USQ PROCESS
POTENTIAL INADEQUACY IN THE SAFETY ANALYSIS (PISA) FORM
(DO NOT USE FOR PROPOSED CHANGES)

Facility or Activity: Advanced Test Reactor

USQ Process No.: TRA-USQ-2004-413

Subject: ATR Seismic Primary Coolant System Break Size Contribution From Letdown Valves

Describe the New Information/Discovery:

The potential for the amount of seismically-induced primary coolant system (PCS) leakage characterized in SAR-153 may be non-conservative when the potential for PCS letdown valves to not fully close is considered. The valves are automatically controlled by the computerized distributed control system (DCS). While the valves are designed to close upon loss of control power or loss of instrument air, the control power is battery-backed and may not fail due to a seismic event. Instrument air compressors may be likely to lose power; however, the air bleed off may not occur rapidly. The DCS has not been designated as Seismic Category I equipment and has not previously been seismically evaluated. The PCV-1-1 valve letdown is limited by a nearby 2.5-in. diameter orifice. The LCV-1-3C letdown valve size is 1-in. The estimated combined break size would be slightly above 2.5-in. equivalent diameter, or approximately 2.7-in. diameter. The elevation of the PCV-1-1 valve is 81 ft; however, the letdown flow line routed from the valve to the degassing tank rises to 88 ft 9 in. The elevation of valve LCV-1-3C on the bypass demineralizer system is 64 ft.

Seismically-induced primary coolant system (PCS) leakage characterized in SAR-153 is based on TRA-ATR-1490 that considered the configuration of the PCS and interfacing piping, the potential range of seismic events, previous evaluations of seismic fragility of piping systems at the ATR, and walkdowns of the PCS interfacing piping. The potential leakage resulting from a seismic event was not expected to exceed an equivalent 1-inch break for seismic events up to and including the design basis earthquake. However, the Chapter 15 accident analysis has included 2.5-in. equivalent diameter and smaller breaks as Condition 3 (unlikely) events, and 3.0-in. equivalent diameter breaks as Condition 4 (extremely unlikely) events. The break locations are analyzed as follows: the 1-in. and 2-in. breaks are analyzed directly on reactor vessel inlet piping, the 2.5-in. and 3-in. breaks were analyzed on the non-radiographed portion of the primary coolant system between the vessel inlet line and the bypass demineralizer, downstream of the orifice. The 3-in. break assumed a 3-in. orifice, and the 2.5-in. break used the actual orifice diameter of 2.5-in. No letdown flows from PCV-1-1 or LCV-1-3C were included in the analyses for evaluating thermal margins for the postulated breaks.

Identify the applicable safety basis document(s) (i.e., DSA, SAR, BIO, TSRs, OSRs):

SAR-153 ATR Upgraded Final Safety Analysis Report, Chapter 15.6 and SAR Addendum EDF-4334

I. PISA ASSESSMENT

a. Is there a potential inadequacy in the analytical methodology and/or tools used in the safety basis?
☐ Yes ☒ No

b. Is there a potential discrepancy between the safety basis assumptions, provisions, or conditions, in the accident analysis and the facility operation or parameters?
☒ Yes ☐ No

c. Is the actual physical condition of the facility different than as described in the safety basis (discrepant as-found condition)?
☐ Yes ☒ No

d. Has an actual or postulated operational event or incident revealed a potential inadequacy in the safety basis?
☐ Yes ☒ No

Provide an explanation of the assessment result:

Seismic loss-of-coolant accident events are represented as design basis events in Chapter 15 of the ATR Upgraded Final Safety Analysis Report (UFSAR). No letdown flows from PCV-1-1 or LCV-1-3C were included in the analyses for evaluating thermal margins for the postulated breaks. The omission could increase the potential

Facility or Activity: Advanced Test Reactor

USQ Process No.: TRA-USQ-2004-413

Subject: ATR Seismic Primary Coolant System Break Size Contribution From Letdown Valves

seismically-induced leakage when combined with seismically-induced pipe breaks, and could result in larger than previously estimated seismically-induced PCS leakage and a reduction in previously calculated thermal margins.

The screening Unreviewed Safety Question result is positive.

T. A. Thatcher	**Original signed on 12/23/04**	
USQ Evaluator Print/Type Name	USQ Evaluator Signature	Date
D. W. Suthers	**Original signed on 1/17/05**	
Nuclear Facility Manager Print/Type Name	Nuclear Facility Manager Signature	Date

If the answer to ALL of the assessment questions above is No, file the completed form.

If the answer to any one of the assessment questions is Yes, there is a potentially inadequate safety analysis (PISA):

- Nuclear Facility Manager document actions (Section II), including interim operating restrictions, taken to place or maintain the facility in a safe condition.
- Nuclear Facility Manager execute reporting process per MCP-190, for PISA.
- USQ evaluator proceed to Section III.

II. DOCUMENT ACTIONS TAKEN TO PLACE OR MAINTAIN FACILITY IN A SAFE CONDITION

III. DETERMINATION

Identify applicable section(s) of the safety basis document(s) (i.e., SAR, BIO, TSRs, OSRs).

SAR-153 ATR Upgraded Final Safety Analysis Report, Chapters 15.6 and 15.16.
EDF-4334, "Summary of Resolution of Advanced Test Reactor Unreviewed Safety Questions – SE-2003-126, SE-2003-145, SE-2003-146, SE-2003-155, SE-2003-156, and SE-2003-171, see page 20 for analysis of 2.5-in. seismic LOCA.

Applicable Technical References

Atkinson, S. A., 199, *Safety Basis for ATR Seismic-LOCA Scenarios*, EDF TRA-ATR-1490, July 1999.

Davis, C. B., and S. T. Polkinghorne, 1999, *Analysis of Seismic SBLOCAs*, EDF TRA-ATR-1489, August 1999.

Polkinghorne, S. T., 2003, *Analysis of a Seismically Initiated Rupture of the ATR's Bypass Demineralizer Line*, EDF-3278, November 2003.

IIIa: POTENTIAL FOR AN INCREASE IN PROBABILITY OR CONSEQUENCE OF AN ACCIDENT OR MALFUNCTION EVALUATED IN THE SAFETY BASIS

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USQ Process No.: TRA-USQ-2004-413

Subject: ATR Seismic Primary Coolant System Break Size Contribution From Letdown Valves

1. Could the PISA increase probability of occurrence of an accident previously evaluated in the safety basis?
Yes ☒ No ☐
Explain:

Seismic loss-of-coolant accident events are represented as design basis events in Chapter 15 of the ATR Upgraded Final Safety Analysis Report (UFSAR). Chapter 15.6.6, Seismically-Induced PCS Leakage" includes a characterization of the potential leakage resulting from a seismic event based on the configuration of the primary coolant system (PCS) and interfacing piping, the potential range of seismic events, previous evaluations of seismic fragility of piping systems at the ATR, and a walkdown of the PCS interfacing piping performed in 1999. This characterization concluded that for seismic events up to the SSE (that is, up to a Condition 4 seismic event), the leakage would be less than that from a 1-inch diameter break in the reactor vessel inlet piping. A 2-inch diameter break in the reactor vessel inlet piping was analyzed to provide a conservative upper bound on the leakage expected following a severe Condition 3 (unlikely) earthquake. A 2.5-in. break downstream of the 2.5-in. diameter orifice (FE-37) was also evaluated for Condition 3 events in EDF-3278 as described in SAR Addendum EDF-4334. A 3-inch diameter break in the reactor vessel inlet piping was analyzed for a severe Condition 4 (extremely unlikely) earthquake.

Upon low pressure, PCS letdown valves, PCV-1-1 and LCV-1-3C, are designed to close. If the valves did not close following the seismic event, the open valves may increase the PCS letdown flow. These valves are controlled by the distributed control system (DCS) that has not been seismically qualified. Depending on the timing of DCS loss of communication or other fault condition, the valves could fail as-is, move to an open or closed position. Conditions during the transient and possible failure modes for the DCS make it difficult to characterize the likelihood of the resulting valve position. The valves would close on loss of power; however, the power supply is battery-backed and may be generally robust. The valves would close on loss of instrument air; however, the timing of loss of instrument air and failure fragility is unknown. Furthermore, a modification had been planned to provide nitrogen backup air for the valves.

The PCV-1-1 valve letdown is limited by a nearby 2.5-in. diameter orifice. The LCV-1-3C letdown valve size is 1-in. The estimated combined break size would be slightly above 2.5-in. equivalent diameter, or approximately 2.7-in. diameter. The elevation of the PCV-1-1 valve is 81 ft; however, the letdown flow line routed from the valve to the degassing tank rises to 88 ft 9 in. The elevation of valve LCV-1-3C on the bypass demineralizer system is 64 ft.

The largest break size evaluated for Condition 3 was 2.5-in.; therefore, the 2.7-in. effective break size due to letdown flow is not bounded by existing analysis, although the locations of the letdown valves may not be the most limiting locations. As the fragility of the DCS system that controls the valves has not been characterized, the likelihood of conditions resulting in PCS leakage following a seismic event may be higher than previously analyzed.

The probability of occurrence of an accident previously evaluated in the safety basis may be increased.

2. Could the PISA increase the consequences of an accident previously evaluated in the safety basis?
Yes ☒ No ☐
Explain:

Seismic loss-of-coolant accident events are represented as design basis events in Chapter 15 of the ATR Upgraded Final Safety Analysis Report (UFSAR). Chapter 15.6.6, Seismically-Induced PCS Leakage" includes a 2-inch diameter break in the reactor vessel inlet piping that was analyzed to provide a conservative upper bound on the leakage expected following a severe Condition 3 (unlikely) earthquake. A 2.5-in. break downstream of the 2.5-in. diameter orifice (FE-37) was also evaluated for Condition 3 events in EDF-3278 as described in SAR Addendum EDF-4334. And, a 3-inch diameter break in the reactor vessel inlet piping was analyzed for a severe Condition 4 (extremely unlikely) earthquake.

USQ PROCESS
POTENTIAL INADEQUACY IN THE SAFETY ANALYSIS (PISA) FORM
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Facility or Activity: Advanced Test Reactor

USQ Process No.: TRA-USQ-2004-413

Subject: ATR Seismic Primary Coolant System Break Size Contribution From Letdown Valves

An earthquake was assumed to trip the reactor at 0.2 s. The first damaging seismic waves reached the ATR at 2.0 s, initiating the small-break LOCA and causing a loss of commercial and diesel power. Low flow in the M-10 emergency coolant pump recirculation line caused the DC-powered M-11 emergency coolant pump to start automatically. EFIS flow was automatically actuated after upper plenum pressure decreased to the EFIS actuation setpoint. Thermal limits were approached twice during the transients: the first approach to thermal limits occurred as the primary coolant pumps (PCPs) coasted down and the flow rate through the core reached a minimum (near the time that the PCP-check valves closed) at approximately 23 seconds after scram for the Condition 3 and 4 small-break LOCAs. Thermal margins then increased because of decreasing decay heat and increasing core flow. The second approach to limits occurred when the DC power supply was depleted, causing the M-11 emergency coolant pump to coast down. Temperatures increased and thermal margins decreased when the flow reversed (from forced downflow to natural circulation upflow).

The minimum thermal margins, both early and late, are summarized in Chapter 15.6.6 Table 15.6-4, for the 2-in. Condition 3 and for the 3-in. Condition 4 small-break LOCAs. The 3-in. Condition 4 small-break LOCA had very low margins to flow instability (1.16 standard deviations) early in the transient (at approximately 23 s).

The largest break size evaluated for Condition 3 was 2.5-in.; therefore, the 2.7-in. effective break size due to letdown flow from PCV-1-1 and LCV-1-3C may not be bounded by existing analysis, although the locations of the letdown valves may not be the most limiting locations. TT 2-3-05

The increased break size for Condition 3 events could increase the consequences of an accident previously evaluated in the safety basis.

3. Could the PISA increase the probability of occurrence of a malfunction of equipment important to safety previously evaluated in the safety basis? Yes ☒ No ☐

Explain:

The potential of PCV-1-1 and LCV-1-3C to remain open rather than close as was assumed is discussed above in questions 1 and 2. The probability of occurrence of a malfunction of equipment important to safety previously evaluated in the safety basis may be increased.

4. Could the PISA increase the consequences of a malfunction of equipment important to safety previously evaluated in the safety basis? Yes ☒ No ☐

Explain:

The potential of PCV-1-1 and LCV-1-3C to remain open rather than close as was assumed is discussed above in questions 1 and 2. The consequences of DCS failure had not been expected to worsen the seismic event analyzed in the safety basis. The consequences of a malfunction of equipment important to safety previously evaluated in the safety basis may be increased.

IIIb: POTENTIAL FOR CREATION OF AN UNANALYZED ACCIDENT OR MALFUNCTION OF A DIFFERENT TYPE

USQ PROCESS
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Facility or Activity: Advanced Test Reactor

USQ Process No.: TRA-USQ-2004-413

Subject: ATR Seismic Primary Coolant System Break Size Contribution From Letdown Valves

5. Could the PISA create the possibility of an accident of a different type than previously evaluated in the safety basis? Yes ☐ No ☒
Explain:

Seismically-induced PCS leakage has been considered in the safety basis. The PISA does not create the possibility of an accident of a different type not previously evaluated in the safety basis.

6. Could the PISA create the possibility of a malfunction of equipment important to safety of a different type than previously evaluated in the safety basis? Yes ☐ No ☒
Explain:

Seismically-induced PCS leakage has been considered in the safety basis. The PISA does not create the possibility of a malfunction of equipment important to safety of a different type than previously evaluated in the safety basis.

IIIc: POTENTIAL FOR REDUCTION IN A MARGIN OF SAFETY

7. Could the PISA reduce a margin of safety as defined in the safety basis? Yes ☒ No ☐
Explain:

Increased PCS break size could reduce the margin of safety needed to protect ATR fuel, but may depend on break (or letdown) location.

IIId: USQ DETERMINATION CONCLUSION

8. Based on the responses to questions 1 – 7 above, does the PISA constitute an Unreviewed Safety Question?
Yes ☒ No ☐
Explain:

The largest break size evaluated for Condition 3 was 2.5-in.; therefore, the 2.7-in. effective break size due to letdown flow may not be bounded by existing analysis. The contribution to seismic PCS leakage reduces the accommodation of possible seismically-induced pipe breaks.

The open valves could result in an estimated 2.7-in. equivalent diameter letdown flow at Condition 3, and pose an Unreviewed Safety Question.

NOTE: If USQ determination result is positive, additional notification for ORPS is required per MCP-190.

IV. APPROVAL/CONCURRENCE SIGNATURES

APPROVAL:

T Thatcher	T Thatcher	1-27-05
USQ Evaluator Print/Type Name	USQ Evaluator Signature	Date

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Facility or Activity: Advanced Test Reactor

USQ Process No.: TRA-USQ-2004-413

Subject: ATR Seismic Primary Coolant System Break Size Contribution From Letdown Valves

J.C. Chapman USQ Reviewer Print/Type Name	JC Chapman USQ Reviewer Signature	2 Feb 2005 Date
D.W. Sutherland Nuclear Facility Manager Print/Type Name	[signature] Nuclear Facility Manager Signature	2/3/05 Date

CONCURRENCE:

Alan P Hoskins Independent Review Committee Chair Print/Type Name	Alan P Hoskins Independent Review Committee Chair Signature	2-3-05 Date