CP&L

Carolina Power & Light Company

Brunswick Steam Electric Plant
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U.S. Nuclear Regulatory Commission
ATTN: Document Control Desk
Washington, DC 20555

BRUNSWICK STEAM ELECTRIC PLANT UNITS 1 AND 2
DOCKET NOS. 50-325 AND 50-324
LICENSE NOS. DPR-71 AND DPR-62
RESPONSE TO INFRACTIONS OF NRC REQUIREMENTS

Dear Dr. Grace:

The Brunswick Steam Electric Plant (BSEP) has received I&E Inspection Report 50-325/88-14 and 50-324/88-14 and finds it does not contain information of a proprietary nature.

This report identified three items that appeared to be in noncompliance with NRC requirements. Enclosed please find Carolina Power & Light Company's response to these violations.

Very truly yours,

C. R. Dietz, General Manager
Brunswick Steam Electric Plant

MJP/srg

Enclosure

cc: Dr. J. N. Grace
Mr. E. D. Sylvester
BSEP NRC Resident Office

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VIOLATION A

Technical Specification 3.4.6 provides pressure/temperature limits for the Reactor Coolant System that are applicable at all times. Under that specification, surveillance requirement 4.4.6.1.1 requires that the reactor vessel shell temperature and reactor vessel pressure be determined to be within the limits at least once per 30 minutes during system heatup, cooldown, and inservice leak and hydrostatic testing operations.

Contrary to the above, on January 25, 1988, reactor vessel pressure and shell temperature were not determined to be within limits once per 30 minutes during system heatup. Specifically, from 0245 hours to 0430 hours, a Reactor Coolant System heatup of about 90 degrees Fahrenheit occurred with no determination during that time that reactor vessel pressure and shell temperatures were within limits.

This is a Severity Level IV violation (Supplement I).

RESPONSE

I. Admission or Denial of the Alleged Violation

CP&L denies that reactor vessel pressure and reactor vessel shell temperature readings were required to be verified within limits every 30 minutes based on the following:

1. Based on the basis section of the technical specification, the terms "heatup" and "cooldown" as used in Technical Specification 4.4.6.1.1 clearly refer to actions related to startup and shutdown. The intent of the surveillance requirement was met. The basis for this section states, "During startup and shutdown, the rates of temperature and pressure change are limited so that maximum specified heatup and cooldown rates are considered with the design assumptions and satisfy the stress limits for cyclic operation." This requirement does not refer or apply to temperature changes made while in the cold shutdown condition.

2. There is no technical concern. The temperature pressure curves do not apply when the reactor vessel is vented. The bulk reactor water temperature increased 90°F over approximately 1.75 hours, representing only an average heatup rate of < 52°F/hr. Actual vessel metal temperature change was less than 10° over the same period or < 6°F/hr.

It is acknowledged that the event which led to the alleged violation was an error and that recurrence is undesirable. The temperature logging requirements while shut down have been increased to reflect a 30-minute interval until future long-term actions are completed. Additionally, in response to this event, an internal review was completed which established real-time training of the event (complete), initiated an evaluation of the adequacy of design of the current control system, established an action item to evaluate need for additional alarms for the operator, including the consideration of a ERFIS shutdown screen to assist the operator in monitoring shutdown parameters, and established an action item to develop a training package outlining items of concern/lessons learned for the shutdown condition. Additionally, appropriate disciplinary action for involved operators was administered.

VIOLATION B

10CFR50.59(a)(1) allows a licensee to make changes in the facility as described in the safety analysis report without prior commission approval, unless the proposed change involves an unreviewed safety question. 10CFR50.59(b) requires the licensee to maintain records of these changes. These records must include a written safety evaluation which provides the basis for the determination that the change does not involve an unreviewed safety question. Final Safety Analysis Report (FSAR) Table 9.2.1-1, Service Water Flow Distribution - One Reactor Plant, lists Reactor Building CCW heat exchanger flow rate from the nuclear service water header during the first 10 minutes following a loss of coolant accident as zero gallons per minute.

Contrary to the above, a written safety evaluation providing the basis for the determination that a change did not involve an unreviewed safety question was not performed. The licensee received information prior to Unit 1 startup on February 20, 1988, that with certain single failures, nuclear service water flow to the Reactor Building CCW heat exchangers would not be zero gallons per minute during the first 10 minutes of a loss of coolant accident. This information constitutes a change in the facility as described in the FSAR. A written safety evaluation was not completed until March 22, 1988, subsequent to being identified by the NRC.

This is a Severity Level IV violation (Supplement I).

RESPONSE

I. Admission of the Alleged Violation

A formal nuclear safety assessment; i.e., 10CFR50.59 evaluation, was not documented as a result of the preliminary engineering assessment presented to and approved by the Plant Nuclear Safety Committee (PNSC); reference PNSC Meeting No. 88-018 of February 12, 1988.

II. Reason for the Violation if Admitted

The preliminary engineering assessment of the SW-V106 issue was performed in accordance with an approved site procedure; i.e., BSEP procedure OI-04, Limiting Condition for Operation (LCO) Evaluations and Follow-up Operating Instruction, revision 27. The assessment was taken to PNSC for concurrence because available information at the time of the assessment resulted in a decision to take compensatory action on the RBCCW SW System; an action taken because of a potential for the Service Water System to be operating outside of the system design bases.

As a result of the presentation to PNSC (Meeting No. 88-018), an action item was initiated for tracking the follow-up evaluation to eliminate/resolve the 5,000 gpm flow restriction (compensatory/interim action) to the RBCCW heat exchangers. The follow-up evaluation would be ultimately documented in accordance with BSEP procedure ENP-12, Engineering Evaluation Procedure, revision 17, and contain a formally documented 10CFR50.59 nuclear safety assessment.

A second engineering party was assigned the task of tracking the action item for formal follow-up evaluation of the SW-V106 issue. BSEP project PID-6329A, Service Water Flow Distribution Verification, was issued to a third engineering organization to perform the follow-up evaluation. The follow-up evaluation was completed in May 1988.

This violation occurred because the Brunswick plant did not have clear policy guidance in effect for determining how to deal with plant conditions which constitute potential deviations from the description of the facility as described in the FSAR. Even though procedural guidance existed for dealing with and assessing the operability significance of such events, no clear link existed between that assessment process and the engineering evaluation process which would have resulted in a justification for continued operation (with attendant 10CFR50.59 evaluation). As a result, no mechanism for timely evaluation to the requirements of 10CFR50.59 existed.

An engineering evaluation, EER 88-0167, was generated during March 1988 to document the basis for continued operation of the Brunswick units pending final resolution of PID-6329A. This EER was generated after concern was raised by the NRC over lack of a formal evaluation.

III. Corrective Steps Which Have Been Taken and Results Achieved

EER 88-0167 was written documenting the basis for continued operation of the Brunswick units, as discussed above.

Project PID-6329A, Service Water Flow Distribution Verification, was completed. This work concluded that adequate margin existed within the Service Water System flow characteristics to provide for adequate design flow to critical components even with valve SW-V106 open, following a design basis event. In addition, EER 88-0617 was revised on June 2, 1988, to reflect the additional information resulting from PID-6329A.

III. Corrective Steps Which Have Been Taken and Results Achieved

Following notification of a potential problem with the valve position, the valve was locked in the open position at 1700 hours on March 4, 1988. In addition, on the same day, a temporary revision to the involved Operating Procedure, OP-24, was implemented to specify in the procedure valve lineup that valve be locked open. This procedure change was made a permanent part of the procedure on March 15, 1988. Due to the procedure change involving OP-24, a procedure change to PT-16.1 was deemed to be unnecessary.

IV. When Full Compliance Will Be Achieved

Full compliance concerning this requirement has been achieved.

IV. Corrective Steps Which Will be Taken and When Full Compliance Will be Achieved

1. Engineering personnel involved in preparation of operability assessments and engineering evaluations will be briefed on this event by August 1, 1988.

2. Improved policy guidance will be developed for dealing with plant conditions which constitute potential deviations from the facility as described in the FSAR. This guidance will provide a clear link to the engineering evaluation/justification for continued operation process as appropriate to ensure 10CFR50.59 evaluations are conducted when required in a timely manner. This guidance will be issued by September 1, 1988.

VIOLATION C

Technical Specification 4.6.6.2.a.2 requires that the Containment Atmospheric Dilution (CAD) System shall be demonstrated to be operable at least once per 31 days by verifying that each valve (manual, power-operated, or automatic) in the flow path not locked, sealed, or otherwise secured in position, is in its correct positon. PT-16.1, revision 12, CAD System Component Test, implements this requirement.

Contrary to the above, the CAD System was not demonstrated to be operable by verifying each manual valve in the flow path not locked was in the correct position. Valve 1-CAC-V168, a flow path valve, was open and not locked on and before February 26, 1988, and was not verified in its correct position (open) by PT-16.1, revision 12.

This is a Severity Level V violation (Supplement I).

RESPONSE

I. Admission or Denial of the Alleged Violation

CP&L admits this violation occurred as described.

II. Reason for the Violation

This violation is attributed to an inadequate technical review of the information package associated with the plant modification which installed 1-CAC-V168. Following installation of the valve, the responsible Engineering reviewer and the Operations procedure writer and technical reviewers failed to recognize the necessity that the valve be controlled in accordance with Technical Specification 4.6.6.2.a.2. This occurred despite the fact that the modification package drawings show the valve as being locked open.