15.5 INCREASE IN REACTOR COOLANT INVENTORY

15.5.1 INADVERTENT HPCI STARTUP

This event is non-limiting, and therefore, it had not been analyzed for each cycle. However, the inadvertent HPCI startup was reanalyzed for using the methods in References 15.5-4 through 15.5-6 for Unit 1, and References 15.5-4, 15.5-5, and 15.5-7 for Unit 2. Based on the results of this analysis, this event is identified as non-limiting at full power EPU conditions. The results of this analysis are reported in Section 15E.

Analyses of the inadvertent HPCI Startup at lower powers have shown that this event is potentially limiting and is evaluated on a cycle specific basis.

15.5.1.1 Identification of Causes and Frequency Classification

15.5.1.1.1 Identification of Causes

Manual startup of the HPCI system is postulated for this analysis, i.e., operator error.

15.5.1.1.2 Frequency Classification

This transient disturbance is categorized as an incident of moderate frequency.

15.5.1.2 Sequence of Events and Systems Operation

15.5.1.2.1 Sequence of Events

Table 15E.5.1-1 lists the sequence of events for Figure 15E.5.1-1.

15.5.1.2.1.1 Identification of Operator Actions

With the recirculation system in either the automatic or manual mode, relatively small changes would be experienced in plant conditions. The operator should, after hearing the alarm that the HPCI has initiated, check reactor water level and drywell pressure. If conditions are normal, the operator should shut down the system.

15.5.1.2.2 System Operation

To properly simulate the expected sequence of events the analysis of this event assumes normal functioning of plant instrumentation and controls, specifically, the pressure regulator and the vessel level control which respond directly to this event.

Required operation of engineered safeguards other than what is described is not expected for this transient event.

The system is assumed to be in the manual flow control mode of operation.

15.5.1.2.3 The Effect of Single Failures and Operator Errors

Inadvertent operation of HPCI results in a mild pressurization. Corrective action by the pressure regulator and/or level control is expected to establish a new stable operating state. The effect of a single failure in the pressure regulator will aggravate the transient depending upon the nature of the failure. Pressure regulator failures are discussed in Subsections 15.1.3 and 15.2.1.

A single failure in the level control system causes level rise or fall by improper control of the feedwater system. Increasing level will trip the turbine and automatically trip the HPCI system off. This trip signature is already described in the failure of feedwater controller with increasing flow. Decreasing level will automatically initiate scram at the L3 level trip and will have a signature similar to loss of feedwater control - decreasing flow.

15.5.1.3 Core and System Performance

15.5.1.3.1 Mathematical Model

The detailed nonlinear dynamic model described in References 15.5-4 through 15.5-6 was used to simulate this transient for Unit 1 and References 15.5-4, 15.5-5, and 15.5-7 for Unit 2.

15.5.1.3.2 Input Parameter and Initial Conditions

This analysis has been performed, unless otherwise noted, with plant conditions tabulated in Tables 15C.0-2 and 15D.0-2.

For the Unit 1 analysis, the water temperature of the HPCI system was assumed to be 40°F with an enthalpy of 11 BTU/Lb. The Unit 2 analysis is discussed in Reference 15.5-8.

Inadvertent startup of the HPCI system was chosen to be analyzed since it provides the greatest auxiliary source of cold water into the vessel.

For the SSES Units, the HPCI is introduced into only one of the feedwater lines. This will cause a non-symmetrical change in the inlet enthalpy. To account for the non-symmetrical introduction of colder water to the core, the HPCI flow assumed for this analysis is conservatively increased by 40% from 19% of the normal feedwater flow to 26% of normal feedwater flow for Unit 1 analyses. Application of the Reference 15.5-7 methodology also conservatively accounts for the core enthalpy asymmetry for U2C21 and later reloads.

15.5.1.3.3 Results

Figure 15E.5.1-1 shows the simulated transient event for the manual flow control mode. It begins with the introduction of cold water into the feedwater sparger. Within 1 second the full HPCI flow is established at approximately 27% of the rated feedwater flow rate. No delays were considered because they are not relevant to the analysis.

Addition of cooler water to the core causes the neutron flux to increase to the value shown in Table 15E.0-1 for this event.

15.5.1.3.4 Consideration of Uncertainties

Important analytical factors including reactivity coefficient and feedwater temperature change have been assumed to be at the worst conditions so that any deviations in the actual plant parameters will produce a less severe transient.

15.5.1.4 Barrier Performance

Figure 15E.5.1-1 indicates a slight pressure increase from initial conditions. The peak pressure is shown in Table 15E.0-1 for this event. Since the peak pressure is well below the design pressure of the RCPB, the RCPB is not threatened.

15.5.1.5 Radiological Consequences

Since no activity is released during this event, a detailed evaluation is not required.

15.5.2 Chemical Volume Control System Malfunction (or operator error)

This section is not applicable to BWR.

15.5.3 BWR Transients Which Increase Reactor Coolant Inventory

These events are discussed in Sections 15.1 and 15.2.

15.5.4 REFERENCES

15.5-1 Linford, R.B., ”Analytical Methods of Plant Transient Evaluations for General Electric Boiling Water Reactor,” April 1973 (NEDO 10802).

15.5-2 F. Odar, “Safety Evaluation for General Electric Topical Report: Qualification of One-Dimensional Core Transient Model for Boiling Water Reactors, ”NEDO-24154, NEDE-24154-P, Vols. I, II, III, dated June 1980.

15.5-3 Deleted

15.5-4 EMF-215B(P)(A) Rev.. 0: “Siemens Power Corporation Methodology for Boiling Water Reactors: Evaluation and Validation of CASMO-4/Microburn-B2,” Siemens Power Corporation, October 1999.

15.5-5 XNF-80-19(P)(A) Volume 4 Revision 1, Exxon Nuclear Methodology for Boiling Water Reactors: Application of the ENC Methodology to BWR Reloads, Exxon Nuclear Company, June 1986.

15.5-6 ANF-913(P)(A), Volume 1 Revision 1, Supplements 2, 3, and 4, “COTRANSA2: A Computer Program for Boiling Water Reactor Transient Analyses,” Advanced Nuclear Fuels Corporation, August 1990.

15.5-7 ANP-10300P-A Revision 1, "AURORA-B: An Evaluation Model for Boiling Water Reactors; Application to Transient and Accident Scenarios," Framatome, January 2018.

15.5-8 ANP-3884P, "Susquehanna Unit 2 Cycle 21 Reload Safety Analysis," Framatome, Inc.