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CP&L

Carolina Power & Light Company

April 23, 1979

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Mr. James P. O'Reilly, Director
U. S. Nuclear Regulatory Commission
Region II
101 Marietta Street, Suite 3100
Atlanta, Georgia 30303

BRUNSWICK STEAM ELECTRIC PLANT, UNIT NOS. 1 & 2
LICENSE NOS. DPR-71 AND DPR-62
DOCKET NOS. 50-325 AND 50-324
RESPONSE TO IE BULLETIN 79-08

Dear Mr. O'Reilly:

Attached you will find CP&L's response to Items 1-10 of Bulletin 79-08 concerning events relevant to boiling water reactors identifed during the Three-Mile Island incident.

I trust that this information is suitable for your use.

Yours very truly,

D B Waters for B J Furr

B. J. Furr
Manager
Generation Department

CSB:men*

Attachment

cc: NRC Office of Inspection & Enforcement

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411 Fayetteville Street • P. O. Box 1551 • Raleigh, N. C. 27602

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BRUNSWICK STEAM ELECTRIC PLANT
UNITS 1 AND 2
DOCKET NOS. 50-325 AND 50-324
RESPONSE TO IE BULLETIN 79-08

Introduction

As required by IE Bulletin 79-08, Carolina Power & Light has conducted reviews and held discussions concerning the events of the Three Mile Island Incident and how they relate to the Brunswick Steam Electric Plant. During the 10-day response period, the reviews conducted were necessarily preliminary, and CP&L commits to having detailed reviews completed and changes in procedures implemented, if required, by July 31, 1979. The following are the response to Items 1-10 of the bulletin as they relate to the Brunswick Steam Electric Plant:

Item 1

Review the description of circumstances described in Enclosure 1 of IE Bulletin 79-05 and the preliminary chronology of the TMI-2 March 28, 1979 accident included in Enclosure 1 to IE Bulletin 79-05A.

a. This review should be directed toward understanding: (1) the extreme seriousness and consequences of the simultaneous blocking of both trains of a safety system at the Three Mile Island Unit 2 plant and other actions taken during the early phases of the accident; (2) the apparent operational errors which lead to the eventual core damage; and (3) the necessity to systematically analyze plant conditions and parameters and take appropriate corrective action.

b. Operational personnel should be instructed to (1) not override automatic action of engineered safety features unless continued operation of engineered safety features will result in unsafe plant conditions (see Section 5a of this bulletin); and (2) not make operational decisions based solely on a single plant parameter indication when one or more confirmatory indications are available.

c. All licensed operators and plant management and supervisors with operational responsibilities shall participate in this review and such participation shall be documented in plant records.

CP&L Response

a. Carolina Power & Light, recognizing the seriousness of the events at TMI, established a Corporate Investigative Team in the days following the accident. The team is responsible to CP&L Management for taking the TMI Unit 2 event as a reference and performing a review of the Company's nuclear plant designs and operating procedures to access and assure present and

continued safety at its nuclear power plants. The team is divided into two subteams: one to evaluate PWRs and the other, BWRs. Each subteam is composed of representatives from the power plant, CP&L general office, reactor vendor, and architect/engineer to assure the orderly and prompt development and dissemination of all information.

On April 19, 1979, the BWR subteam met to discuss the sequence of events which occured at TMI, to identify operating errors at TMI and their seriousness, and to determine if parallel problems could occur at the Brunswick Steam Electric Plant. As more detailed information becomes available, the team will continue to review, performing an overall examination of the plant design, safety, procedures, emergency plan, and training.

A meeting was held on April 24, 1979 for plant management, supervisors with operational responsibilities, and a representative from the CP&L TMI Corporate Investigative Team to transfer the findings of the team to the plant personnel and to enable the plant manager to reaffirm CP&L's commitment to responsible and safe operation. The available information on the chronology of events, and inoperability of safety systems, apparent operator errors, and failure to comprehend the actual plant status will be discussed.

The nonoperations employees will be briefed on the events at TMI at the regularly scheduled monthly employee information meeting on April 26, 1979. These meetings will include a discussion of the sequence of the events and each employee's responsibility toward assuring safe plant operation.

1b, c. All licensed personnel, plant management, and supervisors with operational responsibilities have received the first of two study packages on the events that occurred at TMI-2 on March 28. This first review included the events leading up to and those occurring during the transient, the comparisons between TMI and BSEP systems, and also the importance of not overriding automatic actuation of engineered safety features based on the indication of a single instrument or unless continued operation would result in an unsafe plant condition. Part 2 of this package will be issued when, and if, all changes to plant operations have been developed to prevent and/or mitigate the consequences of a similar event.

Operations personnel were kept informed of available information as the events progressed and this information was available. A formal review of the TMI Incident has been developed and is in progress for all operations personnel. The review will reach personnel on all shifts and will be completed within ten days.

Item 2

Review the containment isolation initiation design and procedures, and prepare and implement all changes necessary to initiate containment isolation, whether manual or automatic, of all lines whose isolation

does not degrade needed safety features or cooling capability, upon automatic initiation of safety injection.

CP&L Response

A review of the primary containment isolation design has been completed. This review verified that a safety injection signal will automatically initiate containment isolation on all valves whose isolation does not degrade needed safety features or cooling capability. In addition, the applicable operating Emergency Instructions were reviewed to ensure that the operators are instructed to verify that all automatic actions do occur.

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Item 3

Describe the actions, both automatic and manual, necessary for proper functioning of the auxiliary heat removal systems (e.g., RCIC) that are used when the main feedwater system is not operable. For any manual action necessary, describe in summary form the procedure, by which this action is taken in a timely sense.

CP&L Response

The auxiliary heat removal systems provided to remove decay heat from the reactor core and containment following loss of feedwater systems are:

High Pressure Coolant Injection (HPCI) System
Reactor Core Isolation Cooling (RCIC) System
Core Spray (CS) System
Residual Heat Removal (RHR) System

The description that follows details the operation of the systems needed to achieve initial core cooling followed by containment cooling and then followed by extended core cooling for a long-term plant shutdown.

Initial Core Cooling

Following a loss of feedwater and reactor scram, a low reactor water level signal (Level 2) will automatically initiate main steam line isolation valve closure. At the same time, this signal will put the HPCI and RCIC Systems into the reactor coolant makeup injection mode. These systems will continue to inject water into the vessel until a high water level signal automatically trips the system.

Following a high reactor water level trip, the HPCI System will automatically reinitiate when reactor water level again decreases to low water Level 2. The RCIC system must be manually reset by the operator in the control room before it will automatically reinitiate after a high water trip.

The HPCI and RCIC Systems have redundant supplies of water. Normally they take suction from the Condensate Storage Tank (CST). The HPCI System suction will automatically transfer from the CST to the suppression pool if the CST water is depleted or the suppression water level increases to a high level.

The RCIC System suction must be manually transferred from the CST to the suppression pool using controls located in the main control room. This action would be taken when control room alarms indicate low CST or suppression pool high water level.

The operator can manually initiate the HPCI and RCIC systems from the control room before the Level 2 automatic initiation level is reached. The operator has the option of manual control or automatic initiation and can maintain reactor water level by

throttling system flow rates. The operator can verify that these systems are delivering water to the reactor vessel by:

a) Verifying reactor water level increases when systems initiate.
b) Verify systems flow using flow indicators in the control room.
c) Verify system flow is to the reactor by checking control room position indication of motor-operated valves. This assures no diversion of system flow to the reactor.

Therefore, the HPCI and RCIC can maintain reactor water level at full reactor pressure and until pressure decreases to where low pressure systems such as the Core Spray (CS) or Low Pressure Coolant Injection (LPCI) can maintain water level. If for some reason, the HPCI and RCIC do not maintain reactor water level, the Automatic Depressurization System (ADS) will initiate depressurization of the reactor quickly such that the CS and LPCI can immediately begin to cool and flood the core.

Containment Cooling

After reactor scram and isolation and establishment of satisfactory core cooling, the operator would start containment cooling. This mode of operation removes heat resulting from HPCI and RCIC turbine operation and Safety-Relief Valve (SRV) discharge to the suppression pool. This would be accomplished by placing the Residual Heat Removal (RHR) System in containment (suppression pool) cooling mode, i.e., RHR suction from and discharged to the suppression pool.

The operator can verify proper operation of the RHR System containment cooling function from the control room:

a) Verifying RHR and Service Water (SW) System flow using system control room flow indicators.
b) Verify correct RHR and SW System flow paths using control room position indication of motor-operated valves.
c) On branch lines that could divert flow from the required flow paths, close the motor-operated valves and note the effect on RHR and SW flow rate.
d) Monitoring RHR and SW temperatures into and out of the RHR-SW heat exchangers using the control room instrumentation.

When the RHR is in the containment cooling mode, core cooling continues to be its primary function. Thus, if a high drywell pressure signal or reactor low water signal is again received at any time during the period when the RHR is in the containment cooling mode, the RHR System will automatically revert to the LPCI injection mode. The Core Spray System would automatically initiate and both the LPCI and CS Systems would inject water into the reactor vessel if reactor pressure is below steam discharge pressure.

Extended Core Cooling

When the reactor has been depressurized to less than 100 psig, the RHR System can be placed in the long-term shutdown cooling mode. The operator manually terminates the containment cooling mode of one of the RHR containment cooling loops and places the loop in

shutdown cooling mode as follows:

a) Trip the RHR pumps.
b) Close motor-operated valves in the suppression pool suction and discharge lines.
c) Rack out the RHR minimum flow valve breaker.
d) Open suction valves from and discharge valves to the reactor vessel.
e. Restart the RHR pumps.

In this operating mode, the RHR System can cool the reactor to cold shutdown. Proper operation and flow paths in this mode can be verified by methods similar to those described for the containment cooling mode.

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Item 4

Describe all uses and types of vessel level indication for both automatic and manual initiation of safety systems. Describe other redundant instrumentation which the operator might have to give the same information regarding plant status. Instruct operators to utilize other available information to initiate safety systems.

CP&L Response

The reactor vessel water level is continuously monitored by indicators or recorders for normal, transient and accident conditions. The monitors used to provide automatic safety equipment initiation are arranged in a redundant array with two instruments in each of two or more independent electronic divisions.

The range of reactor vessel water level from below the top of the active fuel area up to the top of the vessel is covered by a combination of narrow- and wide-range instruments. Level is indicated and/or recorded on eight separate channels in the control room. These level indicators include: two -150" to +60" indicators, one 0" to 400" indicator, one -100" to +200" indicator, and one -100" to +200" recorder.

A separate set of narrow-range level instrumentation channels on separate condensing chambers provides reactor level control via the reactor feedwater system. This set also indicates or records in the control room on three level indicators with a range of 0" to 60", one channel of which feeds a recorder.

Safety-related systems or functions served by a safety-related reactor water level instrumentation are:

Reactor Core Isolation Coolant (RCIC) System
High-Pressure Coolant Injection (HPCI) System
Core Spray (CS) System
Residual Heat Removal/Low-Pressure Coolant Injection (RHR/LPCI)
Automatic Depressurization System (ADS)
Nuclear Steam Supply Shut-off System (Containment Isolation)

All systems automatically initiate on low reactor water level. In addition, the RCIC and HPCI systems shut down on high reactor vessel water level. In all cases, except the RCIC, these systems automatically restart if low reactor level is being reached.

Additional instrumentation, which the operator can use to determine changes in the reactor coolant inventory or other abnormal conditions are:

Drywell high pressure
Drywell high radioactivity levels
Suppression pool high temperature
Safety relief valve discharge high temperature
High feedwater flow rates

High main steam flow
High containment and equipment area temperatures
High differnential flow-reactor water cleanup system
Abnormal reactor pressure
High suppression pool water level
High drywell and containment sump fill and pump out rate
Valve steam leak-off high temperature

Instrumentation that can signal abnormal plant status, but is not necessarily indicative of loss of coolant are:

High neutron flux
High process monitor radiation levels
Main turbine status instrumentation
Abnormal reactor recirculation flow
High electrical current (amperes) to pump motors

Operations personnel have been instructed to compare all available parameters which would indicate abnormal conditions via discrepancy in reactor coolant inventory prior to overriding any automatically actuated safety system.

Item 5

Review the action directed by the operating procedures and training instructions to ensure that:

a. Operators do not override automatic actions of engineered safety features, unless continued operation of engineered safety features will result in unsafe plant conditions (e.g., vessel integrity).

b. Operators are provided additional information and instructions to not rely upon vessel level indication alone for manual actions, but to also examine other plant parameter indications in evaluating plant conditions.

CP&L Response

a. A preliminary review of operating procedures and training information has been conducted with respect to not overriding automatic actions of engineered safety features. This review has not identified any problems with these procedures. In addition, plant operating personnel have been specifically instructed on the potential consequences of overriding safety systems and to make a careful evaluation of all available supporting instrumentation prior to taking such action.

b. A preliminary review of operating procedures and training instructions has been conducted concerning the indications available to the operator other than vessel level indication for initiation of manual actions. This review has not identified any problems with these operating procedures. A number of other indications are available to the operator in addition to reactor vessel level to determine changes in reactor coolant inventory as described in Item 4. The availability of these other indications has been stressed to the operating personnel.

Item 6

Review all safety-related valve positions, positioning requirements and positive controls to assure that valves remain positioned (open or closed) in a manner to ensure the proper operation of engineered safety features. Also review related procedures, such as those for maintenance, testing, plant and system start-up, and supervisory periodic (e.g., daily/shift checks) surveillance to ensure that such valves are returned to their correct positions following necessary manipulations and are maintained in their proper positions during all operational modes.

CP&L Response

A review of safety-related valves and their positioning requirements has been performed. This review encompassed an evaluation of the valves in the various systems, with respect to the normal operating positioning, the position in an emergency, the controlling function for automatic operation of the valves, and whether a manual override exists.

The following systems were included in the review of the safety-related valves and their positioning requirements:

a. Low-pressure coolant injection (residual heat removal and core spray).

b. High-pressure coolant injection.

c. Reactor core isolation cooling.

d. Primary containment isolation.

e. Containment atmospheric control.

f. Service water.

g. Reactor instrument penetrations.

h. Standby gas treatment.

All valves required for proper operation of the engineered safety features, which must reposition, receive an automatic safety system protective action signal.

Valves which are not required to reposition either receive an automatic signal or are keylocked in their proper engineered safety features position, except reactor instrument penetration valves which have indicating lights on reactor-turbine generator board to indicate valve position.

Engineered safety features valves, which are required to be repositioned after performing engineered safety features function, are provided with an override capability which allows operator to manually reposition the valve.

There are currently three procedures which relate to the positioning of valves in safety-related systems following maintenance and/or testing. In addition, the "Clearance Procedure" requires a special review by the shift foreman for safety systems before removing the "clearance." A preliminary review of the "Periodic Test" Procedures reveals that there does exist a requirement for verification of proper valve position at the conclusion of the test.

Item 7

Review your operating modes and procedures for all systems designed to transfer potentially radioactive gases and liquids out of the primary containment to assure that undesired pumping, venting or other release of radioactive liquids and gases will not occur inadvertently.

In particular, ensure that such an occurrence would not be caused by the resetting of engineered safety features instrumentation. List all such systems and indicate:

a. Whether interlocks exist to prevent transfer when high radiation indication exists, and

b. Whether such systems are isolated by the containment isolation signal.

c. The basis on which continued operability of the above features is assured.

CP&L Response

7a. & 7b.

Potentially radioactive gases are transferred from containment through the Containment Atmospheric Control (CAC) System valves. Then, based on activity level, these gases are either passed through the Standby Gas Treatment (SGT) System filters or released to the atmosphere via the plant stack. All but two of the CAC containment isolation valves close on the following signals:

1. Drywell high pressure (2 PSIG) or,

2. Reactor low water (+ 12.5") or,

3. Reactor building vent high radiation

The two valves which do not close on the above signals are CAC V16 and V17. These valves are normally closed isolation valves upstream of the containment to reactor building vacuum breakers CAC X20A and X20B. CAC V16 and V17 are required to automatically open when a negative pressure condition exists inside containment so that the vacuum breakers can perform their design function. Remote position indication for V16 and V17 is monitored in the control room.

In order to permit Post-LOCA venting of the containment, four of the CAC containment isolation valves are provided with the

capability to manually override the automatic closure signal. The remaining CAC valves cannot be opened as long as an isolation signal exists.

In addition to full override via actuation of the override switch, these four valves may be opened directly from the control switch even when an automatic closure signal is present. However, when the valve is fully opened, it will automatically cycle shut.

In the reactor building, bypassing of the SCT filters is prevented by closure of the filter bypass valves simultaneously with the CAC containment isolation valves. In addition, the normal flow path to the filters is closed by these same signals. Post-LOCA venting of the containment through the SGT filters is controlled via two parallel 1/2" remote control valves which are not interlocked or closed by high radiation.

The transfer of potentially radioactive liquids out of the primary containment is accomplished by the drywell floor and equipment drain system. Interlocks, however, are provided that will automatically close the drywell drains, containment isolation valves and trip the pump and prevent them from being started automatically or manually if any of the following conditions exists:

1. Reactor level low (+ 12.5" H_2O DEC.)

2. Drywell pressure high (2 PSIG)

3. Drain isolation valves closed (on loss of air or power)

Once the system has been isolated, the operator cannot activate the pumps until the isolation signal has been cleared and the operator manually resets the drain transfer valves. Manual reset is also required before the system can be returned to normal automatic operation. No interlocks exist to prevent drains transfer due to the presence of a high radiation signal.

7c.

The existing BSEP Technical Specifications require isolation surveillance testing of the CAC and the drywell floor and equipment drain system to verify operability.

Item 8

Review and modify as necessary your maintenance and test procedures to ensure that they require:

a. Verification, by test or inspection, of the operability of redundant safety-related systems prior to the removal of any safety-related system from service.

b. Verification of the operability of all safety-related systems when they are returned to service following maintenance or testing.

c. Explicit notification of involved reactor operational personnel whenever a safety-related system is removed from and returned to service.

CP&L Response

8a.

The BSEP Technical Specifications require "periodic test" of all safety-related and redundant safety-related systems to verify operability. The results of these tests are used to assure operability of redundant systems before removing a safety-related system from service.

8b.

Existing "Operating Work Procedures" require verification of operability of all safety-related systems before returning them to service following maintenance.

8c.

A preliminary review of the "Periodic Tests Procedures" indicates that the shift foreman is notified whenever a safety-related system is removed from and returned to service.

Item 9

Review your prompt reporting procedures for NRC notification to assure that NRC is notified within one hour of the time the reactor is not in a controlled or expected condition of operation. Further, at that time an open continuous communication channel shall be established and maintained with NRC.

CP&L Response

A review of the BSEP Technical Specifications regarding "prompt notification" reveals that, for certain problems identified in the Technical Specifications, prompt notification of the NRC must be accomplished within 24 hours. In order to comply with the one-hour notification requirement, CP&L will work with the NRC to define those conditions where "the reactor is not in a controlled or expected condition of operation."

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Item 10

Review operating modes and procedures to deal with significant amounts of hydrogen gas that may be generated during a transient or other accident that would either remain inside the primary system or be released to the containment.

CP&L Response

A preliminary review of operating modes and procedures to deal with significant amounts of hydrogen gas has been performed.

During normal operation, the reactor pressure vessel dome is filled with steam which flows to the turbine. During reactor isolation, the dome is automatically vented through the safety-related valves to the suppression pool. In addition, the reactor pressure vessel head has a vent line with a valve remotely operated from the control room.

In the event of significant hydrogen release to the primary containment, the containment atmosphere dilution system maintains hydrogen below flammability. In addition, there are several other systems such as the containment atmospheric monitoring system, hydrogen recombiners, and containment purge via standby gas treatment which can be used to provide long-term hydrogen control.