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

(DO NOT USE FOR PROPOSED CHANGES)

Facility or Activity: ATR

USQ Process No.: ATR Complex USQ-2010-537

Subject: Notification of Error in RELAP5/MOD3 Critical Heat Flux Lookup Table

Describe the New Information/Discovery:

As documented in the P. P. Cebull, Letter to D. T. McDaniel, Notification of Error in RELAP5 Critical Heat Flux Lookup Table, June 21, 2010, errors were found in the Groeneveld critical heat flux lookup table implemented in RELAP5/MOD3 code. This code error is present in the version of RELAP5/MOD3 used in some ATR safety analyses.

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

ATR SAR 153. Most ATR accident analyses were done using the RELAP5/MOD2.5 version of the code. ATR SAR 153 Chapter 15 safety analysis utilizes the RELAP5/MOD3 code for analysis in support of accidents identified in SAR Table 15.0-8. Accidents addressed are SAR Section 15.4.4 and 15.4.6, the Blowdown of an In-pile Tube, and a subset of some Loss of Coolant Accidents (LOCA) addressed in SAR Section 15.6. RELAP5/MOD3 was also used to derive the SIPT and LIPT operating limits discussed in SAR Section 10.2. RELAP5/MOD3 was also used in development of the severe accident analysis described in SAR 153 section 15.12.11. As the severe accident analysis work is beyond design basis it will not be addressed.

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.

SAR Sections 10.2 and 15.4:

Versions of RELAP5/MOD3 were used in the analyses described in SAR Sections 10.2 to derive the SIPT and LIPT operating limits, and in Sections 15.4.4 and 15.4.6, which correspond to inpile tube decompressions due to failures of ½-inch piping and > ½-inch piping, respectively. TRA-ATR-850 calculated the reactivity insertion due to the loop voiding which TRA-ATR-841 used to calculate the power history for input to ATR-SINDA. TRA-ATR-850 stated that subcooled boiling was the primary mechanism of heat transfer on the surfaces of the experiment assembly and pressure tube, but that some localized film boiling occurred for short periods of time. Since nucleate boiling maximizes the heat transfer into the fluid, it also maximizes inpile tube voiding and the corresponding excursion in reactor power. Errors in the CHF calculation have the potential to affect the reactor power excursion because the CHF model affects the transition between nucleate and film boiling.

SAR Figures 15.4-3, 15.4-4, 15.4-15, and 15.4-16 indicate that the period of interest for the loop blowdown events is less than 0.25 s, after which time the reactor power is less than the initial value. A review of the analysis described by EDF 4520 showed that the inpile tube pressures varied between about 1750 and 1170 psia (12,100 to 8070 kPa) for the ½-inch break and between 1750 and 950 psia (12,100 to 6580 kPa) for the >1/2-inch break (The specific cases reviewed were from input files siptsys.fiss5.sb2.i and siptsys.noslip.i for the small and large breaks, respectively). The fuel specimens remained in nucleate boiling during the period of interest for both breaks. However, portions of the pressure tube experienced transition boiling for short periods of time during the latter portion of the period of interest. The heat transfer mode from the fuel specimens transitioned from nucleate to film boiling for the larger break cases when all 200 kW of fission power was generated over a single 9.6-inch segment instead of distributed over the entire length of the core. However, EDF 4520 concluded that the transition to film boiling did not significantly affect the power excursion.

The CHF table in RELAP5/MOD3 contains over 5000 CHF values, of which 15 are considered to be in error (P. P. Cebull, Letter to D. T. McDaniel). Only four points in the CHF table, out of 588 in the pressure range of interest,

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were in error for the loop blowdown events. The corrected CHF value was greater than the original value contained in the table for each of the four points of interest. The maximum error was 7.0%. An increase in the CHF will not affect the results for those cases where the experiment and pressure tube surfaces remain in nucleate boiling. The heat transfer could increase for those cases in which the surfaces experienced a transition to film boiling. However, the effect is expected to be small because the film boiling was localized and only occurred for short periods of time during the latter portion of the period of interest. Furthermore, EDF 4520 results indicated that the effect of a localized transition to film boiling was not significant.

Preliminary evaluations were performed using a corrected version of RELAP5/MOD3 for four cases involving SIPT and LIPT large and small break LOCAs. The results of the scoping calcuations indicate that the maximum core energy (MJ) deposition with the corrected version is identical to or differs in the fifth significant digit when compared to the existing calcuations documented in ECAR-402. Therefore, it is concluded that the errors in the CHF table will not affect the results presented in the SAR for Section 10.4 loop operating parameters and the accident analyses in Sections 15.4.4 and 15.4.6.

SAR Section 15.6

SAR Section 15.6.4, Rupture of PCS Outside the Radiographic Limits or an Opening in the PCS Due to Opening a Drain Valve, Relief Valve, or Vent Valve with an Equivalent Diameter of 3-Inch or Less

This event bounds the Condition 2 surge tank vent line break (primary coolant system depressurization) analyzed in EDF-4247, Analysis of ATR Surge Tank Vent Line Break. EDF-4247 RELAP5 calculations were performed using the RELAP5/MOD3 and RELAP5/MOD2.5 thermal-hydraulic codes. Version 3.2.1.2 of the RELAP5/MOD3 code was used to calculate the depressurization rate of the surge tank (for 2- and 3-PCP operation and air volumes of 40 and 60 ft). The ATR version of the RELAP5/MOD2.5 code was used to calculate the overall response of the primary coolant system (PCS). The surge tank pressure histories calculated with RELAP5/MOD3 were used as boundary conditions in the RELAP5/MOD2.5 calculations. Therefore, it is concluded that the errors in the CHF table will not affect the results this accident evaluation.

Section 15.6.5, Three-Inch Opening in PCS with Concurrent Failures

Discussed in this section are failures assumed to occur concurrently with the 3-in. equivalent pipe break discussed in Section 15.6.4 (referred to as the base-case 3-in. LOCA). For the Failure to Shut Off One PCP, it was assumed that one of the operating PCPs continued to run due to partial failure of the LOCA PCP shutoff ESF. Supporting analyses in EDF-4304 demonstrate that potential air entrainment in the primary coolant will not result in significant degraded pump performance. Air pull-through was examined with the RELAP5/MOD3 thermal hydraulics computer code. Air pull-through could result in air being entrained in the liquid to the primary coolant and was determined by implementing an experimentally based correlation into a system thermal hydraulics code such as RELAP5/MOD3. Therefore, it is concluded that the errors in the CHF table will not affect the results of this accident evaluation.

Section 15.6.10.4, Pressure Tube Penetration or Bottom Head Closure Plug Failure

A failure of a bottom head closure plate penetration (closure plug) is a Condition 4 event that could cause a LOCA during reactor shutdown. The analysis of the bottom head LOCA was performed in TRA-ATR-812. The analyses were performed with the RELAP5/MOD3 code. The analyses determined that the leak rate is initially 682.6 gpm (coolant at 100-ft elevation) and slows to 527.6 gpm when the coolant level is at the 86-ft elevation. The minimum available flow from the bottom-head EFIS exceeds this amount, therefore the core remains covered during this event.

This postulated event considers failure due to maintenance activities during shutdown conditions. During maintance activities on the bottom head all fuel is removed from the reactor. Therefore, it is concluded that the errors in the

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CHF table will not affect the results of this accident evaluation

D Gerstner / D McDaniel		**7/6/2010**
USQ Evaluator Print/Type Name	USQ Evaluator Signature	Date
Charles Forshee		7/7/10
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.

A PISA exists. Complete the following actions

- Nuclear Facility Manager document actions (below) including interim operating restrictions taken to place or maintain the facility in a safe condition.
- Nuclear Facility Manager execute reporting process for PISA per LWP-9301 and LWP-13830.
- Qualified USQ evaluator proceed to Section III.

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

Safety Analyst Print/Type Name	Safety Analyst Signature	Date
Nuclear Facility Manager Print/Type Name	Nuclear Facility Manager Signature	Date

Is independent technical review required? Yes ☐ No ☐