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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-2004-396

Subject: ATR Seismic Primary Coolant System Break Size and TRA Support Building Vulnerabilities

Describe the New Information/Discovery:

Recent seismic walkdowns conducted as part of an effort to provide an updated seismic probabilistic risk assessment (PRA) documented in the November 29, 2004 letter report from ARES Corporation "Consulting Services for the Advanced Test Reactor Seismic PRA Final" identified several smaller PCS lines (1-in. dia or less) that would be vulnerable to breakage during PC-3 or PC-4 seismic events. The seismic report also identified the bypass demineralizer shielding block partition wall as not satisfying ATR seismic criteria for a PC-4 seismic event. Also, TRA masonry block buildings constructed in the early 1950's have not been shown to be adequately reinforced and block wall collapse would be expected for PC-3 and PC-4 seismic events.

The potential for seismically-induced leakage and the amount of 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 that had not identified piping deficiencies. 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. The walkdown report identifies approximately 2-in. of leakage that would be expected to occur for Condition 3 or 4 events that was not previously identified. If the bypass demineralizer wall cracking and resulting displacements are sufficient to cause failure of primary coolant system piping or valve stems on the bypass demineralizer, the break size would exceed an equivalent diameter of 3-in. Preliminary fragility assessment provided in the report would indicate wall failure not at Condition 3 but a concern for Condition 4 events.

The November 29, 2004 letter report from ARES Corporation also identified inadequate reinforcement of some TRA masonry block buildings, which has resulted in very low seismic capacities. The likelihood of masonry block wall collapse, including the TRA-619 building that contains two safety related emergency firewater injection pumps, has been estimated to be very high (0.7 probability at the safe shutdown earthquake). Wall collapse in buildings that do not contain safety related equipment could also degrade the emergency firewater injection system inventory and flow capacity due to flow lost from damage to fire water risers in the buildings.
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

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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). Walkdown conclusions documented in the November 29, 2004 letter report from ARES Corporation "Consulting Services for the Advanced Test Reactor Seismic PRA Final" identified several smaller PCS lines (1-in. diameter or less) that would be vulnerable to breakage. The seismic report also identified the bypass demineralizer shielding block partition wall as not satisfying ATR seismic criteria.

Inadequate reinforcement of some TRA masonry block buildings has resulted in very low seismic capacities. The likelihood of masonry block wall collapse, including the TRA-619 building that contains two safety related emergency firewater injection pumps, has been estimated to be very high (0.7 probability at the safe shutdown earthquake). Wall collapse in buildings that do not contain safety related equipment could also degrade the emergency firewater injection system inventory and flow capacity due to flow lost from damage to fire water risers in the buildings. 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

Facility or Activity: Advanced Test Reactor

USQ Process No.: TRA-USQ-2004-396

Subject: ATR Seismic Primary Coolant System Break Size and TRA Support Building Vulnerabilities

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

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-in. diameter break in the reactor vessel inlet piping. A 2-in. 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-in. diameter break in the reactor vessel inlet piping was analyzed for a severe Condition 4 (extremely unlikely) earthquake.

The November 29, 2004 letter report from ARES Corporation "Consulting Services for the Advanced Test Reactor Seismic PRA Final" identified several smaller PCS lines (1-in. dia or less) that would be vulnerable to breakage. The seismic report also identified the bypass demineralizer shielding block partition wall as not satisfying ATR seismic criteria. Bounding estimates of the PCS line breakage could result in a combined leakage of approximately 2-in. equivalent diameter at Condition 3. Seismically-induced pipe failures from deficiencies identified for several small diameter PCS lines or instrument lines exceed the expected line breakage characterized in the safety basis. The combined identified piping discrepancies and the bypass demineralizer wall not meeting PC-4 seismic criteria could result in seismic break sizes exceeding 3-in. equivalent diameter and leave little or no allowance for any unidentified deficiency to increase the PCS leakage.

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The increased break size for Condition 3 and 4 events could increase the consequences of an accident previously evaluated in the safety basis.

Inadequate reinforcement of some TRA masonry block buildings has resulted in very low seismic capacities. The likelihood of masonry block wall collapse, including the TRA-619 building that contains two safety related emergency firewater injection pumps, has been estimated to be very high (0.7 probability at the safe shutdown earthquake). Wall collapse in buildings that do not contain safety related equipment could also degrade the emergency firewater injection system inventory and flow capacity due to flow lost from damage to fire water risers in the buildings. With degraded EFIS flow, the consequences of a seismically-induced small-break LOCA could be increased.

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:

Identified deficiencies regarding PCS piping, the bypass demineralizer wall adjacent to PCS piping, and the vulnerability of masonry block buildings that may result in damage to EFIS piping or additional firewater inventory losses are 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:

Failures resulting in loss-of-coolant accidents such as PCS piping breakage and failure of the EFIS to provide core cooling and makeup are analyzed, and the consequences of a malfunction of equipment important to safety previously evaluated in the safety basis are not increased.

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

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 and EFIS faults have 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 and EFIS faults have 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

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

   Increased PCS break size or inadequate EFIS flow would reduce the margin of safety needed to protect ATR fuel.

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:

   Seismically-induced pipe failures from deficiencies identified for several small diameter PCS lines or instrument lines exceed the expected line breakage characterized in the safety basis. The combined identified piping discrepancies and the bypass demineralizer wall not meeting PC-4 seismic criteria could result in seismic break sizes above those analyzed and leave little or no allowance for any unidentified deficiency. The deficiencies pose an Unreviewed Safety Question.

   The lack of reinforcement of some TRA masonry block buildings can result in very low seismic capacities. The likelihood of masonry block wall collapse, including the TRA-619 building that contains two safety related emergency firewater injection pumps, has been estimated to be very high (0.7 probability at the safe shutdown earthquake). Wall collapse in buildings that do not contain safety related equipment could also degrade the emergency firewater injection system inventory and flow capacity due to flow lost from damage to fire water risers in the buildings. The identified high probability of failure of these buildings and potential for reduced EFIS flow to the core or potential for reduced EFIS inventory poses 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
J.C. Chapman	JC Chapman	2 Feb 2005
USQ Reviewer Print/Type Name	USQ Reviewer Signature	Date
D.W. Sutton	[illegible] Sutton	2/3/05
Nuclear Facility Manager Print/Type Name	Nuclear Facility Manager Signature	Date

CONCURRENCE:

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

Break Size Estimate		Wall&PCS	Valves&pcs	valves/wall/pcs	valves/150gpm	valves	pcs breaks
		Case 1	Case 2	Case 3	Case 3	Case 4	Case 5
Description	Dia (in.	Area (in.^{2})	Area (in.^{2})	Area (in.^{2})	Area (in.^{2})	Area (in.^{2})	Area (in.^{2})
PCV-1-1 FTC	2.5		4.908739	4.908739	4.908739	4.908739	
Bypass demin wall	2.5	4.908739		4.908739			
or							
LCV-1-3C	1		0.785398	0	0.785398	0.785398	
or							
Other bypass demin leakage	0.5						
	2						
Twelve 3/8-in. breaks	0.375	1.325359	1.325359	1.325359			1.325359
150 gpm	0.57				0.255176		
Outlet instrument lines impact	1	0.785398	0.785398	0.785398			0.785398
with drain line	1	0.785398	0.785398	0.785398			0.785398
	0.5	0.19635	0.19635	0.19635			0.19635
	0.5	0.19635	0.19635	0.19635			0.19635
	Total area (in.^{2})	***8.197593***	***8.982991***	***13.10633***	***5.949313***	***5.694137***	***3.288855***
	Equiv. dia. (in.)	**3.230712**	**3.381937**	**4.085034**	**2.752254**	**2.692582**	**2.046338**

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