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

Facility or Activity: Advanced Test Reactor (ATR)

USQ Process No.: ATR Complex-USQ-2009-459

Subject: Inadequate Procedure E-2, Loss of Primary Coolant Flow

Describe the New Information/Discovery:
Step 17 of E-2, Loss of Primary Coolant Flow, directs the operators to establish LDW feed to the reactor after venting the PCS and prior to EFIS actuation. As part of the effort to address an issue from the ATR Design Basis Reconstitution Program, scoping calculations using RELAP and SINDA were performed for a Complete Loss of Flow accident which incorporated some of the steps in the ATR EPN, specifically E-2, Loss of Primary Coolant Flow. RELAP calculates overall plant response, including the hottest fuel element, and provides the input for SINDA, which calculates the results for individual fuel plates.

For this scenario, the following was assumed:

1. The reactor is initially operating at 250 MW with 2-PCP operation and 70 ft3 of air in the surge tank.
2. The accident is initiated by a loss of commercial and diesel power, which causes the PCPs, the SCPs, the pressurizing pumps, and the experiment loop pumps to coast down.
3. Commercial and diesel power are not necessarily lost simultaneously (which would result very fast release of the safety rods). The reactor scrams on low vessel inlet pressure, with no core power increase due to experiment loop heat up.
4. The operators depressurize the PCS to 50 psig, then, 28 minutes after scram, the pressure in the upper plenum is reduced to 11 psia (when GT-D-39 is opened). The operators began to vent the reactor vessel 28 minutes after scram--instead of the required 30 minutes--so that the upper plenum pressure is ~11 psia when ECP M-11 lost power at ~30 minutes.
5. The M-11 ECP loses power (dc power supply depleted) while the upper plenum pressure is at 11 psia. Flow through the core reverses and natural circulation is established inside the reactor vessel.
6. Following the flow reversal, the pressure at the top of the reactor vessel (~101 ft) will be increased to 12.2 psia (i.e., atmospheric pressure). This pressure is maintained for the remainder of the calculation.
7. The vessel outlet temperature when forced flow is lost is used to determine the time that bottom-head EFIS is actuated. It is assumed that efforts to establish LDW flow are not successful.
8. The RELAP calculation is terminated after EFIS flow is established and core temperatures are decreasing. The same bottom-head EFIS flow rate table that was used for the very small seismic LOCA analysis is used.

With the above conditions, RELAP predicts that boiling occurs in the reactor vessel, and SINDA predicts dryout of highest power fuel plates. With dryout, SINDA is not able to perform the calculations that fit the assumptions for the statistical Plant Protection Criteria. With a nominal SINDA calculation, the peak cladding temperature is less than 400 F with no fuel failure.

The result of interest is:

"Attempted a "worst-case" Plate 15 calculation with ATR-SINDA. This has been done in previous analyses in which boiling occurs to demonstrate compliance with the ATR plant protection criteria (see, for example, EDF-6503, "Analysis of ATR Very Small Seismically-Initiated LOCA"). Fuel plate melting occurred at ~30.6 minutes."

The worse-case scenario is used to demonstrate compliance when the normal statistical response is not possible. If the worst-case passes, then no failure would occur. The worst-case scenario uses the combination of parameters that has resulted in the highest cladding temperature in past calculations.

The above scenario is not in the ATR safety basis since SAR-153 uses only EFIS as a water source since it is a safety-related system and not LDW, since it is not safety-related. In addition, depressurization of the reactor vessel in the SAR scenario is not initiated until ~50 minutes, well after ECP M-11 is assumed to lose power, which results in the flow reversal.

SAR-153, Section 15.3.4, presents the Long-Term Complete Loss of Flow accident. In that case, both commercial and diesel power is lost with M-11 providing flow for 30 minutes. Long term success for this Condition 4 event is

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met by manually opening the vent valves prior to the upper plenum temperature reaching 200 degrees F. Following the depressurization, EFIS will supply adequate water to protect the core.

Identify the applicable safety basis document(s) (e. g., DSA, SAR, BIO, TSRs, etc.):
SAR-153, Updated Final Safety Analysis Report for the Advanced Test Reactor

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.
While the RELAP scoping calculations described above indicate that there is a potential for fuel melt to occur, there is no indication that the ATR safety basis is inadequate. As noted in the description above, SAR-153, Section 15.3.4, shows that for a Loss of Flow Accident, EFIS actuation will protect the core since PCS depressurization does not occur until later than the sequence described above. Depressurization occurs later since it is a function of the heat up rate for the upper plenum and the elapsed time from when the loss of flow occurred. At the later time, the steam generation rate is lower due to the lower decay heat and the higher PCS pressure. The inadequacy is not with the ATR safety basis but with the ATR EPNs which could place the plant in an unanalyzed condition if a Loss of Flow accident occurred.

William E. Kohn	JC Rafuse for WE Kohn per e-mail	2 July 2009
USQ Evaluator Print/Type Name	USQ Evaluator Signature	Date
Bruce Criswell		7/2/09
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.