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

Facility or Activity: Advanced Test Reactor

USQ Process No.: RTC-USQ-2005-685

Subject: Frequency of Primary Coolant System Butterfly Valve (BF-A-1-14) Closure Accidents

Describe the New Information/Discovery:

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

I. PISA ASSESSMENT

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

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

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

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

e. Is this an activity or condition not analyzed in the existing safety basis?
☐ Yes ☐ No

Provide an explanation of the assessment result:

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

If the answer to ALL of the assessment questions above is No, file the completed form. Do not complete Section II, III, or IV.
If the answer to any one of the assessment questions is Yes, there is a potentially inadequate safety analysis (PISA):

- Nuclear Facility Manager document actions (Section **II**), including interim operating restrictions, taken to place or maintain the facility in a safe condition.
- Nuclear Facility Manager execute reporting process per MCP-190 or AWP-3.1, for PISA.
- USQ evaluator proceed to Section **III**.

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

Facility or Activity: Advanced Test Reactor

USQ Process No.: RTC-USQ-2005-685

Subject: Frequency of Primary Coolant System Butterfly Valve (BF-A-1-14) Closure Accidents

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

Safety Analyst Print/Type Name	Safety Analyst Signature	Date
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. DETERMINATION

Identify applicable section(s) of the safety basis document(s) (e.g., DSA, SAR, BIO, TSRs, etc.).
SAR-153, Chapter 15.3.4.
SAR-153, Chapter 15.13.2

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:

SAR-153, Chapter 15.3.4 assumed that a mechanical failure of the primary coolant system (PCS) butterfly valve (BF-A-1-14) leading to rapid closure is a Condition 4 (extremely unlikely) fault. SAR-39 (ATR Design Basis Report) and EDF TRA-ATR-779, Revision 2 concluded that closure of this valve to the stop due to operator error, valve controller error, or valve mechanical failure is a Condition 3 (unlikely) fault. Also, Section I of this PISA stated that the supporting documentation did not address other possible failure mechanisms of the valve, such as fatigue. Review of NRC data bases and the operation history of the ATR secondary coolant system butterfly valves indicates that the failure of the valve is a least a Condition 3 fault. Based on the contradictory information found in various SAR-153 supporting references, this PISA could increase the probability of an accident previously evaluated in the safety basis (from a Condition 4 to a Condition 3 fault).

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

Facility or Activity: Advanced Test Reactor

USQ Process No.: RTC-USQ-2005-685

Subject: Frequency of Primary Coolant System Butterfly Valve (BF-A-1-14) Closure Accidents

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

Explain:
The thermal margins resulting from rapid closure of this butterfly valve are presented in Chapter 15.3.4, SAR-153. The thermal margins still meet Condition 2 acceptance criteria. The primary coolant system (PCS) pressure response resulting from rapid closure of this butterfly valve is presented in Chapter 15.13.2, SAR-153. The pressure response is shown to meet Condition 3 acceptance criteria. The consequences are not affected by its classification as either a Condition 3 or a Condition 4 fault and therefore, the new information does not increase the consequences of an accident previously evaluated in the safety basis. (As discussed above in Question 1, review of NRC data bases and operation history of the ATR secondary coolant system butterfly valves indicate that the failure is a Condition 3 fault.)

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

Explain:

The probability of closure of the flow control valve to the stop due to operator error, valve controller error, or valve mechanical failure is reported as a Condition 3 fault in SAR-39 and EDF TRA-ATR-779, Revision 2 and as a Condition 4 fault in SAR-153, Chapter 15.3.4. Review of NRC data bases and a review of the operation history of the ATR secondary coolant system butterfly valves indicates that valve failure of this butterfly valve is a Condition 3 event. Without further analysis to reconcile the probability of failure of the valve, the new information could increase 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:

The thermal margins resulting from rapid closure of the valve is presented in SAR-153, Chapter 15.3.4 and the effect of the high system pressure resulting from rapid closure of the valve is presented in SAR-153, Chapter 15.13.2. The new information did not raise issues associated with the calculation of consequences due to valve malfunction. The valve failure has still been shown to meet Condition 2 acceptance criteria for thermal margins (SAR-153 Chapter 15.3.4) and Condition 3 acceptance criteria margins for the pressure increase (SAR-153 Chapter 15.13.2). Therefore, the consequences are not affected by its classification as either a Condition 3 or a Condition 4 fault. Thus, this PISA does not increase the consequences of a malfunction of equipment important to safety previously evaluated in the safety basis. (As discussed above in Question 1, review of NRC data bases and operation history of the secondary coolant system butterfly valves indicates that the failure is a Condition 3 fault.)

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 accident nor the accident progression has not changed. If the valve shaft were to fail due to fatigue as discussed in Section I of the PISA, the result would be rapid closure of the valve, which has been analyzed in SAR-153, Sections 15.3.4 and Section 15.13.2. Therefore, the new information presented in Section I of this PISA did not create the possibility of an accident of a different type than previously evaluated in the safety basis.

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

Facility or Activity: Advanced Test Reactor

USQ Process No.: RTC-USQ-2005-685

Subject: Frequency of Primary Coolant System Butterfly Valve (BF-A-1-14) Closure Accidents

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 noted that the supporting documentation did not discuss fatigue failures as a possible failure mechanism for the valve. While the reason for valve failure is not discussed, the consequences of the failure from whatever the cause are bounded by analysis resulting from of a rapid closure case. The NRC data bases reviewed, only established the failure rate from all causes and did not differentiate between types of valve failures. Since all failures are grouped together, the PISA did not create the possibility of a malfunction of equipment important to safety of a different type than previously evaluated in the safety basis.

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 results of this event as presented in SAR-153 have not changed. However based on the References presented in SAR-153 the event could be a Condition 3 or a Condition 4 event. If it were concluded by further analysis to be a Condition 3 event, the new information presented in Section I of this PISA could reduce the 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:

Based on the "Yes" responses to questions 1, 3 and 7. This PISA does constitute an Unreviewed Safety Question.

NOTE: If USQ determination result is positive, additional notification for ORPS is required per MCP-190 or AWP-3.1.

IV. APPROVAL/CONCURRENCE SIGNATURES

APPROVAL:

Steven Wagoner USQ Evaluator Print/Type Name	Steven Wagoner USQ Evaluator Signature	2/20/06 Date
MARTIN McDONOUGH Nuclear Facility Manager Print/Type Name	Martin McDonough Nuclear Facility Manager Signature	2/22/06 Date

CONCURRENCE:

Alan P Hoskins Independent Review Committee Chair Print/Type Name	Alan P Hoskins Independent Review Committee Chair Signature	2-23-06 Date

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

Facility or Activity: Advanced Test Reactor

USQ Process No.: RTC-USQ-2005-685

Subject: Frequency of Primary Coolant System Butterfly Valve (BF-A-1-14) Closure Accidents

Describe the New Information/Discovery:

The Advanced Test Reactor (ATR) Primary Coolant System (PCS) flow control valve FCV or butterfly valve (BF-A-1-14) is designed to provide fine adjustment of the primary coolant flow. The FCV is a rotating disk inside the inlet piping. The valve is designed with an external stop (on the valve body rather than internal to the valve) to limit the amount of closure. Since there is not an internal stop, certain failures downstream of the external stop, would allow the valve to rotate freely. The valve disk has 6 holes in it and is smaller than the valve throat; therefore, a complete stoppage of flow cannot occur should the flow stop fail and the valve close completely. Operating data indicate that the flow induced forces on the disk over the normal operating range are in the direction to close the valve. The impact of flow reduction on the core heat removal in the event of FCV closure or failure is addressed in SAR-153 Section 15.3.4, Complete Closure of the Flow Control Butterfly Valve. The effect of the closure or failure on the pressure transient within the piping is considered in SAR-153 Section 15.13.2, The Flow Control Valve (Butterfly) Fails.

Closure of the flow control valve to the stop due to operator error, valve controller failure, or valve mechanical failure was considered an unlikely fault in the Advanced Test Reactor (ATR) Design Basis Report (DBR) (SAR-39). The DBR was the predessor to the accident analysis in SAR-153. Rapid and complete closure of the flow control valve due to valve shaft failure (which bypasses the stop) or closure of the flow control valve with coincident failure of the stop was also considered an unlikely fault in the DBR. The unlikely fault category in the ATR DBR corresponds to the Condition 3 fault category in SAR-153.

A quantitative analysis of the potential for an operator error resulting in closing the valve to the stop was performed to support the process control system upgrade project (TRA-ATR-779 Revision 2 and TRA-ATR-786). These analyses conclude the frequency of an operator error resulting in inadvertent closure of the flow valve to the stop was 7E-03/year which is consistent with an unlikely or Condition 3 category. The event is considered as a Condition 4 event in SAR-153. Condition 4 events have a lower frequency of occurrence than Condition 3.

SAR-153 Section 15.3.4 does not refer to the analyses in TRA-ATR-779 and TRA-ATR-786 that are the design basis of the existing ATR process control system. TRA-ATR-786 is a reference to Section 15.5, 15.5, Increase in Primary Coolant Inventory. SAR-153 does not justify lowering the frequency for inadvertent closure of the valve due to operator error from Condition 3 determined in TRA-ATR-786 to Condition 4.

SAR-153 does refer to an analysis (TRA-ATR-839) that supports classifying mechanical failure of the flow control valve as Condition 4. The analysis in TRA-ATR-839 determines stresses in various components resulting from the expected maximum differential pressure to estimate the likelihood of valve failure. The analysis does not address fatigue failure which would be a credible failure mode for the valve. Also the analysis did not look at the industry experience of failures of similar components in evaluating the failure frequency. The typical approach in assessing component failure frequency is to use an industry experience failure database.

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

SAR-153 Revision 16 Effective on 07-APR-05, Section 15.3.4, Complete Closure of the Flow Control Butterfly Valve

SAR-153 Revision 16 Effective on 07-APR-05, Section 15.13.2, The Flow Control Valve (Butterfly) Fails

Brower, J. O., BF-1-A-14 ATR PCS Flow Control Valve Failure Analysis, TRA-ATR-839, October 26, 1993

Durney, J. L., Evaluation of Pressure Transients Associated With Failure of The Flow Control Valve, TRA-ATR-898, February 3, 1994

Durney, J. L., Safety Analysis Review for Process Control System Upgrade, TRA-ATR-779, Revision 2, August 1993

Galyean, W. J., Probability of Inadvertent Closure of Butterfly Valve BF-A-14, TRA-ATR-786, April 18, 1993

Pafford, D. J., RELAP5 Analysis of the Butterfly Valve Closure Transient for the Advanced Test Reactor Updated Final Safety Analysis Report, TRA-ATR-847, June 29, 1994

Pafford, D. J., RELAP5 Analysis of the Butterfly Valve Closure Transient During two-Pump Operation for the Advanced Test

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

Facility or Activity: Advanced Test Reactor

USQ Process No.: RTC-USQ-2005-685

Subject: Frequency of Primary Coolant System Butterfly Valve (BF-A-1-14) Closure Accidents

Reactor Updated Final Safety Analysis Report, TRA ATR 920, June 23, 1994

I. PISA ASSESSMENT

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

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

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

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

e. Is this an activity or condition not analyzed in the existing safety basis?
☐ Yes ☒ No

Provide an explanation of the assessment result:

The determinations of the frequency of occurrence of inadvertent closure and failure of the ATR PCS FCV are potentially inadequate. The frequency of inadvertent closure in SAR-153 is not consistent with the process control system design basis documentation. The determination of frequency of valve failure is not consistent with probabilistic methods used elsewhere in SAR-153 and does not consider all credible failure modes.

R. T. McCracken USQ Evaluator Print/Type Name	USQ Evaluator Signature	11-17-05 Date
M. B. McDonough Nuclear Facility Manager Print/Type Name	M.B. McDonough Nuclear Facility Manager Signature	11-17-05 Date

If the answer to ALL of the assessment questions above is No, file the completed form. Do not complete Section II, III, or IV.
If the answer to any one of the assessment questions is Yes, there is a potentially inadequate safety analysis (PISA):

- Nuclear Facility Manager document actions (Section II), including interim operating restrictions, taken to place or maintain the facility in a safe condition.
- Nuclear Facility Manager execute reporting process per MCP-190 or AWP-3.1, for PISA.
- USQ evaluator proceed to Section III.

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

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

Facility or Activity: Advanced Test Reactor

USQ Process No.: RTC-USQ-2005-685

Subject: Frequency of Primary Coolant System Butterfly Valve (BF-A-1-14) Closure Accidents

In SAR-153 Section 15.13, PCS High Pressure Anomalies, the acceptance criteria for events that result in an increase in pressure is that the pressure not exceed 110% of design pressure for high-probability events (Condition 2) and to not exceed 120% of design pressure for low-probability events (Condition 3 and 4). The analysis in TRA-ATR-898 determines that the maximum over-pressure for failure of the butterfly valve is 114% of design which is within the acceptance criterion for Condition 3 events. This maximum over-pressure occurs as the valve passes through zero degrees and opens. The pressure transient occurs downstream of the valve. The over-pressure upstream of the valve is less than 105% of design.

SAR-153 Section 15.3, Decrease in Reactor Primary Coolant Flow Rate, refers to analyses in TRA-ATR-847 and TRA ATR 920 which show butterfly valve closure the results in minimum margins that are greater than 3σ to critical heat flux which is the acceptance criterion for a Condition 2 fault. Since the Condition 2 acceptance criterion is met, the consequences meet the less restrictive Condition 3 acceptance criterion.

Since analyses in SAR-153 Sections 15.3.4 and 15.13.2 show the consequences of FCV inadvertent closure or failure meet the acceptance criteria for a Condition 3 event, no actions or interim controls are necessary to maintain the plant in a safe condition.

R. T. McCracken Safety Analyst Print/Type Name	Safety Analyst Signature	11-17-05 Date
M. B. McDonough Nuclear Facility Manager Print/Type Name	M. B. McDonough Nuclear Facility Manager Signature	11-17-05 Date

Is independent technical review required? Yes ☐ No ☐

Nuclear Facility Manager
Signature or Initials

Independent Reviewer Print/Type Name	Independent Reviewer Signature	Date

III. DETERMINATION

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

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:

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

Facility or Activity: Advanced Test Reactor

USQ Process No.: RTC-USQ-2005-685

Subject: Frequency of Primary Coolant System Butterfly Valve (BF-A-1-14) Closure Accidents

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

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:

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

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:

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:

IIIc: POTENTIAL FOR REDUCTION IN A MARGIN OF SAFETY

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

Facility or Activity: Advanced Test Reactor

USQ Process No.: RTC-USQ-2005-685

Subject: Frequency of Primary Coolant System Butterfly Valve (BF-A-1-14) Closure Accidents

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

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:

NOTE: If USQ determination result is positive, additional notification for ORPS is required per MCP-190 or AWP-3.1.

IV. APPROVAL/CONCURRENCE SIGNATURES

APPROVAL:

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

CONCURRENCE:

Independent Review Committee Chair Print/Type Name	Independent Review Committee Chair Signature	Date

has six holes in it and is smaller than the valve throat; therefore, a complete stoppage of flow cannot occur should the flow stop fail and the valve close completely. Operating data indicate that the flow induced forces on the disk over the normal operating range are in the direction to close the valve. The impact of flow reduction on the core heat removal in the event of FCV closure or failure is addressed in SAR-153, Section 15.3.4, Complete Closure of the Flow Control Butterfly Valve. The effect of the closure or failure on the pressure transient within the piping is considered in SAR-153, Section 15.13.2, The Flow Control Valve (Butterfly) Fails.

Closure of the flow control valve to the stop due to operator error, valve controller failure, or valve mechanical failure was considered an unlikely fault in the ATR Design Basis Report (DBR) (SAR-39). The DBR was the predecessor to the accident analysis in SAR-153. Rapid and complete closure of the flow control valve due to valve shaft failure (which bypasses the stop) or closure of the flow control valve with coincident failure of the stop was also considered an unlikely fault in the DBR. The unlikely fault category in the ATR DBR corresponds to the Condition 3 fault category in SAR-153.

A quantitative analysis of the potential for an operator error resulting in closing the valve to the stop was performed to support the process control system upgrade project (TRA-ATR-779, Revision 2 and TRA-ATR-786). These analyses conclude the frequency of an operator error resulting in inadvertent closure of the flow valve to the stop was 7E-03/year which is consistent with an unlikely or Condition 3 category. The event is considered as a Condition 4 event in SAR-153. Condition 4 events have a lower frequency of occurrence than Condition 3.

SAR-153, Section 15.3.4 does not refer to the analyses in TRA-ATR-779 and TRA-ATR-786 that are the design basis of the existing ATR process control system. TRA-ATR-786 is a reference to Section 15.5, Increase in Primary Coolant Inventory. SAR-153 does not justify lowering the frequency for inadvertent closure of the valve due to operator error from Condition 3 determined in TRA-ATR-786 to Condition 4.

SAR-153 does refer to an analysis (TRA-ATR-839) that supports classifying mechanical failure of the flow control valve as Condition 4. The analysis in TRA-ATR-839 determines stresses in various components resulting from the expected maximum differential pressure to estimate the likelihood of valve failure. The analysis does not address fatigue failure which would be a credible failure mode for the valve. ~~Also, the analysis did not look at the industry experience of failures of similar components in evaluating the failure frequency~~. The typical approach in assessing component failure frequency is to use an industry experience failure database.

However, (Insert)

---

16. Is Subcontractors Involved? No

---

17. Operating Conditions of Facility at Time of Occurrence:

The Advanced Test Reactor was at nominal full power for the Cycle 132C-1.

The ATR SINDA-SAMPLE code models the variation in flow rate in the hot fuel plate analysis. The model development did not explicitly address some pertinent sources of uncertainty and therefore may not be conservative. Acceptance criteria in safety basis surveillance procedures and in-service testing of the emergency pump have been based on the values derived in the safety basis.

Occurrence Report No. 14
USQ No.: RTC-USQ-2005-248
Discovered: May 4, 2005, 1630
Categorized: May 4, 2005, 1630

The derivations of the analytical limit setpoint and response time requirements for the ATR radiation monitoring and seal system (RMSS) are not consistent with the system functional performance assumed in the radiological consequence analyses. The derivations are based on limits that are not consistent with the ATR plant protection criteria specified in the safety basis.

The derivation of the analytical limit setpoint and response time are not consistent with the methods used in the radiological consequence analyses presented in SAR-153, Section 15.7 and 15.12. The methodology used for the derivation of the setpoint could allow higher off-site doses than predicted by the radiological consequence analyses. Since these radiological consequence analyses are the basis upon which DOE approved operation of the ATR, the descrepancy represents a potentially inadequate safety analysis.

The current in-plant setting ensure RMSS actuation and confinement isolation prior to the stack discharge rate exceeding the limits for normal operation. The in-plant setting is consistent with the system functional performance assumed in the radiological consequence analyses for accidental releases. The facility is in a safe condition provided the RMSS setting is not raised above the current procedural value.

The current in-plant setting of less than or equal to 45 mR/hr will be maintained as the TSR limit until the next annual SAR update is issued and the applicable TSR and TSR bases are revised. The in-plant setting will be controlled using existing procedure change control and USQ processes.

Occurrence Report No. 15
USQ No.: RTC-USQ-2005-685
Discovered: February 9, 2006, 0936
Categorized: February 9, 2006, 0936

The Advanced Test Reactor (ATR) Primary Coolant System (PCS) flow control valve (FCV) or butterfly valve (BF-A-1-14) is designed to provide fine adjustment of the primary coolant flow. The FCV is a rotating disk inside the inlet piping. The valve is designed with an external stop (on the valve body rather than internal to the valve) to limit the amount of closure. Since there is not an internal stop, certain failures downstream of the external stop would allow the valve to rotate freely. The valve disk

Insert

~~However~~, review of NRC data bases and the operation history of the ATR secondary coolant system butterfly valves indicates that the failure of the BF-A-1-14 valve is a least a Condition 3 fault.