Florida Power
CORPORATION
Crystal River Unit 3
Docket No. 50-302

July 3, 1991
3F0791-05

U. S. Nuclear Regulatory Commission
Attn: Document Control Desk
Washington, D.C. 20555

Subject: Station Blackout Rule Implementation

Reference: A. NRC to FPC letter dated May 6, 1991
B. FPC to NRC letter dated June 13, 1991

Dear Sir:

This letter provides additional information in response to the NRC Staff's Supplemental Safety Evaluation (Reference A) regarding station blackout rule implementation for Crystal River Unit 3. The attachment to this letter provides a summary of the station blackout reactor coolant inventory analysis which documents the applicability of the assumption in Section 2.5 of NUMARC 87-00, Guidelines and Technical Bases for NUMARC Initiatives Addressing Station Blackout at Light Water Reactors. That assumption is that reactor coolant inventory makeup systems in addition to those currently available under blackout conditions are not required. Florida Power Corporation committed to provide this summary (Reference B) in response to a request by the NRC Staff.

Sincerely,

Gary Boldt

G. L. Boldt
Vice President
Nuclear Production

GLB:AEF

Attachment

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

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SUMMARY OF THE STATION BLACKOUT REACTOR COOLANT INVENTORY ANALYSIS FOR CRYSTAL RIVER 3

INTRODUCTION

A station blackout (SBO) event with coincident reactor coolant pump seal leaks on a generic B&W plant model was simulated with the B&W version of RELAP5/MOD2. Operator action was modeled to maintain the plant in a hot, pressurized, shut down condition for approximately four hours after reactor trip.

ASSUMPTIONS

The RELAP5 analysis of an SBO event incorporated the following assumptions:

1. The plant was initially operating at 2772 MWt. (Crystal River 3 is licensed to operate up to 2544 MWt.)

2. The initial indicated pressurizer level was 180". (This is within the allowable operating range, but below the low level alarm setpoint.)

3. Maximum realistic decay heat corresponding to 1.0 times ANS 5.1 of 1979 was assumed.

4. A loss of off-site power initiated the event by tripping the reactor, tripping the turbine, tripping the reactor coolant pumps, and tripping the main feedwater pumps.

5. Reactor coolant pump seal leaks of 25 gpm per pump (100 gpm total) began coincident with the reactor trip. These leaks were modeled as critical flow orifices. Consequently, leak flows were affected by local pressure and fluid density changes.

6. Maximum reactor coolant system leakage allowed by Technical Specifications, 10 gpm identified and 1 gpm unidentified, was also modeled.

7. Letdown and seal return were assumed to be isolated shortly after reactor trip. Consequently, neither flow path was modeled since the integrated new flow was considered insignificant.

8. Steam generator levels were raised to and controlled at 50% on the operate range level indication by the Emergency Feedwater Initiation and Control (EFIC) System following the reactor trip. Steam generator levels are raised to 95% upon loss of minimum adequate subcooling margin of 20 °F, as required by plant procedures.

9. Station batteries were considered available for the entire four hours. Consequently, all instrumentation was available.

10. The power operated relief valve (PORV) was available for the entire four hours.

11. Atmospheric dump valves were available for the entire four hours.

RESULTS

The reactor coolant system experienced a normal post-trip contraction and cooldown due to the reduction in reactor power. The system made a smooth transition to two-loop natural circulation as the steam generator levels were raised to 50% on the operating range.

The pressurizer experienced a normal out-surge following the reactor trip. The pressurizer level and primary pressure continued to decrease due to the reactor coolant pump seal leaks and maximum Technical Specification leakage. Since no make-up flow was available, the pressurizer emptied in about 30 minutes. As a result, the primary system depressurization rate increased. Although liquid in the reactor vessel upper head area began to flash, the system continued to depressurize until the minimum adequate subcooling margin of 20 °F was lost. Upon loss of sub-cooling margin, steam generator secondary liquid levels were raised to 95% on the operating range at 7 to 8 inches per minute. The system continued to depressurize until the hot legs became saturated.

Approximately 117 minutes after the reactor trip, the hot legs began to experience significant voiding and a period of two phase natural circulation began. The reduced natural circulation flow was insufficient to remove decay heat. Consequently, the primary system began to repressurize.

The repressurization continued until expansion of the core exit and hot leg liquid caused the loop B flow to increase, establishing single phase natural circulation at 135 minutes after the reactor trip. System pressure stabilized and the loop A flow stagnated. Single phase natural circulation flow was not reestablished in loop A because some of the hot core liquid which was expanding into the hot legs also entered the pressurizer. Consequently, the loop A riser level remained slightly below the loop B riser level. Also, the loop A riser fluid was not heated as much as the loop B riser fluid. As a result, the loop B hot leg riser liquid density was lower and flow reestablished in that loop. However, since the loop B liquid was very near saturated, natural circulation stopped again at 175 minutes after the reactor trip due to void formation in the loop B hot leg.

Following flow stagnation, the primary system repressurized toward the PORV setpoint. This repressurization was slowed by periodic flow surges in the loops which induced heat transfer in the steam generators and caused primary pressure to decrease for several minutes until the flow stopped.

These flow surges were caused by voiding of the cold leg discharge piping which caused that liquid volume to be displaced up the hot legs. The increase in hot leg riser volume resulted in an increase in the hot leg riser mixture level above the U-bend spill-over elevation. The mixture spilled over to the steam generator side, inducing flow which caused subcooled liquid from the steam generators and cold leg J-bends to flow through the reactor coolant pumps and collapse the steam voids in the cold leg discharge piping. Collapsing the cold leg voids caused a rapid decrease in primary pressure. It also caused the hot leg levels to decrease until they were below the U-bend spill-over elevation and flow stopped. Subsequently, the system repressurized.

The primary system experienced several of these hot leg U-bend spill-overs. Near the end of the analysis, the repressurization/depressurization cycles were

smaller in magnitude as high elevation boiler-condenser mode cooling slowly became a greater contributor to the core heat removal. Primary side steam generator liquid levels fell below the upper tube sheets inducing condensation until cold leg voiding eliminated the condensing surface by pushing liquid into the steam generators and raising the level above the tube sheets. However, at the end of the analysis (270 minutes after the reactor trip) the primary side steam generator liquid levels were below the tube sheets and high elevation boiler-condenser mode cooling was established. Also, at the end of the analysis there was approximately 2400 cubic feet of primary system liquid available to compensate for seal leakage. This includes the liquid above the top of the core in the reactor vessel, the liquid in the hot leg risers and pressurizer, and the liquid in the steam generators above the reactor coolant pump spill-over elevation. This means that core uncovering would not occur for another three hours.

CONCLUSIONS

This analysis shows that Crystal River 3 can cope with a station blackout with coincident leakage of 111 gpm by following current guidelines. All acceptance criteria were met. Placing the plant in two-loop natural circulation and holding the steam generators at constant pressure will keep the core covered for over four hours with this amount of leakage. Also, the PORV only lifted one time and passed steam when it was open. Consequently, this does not represent a challenge to the valve's ability to operate correctly.