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Technical Evaluation Study

Evaluation of the Safety of the Situation for Resolution of RTC-USQ-2008-553, "Error in RELAP5 Reactor Kinetics Model"

Idaho National Laboratory

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ACRONYMS

AHTL	ATR high temperature loop
ATR	Advanced Test Reactor
ATRC	Advanced Test Reactor Critical Facility
ATWS	anticipated transient without scram
AWIT	aluminum water ignition threshold
BEA	Battelle Energy Alliance
CHF	critical heat flux
DOE	Department of Energy
DR	deficiency report
ECAR	engineering calculations and analysis report
EDF	engineering design file
ESS	evaluation of the safety of the situation
FI	flow instability
ICARE	Issue Communication and Resolution Environment
IPT	Inpile Tube
ISI	Inservice Inspection
kW	kilowatt
LIPT	large inpile tube
LOCA	loss-of-coolant accident
LOFT	Loss of Fluid Test facility
MJ	megajoule
MW	megawatt
MUCH	Maximum Useful Capacity Holder
N_F	The reactor operating steady power level (pressurized operation, up to and including 250 MW).
N_L	For power operation and low power operation, a power level which corresponds to 1% N_F. For depressurized operation, the maximum allowed reactor power level (500 kW)
PCP	primary coolant pump
PISA	potential inadequacy in the safety analysis
PPC	plant protection criteria
RELAP	Reactor Excursion and Leak Analysis Program
RIA	reactivity insertion accident
RSS	reactor shutdown system
RTC	Reactor Technology Complex (now ATR Complex)
SAR	safety analysis report
SIPT	standard inpile tube
TEV	technical evaluation study
TSR	technical safety requirements

USQ	unreviewed safety question
USQD	USQ determination
WRS	wide range subsystem
σ	sigma (standard deviation)
$	dollars of reactivity

1. INTRODUCTION

This document provides the evaluation of the safety of the situation (ESS) for resolution of the potential inadequacy in the safety analysis (PISA) identified in unreviewed safety question (USQ) RTC-USQ-2008-553, *"Error in RELAP5 Reactor Kinetics Model"* (Reference 1).

2. ESS PURPOSE/OBJECTIVES

This document provides an ESS resulting from the contractor declaration that a PISA exists. The specific objectives of this ESS are as follows:

- Provide the background that led to the PISA declaration and resultant USQ,
- Provide the status of corrective actions to resolve the USQ,
- Provide a summary of the engineering analysis that resolves the USQ,
- Complete the USQ process,
- Provide the basis to support the request for changes to the Advanced Test Reactor (ATR) and ATR Critical (ATRC) Facility safety basis documents, and
- Provide a draft markup of required changes to the ATR and ATRC Facility safety basis documents.

3. BACKGROUND

The Reactor Excursion and Leak Analysis Program (RELAP5) computer code is the thermal-hydraulic accident analysis code used for the safety basis accident analyses for the ATR and the ATRC. Two versions of this computer code are currently used for ATR and ATRC accident analysis: RELAP5/MOD2.5 (for ATR and ATRC accident analyses) (Reference 2) and Version 3.2.1.2 of RELAP5/MOD3 (for ATR accident analyses) (Reference 3).

Two potential errors in the RELAP5 reactor kinetics model were originally identified in RTC-USQ-2008-553, the first in the calculation of terms associated with one of the six delayed neutron groups and the second in the logic that determines when to apply a quasi-steady form of the point kinetics equations rather than the fully transient form. The USQ identified that the errors could result in maximum core power and energy deposition being greater than calculated in the safety basis analyses.

Although the error discovered in the RELAP5 versions used for safety analyses could cause the ATR and ATRC maximum core power and energy deposition to be greater than some of the previous analyses, preliminary calculations performed in support of RTC-USQ-2008-553 using a corrected version of RELAP5 showed that the energy deposition of the bounding ATR Condition 2, 3, and 4 accidents were essentially unchanged or lower. The preliminary calculations provided assurance that ATR could operate with sufficient margin. Therefore, no interim operating restrictions or controls were required in RTC-USQ-2008-553 for either ATR or ATRC operations.

No impact to the safety of the workers, public or environment was identified. PISA RTC-USQ-2008-553 was declared July 24, 2008. This PISA was reported to the Department of Energy (DOE) and is being tracked in NTS-ID-BEA-ATR-2008-0005, Occurrence Report NE-ID-BEA-ATR-2008-0018, and the contractor's issues management tracking system Issue Communication and Resolution Environment (ICARE) Deficiency Report (DR) 43025.

4. USQ RESOLUTION PLAN AND CURRENT STATUS

The following corrective actions for RTC-USQ-2008-553 were identified:

1. Evaluate the effect of the RELAP5 error on the existing ATR and ATRC accident analyses.

   Status: An analysis was conducted and documented in Engineering Calculations and Analysis Report (ECAR)-402 (Reference 4). The results of that analysis are summarized in Section 5.2.

2. Because of the error in the RELAP5 code, previous ATR experiment-specific safety analyses may no longer be valid and should not be referenced. Ensure that the affected analyses are identified and cancelled.

   Status: The affected safety analyses have been identified in ECAR-402. The analyses have been cancelled in the Idaho National Laboratory Electronic Document Management System.

3. Complete an ESS, completing the USQ process.

   Status: This ESS summarizes the results of ECAR-402, and provides the basis for resolution of RTC-USQ-2008-553.

4. Prepare necessary revisions to the ATR and ATRC safety basis documentation.

   Status: Appendix B provides a markup of Safety Analysis Report (SAR)-153 (Reference 5) and Appendix C provides a markup of SAR-192 (Reference 6) to reflect the analyses and conclusions in ECAR-402. No changes to Technical Safety Requirements (TSR)-186 (Reference 7) or TSR-192 (Reference 8) are required.

   This markup is provided for information only. The SAR changes will be incorporated into a future annual update and submitted to DOE for review and approval.

5. RESOLUTION OF THE PISA IDENTIFIED IN RTC-USQ-2008-553

5.1 Error in RELAP5 Reactor Kinetics Model

RELAP5/MOD2.5 is used to simulate most of the events described in SAR-153 and all the reactivity events described in the SAR-192, while RELAP5/MOD3 Version 3.2.1.2 is used to simulate some of the pressurized water loop experiment voiding events described in the SAR-153. RELAP5/MOD3 Version 3.2.1.2 is also generally used in the experiment safety analyses that support the irradiation of various tests.

Both versions of RELAP5 contain a point reactor kinetics model that has been used to simulate power excursion transients at the ATR and ATRC. The point kinetics model was validated against numerous exact solutions and calculations from other computer codes as described in Reference 9. All of the validation calculations showed excellent performance of the RELAP5 point reactor kinetics model. However, errors in the point kinetics model were reported to the RELAP5 code developers in 2007. The errors were originally reported by researchers at Purdue University who were studying reactivity transients in an Argentine reactor. The errors were reported to rarely have a significant effect on calculated results. The first error was in the calculation of terms associated with one of the six delayed neutron groups. The second error was related to logic that determined when to apply a quasi-steady form of the point kinetics equation rather than the fully transient form. A detailed description of the errors is presented in Appendix A of this ESS.

As discussed in Appendix A, corrected versions of the RELAP5/MOD2.5 and RELAP5/MOD3 Version 3.2.1.2 codes were developed (2007 version as described in Appendix A). The verification and validation calculations for the original versions of RELAP5 were repeated with the corrected codes. Preliminary testing during this evaluation showed that the correction to the first error reported in 2007 actually caused slightly worse results for the reactivity step insertion in that the power remained constant during the first time step. Physically, the power should increase immediately following a step insertion of reactivity. The original developer of the point kinetics model was consulted about this anomaly. He re-evaluated the corrections generated in 2007 and concluded that the first reported error was not actually an error. The contributions of the delayed neutron groups were correctly calculated in the original code, although subtleties in the coding could easily cause one to come to the opposite conclusion following a cursory examination. This conclusion of the original developer was confirmed by calculations performed during this evaluation. The code version with the corrections to the second reported error but without the corrections to the first reported error (2008 version as discussed in Appendix A) produced results that were equivalent to or slightly better than the version with the corrections to both reported errors (2007 version as described in Appendix A). In particular, the code version without the first "correction" converged to the exact solution at larger time steps during the reactivity ramps. The corrected versions of RELAP5/MOD2.5 and RELAP5/MOD3 Version 3.2.1.2 without the first "correction" were verified and validated as documented in Reference 10 and as discussed in Appendix A.

Calculations were also performed in Reference 10 to demonstrate that the original and corrected versions of RELAP5 produced identical results for cases that did not use the point reactor kinetics model. The simulated cases included blowdown experiments in the LOFT and Semiscale facilities (Reference 11) and a loss-of-coolant accident in a hypothetical pressurized water reactor. Thus, the corrections to the point kinetics model do not affect the basic hydrodynamic solution.

5.2 Evaluation of the Effects of the Error on the ATR and ATRC Accident Analysis

5.2.1 Summary of the Analysis Conducted in ECAR-402

ECAR-402 (Reference 4) evaluates the effect of the error on existing ATR and ATRC accident analyses. Several reactivity insertion accident (RIA) calculations discussed in SAR-153, SAR-192, and Engineering Design File (EDF) -6937 (Reference 12) (Analysis of ATR High Temperature Loop (AHTL) Flow Coastdown with New Pumps) were rerun with the corrected RELAP5 codes. A non-RIA calculation (ATR loss of commercial power) was also rerun. In a few cases as discussed in the following sections, the maximum calculated core power and/or core energy deposition were higher than calculated previously (the effect of the RELAP5 error on the non-RIA calculation was negligible). However, in all cases, the ATR and ATRC plant protection criteria were met. The RELAP5 error does not change the conclusions of SAR-153, SAR-192, or EDF-6937.

5.2.2 Analysis Cases

SAR-153, SAR-192, and EDF-6937 were reviewed in ECAR-402 to determine which RELAP5 calculations should be rerun.

SAR-153, Section 15.4 (Reactivity and Power Distribution Anomalies)

The following RELAP5 calculations were rerun:

- Experiment Loop 1/2-in. Pipe Break (SAR-153, Section 15.4.4). This is the limiting Condition 2 reactivity fault. Only 3-PCP operation was simulated since it bounds 2-PCP operation.
- 0.50$ Step Insertion (SAR-153, Section 15.4.5). This is a hypothetical Condition 3 reactivity fault that bounds several Condition 2, 3, and 4 faults. Only 3-PCP operation was simulated since it bounds 2-PCP operation.
- Experiment Loop Large Pipe Break (SAR-153, Section 15.4.6). This is the limiting Condition 4 reactivity fault. Only 3-PCP operation was simulated since it bounds 2-PCP operation.
- Pressure Tube Flow Bypass Due to MUCH Test Train Failure with a Loop Scram (SAR-153, Section 15.4.7.10). The RELAP5 calculation was rerun to confirm that this Condition 3 fault is bounded by a 0.50$ step.
- Withdrawal of All Outer Shims from 10^{-10} N_F with Failure of the WRS (SAR-153, Section 15.4.7.14). The RELAP5 calculation was rerun to confirm that this Condition 4 fault is bounded by a 0.30$/s ramp (i.e., maximum core power < 321.6 MW).
- Withdrawal of All Outer Shims and Neck Shims from 10^{-5} N_F Coupled with Rapid Regulating Rod Withdrawal at 10^{-3} N_F (SAR-153, Section 15.4.7.17). The RELAP5 calculation for this Condition 4 fault was rerun to confirm that the power does not reach 10^{-3} N_F.
- Voiding in All IPTs Due to Simultaneous Flow Coastdown or Loss-of-Temperature Control in all Experiment Loops (SAR-153, Section 15.4.7.19). The flow coastdown calculation was rerun to confirm that this Condition 4 fault is bounded by a 0.30$/s ramp (i.e., maximum core power < 321.6 MW).

SAR-153, Section 15.9 (Depressurized Operations Events)

The following RELAP5 calculations were rerun:

- 0.30$/s Ramp Insertion. Most of the reactivity accidents discussed in SAR-153, Section 15.9.3 are bounded by the 0.30$/s ramp in EDF TRA-ATR-1835 (Reactivity Accident Analysis for ATR Depressurized Operation). The 0.30$/s ramp was rerun to confirm that the maximum core power is less than 7.5 MW, which is the power limit applied to all of the reactivity accidents in SAR-153, Section 15.9.3.
- Rapid Regulating Rod Withdrawal (SAR-153, Section 15.9.3.2). The reactivity insertion rate during this Condition 3 fault is 1.05$/s, with a total insertion of 0.57$. The maximum core power was previously calculated to be less than that for a 0.30$/s ramp. The RELAP5 calculation was rerun to confirm this result.
- Perched Fuel Element Drop from within Measurement Accuracy (One Inch) (SAR-153, Section 15.9.3.8). This Condition 3 fault and the Condition 4 step insertions discussed in SAR-153, Section 15.9.3.12, are bounded by a 0.10$ step. The maximum core power following a 0.10$ step was previously calculated to be less than that for a 0.30$/s ramp. The RELAP5 calculation was rerun to confirm this result.

SAR-153, Section 15.12 (Severe Accident Analyses)

The following RELAP5 calculations were rerun:

- Anticipated Transient Without Scram (ATWS) (SAR-153, Section 15.12.5). Two ATWS calculations are discussed in this section. They are: (a) primary pump coastdown due to a loss of commercial power and (b) positive reactivity insertion due to an experiment loop 1/2-in. pipe break. Neither of these accidents results in core damage. The RELAP5 calculations were rerun primarily to investigate the effect of the reactor kinetics errors.
- Very Large RIA (SAR-153, Section 15.12.6). A 1.30$ step was previously analyzed with the RELAP4 code. As noted earlier, there are no errors in the RELAP4 reactor kinetics model. A 1.30$ step was simulated with RELAP5 (with and without the errors) to investigate the effect of the errors on a very large RIA.
- Direct Damage Large RIA (SAR-153, Section 15.12.7). The RELAP5 model that was used to simulate this accident is no longer available. This accident is initiated by the rupture of an IPT inside the core. It is described as being "nearly equivalent to a 0.90$ step." A 0.90$ step was simulated with the current RELAP5 model to investigate the effect of the reactor kinetics errors on a large RIA.

SAR-153, Section 10.2 (Pressurized Water Loop Experiment Facilities)

Numerous IPT blowdown calculations have been performed with RELAP5 to establish operating limits for pressurized water loop experiments (SAR-153, Section 10.2.6.2.2). The worst Condition 2 and 4 (i.e., small and large experiment loop pipe break) calculations were rerun for each IPT configuration (SIPT, LIPT, Standard MUCH, and AHTL MUCH). These calculations, eight total, were compared to the limiting Condition 2 and 4 RIA calculations in SAR-153, Section 15.4, to confirm that they are bounded.

SAR-192, Section 8.4 (Accident Analysis)

A core energy deposition limit (15 MJ) is used to demonstrate that no fuel melting occurs in ATRC reactivity accidents. The following RIA calculations were rerun. They are Condition 4 faults and are the only RIAs discussed in the SAR-192 that result in significant core energy deposition (> 1 MJ).

- Air Voiding of a Large Inpile Tube (SAR-192, Section 8.4.8.1).
- Aluminum Filler Piece Drop (SAR-192, Section 8.4.8.2).

EDF-6937 (Analysis of AHTL Flow Coastdown with New Pumps)

The AHTL pumps were replaced recently. The RELAP5 calculations in EDF-6937 show that RIAs initiated by coast down of the new AHTL pumps and by simultaneous coast down of all of the experiment loop pumps are bounded by the limiting Condition 2 and 4 reactivity faults discussed SAR-153, Section 15.4. This was confirmed by rerunning the two worst RIA calculations in EDF-6937.

In addition, the calculations in ECAR-402 incorporate a revised time step size of 0.0001 s to ensure that the scram time is calculated more accurately. For fast reactivity transients, Reference 13, recommends 0.0001 s. Reference 10 shows that more accurate solutions to the reactor kinetics equations are obtained (with the corrected code) when the time step size is reduced. ECAR-402 therefore evaluates the effect of the error on existing ATR and ATRC accident analyses as well as incorporating a revised time step. Specific conclusions from ECAR-402 are summarized in the following sections.

5.2.3 Reactivity and Power Distribution Anomalies (SAR-153, Section 15.4)

Except as noted, the scram setpoint for the calculations discussed in this section was 20% overpower (300 MW), with a response time of 0.025 s.

5.2.3.1 Inpile Tube (IPT) Decompression Due to Experiment Loop 1/2-in. Pipe Break (SAR-153, Section 15.4.4)

This is the limiting Condition 2 reactivity fault in SAR-153. Only the calculation with three primary coolant pumps (PCPs) operating was updated since it bounds the 2-PCP calculation. The decompression and subsequent voiding of an IPT as a Condition 2 event can occur as a result of operator error or mechanical failure of components. Piping failures considered are 1/2-in. or less for this probability classification. The evaluation envelops low pressures resulting from control system failures as well. The analysis can be shown to envelop other failures in large diameter pipes that are isolated from the IPT by substantial piping lengths. Such evaluations may be completed on a case basis to support other activities (e.g., the application of the Inservice Inspection (ISI) plan to the loops).

The limiting failure is a shear of a 1/2-in. pipe in the drain manifold attached to the loop piping at the heater legs. This event results in IPT voiding and a positive reactivity insertion. A scram occurs on high power and is initiated by the neutron level subsystem. The dominant shutdown mechanism is feedback from the fuel elements. The safety rod insertion provides some mitigation of the event and ensures the reactor remains subcritical.

The original calculations in Reference 14 and discussed in SAR-153, Section 15.4.4, were updated in ECAR-402 with and without the RELAP5 code error correction. In addition, the calculation with the corrected code incorporates a revised time step size of 0.0001 s. The original calculation was performed with a time step size of 0.005 s. The maximum core power and core energy deposition results from ECAR-402 are as follows:

Table 1. ECAR-402 Results for Experiment ½-in. Pipe Break.

Parameter	Calculation		
	Original	Corrected	Revised Time Step
Scram Time (s)	0.0750	0.0750	0.0690
Maximum Core Power (MW)	464.54	464.66	456.91
Maximum Energy Deposition (MJ)	27.553	27.556	25.879

As shown above in Table 1 and in Figures 1 and 2, the original and corrected calculations were nearly identical and both calculations bounded the revised time step calculation (the most accurate of the calculations). The maximum core power and core energy deposition decreased when the time step size was reduced, primarily because the scram time was calculated more accurately.

The original ATR-SINDA and SINDA-SAMPLE calculations (SAR-153, Section 15.4.4) show that the event meets the acceptance criteria for a Condition 2 event (margins to critical heat flux (CHF) and flow instability (FI) exceed 3σ). As shown in Figures 1 and 2, since the revised calculations in ECAR-402 show that the maximum core power and core energy deposition with a time step size of 0.0001 s are bounded by the original ATR-SINDA and SINDA-SAMPLE calculations, it is concluded that the analysis in SAR-153, Section 15.4.4, is bounding, and the ATR plant protection criteria (PPC) are met.

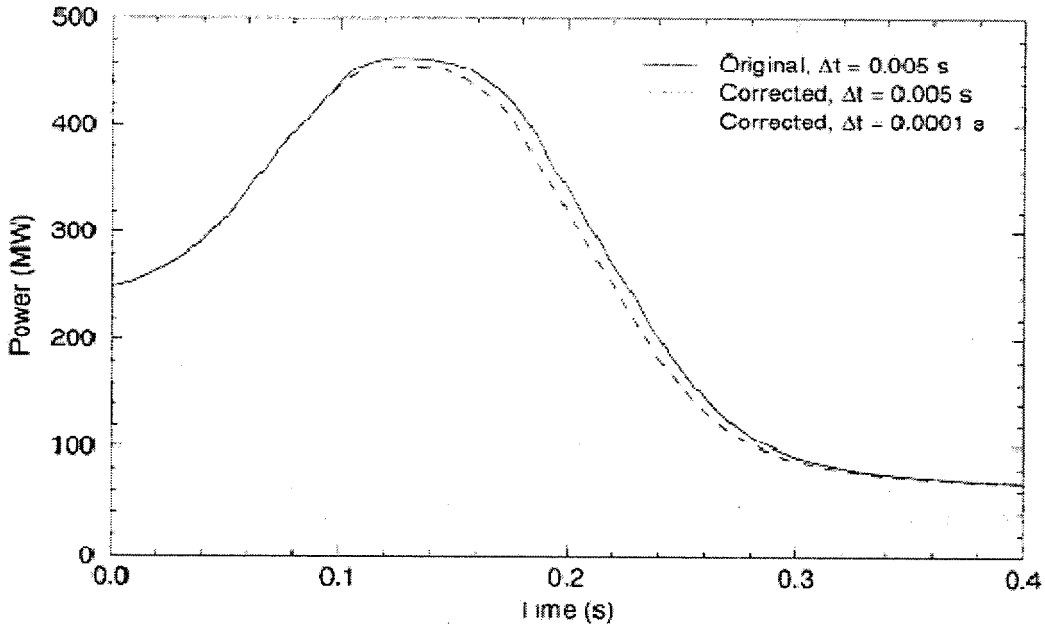

Figure 1. Core power vs. time during limiting Condition 2 RIA (experiment loop 1/2-in. pipe break).

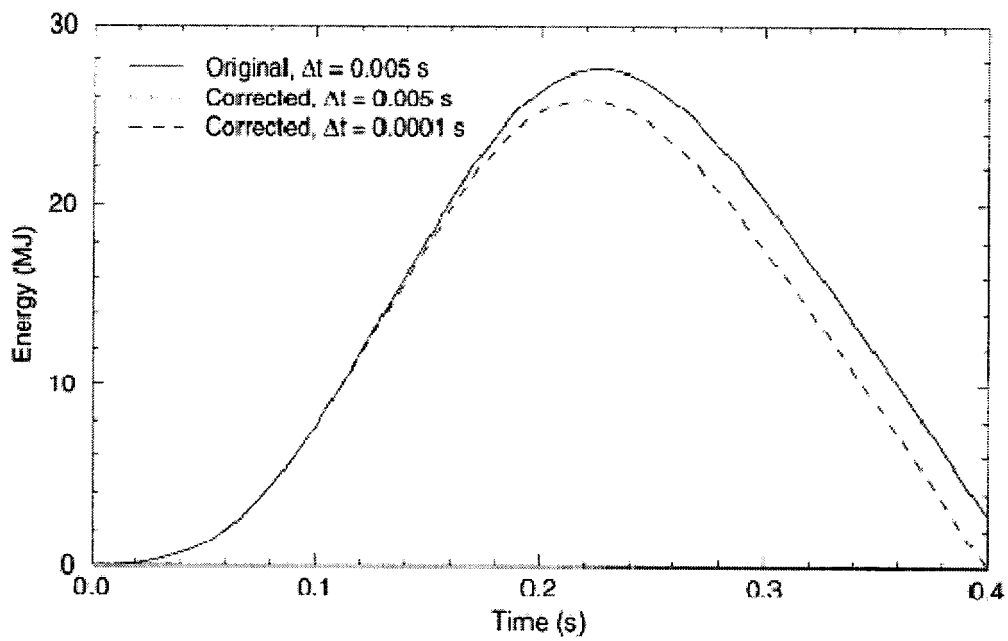

Figure 2. Core energy deposition vs. time during limiting Condition 2 RIA (experiment loop 1/2-in. pipe break).

5.2.3.2 0.50$ Step Insertion (SAR-153, Section 15.4.5)

This is the limiting Condition 3 reactivity fault in SAR-153. Only the 3-PCP calculation was rerun since it bounds the 2-PCP calculation. The 0.50 s step is a hypothetical fault which is used as a bounding event for several Condition 2, 3 and 4 faults. The reactivity insertion of 0.50$ was inserted at time zero as an instantaneous step. A reactor trip occurred from the neutron level subsystem at 1.2 N_F with a response time of 0.025 s.

The calculations in Reference 14 and discussed in SAR-153, Section 15.4.5, were updated in ECAR-402 with and without the RELAP5 code error correction. In addition, the calculation with the corrected code incorporates a revised time step size of 0.0001 s. The original calculation was performed with a time step size of 0.005 s. The maximum core power and core energy deposition results from ECAR-402 are as follows:

Table 2. ECAR-402 Results for 0.50$ Step Insertion.

Parameter	Calculation		
	Original	Corrected	Revised Time Step
Scram Time (s)	0.0400	0.0350	0.0283
Maximum Core Power (MW)	468.44	468.44	467.39
Maximum Energy Deposition (MJ)	24.448	23.611	22.823

As shown above in Table 2 and in Figures 3 and 4, the original and corrected calculations were nearly identical and both calculations bounded the revised time step calculation (the most accurate of the calculations). The maximum core power and core energy deposition decreased when the time step size was reduced, primarily because the scram time was calculated more accurately.

The original ATR-SINDA and SINDA-SAMPLE calculations (SAR-153, Section 15.4.5) show that the event meets the acceptance criteria for a Condition 3 event (2σ to buckling). As shown in Figures 3 and 4, since the revised calculations in ECAR-402 show that the maximum core power and core energy deposition with a time step size of 0.0001 s are bounded by the original ATR-SINDA and SINDA-SAMPLE calculations, it is concluded that the analysis in SAR-153, Section 15.4.5, is bounding, and the ATR PPC are met.

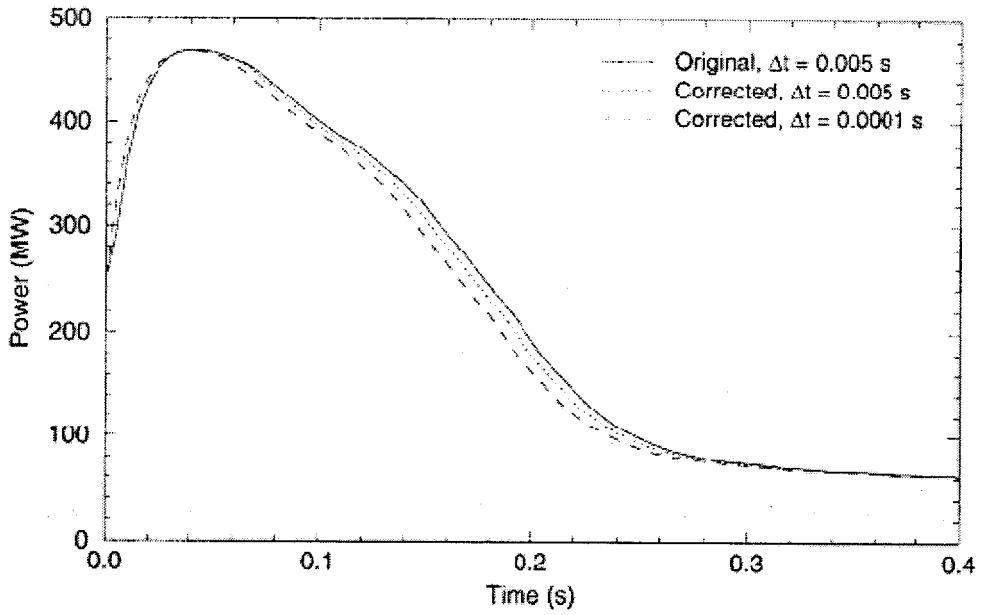

Figure 3. Core power vs. time during limiting Condition 3 RIA (0.50$ step).

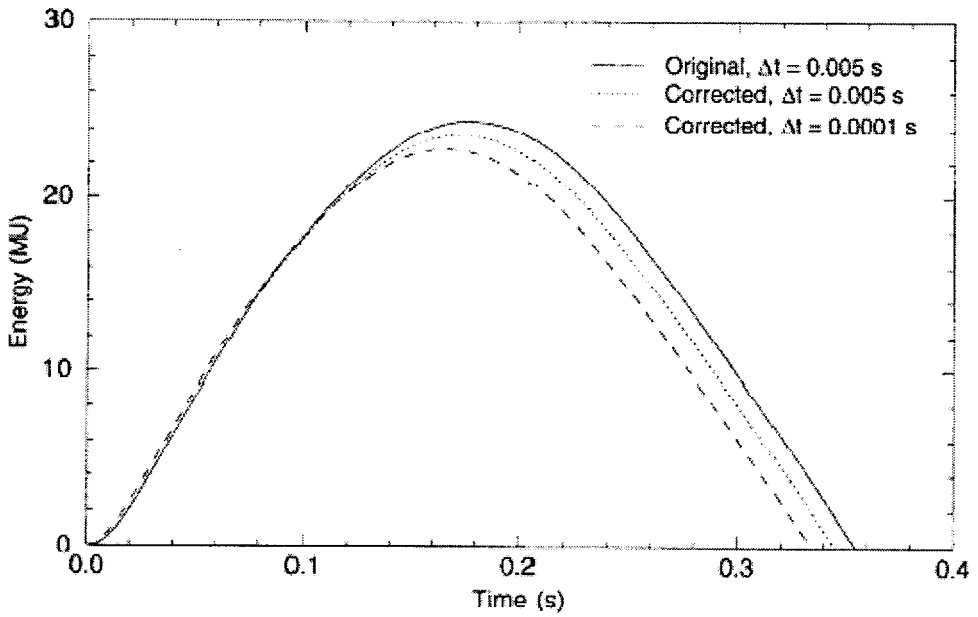

Figure 4. Core energy deposition vs. time during limiting Condition 3 RIA (0.50$ step).

5.2.3.3 IPT Decompression Due to Experiment Loop Large Pipe Break (SAR-153, Section 15.4.6)

This is the limiting Condition 4 reactivity fault in SAR-153. This event is initiated by failure of the piping in a pressurized water loop facility. Failure of the pipe with the resultant decompression is assumed to result in failure of the experiment hardware. The load results from blowdown forces on the test. The hardware failure results in a relocation dependent upon the distance between the bottom of the test and the experiment stop below the test. The relocation results in a reactivity insertion in addition to the voiding insertion. This insertion is bounded as a 0.10$ step. During the transient, temperature changes in the other flux trap facilities result in further positive reactivity insertion as a result of cascading. This contribution is bounded by previous analyses for cascading associated with a 0.75$ step. A scram occurs on high power and is initiated by the neutron level PPS subsystem. The event is terminated by feedback from the fuel elements. The safety rod insertion ensures the reactor remains subcritical. Only the 3-PCP calculation was rerun since it bounds the 2-PCP calculation. The calculations in Reference 14 and discussed in SAR-153, Section 15.4.6, were updated in ECAR-402 with and without the RELAP5 code error correction. In addition, the calculation with the corrected code incorporates a revised time step size of 0.0001 s. The original calculation was performed with a time step size of 0.005 s. The maximum core power and core energy deposition results from ECAR-402 are as follows:

Table 3. ECAR-402 Results for Experiment Loop Large Pipe Break.

Parameter	Calculation		
	Original	Corrected	Revised Time Step
Scram Time (s)	0.0450	0.0400	0.0390
Maximum Core Power (MW)	572.37	563.47	560.58
Maximum Energy Deposition (MJ)	37.304	35.505	35.180

As shown above in Table 3 and in Figures 5 and 6, the original and corrected calculations were nearly identical and both calculations bounded the revised time step calculation (the most accurate of the calculations). The maximum core power and core energy deposition decreased when the time step size was reduced, primarily because the scram time was calculated more accurately.

The original ATR-SINDA and SINDA-SAMPLE calculations (SAR-153, Section 15.4.6) show that the event meets the acceptance criteria for a Condition 4 event, (greater than 3σ margin to the AWIT temperature), thus meeting the acceptance criterion for Condition 4 faults.

As shown in Figures 5 and 6, since the revised calculations in ECAR-402 show that the maximum core power and core energy deposition with a time step size of 0.0001 s are bounded by the original ATR-SINDA and SINDA-SAMPLE calculations, it is concluded that the analysis in SAR-153, Section 15.4.6, is bounding, and the ATR PPC are met.