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

July 18, 2013

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

SUBJECT: SPENT FUEL POOL STUDY

Dear Chairman Macfarlane:

During the 606th meeting of the Advisory Committee on Reactor Safeguards, July 9-12, 2013, we reviewed the NRC staff's report, "Consequence Study of a Beyond-Design-Basis Earthquake Affecting the Spent Fuel Pool for a U.S. Mark I Boiling Water Reactor." This study is commonly referred to as the Spent Fuel Pool Study (SFPS). Our Materials, Metallurgy, and Reactor Fuels and Reliability and PRA Subcommittees jointly reviewed the methods, approaches, and findings of this study on May 8, 2013. During these meetings we had the benefit of discussions with representatives of the NRC staff. We also had the benefit of the documents referenced.

CONCLUSIONS AND RECOMMENDATIONS

1. The SFPS has been performed in a thorough and systematic manner, and provides a state-of-the-practice assessment of the consequences of a beyond-design-basis seismic event on the spent fuel pool in a reference boiling water reactor containing either high-density or low-density fuel loading.

2. The study has demonstrated that health effects from seismically initiated spent fuel pool damage scenarios are very low for both low-density and high-density pool loadings.

3. We agree with the staff's conclusion that expedited transfer of spent fuel from the pool to dry cask storage does not provide a substantial safety enhancement for the reference plant.

4. An important insight from the SFPS is that the less conventional (1x8) high-density fuel loading configuration actually used in the reference plant can significantly reduce the consequences of seismically induced damage. This approach should be further explored as a measure to provide additional defense in depth against spent fuel pool accidents.

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5. The SFPS provides sound approaches, tools, and insights for a broader evaluation of the consequences of severe seismic events on spent fuel pools of different design and will be valuable in determining whether expedited transfers to dry cask storage systems (DCSSs) produce substantial safety benefit for U.S. BWRs and PWRs.

6. The SPFS should be issued.

BACKGROUND

One of the early NRC concerns following the March 11, 2011 Great East Japan Earthquake and tsunami was the possibility that the spent fuel pools at Fukushima had been severely damaged and drained, that fuel cladding had failed due to overheating, and could be releasing very large amounts of radioactive material. This early concern was not unreasonable in view of the unexpected detonation at Unit 4 in which all the fuel was in the spent fuel pool. Within a short time it became clear that the spent fuel pools at Fukushima were not damaged or leaking, the fuel cladding was undamaged, and uncontrolled radiological releases had not occurred. Despite these observations, concerns remained that spent fuel in the pools constituted a major threat to public health and safety. As a result, expedited transfer of spent fuel from the pools in U.S. reactors to DCSSs has been proposed.

The primary objective of the SFPS was to determine the safety benefit of expedited transfer of spent fuel from the pool of a reference BWR (of similar design to those at Fukushima) to DCSSs. The staff reviewed past consequence and risk assessments related to spent fuel storage. These studies showed that risk of fuel uncovery was low, but that the consequences could be large. These prior studies also showed that seismic hazard was the dominant contributor to spent fuel pool risk. For these reasons, the staff selected a study of the consequences of a beyond-design-basis seismic event at a General Electric BWR-4 Mark I reference plant as the starting point to assess the continued applicability of past studies and to evaluate the safety benefit of low-density versus high-density fuel loading in spent fuel pools. Depending on the results gained from the study, it was recognized that additional work would be needed to reach generally applicable conclusions for U.S. BWRs and PWRs.

In a Staff Requirements Memorandum (SRM), dated July 16, 2012, the Commission directed the staff to incorporate human reliability analysis (HRA) and conduct a comparative assessment of the results of the SFPS with previous studies of the safety consequences associated with loading, transfer, and long-term storage in DCSSs. In addition, the SRM stated that the SFPS should consider comparisons of the performance of spent fuel pools during the real incidents identified by the ACRS members in the April 25, 2012, letter with the predicted behavior of the reference spent fuel pool.

DISCUSSION

The SFPS is a detailed analysis of the consequences of a severe, beyond-design-basis, seismic event on the spent fuel pool in a BWR-4 Mark 1 reference plant at a specific site. The analysis was intended to be as realistic as possible with minimal deliberate additions of conservatism. The study examines the benefits of 10 CFR 50.54(hh)(2) mitigation procedures and equipment required by the Power Reactor Security Rulemaking, although it does not consider the capabilities associated with the FLEX program or the requirements of Orders EA-12-051 and EA-12-049.

Seismic and Structural Analyses

The seismic initiator selected for the study was an event with a peak ground acceleration (PGA) range of 0.5 to 1.0 g, a geometric mean PGA of 0.7 g, and a likelihood (based on PGA) of 1 in 60,000 years. The staff concluded that this event provided a good compromise between events with higher occurrence frequencies that would lead to little or no damage versus higher consequence events with very low frequencies. This selection was consistent with prior spent fuel pool probabilistic risk analyses and similar to the seismic event chosen for the State-of-the-Art Reactor Consequences Analysis short-term station blackout study. The event is considerably more severe than the current design basis Safe Shutdown Earthquake (SSE) of the reference plant (PGA of 0.12 g) and the SSEs of most U.S. plants. The horizontal response spectrum is the Generic Issue-199 spectral shape, which used the 2008 U.S. Geological Survey model, scaled to a PGA of about 0.7 g.

Structural modeling and analyses were performed to determine the damage states that would be produced by the reference seismic event and to define the initial conditions for the damage progression analysis. Cracking of the reinforced concrete pool and tearing of its stainless steel liner at the bottom of the pool had the greatest potential for creating leaks and potentially draining the pool, but other damage states were not ignored. The study included quantitative assessments of: the amount of water lost by sloshing, damage to the refueling gate, damage to support systems and penetrations, damage to the reactor building, and damage to other relevant structures. Qualitative assessments were also made of damage to spent fuel racks and stored fuel assemblies.

In-structure response spectra (ISRS) at elevations of interest were obtained by scaling the ISRS developed for the reference plant for the NUREG-1150 study. The loads on the spent fuel pool structure included the static weight of concrete, steel, water, storage racks, and fuel; the dynamic forces from the ground motion; and hydrodynamic forces from the water in the pool. These loads and best-estimate mechanical properties of all pool materials including concrete, structural steel, and the liner were used as inputs to a three dimensional nonlinear finite element analysis of the pool structure and its supports to determine the extent of cracking of the concrete and deformation or tearing of the liner.

The integrity of the spent fuel pool liner is the most important factor influencing the rate and extent of damage following the seismic event. To address liner integrity, the staff constructed a detailed finite element model of the reference plant liner, including floor and wall attachments, using elements as small as 0.15 inches. This model was integrated into the concrete base model to calculate the strains in the liner. The maximum liner strains were concentrated at the attachments to the backup plates embedded in the concrete. These maximum strains however were less than the expected failure strain for the liner material. The staff performed an analysis which considered variability in the failure strain, uncertainties in the ISRS accelerations, uncertainties in concrete properties, and reductions in spectral accelerations to account for ground motion incoherency and nonlinear effects.

The results of these analyses confirmed that a state with no liner tearing was most likely with a relative likelihood in excess of 90%. For the state in which there was a 10% likelihood of liner tearing, the staff assumed that there was an equal likelihood that tearing of the liner would result in small leaks or moderate leaks. The staff estimated the average flow rate for a small leak to a height of about 16 ft above the spent fuel pool floor is about 200 gallons per minute. Their estimate of the average flow rate for the moderate leak to a height of about 16 ft above the spent fuel pool floor is about 1,500 gallons per minute.

The analyses of the likelihood of liner tearing and the resulting flow rates are subject to considerable uncertainty associated with the characteristics of the cracking in the concrete, the loading on the structure, the initial condition of the liner and the concrete, the prediction of failure in the liner material under complex stress states, and the leak rates through cracks in concrete and steel.

The staff believes that this assessment is somewhat conservative. We concur with this assessment. The impact of such uncertainties should be considered in the context of the overall behavior of the system. Caution should be exercised in extrapolating the fragility results from the analysis of this particular spent fuel pool to other pool configurations.

The results of the seismic and structural analysis can be described in terms of three initial damage states which are used as inputs to follow-on analyses of accident progression:

- A state with no leakage. In this state, the liner would be intact, but through-wall cracking of the concrete would occur at the junction of floor and wall of the pool. The likelihood of this state was determined to be 90%.

- A state with small leakage. In this state the rate of leakage would be controlled by the number and dimensions of localized tears in the liner. The likelihood of this state was assumed to be 5%. Absent any mitigation, the pool would drain completely in approximately 42-62 hours.

- A state with moderate leakage. In this state the liner would be severely damaged and the rate of leakage would be controlled solely by the dimension of cracks in the concrete. The likelihood of this state was assumed to be 5%. Absent any mitigation, the pool would drain completely in approximately 6-9 hours.

Accident Progression Analyses and Radiological Releases

Detailed analyses of accident progression were performed using the MELCOR code. Various models within the code were used to evaluate major factors affecting accident progression. These included the pool geometry, fuel design, fuel loading configurations, rack design, decay heat, radionuclide inventory, thermal radiation, air oxidation, hydraulic resistance, fission product release and transport, and the effects of mitigative sprays.

To address the varying heat load and fission product content of the spent fuel pool, the plant operating cycle, was divided into five operating cycle phases (OCPs):

- OCPs 1 and 2 include the refueling period during which the spent fuel pool is connected to the reactor, and the decay heat generation rate is at its highest. These phases account for 3% of the plant operating cycle.

- OCP 3 is a post outage phase in which the spent fuel pool is no longer connected to the reactor, and decay heat is decreasing but still high. This phase represents 5% of the plant operating cycle.

- OCPs 4 and 5 cover the remaining 92% of the plant operating cycle during which decay heat is slowly decreasing. A 1x4 configuration (one recently discharged assembly surrounded by four low power assemblies) was used throughout the study for the high-density pool loading cases, although the reference plant uses a more favorable 1x8 configuration.

For each OCP, the staff analyzed scenarios in which combinations of damage state (no leak, small leak, or moderate leak) mitigation (effective or ineffective) and fuel loading (high-density or low-density) were evaluated. The analyses showed:

- No radiological release occurred for the most likely, no-leak, damage state even with high-density fuel loading. Coolant losses due to seismic induced sloshing and boil-off were small. Even during OCP 1 the fuel remained covered with 15 feet of water after 72 hours.

- For the small leak damage state, no releases occurred for any operating cycle phase for mitigated high-density or low-density fuel loading scenarios. This was not the case for unmitigated, high-density or low-density loadings during OCPs 1, 2 and 3. For these scenarios timely deployment of 10 CFR 50.54(hh)(2) spent fuel pool makeup was required to prevent release. If left unmitigated, radiological release would begin within 40 to 60 hours of the seismic event.

- A significant advantage of the low-density loading was that hydrogen deflagrations were not predicted for any of the small leak unmitigated cases, whereas deflagrations and substantially higher releases were predicted for unmitigated high-density pool loading using the 1x4 configuration. In separate sensitivity analyses the staff evaluated the 1x8 high-density configuration actually used by the reference plant and found that this configuration resulted in major reductions in heat-up rates and maximum fuel temperatures, no deflagrations, and no radiological releases. Such loadings should be evaluated further to determine whether they are broadly beneficial to BWR and PWR spent fuel pools.

- For the moderate leak damage state, mitigation was effective in preventing release during OCPs 2 and 3 for both high-density and low-density fuel loading but not effective during the first week of the plant operating cycle (OCP 1).

- For the moderate leak damage state, mitigation was not necessary to prevent release during OCPs 4 and 5. Both high-density and low-density fuel loadings were air-coolable.

<u>Mitigation and Human Reliability Analyses</u>

For scenarios in which mitigation is successful, the study credited the use of 10 CFR 50.54(hh)(2) equipment and the use of both on-site capabilities and off-site resources to extend the utilization of this equipment until on-site capabilities are recovered. None of these capabilities were credited in the assessment of unmitigated scenarios.

The SFPS shows that the 10 CFR 50.54(hh)(2) equipment is capable of mitigating most scenarios. However, such equipment may not be available due to deployment for other purposes. Order EA-12-049, while not evaluated in the SFPS, specifically requires a capability to maintain spent fuel pool cooling. Responses to the order should be evaluated to ensure that they provide the capability to deal with leakage on the order of that expected in a beyond-design-basis seismic event such as that considered in the SFPS.

To gain insights into the likelihood that successful mitigation could be accomplished, the staff performed a simplified HRA. Unfortunately the analysis does not address complexities expected to affect human performance during a severe seismic event, and it does not account for the effects of uncertainties. Therefore, the calculated human error probabilities are optimistic and cannot be considered as realistic estimates of the probability of success. Nevertheless, the descriptive analysis of the actions required to implement mitigation provides a useful structure for understanding these activities.

Offsite Consequence Analyses

The SFPS shows that for all scenarios no early fatalities are predicted to occur. Even in the unmitigated scenarios, the predicted releases are neither fast enough nor large enough to significantly exceed offsite dose levels that can cause early fatalities. As modeled in the SFPS, personnel evacuation and hotspot relocation are always effective in preventing significant exposure to all individuals in the areas of most concern. A more thorough consideration of uncertainties could identify cases where evacuations are not as effective.

The SFPS also shows that average individual latent cancer fatality risk based on the linear no threshold dose response model is low for both high and low-density loadings without successful mitigation. These individual risks are dominated by long-term exposures to contaminated areas for which doses are small enough to be considered habitable. Use of alternate dose models results in significant reduction in the individual latent cancer fatality risk estimates.

The amount of land interdiction for the studied scenarios could be up to two orders of magnitude greater for the high-density loading configuration compared to the low-density loading configuration. However, successfully deployed mitigation in the high-density loading configuration is predicted to reduce the amount of land interdiction.

Overall, the SFPS provides a technically sound basis for the staff's conclusion that expedited transfer of spent fuel from the pool to dry cask storage does not provide a substantial safety enhancement for the reference plant. The study also provides approaches, tools, and insights for a broader evaluation of the consequences of severe seismic events on spent fuel pools of different design, and will be valuable in determining whether expedited transfers to DCSSs produce substantial safety benefit for U.S. BWRs and PWRs. We commend the staff for these accomplishments.

Sincerely,

/RA/

J. Sam Armijo
Chairman

REFERENCES

1. Consequence Study of a Beyond-Design-Basis Earthquake Affecting the Spent Fuel Pool for a U.S. Mark I Boiling Water Reactor, June, 2013 (ML13133A132)

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5. U.S. Nuclear Regulatory Commission, NRC Order EA-12-049, "Issuance of Order to Modify Licenses with Regard to Requirements for Mitigation Strategies for Beyond-Design-Basis External Events," dated March 12, 2012 (ML12054A736)

6. U.S. Nuclear Regulatory Commission, NRC Order EA-12-051, "Issuance of Order to Modify Licenses with Regard to Reliable Spent Fuel Pool Instrumentation," dated March 12, 2012 (ML12054A682)

7. RES Memorandum, Subject: Project Plan for Spent Fuel Pool Scoping Study, July 26,2011 (ML111570370)