which assumed an instantaneous release of fission products. Improved understanding of severe accidents shows that fission product releases to containment do not occur instantaneously and that the bulk of the releases may not take place for about an hour or more. In order to maintain a meaningful ESF performance measure, the use of more realistic source terms (e.g., those in NUREG-1465) should be coupled with the 2-hour time period that results in the highest potential radiological consequence at or beyond the exclusion area boundary.

Therefore, the dose to a hypothetical individual at or beyond the exclusion area boundary should not be in excess of the dose guidelines for any 2-hour period after the appearance of fission products within containment. This position was previously outlined to the Commission with the same proposed rule change to 10 CFR Parts 50, 52, and 100 discussed above. Therefore, the staff plans to use the two hour period that would result in the highest potential radiological consequence in its evaluation of the AP600 design.

Westinghouse has suggested that the 25 rem TEDE dose guideline value identified in the proposed revisions to 10 CFR Part 50 and Part 100 be increased. The proposed rule change uses 25 rem TEDE as a single value dose reference criterion (to be consistent with the recent revision to Part 20) rather than the existing Part 100 dose criteria of 25 rem whole-body and 300 rem to the thyroid. The TEDE is defined as the deep dose equivalent (for external exposures) plus the committed effective dose equivalent (for internal exposures). The deep dose equivalent is the same as the present whole-body dose, while the committed effective dose equivalent is the sum of the doses to six selected body organs times the weighting factors for each organ.

The use of the TEDE criterion would ensure (1) a risk-consistent methodology to assess the radiological impact of all relevant nuclides on all body organs, (2) a uniformity and consistency in assessing radiation risk that may not exist with the separate whole-body and thyroid organ dose reference values in the present regulation, and (3) the effective application of revised accident source terms that reflect additional low-volatile nuclides, other than noble gases and iodine, to be released into the containment. Therefore, the staff plans to use 25 rem TEDE at the exclusion area boundary and 5 rem TEDE for the control room as the reference dose criteria in its evaluation of the AP600 design.

The staff plans discussed above are based on the proposed 10 CFR Parts 50 and 100 framework. The comment period for the proposed rule revision ended, and many comments, including those from Westinghouse, were received by the staff. If Commission-approved changes are made to the proposed rule revision in a time frame consistent with the AP600 review, the staff will incorporate those changes into the AP600 review. The results of the staff's review of this issue will be given in the AP600 FSER.

On May 31 and June 9, 1995, the staff briefed the ACRS on this issue. On June 15, 1995, the staff received a letter from the ACRS addressing this issue. The Committee states:

> We believe that the issues associated with the potential for radioactive leakage and the source term should be treated separately. We believe that the staff position on the source term is appropriate. The radioactive leakage from the proposed containment design, however, should be considered with respect to public risk and the safety goals.

The staff notes the Committee's comments; however, the staff believes its approach is technically sound and appropriate for resolving this issue on the AP600.

SECY-92-277, "Final Safety Evaluation Report for Volume II of the Electric Power Research Institute's Advanced Light Water Reactor Requirements Document," August 10, 1992.

SECY-92-287, "Form and Content for a Design Certification Rule," August 18, 1992.

SECY-92-292, "Advance Notice of Proposed Rulemaking on Severe Accident Plant Performance Criteria for Future LWRs," August 21, 1992.

SECY-92-294, "Acceptance Review of the Westinghouse Electric Corporation's Application for Final Design Approval and Design Certification for the AP600 Design," August 24, 1992.

SECY-92-299, "Development of Design Acceptance Criteria (DAC) for the Advanced Boiling-Water Reactor (ABWR) in the Areas of Instrumentation and Controls (I&C) and Control Room Design," August 27, 1992.

SECY-92-327, "Reviews of Inspections, Test, Analyses, and Acceptance Criteria (ITAAC) for the General Electric (GE) Advanced Boiling Water Reactor (ABWR)," September 22, 1992.

SECY-92-339, "Evaluation of the General Electric Company's (GE's) Test Program to Support Design Certification for the Simplified Boiling-Water Reactor," October 6, 1992.

SECY-92-368, "Final Rule Amending 10 CFR Part 52," October 29, 1992.

SECY-92-381, "Rulemaking Procedures for Design Certification," November 10, 1992.

SECY-92-403, "Acceptance Review of GE Nuclear Energy's (GE's) Application for Final Design Approval (FDA) and Design Certification (DC) of the Simplified Boiling-Water Reactor Design (SBWR)," December 3, 1992.

Westinghouse Electric Corporation
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DCP/NRC0346

June 9, 1995

Document Control Desk
United States Nuclear Regulatory Commission
Washington, D.C. 20555-001

ATTENTION: MR. DENNIS M. CRUTCHFIELD

SUBJECT: DRAFT COMMISSION PAPER ON STAFF POSITIONS ON TECHNICAL ISSUES PERTAINING TO THE WESTINGHOUSE AP600 STANDARD PRESSURIZED REACTOR DESIGN

Dear Mr. Crutchfield:

Westinghouse appreciates the opportunity to comment on the draft policy paper provided in your letter to Mr. Liparulo of May 18, 1995. Specific comments on each of the 10 issues is presented in the attachment 1. The Westinghouse formal comments on the proposed changes to 10 CFR Parts 50, 52 and 100 are presented in attachment 2.

Please contact me if you have any questions concerning these comments.

Brian A. McIntyre, Manager
Advanced Plant Safety and Licensing

/nja

Attachments

Attachment 2

2468A

ATTACHMENT 1

DRAFT COMMISSION PAPER ON STAFF POSITIONS ON TECHNICAL ISSUES PERTAINING TO THE WESTINGHOUSE AP600 STANDARD PRESSURIZED REACTOR DESIGN

WESTINGHOUSE COMMENTS

I. Leak Before Break Approach

The proposed staff position on leak-before-break approves a leakage rate limit of 0.5 gpm for the AP600 at the design certification stage, provided that a margin of at least 1.41 is available between the maximum combined load calculated for the piping and the load at which crack stability is established. The staff believes this margin is necessary to cover as-built pipe routings and materials. Discussions with the staff have indicated that the leak-before-break methodology will be a Tier 2* item in the AP600 design control document. This implies that the methodology can be changed, with staff approval, by the combined license (COL) applicant at the time of COL application.

Westinghouse does not agree with the draft policy position on leak-before-break that has been prepared by the NRC staff. The draft position requirement to use a factor of 1.41 on loads will likely lead to more plant hardware that may be adverse to overall plant safety. This may include pipe supports, pipe whip restraints, and jet shields for the primary coolant loop piping, main steam line, main feedwater line and reactor coolant system auxiliary piping four inches and larger in diameter located inside containment. Westinghouse requests approval to use the absolute sum of the various loadings in order to provide margin on loads. This approach to margin on loads is consistent with General Design Criteria 4, draft Standard Review Plan section 3.6.3, and NRC approved applications of leak-before-break to operating plants. The Westinghouse margin on loads will result in improved safety, plant simplification, and a more economical plant. The design process for AP600 assures that once leak-before-break is demonstrated during the design certification stage, it will be met in all subsequent stages of design and construction.

The safety benefits that can be realized by successful application of leak-before-break include reduced plant congestion, better access for in-service inspection and maintenance, lower radiation dosages, and lower risk of unintended resistance to thermal growth. This is achieved by reducing the number of pipe supports, pipe whip restraints, and jet shields and simplifying the plant design. The result is a safer and more economical plant.

In our discussions with the NRC staff subsequent to issuance of the draft policy paper, we understand that they have the following concerns which resulted in their requirement for the additional margin on load:

1. Leak-before-break application is deterministic in nature and would require a higher level of margins.

2. Application of leak-before-break at 0.5 gpm leak rate allows smaller pipes to be qualified. Smaller pipes require additional margins.

3. Only preliminary piping analyses will be available during the design certification review of the AP600. The staff would like additional margins at this stage to ensure that the as-built AP600 meets the requirements applied to current plants.

The following paragraphs delineate how the AP600 addresses these staff concerns:

The Westinghouse position on leak-before-break includes adequate margins for uncertainties. A margin of 10 on leak rate is used to address uncertainties in detecting and calculating the leakage from a through wall crack. The leakage crack length will produce a leak rate that is 10 times the detectable leak rate. Uncertainties in the calculations for crack stability are addressed by applying the absolute sum of the loads to a through wall crack that is twice as long as the leakage crack. Absolute summation of loads would result in an added conservatism in the range of a factor of 1.1 on the average.

The NRC staff position on leak-before-break adds complexity to the Westinghouse position by requiring that crack stability also be verified when 1.41 times the algebraic sum of the loads is applied to the leakage crack. To meet this criterion, the stresses in the pipe would need to be reduced by using a more complicated pipe layout and support configuration. This required reduction in pipe allowable stress increases with the diameter of the pipe. There are other factors that affect the pipe layout and support configuration, such as plant access, maintenance, inspection, and laydown space. The combination of these factors, along with the 1.41 margin on loads in the NRC staff position, will result in more supports on some of the pipe lines and could cause some lines to not qualify for leak-before-break. Pipe whip restraints and jet shields would then be required for those lines that do not qualify.

The Westinghouse design process for the AP600 assures that leak-before-break acceptance criteria can be met at all stages of the plant design and construction. The preliminary pipe stress analysis is completed during design certification. The pipe layout and support configuration are controlled by three dimensional electronic model of the plant. Preliminary vendor data is used for valves and equipment to assure design feasibility. Pipe materials and welding processes are selected. Piping isometric drawings are prepared for input into the pipe stress analysis. The pipe stress analysis is performed for the limiting loading conditions and minimum material strength properties, and the leak-before-break bounding analysis curves are met.

During the beyond-design-certification engineering stage, the final pipe stress analysis is performed. Firm vendor data is used for valves and equipment. Minor changes to the pipe layout and support configuration are controlled by the electronic plant model. Revised isometric drawing are prepared, as required, for input into the pipe stress analysis. The ASME piping Design Specifications are prepared to provide a complete basis for the piping design. Pipe stress analysis is performed in accordance with the Design Specifications to satisfy all ASME Code requirements. Refinements in the pipe stress analysis methods may be used to meet the leak-before-break bounding analysis curves. Support design and stress analysis is performed to confirm the design interface for the pipe stress analysis. The COL applicant will use this final analysis to obtain the combined license.

During construction the procurement documents are reviewed to assure conformance with the final pipe stress analysis. New pipe stress analysis is performed for a limited number of lines as required. Construction tolerances are utilized to assure that construction closely follows the design. After construction the as-built configuration and materials are reviewed to assure conformance with the final pipe stress analysis. As-built tolerances for pipe stress analysis are used to reconcile the as-built configuration with the final pipe stress analysis.

The Westinghouse position is that this margin should not be taken from the plant design organization so that the full benefit of leak-before-break can be applied to the design stage rather than at the time of each COL application. There is a greater benefit in not designing additional supports than removing them at the COL stage.

In summary, the Westinghouse approach is to use the absolute sum of the various loadings in order to provide margin on loads. This approach is consistent with General Design Criteria 4, draft Standard Review Plan section 3.6.3, and NRC approved applications of leak-before-break to operating plants. The Westinghouse approach will result in improved safety, plant simplification, and a more economical plant. The design process for AP600 assures that once leak-before-break is demonstrated during the design certification stage it will be met in all subsequent stages of design and construction.

II. Security Design

The staff's understanding of the proposed approach to security for the AP600 is correct. Westinghouse is updating the AP600 SSAR to reflect this change to the design and is in the process of developing a Security Design Report that will be submitted to the staff for review.

III. Technical Specifications

Westinghouse agrees with the staff's assessment of the status and approach to resolution for this issue.

IV. Initial Test Program

Westinghouse agrees with the staff's assessment of the status and approach to resolution for this issue. Meetings have been held with the staff to discuss their concerns in this area. Westinghouse is developing new abstracts for the initial test program and will submit these to the staff for review.

V. Passive System Thermal-Hydraulic Performance Reliability

Westinghouse agrees with the staff's assessment of the status and approach to resolution for this issue. A schedule has been developed and submitted to NRC for the efforts associated with resolution of this issue.

VI. Regulatory Treatment of Nonsafety Systems

The position presented in the paper delineates the key issues associated with resolution of the RTNSS issue. These issues are; post 72 hour actions, acceptance of the baseline PRA, adverse systems interactions and technical specifications for the RTNSS important systems. Westinghouse agrees that these are the important issues and that the staff and Westinghouse are meeting to pursue resolution.

VII. Containment Performance

Westinghouse agrees with the staff's assessment of the status and approach to resolution for this issue using both probabilistic and deterministic criteria.

VIII. External Reactor Vessel Cooling

Westinghouse agrees with the staff's assessment of the status and approach to resolution for this issue. Meetings have been held with the staff concerning the technical aspects of in-vessel retention.

IX. Passive Hydrogen Control Measures

Westinghouse agrees with the staff's assessment of the status and approach to resolution for this issue. Additional information will be provided to the staff on the performance of autocatalytic hydrogen recombiners.

X. DBA and Long Term Severe Accident Radiological Consequences

The first sentence of the third paragraph refers to the higher leakage rates that would exist for the AP600. This is misleading and incorrect. The sentence should be revised to read, "The effect of higher concentrations of suspended radionuclides for longer periods could lead to ...".

In the fifth paragraph there is the statement that the staff " . . . will allow flexibility in some aspects of DBA dose calculations." This does not specifically identify what the flexibility will be.

In the seventh and eighth paragraphs the NRC refers to Westinghouse's use of the EPRI source term. While Revision 3 of the SSAR reflects the EPRI source term, Westinghouse has presented to the NRC a commitment to change source terms to use the new NRC source term from NUREG-1465 in almost every aspect. The results of this revised analysis will be submitted to the staff in June, 1995. The paragraphs present an image of Westinghouse as holding onto an approach that the NRC is opposed to and that we have made no progress since submitting the SSAR in 1992. It is recommended that the first three sentences of paragraph seven be deleted since they do not reflect Westinghouse positions that have been presented to the staff. It is also recommended that following be added to the end of the paragraph:

Westinghouse's proposal for performing the DBA dose calculations agrees with the NRC approach in most ways. There are three departures. First, Westinghouse contends that the duration for the site boundary accident doses should be the first two hours of the accident or, alternatively, the first portion of the accident including the time up to the initiation of the core damage sequence plus the first two hours after onset of core damage. Secondly, Westinghouse proposes that core release fractions identified in NUREG-1465 for low-volatile elements be reduced (reduction factor of five for Sr, Ba, and the cerium group and a reduction factor of two for the lanthanide group). Lastly, Westinghouse has suggested that the 25 rem TEDE dose limit identified in the proposed revisions to 10 CFR 100 and 10 CFR 50 be increased.

Paragraph eight should have the first sentence deleted since the referenced EPRI source term is no longer germane to the AP600. The table of release fractions that follows paragraph eight should also be deleted.

The third sentence of paragraph ten should be revised, replacing "Westinghouse proposes to calculate doses ..." with "In past practice doses have been calculated ...".

Also in the tenth paragraph, reference to a 25 rem TEDE dose limit should be removed. Its presence in this paragraph is not appropriate since it should be addressed as a separate subject (and it is the main subject of the following paragraph). Its use presumes a final acceptance of the limit when it is only a proposed limit at this time. Therefore, the phrasing should be changed from " . . . the dose to an individual should not be in excess of 25 rem TEDE . . . " to " . . . the dose to an individual should not be in excess of the defined limits . . . ".

In paragraph eleven, the use of "proposed" should be revised to "proposes". The current wording implies that the dose limit proposed in 60 FR 10810 - February 28, 1995 has been finalized. Adopting this position prior to the Commission dispositioning public comments on the proposed changes to 10 CFR 50, 52 and 100 is premature.

ATTACHMENT -

NTD-NRC-95-4474

Westinghouse
Electric Corporation

Energy Systems

Nuclear Technology Division

Box 355
Pittsburgh Pennsylvania 15230-0355

June 2, 1995

Mr. John C. Hoyle
Secretary
U.S. Nuclear Regulatory Commission
Washington, DC 20555-0001

Attention: Docketing and Service Branch

Dear Mr. Hoyle:

Subject: Comments on Proposed Rule -- 10 CFR Parts 50, 52, and 100, "Reactor Site Criteria Including Seismic and Earthquake Engineering Criteria for Nuclear Plants" (60 *Federal Register* 10810 - February 28, 1995)

We believe that the proposed rulemaking, associated regulatory guides, and standard review plans should contain unambiguous language, and clear and consistent technical guidance to establish a stable licensing basis for the siting of future nuclear power plants. And, we further believe that a stable licensing basis will create the environment for a successful rebirth of nuclear power plant construction.

As a major NSSS vendor who has a major stake in the success of nuclear power industry, Westinghouse personnel over the past four years have participated in industry advisory committees and task forces associated with NRC's proposed rulemaking referenced above. Through this participation, many Westinghouse concerns regarding the proposed rulemaking have been addressed. Specifically, the concerns were captured in the letter from the Nuclear Energy Institute (W.H. Rasin to J.C. Hoyle) dated May 12, 1995 and in the letter from EPRI ALWR Programs (A. Machiels to J.C. Hoyle) dated May 12, 1995.

Since the proposed rule revision will have an enormous impact on the future of nuclear power, we are augmenting the comments provided by the industry associations with our own comments. Our comments are provided in Enclosure 1.

We want to commend the NRC for addressing industry concerns in a very professional manner through the previous round of comments on this proposed rule. We hope that the Staff will do an equally diligent job in addressing the concerns of the industry through this round. Westinghouse would be pleased to meet with the Staff to discuss any comments offered in this letter.

Very truly yours,

N. J. Liparulo, Manager
Nuclear Safety Regulatory and Licensing Activities

Enclosure

cc: James S. Taylor, NRC
William T. Russell, NRC
David L. Morrison, NRC
Leonard Soffer, NRC
Andrew J. Murphy, NRC

ENCLOSURE 1

Comments on Proposed Rule -- 10 CFR Parts 50, 52, and 100, "Reactor Site Criteria Including Seismic and Earthquake Engineering Criteria for Nuclear Plants"

NON-SEISMIC

Comments on the proposed revision (59 Federal Register 52255, October 17, 1994) to paragraph (a)(1) of part 50.34 of the Code of Federal Regulations.

1.1 Comments on the identification of the two hour interval to be used for calculating the dose at the Exclusion Area Boundary

The proposed time period for calculating the Exclusion Area Boundary (EAB) dose is "any 2 hour period following the onset of the postulated fission product release." Since the dose calculated over the time interval has value only relative to the potential for an individual to be exposed, the dose interval should bear a relationship to the presence of a population in the vicinity to the plant. Presently, EAB doses are calculated based on the assumption that the people in the low population zone are at the site boundary for a two hour period at the beginning of the accident, thus conservatively calculating the potential dose during the two hour interval over which evacuation of this zone is assumed to take place.

With the implementation of the source term described in NUREG-1465, "Accident Source Terms for Light-Water Nuclear Power Plants," February 1995, the release of activity from the core is modeled as occurring over a period of time instead of the instantaneous release assumed in Regulatory Guides 1.3 and 1.4. This creates the likelihood that the calculation of the dose over the 0 to 2 hour time interval will not result in the most conservative determination. Additionally, the development of passive plant designs has demonstrated that it is feasible to design plants to substantially delay the onset of core damage in the event of an accident. Instead of core damage initiating at the very beginning of the postulated large break Loss-of-Coolant Accident, it has been shown that the core damage will be delayed for approximately an hour. This is a substantial improvement in plant safety relative to currently operating plants. Theoretically, it would be possible to design a plant such that the onset of core damage would be more than two hours after accident initiation. In this instance, a two hour dose calculated from the beginning of the accident would not be significant since the only radioactive material that would be available for release would be the activity from the reactor coolant which would enter the containment building as a result of the accident.

With the "sliding dose window" concept, the interval over which doses would be calculated is not linked to any specific occurrence; not to the beginning of the accident, not to the onset of the gap release phase, and not to the onset of the core melt phase. Specifying that the interval for the EAB dose determination should be the two hours over which the highest doses would be accumulated is conservative but, since there is no direct link to any particular aspect of the accident sequence, there is a sense of the arbitrary that detracts from the technical authority that should be present in this document. The "sliding dose window" ignores the dose that would be accumulated during the time period between the accident initiation and the two hour interval of highest dose. It could, of course, be argued that the population in the vicinity of the plant leaves but that other members of the public initially at a far distance from the plant site choose to approach the plant during the time of the site emergency and at the precise time interval when they will accumulate the maximum dose. While theoretically possible, this is not an appropriate model.

A more reasonable approach would be to modify the use of the two hour dose concept, replacing it with a time interval of two hours starting at the onset of core damage plus the time interval between accident initiation and the onset of core damage. This has the advantage that it is linked to the beginning of the accident and thus has a rational connection with the concept of notification of the public and their evacuation. It does not ignore the dose accumulation that would occur prior to core damage. It also has the regulatory advantage in that the EAB dose calculation is not susceptible to being made trivial due to an extensive delay in reaching the beginning of core degradation.

The proposed use of a "sliding dose window" attempts to address the issue of evaluating the capabilities of the containment and other safety features to limit release of activity to the environment but, as discussed above, in its attempt to capture the period of greatest activity release, it introduces distortions into the determination of potential dose to the public. The evaluation of the ability of the plant to limit accident releases to the environment should encompass a sufficient portion of the accident duration to be said to characterize the event. For the postulated large break LOCA with core melt, an appropriate time interval would be 24 hours since, after this point, the accident is essentially complete. There would be continuing minor releases of activity to the environment but these would not be significant when compared to the first 24 hours. The releases over this 24 hour period would have to be demonstrated as being within some bounding value. One approach would be to specify that the EAB dose over the first 24 hours not exceed twice the identified dose limit for the "two hour" EAB dose. The calculation of a 24 dose at the EAB should not be construed as the consideration of the presence of any person at the EAB for that period; the determination of dose is only a surrogate used for evaluating the ability of the containment and other safety features to appropriately limit the release of activity to the environment.

1.2 Comments on the TEDE (Total Effective Dose Equivalent) dose limit

The proposed revision specifies a dose limit of 25 rem TEDE. From SECY-94-194, the approach used in determining this limit is based on starting with the current dose criteria of 300 rem thyroid and 25 rem whole body and determining the risk of latent cancer fatality associated with these combined doses. The resultant risk of latent cancer fatality is 2.7×10^{-2} (2.5×10^{-2} from the whole body dose and 2.0×10^{-3} from the thyroid dose). It is noted that this risk determination neglects the dose contribution from the remainder of the source term identified in TID-14844. That source term includes, in addition to the iodines and noble gases, one percent of the solids in the fission product inventory. This portion of the source term was not taken into consideration in the calculation of whole body doses or thyroid doses because the contribution is not significant but it has been included in the evaluation of in-containment radiation environment following the postulated accident. If these "other nuclides" are taken into consideration, the risk associated with the current dose methodology and source term is greater than the 2.7×10^{-2} that was determined and would lead to higher TEDE dose limits.

In addition to assuming a risk factor of 2.7×10^{-2}, SECY-94-194 also assumes that the dose is quickly accumulated over the designated two hour interval, thus justifying the risk coefficient of 10^{-3} per rem instead of the risk coefficient of 5×10^{-4} per rem that is associated with dose accumulation over the longer term (i.e., a period of days or more). Using this approach, the dose limit was calculated to be 27 rem but was reduced to 25 rem.

The calculation of 27 rem TEDE is based on the inappropriate assumption that the dose accumulation occurs over a short time. For the postulated two hour exposure interval, most of the anticipated dose would be as a result of long term dose accumulation from the nuclide body burden. Only a small fraction of the total dose would be acute dose from the immersion in the cloud of activity. This is especially true when taking into account the new source term set forth in NUREG-1465. Instead of basing the TEDE dose limit on the risk associated with short term dose accumulation, it should be based on an appropriate combination of short and long term dose accumulation. It is appropriate to assume that the acute dose is substantially below this value - say 10 rem. Based on studies for the AP600 plant, the acute dose is expected to be well below this value. This 10 rem gamma body dose translates to a 1×10^{-2} risk for latent cancer fatality and corresponds to 10 rem TEDE. Using a risk coefficient of 5×10^{-4} per rem, the additional allowable TEDE dose that could accumulate over the long term without exceeding a risk of 2.7×10^{-2} would be 34 rem. The total is thus 44 rem TEDE.

However, this approach still does not accurately reflect the dose limit that is associated with the identified level of risk. As indicated in SECY-94-194, in the original determination of the level of risk, the risk identified for the 300 rem thyroid dose is 2×10^{-3}. Since 300 rem thyroid translates to 9 rem TEDE, the risk coefficient associated with this exposure is 2.2×10^{-4} per rem. For the postulated LOCA with core melt, more than half of the accumulated TEDE dose is expected to be from dose to the thyroid. If it is conservatively assumed that only 25% of the non-acute dose is from thyroid dose (and thus, 25% of the risk of 1.7×10^{-2} allocated for the non-acute dose is associated with the thyroid dose), this results in 19.3 rem TEDE associated with the thyroid dose and 25.5 rem TEDE associated with the remaining organ contributors. The resulting total of 55 rem TEDE could be rounded down to 50 rem. This would be a more appropriate TEDE dose limit than the 25 rem specified in the proposed revision to 10 CFR 50.

1.3 Comments on the concept of "caps" for specific organ doses

The idea of "caps" on the fraction of TEDE dose limit that could be associated with any specific organ is presented for discussion in SEC-94-194. This concept of having specific organ dose limits in addition to the overall TEDE dose limit adds to the complexity of the approach and implies that the methodology used in generating the TEDE dose limit is not viewed as valid. The TEDE dose limit is based on an identified level of risk. If the basis for the TEDE dose limit is valid, there is no need for caps on specific organ doses. The use of limits on specific organ doses in addition to the overall TEDE dose limit would result in an unnecessary complication to the rule and would not reduce the risk to the public.

SEISMIC

Westinghouse supports NRC's decision to move guidance material from the proposed rule to the proposed regulatory guides. We also support NRC's decision to eliminate the "dual" deterministic and probabilistic analyses from the proposed rule. We, however, are concerned that retaining deterministic evaluations in SRP 2.5.2 will lead to confusion as to whether future licensees will also need to perform a deterministic analysis even though such an analysis is only recommended for NRC staff to perform as a "sanity" check. This additional deterministic analysis will add to instability in the licensing process and increase a future license applicant's seismic analysis costs (in defending its probabilistic analyses) without any additional benefit to public health and safety. We recommend that references to deterministic analyses be removed from all documentation associated with the proposed rule revision.

Westinghouse shares NEI's concern with respect to the type of analyses needed to construct a new plant on an existing approved site, using the proposed rule and associated proposed regulatory guides. We also believe that site characterization analysis for existing sites should be confirmatory in nature and of "limited scope," rather than "full scope" as required for new sites.

There are several phrases that are used in the proposed rule that should be modified to make the rule more stable from a licensing point of view. Since these phrases are used in several places, only the phrase and not the location, are identified below. We suggest that these phrases and others that are similar in nature be modified as well.

1. "...certain structures, systems, and components" should read: "certain structures, systems, and components as identified in Regulatory Guides xxx." By referencing the regulatory guides, the vagueness of the statement is eliminated from the rule and the description of the structures, systems and components can be changed, if necessary, via changes to the regulatory guides.

2. "...without loss of capability to perform their safety functions" should read: "...without loss of capability to perform their ~~safety~~ intended functions." The components perform a function and not a "safety" function -- components may be a part of a safety system or a non-safety system. There are other sentences which have similar phraseology -- for example, item 3 below. These sentences should be similarly modified.

3. "The required safety functions of structures, systems, and components must be assured..." should read: "The required ~~safety~~ functions of structures, systems, and components must be assured per the guidance provided in Regulatory Guide xxx..." The underlined phrase shows that the regulatory guide contains guidance as to how a future license applicant can provide "assurance."