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

Facility or Activity: Advanced Test Reactor

USQ Process No.: ATR Complex-USQ-2010-497

Subject: Seismic-Induced Experiment Loop Leakage

Describe the New Information/Discovery:
Hendrickson (EDF TRA-ATR-1166) supports the seismic ruggedness of the expile experiment loop piping. Hendrickson concludes the catastrophic failure of loop piping from the safe shutdown earthquake (SSE) is incredible (beyond design basis). Hendrickson, however, postulates some leakage and small cracks due to the SSE in loop piping. Seismic induced experiment loop loss-of-coolant-accidents (LOCAs) are not specifically addressed in the ATR accident analysis. However, unreviewed safety question ATR Complex-USQ-2009-714, "Evacuation System Seismic Requirements not Supported by Seismic Design of Buildings," postulated that seismic induced experiment loop piping leakage due to the Condition 4 earthquake (SSE) would be enveloped by the limiting Condition 2, ½-in. loop piping LOCA consequence presented in section 15.4.4 of SAR-153. Hence, a seismic induced loop LOCA could be considered within the accident category of existing experiment loop LOCA addressed in section 15.4. The applicability of the consequence analysis of the limiting Condition 2, ½-in. loop LOCA to the consequence of a seismic induced loop LOCA is questionable.

Identify the applicable safety basis document(s) (e. g., DSA, SAR, BIO, TSRs, etc.):
SAR-153, Upgraded Final Safety Analysis Report for the Advanced Test Reactor, Revision 27
TSR-186, Technical Safety Requirements for the Advanced Test Reactor, Revision 23

I. REASONABILITY DETERMINATION (Reasonability determinations made as part of the ATR Design Basis Reconstitution effort may be documented per SP-10.2.4.9)

Is it reasonable that the potential for an inadequate safety analysis exists based on the new information or discovery described above? ☒ Yes ☐ No

If "Yes," proceed to Sections II, III, and IV.
If "No," provide an explanation and basis for the reasonability determination.

USQ Evaluator Print/Type Name	USQ Evaluator Signature	Date
Nuclear Facility Manager Print/Type Name	Nuclear Facility Manager Signature	Date

If the answer to the question above is "No," file the completed form. Do not complete Section II, III, or IV.

If the answer to the question above is "Yes," complete Section II, III, and IV.

II. PISA DECLARATION

What is the basis for concern that the safety basis may not be bounding or may be otherwise inadequate? Provide an explanation.

Two assumptions of the SAR-153 ½-in. loop LOCA consequence analysis are not valid for Condition 4 seismic events. First, the consequence analysis presented in SAR-153, section 15.7.1, assumes ATR Complex evacuation. ATR Complex-USQ-2009-714 documents the real possibility that the evacuation system will be impaired by a large earthquake. Therefore, evacuation sirens may not be available to alert personnel to evacuate the ATR Complex. The supporting consequence analysis (EDF TRA-ATR-1562),

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however, considered non-evacuating personnel. The resolution of ATR Complex-USQ-2009-714 will add the consequence to non-evacuating personnel to SAR-153, Section 15.7.1.

Second, the consequence analysis assumed ATR confinement. However, the 8-in. block wall, which separates the reactor main floor from the canal areal, is not expected to maintain its "gas tight boundary" integrity following the Condition 4 earthquake. Additionally, the consequence analysis that is assumed to bound the seismic induced loop piping leakage assumes only one loop fails. The five expile experiment loops are essentially the same for the purpose of this discussion, e.g., piping material and sizes, loop configurations, pumps, and piping supports. The seismic event that may cause loop piping damage in one loop would also be expected to cause loop piping damage in the other four experiment loops.

Although experiment loop ruptures are addressed in SAR-153, Section 15.4, seismic induced loop piping failure must be considered with accompanying failures of the evacuation system (ATR Complex-USQ-2009-714), of the 8-in. confinement block wall, and of the four other experiment loops.

References:

Hendrickson, M. B., Evaluation of Seismic Criteria to be Used for Loop Piping Systems, TRA-ATR-1166, June 3, 1997

Knudson, D. L, Radiological Analysis Supporting an ATR UFSAR Revision, TRA-ATR-1562, May 30, 2000

A PISA exists. Complete the following actions

- Nuclear Facility Manager document actions (below) including interim operating restrictions taken to place or maintain the facility in a safe condition.
- Nuclear Facility Manager execute reporting process for PISA per LWP-9301 and LWP-13830.
- Qualified USQ evaluator proceed to Section III.

ACTIONS TAKEN TO PLACE OR MAINTAIN FACILITY IN A SAFE CONDITION AND THE BASIS FOR THE SELECTION OF ANY INTERIM CONTROLS.

ATR was in the 147-A outage at the time of discovery. No immediate operational restrictions were needed.

EDF TRA-ATR-1562, the consequence analysis referenced in SAR-153, Section 15.7.1, also calculated the maximum dose for full plume passage at the near site boundary (NSB). The NSB dose consequence analysis assumed confinement and one experiment loop failure; the NSB dose was 192 mrem. Although the analysis assumed confinement, the analysis also assumed full plume passage, instead of the two-hour dose as assumed in SAR-153, Section 15.12.10. Because full plume passage was assumed in the analysis and because of the plume travel time to the NSB, the affect of not having an intact confinement on the NSB dose would not be significant. The effect of all five experiment loops containing 200 kW fueled tests and all five loop experiencing leakage from the Condition 4 earthquake is estimated to be five times that of a single loop, or approximately 1 rem. The Condition 4 ATR Plant Protection Criteria (PPC) is 25 rem, whole body, for the off-site public. The maximum off-site whole body doses for all SAR-153 design basis accident analyses are 2.3 rem (NSB) and 4.1 rem (low-population-zone). Because the potential consequence is less than 1/2 at the NSB and 1/4 at the LPZ of the maximum analyzed consequences and less than 1/25 of the Condition 4 PPC, no interim controls are required.

J. C. Chapman	see original for signature (EDMS)	
Safety Analyst Print/Type Name	Safety Analyst Signature	Date

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E. J. Schuebert	see original for signature (EDMS)	
Nuclear Facility Manager Print/Type Name	Nuclear Facility Manager Signature	Date

Is independent technical review required? Yes ☐ No ☒

Nuclear Facility Manager Signature or Initials

Independent Reviewer Print/Type Name	Independent Reviewer Signature	Date

III. USQ DETERMINATION

Identify applicable section(s) of the safety basis document(s) (e.g., DSA, SAR, BIO, TSRs, etc.).

SAR-153 §ES-3.2.1 Design Basis Accidents

Evaluation of Condition 2, Condition 3, and Condition 4 events in Chapter 15 showed no events where direct fuel damage occurred concurrent with a breach of the PCS. The off- site and on site radiological consequences were reanalyzed only for the hypothetical beyond design basis loss of coolant accident resulting in 100% core damage. Radiological consequences from this event bound all design basis events.

SAR-153 Table ES-5 Summary table of worst case events resulting in a radioactive consequence

Identifies the canal fuel damage event described in SAR-153 §15.8.7 as the design basis event with the maximum off-site dose consequences. The TEDE is 2.3 rem at the NSB and 4.1 rem at the LPZ.

SAR-153 §2.1.3.4 Low Population Zone

The low population zone (LPZ) is the area between the INL boundary and three-fourths the distance to the nearest population center with a population greater than 25,000, which is Idaho Falls. The distance from the ATR Complex to the boundary of the LPZ is 34.5 mi (55.5 km).

SAR-153 §3.1.1.2.3 Seismic Analysis

The Design Basis Earthquake (DBE)/safe shutdown earthquake (SSE) for ATR as the maximum postulated seismic event for which it should be demonstrated that the reactor can be safety shut down and maintained in a safe condition. Currently, the DBE/SSE for ATR is defined by the site-specific probabilistic DBE documented in Payne et al. (2002) for PC3 and 4 SSCs.

SAR-153 §10.1.5 Classification of Experiment Structures, Systems, and Components

Loop piping is identified as important to safety and shall meet applicable overall requirements of General Design Criteria 2 for design bases for protection against natural phenomena. For loop piping, sudden catastrophic failure or rupture caused by a seismic event is beyond-design-basis (Hendrickson 1997a). After a seismic event at a level that would cause the PPS seismic subsystem to trip, the loop facilities will be inspected using Hendrickson (1997b) to ensure the loop facilities were not damaged during the event.

SAR-153 §10.1.7.1 Primary Experiment Safety Analyses Criterion

The consequences of normal operation of the experiment and of any experiment fault must be bounded by the ATR Plant Protection Criteria for the same operating condition (i.e., Conditions 1, 2, 3, and 4 as defined in Chapter 15 [Accident Analyses]) whenever the experiment is within the ATR facility.

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SAR-153 §10.2.2.3 Out-of-Pile Loop Evaluations

Sections states that sudden catastrophic failure or rupture of loop piping caused by a seismic event is beyond design basis (Hendrickson 1997a).

SAR-153 §15.0.3 Classification of Plant Conditions

Condition 1 is normal operation and operational transients that are expected to occur frequently or regularly in the course of reactor operation, refueling, or maintenance. Condition 2 events are faults of moderate frequency that is anticipated to occur once or more during the lifetime of the facility die to an expected single fault. Condition 3 events are infrequent faults that may occur infrequently during the life of the plant that result from single low frequency failures and combinations of independent high frequency failures. Condition 4 events are limiting faults that are not expected to occur but are postulated because their consequences include the potential for release of significant quantities of radioactive material.

According to 10 CFR 100, the exclusion area is the area surrounding the reactor in which authority exists to control all activities including the removal of personnel and property. Because the DOE has such authority within the INL borders, the INL site boundary was assumed to be the ATR exclusion boundary. For Condition 3 and 4 events, the off-site exposures at the NSB and the LPZ can differ in the time allowed to receive the dose. The exposure at the NSB (10.8 km from ATR) can be limited to two hours while the exposure at the LPZ (55.5 km from ATR) must be based on the entire passage of the radioactive plume.

SAR-153 §15.0.6 Safety Rod Insertion Characteristics

The negative reactivity insertion following a reactor trip is a function of the axial position of the safety rod at the time of trip, acceleration of the safety rods, and safety rod worth as a function of core position. For accident analysis, the critical parameter for safety rods is the time for insertion of reactivity to make the core subcritical or begin to balance a positive condition. For ATR, this reactivity is typically realized within the first 12 in. or approximately 33% of the safety rod travel. A trip signal to the safety rod subsystem will release the safety rods within 25 msec for pressurized operation and within 35 msec for depressurized operation. The safety rods will insert 12 in. into the core and effectively terminate power operation within 150 msec (for three pump operation) and 170 msec (for two pump operation). The safety rods will insert 12 in. into the core within 300 msec for depressurized operation.

SAR-153 §15.0.13 Operator Actions

For recovery from LOCA events, operator action is needed to sustain firewater injection by maintaining raw water inventories via deep well pumps and keeping firewater system fuel tanks replenished. The specific actions required are specified in the ATR Water Management Plan, which is included as Appendix C in FRP-C.1, "Inadequate Core Cooling".

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SAR-153 Table15.0-1

Table defines failures due to very low probability events, such as large earthquakes, as extremely unlikely faults (Condition 4).

SAR-153 Table15.0-6

Table identifies limiting (worst case) trip setpoints assumed in the accident analyses, and the assumed response time (delay time) for each trip function. Specifically, the high neutron level analysis trip point is 120% with a response time of 0.025 sec (0.035 sec for depressurized ops).

SAR-153 §15.4.4 Inpile Tube Decompression with Subsequent Voiding Due to Failure of the ½-Inch Piping

A Condition 2 event involving decompression and subsequent voiding of an IPT resulting from mechanical failure of any ½ in. or smaller piping. The limiting failure is a shear of a ½ in. pipe in the draining manifold attached to the loop piping at the heater leg which results in IPT voiding and a reactivity insertion. The pipe break is assumed to occur instantaneously resulting in a double-ended offset shear. A scram occurs on high power and is initiated by the neutron level subsystem. The dominant shutdown mechanism is feedback from the fuel elements. The safety rod insertion provides some mitigation of the event and ensures the reactor remains subcritical. Diverse protection is provided by the wide range subsystem.

The analyses assume a scram with a response time of 0.025 s occurring at 1.2 NF. An additional positive reactivity insertion is added as a ramp beginning at zero time and inserting 0.05$ in 0.13 s (it is constant after 0.13 s) to conservatively represent cascading from other loops. The RELAP5 (Nielsen 1994, Terry 1994) results for reactor power show a maximum of 428 MW at 0.12 s for the two-pump case and 365 MW at 0.13 s for the three-pump case. The ATR-SINDA and SINDA-SAMPLE calculations (Polkinghorne 1994) show that the event meets the acceptance criteria for a Condition 2 event (margins to CHF and FI exceed 3σ) as well as the criteria for a Condition 3 event (2σ to buckling). Results of the calculation are as follows:

Parameter	2-pump	3-pump
Peak fuel plate centerline temperature	635°F	660°F
CHF margin	3.40σ	3.3σ
Flow instability margin	4.3σ	4.3σ
Fuel plate buckling margin	3.3σ	2.7σ

The protective criteria margins will be preserved with maximum effective Plate 15 powers of 443 MW and 417 MW for the three-pump operation and the two-pump operation, respectively.

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SAR-153 §15.7.1 Pressurized Water Loop or Experiment Handling Failure

A LOCA in the experiment loop could result in melting of the test. A 1/2 in. experiment loop LOCA is considered a Condition 2 occurrence. A larger than 1/2-in. experiment loop LOCA is considered a Condition 4 event. If test damage were to occur, the LOCA events would allow fission product release into the loop cubicle and thus, the environment.

Analyses of the off-site consequences of a loop facility LOCA with fuel melt for a test with 200 kW fission power were completed (Knudson 2000). The computer codes employed were comparable to that used for dose analyses in SAR-153 Section 5.12. The ORIGEN code was used for the source term calculation and the RSAC 5 code was used for the dose analysis. The analysis assumed 100% of the noble gas inventory and 25% of the iodine inventory in the fuel was released to the ATR building. RMSS actuation was assumed. Worst-case (99.5%) site specific meteorology was used. The resulting dose was 192 mrem , TEDE, for the maximally exposed individual at the INL NSB. This result meets the Condition 2 ATR Plant Protection Criteria of 500 mrem off site dose.

The dose for a worker outside of the loop cubicle including inhalation and external dose was determined (Lucas and Wagoner 2001). The analysis assumed the worker evacuated the area in 5 minutes in response to radiation and air monitoring equipment alarms. The resulting dose was 1.95 rem, TEDE, which is below the Condition 2 ATR Plant Protection Criteria of 5 rem worker dose. The dose to other ATR Complex personnel was 7.4 mrem, TEDE. The dose to workers at INTEC was 15.8 mrem TEDE. INTEC doses are higher due to a longer evacuation time.

SAR-153 §15.8.7 Significant Melting of One Entire Fuel Element

Section includes discussion of a Condition 4 event that results in the maximum off-site dose consequences.

Potential doses due to significant fuel damage and melting in the storage canal (without canal draining) is enveloped by the analysis for the large-break LOCA discussed in SAR-153 Section 15.12.10. The dose calculated for 100% core melt considers release of 64.5% of the source term in the first day and the remainder over the next 10 days; the total dose is 185 rem thyroid and 13.2 rem EDE (whole body) at the LPZ. Assuming that all the calculated dose results from the first day release provides an estimate of the dose from 64.5% of the core or 25.8 fuel elements. If the event affects eight fuel elements, the fraction of the dose resulting would be 0.31 (the ratio of 8/25.8). The estimated dose at the LPZ is 57.4 rem thyroid and 4.1 rem whole body for failure of eight fuel elements. **Section 15.12.10 (Knudson 2000) identifies a dose at the NSB of 82.7 rem thyroid and 7.53 rem whole body. Scaling this dose as noted above results in a 25.6 rem thyroid and 2.3 rem whole body. (See ICAMS SD-001313).**

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SAR-153 §15.12.10 Radiological Analysis

A radiological analysis of a postulated hypothetical large break LOCA in the ATR is documented and provides the basis for a bounding radiological analysis. The accident is considered beyond design basis.

The INL site boundary was assumed to be the ATR exclusion boundary. The outer edge of the LPZ is defined as located at three quarters of the distance to the nearest population center of 25,000 or more. At 74 km from the ATR, Idaho Falls is the nearest qualifying population center. Consequently, the outer edge of the nearest LPZ lies at a radius of 55.5 km from ATR. The radiological limits of 10 CFR 100 (25 rem whole body, 300 rem thyroid) are assumed to be applicable to both off site personnel and the evacuating personnel on site. Centerline doses were calculated for exposure to plume passage at the outer edge of the nearest LPZ, for exposure at the nearest exclusion boundary, and for exposure of evacuating workers.

Potential doses at the LPZ were calculated for exposure to the maximum plume concentration during passage of the entire release. Exposure to the entire release corresponds to the assumption that individuals beyond the outer edge of the LPZ will not be evacuated. (It should also be noted that credit is not allowed for any shielding or sheltering.)

For exclusion boundary exposure, receptors were assumed to be located at that distance for a period of 2 h following the onset of the release. That is equivalent to assuming that individuals from the exclusion boundary to the outer edge of the LPZ will be evacuated within 2 h.

References

FRP-C.1, "Inadequate Core Cooling," Rev. 12, March 16, 2010.

Hendrickson, M. B., 1997a, "Evaluations of Seismic Criteria to be Used for Loop Piping Systems," EDF TRA-ATR-1166, June 3, 1997.

Hendrickson, M. B., 1997b, "ATR Loop Restart Inspection Criteria after a Large Seismic Event (6.0 Richter Scale)," EDF TRA-ATR-1231, June 17, 1992.

Knudson, D. L., 2000, "Radiological Analysis Supporting an ATR UFSAR Revision," EDF TRA-ATR-1562, May 30, 2000.

Lucas, D. S., 2000, "ATR Confinement Building Leak Rate Calculations," EDF TRA-ATR-1543, February 2000. Note: This reference is provided as referenced in Knudson (2000); however, revision 0 of this EDF is dated May 2000. The discrepancy in dates is believed to be a typographical error in Knudson (2000) resulting from the different signature dates of the performers on the EDF, specifically, the author and independent verifier signed the EDF in February 2000, while the requestor signed the EDF in May 2000.

Lucas, D. S. and S. R. Wagoner, 2001, "SAR Addendum for Unreviewed Safety Questions (USQ) on the ATR Confinement Leak Rate," EDF TRA-ATR-1588, Rev. 2, March 1, 2001.

Nielsen, H. H., 1994, "ATR NR Loop Reactivity Insertions from Voiding," EDF TRA-ATR-850, April 29, 1994.

Payne, S. J., et al., 2002, *Development of Probabilistic design basis Earthquake (DBE) Parameters for Moderate and High Hazard Facilities at INEEL,* INEEL/EXT-99-00775, Rev. 2, June 2002.

Polkinghorne, S. T., 1994, "ATR-SINDA and SINDA-SAMPLE Calculations for Chapter 15 of the ATR's Updated FSAR," EDF TRA-ATR-840, February 11, 1994.

Terry, W. K., 1994, "Reactivity Insertion Analyses with RELAP5," EDF TRA-ATR-841, April 19, 1994.

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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:

The pertinent accidents of concern are a loop loss of coolant accident and a loop voiding reactivity insertion accident. A small, less than ½ in., loop LOCA and the decompression and subsequent voiding of an IPT are both considered Condition 2 frequency accidents. The seismic event postulated to initiate loop piping leakage is considered Condition 4. In both instances, the accident initiator identified in the PISA is a postulated seismic event. A Condition 4 initiating event would not increase the probability of occurrence of the Condition 2 loop LOCA and reactivity insertion events. Because the frequency of occurrence of such a seismic event remains unchanged, the probability of the accident(s) also remains unchanged. Thus, the PISA does not increase the probability of occurrence of an accident previously evaluated in the safety basis.

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

SAR-153 Section 15.7.1 describes a Condition 2 postulated ½ in. experiment loop LOCA. Because Hendrickson (1997a) supported the seismic ruggedness of the out-of-pile experiment loop piping by concluding that catastrophic failure from the SSE is incredible, unreviewed safety question ATR Complex-USQ-2009-714 postulated that seismic induced experiment loop piping leakage due to the Condition 4 earthquake (SSE) would be enveloped by the limiting Condition 2, ½ in. loop piping LOCA consequence. The analysis for the potential off-site dose consequence (Knudson 2000) overestimated the dose at the NSB by calculated full plume passage in lieu of the 2 hr dose; however, the analysis only considered the dose consequence resulting from the failure of the piping from one experiment loop. Because the seismic event that may cause loop piping damage in one loop would also be expected to cause loop piping damage in the other four experiment loops, the amount of radioactive material released is underestimated. In addition, the analysis took credit for some confinement by assuming a confinement leakage of 51.4% per day based on Lucas (2000); however, the block wall separating the reactor main floor from the canal area is not expected to maintain its "gas tight boundary" integrity following the Condition 4 earthquake. Thus, the confinement leakage rate assumed in the analysis is too low. Because the dose calculated in Knudson underestimated the rate of confinement leakage and the amount of radioactive material released, the dose consequence is also underestimated. Likewise, the analysis for the potential on-site consequence also only considered the failure of one experiment loop and assumed evacuation as indicated in ATR Complex-USQ-2009-714.

Of additional concern is a Condition 4 seismic event that results in postulated LOCAs in both the experiment loops and the PCS. SAR-153 Chapter 15.6 evaluates decrease in primary coolant inventory accidents, which includes seismically-induced PCS leakage. For recovery from LOCA events, operator action is needed to sustain firewater injection by maintaining raw water inventories via deep well pumps and keeping firewater system fuel tanks replenished (SAR-153 Section 15.0.13). For Condition 3 and 4 events, on-site dose criteria are applied to evacuating workers in neighboring facilities in lieu of in-facility workers. Because of the need to perform water management duties in the event that EFIS is actuated in response to PCS leakage, worker evacuation will be purposely delayed for some key personnel. Thus, the consequence analysis must consider on-site emergency action dose criteria for personnel at ATR Complex, not just neighboring facilities. SAR-153 does not evaluate the ability of non-evacuating personnel at ATR Complex to perform water management duties in response to the seismically-induced PCS leakage concurrent with the radiological release resulting from the loop LOCA.

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The increase in the amount of radioactive material released, the lack of confinement, and the potential consequence to non-evacuating personnel at ATR Complex will contribute to an increase in the consequence of the accident. Thus, the PISA increases the consequence 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:

SAR-153 Section 15.4.4 evaluates the Condition 2 event involving decompression and subsequent voiding of an IPT resulting from mechanical failure of any ½ in. or smaller piping. The analysis considered the reactivity insertion resulting from voiding of one experiment loop. Because the seismic event that may cause loop piping damage in one loop would also be expected to cause loop piping damage and subsequent voiding in the other four experiment loops, the amount of reactivity insertion is underestimated.

The ATR fuel element cladding is the primary barrier against the release of fission products. Adequate heat removal is required during and following accident conditions to ensure the integrity of the fuel element cladding is maintained. An increase in the amount of the reactivity insertion will result in an increase in energy deposition to the fuel elements which could challenge the ability of the fuel element cladding to contain fission products. The increased reactivity insertion could increase the probability of fuel element cladding failure. Thus, the PISA increases the probability of occurrence of a malfunction of equipment important to safety previously evaluated in the safety basis.

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

Explain:

Experiment loop piping protects against a release of radioactive material and is considered equipment important to safety. As discussed in question 2, SAR-153 Section 15.7.1 describes a Condition 2 postulated ½ in. experiment loop LOCA; however, the analysis (Knudson 2000) only considered the dose consequence resulting from the failure of the piping from one experiment loop. Because the seismic event that may cause loop piping damage in one loop would also be expected to cause loop piping damage in the other four experiment loops, the amount of radioactive material released is underestimated. Thus, the consequence of a malfunction of experiment loop piping is increased.

Likewise, the ATR confinement is considered equipment important to safety to protect against the release of radioactive material. Knudson (2000) took credit for some confinement by assuming a confinement leakage of 51.4% per day based on Lucas (2000); however, the block wall separating the reactor main floor from the canal area is not expected to maintain its "gas tight boundary" integrity following the Condition 4 earthquake. Thus, the confinement leakage rate assumed in the analysis is too low. The safety basis does not evaluate the potential dose consequence of a seismic event that results in the failure of five loops concurrent with a loss of confinement. Thus, the consequence of a failure of confinement during the Condition 4 seismic event is increased.

The primary coolant system (PCS) boundary is also considered equipment important to safety to protect against the release of radioactive material. A Condition 4 seismic event could result in postulated LOCAs in both the experiment loops and the PCS. As discussed in question 2, SAR-153 Chapter 15.6 evaluates decrease in primary coolant inventory accidents, which includes seismically-induced PCS leakage. For recovery from LOCA events, operator action is needed to sustain firewater injection by maintaining raw water inventories via deep well pumps and keeping firewater system fuel tanks replenished (SAR-153 Section 15.0.13). For Condition 3 and 4 events, on-site dose criteria are applied to evacuating workers in neighboring facilities in lieu of in-facility workers. Because of the need to perform water management duties in the event that EFIS is actuated in response to PCS leakage,

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worker evacuation will be purposely delayed for some key personnel. Thus, the consequence analysis must consider on-site emergency action dose criteria for personnel at ATR Complex, not just neighboring facilities. SAR-153 does not evaluate the ability of non-evacuating personnel at ATR Complex to perform water management duties in response to the seismically-induced PCS leakage concurrent with the radiological release resulting from the loop LOCA. Thus, the consequence of a failure of the PCS boundary may be increased.

The PISA has the potential to increase the consequence of failures of experiment loop piping, confinement, and the PCS boundary. Thus, the PISA increase the consequences of a malfunction of equipment important to safety previously evaluated in the safety basis

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:

The PISA discusses seismic-induced experiment loop leakage from damage to all five experiment loops. In addition to an increased radioactive material release, a loop LOCA would also result in inpile tube voiding and an increased reactivity insertion.

SAR-153 analyzes experiment loop LOCAs, experiment loop reactivity insertion events, and PCS LOCAs; however, the SAR does not analyze a reactivity insertion event concurrent with a PCS LOCA. Because a Condition 4 seismic event could result in LOCAs in both the experiment loops and the PCS; the potential reactivity insertion resulting from the experiment loop leakage concurrent with the PCS LOCA represents a new type of accident. Thus, the PISA does create the possibility of an accident of a different type than 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:

The PISA addresses seismic-induced experiment loop leakage from damage to all five experiment loops. SAR-153 addresses failures of experiment loop piping, confinement, and the PCS boundary. Although the PISA proposes that all five experiment loops fail, the out-of-pile experiment loop piping is located in separate cubicles: 2 in the ATR first basement and 3 in the ATR second basement. Because of their location and separation, the amount of leakage is not anticipated to adversely impact other equipment important to safety. Thus, 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:

The PISA discusses seismic-induced experiment loop leakage from damage to all five experiment loops. In addition to an increased radioactive material release, a loop LOCA would also result in inpile tube voiding and an increased reactivity insertion.

Likewise, SAR-153 Section 15.4.4 evaluates the effects of a reactivity insertion event resulting from decompression and subsequent voiding of an IPT resulting from mechanical failure of any ½ in. or smaller piping. The ATR-SINDA and SINDA-SAMPLE calculations estimate margins to CHF of 3.40σ and 3.3σ for 2-pump and 3-pump operations

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respectively, margins to FI of 4.3σ for both 2-pump and 3-pump operations, and margins to fuel plate buckling of 3.3σ and 2.7σ for 2-pump and 3-pump operations respectively. The protective criteria margins will be preserved with maximum effective Plate 15 powers of 443 MW and 417 MW for the three-pump operation and the two-pump operation, respectively. However, the PISA proposes that all five experiment loops will fail, which may result in a larger reactivity insertion and increased energy deposition in the fuel. Thus, the margins to CHF, FI, and fuel plate buckling will be reduced.

The PISA may reduce margins to CHF, FI, and fuel plate buckling. Thus, the PISA will reduce a margin of safety as defined in the safety basis.

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 PISA addresses seismic-induced experiment loop leakage from damage to all five experiment loops and the resulting increased reactivity insertion and radioactive material release.

As discussed above, the PISA increases the consequence of an accident previously evaluated in the safety basis, increases the probability of occurrence of a malfunction of equipment important to safety previously evaluated in the safety basis, increases the consequences of a malfunction of equipment important to safety previously evaluated in the safety basis, and reduces a margin of safety of safety as defined in the safety basis. In addition, the PISA creates the possibility of an accident of a different type than previously evaluated in the safety basis. Thus, the PISA constitutes an unreviewed safety question.

If "No," is a Safety Basis document revision required at next annual update? ☐ Yes ☐ No

NOTE: If USQ determination result is positive, additional notification for ORPS is required per LWP-9301.

IV. APPROVAL/CONCURRENCE SIGNATURES

APPROVAL:

Print/Type Name	Signature	Date
Anne K. McCartin / John C. Chapman (USQ Evaluator)	Anne K. McCartin / John C. Chapman (USQ Evaluator)	6-23-2010
Edmond J. Schuebert (Nuclear Facility Manager)	E J Schuebert (Nuclear Facility Manager)	6/23/10 1730

CONCURRENCE:

Print/Type Name	Signature	Date
Jack A. Jacobi (Independent Review Committee Chair)	Jack A. Jacobi (Independent Review Committee Chair)	6/24/2010 1326