UNITED STATES
NUCLEAR REGULATORY COMMISSION
ADVISORY COMMITTEE ON REACTOR SAFEGUARDS
WASHINGTON, DC 20555 - 0001

November 7, 2012

The Honorable Allison M. Macfarlane
Chairman
U.S. Nuclear Regulatory Commission
Washington, DC 20555-0001

SUBJECT: LONG-TERM CORE COOLING FOR THE SOUTH TEXAS PROJECT ADVANCED BOILING WATER REACTOR COMBINED LICENSE APPLICATION

Dear Chairman Macfarlane:

During the 599th meeting of the Advisory Committee on Reactor Safeguards (ACRS, Committee), November 1-3, 2012, we completed our review of the NRC staff's safety evaluation regarding the adequacy of long-term core cooling for the certified Advanced Boiling Water Reactor (ABWR) design in the South Texas Project (STP) Units 3 and 4 Combined License Application (COLA). The STP Units 3 and 4 (STP 3 and 4) long-term core cooling performance was also reviewed during subcommittee meetings held on June 23-24, 2010, March 8, June 21, October 4, 2011, and October 2, 2012. During these meetings, we had the benefit of discussions with the representatives of the NRC staff and the Nuclear Innovation North America, the STP applicant. We also had the benefit of the documents referenced.

CONCLUSION AND RECOMMENDATION

1. Long-term core cooling for design basis conditions for STP 3 and 4 will be adequately met pending successful resolution of the downstream effects test program.

2. The downstream effects testing program is based on the applicant's commitment to maintain low levels of fibrous materials and other deleterious materials in the containment. Any future relaxation of these cleanliness requirements would have to be addressed by additional test data and associated analysis.

3. While STP has committed to use test procedures and protocols consistent with current industry practice at the time of the tests, the Committee wants to review the STP downstream fuel effects test procedure prior to testing.

BACKGROUND

On May 8, 2008, the Commission issued a Staff Requirements Memorandum (SRM) stating "The ACRS should advise the staff and Commission on the adequacy of the design basis long-term core cooling approach for each new reactor design based, as appropriate, on either its review of the design certification or the first license application referencing the reactor design." The main focus was the adequacy of the safety systems to provide adequate core cooling over extended time periods when the Emergency Core Cooling System (ECCS) recirculation mode is activated during a design basis accident (DBA).

The ABWR design was certified by the NRC in 1997. It incorporated improvements from the currently operating boiling water reactor (BWR) design. The ABWR design eliminates recirculation piping external to the reactor pressure vessel (RPV). Also, ABWR main steam and feedwater piping connects to the RPV above the core, thus eliminating a large break loss of coolant accident (LOCA) below the top of active fuel.

There are several potential issues associated with long-term core cooling under design basis accident conditions: e.g., availability of water to ECCS pumps; containment accident pressure credit; the potential for non-condensable gases in ECCS piping; and adequate cooling flow through core fuel channels.

DISCUSSION

The general response to loss of core cooling under high-pressure conditions is to depressurize the primary system and provide core cooling using the ECCS. The safety relief valves discharge reactor pressure vessel steam into the suppression pool. The ECCS first draws cooling water from the condensate storage tank. After the condensate storage tank is depleted, ECCS suction is automatically switched to a recirculation mode from the suppression pool. The flow of water into the RPV and the core fuel assemblies requires the combined actions of pumps and valves in the ECCS. Lack of an adequate net positive suction head (NPSH) for a pump or blockage of flow into the core fuel assemblies must be avoided to assure adequate water flow through the core to remove decay heat without fuel overheating and damage.

For long-term core cooling over many days or weeks, the heat released into the suppression pool is removed by the Residual Heat Removal (RHR) system, which draws suction from the suppression pool and pumps the water through heat exchangers cooled by service water from the ultimate heat sink. Interruption of long-term core cooling can occur by disruption of the heat sink or the interconnecting piping, failure of important components, or blockage of the suction line from the suppression pool. These issues are addressed by diversity and redundancy in the design, and by designing for the worst-case environmental conditions. The STP 3 and 4 ultimate heat sink is designed to maintain reactor service water temperature below 35 °C (95 °F) with no makeup for 30 days.

Boiling Water Reactor (BWR) strainer performance issues were evaluated in the mid 1990s after some incidents at foreign and domestic BWRs led to concerns about strainer performance. Evaluation of these issues led to the installation of larger strainers. The extensive work performed on pressurized water reactors (PWRs) to address Generic issue, GSI-191, "Assessment of Debris Accumulation on PWR Sump Performance" has enhanced the level of knowledge of various aspects of ECCS strainer performance. In particular, this work raised additional concerns with respect to the potential blockage of the ECCS suction strainers and downstream fuel assemblies by debris transported through the suppression pool and piping during an ECCS recirculation mode of operation. The main sources of this debris during a design basis accident are: 1) latent containment debris, such as clothing fibers or other fibrous material; 2) debris generated by LOCA jets impinging on surfaces such as insulation or coatings; 3) debris from dirt and sludge that is transported by convective flow in the suppression pool, and 4) chemical precipitates formed in the recirculating water stream. The applicant is a member to the BWR Owners Group and has benefitted from their work on strainer blockage and from the findings of the GSI-191 activities for the fleet of operating PWRs to achieve maximum cleanliness and minimal debris sources.

In STP 3 and 4, no fibrous or calcium silicate insulation is allowed inside the containment, and only reflective metallic insulation is used. To minimize chemical effects due to dissolved metals, no aluminum is allowed inside the containment. The only source of zinc is the inorganic zinc primer in qualified coatings. Thus, in comparison to current operating reactors, minimal debris will be generated from any LOCA blowdown event. Mobilization of latent debris will be reduced, and favorable water chemistry will minimize formation of chemical products. Nevertheless the applicant conservatively assumed a small amount of latent aluminum based on the minimal amount that could be detected and excluded by the STP foreign material exclusion and containment cleanliness programs. The applicant also assumed 1 ft^3 of latent debris fibers will be present. This amount was conservatively based upon operational experience in Japanese ABWRs. The applicant has committed to an operational program to ensure that the primary containment is free from debris consistent with industry guidance reflecting ABWR Operating Experience, Electric Power Research Institute (EPRI) guidelines contained in EPRI TR 1016315, "Nuclear Maintenance Applications Center: Foreign Material Exclusion Guidelines" and Institute of Nuclear Power Operations (INPO) guidance in INPO 07-008, "Guidelines for Achieving Excellence in Foreign Material Exclusion (FME).

To assure adequate performance of suction strainers, the applicant proposed to use the reference Japanese ABWR ECCS suction strainers, sized in accordance with BWR Owners Group Guidance, and supported by testing compliant with Regulatory Guide 1.82, Revision 3. The load combinations of the ECCS strainer met the load combinations in Table 3.9.2 of the design control document. To ensure structural integrity of the strainers, hydrodynamic load development will follow the ABWR design certification methodology, and stresses will be compared to the applicable American Society of Mechanical Engineers Boiler and Pressure Vessel (ASME B&PV) Code Section III allowable for limiting load combinations. The design will meet ASME B&PV Section III, Subsection NC, design requirements for local membrane stresses for the required load combinations, which ensures no local damage to strainer pockets due to loads imposed during vent clearing, condensation bubble collapse, and condensation oscillation. The applicant will use the accepted stress analysis methodology in the non-mandatory ASME B&PV Code, Section III, Appendix A-8000, to account for modeling of the perforated metal sheets in the ECCS strainer. Inspections, tests, analyses, and acceptance criteria (ITAAC) will verify the acceptability of the ASME design report when submitted. Similar cassette-type strainers are used in many PWR applications in the US. The ECCS pump NPSH calculations do not take credit for containment pressure during an accident.

The ECCS strainers are sized based on full-fiber load for the reference Japanese ABWR design, which is much larger than what is required at STP. Large surface area and convoluted suction surfaces disrupt formation of debris "thin bed," protect the strainers, and preserve NPSH margin. To determine debris loading for downstream effects, the STP analysis assumed particulate loadings (primarily from qualified epoxy containment coatings, dirt, and corrosion product sludge in the suppression pool water), reflective metallic insulation shards, a small source of latent aluminum, and 1 ft^3 of latent debris fibers. The erosion products from the exposed concrete and zinc primer in the zone of influence of pipe break as well as the aluminum were considered in the chemical effects analysis. No credit is taken for solubility of aluminum corrosion products or zinc oxide solubility. Because of the large strainer size, blockage is very unlikely and any potential problems with long term core cooling would only be due to flow blockage in the reactor core fuel assemblies by materials that pass through the strainers.

The applicant has committed to perform downstream effects tests with the final fuel design and confirmatory analyses for the ECCS components. The tests will be performed according to the plan developed and implemented for the PWR Owner's Group downstream effects debris testing with regard to debris preparation, the addition of debris, and pressure drop monitoring. For the proposed downstream fuel effects test, the applicant conservatively assumed that all the 1 ft^3 of latent debris fibers will pass through the strainers. Latent aluminum and zinc inventories are consistent with PWR Owners Group assumptions. The applicant will conduct a series of tests in a pumped flow loop similar to facilities used by the industry for PWR. The loop will incorporate a part-length fuel assembly with inlet and spacer geometries representative of the actual fuel design. Flow rates will span the range of the calculated transient mass flow rate through the core during long term core cooling. An approved surrogate material such as aluminum oxyhydroxide will be added over a period of time to simulate the effect of chemical precipitates that might form from aluminum and zinc. The reference experimental protocol would follow a bounding sequence of events expected for the long-term recirculation phase of a design basis accident as predicted by LOCA analyses.

The results of the tests may be affected by factors such as: water chemistry, the use of NUKON fibers as a surrogate for the actual wide-range of organic materials (hair, clothing fiber), the use of silicon-carbide particles as a surrogate for dirt, sludge and corrosion products, the use of a single part-length fuel assembly, the debris addition protocol, the pressure drop modeling of the debris bed, the use of room temperature during the downstream effects testing as well as the use of aluminum oxyhydroxide as an appropriate surrogate for zinc oxide. The detailed test procedure will be provided to the NRC at least six months prior to performing the test and will reflect industry experience with performance of such tests, for example consideration of fuel assembly geometry, debris addition and test protocol, updates on debris surrogate composition or size distribution, the number of tests, and provisions for assessing test variability. The applicant committed to perform multiple tests at the same conditions to quantify test reproducibility.

The applicant's confirmatory analysis is following accepted methodology for downstream effects of debris on ECCS components. The applicant has performed analyses to determine an acceptable maximum level of flow blockage. The acceptance criterion was developed conservatively and specifies that the core void fraction, both in the hot assembly and in the average assembly, remains less than 0.95. The staff imposed a license condition to ensure the issue of downstream fuel test is adequately resolved. We concur with this approach and want to review the detailed test procedure that will be submitted prior to testing.

Additional defense-in-depth analyses showed supplemental cooling from the high-pressure core flooder or engineered bypass paths can each provide sufficient cooling in the event of complete blockage of assembly inlet, and deposition on fuel does not cause significant clad heat-up. In addition, the alternate AC independent water addition mode of RHR allows water from the Fire Protection System to be pumped to the vessel and sprayed in the wetwell and drywell from diverse water sources to maintain cooling of the fuel and containment. The wetwell can also be vented at low pressures to assist in cooling the containment.

In summary, debris generation during DBAs is minimized by the ABWR design and the use of reflective metallic insulation and qualified epoxy coatings. These, together with the large flow-area sump screens, result in acceptable ECCS pump NPSH. The applicant's plan for future STP 3 and 4 specific downstream fuel effects testing will reflect industry experience with the performance of such tests. STP 3 and 4 meet requirements for containment integrity and ECCS NPSH with no credit for containment accident pressure. The ABWR design and monitoring minimize the potential for non-condensable gas accumulation.

We concur with the staff's assessment that long-term core cooling for design basis conditions have been adequately met pending successful resolution of the downstream effects testing.

Sincerely,

/RA/

J. Sam Armijo
Chairman

REFERENCES

1. Staff Requirements Memorandum, "Periodic Briefing on New Reactor Issues, 1:00 p.m., Wednesday, April 30, 2008, Commissioners' Conference Room, One White Flint North, Rockville, Maryland," May 8, 2008 (ML081290255)

2. NRR Memorandum, “South Texas Project Units 3 and 4, Combined License Application – Advanced Safety Evaluation for Chapter 4, Reactor,” February 7, 2011 (ML110340381)

3. NRR Memorandum, “South Texas Project Units 3 and 4, Combined License Application – Advanced Safety Evaluation for Chapter 5, “Reactor Coolant System and Connected Systems” February 7, 2011 (ML110350169)

4. NRR Memorandum, “South Texas Project Units 3 and 4, Combined License Application – Advanced Safety Evaluation for Chapter 6, “Engineered Safety Features” February 7, 2011 (ML110310194)

5. NINA Letter, “South Texas Project Units 3 and 4 Docket Nos. 52-012 and 52-013 Submittal of Combined License Application Revision 7,” February 1, 2012 (ML12048A714, ML12048A686)

6. Certified Advanced Boling-Water Reactor (ABWR) design control document (DCD), Revision 4, referenced in Title 10 of the *Code of Federal Regulations* (10 CFR) Part 52, “Licenses, Certifications, and Approvals for Nuclear Power Plants,” Appendix A, “Design Certification Rule for the U.S. Advanced Boiling Water Reactor” (ML11126A173, ML11126A129)

7. Regulatory Guide 1.82, Revision 3, “Water Sources for Long-Term Recirculation Cooling Following a Loss-of-Coolant Accident”

1. NRR Memorandum, "South Texas Project Units 3 and 4, Combined License Application – Advanced Safety Evaluation for Chapter 4, Reactor," February 7, 2011 (ML110340381)

2. NRR Memorandum, "South Texas Project Units 3 and 4, Combined License Application – Advanced Safety Evaluation for Chapter 5, "Reactor Coolant System and Connected Systems" February 7, 2011 (ML110350169)

3. NRR Memorandum, "South Texas Project Units 3 and 4, Combined License Application – Advanced Safety Evaluation for Chapter 6, "Engineered Safety Features" February 7, 2011 (ML110310194)

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6. Regulatory Guide 1.82, Revision 3, "Water Sources for Long-Term Recirculation Cooling Following a Loss-of-Coolant Accident"

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