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-2006-655

Subject: Inconsistencies in the Maximum Hypothetical Accident Analysis

Describe the New Information/Discovery:

The ATR Design Basis Reconstitution Program (DBRP) review of the radiological consequence analysis in Section 15.12, Severe Accident Analyses, of SAR-153 resulted in numerous observations. The observations can be lumped into three general categories.

APPLICATION OF ANALYTICAL BASES

There are issues with the way in which the analytical bases were applied in the determination of the fission product source term available for release from the ATR confinement building.

- Section 15.12 states that the radiological consequences are based on 100% core melt; however, the analytical basis (Technical Report PG-T-92-112) is a large outlet break loss-of-coolant accident (LOCA) calculation that results in melting of 30% of the core fuel in the high power quadrants. Since these are the high power quadrants this 30% of the core fuel contains 45% of the total core inventory. The results from the analytical basis were not adjusted upward for the stated SAR-153 assumption of 100% core melt.

- The analytical basis predicts about 30 grams of iodine are released from the fuel that melts. The total core iodine inventory is 131 grams. The analysis then predicts that 3.1 grams of iodine are released to the ATR confinement atmosphere. This represents 10.5% of the iodine released from the melted fuel. SAR-153 argues that the basis calculations is conservative and reduces the 10.5% factor to 5.3%. Only brief qualitative arguments are provided for the factor of 2 reduction in the release to the confinement building. These qualitative arguments do not address all of the significant parameters that affected the release in the analytical basis calculation.

- In the SAR-153 radiological consequence analysis, the 5.3% factor is applied to the total core iodine inventory not the iodine released from the melted fuel. This results in 7 grams of iodine available for release from the confinement building. This misinterpretation of the analytical basis results in off-setting the above issue where no upward adjustment was made for the assumed increase in fuel melt. For a 100% core melt assumption about 67 grams of iodine would be released from the fuel (using the data from the analytical basis calculation). 7 grams would be available for release from confinement for the 10.5% factor and 3.5 grams would be available for the 5.3% factor.

- In the analytical basis calculation the pipe corridor (primary coolant piping space below the reactor vessel) is characterized as a relatively closed volume within the confinement. This small volume is used to determine the partition coefficient of iodine from the water into the confinement atmosphere. This makes the analysis very dependent on break location and confinement isolation. The confinement volume containing the large primary coolant system piping (pipe corridor, pipe tunnel, and heat exchanger area) has a significantly larger volume than the pipe corridor.

PRESENTATION OF THE ANALYSIS

Section 15.12 of SAR-153 presents the analysis of radiological consequences as bounding and conservative. The results are portrayed as "maximum hypothetical accident" consequences. The analyses of the fission product inventory and fission product source term available for release from the ATR confinement building are based on numerous nominal and best estimate assumptions. Use of nominal and best estimate analyses in a facility safety basis is typically accompanied by a determination and accounting for uncertainties in the analyses. There is no determination and accounting for uncertainty in the Section 15.12 analyses.

Facility or Activity: Advanced Test Reactor

USQ Process No.: RTC-USQ-2006-655

Subject: Inconsistencies in the Maximum Hypothetical Accident Analysis

A rigorous uncertainty analysis was not possible for this work because the computer codes used in the calculations (ORIGEN2 , CORSOR , IRATE, EQUILIBRIUM, and TRAPMELT/PULSE) have not yet been subjected to a formal validation and verification (V&V), especially under the unique conditions represented by the ATR . For this reason, a best-estimate analysis was performed and then sensitivity analyses were performed to determine whether the best-estimate calculation was conservative.

REGULATORY OBSERVATIONS

Section 15.12 presents the analysis as consistent with Regulatory Guide 1.4, Assumptions Used for Evaluating the Potential Radiological Consequences of a Loss of Coolant Accident for Pressurized Water Reactors, and states that where a deviation was taken the deviation is discussed. There are deviations from Regulatory Guide 1.4 that are not discussed. There are also deviations discussed that are not clearly identified as deviations.

10 CFR 830 specifies Regulatory Guide 1.70, Standard Format and Content of Safety Analysis Reports for Nuclear Power Plants, as the safe harbor standard for the documented safety analysis for DOE reactors. With regard to radiological consequences, Regulatory Guide 1.70 states, "when calculating the radiological consequences of a loss-of-coolant accident (LOCA), it is suggested that the assumptions given in ... Regulatory Guide 1.4, "Assumptions Used for Evaluating the Potential Radiological Consequences of a Loss-of-Coolant Accident for Pressurized Water Reactors," be used. This analysis should be referred to as the "design basis analysis." There may be instances in which the applicant will not agree with the conservative margins inherent in the design basis approach approved by the NRC staff or the applicant may desire to provide a "realistic analysis" for comparison purposes. If this is the case, the applicant may provide an indication of the assumptions he believes to be adequately conservative, but the known NRC assumptions should nevertheless be used in the design basis analysis. Any "realistic analysis" provided will help quantify the margins that are inherent in the design basis approach."

Section 15.12 presents a "realistic analysis" for the fission product inventory and fission source term available for release from the confinement building.

Identify the applicable safety basis document(s) (e. g., DSA, SAR, BIO, TSRs, etc.):
SAR-153, Section 15.12

I. PISA ASSESSMENT

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-2006-655

Subject: Inconsistencies in the Maximum Hypothetical Accident Analysis

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:
Section 15.12 analysis (i.e., the "maximum hypothetical accident" consequence analysis) is used to bound the consequences of the design basis accidents. Section 15.12 analysis would have bounded the consequences of the ATR design basis accidents. However, questions, inaccuracies, and inconsistencies in the discussion of the analysis and assumptions used in the analysis bring into question the appropriateness of using Section 15.12 analysis.

J. C. Chapman USQ Evaluator Print/Type Name	J C Chapman USQ Evaluator Signature	27 Sept 2006 Date
M. B. McDonough Nuclear Facility Manager Print/Type Name	M. B. McDonough Nuclear Facility Manager Signature	27SEP2006 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

The worst case design basis accident for offsite consequence is the Condition 4 seismic loss-of-coolant accident (LOCA). The worst case accident calculated margins from critical heat flux (CHF) and flow instability (FI), the ATR Plant Protection Criteria, are 3.40 standard deviations and 1.64 standard deviations (EDF-5668). These margins meet the Condition 4 plant protection criteria.

Primary analysis indicates that reducing the maximum effective plate power (EPP) for the ATR fuel elements by 13% will increase the margins from CHF and FI to 3.49 standard deviations and 3.34 standard deviations. Preliminary analysis results are attached. The Condition 2 ATR Plant Protection Criteria are 3 standard deviations from CHF or FI. Operations of the ATR with the 13% reduced EPP will result in the worst case Condition 4 accident meeting the Condition 2 plant protection criteria, and consequently, will eliminate any credible possibility of fuel damage from the worst case design basis LOCA. Other postulated design basis ATR fuel damage events occur with an intact primary

Facility or Activity: Advanced Test Reactor

USQ Process No.: RTC-USQ-2006-655

Subject: Inconsistencies in the Maximum Hypothetical Accident Analysis

coolant system (PCS). Any fission product release will be contained in the PCS and detected by the fission break and stack monitors resulting in reactor scram and confinement isolation.

Reactor shall be operated with the following limits on ATR fuel element effective plate powers (EPPs):
For greater than or equal to 2 standard deviations to reflector ligament craking,
3 primary coolant pump (PCP) operations plate 19 EPP less than or equal to 387 MW;
3 PCP operations inner plates EPP less than or equal to 385 MW;
2 PCP operations plate 19 EPP less than or equal to 362 MW;
2 PCP operations inner plates EPP less than or equal to 362 MW; and
for less than to 2 standard deviations to reflector ligament craking,
3 PCP operations plate 19 EEP less than or equal to 310 MW;
3 PCP operations inner plates EPP less than or equal to 310 MW;
2 PCP operations plate 19 EPP less than or equal to 311 MW;
2 PCP operations inner plates EPP less than or equal to 354 MW.

J. C. Chapman		
Safety Analyst Print/Type Name	Safety Analyst Signature	Date
M. B. McDonough		
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.).

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-2006-655

Subject: Inconsistencies in the Maximum Hypothetical Accident Analysis

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-2006-655

Subject: Inconsistencies in the Maximum Hypothetical Accident Analysis

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.

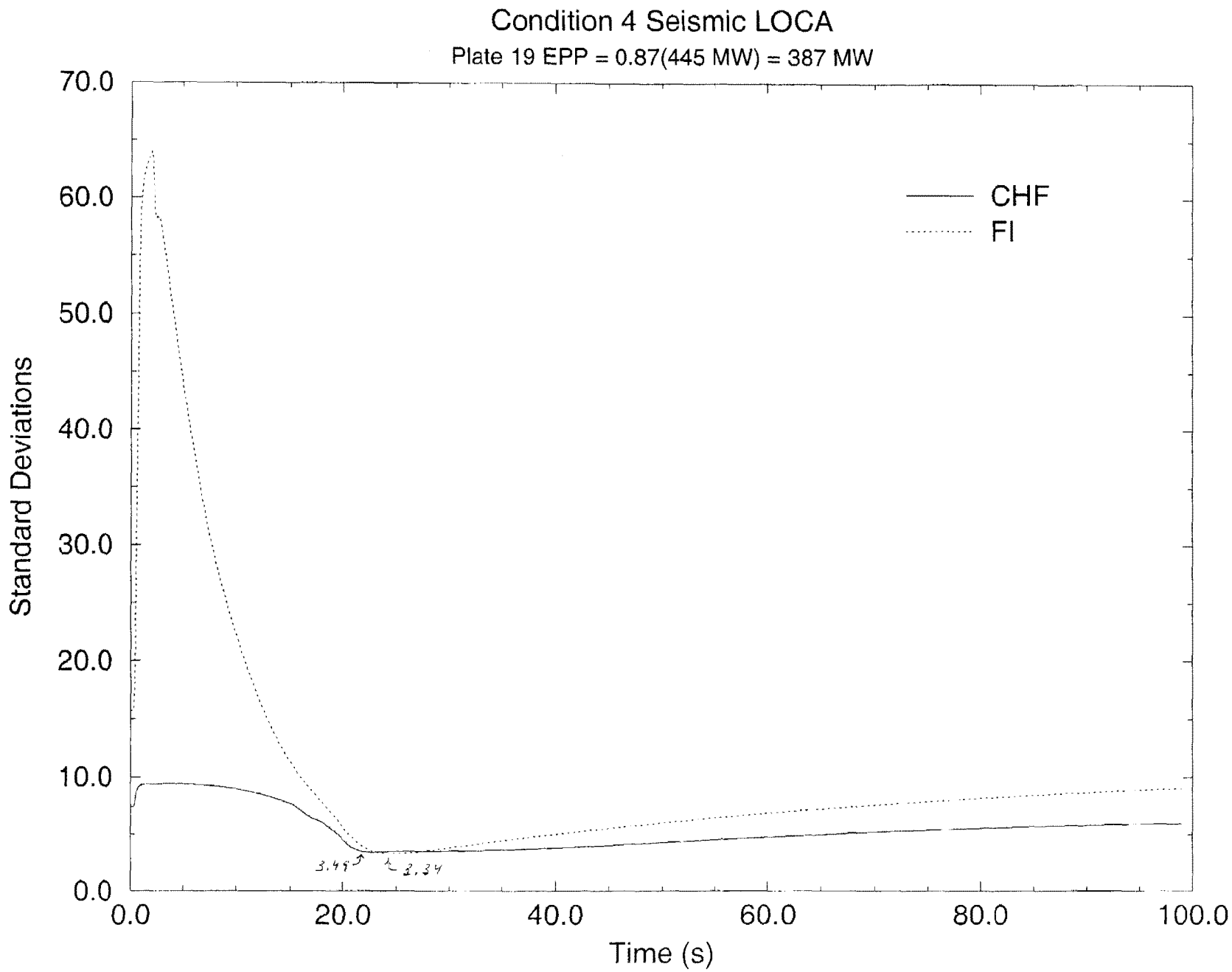
IV. APPROVAL/CONCURRENCE SIGNATURES

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


Condition 4 Seismic LOCA
Plate 19 EPP = 0.87(445 MW) = 387 MW
Standard Deviations
Time (s)
CHF
FI
70.0
60.0
50.0
40.0
30.0
20.0
10.0
0.0
0.0
20.0
40.0
60.0
80.0
100.0
3.49
2.34

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POTENTIAL INADEQUACY IN THE SAFETY ANALYSIS (PISA) FORM
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USQ Process No.: RTC-USQ-2006-655

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- The analytical basis predicts about 30 grams of iodine are released from the fuel that melts. The total core iodine inventory is 131 grams. The analysis then predicts that 3.1 grams of iodine are released to the ATR confinement atmosphere. This represents 10.5% of the iodine released from the melted fuel. SAR-153 argues that the basis calculations is conservative and reduces the 10.5% factor to 5.3%. Only brief qualitative arguments are provided for the factor of 2 reduction in the release to the confinement building. These qualitative arguments do not address all of the significant parameters that affected the release in the analytical basis calculation.

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Section 15.12 of SAR-153 presents the analysis of radiological consequences as bounding and conservative. The results are portrayed as "maximum hypothetical accident" consequences. The analyses of the fission product inventory and fission product source term available for release from the ATR confinement building are based on numerous nominal and best estimate assumptions. Use of nominal and best estimate analyses in a facility safety basis is typically accompanied by a determination and accounting for uncertainties in the analyses. There is no determination and accounting for uncertainty in the Section 15.12 analyses.

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REGULATORY OBSERVATIONS

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Section 15.12 presents a "realistic analysis" for the fission product inventory and fission source term available for release from the confinement building.

Identify the applicable safety basis document(s) (e. g., DSA, SAR, BIO, TSRs, etc.):
SAR-153, Section 15.12

I. PISA ASSESSMENT

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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-2006-655

Subject: Inconsistencies in the Maximum Hypothetical Accident Analysis

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:
Section 15.12 analysis (i.e., the "maximum hypothetical accident" consequence analysis) is used to bound the consequences of the design basis accidents. Section 15.12 analysis would have bounded the consequences of the ATR design basis accidents. However, questions, inaccuracies, and inconsistencies in the discussion of the analysis and assumptions used in the analysis bring into question the appropriateness of using Section 15.12 analysis.

J. C. Chapman		
USQ Evaluator Print/Type Name	USQ Evaluator Signature	Date
M. B. McDonough		
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.
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II. DOCUMENT ACTIONS TAKEN TO PLACE OR MAINTAIN FACILITY IN A SAFE CONDITION AND PROVIDE THE BASIS FOR THE SELECTION OF ANY INTERIM CONTROLS

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Preliminary ~~Primary~~ **analysis indicates that reducing the maximum effective plate power (EPP) for the ATR fuel elements by 13% will increase the margins from CHF and FI to 3.49 standard deviations and 3.34 standard deviations. Preliminary analysis results are attached. The Condition 2 ATR Plant Protection Criteria are 3 standard deviations from CHF or FI. Operations of the ATR with the 13% reduced EPP will result in the worst case Condition 4 accident meeting the Condition 2 plant protection criteria, and consequently, will eliminate any credible possibility of fuel damage from the worst case design basis LOCA. Other postulated design basis ATR fuel damage events occur with an intact primary**

ABH 10-5-06 MBM 10/5/06

Facility or Activity: Advanced Test Reactor

USQ Process No.: RTC-USQ-2006-655

Subject: Inconsistencies in the Maximum Hypothetical Accident Analysis

coolant system (PCS). Any fission product release will be contained in the PCS and detected by the fission break and stack monitors resulting in reactor scram and confinement isolation.

Reactor shall be operated with the following limits on ATR fuel element effective plate powers (EPPs):
For greater than or equal to 2 standard deviations to reflector ligament craking,
3 primary coolant pump (PCP) operations plate 19 EPP less than or equal to 387 MW;
3 PCP operations inner plates EPP less than or equal to 385 MW;
2 PCP operations plate 19 EPP less than or equal to 362 MW;
2 PCP operations inner plates EPP less than or equal to 362 MW; and
for less than to 2 standard deviations to reflector ligament craking,
3 PCP operations plate 19 EEP less than or equal to ~~310~~ 312 MW;
3 PCP operations inner plates EPP less than or equal to ~~318~~ 322 MW;
2 PCP operations plate 19 EPP less than or equal to 311 MW;
2 PCP operations inner plates EPP less than or equal to ~~354~~ 353 MW.

AOH 10-5-06
JC 10-5-06
MBM 10/5/06

J. C. Chapman		
Safety Analyst Print/Type Name	Safety Analyst Signature	Date
M. B. McDonough		
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, Section 15.12.10, "Radiological Analysis," Section 15.6, "Decrease in Primary Coolant Inventory," Section 15.8.7, Significant Melting of One Entire Fuel Element (or Partial Melting in More than One Element) Due to Crushing to Other Accidents,"and Section 15.10.3, "Fuel Channel Blockage or Fuel Damage Due to the Failure of Large Structural Elements Above or Within the Core"

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 PISA questioned the appropriateness of using the Section 15.12.10 large break loss-of-coolant accident (LOCA), "maximum hypothetical accident," analysis to bound the consequence to the design basis accidents. Section 15.12, "Severe Accident Analyses," discusses beyond design basis accidents. The PISA did not question the probability of occurrence of any accident.

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-2006-655

Subject: Inconsistencies in the Maximum Hypothetical Accident Analysis

2. Could the PISA increase the consequences of an accident previously evaluated in the safety basis?
Yes ☒ No ☐
Explain:
The PISA questioned the appropriateness of using Section 15.12.10 LOCA analysis to bound the consequence of design basis accidents. Due to concerns about assumptions, inaccuracies and inconsistencies of that analysis, the results of the analysis may not be conservative when used as the bounding consequence. Although engineering judgment suggests that the results of the Section 15.12.10 LOCA analysis would bound the consequences of the design basis accidents, the possibility exists that when the issues with the Section 15.12.10 analysis are resolved the consequences of the LOCA may increase which in effect will increase the consequence of the bounded accidents.

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 PISA questioned the appropriateness of using the Section 15.12.10 LOCA analysis to bound the consequence to the design basis accidents. Section 15.12, "Severe Accident Analyses," discusses beyond design basis accidents. The PISA did not question the probability of occurrence of a malfunction of any important to safety equipment.

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 PISA questioned the appropriateness of using Section 15.12.10 LOCA analysis to bound the consequence of design basis accidents. Due to concerns about assumptions, inaccuracies and inconsistencies of that analysis, the results of the analysis may not be conservative when used as the bounding consequence. The possibility exists that when the issues with the Section 15.12.10 analysis are resolved the consequences of the LOCA may increase which in effect will increase the consequence of the bounded accidents.

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 questioned the appropriateness of using the Section 15.12.10 LOCA analysis to bound the consequence to the design basis accidents. Section 15.12, "Severe Accident Analyses," discusses beyond design basis accidents. The PISA does not create the possibility of any accident of a different type.

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 questioned the appropriateness of using the Section 15.12.10 LOCA analysis to bound the consequence to the design basis accidents. Section 15.12, "Severe Accident Analyses," discusses beyond design basis accidents. The PISA does not create the possibility of any malfunction of equipment important to safety.

INL USQ PROCESS
POTENTIAL INADEQUACY IN THE SAFETY ANALYSIS (PISA) FORM
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Facility or Activity: Advanced Test Reactor

USQ Process No.: RTC-USQ-2006-655

Subject: Inconsistencies in the Maximum Hypothetical Accident Analysis

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 questioned the appropriateness of using Section 15.12.10 LOCA analysis to bound the consequence of design basis accidents. Due to concerns about assumptions, inaccuracies and inconsistencies of that analysis, the results of the analysis may not be conservative when used as the bounding consequence. Although engineering judgment suggests that the results of the Section 15.12.10 LOCA analysis would bound the consequences of the design basis accidents, the possibility exists that when the issues with the Section 15.12.10 analysis are resolved the consequences of the LOCA may increase which in effect will decrease 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 questioned the appropriateness of using Section 15.12.10 LOCA analysis to bound the consequence of design basis accidents. Although engineering judgment suggests that the results of the Section 15.12.10 LOCA analysis would bound the consequences of the design basis accidents, the possibility exists that when the issues with the Section 15.12.10 analysis are resolved the consequences of the LOCA may increase. Since the consequences of some of the design basis accidents are scaled from the Section 15.12.10 LOCA consequences, the consequence of those design basis accidents may similarly increase.

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

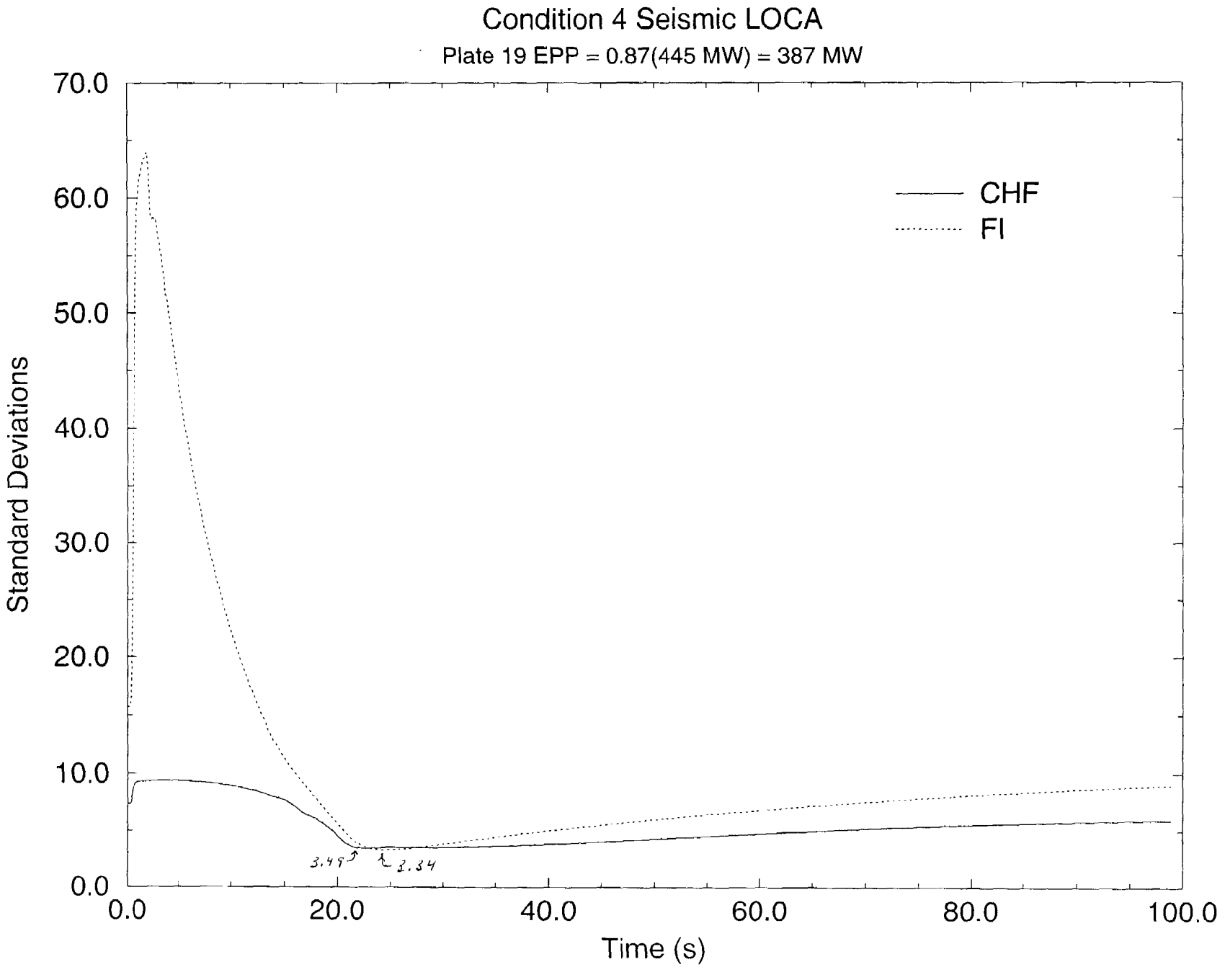
IV. APPROVAL/CONCURRENCE SIGNATURES

APPROVAL:

J. C. Chapman USQ Evaluator Print/Type Name	JC Chapman USQ Evaluator Signature	4 Oct 2006 Date
M. B. McDonough Nuclear Facility Manager Print/Type Name	M. B. McDonough Nuclear Facility Manager Signature	4/OCT/2006 Date

CONCURRENCE:

A. P. Hoskins Independent Review Committee Chair Print/Type Name	Alan P. Hoskins Independent Review Committee Chair Signature	10-5-06 Date


Condition 4 Seismic LOCA
Plate 19 EPP = 0.87(445 MW) = 387 MW
CHF
FI
Standard Deviations
Time (s)
70.0
60.0
50.0
40.0
30.0
20.0
10.0
0.0
0.0
20.0
40.0
60.0
80.0
100.0
3.49
3.34