15.5 INCREASE IN REACTOR COOLANT INVENTORY

Discussion and analysis of the following events is presented in this section:

a. inadvertent operation of emergency core cooling system during power operation,

b. chemical and volume control system malfunction that increases reactor coolant inventory, and

c. a number of BWR transients (not applicable to the Byron/Braidwood Stations).

These events, considered to be ANS Condition II, cause an increase in reactor coolant inventory. Subsection 15.0.1 contains a discussion of ANS classifications.

15.5.1 Inadvertent Operation of Emergency Core Cooling System During Power Operation

15.5.1.1 Identification of Causes and Accident Description

Inadvertent operation of the emergency core cooling system (ECCS) at power could be caused by operator error, test sequence error, or a false electrical actuation signal. A spurious signal initiated after the logic circuitry in one solid-state protection system train for any of the following engineered safety feature (ESF) functions could cause this incident by actuating the ESF equipment associated with the affected train.

a. High containment pressure,

b. Low pressurizer pressure, or

c. Low steamline pressure.

Following the actuation signal, the suction of the coolant charging pumps diverts from the volume control tank to the refueling water storage tank. Simultaneously, the valves isolating the charging pumps from the injection header automatically open and the normal charging line isolation valves close. The charging pumps force the borated water from the refueling water storage tank (RWST) through the pump discharge header, the injection line, and into the cold leg of each loop. The passive accumulator tank safety injection and low head system are available. However, they do not provide flow when the reactor coolant system (RCS) is at normal pressure.

A safety injection (SI) signal normally results in a direct reactor trip and a turbine trip. However, any single fault that actuates the ECCS will not necessarily produce a reactor trip.

If an SI signal generates a reactor trip, the operator should determine if the signal is spurious. If the SI signal is determined to be spurious, the operator should terminate SI and maintain the plant in the hot-standby condition as determined by appropriate recovery procedures. If repair of the ESF actuation system instrumentation is necessary, future plant operation will be in accordance with the Technical Specifications. If the SI results in discharge of coolant through the pressurizer safety valves, the operators will bring the plant to cold shutdown in order to inspect the valves.

If the reactor protection system does not produce an immediate trip as a result of the spurious SI signal, the reactor experiences a negative reactivity excursion due to the injected boron, which causes a decrease in reactor power. The power mismatch causes a drop in T_{avg} and consequent coolant shrinkage. The pressurizer pressure and water level decrease. Load decreases due to the effect of reduced steam pressure on load after the turbine throttle valve is fully open. If automatic rod control is used, these effects will lessen until the rods have moved out of the core. The transient is eventually terminated by the reactor protection system low pressurizer pressure trip or by manual trip.

The time to trip is affected by initial operating conditions. These initial conditions include the core burnup history which affects initial boron concentration, rate of change of boron concentration, and Doppler and moderator coefficients.

15.5.1.2 Analysis of Effects and Consequences

Method of Analysis

Inadvertent operation of the ECCS is analyzed using the LOFTRAN computer code (Reference 1). The code simulates the neutron kinetics, RCS, pressurizer, pressurizer relief and safety valves, pressurizer spray, the feedwater system, the steam generator, and steam generator safety valves. The code computes pertinent plant variables including temperatures, pressures, and power level.

Inadvertent operation of the ECCS at power is classified as a Condition II event, a fault of moderate frequency. The criteria established for Condition II events include the following.

a. Pressure in the reactor coolant and main steam systems should be maintained below 110% of the design values,

b. Fuel cladding integrity shall be maintained by ensuring that the minimum departure from nucleate boiling ratio (DNBR) remains above the DNBR limit, derived at a 95% confidence level and 95% probability, and

c. An incident of moderate frequency should not generate a more serious plant condition without other faults occurring independently.

The inadvertent ECCS actuation at power event is analyzed to determine the maximum RCS pressure encountered throughout the accident. The most limiting case with respect to RCS pressure is an SI at Hot Full Power coincident with a reactor trip. Because of the pressure reduction from the reactor trip, the SI flow is maximized. The SI flow refills the pressurizer until the pressurizer is water solid, and the SI flow results in liquid discharge through the pressurizer safety valves.

The performance of the pressurizer safety valve system and the loads on pressurizer safety valves, associated piping, and supports as a result of liquid discharge through the pressurizer safety valves, was determined to be acceptable (References 4 and 5).

The Inadvertent Operation of the ECCS During Power Operation event does not progress into a stuck open Pressurizer Safety Valve LOCA event. All three valves may lift in response to the event, but they will reclose. The resulting leakage from up to three pressurizer safety valves that are seated is bounded by flow through one fully open valve. The consequences of the event are bounded by the analysis described in UFSAR Section 15.6.1, "Inadvertent opening of Pressurizer Safety or Relief Valve" (References 6 and 7). This event is also classified as an event of moderate frequency.

American Nuclear Society standard 51.1/N18.2-1973 (Reference 2) describes example 15 of a condition II event as a "minor reactor coolant system leak which would not prevent orderly reactor shutdown and cooldown assuming makeup is provided by normal makeup systems only." In Reference 2, normal makeup systems are defined as those systems normally used to maintain reactor coolant inventory under respective conditions of startup, hot standby, power operation, or cooldown, using onsite power. Since the cause of the water relief is the ECCS flow, the magnitude of the leak will be less than or equivalent to that of the ECCS (i.e., operation of the ECCS maintains RCS inventory during the postulated event and establishes the magnitude of the subject leak). Therefore, the above example of a Condition II event is met.

The inadvertent ECCS actuation at power event is also analyzed to determine the minimum DNBR value.

The most limiting case is a minimum reactivity feedback condition with the plant assumed to be in manual rod control. Because of the power and temperature reduction during the transient, operating conditions do not approach the core limits.

The minimum DNBR was obtained at time zero for both units. The Unit 1 specific results are presented here. However, they are representative of the results for both Unit 1 and Unit 2.

The analysis assumptions for the DNBR evaluation are as follows:

a. Initial Operating Conditions

The event is analyzed with the revised thermal design procedure as described in WCAP-11397-P-A (Reference 3). Initial reactor power, RCS pressure and temperature are assumed to be at the nominal full power values. With the exception of the RCS average temperature bias, which is explicitly modeled in the analysis, uncertainties in initial conditions are included in the limit DNBR as described in Reference 3.

b. Moderator and Doppler Coefficients of Reactivity

The analysis assumes a zero moderator temperature coefficient and a low absolute value Doppler power coefficient at beginning of life.

c. Reactor Control

The reactor is assumed to be in manual rod control.

d. Pressurizer Pressure Control

Pressurizer heaters are assumed to be inoperable. This assumption yields a higher rate of pressure decrease which is conservative. Pressurizer spray and PORVs are assumed available in order to minimize RCS pressure.

e. Boron Injection

At the initiation of the event, two charging pumps inject borated water into the cold leg of each loop. The analysis assumes zero injection line purge volume for calculational simplicity; thus, the boration transient begins immediately in the analysis. The positive displacement charging pump is assumed to be inoperable at event initiation.

f. Turbine Load

The turbine load remains constant until the governor drives the throttle valve wide open. After the throttle valve is full open, turbine load decreases as steam pressure drops.

g. Reactor Trip

Reactor trip is initiated by a low pressurizer pressure signal at 1860 psia.

h. Decay Heat

The decay heat has no impact on the DNB case (i.e., minimum DNBR occurs prior to reactor trip). A conservative core residual heat generation based upon long-term operation at the initial power level is assumed.

i. Operator Action Time

Operator action is not required to mitigate the consequences of this event. Operator action is assumed to occur after the event to stabilize the plant in accordance with approved procedures to bring the plant to the applicable condition.

j. Pressurizer Safety Valves

The safety valves setpoints do not impact the minimum DNBR since the PORVs are assumed available to maintain low RCS pressure; this assumption is conservative with respect to DNBR.

k. Auxiliary Feedwater

Auxiliary feedwater was not credited.

l. Main Steam Safety Valves

The main steam safety valves are assumed conservatively to open at +5% above their nominal set pressure for the DNB case. No credit for steam dump is assumed in this analysis.

Plant systems and equipment credited for mitigating the effects of the accident are discussed in Subsection 15.0.8 and listed in Table 15.0-7. No single active failure in any of these systems or equipment will adversely affect the consequences of the accident.

Results

The transient response is shown in Figures 15.5-1 through 15.5-3. Table 15.5-1 shows the calculated sequence of events.

Nuclear power starts decreasing immediately due to boron injection, but steam flow does not decrease until later in the transient when the turbine throttle valve is wide open. The mismatch between load and nuclear power causes T_{avg}, pressurizer

water level, and pressurizer pressure to drop. The reactor trips and control rods start moving into the core when the pressurizer pressure reaches the pressurizer low pressure trip setpoint. The DNBR increases throughout the transient.

15.5.1.3 Radiological Consequences

There are only minimal radiological consequences associated with inadvertent ECCS operation. The reactor trip causes a turbine trip and heat is removed from the secondary system through the steam generator power relief valves or safety valves. Since no fuel damage is postulated to occur for this transient, the radiological consequences associated with an atmospheric steam release from this event would be less severe than the steamline break event analyzed in Subsection 15.1.5.3.

Water relief from the pressurizer PORVs and safeties may result in overpressurization of the pressurizer relief tank (PRT), breaching the rupture disk and spilling contaminated fluid into containment. The radiological releases (offsite doses) resulting from breaking the PRT rupture disk are limited by isolation of the containment.

15.5.1.4 Conclusions

Results of the analysis show that spurious ECCS operation at measurement uncertainty recapture conditions without immediate reactor trip does not present any hazard to the integrity of the core or the RCS with respect to DNBR. The minimum DNBR is never less than the initial value. If the reactor does not trip immediately, the low pressurizer pressure reactor trip will provide protection. This trips the turbine and prevents excess cooldown, which expedites recovery from the incident.

With respect to pressurizer filling, RCS pressure will stabilize well below the RCS pressure safety limit of 2735 psig. The performance of the pressurizer safety valve system and the loads on pressurizer safety valves, associated piping, and supports will be within acceptable limits.

15.5.2 Chemical and Volume Control System Malfunction That Increases Reactor Coolant Inventory

An increase in reactor coolant inventory which results from the addition of cold, unborated water to the reactor coolant system is analyzed in Subsection 15.4.6, chemical and volume control system malfunction that results in a decrease in boron concentration in the reactor coolant. An increase in reactor coolant inventory which results from the injection of highly borated water into the reactor coolant system is analyzed in Subsection 15.5.1, inadvertent operation emergency core cooling system during power operation.

15.5.3 A Number of BWR Transients

(Not applicable to the Byron/Braidwood Stations.)

15.5.4 References

1. Burnett, T. W. T., et al., "LOFTRAN Code Description," WCAP-7907-P-A (Proprietary), WCAP-7907-A (Non-Proprietary), April, 1984.

2. ANS-51.1/N18.2-1973, "Nuclear Safety Criteria for the Design of Stationary Pressurized Water Reactor Plants."

3. Freidland, A. J. and Ray, S., "Revised Thermal Design Procedure," WCAP-11397-P-A (Proprietary), WCAP-11397-A (Non-Proprietary) April 1989.

4. "Technical Evaluation Report, TMI Action Plan - NUREG-0737 (II.D.1) Braidwood Unit 1 & 2.. Docket No. 50-456, 50-457," G. K. Miller et. al, Idaho National Engineering Laboratory, January 1988.

5. NRC Letter from Mr. Leonard N. Olshan to ComEd Henry E. Bliss, dated August 18, 1988, Subject: NUREG-0737, Item II.D.1, Performance Testing on Relief and Safety Valves for Byron Station, Units 1 and 2 (TAC Nos. 56200 and 63240) transmitting Technical Evaluation Report (TER) providing the results of the NRC's review on Byron Units 1 and 2 response to NUREG-0737, Item II.D.1.

6. NRC Letter from Mr. George F. Dick (NRR) to Oliver D. Kingsley (Exelon), dated May 4, 2001. Subject: Issuance of Amendments; Increase in Reactor Power, Byron Station, Units 1 and 2, and Braidwood Station, Units 1 and 2" (TAC Nos. MA9428, MA9429, MA9426 and MA9427).

7. NRC Letter from Mr. George F. Dick (NRR) to Mr. Christopher M. Crane (Exelon), dated August 26, 2004. Subject: Byron Station, Units 1 and 2, and Braidwood Station Units 1 and 2 - Issuance of Amendments, RE: Pressurizer Safety Valve Setpoints, (TAC Nos. MB9762, MB9763, MB9760 and MB9761).

TABLE 15.5-1

TIME SEQUENCE OF EVENTS FOR INCREASE IN REACTOR COOLANT INVENTORY EVENTS

ACCIDENT	EVENT	TIME (sec.)
Inadvertent Actuation of ECCS During Power Operation		
	Spurious SI signal generated; two charging pumps begin injecting borated water	0
	Turbine throttle valve wide open, load begins to drop with steam pressure	51.5
	Low pressurizer pressure reactor trip setpoint reached	74.1
	Control Rod Motion Begins	76.1
	Minimum DNBR occurs	(*)

(*) - DNBR does not decrease below its initial value