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

Facility or Activity: ATR and ATR Critical Facility

USQ Process No.: RTC-USQ-2008-553

Subject: Error in REAP5 Reactor Kinetics Model

Describe the New Information/Discovery:
It has been discovered that there is an error in the reactor kinetics model in the RELAP5 computer code. RELAP5 is the thermal-hydraulic accident analysis code used for the safety basis accident analyses for the Advanced Test Reactor (ATR) and the ATR Critical Facility. The error is in the summation of the delayed neutron terms. Also poor coding logic was used in the switch between the steady-state and the transient solutions. This error and poor coding logic, henceforth simply called "error," could result in maximum core power and energy deposition being greater than calculated in the safety basis analyses. This error is present in the ATR version of RELAP5/MOD2.5 (used for ATR and ATR Critical Facility accident analyses) and Version 3.2.1.2 of RELAP5/MOD3 (used for ATR accident analyses).

Identify the applicable safety basis document(s) (e. g., DSA, SAR, BIO, TSRs, etc.):
SAR-153 Upgraded Final Safety Analysis Report for the Advanced Test Reactor
TSR-186 Technical Safety Requirements for the Advanced Test Reactor
SAR-192 Safety Analysis Report for the Advanced Test Reactor Critical Facility (ATRC)
TSR-192 Technical Safety Requirements for the Advanced Test Reactor Facility (ATRC)

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
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.

Since this error in the kinetics model is present in version RELAP5/MOD2.5, which was used for both ATR and ATR Critical Facility accident analyses, and Version 3.2.1.2 of RELAP5/MOD3, which was used for ATR accident analyses, this error could result in maximum core power and energy deposition being greater than calculated in the safety basis analyses.

A PISA exists. Complete the following actions

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- 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.

ATR was in two pump operation, Cycle 142B, at the time of discovery. The maximum effective plate power (EPP) for this operating cycle was 253 MW. The safety basis analyses for two primary coolant pump (PCP) operation assumed that the maximum EPP was 417 MW. Based on preliminary calculations using a version of RELAP5 with this error corrected, the potential error was predicted to only have minimal effect. Considering the large margin between the maximum EPP assumed the safety basis analyses and the actual cycle EPP (417 MW vs 253 MW), no immediate action was need to place the ATR in a safe condition. The ATR Critical Facility was not operating at the time of the discovery.

As noted above, the error in RELAP5 was corrected and scoping calculations were performed to assess the effect of this error on various accidents. The bounding Condition 2, 3 and 4 reactivity insertion accidents for ATR were recalculated using the corrected version of RELAP5. As seen in the attached figures, the energy deposition using the corrected RELAP5 version essentially overlays the results of the safety analysis version for the Condition 2 and 4 bounding faults (Figures 1 and 3) and is less than the results of the safety analysis version for the Condition 3 bounding fault (Figure 2). A reactivity ramp fault, withdrawal of all outer shims and neck shims with the failure of the Wide Range System, a bounded event, was also recalculated using the corrected RELAP5 version. The core power response calculated using the corrected version essentially overlays the result of the safety analysis version (Figure 4).

The effect of this error on the generic Condition 4 experiment loop voiding analyses was also evaluated. The generic experiment loop analyses were performed to establish an envelope for loop experiments. If a loop experiment is not within the established envelope, an experiment specific voiding analysis must be performed to ensure that voiding of a particular loop experiment meets the ATR Plant Protection Criteria. For both the Standard inpile tube (IPT) and the Large IPT loop experiment voiding faults, the maximum energy depositions using the corrected RELAP5 version remained less than the Chapter 15 safety analysis. Thus, the generic Condition 4 analyses will remain bounded by SAR-153 safety analysis. The corrected version calculated energy depositions, however, exceeded the previously calculated generic Condition 4 loop voiding analyses (Figures 5 and 6); but, as stated above, will remain bounded by the energy deposition assumed in the SAR-153 accident analyses.

Of particular concern is the accident analysis for one of the experiments currently in the reactor (2E-NW-158). The maximum void worth of the IPT during Experiment 2E-NW-158 could potentially be greater than that allowed in the ATR SAR. Therefore, per SAR Section 10.2.6.4.1, an analysis was performed to demonstrate that the consequences of an accident with this experiment inserted are acceptable. (Exhibit 1 is the cover page of ECAR-208, Inpile Tube Voiding Analysis for ATR Experiment 2E-NW-158.) The void worth of the IPT was assumed to be 1.10$ instead of 1.00$ (i.e., the SAR limit for the Large IPT). The safety analysis (using RELAP5 version with the error) showed that the ATR Plant Protection Criteria are met. With the error in RELAP5 corrected, however, thermal safety margins for the limiting Condition 4 reactivity fault (>½-in. experiment loop pipe break) may be less than demonstrated to be acceptable in SAR Chapter 15.

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Figure 7 shows that the energy deposition calculated using the corrected RELAP5 version exceeds that calculated in ECAR-208. The safety analysis for Experiment 2E-NW-158, ECAR-208, however, is very conservative. The maximum effective plate power (EPP) was 443 MW and a step reactivity insertion of 0.10$ was included in the RELAP5 calculation to conservatively bound the effect of test train failure. (NW lobe power assumed for this analysis was 34 MW.) The Core Safety Assurance Package (CSAP) for the current cycle shows that the maximum EPP is 253 MW and that failure of the test train during Experiment 2E-NW-158 would not cause a reactivity insertion. Figure 7 shows the calculated integrated core power vs. time (i.e., core energy deposition) for a >½-in. loop pipe break for Experiment 2E-NW-158, with and without the RELAP5 error correction and with and without a 0.10$ step insertion due to test train failure. The calculation with the corrected RELAP5 code and without a 0.10$ step is bounded by the accident analysis in ECAR-208. In addition, the calculation assumed the NW lobe power was 34 MW. (The nominal northwest (NW) lobe power for this cycle is 23 MW; the maximum NW lobe power is 26 MW.) The ATR is currently operating with two primary coolant pumps (PCPs) and a maximum EPP of 253 MW. The EPPs considered in Chapter 15 safety analysis are considerably greater than 253 MW; (i.e., 417 MW for 2-PCP operation and 443 MW for 3-PCP operation). The recalculated energy deposition using the corrected RELAP5 version with the assumed NW lobe of 26 MW would be bounded by the accident analysis in ECAR-208. Based on the relatively low power at which the reactor is operating and several scoping calculations, the error in RELAP5 will not result in exceeding the ATR Plant Protection Criteria.

JEC 8-4-08

The effect of the RELAP5 error on ATR depressurized operations was evaluated by recalculating, using the corrected version, the enveloping 0.30$/sec ramp insertion which is the basis for comparing the accident analyses to the ATR Plant Protection Criteria. (Exhibit 2 is the cover page of the EDF TRA-ATR-1835, Reactivity Accident Analysis for ATR Depressurized Operation.) The recalculated maximum power is approximately 0.1 MW higher than the safety analysis calculation (Figure 8); the recalculated maximum core power, however, is well below the safety basis limit of 7.3 MW, which was demonstrated to meet the ATR Plant Protection Criteria.

Reactive insertion accidents for the ATR Critical Facility (SAR-192) were also recalculated using the corrected RELAP5 version. The effect on Condition 4 Large IPT Voiding and the Filler Piece Drop accidents were evaluated. The recalculated maximum power and the energy deposition (integrated core power) exceed the calculated safety analysis RELAP5 version values for the Larger IPT voiding analysis (Figures 9 and 10) and essentially overlay the Filler Piece Drop accident analysis (Figures 11 and 12). The ATR Critical Facility accident acceptance criterion, however, is an energy disposition of less than or equal to 15 MJ. As shown in Figures 10 and 12, the recalculated energy deposition for these Condition 4 accidents is well below the ATR Critical Facility acceptance criterion.

Although the error discovered in the RELAP5 versions used for safety analyses apparently caused the maximum core power and energy deposition to be greater than some of the previous analyses, based on the scoping calculations that used the corrected RELAP5 version, this error will not challenge the conclusions of either the ATR or the ATR Critical Facility safety basis. Therefore, no interim operating restrictions or controls are required for either ATR or ATR Critical Facility operations.

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J. C. Chapman		**Signed 7/24/08**
Safety Analyst Print/Type Name	Safety Analyst Signature	Date
M. B. McDonough	M. B. McDonough	**Signed 7/24/08**
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. USQ DETERMINATION

Identify applicable section(s) of the safety basis document(s) (e.g., DSA, SAR, BIO, TSRs, etc.).
SAR-153 Section 10.1.6 "In some cases, the experiment specific initial conditions or potential reactivity insertions may not be clearly bounded by those in Chapter 15 analysis. In these cases, the ESA [experiment safety analysis] includes detailed analysis of the specific experiment parameters and operation to show compliance with the ATR Plant Protection Criteria." This analysis includes a RELAP5 analysis for the postulated Condition 2, 3, and 4 reactive insertion events.

SAR-153 Section 10.2.6.2.2 "Prior to reactor operation, the operation of each PWL [pressurized water loop] facility that is operated at greater than 200°F or greater than 750 psig (1500 psig for AHTL [ATR High Temperature Loop]) is compared with the IPT [inpile tube] and PWL evaluations (Section 10.2.1.3 and 10.2.2.3) and the loop blowdown envelope analyses to ensure that the operation is consistent with the assumptions and results of the analyses. The table below is a compilation of these analyses with the more restrictive controlling limits listed. If the experiment parameters are within the blowdown analyses, then operation of the experiment is within the assumptions of the analyses in UFSAR Chapter 15 for a loop blowdown."

"Operations of an experiment outside the limits below may be shown to be acceptable (usually by limiting other experiment conditions) if analyses demonstrate that conducting the experiment is within the ATR Plant Protection Criteria as discussed in Chapter 15 (Accident Analyses).

SAR-153 Section 10.2.6.4.3 "The reactivity insertion from this event is bounded by analyses used in Chapter 15 (Accident Analyses) if the reactivity insertion ≤ 0.10$. Whenever the reactivity insertion exceeds 0.10$, additional analyses is performed to show that the combined reactivity insertion from a loop decompression and the experiment failure is within the ATR Plant Protection Criteria for a Condition 4 event."

SAR-153 Section 10.3.3 "The accident analyses of the PWL facilities and experiments in Chapter 15 (Accident Analyses) provide reactivity transients that can be used to envelop potential effects from the capsule facilities and experiments. The PWL reactivity insertion accidents have been fully analyzed to determine the resulting power transients and margins for fuel element performance."

"As part of the analyses for each capsule facility ESA, potential failures are identified and compared against the results in Chapter 15 (Accident Analyses) for the PWL facilities. Additional analyses are completed as necessary to demonstrate compliance with the ATR Plant Protection Criteria for all operating conditions."

SAR-153 Section 10.3.5.1 "The potential reactivity addition of experiment cooled by the reactor coolant must be evaluated relative to the accepted reactivity insertion events in Chapter 15 (Accident Analyses) and shown to be enveloped, or specific analyses must be completed."

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SAR-153 Section 10.3.5.1.1 "The potential reactivity insertion rate shall not exceed the reactivity insertion rate of the limiting event in each fault category analyzed in the UFSAR without additional analyses to show acceptable consequences."

SAR-153 Section 15.4.1.1 "The analysis of reactivity and power distribution anomaly events was completed by the use of RELAP5, ATR-SINDA, and SINDA-SAMPLE. The power transients resulting from the reactivity insertions must be generated from the RELAP5 reactor model and input to ATR-SINDA."

"In order to consider the effect of this cascading, an additional reactivity input is included in RELAP5."
"The reactivity insertion rate from thermal-hydraulic upsets in the IPT was calculated with the RELAP5 models of the loop facilities with an assumed total reactivity insertion for complete voiding of the IPT (0.80$ for standard and AHTL IPT and 1.0$ for LIPT [large inpile tube])."

SAR-153 Section 15.4.2.2 "The DBR (SAR-39) contains analysis results for various ramp rates that demonstrate the maximum reactor power which would be attained with pressurized and depressurized operation. The results for pressurized operation are illustrated in Figure 15.4-2. The data in this figure
were generated with a RELAP4 model, however, as discussed in Terry, there is very good agreement of results between the newly-developed RELAP5 model and the previous models."

SAR-153 Section 15.4.4.2.2, "The loop blowdown analyses showed that the bounding [Condition 2] reactivity insertion occurred for the standard loop with the MUCH [Maximum Useful Capacity Holder] configuration. The insertion rate was initially about 6$/second and reached 0.61$ total insertion in 0.2 seconds for an average rate of about 3$/second."

"The RELAP5 results for reactor power show a maximum of 428 MW at 0.12 seconds for two-pump case and 365 MW at 0.13 seconds for the three-pump case."

SAR-153 Section 15.4.5.1 "The 0.50$ step is a hypothetical fault which is used as a bounding event for the following several near step events: A. Limiting perched fuel element (Condition 2), B. Loop experiment hardware failure (Condition 3), C. Loss of reflector coolant (Condition 3), D. Reflector movement toward core (Condition 3), E. Perched fuel drop to two fuel elements from within measurement accuracy (Condition 3), F. Cold water addition from opening of primary pump discharge valve (Condition 3), G. Gas release into the core (Condition 3), H. Perched fuel element (greater than the measurement uncertainty) (Condition 4), I. Movement of two reflector blocks (Condition 4)."

SAR-153 Section 15.4.5.2.2 "The RELAP5 results [0.50$ step reactivity insertion] show a peak power of 435 MW occurs at 0.04 seconds for the two pumps and 369 MW at 0.04 seconds for three pumps."

SAR-153 Section 15.4.6.2.2 "The analyses show that the bounding reactivity insertion [Condition 4] occurs as a result of the DEOS [double-ended offset shear] of the SIPT [standard inpile tube] at the pump discharge."

"The RELAP5 results for the reactor power show a maximum of 526 MW at 0.1 seconds for two-pump operations and 451 MW at 0.1 seconds for three-pump operations."

SAR-153 Section 15.4.7, Bounded Events – Reactivity and Power Distribution Anomalies;
Section 15.4.7.1 - Rapid Regulating Rod Withdrawal
Section 15.4.7.2 – Withdrawal of all Outer Shims from 10-10 NF
Section 15.4.7.3 - Withdrawal of all Outer Shims and One Neck Shim from NL
Section 15.4.7.4 – Cold Water Injection
Section 15.4.7.5 – Inpile Tube Voiding Due to a Rupture Disk or Relief Valve Failure or an Open and Accessible

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Valve
Section 15.4.7.6 – Loop Flow Coastdown or Loop Loss of Temperature Control With a Loop Instrumentation Initiated Reactor Trip
Section 15.4.7.7 – Powered Axial Locating Mechanism Drive System Failures
Section 15.4.7.8 – Withdrawal of All Outer Shims and Neck Shims from 10-10 NF
Section 15.4.7.9 – Inpile Tube Voiding Due to Opening of a Normally Inaccessible Valve
Section 15.4.7.10 – Pressure Tube Flow Bypass Due to a Maximum Useful Capacity Holder Test Train Failure with Loop Scram
Section 15.4.7.11 – Loss of Loop Temperature Control Due to Heat Exchanger Failure or Line Heater Sticking on Without Loop Scram
Section 15.4.7.12 – Slow Lobe Power Balance Shift Due to Shim, Lobe Power Indicating System or Operator Failure with Operator Compensation
Section 15.4.7.13 – Withdrawal of all Shims and Safety Rods from 10-10 NF
Section 15.4.7.14 – Withdrawal of all Outer Shims from 10-10 NF with Failure of the Wide Range Subsystem
Section 15.4.7.15 – Withdrawal of All Outer Shims and Neck Shims from 10-5 NF with Failure of the Wide Range Subsystem
Section 15.4.7.16 – Rapid Regulating Rod Withdrawal with Failure of the Wide Range Subsystem
Section 15.4.7.17 – Withdrawal of All Outer Shims and Neck Shims from 10-5 NF Coupled with Rapid Regulating Rod Withdrawal at 10-3 NF
Section 15.4.7.18 – Driven Test Loop Blowdown with Experiment Hardware Failure
Section 15.4.7.19 – Voiding in All Inpile Tubes Due to a Simultaneous Flow Coastdown or Loss of Temperature Control in All Loops
Section 15.4.7.20 – Loss of Loop Temperature Control in the ATR High Temperature Loop Due to Heat Exchanger Failure or Line Heater Stuck Without a Loop Scram
Section 15.4.7.21 – Loop Flow Coastdown Without a Loop Scram
Section 15.4.7.22 – Pressure Tube Cooling Flow Bypass Due to a Maximum Useful Capacity Holder Test Train Failure Without a Loop Scram.

SAR-153 Section 15.9.3 "Polkinghorne calculated a conservative overpower limit for application of depressurized operation using the ATR version of RELAP5/MOD2.5 and ATR-SINDA Version B Revision 1 and SINDA-SAMPLE Version B Revision 1. The thermal analysis for steady-state depressurized operation with 3,600 gpm of emergency coolant flow and 125°F at the vessel inlet determined that depressurized operation could occur up to a power of 7.3 MW in an extreme 70/20 lobe power split while maintaining margin from CHF and FI of at least three standard deviations. Therefore, a large margin exist for depressurized operation at a power level of 500 kW before approaching core damage thresholds. The core power limit (7.3 MW) is applied in each of the reactivity insertion analyses for depressurized operation listed below."

"Table 15.9-2 provides various ramp reactivity insertion rates and the corresponding limiting initial and maximum core power levels. The maximum core power level as a function of the reactivity insertion ramp is also presented in Figure 15.9-1."

SAR-192 Section 8.4.6.1.3 "A transient analysis was performed using RELAP5 to quantify the transient response to an uncontrolled withdrawal of the OSCCs [outer shim control cylinders]. The withdrawal of the OSCCs was simulated by modeling a constant reactivity insertion rate equal to the peak reactivity insertion rate from Figure 8.5. The maximum reactivity insertion rate for all OSCCs simultaneous withdrawal (0.077$/s) was conservatively assumed."

SAR-192 Section 8.4.6.2.3 "An analysis was performed using RELAP5 to quantify the transient response to a 0.25$ reactivity step insertion."

SAR-192 Section 8.4.7.1.3 "A transient analysis was performed using RELAP5 to quantify the transient response to

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a simultaneous uncontrolled withdrawal of the OSCCs and four neck shim rods. The withdrawal of the OSCCs and four neck shim rods was simulated by modeling a constant reactivity insertion rate equal to the combined peak reactivity insertion rate from the OSCC pairs and four neck shim rods. The maximum reactivity insertion rate for all OSCC pairs and four neck shim rods simultaneously withdrawing (0.11.$/s) was conservatively assumed."

SAR-192 Section 8.4.7.2.3 "An analysis was performed using RELAP5 to quantify the transient response to a 0.75$ reactivity step insertion."

SAR-192 Section 8.4.8.1.3 "A transient analysis was performed using RELAP5 to quantify the transient response to air voiding of the LIPT [large inpile tube]. The voiding event was simulated by modeling a total reactivity of 1.54$ insertion into the core at a constant rate of 0.26$/sec. This represents the maximum possible reactivity insertion, and insertion rate, that can reasonably occur."

SAR-192 Section 8.4.8.2.3 "A transient analysis was performed using RELAP5 to quantify the transient response to an aluminum filler piece dropped into an LIPT. The filler drop event was simulated by modeling a rapid total reactivity of 1.2$ insertion. This bounds the measured worth of 1.18$ for a filler piece."

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 error in RELAP5 is in the reactor kinetics model. This error could affect the calculated core power for reactivity insertion accidents, i.e. accident scenarios in which core reactivity is increasedin an uncontrolled manner. RELAP5 is a thermal-hydraulic transient analysis computer code. RELAP5 is used to simulate the thermal-hydraulic response of a pressurized water reactor (PWR) system to hypothesized accident scenarios. RELAP5 is not used to establish the probability of accident sequences. RELAP5 is not used in any manner to control reactor operation. RELAP5 is not an initiator of any accident sequence. The error in the summation of the delayed neutron terms, or the poor logic in the transition from the steady state solution to the transient simulation, would not affect the assumed frequency or probability of occurrence of an accident.

2. Could the PISA increase the consequences of an accident previously evaluated in the safety basis?
Yes ☒ No ☐
Explain:
The error in the RELAP5 analysis computer code could result in under predicting the maximum power and the total energy deposition that were calculated in the accident safety analyses, and thus, under predicting the accident consequences. The preliminary calculations, however, using a corrected version of RELAP5 show that the energy deposition of the bounding ATR Condition 2, 3 and 4 accidents is essentially unchanged or lower, and the preliminary calculation provide assurance that ATR can operate with sufficient margin. The USQ determination, however, can not be based on preliminary results.

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:
For reactivity insertion accident scenarios the equipment important to safety are: a) fuel element cladding (primary barrier preventing the release of fission products) and structure (heat removal), b) the Reactor Shutdown System (RSS) (mitigation of reactivity insertion), c) ATR primary coolant system pumps and ATRC canal water level (heat removal), d) ATR primary coolant system pressure boundary (barrier preventing fission product release and heat removal path), and e) ATR heat exchangers and secondary (heat removal). The error in RELAP5 is in the reactor kinetics model. RELAP5 is a thermal-hydraulic transient analysis computer code and is used to simulate the

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thermal-hydraulic response of a PWR system to hypothesized accident scenarios. This error could affect the calculated core power for reactivity insertion accident. RELAP5 is not used to establish the probability of failure or malfunction of equipment. RELAP5 is not used in any manner to control equipment. RELAP5 is not an initiator of any malfunction of equipment. The results of RELAP5 simulations were not used to establish environmental conditions for the equipment design bases. Thus, the error could not result in an error in the assumed probability of occurrence of a malfunction of the identified equipment important to safety.

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 equipment important to safety are: a) fuel element cladding (primary barrier preventing the release of fission products) and structure (heat removal), b) the Reactor Shutdown System (RSS) (mitigation of reactivity insertion), c) outer shim control cylinders, neck shim rods and regulating rods (reactivity control), d) ATR primary coolant system pumps and ATRC canal water level (heat removal), e) ATR primary coolant system pressure boundary (barrier preventing fission product release and heat removal path), and f) ATR heat exchangers and secondary (heat removal). The error in RELAP5 is in the reactor kinetics model. RELAP5 is a thermal-hydraulic transient analysis computer code and is used to simulate the thermal-hydraulic response of a PWR system to hypothesized accident scenarios. This error could affect the calculated core power for reactivity insertion accident. The error could result in under predicting the consequence of the malfunction of the reactivity control systems and the RSS as currently described in the accident safety analyses.

The accident analyses are used to verify the effectiveness of RSS setpoints, the error could result in RSS setpoints that are too high to be as effective as currently demonstrated by the safety analyses. Higher RSS setpoints would result in reductions in the margin between the analytical results and the ATR Plant Protection Criteria and the ATRC core power Safety Limit. But as demonstrated by the preliminary calculation with a corrected RELAP5 version, the expected magnitude of the effect of the error is minimal; so that, the higher RSS setpoints would probably have a minimal affect on the consequences of the analyzed 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:

RELAP5 is a thermal-hydraulic computer accident analysis code used to simulate the integrated thermal-hydraulic response for hypothesized accident scenarios in PWR. The error in RELAP5 is in the reactor kinetics model. This error could affect the calculated core power for reactivity insertion accident. Reactivity anomalies are analyzed accidents in SAR-153 and SAR-192. RELAP5 is not used in any real-time reactor control mechanisms. RELAP5 is not an initiator of any accident sequence. The error in RELAP5 accident analysis code would not create a condition for an accident of a different kind than analyzed in the safety basis accident analyses.

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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 equipment important to safety are: a) fuel element cladding (primary barrier preventing the release of fission products) and structure (heat removal), b) the Reactor Shutdown System (RSS) (mitigation of reactivity insertion), c) outer shim control cylinders, neck shim rods and regulating rods (reactivity control), d) ATR primary coolant system pumps and ATRC canal water level (heat removal), e) ATR primary coolant system pressure boundary (barrier preventing fission product release and heat removal path), and f) ATR heat exchangers and secondary (heat removal). RELAP5 is a thermal-hydraulic computer accident analysis code used to simulate the integrated thermal-hydraulic response for hypothesized accident scenarios in PWR. The error in RELAP5 is in the reactor kinetics model. This error could affect the calculated core power for reactivity insertion accident. RELAP5 was used to simulate the reactivity anomaly accident sequences, such as withdrawal of reactivity controls rods or shims and voiding of an experiment position in the core. The RELAP5 analyses were also used to verify the effectiveness of the RSS setpoints. The RELAP5 accident analyses were not used to eliminate equipment from the possible important to safety equipment identified in the safety analyses.

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:

Since the accident analyses are used to verify the effectiveness of RSS setpoints, the error could result in RSS setpoints that are too high to be as effective as currently demonstrated by the safety analyses. Higher RSS setpoints would result in reductions in the margin between the analytical results and the ATR Plant Protection Criteria and the ATRC core power Safety Limit.

However, for the ATR Critical Facility, the energy deposition acceptance criterion is 15 MJ, which is a Technical Safety Requirement Safety Limit, SL 2.192.1, Core Energy Deposition. The energy deposition is based on the amount of energy to raise the fuel element cladding temperature to 855 K, which is treated as the threshold clad melt temperature. The energy deposition need to raise the fuel cladding temperature to 855 K was calculated to be 30 MJ. The allowable energy deposition was chosen to be half that value or 15 MJ. As shown in preliminary calculations using a corrected REALP5 version, the results do not approach 15 MJ. The USQ determination, however, can not be based on preliminary results.

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 error discovered in the RELAP5 the reactor kinetics model could affect the consequences of accidents analyzed and of the failure or malfunction of equipment important to safety analyzed in the safety basis analyses. This error could result in the decrease in the margin of safety for the ATR and the ATR Critical Facility.

This error does not affect the probability of occurrence of accidents or the occurrence of the malfunction of equipment important to safety analyzed in the safety basis accident analyses. This error does not create the possibility of malfunction of equipment or create the possibility of a failure of a different type.

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

INL USQ PROCESS
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APPROVAL:

J. C. Chapman USQ Evaluator Print/Type Name	John C. Chapman USQ Evaluator Signature	11 Aug 2008 Date
M. B. McDonough Nuclear Facility Manager Print/Type Name	M. B. McDonough Nuclear Facility Manager Signature	11 AUG 2008 Date

CONCURRENCE:

J. A. Jacobi Independent Review Committee Chair Print/Type Name	Jack A. Jacobi Independent Review Committee Chair Signature	11 Aug 2008 Date

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Rev. 02
Use with LWP-18001

INL USQ PROCESS REASONABILITY DETERMINATION/POTENTIAL INADEQUACY IN THE SAFETY ANALYSIS (PISA) FORM

(DO NOT USE FOR PROPOSED CHANGES)

Facility or Activity: ATR and ATR Critical Facility

USQ Process No.: RTC-USQ-2008 553

Subject: Error in RELAP5 Reactor Kinetics Model

Describe the New Information/Discovery:
It has been discovered that there is an error in the reactor kinetics model in the RELAP5 computer code. RELAP5 is the thermal-hydraulic accident analysis code used for the safety basis accident analyses for the Advanced Test Reactor (ATR) and the ATR Critical Facility. The error is in the summation of the delayed neutron terms. Also poor coding logic was used in the switch between the steady-state and the transient solutions. This error and poor coding logic, henceforth simply called "error," could result in maximum core power and energy deposition being greater than calculated in the safety basis analyses. This error is present in the ATR version of RELAP5/MOD2.5 (used for ATR and ATR Critical Facility accident analyses) and Version 3.2.1.2 of RELAP5/MOD3 (used for ATR accident analyses).

Identify the applicable safety basis document(s) (e. g., DSA, SAR, BIO, TSRs, etc.):
SAR-153 Upgraded Final Safety Analysis Report for the Advanced Test Reactor
TSR-186 Technical Safety Requirements for the Advanced Test Reactor
SAR-192 Safety Analysis Report for the Advanced Test Reactor Critical Facility (ATRC)
TSR-192 Technical Safety Requirements for the Advanced Test Reactor Facility (ATRC)

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
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.

Since this error in the kinetics model is present in version RELAP5/MOD2.5, which was used for both ATR and ATR Critical Facility accident analyses, and Version 3.2.1.2 of RELAP5/MOD3, which was used for ATR accident analyses, this error could result in maximum core power and energy deposition being greater than calculated in the safety basis analyses.

A PISA exists. Complete the following actions

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- 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.

ATR was in two pump operation, Cycle 142B, at the time of discovery. The maximum effective plate power (EPP) for this operating cycle was 253 MW. The safety basis analyses for two primary coolant pump (PCP) operation assumed that the maximum EPP was 417 MW. Based on preliminary calculations using a version of RELAP5 with this error corrected, the potential error was predicted to only have minimal effect. Considering the large margin between the maximum EPP assumed the safety basis analyses and the actual cycle EPP (417 MW vs 253 MW), no immediate action was need to place the ATR in a safe condition. The ATR Critical Facility was not operating at the time of the discovery.

As noted above, the error in RELAP5 was corrected and scoping calculations were performed to assess the effect of this error on various accidents. The bounding Condition 2, 3 and 4 reactivity insertion accidents for ATR were recalculated using the corrected version of RELAP5. As seen in the attached figures, the energy deposition using the corrected RELAP5 version essentially overlays the results of the safety analysis version for the Condition 2 and 4 bounding faults (Figures 1 and 3) and is less than the results of the safety analysis version for the Condition 3 bounding fault (Figure 2). A reactivity ramp fault, withdrawal of all outer shims and neck shims with the failure of the Wide Range System, a bounded event, was also recalculated using the corrected RELAP5 version. The core power response calculated using the corrected version essentially overlays the result of the safety analysis version (Figure 4).

The effect of this error on the generic Condition 4 experiment loop voiding analyses was also evaluated. The generic experiment loop analyses were performed to establish an envelope for loop experiments. If a loop experiment is not within the established envelope, an experiment specific voiding analysis must be performed to ensure that voiding of a particular loop experiment meets the ATR Plant Protection Criteria. For both the Standard inpile tube (IPT) and the Large IPT loop experiment voiding faults, the maximum energy depositions using the corrected RELAP5 version remained less than the Chapter 15 safety analysis. Thus, the generic Condition 4 analyses will remain bounded by SAR-153 safety analysis. The corrected version calculated energy depositions, however, exceeded the previously calculated generic Condition 4 loop voiding analyses (Figures 5 and 6); but, as stated above, will remain bounded by the energy deposition assumed in the SAR-153 accident analyses.

Of particular concern is the accident analysis for one of the experiments currently in the reactor (2E-NW-158). The maximum void worth of the IPT during Experiment 2E-NW-158 could potentially be greater than that allowed in the ATR SAR. Therefore, per SAR Section 10.2.6.4.1, an analysis was performed to demonstrate that the consequences of an accident with this experiment inserted are acceptable. (Exhibit 1 is the cover page of ECAR-208, Inpile Tube Voiding Analysis for ATR Experiment 2E-NW-158.) The void worth of the IPT was assumed to be 1.10$ instead of 1.00$ (i.e., the SAR limit for the Large IPT). The safety analysis (using RELAP5 version with the error) showed that the ATR Plant Protection Criteria are met. With the error in RELAP5 corrected, however, thermal safety margins for the limiting Condition 4 reactivity fault (>½-in. experiment loop pipe break) may be less than demonstrated to be acceptable in SAR Chapter 15.

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Figure 7 shows that the energy deposition calculated using the corrected RELAP5 version exceeds that calculated in ECAR-208. The safety analysis for Experiment 2E-NW-158, ECAR-208, however, is very conservative. The maximum effective plate power (EPP) was 443 MW and a step reactivity insertion of 0.10$ was included in the RELAP5 calculation to conservatively bound the effect of test train failure. (NW lobe power assumed for this analysis was 34 MW.) The Core Safety Assurance Package (CSAP) for the current cycle shows that the maximum EPP is 253 MW and that failure of the test train during Experiment 2E-NW-158 would not cause a reactivity insertion. Figure 7 shows the calculated integrated core power vs. time (i.e., core energy deposition) for a >½-in. loop pipe break for Experiment 2E-NW-158, with and without the RELAP5 error correction and with and without a 0.10$ step insertion due to test train failure. The calculation with the corrected RELAP5 code and without a 0.10$ step is bounded by the accident analysis in ECAR-208. In addition, the calculation assumed the NW lobe power was 34 MW. (The nominal north-west (NW) lobe power for this cycle is 23 MW; the maximum NW lobe power is 26 MW.) The ATR is currently operating with two primary coolant pumps (PCPs) and a maximum EPP of 253 MW. The EPPs considered in Chapter 15 safety analysis are considerably greater than 253 MW; (i.e., 417 MW for 2-PCP operation and 443 MW for 3-PCP operation). The recalculated energy deposition using the corrected RELAP5 version with the assumed NW lobe of 26 MW would be bounded by the accident analysis in ECAR-208. Based on the relatively low power at which the reactor is operating and several scoping calculations, the error in RELAP5 will not result in exceed the ATR Plant Protection Criteria.

The effect of the RELAP5 error on ATR depressurized operations was evaluated by recalculating, using the corrected version, the enveloping 0.30$/sec ramp insertion which is the basis for comparing the accident analyses to the ATR Plant Protection Criteria. (Exhibit 2 is the cover page of the EDF TRA-ATR-1835, Reactivity Accident Analysis for ATR Depressurized Operation.) The recalculated maximum power is approximately 0.1 MW higher than the safety analysis calculation (Figure 8); the recalculated maximum core power, however, is well below the safety basis limit of 7.3 MW, which was demonstrated to meet the ATR Plant Protection Criteria.

Reactive insertion accidents for the ATR Critical Facility (SAR-192) were also recalculated using the corrected RELAP5 version. The effect on Condition 4 Large IPT Voiding and the Filler Piece Drop accidents were evaluated. The recalculated maximum power and the energy deposition (integrated core power) exceed the calculated safety analysis RELAP5 version values for the Larger IPT voiding analysis (Figures 9 and 10) and essentially overlay the Filler Piece Drop accident analysis (Figures 11 and 12). The ATR Critical Facility accident acceptance criterion, however, is an energy disposition of less than or equal to 15 MJ. As shown in Figures 10 and 12, the recalculated energy deposition for these Condition 4 accidents is well below the ATR Critical Facility acceptance criterion.

Although the error discovered in the RELAP5 versions used for safety analyses apparently caused the maximum core power and energy deposition to be greater than some of the previous analyses, based on the scoping calculations that used the corrected RELAP5 version, this error will not challenge the conclusions of either the ATR or the ATR Critical Facility safety basis. Therefore, no interim operating restrictions or controls are required for either ATR or ATR Critical Facility operations.

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J. C. Chapman	J C Chapman	24 July 2008
Safety Analyst Print/Type Name	Safety Analyst Signature	Date
M. B. McDonough	E Schuebert for	7/24/08 1607
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. USQ 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:

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:

INL USQ PROCESS
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USQ Process No.: RTC-USQ-2008 553

Subject: Error in REAP5 Reactor Kinetics Model

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

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:

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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:

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

RELAP5/MOD2.5 ATR Version

Limiting SAR Condition 2 Fault (SLIPT MUCH)

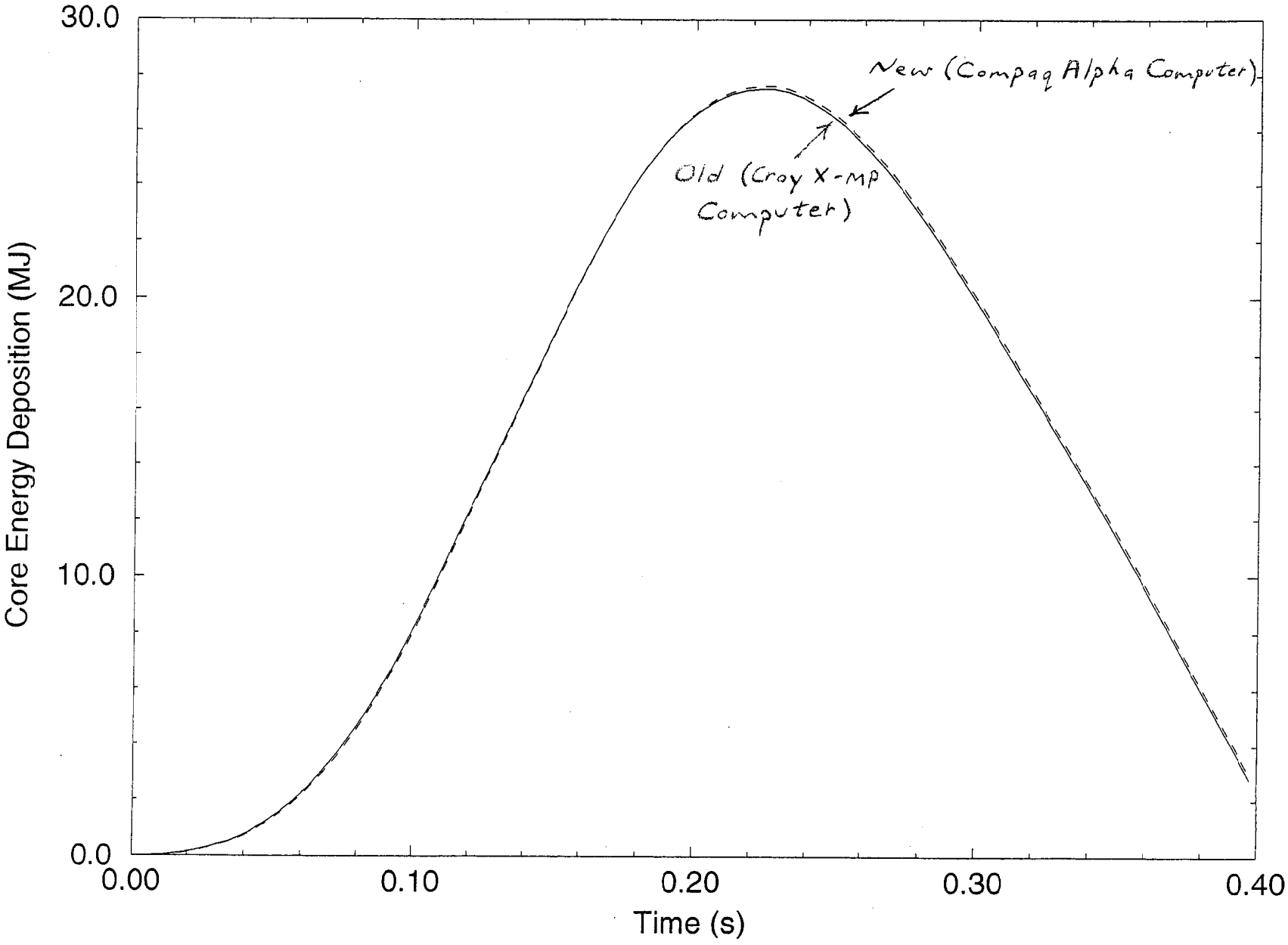

Figure 1

RELAP5/MOD2.5 ATR Version

Limiting SAR Condition 3 Fault (0.50$ step)

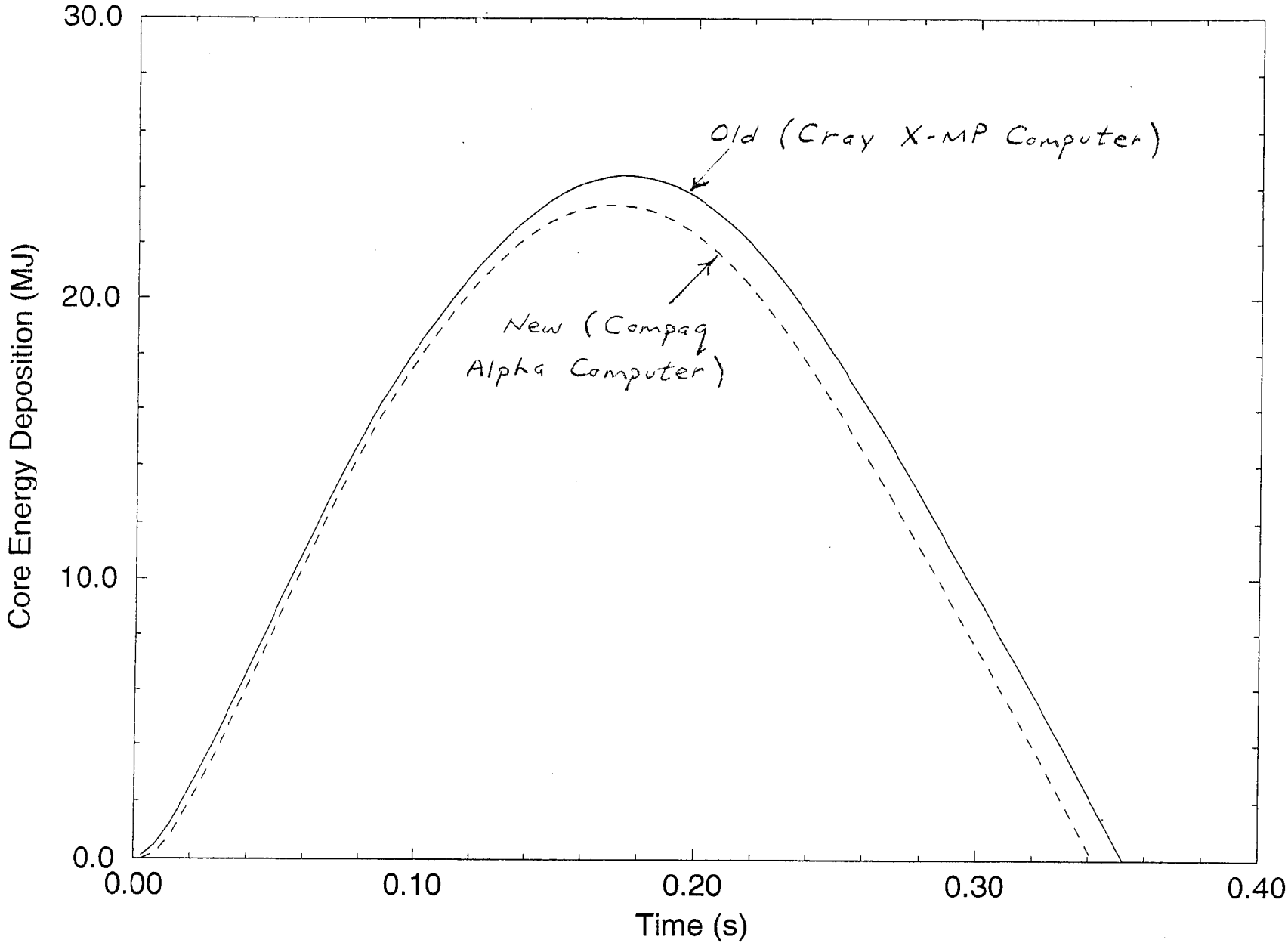

Figure 2

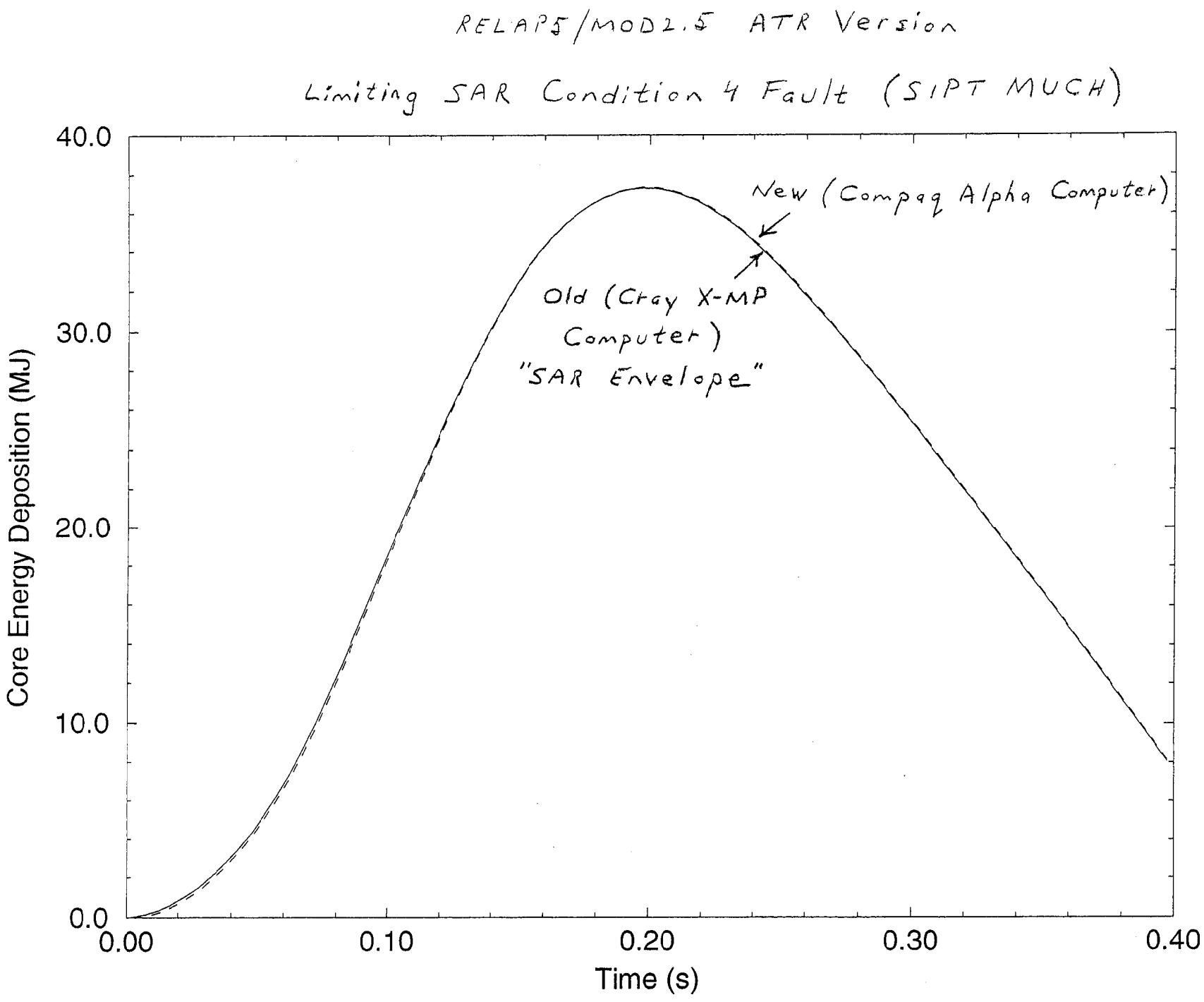
RELAP5/MOD2.5 ATR Version
Limiting SAR Condition 4 Fault (SIPT MUCH)
New (Compaq Alpha Computer)
Old (Cray X-MP Computer)
"SAR Envelope"
Core Energy Deposition (MJ)
40.0
30.0
20.0
10.0
0.0
0.00
0.10
0.20
0.30
0.40
Time (s)

Figure 3

RELAP5/MOD2.5 ATR Version

0.079 $/s from 10^{-10} N_F (Bounded Event)

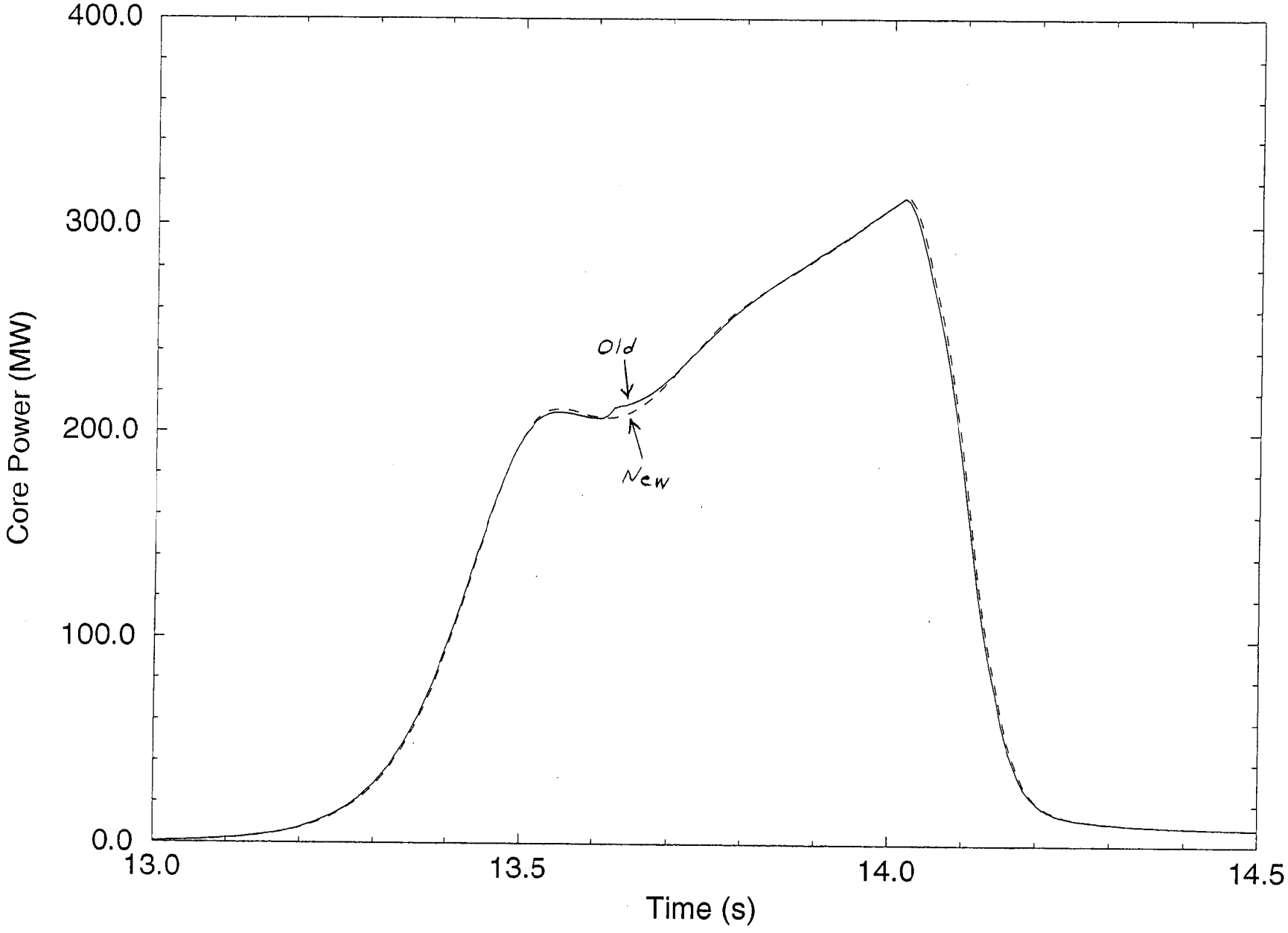

Figure 4

RELAP5/MOD3 Version 3.2.1.2

SIPT Condition 4 Fault

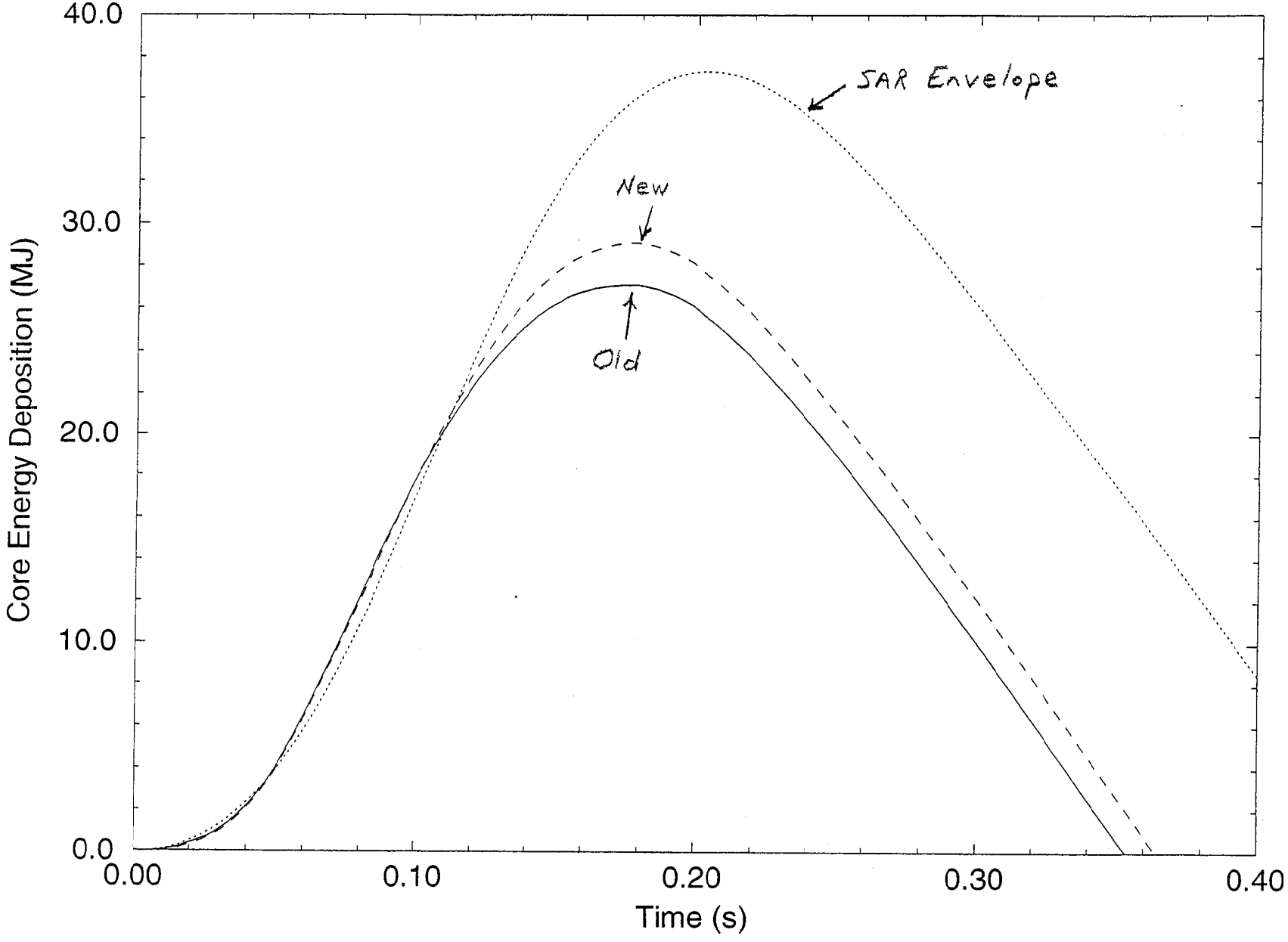

Figure 5

RELAP5/MOD3 Version 3.2.1.2

LIPT Condition 4 Fault

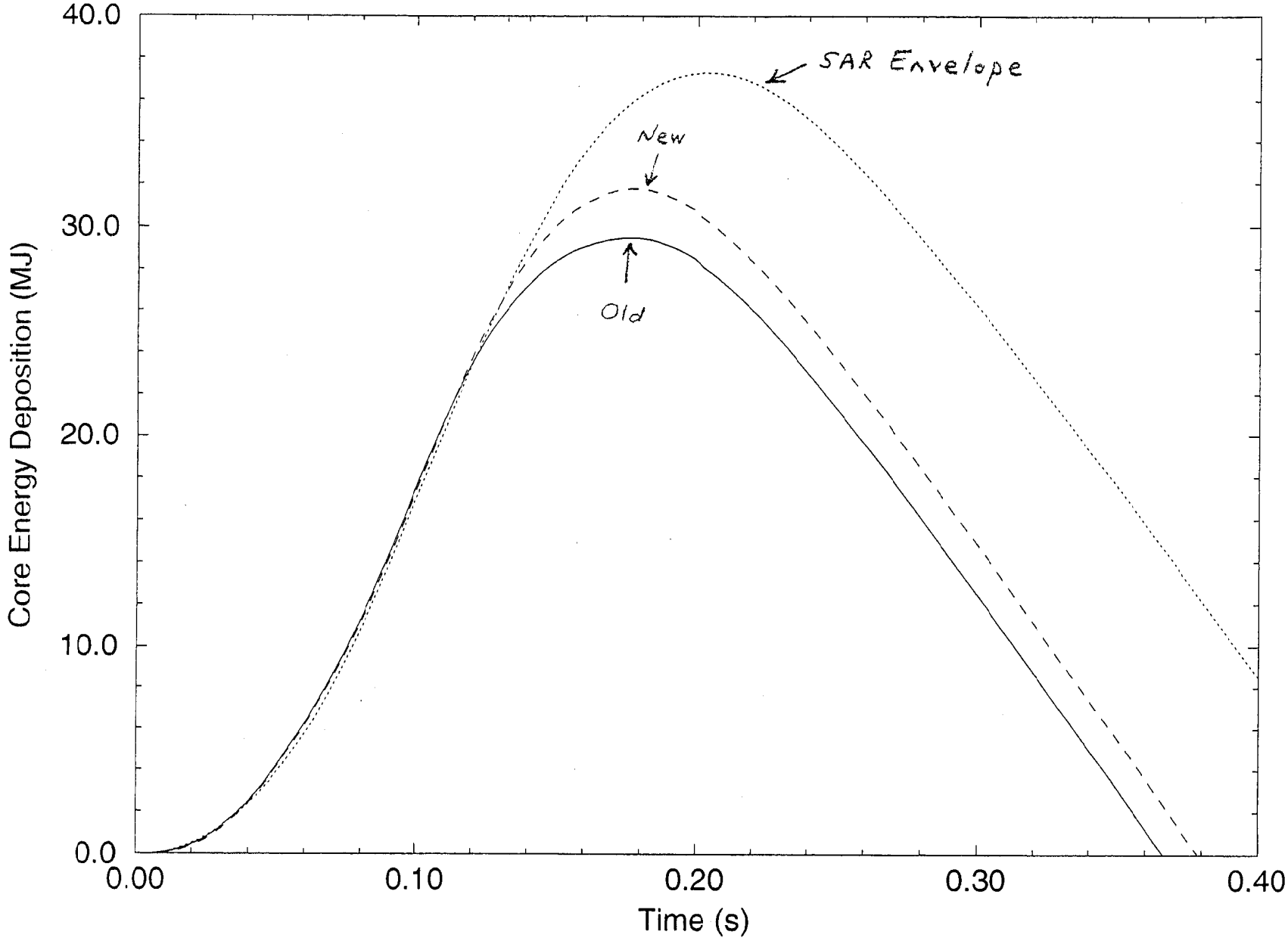

Figure 6

ask Jim A. about 10¢ step. relavency

RELAP5/MOD3 Version 3.2.1.2

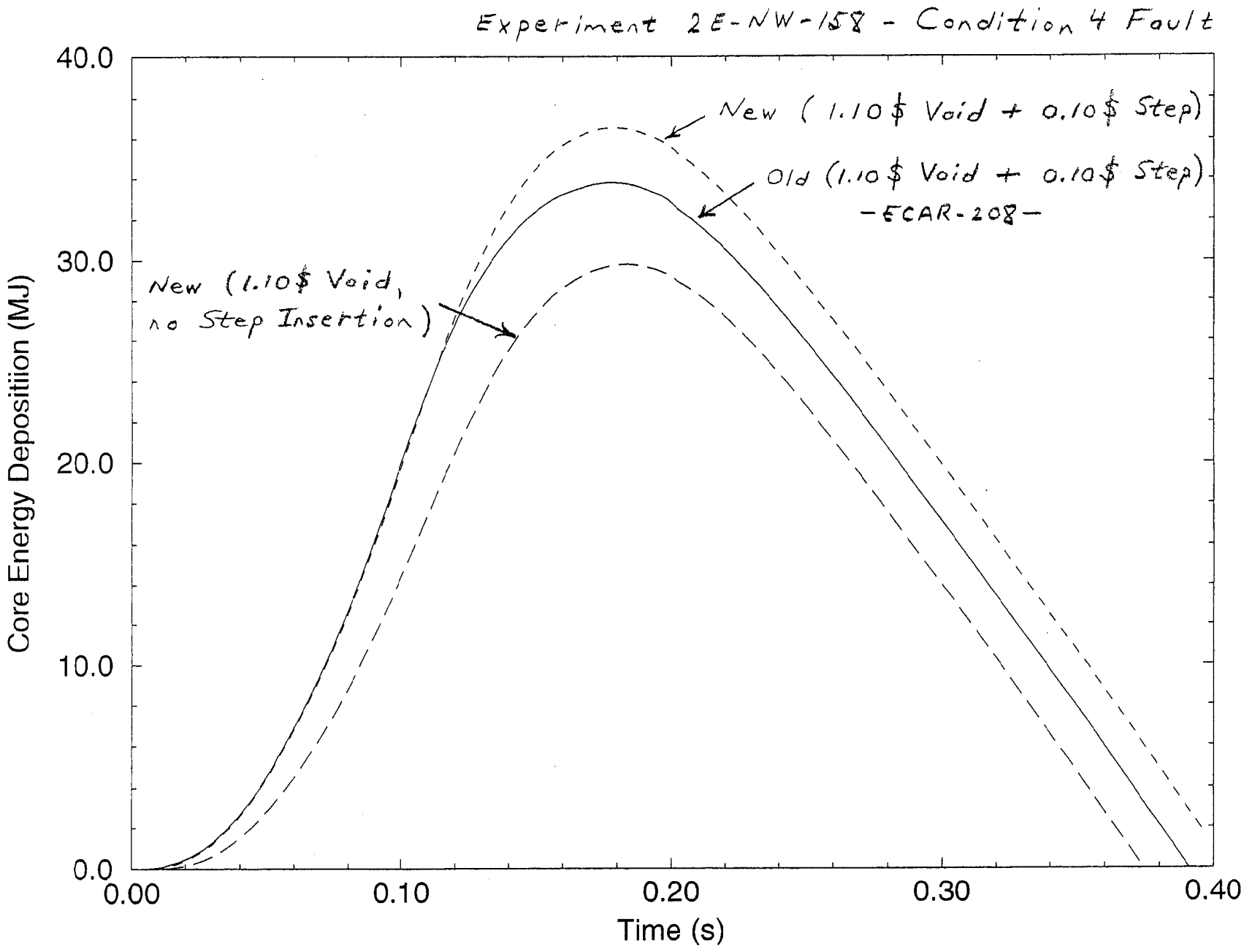

Figure 7

RELAP5/MOD 2.5 ATR Version

Depressurized Operation - 0.30 $/s

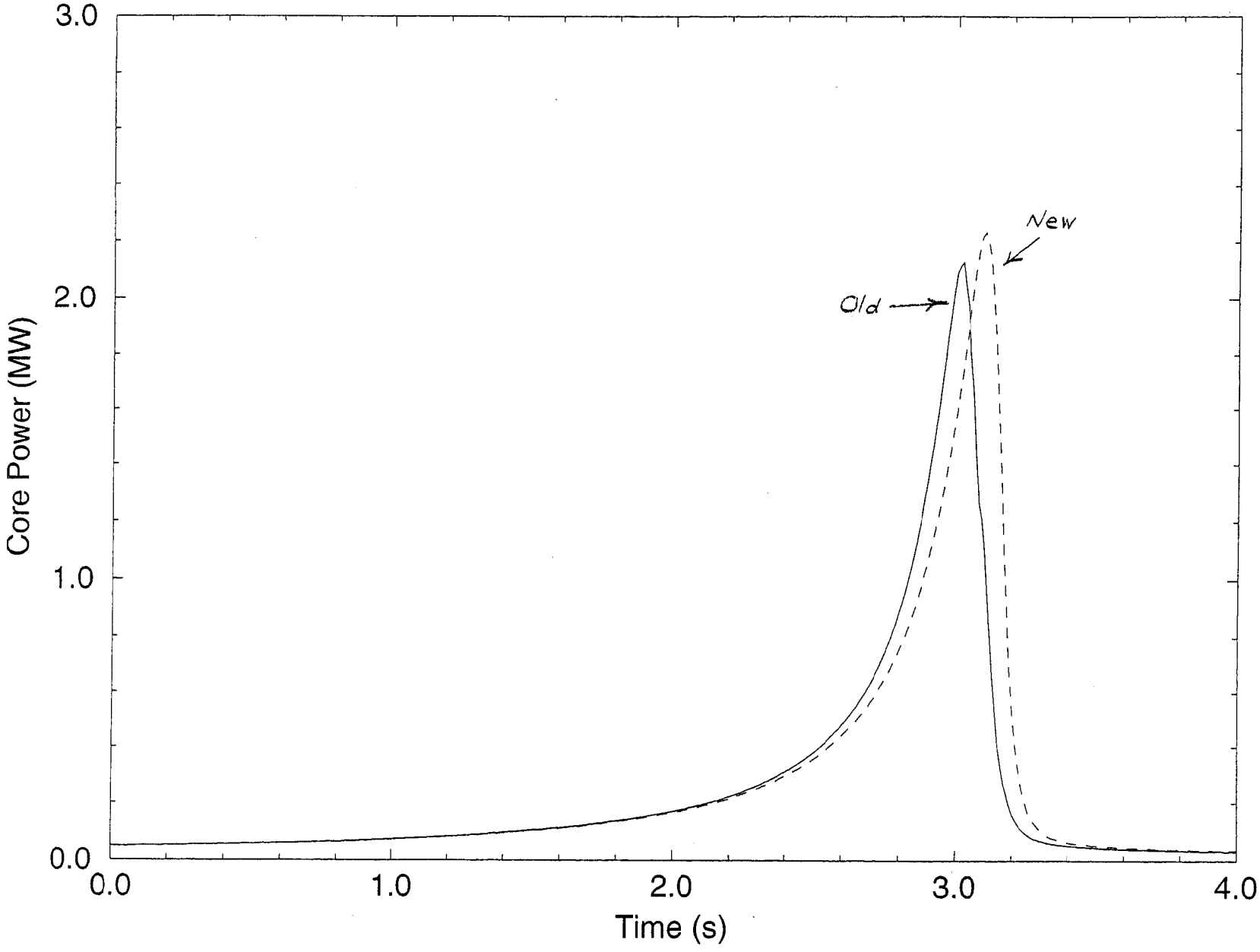

Figure 8

RELAP5/MOD2.5 ATR Version

ATRC LIPT Air Voiding Accident

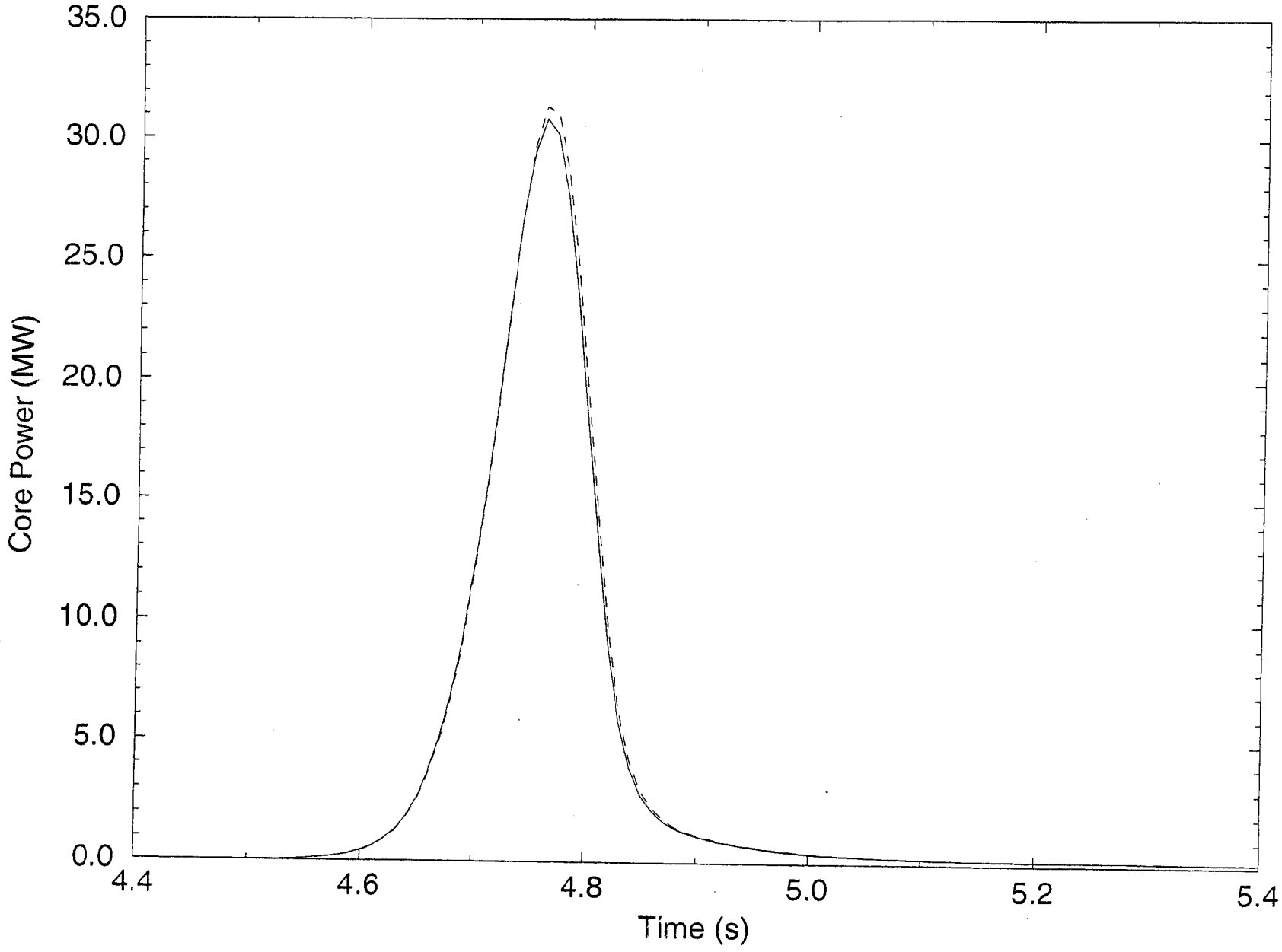

Figure 9

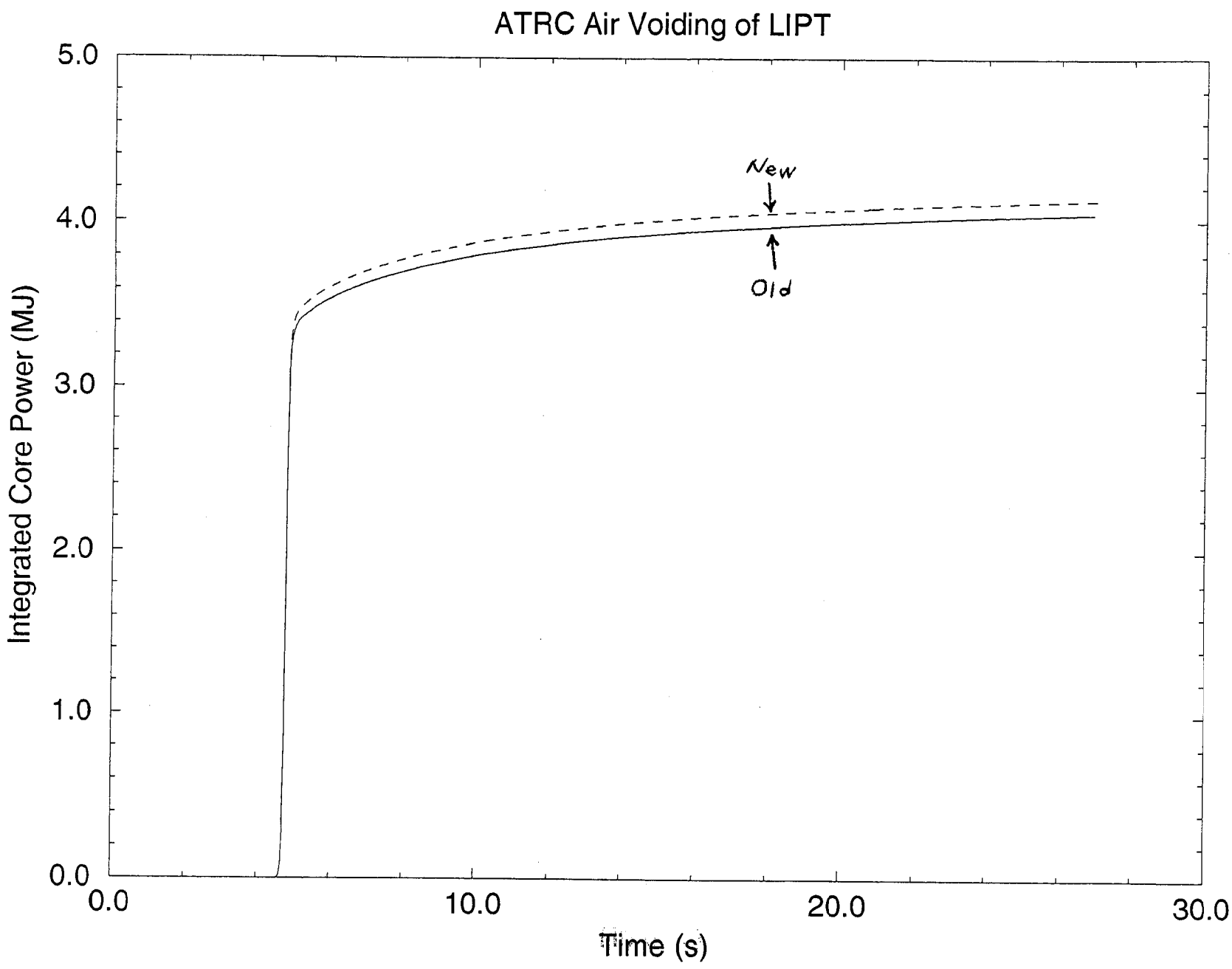
ATRC Air Voiding of LIPT
Integrated Core Power (MJ)
5.0
4.0
3.0
2.0
1.0
0.0
0.0
10.0
20.0
30.0
Time (s)
New
Old

Figure 10

RELAP5/MOD2.5 ATR Version

ATRC Filler Piece Drop Accident

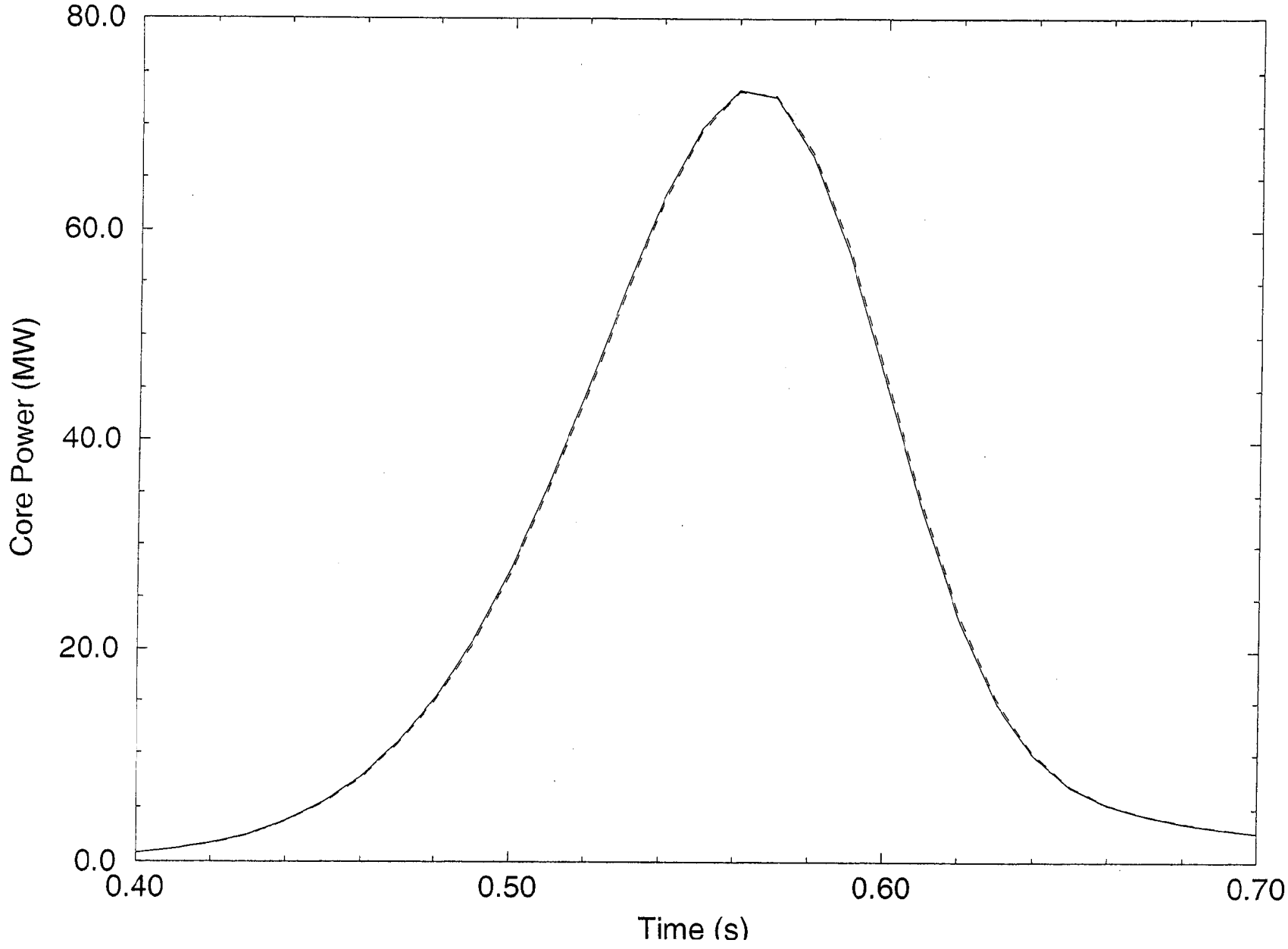

Figure 11

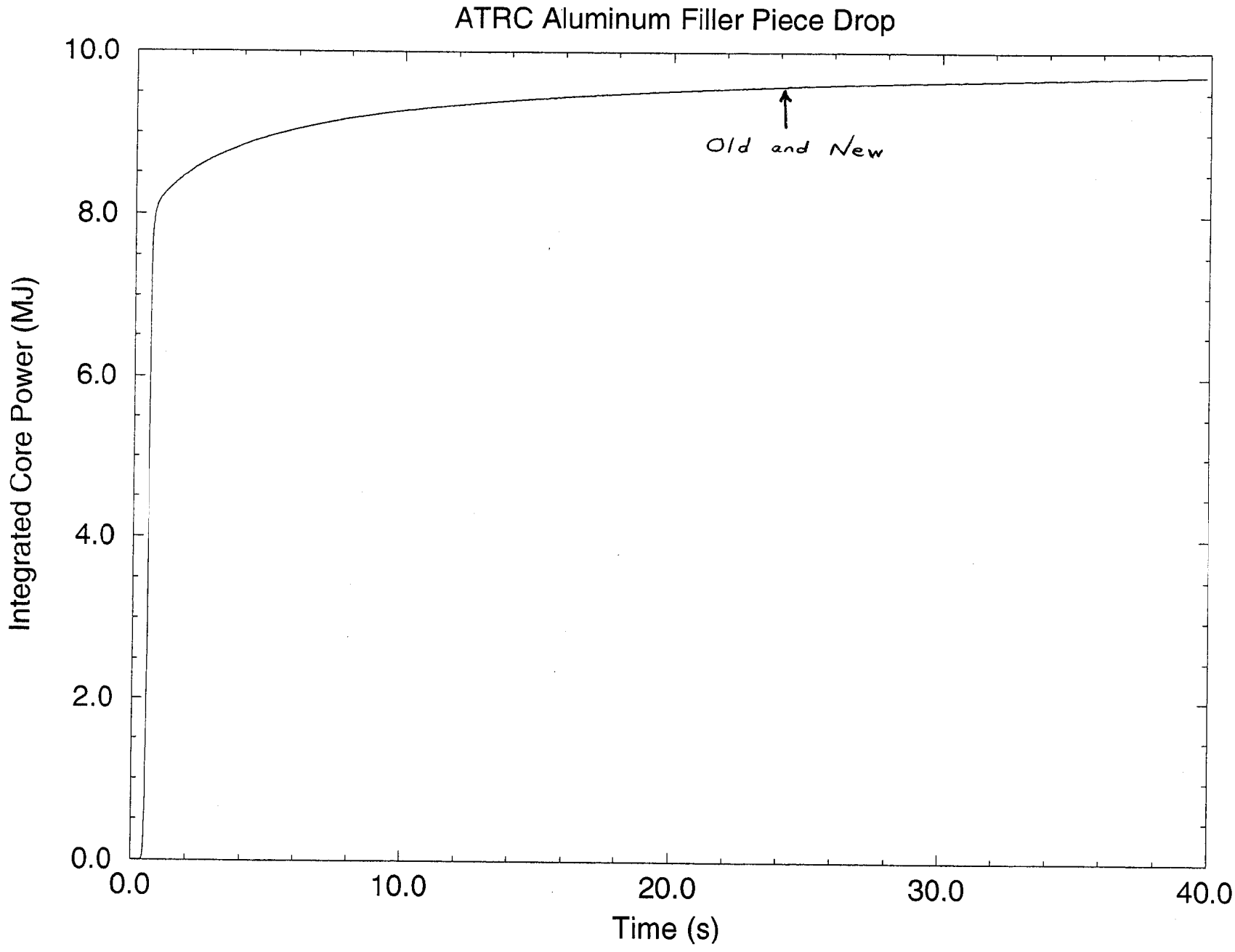
ATRC Aluminum Filler Piece Drop
Integrated Core Power (MJ)
10.0
8.0
6.0
4.0
2.0
0.0
0.0
10.0
20.0
30.0
40.0
Time (s)
Old and New

Figure 12

ENGINEERING CALCULATIONS AND ANALYSIS REPORT

ECAR No.: 208 ECAR Rev. No.: 0 Project File No.: Date: 4/14/08

Title: Inpile Tube Voiding Analysis for ATR Experiment 2E-NW-158

1. Index Codes

Building/Type: TRA-670 SSC ID: Site Area: RTC

2. Quality Level: 1 *(Required Element)*

3. Objective/Purpose

The void worth of the inpile tube (IPT) during ATR Experiment 2E-NW-158 may be greater than allowed in Section 10.2.6.2.2 of the ATR Safety Analysis Report (SAR). Therefore, per SAR Section 10.2.6.4.1, an analysis is required to demonstrate that the consequences of an accident during this experiment are acceptable. The limiting Condition 2 and 4 IPT decompression accidents discussed in Chapter 15 of the SAR (≤½-in. and >½-in. experiment loop pipe breaks, respectively) were analyzed specifically for Experiment 2E-NW-158 assuming that the void worth of the IPT was 1.10$ instead of 1.00$ (i.e., the SAR limit for the Large IPT).

4. Conclusions/Recommendations

It is shown that Experiment 2E-NW-158 can be conducted safely if experiment operation is limited as follows:

Loop flow rate	≥100 gpm
Flow split	≥56:44
IPT inlet pressure	1750 to 2250 psig
Maximum IPT coolant temperature	$<T_{sat}$
NW lobe power	≤34 MW
Test fission power	≤200 kW
Fission power over 9.6 in.	≤56 kW
Water fraction inside shrouds	≥0.30
IPT void worth	≤1.10$
Reactivity insertion due to test hardware failure	≤0.10$

5. Review (R) and Approval (A) and Acceptance (Ac)[1]:

		Typed Name/Organization	Signature/Date[2]
Performer/Author		S. T. Polkinghorne / G121	Steve Polkinghorne 4/10/08
Technical Checker	R	P. A. Roth / G120	Pages checked: All 4/14/08
Independent Peer Reviewer[3]	R	G. L. Sharp / G125	4/14/08
Performer's Manager	A	L. Harrison / G121	4/15/08
Nuclear Safety[3]	Ac	J. C. Chapman / G120	15 Apr 2008
Document Control		Laura Smith	4/16/08

1 Review and Approval are required. See LWP-10200 for definitions and responsibilities.
2 An Electronic Change Request (ECR) indicating final review and concurrence by the listed individuals can be used in lieu of signatures.
3 If Required, per LWP-10200.

6. Additional Distribution: (Name and Mail Stop)	Document Control: Reactor Programs Document Management (J. M. Lewis) (original + 1); CSAP, MS 7136; A. W. LaPorta, MS 7136; SORC, MS 7114; T. M. Stumpf, MS 7101

Exhibit 1

ENGINEERING DESIGN FILE

1. Title: Reactivity Accident Analysis for ATR Depressurized Operation

2. Project File No.: N/A

3. Index Codes:
   Building/Type TRA-670 SSC ID N/A Site Area 530

4. Summary:

The RELAP5 computer code was used to analyze several reactivity insertion accidents assumed to occur during ATR depressurized operation. The accidents analyzed were: (1) ramp reactivity insertions with insertion rates ranging from 0.04 to 0.30$/s, (2) rapid withdrawal of the regulating rod (0.57$ in 0.543 s), and (3) one-inch drop of a perched fuel element (0.10$ step). The worst initial power was determined for each accident. In all cases, a high-neutron-level scram was assumed to occur 0.035 s after the power reached 2.3N_L (1.15 MW). The results were as follows:

Accident	Initial Core Power (kW)	Maximum Core Power (MW)
Ramp reactivity insertion		
0.04$/s	0.73	1.359
0.10$/s	10.34	1.513
0.15$/s	20.68	1.647
0.20$/s	30.69	1.788
0.25$/s	40.05	1.946
0.30$/s	48.80	2.130
Rapid reg-rod withdrawal	500.0	1.318
Perched fuel element drop	500.0	1.150

The maximum power in each case was well below the steady-state power that would result in three standard deviation (3σ) margins to flow instability (FI) or the critical heat flux (CHF) (i.e., 7.3 MW). Therefore, the ATR protective criteria for Condition 2 faults (≥3σ to FI or CHF) are satisfied, even though the accidents shown above are less-probable Condition 3 or 4 faults.

A RELAP5 calculation was also performed to determine core power as a function of time following reactor scram, assuming that the reactor had operated for 2 hrs at 500 kW (N_L). By 10 minutes, the power had decreased to 1.36% of the initial power, and by 20 minutes, the power had decreased to 0.983%.

The results of these calculations will be used in ATR SAR Section 15.9, "Depressurized Operations Events," which is currently being updated.

5. Review (R) and Approval (A) and Acceptance (Ac) Signatures:
   (See instructions for definitions of terms and significance of signatures.)

	R/A	Typed Name/Organization	Signature	Date
Performer		S. T. Polkinghorne / Nuclear Systems Safety Analysis	Steve Polkinghorne	10/28/02
Checker	R	C. B. Davis / Nuclear Systems Safety Analysis	CB Davis	10/31/02
Requester	Ac	G. L. Sharp / TRA Nuclear Engineering	Dwayne Sharp	11/07/02
Approver	A	R. T. McCracken / TRA Nuclear Engineering	R.T. McCracken	1-13-03

6. Distribution: (Name and Mail Stop)
   Reactor Programs Document Management (original + 1); R. G. Ambrosek, MS 3885; C. B. Davis, MS 3890; D. E. Hale, MS 7136; R. T. McCracken, MS 7136; S. T. Polkinghorne, MS 3885; G. L. Sharp, MS 7136; SORC, MS 7114; S. R. Wagoner, MS 7101; J. R. Wolf, MS 3890

Doc Control Ac R.K. McCornell Lura Kay McCornell 1/29/03

Exhibit 2