Florida
Power
CORPORATION
Crystal River Unit 3
Docket No. 50-302
Operating License No. DPR-72

August 22, 2000
3F0800-14

U.S. Nuclear Regulatory Commission
Attn: Document Control Desk
Washington, DC 20555-0001

Subject: Pressurizer Heaters Emergency Power Supply, NUREG-0578, TMI-2 Lessons Learned Task Force Status Report and Short-Term Recommendations, Item 2.1.1 and NUREG-0737, Clarification of TMI Action Plan Requirements, Item II.E.3.1

Reference: FPC to NRC letter, 3F0700-08, dated July 20, 2000

Dear Sir:

The purpose of this letter is to respond to NRC questions concerning the referenced letter. In discussions with the NRC Project Manager and others of the NRC staff, several areas of clarification were requested. The questions and responses are contained in the attachment to this letter.

This submittal contains no new regulatory commitments.

If you have any questions regarding this submittal, please contact Mr. Sid Powell, Manager, Nuclear Licensing at (352) 563-4883.

Sincerely,

JJHolden

John J. Holden
Vice President and Site Director

JJH/rmb

Attachment: Responses to NRC Requests for Additional Information

xc: Regional Administrator, Region II
NRR Project Manager
Senior Resident Inspector

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Responses to NRC Requests for Additional Information

Questions 1, 2, and 4:

How will the licensee ensure natural circulation is not lost following a loss of offsite power scenario with RCV-8 leakage above 6 gpm? What will the licensee rely on for controlling pressure in this situation? If natural circulation capability is lost, how will the licensee ensure core cooling? (Provide discussion of existing procedures, compensatory actions, and recent and ongoing training to ensure heightened operator awareness and adequate capability to deal with the loss of offsite power scenario with an existing leak greater than 6 gpm.)

Response to Questions 1, 2, and 4:

To ensure natural circulation is not lost following a Loss Of Offsite Power (LOOP) with pressurizer relief valve (RCV-8) leakage above 6 gallons per minute (gpm), the operators will maintain subcooling by lowering reactor coolant system (RCS) temperature and pressure. Lowering RCS pressure will decrease the leak rate out of RCV-8. The decreased leak rate will require less heater capability to maintain pressure control. The Operators currently have administrative guidance in place that discusses compensatory actions for cooldown and pressure reduction of the RCS using approved Emergency Operating Procedures (EOPs). The administrative guidance provides a graph of target RCS pressures necessary to sustain Mode 3 (Hot Standby) for a given leak rate. The administrative guidance does not inhibit continuing with a cooldown and pressure reduction to lower Modes and decay heat removal system initiation if considered necessary. During licensed operator re-qualification classroom training, it was emphasized that it is not necessary to continue with a pressure reduction if RCS pressure can be stabilized and it is desired to stay at higher pressures.

The Improved Technical Specification (ITS) limit for Identified Leakage is 10 gpm. If all of the 10 gpm leak rate came from the pressurizer steam space, 420 kW of pressurizer heaters would be necessary to maintain subcooling after a LOOP. This is available by relying on both emergency diesel generators to supply power to the essential pressurizer heaters. The current specification requires a minimum of 252 kW for each train for a total of 504 kW. The current available emergency heater capacity is 378kW from each train for a total of 756 kW nominal.

If only one emergency diesel generator is available, the operators will still be able to successfully depressurize and cooldown the plant. During licensed operator re-qualification simulator training (July 12 -August 16, 2000), the operators were trained on pressurizer steam space leak rates of 6 gpm and 8 gpm through RCV-8 (3 crews at 6 gpm and 3 crews at 8 gpm). The 8 gpm steam space leak also included 1+ gpm leakage from the Decay Heat Removal System suction valve (DHV-3) for a total

identified leakage rate of slightly less than the ITS limit of 10 gpm. There are no significant operational differences between the two leak rates during plant pressure reduction and cooldown since identical procedures and step progressions are used. The only difference between a 6 gpm and 8 gpm leak is that the licensing bases requirement to maintain Hot Standby may not be met at leak rates above 6 gpm with only one emergency diesel generator available. The analysis has been shown to be conservative in that pressure control is obtained sooner than predicted during the plant simulator cooldown scenarios. The analysis did not model secondary side cool down by the emergency feedwater system (EFW) but did assume a constant average RCS temperature (T_{ave}).

The time to initiate cooldown and depressurization before subcooling is lost is approximately 3.5 hours with a 6 gpm leak. This time frame is well beyond the two-hour time period assumed in ITS Bases 3.4.8 for recovery of pressurizer heaters fed by emergency diesel generator power. Since the diesel-driven emergency feedwater pump (EFP-3) remains available, the RCS may be cooled down rather than maintained at constant temperature and depressurized at leak rates higher than 6 gpm. Thus, subcooling can be maintained until the heaters are returned to service or the plant is depressurized and cooled down utilizing normal letdown and make up. During the performance of EOP -9, Natural Circulation Cooldown, guidance is provided to the operators on how to control pressure with no pressurizer heaters. This is accomplished by controlling pressurizer inventory with letdown and makeup. With no pressurizer heaters available, RCS pressure will continually decrease due to the loss of inventory out of RCV-8. Subcooling can still be maintained by diesel-backed emergency feedwater (EFP-3)-secondary side cooling.

If the capability to maintain natural circulation is lost, the reactor coolant system (RCS) will begin to heat-up. This symptom of inadequate heat transfer requires the operator to enter EOP-4, Inadequate Heat Transfer. EOP-4 will direct the operator to initiate steps to re-establish primary to secondary heat transfer through lowering steam generator pressure to increase the temperature difference (delta T) between RCS and steam generator. If primary heat transfer can not be re-established and adequate subcooling exists, the hot leg vents are opened to remove non-condensable gases. Also, the steam generator level is increased and the primary to secondary delta T is increased further to promote natural circulation. At this point in EOP-4, the operators will wait for an indication of T_{incore} that reveals lowering temperatures due to steaming or feeding a steam generator. If loss of adequate subcooling occurs during this time frame, EOP-4 will direct the operators to establish High Pressure Injection / Power Operated Relief Valve (HPI / PORV) cooling, close the high point vents, and transition to EOP-8, LOCA Cooldown.

The preceding discussion was based on having EFW and secondary side integrity on at least one steam generator. If EFW and steam generator integrity is not available or cannot be established, HPI / PORV cooling will be initiated. Therefore, if natural

circulation cooling is lost and cannot be re-established, HPI / PORV cooling will be established.

The specific incident of proceeding from a LOOP to a loss of subcooling has not been included in operator training scenarios. However, events with loss of heat transfer and use of EOP-4 / EOP-8 have been included in operator training scenarios. Administrative Procedure (AI) -505, Conduct of Operations During Abnormal and Emergency Events, defines the priority of symptoms and directs the transition from one EOP to another when more important symptoms are present. The operators regularly receive training and evaluations on symptom monitoring and use of EOP-4 / EOP-8. Transition from a solid RCS and HPI / PORV cooling to steam generator cooling has also been demonstrated by the operators on the simulator.

Question 3:

Describe the impact on pressure control, if any, and what actions are taken to mitigate the impact, if one or both pressurizer heaters fail while the plant is at full power and RCV-8 leakage is greater than 6 gpm.

Response to Question 3:

If one feeder breaker were to open at hot full power conditions, one train of heaters would be inoperable. This would reduce the current capability from 1344 kW to approximately 644 kW on the "A" emergency buss or 700 kW on the "B" emergency buss. The current ambient losses from the pressurizer are 210 kW and the losses due to a 6 gpm leak from the pressurizer steam space at power would be 446 kW for a total of 656 kW.

If the "B" emergency buss were to fail and the leak rate was greater than 6 gpm, or if both trains of heaters were lost, RCS pressure would decay. Abnormal Procedure (AP) -520, Loss of RCS Coolant or Pressure, directs operators to initiate restoration of pressurizer heaters. If the heaters could not be restored and RCS pressure control could not be maintained, the operator is directed to initiate a reactor trip. At this point, the operator would perform a cooldown and depressurization of the RCS as previously described. If no heaters were available, the operator would control RCS pressure and pressurizer level using letdown and makeup.

The following is a description of the pressurizer heater groups and possible failure modes:

Under normal (offsite) power lineup from the Start-Up Transformer, six pressurizer heater groups (644 kW) are powered from the "A" side and seven pressurizer heater groups (700 kW) are powered from the "B" side. Following a LOOP, power is supplied by the emergency diesel generators to 378 kW of heaters on each train.

The failure modes that could potentially lose a train of pressurizer heaters are a failure of either of the Reactor Auxiliary Busses, the feeders from the Reactor Auxiliary Busses to the pressurizer motor control centers or the motor control centers themselves. All other components have an alternate distribution path in the emergency mode. Catastrophic failure of the Reactor Auxiliary Busses or the motor control centers is remote. Failure of the feeder breaker is more likely but the feeder breaker is a 1600 amp frame size breaker that, under maximum normal conditions, would be loaded at approximately 950 amps (approximately 850 amps for 700 kW of pressurizer heaters and approximately 100 amps for a non-1E battery charger). At full power with a 6 gpm RCV-8 leak, approximately 656 kW is required (210 kW for ambient losses and 446 kW for RCV-8 pressurizer steam space leakage). Half of this would be powered from the "A" side and half from the "B" side. This equates to approximately 500 amps through the breaker (400 amps pressurizer heaters and 100 amps maximum battery charger loading (not usually this high)). Under maximum usage, emergency conditions the breaker would see approximately 455 amps (378 kW pressurizer heaters).

In summary, the components are oversized and not taxed by normal or emergency operating conditions. The breakers were re-furbished within the past eight years and are part of our preventive maintenance program. There have been no failures since the breakers were re-furbished. The breakers have been well maintained and the failure of any of these components during the two-month period is remote.

Question 5:

To avoid creating a LOOP and to ensure that the equipment relied upon in the submittal is available during the projected two-month period, does the licensee plan to curtail activities (e.g., maintenance) and/or increase surveillances associated with or near this equipment, such as ac and dc power busses and supply busses (including the switchyard), emergency diesel generators, and the makeup/letdown system?

Response to Question 5:

The twelve-week schedule has been reviewed and potential work that may affect the switchyard prior to the planned outage to replace RCV-8 has been removed. AI-1300, Engineering, Maintenance, and Support Interfaces, controls the work performed in the switchyard so that such activities are in compliance with Crystal River Unit 3 (CR-3) procedural controls and are integrated into the CR-3 work control process. This will ensure that only necessary work will be allowed under strict controls. Required surveillances will be performed at the normal frequency. No increases in surveillance activities are planned.

Question 6:

The submittal provides an average probability for a loss of offsite power (LOOP) over any two-month period, which does not address seasonal conditions (e.g., hurricane season). What is the probability of a LOOP for the projected two-month period (mid-September through mid-November) in which RCV-8 leakage is projected to be greater than 6 gpm?

Response to Question 6:

The values in the earlier submittal were provided to show the low probability of a LOOP during any two-month period. If it is conservatively assumed that all risk of a LOOP occurs during hurricane season (six months from June to November), then the probability of a LOOP during any two months of this time period is doubled from 4.3×10^{-3} to 8.6×10^{-3}. Increasing the probability of a LOOP, although not desirable, is acceptable since a LOOP does not lead to core damage. A LOOP with leaks in excess of 6 gpm from the pressurizer steam space are bounded by the Station Blackout event described in the response to Question 7.

Question 7:

The submittal addresses the probability that both emergency diesel generators (EDGs) would not be available with offsite power not recovered within 2 hours. Have other system or component failures been considered, such as a loss of either pressurizer heater bank (or loss of control power or control logic to either pressurizer heater bank) following a LOOP? Explain if and how other system or component failures were considered or why they are not considered.

Response to Question 7:

The pressurizer heaters were not credited in our risk model and the failure to perform their function results in no increase to core damage frequency. This submittal is deterministic based on bounding prior analysis. Therefore, no other system or component failures were considered or evaluated. The following discusses the docketed analysis for a Station Blackout (SBO) event:

For the beyond design bases event of an SBO, analysis and the simulator have shown that subcooling is maintained for up to 3.5 hours with no operator action to cooldown and depressurize the RCS. After this time, the reactor coolant system would become saturated and maintain saturated conditions. The diesel-driven emergency feedwater pump (EFP-3) remains available during an SBO and subcooling could be maintained. The current docketed SBO analysis includes a four-hour coping requirement with a 111 gpm leak from the reactor coolant system. Subcooling is not maintained but saturated conditions are maintained with Boiler / Condenser Cooling. The current discussion on a loss of all pressurizer heater

capabilities with up to a 10 gpm pressurizer steam space leak is bounded by the current docketed analysis.