July 22, 1996

Mr. Regis A. Matzie
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Combustion Engineering, Inc.
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Dear Mr. Matzie:

I am pleased to enclose, for your information, an advance copy of a paper entitled, "Safety Reviews of Next-Generation Light-Water Reactors." I will present this paper at the forthcoming ENS Topical Meeting, TOPNUX'96 "Economic Power for the 21st Century: The Coming Generation of Nuclear Reactors," on September 30, 1996. At this meeting, I will be discussing the NRC's licensing approach for design certification applications.

Sincerely,

Original Signed By
WILLIAM T. RUSSELL

William T. Russell, Director
Office of Nuclear Reactor Regulation

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SAFETY REVIEWS OF NEXT-GENERATION LIGHT-WATER REACTORS

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ABSTRACT

The U.S. Nuclear Regulatory Commission (NRC) is reviewing three applications for design certification under its new licensing process. The Advanced Boiling Water Reactor design and the System 80+ design have received final design approvals. The AP600 design review is continuing. The goals of design certification are to achieve early resolution of safety issues and to provide a more stable and predictable licensing process. NRC also reviewed the Utility Requirements Document of the Electric Power Research Institute and determined that its guidance does not conflict with NRC requirements. This review led to the identification and resolution of many generic safety issues.

The NRC has determined that next-generation reactor designs should provide a higher level of safety for selected technical and severe accident issues. Accordingly, NRC developed new safety requirements for these designs based on (1) operating experience, including the accident at Three Mile Island, Unit 2; (2) the results of probabilistic risk assessments of current and next-generation reactor designs; (3) early efforts on severe accident rulemaking; and (4) research conducted to address previously identified generic safety issues. These additional requirements will become design-specific regulations for each design in the design certification rulemakings.

The U.S. Nuclear Regulatory Commission (NRC) is reviewing three applications for design certification under its new licensing process (Ref. 1). The first application for design certification was submitted by GE Nuclear Energy for the U.S. Advanced Boiling Water Reactor (ABWR) standard design. The second application was submitted by ABB Combustion Engineering, Inc., for the System 80+ standard design. The NRC staff has completed its technical review of these two evolutionary designs and has issued final safety evaluation reports (FSERs). The FSERs for the ABWR (Ref. 2) and System 80+ (Ref. 3) were issued in July and August 1994, respectively. The third application was submitted by Westinghouse Electric Corporation for certification of the AP600 standard design. The NRC staff issued a draft safety evaluation report (DSER) on the AP600 (Ref. 4) in November 1994 and a DSER supplement (Ref. 5) addressing testing and analytical code applicability in April 1996. The staff is continuing its detailed technical review on the remaining design issues for AP600, including the reliability of passive system thermal-hydraulic performance, the regulatory treatment of non-safety-related systems, and analytical codes. Policy issues affecting the AP600 review (Ref. 6) were submitted to the Commission for its review in June 1996. NRC expects to complete its review of the AP600 design and issue the FSER in 1997.

DESIGN CERTIFICATION

On May 18, 1989, NRC issued 10 CFR Part 52 (Ref. 1), which reformed the licensing process for future nuclear power plants. This new licensing framework provides the opportunity to resolve siting and design issues before large commitments of resources are made to construct and operate new nuclear power plants. The key procedural device for bringing about enhanced safety and the early resolution of licensing issues is the design certification process. Design certification provides a more stable and predictable licensing process by resolving all of the safety issues for an essentially complete nuclear power plant design and approving the design through the NRC's rulemaking process.

The new design certification process significantly improves the prior design approval process (Ref. 7). First of all, the scope of the design to be certified is different. In the past, the NRC would approve either a nuclear steam supply system (NSSS) or the balance-of-plant portion of the nuclear plant design. For design certification, the NRC requires an essentially complete power plant design that includes all of the structures and systems of the plant, except for site-specific design features such as the ultimate heat sink. This requirement minimizes the number of design interfaces that must be reviewed and verified. Another improvement in the process is the use of rulemaking for certification of designs rather than approval of a standard design at the staff level. As a result, applicants who reference a certified design in their applications are assured that the safety issues that are resolved during design certification will not be reconsidered during the plant licensing process. Finally, the Commission certifies the standard designs for a duration of 15 years and applies a more stringent backfit standard than was used in previous design approvals. This new approach of resolving all safety issues, placing the resolutions under a restrictive change process that applies to both the regulator and the applicant for design certification, and extending the duration of approval provides a more stable and predictable licensing process.

UTILITY REQUIREMENTS DOCUMENT

In parallel with its review of the design certification applications, the NRC also reviewed the "Advanced Light Water Reactor (ALWR) Utility Requirements Document (URD)" that was prepared by the Electric Power Research Institute (EPRI). This compendium of technical requirements, which are applicable to both evolutionary and passive light-water reactor (LWR) designs, is intended to be a comprehensive statement of utility requirements for the design, construction. and performance of nuclear plants for the 1990s and beyond. Participants in the program include utilities with nuclear plant experience, NSSS vendors, architect-engineering firms, and consultants in related fields.

EPRI began its program in 1983 by working with the NRC staff to identify and resolve key safety and licensing issues. This joint effort resulted in a process whereby proposed resolutions to the unresolved and generic safety issues applicable to next-generation LWRs (Refs. 8 and 9) were identified. In 1985, two new phases were added to the EPRI program: the development of EPRI's URD for evolutionary plants and the assessment of small-plant options. This assessment resulted in the development of a URD for passive plant designs.

EPRI's URD was designed to consistently resolve common operating problems, issues generically applicable to designs, severe accident issues, and certain unresolved and generic safety issues. The URD represented the consensus of industry regarding the best approach for resolving these concerns. In addition, EPRI recognized the need to establish a higher standard for advanced designs. Therefore, additional standards were developed for designers to use to address events beyond the design basis of a nuclear plant. These events are commonly referred to as "severe accidents." The URD is to be used with companion documents, such as utility procurement specifications, that cover the remaining technical requirements applicable to new plant projects. It was also designed to identify early in the design process any major concerns about design concepts for LWRs in which passive safety systems will be used. However, because EPRI's URD is an agreement between the vendors and the nuclear power utilities, it only identifies those features that utilities want in future designs. The URD has no legal or regulatory status. It is not intended to demonstrate complete compliance with the NRC's regulations, regulatory guidance, or policies, nor is it intended to be used as a basis for supporting either a final design approval or a design certification for a specific reactor design.

The NRC staff's review of the URD was performed in accordance with its standard review plan and other review guidance (Refs. 9 and 10). In addition, the Commission directed the NRC staff (Ref. 11) to review the URD to ensure that it is sufficient to allow the staff to evaluate the resolution of severe accident issues and the incorporation of experience from operating events in current designs. The NRC staff used both deterministic and probabilistic methods of evaluation in considering how the URD addressed these issues through the specific design criteria and guidelines that it provided for performing a probabilistic risk assessment (PRA). The NRC staff concluded in its safety evaluation report (Ref. 12) that the guidance established in the URD for evolutionary and passive plant designs does not conflict with current regulatory requirements, has contributed to the improved safety of next-generation reactor designs that used the URD during their development, and, subject to the resolution of the identified design-specific issues, is acceptable.

ENHANCED SAFETY

In addition to its efforts to improve the stability and predictability of the licensing process through design certification, NRC determined that next-generation reactor designs should achieve a higher level of safety for selected technical and severe accident issues than current operating plants. In its policy statement on the resolution of severe accident issues (Ref. 13), the Commission expressed its expectation that vendors would achieve a higher standard of severe accident safety performance for new standard plant designs than for previous designs. The policy affirmed the Commission's belief that reactor designers could show that new designs are acceptable with regard to severe accident concerns if the design meets the current NRC regulations, including requirements stemming from the accident at Three Mile Island (TMI); if it demonstrates technical resolution of unresolved safety issues and the medium- and high-priority generic safety issues (Ref. 8); and if it considers the severe accident vulnerabilities from a design-specific PRA and the insights from severe accident research. The Commission also indicated its intent to continue a defense-in-depth philosophy and to maintain an appropriate balance between accident prevention and consequence mitigation (Ref. 13).

On the basis of this policy guidance, a new review approach evolved for the next-generation LWR designs. First of all, in addition to the existing safety requirements, the NRC staff also reviewed the design certification applications for incorporation of important lessons learned from operating experience by using NRC bulletins and generic letters. Then, additional requirements for selected technical and severe accident issues for next-generation reactor designs were developed, based on (1) operating experience, including the TMI-2 accident; (2) the results of PRAs of current and next-generation reactor designs; (3) early efforts on severe accident rulemaking; and (4) research conducted to address previously identified generic safety issues (Refs. 14 and 15). These issues included anticipated transients without scram, mid-loop operation, station blackout, fire protection, intersystem loss-of-coolant accidents, and hydrogen generation and control. In addition to these issues, other issues evolved as a result of the staff's review of the specific plant designs. The additional requirements related to technical and severe accident issues were resolved during the individual design reviews and will be codified in the rulemakings for design certification (Ref. 16).

During the same period in which the NRC staff was developing new requirements to address severe accident issues, EPRI was describing how these issues were addressed in its URD. EPRI's containment performance study (Ref. 17) provided an overview of the basis for closure of severe accident containment performance issues and presented a systematic review of 23 containment challenges as well. EPRI's study showed how the URD addressed each of the identified challenges. In a sense, this effort demonstrated that the design requirements proposed by EPRI included an adequate safety margin to address severe accidents.

The NRC's new review approach recognized the wealth of information from severe accident research that had been generated since the accident at TMI. General agreement on the major severe accident challenges to the reactor and containment designs had been reached, based upon extensive international and NRC research. However, uncertainties remained regarding initiation and progression of severe accidents, and research to address these uncertainties is continuing. Therefore, the challenge for design certification is to resolve severe accident issues, notwithstanding these uncertainties. Severe accident research and knowledge had increased to a level such that a plant designer could include measures to further reduce the risk from severe accidents. As a result, the approach to closure of severe accidents under design certification was to review the next-generation designs for severe accident requirements that apply to current operating reactors, such as the hydrogen control requirements (Ref. 18), and for additional severe accident challenges to the nuclear reactor and containment. These challenges include a design for ex-vessel core coolability to provide an adequate means of spreading core debris and for flooding the reactor cavity or drywell, and maintenance of containment integrity, or leak tightness, for a specified period following the onset of reactor core damage. Challenges to containment integrity considered during the review included steam explosion, core-concrete interaction, high-pressure melt ejection, hydrogen detonation, and containment bypass.

Information used by the NRC staff in its new review process also included PRA findings, deterministic evaluations, and existing technology relative to experimental data. The role of PRA is an excellent example of this process. The NRC, as previously noted, required a supporting PRA for each design. For the first time, PRA would be available early in the development of the design when modifications could be most effectively implemented. The insights developed during the design-specific PRA review resulted in changes to the design to reduce risk and also identified important information to be considered during construction, testing, and operation of the facility. These insights were captured and documented during the PRA review. As a result of the PRA, the staff focused on those issues that have an impact on the overall safety of the design.

The deterministic analyses were used in addition to the PRA to develop a better understanding of the phenomena of a severe accident event. From a containment performance perspective, an assessment of ex-vessel sequences was essential to determine the ability of the containment to withstand the anticipated thermodynamic as well as hydrodynamic loads. To support the deterministic analyses, the NRC staff used the experimental database to aid in the understanding of the loads associated with the various ex-vessel core-melt phenomena. In situations in which the database was incomplete, the staff assumptions are believed to bound the phenomena in question. The staff concluded for the ABWR and System 80+ designs that sufficient understanding of severe accident phenomena was available to resolve severe accident issues for these designs. That is not to say that all aspects of each individual response of the design to a severe accident is fully understood. Rather, there is sufficient understanding of the phenomena for the staff to conclude that these designs are acceptable.

The following discussion provides examples of how severe accident issues were resolved during the reviews of the evolutionary designs. The first example pertains to the resolution of postulated steam explosions occurring external to the reactor vessel. A quick look would indicate that only one evolutionary design could be acceptable since the ABWR reactor cavity is designed to be dry at the onset of the ex-vessel release, while System 80+ is designed to be wet.

The ABWR has fusible plugs within its drain lines to assure that the quench water will not be released until there is a substantial amount of corium on the floor to cause melting of the fuse plugs. Therefore, the steam explosion issue was determined to be resolved based on a design feature that prevents water in the reactor cavity prior to ex-vessel release. However, in an effort to fully explore the capabilities of the design, the ABWR was analyzed to determine if the critical structures could survive a steam explosion. The analysis demonstrated that the reactor vessel and containment would survive intact. System 80+, on the other hand, is designed to have a flooded reactor cavity prior to ex-vessel release, thereby maximizing cooling of the entering corium. For this design approach, water is expected to be present and, therefore, a steam explosion is also more likely. However, the construction of the reactor cavity walls are unique in that the immediate walls are not the major reactor vessel supports. The results of the structural loads analysis resulting from a steam explosion demonstrated that these immediate walls could be partially destroyed but not the primary support structures. In fact, these immediate walls protected the primary structures. Therefore, it was demonstrated that if a steam explosion were to occur, the System 80+ design would accommodate the expected loads. This example demonstrates the flexibility of the NRC's review approach because the NRC has been able to determine with confidence that the different evolutionary designs can both accommodate external steam explosions.

Core-concrete interaction (CCI) involves the decomposition and chemical interaction of core debris with the concrete containment floor. The extent of CCI and its effect on the containment are influenced by many factors, including the amount of core debris, core debris superheat, core debris composition, the amount of metals within the core debris and concrete floor, the availability of water, the type of concrete, heat transfer mechanisms, debris morphology, and the thickness of the core debris layer. Although many of these factors are specific to the accident sequence and are dependent upon core-melt progression, several of them can be controlled or optimized through the containment design. These factors include providing for actively and passively flooding the reactor cavity, optimizing the reactor cavity floor space by furnishing a large unobstructed area for core debris to spread, supplying a thick layer of concrete to prevent containment liner melt-through in the event of continued CCI, and selection of a type of concrete that either decreases the amount of non-condensible gases generated during decomposition or inhibits radial and axial erosion. In the review of the evolutionary designs, all these factors were evaluated through a combination of analyses and design review. With respect to the analyses, the nuclear industry generally used the MAAP code, whereas the NRC relied upon MELCOR and CORCON. All analyses were performed to achieve the best estimate, and consideration was given to uncertainties in severe accident progression and phenomena. Analytical modeling took into account uncertainty and sensitivity analyses.

High-pressure melt ejection (HPME) and direct containment heating are associated with severe accident sequences at high reactor coolant system pressure that result in vessel failure and core debris ejection, fragmentation, and entrainment into the containment atmosphere. The fragmented core debris mixes and reacts with the atmosphere, causing large pressure and temperature increases that may challenge containment integrity. In addition, the core debris may relocate into contact with the containment shell, leading to melt-through. HPME has generally been associated with pressurized-water reactors because they lack the depressurization ability associated with boiling-water reactors. Therefore, to eliminate HPME as a credible threat to containment integrity for the evolutionary reactor designs, NRC required installation of reliable depressurization systems on all designs. In addition, design features were provided to ensure that a direct pathway did not exist for core debris to be transported from the reactor cavity to the upper containment. The NRC's evaluation of HPME focused on ensuring a reliable depressurization system. This evaluation included an assessment of the power sources (both electrical and air), capacity, valve design, operations and controls, and incorporation of the findings into the emergency operating procedures.

Hydrogen generation and control for the evolutionary reactor designs followed the precedent set by the TMI-2 requirement (Ref. 18). In particular, this regulation specifies the amount of hydrogen generated that must be accommodated through a control system. The hydrogen control system is typically either an inert atmosphere to preclude hydrogen ignition or an engineered ignition system to control the effects of a hydrogen burn. Inertion ensures that regardless of the hydrogen concentration, hydrogen recombination will not occur. An ignitor system deliberately ignites the hydrogen at a low-enough concentration, such that sufficient hydrogen cannot accumulate and recombine in a manner that could challenge the containment integrity.

CONCLUSIONS

The NRC has demonstrated its ability to achieve early resolution of design issues, subject to satisfactory verification of construction and testing, and to provide a more stable and predictable licensing process through its design certification reviews. The interactions between the nuclear industry and the NRC during the parallel reviews of the design certification applications and

EPRI's Utility Requirements Document provided an effective means for resolving the severe accident issues and selected generic safety issues. The site specific issues were bounded to allow for separation of siting reviews from design reviews. Through these efforts, the evolutionary designs (ABWR & System 80+) have achieved a higher level of design safety than currently operating nuclear power plants.

REFERENCES

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6. U.S. NRC, "Policy and Key Technical Issues Pertaining to the AP600 Standardized Passive Reactor Design," SECY-96-128, June 12, 1996.

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10. U.S. NRC, "Standard Review Plan for the Review of Safety Analysis Reports for Nuclear Power Plants," NUREG-0800, July 1981.

11. U.S. NRC Staff Requirements Memorandum, S. Chilk to J. Taylor, February 15, 1991, pertaining to SECY-90-377, "Requirements for Design Certification Under 10 CFR Part 52."

12. U.S. NRC, "NRC Review of Electric Power Research Institute's Advanced Light Water Reactor Utility Requirements Document," NUREG-1242, August 1992.

13. U.S. Federal Register Notice, "Policy Statement on Severe Reactor Accidents Regarding Future Designs and Existing Plants," August 8, 1985, 50 FR 32138.