Summary of Final Resolution of Loss-of-Coolant Accident Unreviewed Safety Questions for the Advanced Test Reactor

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Unreviewed Safety Questions (USQs) were resolved by a combination of analysis revision and plant design modifications while under interim restrictions on operation. The effort involved many parallel path activities, culminating in updated analytical models, updated accident analyses, identification, design, and completion of plant modifications to support the updated analyses, and revision of facility Technical Safety Requirements and associated procedures.

A late-heatup phenomenon during design-basis loss of coolant accidents (LOCAs) was identified that had not been discovered while updating the Advanced Test Reactor (ATR) Updated Final Safety Analysis Report (UFSAR). Identification of the late-heatup phenomenon constituted a USQ for new information and an interim limit reducing the allowable reactor power level by 50% was imposed in order to meet the ATR Plant Protection Criteria in the approved UFSAR. The initial approach taken to resolve the late heatup USQ included demonstrating that the primary coolant pumps (PCPs) would continue to provide flow when PCP net positive suction head (NPSH) fell below 50 ft and to model that degraded flow with the PCP in cavitation. Some flow from the PCP is required to prevent the late heatup after the initiation of a design-basis LOCA. The original UFSAR analysis had assumed the PCP output dropped to zero when NPSH dropped below 50 ft.

A second phenomenon was identified when the PCP cavitation model was applied to the design-basis LOCA analyses. Continued operation of the PCP would result in very low primary coolant system (PCS) pressures. For some analytical cases, pressures were so low that the air volume in the PCS surge tank expanded into the PCS piping creating concern for potential degradation of flow from the operating emergency coolant pump. The surge tank provides damping of pressure variations during normal operation. This evaluation concluded that the potential for surge tank draining during a design-basis LOCA under the interim effective-fuel-plate-power limits also constituted a USQ for new information. Interim resolution was developed based on a manual trip of the PCPs after a LOCA and tightened controls on surge tank level.

For final resolution of the USQs, the low-pressure actuation setpoint for the emergency firewater injection system was raised, surge tank operating levels were restricted, and a new Engineered Safety Feature (ESF) to trip the PCPs nominally 65 seconds after the Plant Protective System tripped on low vessel inlet or outlet pressure was designed and installed. Probabilistic risk assessment was used to evaluate the new ESF reliability and the risk-significance of the facility modifications. An increased vulnerability to heavy load drops was identified that required relocation of emergency coolant pump power conduits as part of the final resolution. The effort resulted in a return to full power operation to achieve needed irradiation capability.

1 INTRODUCTION

This paper provides a summary of the final resolution of the two unreviewed safety questions (USQs) resulting from new information about the progression of loss-of-coolant accidents (LOCAs) in the Advanced Test Reactor (ATR). The final resolution of these USQs required:

- development and revision of analytical models
- review of plant operations and potential accident sequences to determine additional considerations
- completion of accident analyses using the updated models
- identification, design, and completion of plant modifications to support the assumptions of the updated analyses, and
- revision of facility Technical Safety Requirements and procedures to implement the assumptions of the updated accident analyses.

1.1 Background

The first USQ concluded there was a late-heatup phenomenon during design-basis LOCAs that was not addressed in the Updated Final Safety Analysis Report (UFSAR). Identification of this late heatup phenomenon constituted an USQ for new information. The late-heatup phenomenon results from a power-coolant mismatch that occurs in the currently approved UFSAR analytical models at about 20 to 40 seconds after initiation of a design-basis LOCA (i.e., the 3-inch pipe break or 6-inch reactor top-head failure). Cooling of the fuel element hot channel was not adequate for the predicted combination of low pressure, relatively high decay heat, and low flow when flow was reduced to one emergency coolant pump in the analysis.

Interim controls provided a basis for interim operation with an interim limit on effective fuel-plate power pending final resolution of the late heatup USQ. The interim effective-fuel-plate-power limit was derived using the currently approved UFSAR methodology and ATR Plant Protection Criteria. The currently approved UFSAR and the documentation supporting the interim limit both show “Condition 2” or anticipated event margins are met for both bounding design-basis LOCA events analyzed in the UFSAR. The interim limit effectively reduced the allowable reactor power level by 50%.

The initial approach for resolution required demonstrating that the primary coolant pumps (PCPs) would continue to provide flow when PCP net positive suction head (NPSH) fell below 50 ft and to model that degraded flow with the PCP in cavitation. Some flow from the PCP is required to prevent the late heatup after the initiation of a design-basis LOCA. The currently approved UFSAR analysis assumed the PCP output dropped to zero when NPSH dropped below 50 ft.

A second phenomenon was identified when the PCP cavitation model was applied to the design-basis LOCA analyses. Continued operation of the PCP results in very low primary coolant system (PCS) pressures. For some analytical cases, pressures were so low that the air volume in the PCS surge tank expanded into the PCS piping. This resulted in a concern for potential degradation of flow from the operating emergency coolant pump. The concern for air in the PCS piping was applicable to both the full-power USQ resolution and interim operation. The second USQ evaluation for new information was completed relative to the interim operating limits. This evaluation concluded that the potential for surge tank draining during a design-basis LOCA under the interim effective-fuel-plate-power limits constituted an unreviewed safety question for new information. Additional interim resolutions were developed for second USQ based on a manual trip of the PCPs after a LOCA and tight controls on surge tank level. The interim resolutions including the interim effective-fuel-plate-power limits were incorporated into the ATR Technical Safety Requirements (TSRs).

1.2 Summary of the Final Resolution Efforts

The two USQs involved the late-heatup phenomenon and the concern for surge tank draining during design-basis LOCA events. The basic approach to resolving these USQs was to more realistically model the PCP behavior to show there would be sufficient PCS flow to prevent the late heatup in the early stages of a design-basis LOCA. The more realistic modeling of the PCP, based on cavitation data from the pump vendor, showed the PCP would continue to provide flow in the early stage of the design-basis LOCA. However, the PCPs were so effective that a potential to drain the PCS surge tank and introduce air into the PCS was identified. For the final resolution, this concern was eliminated by raising the low-pressure actuation setting for the EFIS, establishing a control on the surge tank operating level, and by installing a new Engineered Safety Feature (ESF) to trip the PCPs nominally 65 seconds after actuation of one of the Plant Protective System (PPS) low reactor-vessel pressure subsystems. An auxiliary modification providing a bypass of the EFIS low-pressure actuation function was installed and controls on use of this bypass were developed. This modification was designed to relieve burdensome operational requirements to prevent inadvertent actuation of the EFIS during normal PCS startup and shutdown operations.

Once this basic approach was established, the detailed resolution effort identified and resolved all of the issues related to this basic approach. The ATR RELAP5 model was upgraded to incorporate the PCP cavitation model and LOCA PCP shutoff ESF and to improve the PCS surge tank model. Vendor data were obtained to support the PCP cavitation model. Because of the sensitivity of the results to EFIS injection, a more realistic model of the EFIS injection was developed. Facility test data were obtained to support this model. The results are also extremely sensitive to emergency coolant pump flow so facility test data were obtained and the ATR RELAP5 model was benchmarked and adjusted to agree with the test data.

The design-basis accident sequence definitions were reviewed. The ATR RELAP5 pipe break model for the 3-inch LOCA was modified to more realistically, yet conservatively, model the design-basis LOCA. Additionally, the initiation frequency of the reactor top-head LOCA was quantitatively reviewed and shown to be much lower than previously assumed. Reactor accident sequences in the UFSAR and the Probabilistic Risk Assessment (PRA) were reviewed to assure that effects on other accident sequences addressed in the UFSAR and other considerations resulting from the USQ resolution were identified and analyzed. A review of the reliance on continued operation of the PCPs for a brief period after a low-pressure scram and the trip of the PCPs at 65 seconds after a low-pressure scram resulted in identifying several credible concurrent events and an additional concern for a seismic event. Thus, RELAP5 analyses were performed for the 3-inch LOCA alone, and in combination with credible concurrent events including upper or lower vessel emergency firewater injection systems out of service, failure of one PCP to be shutoff by the new ESF, shutdown while emergency firewater injection is in manual mode, or failure of the pressurizing/makeup system.

The PCS has been analyzed for the Safe Shutdown Earthquake (SSE) and appropriate upgrades to the PCS have been completed to meet code requirements at the time of the analysis. However, interfacing piping of less than 3-inches diameter or separated from the PCS by an orifice with a diameter of 3-inches or less or a closed valve or check valve has not been seismically analyzed. Significant seismic events up to the magnitude of the SSE are expected to cause some leakage in this interfacing piping. Full rupture of the small-diameter PCS interfacing piping is not expected during earthquakes up to and including the SSE. However, a severe earthquake may result in some cracking of the piping connected to the PCS. The expected leakage was not previously quantified but was considered bounded by the maximum leak diameter of 3 inches.

The potential leakage was characterized based on the configuration of the PCS and interfacing piping, the potential range of seismic events, previous evaluations of seismic fragility of piping systems at ATR, and a walk-down of PCS interfacing piping. This characterization concluded a 1-inch-equivalent-diameter break is the best estimate of the equivalent leakage caused by a seismic event up to the safe shutdown earthquake (up to a Condition 4 seismic event). The evaluation further concluded Condition 2 seismic events (i.e., high frequency but low magnitude) would not result in a seismic SBLOCA sequence. A 2-inch-equivalent-diameter break was recommended for safety analyses of a seismic SBLOCA as a conservative bound for the uncertainty in the best estimate of the equivalent leakage. Since the 3-inch inlet break is the design-basis SBLOCA for the ATR, the same event was recommended as the design-basis seismic SBLOCA for consistency. The 1- and 2-inch seismic SBLOCA consequences are evaluated according to the ATR Plant Protective Criteria for Condition 3 events.

Seismic SBLOCA events were analyzed with RELAP5, ATR-SINDA and SINDA-SAMPLE. The break models for the 1-inch and 2-inch diameter breaks were connected directly to the vessel inlet piping just downstream of the butterfly valve. The 3-inch diameter break model was the same as the base-case 3-inch design-basis LOCA. The seismic event was conservatively assumed to cause a loss of AC power that tripped the PCPs, M-10 emergency coolant pump, and secondary coolant pumps. Not all Condition 3 seismic events are expected to result in a loss of all AC power. The coastdowns of the PCPs and the M-10 emergency coolant pump were calculated with the RELAP5 torque/inertia model. The coastdown of the secondary coolant pumps was simulated by linearly reducing the secondary coolant flow to zero over a 10-second period. The startup of the DC-powered M-11 emergency coolant pump, which starts on low flow in the recirculation line of the M-10 emergency coolant pump, was simulated by linearly increasing the pump speed from zero to its rated value over a 10-second period.

The frequency analysis for the early complete-loss-of-flow accident (CLOFA) was updated to consider the impacts of the LOCA PCP shutoff ESF and the potential for surge tank draining to degrade pump performance. The plant modifications were evaluated using failure mode and effects analysis (FMEA) and fault tree analysis. PRA event trees and fault trees were revised, and the acceptability of risk increases was evaluated using the risk evaluation guidelines that had been previously developed analogous to NRC Regulatory guide 1.174. An increased location-dependent heavy load drop vulnerability was identified in the PRA that required relocation of power conduit for emergency coolant pumps to control risk level.

The review of the UFSAR accident sequences also included the severe, beyond-design-basis accidents. Selected cases from the PCS break-spectrum analyses were run using the upgraded RELAP5 model to examine impacts of the modeling changes (i.e., continued operation of the PCPs in the cavitation regime, EFIS flow updates, higher EFIS actuation pressure, and upgraded surge tank modeling) on the break-spectrum results. The PCS break-spectrum results are used to support the analysis of the frequency of a direct-damage LOCA (DDLOCA). The revised break-spectrum results along with additional update information were used to update the frequency analyses for the DDLOCA.

1.3 Closing Summary

This effort spanned 10 months from identification of the first USQ to completion of analysis, design and installation of plant modifications, changes to the TSR and its bases as well as procedure changes. USQ issues were identified and processed in a compliant manner, and a concerted effort was made to provide effective, timely, and affordable solutions. The effort involved many parallel path activities, many of them providing essential information for concurrent activities, and requiring daily and weekly statusing. Drawing on staff with generally more than 12 years of ATR experience, and often more than 20 years of ATR experience, the effort was successful in resolving the USQ issues and allowing a return to full power operation.