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

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Facility or Activity: Advanced Test Reactor

USQ Process No.: ATR Complex-USQ-2009-714

Subject: Evacuation System Seismic Requirements not Supported by Seismic Design of Buildings

Describe the New Information/Discovery:

Application of Natural Phenomenon Hazard (NPH) requirements to the evacuation system is not consistent with the seismic design of several buildings at the ATR Complex which house and physically support the evacuation sirens. For example, evacuation sirens required to meet NPH Performance Category 2 (PC-2), in accordance with DOE-STD-1021-93, are mounted within office buildings that are not designed to withstand an equivalent seismic event. Hence, building design may prevent some evacuation sirens from functioning following a seismic event. Furthermore, post seismic evacuation of all collocated personnel, including those at neighboring facilities, is not guaranteed considering the potential personnel injuries and damage to occupied buildings. The evacuation system seismic designation also complicates construction of new buildings which potentially require internal installation of PC-2 evacuation sirens.

The evacuation system sirens are relied on in the safety basis for the Advanced Test Reactor (ATR) to notify personnel and initiate an ATR Complex wide evacuation. That is, some safety basis radiological consequence analyses, identified in the ATR Safety Analysis Report (SAR-153), assume that ATR Complex personnel are evacuated in order to demonstrate that the calculated exposures are within the limits of the ATR Plant Protection Criteria (PPC). Demonstrating the safety of collocated workers is required by the Department of Energy (DOE) and is incorporated into the ATR PPC. The accident analyses that rely on evacuation include postulated events initiated by a seismic event.

In recent years the evacuation system performance category for Natural Phenomenon Hazards (NPH) was reviewed and determined to be PC-2, not PC-4 as previously considered. This designation is in part based on limited reliance on the evacuation system to provide protection to ATR Complex worker only following a seismic event. The evacuation system does not provide a public safety function. The PC-2 designation and basis were provided informally to the Department of Energy Idaho Operations Office (DOE-ID).

Applicable SAR-153 safety basis analyses for postulated events that rely on the evacuation system following a seismic event include:

- Decrease in Primary Coolant Inventory (i.e., primary coolant system LOCA) (Section 15.6),
- Pressurized Water Loop or Experiment Handling Failure (i.e., experiment loop LOCA) (Section 15.7), and
- Fuel Storage Canal and Cask Handling Events (Section 15.8).

Note that the evacuation system is not relied on to limit radiological exposures to the public and, therefore, is not designated as a safety-related or safety class structure, system or component (SSC). The Technical Safety Requirements for the ATR (TSR-186) also does not impose any performance or operability controls related to the evacuation system. The worker protection safety function of the evacuation system is established in SAR-153. SAR-153 only identifies those SSCs designated at Seismic Category 1, but does not identify SSCs with lesser seismic designations.

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

SAR-153, Safety Analysis Report for the Advanced Test Reactor
TSR-186, Technical Safety Requirements for the Advanced Test Reactor

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

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

N/A		
USQ Evaluator Print/Type Name	USQ Evaluator Signature	Date
N/A		
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.

The design of some support buildings, including office buildings, is not consistent with the seismic requirements of the evacuation system. The seismic vulnerability of some support buildings at the ATR Complex could result in failure of the evacuation system sirens to perform their intended safety function following a PC-2 seismic event.

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.

No immediate action is necessary to place the facility in a safety condition.

As identified previously, there are three categories of postulated accidents which potentially rely on the evacuation system following a seismic event. The following evaluations provide assurance that the ATR PPC are currently met for ATR Complex workers without reliance on the evacuation system sirens. Therefore, no restrictions on reactor operation are required (e.g., no power reduction); however, an interim control is required with respect to the safety basis PPC for Condition 4 events (discussed below).

Decrease in Primary Coolant Inventory (i.e., primary coolant system LOCA) (Section 15.6)

The current seismic LOCA analysis for the ATR is presented in EDF-5668. This analysis supports the interim safety basis document in an addendum to SAR-153 (see EDF-5614). The seismic LOCA calculations in EDF-5668 analyze the postulated failure of seismically vulnerable plant equipment (valves PCV-1-1 and LCV-1-3C) in conjunction with a 1-inch equivalent PCS break due to seismically-initiated failure of small bore piping. The minimum critical heat flux (CHF) and flow instability (FI) margins for the Condition 3 seismic LOCA

are reported as 4.49σ and 5.34σ, respectively. With such margins, fuel damage is not a credible concern.

For the Condition 4 LOCA, the minimum CHF and FI margins are 3.40 σ and 1.64σ, respectively. Since the CHF and FI margins are ≥ 1.2σ for the Condition 4 fault, the margin to the aluminum-water ignition threshold (AWIT) temperature is >3σ (see SAR-153 Section 15.0.14). In addition, the fuel plate cladding temperatures are well below the buckling temperature (EDF-5668 Figures 12, 18 and 22). Since the cladding temperature is well below the buckling temperature, no fuel significant fuel damage is predicted.

An additional LOCA analysis was prepared subsequent to EDF-5668. The analysis for a very small seismically-initiated LOCA (equivalent to a complete loss of flow with seismically-induced leakage) is documented in EDF-6503. EDF-6503 demonstrates that for the worst-case condition temperatures are well below the fuel plate buckling temperature and, therefore, the margin to fuel plate buckling is > 2σ. Therefore, the Condition 3 PPC are met and no significant fuel damage is predicted to occur.

Finally, ECAR-226 documents the analysis for seismically-induced LOCA and for a very small seismically-initiated LOCA (complete loss of flow with seismically-induced leakage) during operation with an aged reflector (i.e., with assumed flow blockage and a reduced plate power limit). ECAR-226 (see Cases 4 and 5) demonstrates that the results of a seismically-initiated Condition 3 or Condition 4 LOCA, during operation with an aged reflector, are bounded by the corresponding cases analyzed in EDF-5668. The analysis also concludes (see Case 6) that the consequences of a very small seismically-initiated LOCA continue to meet the Condition 3 PPC. Therefore, no significant fuel damage is predicted to occur.

Note that the large PCS piping is not predicted to fail during a seismic event. That is, a seismic event is not postulated to cause a credible large break LOCA. Therefore, analyses and results for direct damage LOCAs that apply the alternate Condition 4 criteria associated with reactor operation at high power, such as those analyzed for the break spectrum study, are not affected by the evacuation system seismic requirements or evaluation herein.

Pressurized Water Loop or Experiment Handling Failure (i.e., experiment loop LOCA) (Section 15.7)

Two experiment loop failures are considered (Section 15.7.1) for the experiment loops LOCAs. These are a ½-inch or smaller break, which is considered a Condition 2 event, and a greater than ½-inch break, which is considered a Condition 4 event. Although the loop piping is not designed to modern seismic standards, a qualitative evaluation of its seismic ruggedness has been documented in EDF TRA-ATR-1166. This evaluation allows that a design basis earthquake may create or cause to progress "small, slowly progressing cracks", and concludes that, "though catastrophic failure of the loop piping is beyond design basis, small cracks or leaks are possible." The seismic damage described in EDF TRA-ATR-1166 is consistent with the Condition 2, less than ½ inch LOCA. The larger Condition 4 loop LOCA would be greater than a "small crack or leak" and therefore a seismic initiator of an experiment loop large LOCA would be beyond design basis.

For the Condition 2 experiment LOCA the worker dose for evacuating (7.41 mrem) and non-evacuating (3.71 rem) scenarios are calculated in TRA-ATR-1562. Although the evacuating dose is presented in the SAR, the results for each scenario satisfy the Plant Protection Criteria (PPC) worker whole body dose limit of ≤ 5 rem. The ATR PPC for Condition 2 events do not establish a thyroid dose limit; however, thyroid dose limits are defined for Condition 3 (75 rem) and Condition 4 (300 rem) events. Although there is no established thyroid limit for a Condition 2 event, it is not acceptable for the consequences of a Condition 2 event to exceed the thyroid dose limit for a Condition 3 or 4 event. TRA-ATR-1562 includes the calculated thyroid dose of 97.6 rem for personnel at the ATR Complex for full plume passage (no worker evacuation). This exceeds the Condition 3 limit of 75 rem. However, full plume passage is conservatively modeled as 47 hours without consideration for limited exposure duration due to the normal work shift. Credible exposure to the plume would be less than 24 hours (normal work shift for reactor operators is 12 hours). Considering a reduced (credible) duration of the

plume exposure, the thyroid dose would be shown to be within the 75-rem limit.

Therefore, the consequences of a seismically initiated experiment loop LOCA, without reliance on the evacuation sirens or ATR Complex evacuation, or evacuation of other INL desert site facilities, are within the PPC.

Fuel Storage Canal and Cask Handling Events (Section 15.8)

Irradiated experiments can contain quantities of radionuclides which may be harmful if released to the environment. A further consideration for this class of accident is that the direct radiation dose rate from an unshielded irradiated experiment could be great enough to require evacuation of workers to satisfy Plant Protection Criteria for worker dose.

When not well-shielded by water an irradiated experiment is contained in a closed cask. A crane or winch is used to handle the loaded cask. Dropping the experiment would be the result of a hardware failure or operator error. The ATR bridge crane structures were shown not to collapse under SSE loading (EDF-7210 Revision 3). Earthquakes do not apply significant lateral loads to cranes, because the loads are frequency isolated. The vertical demand of an earthquake on a suspended load is limited to 1.0 g, above which the response will deamplify by formation of slack on the rope. Standard hoisting and rigging factors of safety are at least 3.0, which allows ample margin for a 1.0 g vertical seismic load. For these reasons, seismic experience has shown bridge cranes and winches to be robust. Therefore, dropping an irradiated experiment is considered a random event independent from seismic activity.

Given that during the 40-year operating history of the ATR no irradiated experiment handling accident has required plant evacuation, a Bayesian probability estimate (see EDF-6722) that such an accident would occur on any given day is approximately 3.4E-05. The probability of occurrence of a major earthquake (10 CFR 100) on any given day is approximately 5.5E-05. Combining the two independent random events gives a joint frequency on the order of 1.9E-09 per day or 6.8E-07 per year. This low probability makes an experiment handling accident, concurrent in terms of consequence and response with a major earthquake, a beyond design basis scenario.

Irradiated fuel (as defined in TSR-186), unirradiated fuel, and experiments are temporarily stored in the ATR canal. The canal walls, liner, and bellows are structurally qualified to withstand the safe shutdown design basis earthquake (ECAR-380 and Paul C Rizzo Associates Inc. Report 053472-02 Revision 3). Therefore, damage to the canal resulting in a draining event as a direct result of a seismic event is beyond design basis.

Damage to the canal is also postulated from accidental drops of heavy objects. There are administrative controls in place to limit the risk of this type of accident. The irradiated fuel is also physically protected from the consequence of canal draining by the short bulkheads. Due to physical and administrative barriers, uncovering irradiated fuel as a result of a dropped object damaging the canal is beyond design basis, and is particularly unlikely to be caused by a seismic event. Potential canal drain and uncovering unirradiated fuel is also a beyond design basis event. As discussed above under "Experiment Handling Accidents," dropping a crane-suspended load is considered a random event independent of seismic activity.

Another plausible mechanism for radionuclide release from stored fuel is direct damage to the irradiated or unirradiated fuel, without concurrent canal draining. The only postulated seismic initiator for such damage would be dropping a heavy object or some type of architectural building debris onto the fuel. Walkdowns of the canal area conducted for the ATR Seismic Assessment project considered this possibility and concluded that there are no credible seismic interaction hazards to canal area safety related SSCs (ARES Corporation Report 060230101-002). Lifting heavy objects over irradiated fuel, or over fuel storage grids containing irradiated fuel, is not permitted. Lifting heavy objects over unirradiated fuel is also not permitted. Postulated heavy drop of

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any type is an infrequent occurrence. It is not credible that these independent improbable random events (lifting heavy object over fuel, dropping heavy object, occurrence of major earthquake) would occur within a short time period such that their consequences would be concurrent and additive.

Other mechanisms for stored irradiated fuel damage, such as improper cutting, either would not release large activities of radionuclides, would have no plausible seismic initiator, or both.

For these reasons, seismically initiated damage to stored fuel or experiments requiring ATR Complex evacuation is beyond design basis, and these postulated accidents do not rely on the evacuation system sirens following a seismic event.

The above evaluations demonstrate that the radiological exposures to ATR Complex collocated personnel (not considered directly at the location of the accident) continue to be met without reliance on the evacuation system sirens to initiate a complex-wide evacuation. However, the Condition 4 PPC for reactor accidents allow significant melting and fission product release from the fuel. Hence, the PPC do not ensure on a continuing basis that the calculated consequences of postulated events (i.e., future analyses) do not require evacuation. Therefore, an interim control is required to ensure that Condition 4 events do not result in significant fuel damage or that an accident-specific radiological analysis be performed (without reliance on the evacuation system sirens) if significant fuel damage is postulated. The follow additional acceptance criterion is to be applied to Condition 4 accident analyses:

"For Condition 4 postulated accidents, initiated by a seismic event, new or revised accident analyses implemented into the safety basis shall demonstrate that potential fuel damage is bounded by existing safety basis analyses."

G. L. Sharp Safety Analyst Print/Type Name	on file (EDMS) Safety Analyst Signature	Date
E. J. Schuebert Nuclear Facility Manager Print/Type Name	on file (EDMS) Nuclear Facility Manager Signature	Date

Is independent technical review required? Yes ☐ No ☒

on file (EDMS)
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; Executive Summary, Chapters 3, 9, 13, 15, and 16

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 seismic design characteristics of ATR Complex buildings, the ATR Complex evacuation system, or buildings at neighboring facilities are not accident initiators. The accident initiators relative to post-seismic personnel evacuation are based in the frequency of occurrence of postulated seismic events. Therefore, there is no increase in 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 presents the results for a Condition 2 postulated experiment loop failure resulting from a seismic event. The radiological consequences to collocated workers, presented in Section 15.7.1, are based on their evacuation. Although continued operation of the ATR has been supported by demonstrating that radiological consequences remain within acceptable limits (see Section II), an increase in the exposure to collocated workers above that reported in SAR-153 is possible without reliance on evacuation. In addition, the basis for the Condition 4 PPC inherently assumes that damage to support buildings does not inhibit evacuation of collocated workers, if necessary, which may not be achievable following a seismic event. Therefore, the PISA 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:

Important to safety equipment is defined in SAR-153 Section 3.2 as structures, systems, or components (SSC) "that provide reasonable assurance that the facility can be operated without undue risk to the health and safety of the public." Seismic design requirements of ATR Complex support buildings and the evacuation system do not increase or reduce the risk to the health and safety of the public. Hence, based solely on the definition of important to safety in SAR-153 support buildings and the evacuation system would not be designated as important to safety. However, the meaning of "important to safety" equipment as applied in the USQ process is not limited to SSC which affect risk to the public. Therefore, postulated failure of ATR Complex support buildings or evacuation system components represents a potential increase in the probability of occurrence of a malfunction of equipment important to safety previously evaluated in the safety basis.

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

Important to safety equipment is defined in SAR-153 Section 3.2 as structures, systems, or components (SSC) "that provide reasonable assurance that the facility can be operated without undue risk to the health and safety of the public." Seismic design requirements of ATR Complex support buildings and the evacuation system do not increase or reduce the risk to the health and safety of the public. Hence, based solely on the definition of important to safety in SAR-153 support buildings and the evacuation system would not be designated as important to safety. However, the meaning of "important to safety" equipment as applied in the USQ process is not limited to SSC which affect risk to the public.

SAR-153 Section 15.7.1 presents the results for a Condition 2 postulated experiment loop failure resulting from a seismic event. The radiological consequences to collocated workers, presented in Section 15.7.1, are based on their evacuation. Although continued operation of the ATR has been supported by demonstrating that radiological consequences remain within acceptable limits (see Section II), an increase in the exposure to collocated workers above that reported in SAR-153 is possible without reliance on evacuation. In addition, the basis for the Condition 4 PPC inherently assumes that damage to support buildings does not inhibit evacuation of collocated workers, if necessary, which may not be achievable following a seismic event. Therefore, the PISA could 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:

Seismic vulnerability of ATR Complex support buildings and the evacuation system performance potentially impacts the radiological exposure of evacuating collocated workers during postulated accidents. Support building design and evacuation system performance do not initiate any reactor accident sequence or introduce 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:

Important to safety equipment is defined in SAR-153 Section 3.2 as structures, systems, or components (SSC) "that provide reasonable assurance that the facility can be operated without undue risk to the health and safety of the public." Seismic design requirements of ATR Complex support buildings and the evacuation system do not increase or reduce the risk to the health and safety of the public. Hence, based solely on the definition of important to safety in SAR-153 support buildings and the evacuation system would not be designated as important to safety. However, the meaning of "important to safety" equipment as applied in the USQ process is not limited to SSC which affect risk to the public.

SAR-153 Section 15.7.1 presents the results for a Condition 2 postulated experiment loop failure resulting from a seismic event. The radiological consequences to collocated workers, presented in Section 15.7.1, are based on their evacuation. Potential failure of support buildings and the evacuation system due to a seismic event is not currently assumed in the safety basis. Therefore, the PISA could create the possibility of a malfunction of equipment important to safety of a different type than previously evaluated in the safety basis.

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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:

SAR-153 Section 15.7.1 presents the results for a Condition 2 postulated experiment loop failure resulting from a seismic event. The radiological consequences to collocated workers, presented in Section 15.7.1, are based on their evacuation. Although continued operation of the ATR has been supported by demonstrating that radiological consequences remain within acceptable limits (see Section II), an increase in the exposure to collocated workers above that reported in SAR-153 is possible without reliance on evacuation. In addition, the basis for the Condition 4 PPC inherently assumes that damage to support buildings does not inhibit evacuation of collocated workers, if necessary, which may not be achievable following a seismic event. Therefore, the PISA could increase the radiological consequences and, thereby, reduce the margin of safety.

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 could increase the consequences of an accident previously evaluated in the safety basis (Question 2).

The PISA could increase the probability of occurrence of a malfunction of equipment important to safety previously evaluated in the safety basis (Question 3).

The PISA could increase the consequences of a malfunction of equipment important to safety previously evaluated in the safety basis (Question 4).

The PISA could create the possibility of a malfunction of equipment important to safety of a different type than previously evaluated in the safety basis (Question 6).

The PISA could reduce a margin of safety as defined in the safety basis (Question 7).

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:

G. L. Sharp USQ Evaluator Print/Type Name	USQ Evaluator Signature	29 Sep 2009 Date
E. J. Schuebert Nuclear Facility Manager Print/Type Name	Nuclear Facility Manager Signature	9/29/09 1110 Date

CONCURRENCE:

J. A Jacobi Independent Review Committee Chair Print/Type Name	Independent Review Committee Chair Signature	9/29/09 1130 Date