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.: RTC-USQ-2007-726

Subject: Evaluation of Confinement Pressure Transient Capability (GAP-013-07)

Describe the New Information/Discovery:

ATR Design Basis Reconstitution Project identified five issues with the ATR safety-basis evaluation of potential confinement over pressurization:

Issue 1: SAR-153 over-states the capability of the confinement to withstand an over-pressure event.
Issue 2: The RMS-2 function for confinement over-pressure protection was eliminated without adequate evaluation.
Issue 3: In appropriate extrapolation of confinement performance data.
Issue 4: Inadequate accounting for potential confinement heat sources.
Issue 5: Confinement under pressure events have not been evaluated

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

Atkinson, S. A., ATR Containment Failure Potential during an ATR Accident, TR-808, March 25, 1976

Burr, T. K., letter to D. R. Mousseau, ATR Confinement Area Design, TKB-19-87, September 9, 1987

Close, J. A., ATR Hydrogen Behavior and Confinement Performance, PG-T-88-020, September 1988

Close, J. A., ATR Hypothetical Accident Description, PG-T-88-021, September 1988

Lucas, D.S , ATR Confinement Building Leak Rate Calculations, TRA-ATR-1543, Rev. 2, March 14, 2005

Rolfe, R. L. letter to Distribution, ATR Confinement Upgrade Study, RLRO-07-88, May 24, 1988

Safety Analysis Report Advanced Test Reactor - Volume 1 of 2, IDO-17021, April 1965

Unreviewed Safety Question Evaluation, Elimination of RMS-2 Action Check, TRA-USQ-SE-98-050, August 26, 1998

Upgraded Final Safety Analysis Report for the Advanced Test Reactor, SAR-153, Current Issue

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

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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.

Issue 1: SAR-153 over-states the capability of the confinement to withstand an over-pressure event.

SAR-153 General Design Criterion 16, Confinement Design, requires that reactor confinement and associated systems shall be provided to establish a barrier against the uncontrolled release of radioactivity to the environment and to assure that the confinement design conditions important to safety are not exceeded for as long as postulated accident conditions require.

The confinement integrity analyses in TR-808, PG-T-88-020, and PG-T-88-021 evaluate various severe accidents resulting in confinement pressurization. These analyses conclude the resulting over-pressurizations are acceptable based on a confinement design pressure limit of 7.5 inches of water. However, there is no verification that the confinement can maintain the safety design leak rate integrity after an over-pressure event reaching 7.5 inches.

The confinement leak rate used in the safety analysis, TRA-ATR-1543, is based on measured leak rate data at very low pressures (i.e., less than 1 in. of water). There are no test data supporting the conclusion that the confinement leak integrity will be maintained after an elevated pressure transient.

This issue was previously identified and evaluated in 1988. RLRO-07-88 concluded that severe structural damage to the confinement structure was unlikely at pressures near 7.5 inches of water; however, confinement seal degradation was expected. RLRO-07-88 concluded installation of a confinement over-pressure protection feature was preferable to performing proof testing at elevated pressures. An over-pressure protection feature has not been installed.

Issue 2: The RMS-2 function for confinement over-pressure protection was eliminated without adequate evaluation.

One of the design functions of the Radiation Monitoring System (RMS) was to mitigate potential confinement structure over-pressure from selected events that might involve release of radioactivity coupled with release of reactor or pressurized water loop coolant (SAR-153 Section 12.5.2.2.1, RMS Protective Function).

The analysis supporting this function is in the 1965 Advanced Test Reactor Safety Analysis Report, IDO-17021. This analysis is based on the performance characteristics of the RMS-2 feature. The RMS-2 feature provided this function by initiating a trip of the ventilation supply while the exhaust was still operating.

The evaluations in IDO-17021 are flawed in that they assume there is a release coincident with the over-pressurization. For the pressurized water loop facility cases considered, the fuel failure and radioactive material release could occur well after the blowdown of the loop facility.

The RMS-2 function was removed in 1998. The unreviewed safety question evaluation supporting removal of the RMS-2 function (USQ-RTC-SE-98-050) argued that the over-pressure mitigation function was provided by automatic action of the HVS-1 variable speed control feature. While this may be an adequate position, the functional performance of this control feature was not evaluated, the credit taken for this control feature was not evaluated to establish the safety category of the control feature, and SAR-153 was not updated to reflect elimination of this protective function from the RMS section.

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Issue 3: Inappropriate extrapolation of confinement performance data.

PG-T-88-020 and SAR-153 Section 6.2.3.2, System Performance, both extrapolate confinement leak rate data far beyond the range of measured data. PG-T-88-020 extrapolates from data measured at less than 1 inch of water to 7.5 inches of water. SAR-153 extrapolates from data measured at less than 0.5 inches of water to 7.5 inches of water. Also the SAR-153 extrapolation is based on an upper bound fit of the measured confinement leak rate. This is inappropriate and non-conservative for the pressure relief assessment. Pressure relief capability should be based on minimum rather than maximum measured leak rate data.

While the confinement structural integrity should be maintained at up to 9.0 inches of water (TKB-19-87), the design basis leak rate integrity probably would not be maintained at this elevated pressure. Some seal materials would be expected to fail at 7.5 inches of water (RLRO-07-88).

Issue 4: Inadequate accounting for potential confinement heat sources.

SAR-153 Section 3.1.4.9, Criterion 38, Confinement Heat Removal, requires that a system to remove heat from the reactor containment (sic) shall be provided. The system safety function shall be to reduce rapidly, consistent with the functioning of other associated systems, the containment (sic) pressure and temperature following any loss of coolant accident and maintain them at acceptable low levels.

In discussing compliance with this requirement SAR-153 states, "Since the ATR is a moderate pressure, low temperature facility, a confinement heat removal system is not needed," (i.e., GDC 38 is not applicable).

The discussion neglects the potential for failures in the pressurized water loop experiment facilities as a potential heat source. The effect of the loop blowdown on confinement over-pressure was evaluated in IDO-17021, however, the evaluation took credit for the RMS-2 over-pressure protection function which has been eliminated. (See Issue 2.)

Issue 5: Confinement under pressure events have not been evaluated

Failure of the reactor building main supply fan (HVS-1) with the exhaust fan running results in a negative pressure of the confinement. This negative pressure has the potential to damage confinement penetration sealing materials. However, confinement under-pressure events have not been analyzed.

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 action was necessary to place the plant in a safe condition. The plant will continue to operate.

Any failure of HVS-1 which result in a building under-pressure will require the following:

If the reactor is in an outage, a building leak-rate test is to be performed prior to entering confinement applicability.

If the reactor is in operation, walk-down of the accessible areas to inspect for damaged confinement will be performed; perform a building leak-rate test prior to the next scheduled reactor startup.

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The assessments of maximum confinement pressures in TR-808, PG-T-88-020, and PG-T-88-021 are based on severe or beyond design basis accidents. The safety design basis for the ATR confinement should be that the confinement must withstand the pressures and temperatures of the limiting design basis accident without exceeding the design leakage rate defined in TRA-ATR-1543. This would make the ATR confinement design basis consistent with the design basis for commercial nuclear power reactors. Commercial reactor containment over-pressure verifications are based on limiting design basis accidents not severe accidents.

Current normal leak rate testing procedures of the ATR confinement include over-pressurization up to about 0.5 inches of water. These leak rate tests monitor confinement performance to assure the design basis leak rate is maintained. Based on engineering judgment it is unlikely that any design basis accident that may result reactor or pressurized water loop experiment fuel damage would result in a confinement over-pressurization that would damage the confinement such that the acceptance criteria in TRA-ART-1543 would not be met. Confinement over-pressurization during design basis loss of coolant, loss of heat sink and loss of flow accidents can be characterized as a volume displacement from emergency firewater injection. The primary coolant system bulk temperature remains below saturation during these accidents. Pressurized water loop depressurizations would result in expansion of the high-pressure, high-temperature loop coolant into the confinement volume. The total volume of a pressurized water loop is small (~250 gallons) compared to the confinement volume (1.64E+6 ft^3) and there would be substantial condensation of the vapor released from the loop piping. The loop piping is almost entirely within small basement compartments inside of confinement which would tend to minimize the global over-pressure effect inside of confinement volume.

J. C. Chapman	**(See PISA declaration)**	**10/24/07**
Safety Analyst Print/Type Name	Safety Analyst Signature	Date
M. B. McDonough	**(See PISA declaration)**	**10/24/07**
Nuclear Facility Manager Print/Type Name	Nuclear Facility Manager Signature	Date

Is independent technical review required? Yes ☒ No ☐

Nuclear Facility Manager Signature or Initials

A. P. Hoskins	**(See PISA declaration)**	**10/25/07**
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 Revision 21 Sections 3.1.2.7, 3.1.2.7.2, 3.1.4.9, 3.1.5.1.2, 3.1.5.3.2, 3.1.5.4.1, 3.8.1.3, Table 3.8-1, 6.2-1, 6.2.2.1, Table 6.2-1, 6.2.2.9, 6.2.3.2, 12.5.2.2.1, 15.7, 15.8.7, 15.12, 16.2.2.6
TSR-186 Revision 11 LCO 3.8.1
TRA-USQ-SE-1998, TRA-USQ-1998-050, TRA-ATR-1543 Rev 2, TRA-ATR-1588 Rev 1

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:

Issue 1

No. Issue 1 notes that GDC 16 requires, in part, that the confinement building is required to ensure that confinement design conditions important to safety are not exceeded for as long as postulated accident conditions require. It then describes a concern that the confinement leak rate may change after an over-

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pressurization event. It also notes that a previously recommended overpressure protection feature has not been installed.

The system in question, the confinement building, performs its function after an accident has already occurred. It cannot be the initiator of an accident. The concern described by Issue 1 cannot increase the probability of occurrence of an accident previously evaluated in the safety basis.

Issue 2

No. Issue 2 notes that the RMS-2 function, which is stated to provide some measure of confinement overpressure protection, was removed without adequate evaluation of a replacement system for this function.

The presence, or absence, of a confinement overpressure protection system affects only the mitigation of a possible accident. It cannot serve as the initiator of an accident. The concern described by Issue 2 cannot increase the probability of occurrence of an accident previously evaluated in the safety basis.

Issue 3

No. Issue 3 notes that SAR-153, and an older accident analysis, extrapolate confinement leak rate data to possibly inappropriate values. It also notes that the SAR discussion uses upper bound leak rate values rather than the more appropriate minimum leak rate data. Finally, Issue 3 indicates that some seal materials would be expected to fail at elevated confinement differential pressures. Such failure could invalidate the extrapolated data. Seal failures due to overpressure event would exacerbate Issue 1.

The pressure retention capability of the confinement, or the capability to relieve pressure, would only be important after an accident occurs. It cannot be the initiator of an accident. The concern described by Issue 3 cannot increase the probability of occurrence of an accident previously evaluated in the safety basis.

Issue 4

No. Issue 4 notes that GDC 38 requires the existence of a confinement heat removal system to control confinement pressure and temperature. The ATR confinement lacks such a system. Issue 4 also notes that failure of a pressurized water loop experiment facility would result in a potential heat source.

The presence or absence of a confinement heat removal source would be of importance only after an accident has occurred. Absence of such a system cannot be the initiator of an accident. The concern described by Issue 4 cannot increase the probability of occurrence of an accident previously evaluated in the safety basis.

Issue 5

No. Issue 5 notes that a confinement building underpressure event could occur if the reactor building main supply fan (HVS-1) fails while the exhaust fan keeps running. Such an underpressure event has not been previously evaluated.

Confinement pressure, either above or below outside ambient air pressure, is not an accident initiator at the ATR. The possibility of an underpressure event does not increase the probability of occurrence of any previously evaluated accident in the safety basis.

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

Issue 1

No. SAR-153 discusses several radiological accidents (Sections 15.7, 15.8.7, 15.12). All of these accidents assume near steady-state releases of radioactive material from the confinement. These releases are driven by emergency firewater injection system (EFIS) flow into the building via the primary coolant system (PCS),

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wind-induced building differential pressure, and building-to-outside pressure and temperature equalization. None of the accidents lead to a confinement building pressurization. Without confinement pressurization, this concern does not exist, thus it cannot increase the consequences of an accident previously evaluated in the safety basis.

As stated in the description for Issue 1, there are several older accident analyses that were performed, in part, to determine if the confinement design pressure could be attained or exceeded. These analyses involve beyond design basis accidents, which are not part of the safety basis for the ATR. While the consequences of these hypothetical accidents could be increased by the concern described in Issue 1, these results of these analyses are not required to be acceptable. The PISA, however, does not questioned the functionality of the confinement for design basis accidents.

Issue 2

No. General Design Criterion 16 requires, in part, that the confinement building is required to ensure that confinement design conditions important to safety are not exceeded for as long as postulated accident conditions require. A confinement overpressure protection system would be intended to keep confinement pressure below the level that would cause a change in the building leak rate. Such a system is not required at the ATR because no design basis events result in confinement pressurization.

As described for Issue 1 (above), none of the radiological analyses involve confinement pressurization. For these accidents, an overpressure protection system is not necessary. Nor is it necessary for beyond design basis accidents that might result in pressurization; the ATR confinement building is not required to, nor is it designed to, survive such events unscathed. Thus, the absence of a confinement overpressure protection system does not increase the consequences of an accident previously evaluated in the safety basis.

Issue 3

No. Confinement pressure capability is only a factor in beyond design basis accidents. No accident analyses in the safety basis credit the capability of the confinement to retain, or relieve, pressure. Inappropriate extrapolation of leak rate data to differential pressures greater than measured in leak rate tests has no effect on the consequences of these accidents because the analyses do not use the extrapolated data. Data extrapolated to 0.5 in. of water from TRA-ATR-1543, Rev. 2 indicate a confinement leakage of greater than 6000 cfm which is greater then that required for the postulated Large Break Loss of Coolant Accident with Emergency Firewater Injection, 5530 cfm.

In addition, SAR-153 Section 6.2.3.2 discusses the ability of the confinement to withstand the effects of a design basis LOCA. The simplistic calculation contained therein is extremely conservative. The peak mass flow from the break occurs for a very short period of time at the beginning of the accident and is not representative of the mass flow that must be relieved by the confinement building. Also, the peak mass flow and EFIS flow do not occur at the same time. By including these conservatisms, the calculation indicates the confinement must be able to relieve more volume displacement than necessary. .

Issue 4

No. General Design Criterion 38 requires commercial nuclear power plants to have a containment heat removal system because the containment building is designed to contain all the mass and energy released from primary or secondary system breaks inside the containment building. Without a heat removal system, the pressure and temperature inside the containment could remain elevated for substantial periods of time, which might adversely affect containment leakage or equipment qualification lifetimes.

At the ATR, the confinement is not designed to contain the mass and energy released from a primary coolant system or experiment loop break. The confinement is not designed to retain pressure. Additionally, the water in the primary coolant system is subcooled. For primary system breaks, insufficient energy is released to the confinement to require a heat removal system. For a pressurized water loop (PWL) experiment facility, water temperatures are significantly higher. The total quantity of water, however, is much less (about 250 gallons) and would condense quickly on cold confinement surfaces. Thus, after a

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PWL failure, the mass-energy release to the confinement is relatively small. The large volume of the gas-tight area of the confinement (1.64 E6 ft^3) would be expected to absorb the release with little increase in pressure or temperature, even if all of the water were to flash to steam. A confinement heat removal system is not needed. Thus, absence of such a system cannot increase the consequences of an accident previously evaluated.

Issue 5

Yes. An underpressure event could conceivably result in changes to confinement building seals. These changes could result in a leak rate larger than assumed in the accident analyses. This potentially higher leak rate would remain undetected since confinement leak rate testing is not required after an underpressure event. Increased leak rates could increase the consequences of the radiological accidents analyzed in the safety basis. Although the PISA does not address possible structural damage from an underpressure event, it is very unlike that events such as exhaust fans continuing to run would cause any structural damage of the confinement.

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:

Issue 1

No. There is no equipment that might malfunction more frequently if the confinement leak rate increased following a confinement pressurization event. Increased confinement leakage would not adversely affect any equipment inside or outside the confinement building. Additionally, since the confinement building has been determined to be able to withstand internal pressures of up to 9.0 inches of water column (in. w.c.), pressurization events up to the design pressure of 7.5 in. w.c. are not expected to result in structural failure of the building. Thus, "parts" of the building would not be expected to fall onto other equipment. In addition, the functionality of the confinement seal system is not expected to be affected by design basis accidents.

Issue 2

No. A confinement overpressure protection system would only be of value in those beyond design basis accidents that result in building pressurization. The evaluation of these accidents is not required to be part of the safety basis. Since the safety basis only evaluates design basis accidents, the probability of occurrence of malfunctions of equipment important to safety in those accidents cannot be increased by the absence of a system that would only function in beyond design basis accidents.

Issue 3

No. The issue of inappropriate extrapolation of confinement leak rate data to higher differential pressures applies only to accidents wherein the confinement building is pressurized. Although confinement seals are expected to be damaged by high overpressure accidents, such accidents are beyond design basis for the ATR.

Issue 4

No. Prior arguments (see Question 1) indicate that a confinement heat removal system is not necessary for the ATR. Absence of a system that is not needed cannot increase the probability of any equipment malfunction.

Issue 5

Yes. Issue 5 raises the possibility that confinement building seal materials could be damaged by an underpressure event. Such damage would be considered a malfunction. Examination of Drawing 120314 suggests that the confinement building penetration seals are robust structures that should be able to withstand the same amount of differential pressure in either a forward or reverse direction. However, an engineering analysis has not been done to support this supposition. In addition, increase confinement leakage would remain undetected after an underpressure accident, since testing after such an accident is not required.

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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:

Issue 1

No. Issue 1 is concerned with the confinement building leak rate following an overpressure event. Such events are beyond the design basis of the ATR. In addition, the functionality of the confinement seal system is not expected to be affected by design basis accidents. However, it is possible that, should the confinement leak rate increase, the radiological consequences to evacuating workers could be increased if the RTC evacuation system were to malfunction. Redundant evacuation systems (voice paging systems, security vehicles with loudspeakers) would, however, mitigate a loss of the evacuation system at RTC.

Issue 2

No. Issue 2 is concerned with the lack of a confinement overpressure protection system at the ATR. Such protection is not required for any analyzed radiological events in the safety basis. Absence of such a system cannot affect previously evaluated equipment malfunctions since such protection has not been credited in the accident analyses.

Issue 3

No. Issue 3 is concerned with inappropriate extrapolation of confinement leak rate data to higher differential pressures. Such extrapolation could lead to erroneous conclusions about the ability of the confinement building to withstand overpressure events. This is akin to not having an overpressure protection system. See Issue 2 above. Although confinement seals are expected to be damaged by high overpressure accidents, such accidents are beyond design basis for the ATR.

Issue 4

No. Prior arguments (see Question 2) indicate that a confinement heat removal system is not necessary for the ATR. Absence of a system that is not needed cannot increase the consequences of any equipment malfunction.

Issue 5

Yes. See Question 2.

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:

Issue 1

No. As described under Question 1, Issue 1 is not an accident initiator.

Issue 2

No. As described under Question 1, Issue 2 is not an accident initiator.

Issue 3

No. As described under Question 1, Issue 3 is not an accident initiator.

Issue 4

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No. As described under Question 1, Issue 4 is not an accident initiator.

Issue 5

Yes. A confinement underpressure event would not be expected to initiate an accident of any type. However, confinement underspressure, since the condition could go undetected, could exacerbate an accident scenario.

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:

Issue 1

No. The safety basis already evaluates the ability of the confinement building to withstand a pressurization event. Additionally, none of the radiological accidents evaluated in the safety basis involve confinement pressurization. Issue 1 (change in leak rate after pressurization / lack of overpressure protection) cannot create the possibility of a different type of malfunction of equipment important to safety (i.e. the confinement building). The functionality of the confinement seal system is not expected to be affected by design basis accidents.

Issue 2

No. While the lack of a confinement overpressure protection system could conceivably cause unforeseen malfunctions of equipment, such malfunctions would only occur in beyond design basis events.

Issue 3

No. Inappropriate extrapolation of confinement leak rate data does not create the possibility of a different type of malfunction of equipment important to safety. Nothing physical is changed. Although, confinement seals are expected to be damaged by high overpressure accidents, such accidents are beyond design basis for the ATR.

Issue 4

No. No credit has been taken for confinement heat removal. Absence of confinement heat removal system cannot create the possibility of a different type of equipment malfunction than previously evaluated. Nothing physical is changed.

Issue 5

Yes. The only equipment malfunctions that could be affected by a postulated confinement underpressure event would involve increased seal leakage. Undetected seal leakage could exacerbate an accident scenario. However, lose of confinement, confinement leakage greater than assumed, has not been considered in accident scenarios.

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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:

Issue 1

No. The potential change in confinement leak rate following a pressurization event does not reduce the margins of safety associated with the ATR. The functionality of the confinement seal system is not expected to be affected by design basis accidents. Data extrapolated to 0.5 in. of water from TRA-ATR-1543, Rev. 2 indicate a confinement leakage of greater than 6000 cfm which is greater then that required for the postulated Large Break Loss of Coolant Accident with Emergency Firewater Injection, 5530 cfm. Thus the confinement will be capability of relieving the postulated over-pressurization. None of the radiological accidents analyzed in the safety basis involve significant confinement pressurization. Any events that might result in significant confinement pressurization are beyond design basis events.

Issue 2

No. While SAR-153 Section 12.5.2.2.1 indicates that part of the confinement ventilation system "serves to relieve any possible pressure buildup in the confinement area that might accompany a radioactive release," none of the analyses in the safety basis credit the existence or performance of a confinement overpressure protection system. Because overpressure protection is not credited, its absence does not affect any margins of safety in the safety basis.

Issue 3

No. Issue 3 deals with certain calculations related to confinement pressure response performance. The margins of safety defined for the confinement in TSR-186 relate to the leak rate under postulated design basis events. Although, confinement seals are expected to be damaged by high overpressure accidents, such accidents are beyond design basis for the ATR. The postulated accidents, even the scenario postulated in TRA-ATR-1543 Rev. 2, do not predict overpressure near extrapolated differential pressures described in Issue 3.

Issue 4

No. Issue 4 is concerned with the absence of a confinement heat removal system and the possible heat source provided by a pressurized water loop experiment facility failure. No credit, or conservatism, has been taken with respect to confinement heat removal. For this reason, the presence, or absence, of such a system does not affect any margin of safety.

Issue 5

Yes. A postulated confinement underpressure event could, in theory, affect confinement building seal leakage. Increases in seal leakage without detection could result in increased radiological consequences and decreased dose margins.

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 constitutes an Unreviewed Safety Question on the basis that a postulated confinement underpressure event could potentially lead to seal damage. Such undetected seal damage could increase the confinement leakrate over that measured from the most recent leak rate test. Increased leakage could result in increased radiological consequences following certain accident scenarios. The existing controls, as presented above, would not detect increased leakage until after the reactor is shutdown and the confinement building tested. It is in this interim period between the underpressure event and reactor shutdown that the potential for increased radiological consequences exists.

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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:

Steven D. Winter (trainee) / J. C. Chapman USQ Evaluator Print/Type Name	J C Chapman USQ Evaluator Signature	7 Nov 2007 Date
M. B. McDonough Nuclear Facility Manager Print/Type Name	M. B. McDonough Nuclear Facility Manager Signature	7 NOV 07 Date

CONCURRENCE:

A. P. Hoskins Independent Review Committee Chair Print/Type Name	Alan P. Hoskins Independent Review Committee Chair Signature	11-7-07 Date