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UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D. C. 20555

October 26, 1979

Docket Nos.: 50-325
50-324

Mr. J. A. Jones, Executive Vice-President
Carolina Power and Light Company
336 Fayetteville Street
Raleigh, North Carolina 27602

Dear Mr. Jones:

SUBJECT: NRC STAFF EVALUATION OF CP&L RESPONSES TO IE BULLETIN 79-08 FOR BRUNSWICK STEAM ELECTRIC PLANT, UNITS 1 AND 2

We have completed our review of the information that you provided in your letters dated April 23 and May 15, 1979 in response to IE Bulletin 79-08 for the Brunswick Steam Electric Plant, Units 1 & 2. We have also completed our review of the supplemental information that you provided in your letter of August 3, 1979.

We have concluded that you have taken the appropriate actions to meet the requirements of each of the eleven action items identified in IE Bulletin 79-08. A copy of our evaluation is enclosed.

However, the NRC staff review of the Three Mile Island, Unit 2 (TMI-2) accident is continuing and other corrective actions may be required at a later date. In this regard, the Bulletins and Orders Task Force is conducting a generic review of operating boiling water reactor plants. In addition, the Lessons Learned Task Force is identifying and evaluating those safety concerns originating with the TMI-2 accident that require licensing actions for operating plants and pending operating licenses and construction permit applications. Specific requirements for your facility that result from these and other TMI-2 investigations will be addressed to you in separate correspondence.

Sincerely,

Tippolito

Thomas A. Ippolito, Chief
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Enclosure:
NRC Staff Evaluation

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EVALUATION OF LICENSEE'S RESPONSES

TO

IE BULLETIN 79-08

CAROLINA POWER & LIGHT COMPANY

BRUNSWICK STEAM ELECTRIC PLANT, UNITS 1 AND 2

DOCKET NOS. 50-325 AND 50-324

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Introduction

By letter dated April 14, 1979, we transmitted IE Bulletin 79-08 to Carolina Power & Light Company (CP&L or the licensee). IE Bulletin 79-08 specified actions to be taken by the licensee to avoid occurrence of an event similar to that which occurred at Three Mile Island, Unit 2 (TMI-2) on March 28, 1979. By letter dated April 23, 1979, CP&L provided responses to action items 1 through 10 of IE Bulletin 79-08 for the Brunswick Steam Electric Plant, Units 1 & 2 (BSEP 1 & 2). CP&L supplemented this response by a letter dated May 15, 1979 to provide the response to action item 11 of IE Bulletin 79-08.

The NRC staff review of the CP&L responses led to the issuance of requests for additional information regarding the CP&L responses to certain action items of IE Bulletin 79-08. These requests were contained in a letter dated July 20, 1979. By letter dated August 3, 1979, CP&L responded to the staff's requests for additional information.

The CP&L responses to IE Bulletin 79-08 provided the basis for our evaluation presented below.

Evaluation

Each of the 11 action items requested by IE Bulletin 79-08 is repeated below followed by our criteria for evaluating the response, a summary of the licensee's response and our evaluation of the response.

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 led to the eventual core damage; and (3) the necessity to systematically analyze plant conditions and parameters and take appropriate corrective action.

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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.

The licensee's response was evaluated to determine that (1) the scope of review was adequate, (2) operational personnel were properly instructed and (3) personnel participation in the review was documented in plant records.

The licensee's response dated April 23, 1979 described the composition of an investigative team established to perform the required review, including (1) the sequence of events which occurred at TMI, (2) operating errors and their significance and (3) a systematic evaluation to determine if similar problems could occur at BSEP 1 and 2. The status of instruction received by operational personnel was provided. A supplemental response dated August 3, 1979 confirmed that personnel participation in the required reviews had been documented in the plant records.

We conclude that the licensee's scope of review, instructions to operating personnel and documented participation satisfies the intent of IE Bulletin 79-08, Item 1.

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.

The licensee's response was evaluated to verify that containment isolation initiation design and procedures had been reviewed to assure that (1) manual or automatic initiation of containment isolation occurs on automatic initiation of safety injection and (2) all lines (including those designed to transfer radioactive gases or liquids) whose isolation does not degrade cooling capability or needed safety features were addressed.

The licensee's April 23, 1979 response noted that a review of the primary containment isolation design had been completed. This review verified that a safety injection signal will automatically initiate containment isolation of all valves whose isolation does not degrade needed safety features or cooling capability. In addition, the licensee stated that the applicable operating emergency instructions were reviewed to assure that the operators are instructed to verify that all automatic actions do occur. In a supplemental response dated August 3, 1979, the licensee confirmed that the review included all lines penetrating primary containment and that the review included the applicable emergency instructions and operating procedures. No changes to design or procedures were reported by the licensee.

We conclude that the licensee's review of containment isolation initiation design and procedures satisfy the intent of IE Bulletin 79-08, Item 2.

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.

The licensee's response was reviewed to assure that (1) it described the automatic and manual actions necessary for the proper functioning of the auxiliary heat removal systems when the main feedwater system is not operable and (2) the procedures for any necessary manual actions were described in summary form.

The licensee, in its response dated April 23, 1979, stated that following a loss of feedwater and reactor scram, a low water level signal would automatically initiate main steam line isolation valve closure and also initiate operation of the high pressure coolant injection system and the reactor core isolation cooling system. These systems would inject water into the reactor vessel until a high water level signal trips the systems. We acknowledge the capability of these systems to provide the required heat removal action.

Following a high reactor water level trip, the high pressure coolant injection system will automatically reinitiate when reactor water level again decreases to low water level. The reactor core isolation cooling system must be manually reset by the operator in the control room before it will automatically reinitiate after a high water trip.

The high pressure coolant injection and reactor core isolation cooling systems have redundant supplies of water. Normally they take suction from the condensate storage tank. The high pressure coolant injection system suction will automatically transfer from the condensate storage tank to the suppression pool if the condensate storage tank water is depleted or the suppression pool water level increases to a high level. The reactor core isolation cooling system suction must be manually transferred from the condensate storage tank to the suppression pool using controls located in the main control room. This action would need to be taken when control room alarms indicate condensate storage tank low water level or suppression pool high water level.

The operator can manually initiate the high pressure coolant injection and reactor core isolation cooling systems from the control room before automatic initiation from low water 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 determine that these systems are delivering water to the reactor vessel by verifying the following:

- Reactor water level increases when systems initiate.
- System flows using flow indicators in the control room.
- Control room position indication of motor-operated valves.

Therefore, the high pressure coolant injection and reactor core isolation cooling systems can maintain reactor water level at high reactor pressures.

When pressure decreases, low pressure systems such as the core spray or low pressure coolant injection systems can maintain water level. If for some reason, the high pressure coolant injection and reactor core isolation cooling systems do not maintain reactor water level, the automatic depressurization system will initiate depressurization of the reactor quickly such that the core spray and low pressure coolant injection systems can immediately begin to cool and flood the core.

We conclude that the licensee's procedural summary of automatic/manual actions necessary for the proper functioning of auxiliary heat removal systems used when the main feedwater system is inoperable satisfies the intent of IE Bulletin 79-08, Item 3.

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.

The licensee's response was evaluated to determine that (1) all uses and types of vessel level indication for both automatic and manual initiation of safety systems were addressed, (2) it addressed other instrumentation available to the operator to determine changes in reactor coolant inventory and (3) operators were instructed to utilize other available information to initiate safety systems.

The licensee's April 23, 1979 response stated that 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 inches indicators
- one zero to +400 inches indicator
- one -100 to +200 inches indicator
- one -100 to +200 inches recorder

A separate set of narrow-range level instrumentation channels, each with a separate condensing chamber provides reactor level control via the reactor feedwater system. Reactor water level is indicated in the control room on three level indicators with a range of zero to +60 inches, with one of these channels recorded.

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

- Reactor core isolation cooling system
- High pressure coolant injection system
- Core spray system
- Residual heat removal/low pressure coolant injection system
- Automatic depressurization system
- Nuclear steam supply shut-off system

The above systems automatically initiate on low reactor water level. In addition, the reactor core isolation cooling and high pressure coolant injection systems shut down on reactor high water level. Except for the reactor core isolation cooling system these systems automatically restart if reactor low water level is sensed by the instrumentation.

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

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- Safety-relief valve discharge high temperature
- High feedwater flow rates
- High main steam flow
- High containment and equipment area temperatures
- High differential 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

The following instrumentation can signal abnormal plant status but is not necessarily indicative of loss of coolant:

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

Operations personnel have been instructed to compare all available parameters which would indicate abnormal conditions prior to overriding any automatically actuated safety system.

We conclude that the licensee's description of the uses and types of reactor vessel level/inventory instrumentation and instructions to operators regarding the use of this information satisfies the intent of IE Bulletin 79-08, Item 4.

5. Review the actions 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 evalating plant conditions.

The licensee's response was evaluated to determine that (1) it addressed the matter of operators improperly overriding the automatic actions of engineered safety features, (2) it addressed providing operators with additional information and instructions to not rely upon vessel level indication alone for manual actions and (3) that the review included operating procedures and training instructions.

The licensee in its April 23, 1979 response stated that a preliminary review of operating procedures and training information had been conducted with respect to not overriding automatic actions of engineered safety features and that the review did not identify 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.

The licensee also reported that a preliminary review of operating procedures and training instructions was conducted concerning the indications available to the operator other than vessel level indication for initiation of manual actions. This review did not identify 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 the licensee's response to IE Bulletin 79-08, Item 4. The licensee reported that the availability of these other indications has been stressed to the operating personnel.

We conclude that the licensee's review of operating procedures and training instructions satisfies the intent of IE Bulletin 79-08, Item 5.

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.

The licensee's response was evaluated to assure that (1) safety-related valve positioning requirements were reviewed for correctness, (2) safety-related valves were verified to be in the correct position and (3) positive controls were in existence to maintain proper valve position during normal operation as well as during surveillance testing and maintenance.

The licensee's response dated April 23, 1979 described the review of safety-related valve positioning requirements. A supplemental response dated August 3, 1979 indicated that safety-related valves were verified to be in their correct positions by the performance of valve line-up checks. The supplemental response confirmed that procedural controls have been reviewed and determined to be adequate to maintain proper valve position during operation, test and maintenance.

We conclude that the licensee's review of safety-related valve positioning requirements, valve positions and positive controls to maintain proper valve positions satisfies the intent of IE Bulletin 79-08, Item 6.

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.

The licensee's response was evaluated to determine that (1) it addressed all systems designed to transfer potentially radioactive gases and liquids out of primary containment, (2) inadvertent releases do not occur on resetting engineered safety features instrumentation, (3) it addressed the existence of interlocks, (4) the systems are isolated on the containment isolation signal, (5) the basis for continued operability of the features was addressed and (6) a review of the procedures was performed.

In the April 23, 1979 response, the licensee reported that potentially radioactive gases are transferred from containment through the containment atmospheric control system valves. Then, based on activity level, these gases are either passed through the standby gas treatment system filters or released to the atmosphere via the plant stack. All but two of the containment atmospheric control system containment isolation valves close on the following signals:

- Drywell high pressure (two pounds per square inch, gauge) or,
- Reactor low water level (+ 12.5 inches) or,
- 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-loss of coolant accident venting of the containment, some of the containment atmospheric control system isolation valves are provided with the capability to manually override the automatic closure signal. The remaining containment atmospheric control system valves cannot be opened as long as an isolation signal exists.

In addition to full override via actuation of the override switch, these 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 standby gas treatment system filters is prevented by closure of the filter bypass valves simultaneously with the containment atmospheric control system containment isolation valves. In addition, the normal flow path to the filters is closed by these same signals. Post-loss-of-coolant accident venting of the containment through the standby gas treatment system filters is controlled via two parallel one-half inch 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 drain containment isolation valves and trip the pumps and prevent them from being started automatically or manually if any of the following conditions exists:

- Reactor low water level (+ 12.5 inches)
- Drywell high pressure (two pounds per square inch, gauge)
- Drain isolation valves closed (on loss of air or power)

Ohce 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 drywell floor and equipment drains system transfer due to the presence of a high radiation signal.

The licensee also reported that the existing Technical Specifications require isolation surveillance testing of the containment atmospheric control system and the drywell floor and equipment drain system to verify operability.

The licensee in its supplemental response of August 3, 1979 reported that present logic associated with the containment atmospheric control system will allow various valves to open when the isolation signal is cleared if the control switch is in the open position (i.e., the valve was open when the isolation occurred). Other valves in the system use spring-return-to-normal type switches and will remain closed when the isolation clears.

The licensee also stated that all appropriate procedures are being revised to instruct the operator to place the non-spring-return-to-normal switches to "close" when any containment atmospheric control system isolation signal is received. The licensee reported that these revisions will be completed by August 31, 1979. A plant modification is being prepared to eliminate the problem of these valves opening without operator action when the isolation signal clears. The licensee required that the plant modification will be completed by November 1, 1979 if no problem is experienced in component procurement.

We conclude that the licensee's review of systems designed to transfer radioactive gases and liquids out of primary containment to assure that undesired pumping, venting, or other release of radioactive liquids and gases will not occur satisfies the intent of IE Bulletin 79-08, Item 7.

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 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.

The licensee's response was evaluated to determine that operability of redundant safety-related systems is verified prior to the removal of any safety-related system from service. Where operability verification appeared only to rely on previous surveillance testing within Technical Specification intervals, we asked that operability be further verified by at least a visual check of the system status to the extent practicable, prior to removing the redundant equipment from service. The response was also evaluated to assure provisions were adequate to verify operability of safety-related systems when they are returned to service following maintenance or testing. We also checked to see that all involved reactor operational personnel in the oncoming shift are explicitly notified during shift turnover about the status of systems removed from or returned to service since their previous shift.

The licensee's response dated April 23, 1979 indicated that operability of redundant safety-related systems was verified through reliance on Technical Specification periodic tests. A supplemental response dated August 3, 1979 committed to a visual check of the system on the control board, to be implemented by revising the Limiting Condition for Operation Evaluation Checksheet. The licensee reported that the revision was to be in effect by September 30, 1979.

The April 23, 1979 response indicated that Operating Work Procedures require verification of system operability prior to returning a safety-related system to service. The August 3, 1979 supplement indicated that Administrative Procedure 4.1.12 provided for explicit notification of the entire watch regarding systems removed from or returned to service. Upon inspection at Brunswick 1 & 2 by the NRC Resident Inspector, we found that AP 4.1.12 required revision to assure that all involved reactor operational personnel in the oncoming shift are explicitly notified about system status since their last shift. The licensee has committed to revise the procedure to achieve the intended result by September 30, 1979.

We conclude that the licensee's review and modification of maintenance, test and administrative procedures to assure the availability of safety-related systems and operational personnel knowledge of system status satisfies the intent of IE Bulletin 79-08, Item 8.

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.

The licensee's response was evaluated to determine that (1) prompt reporting procedures required or were to be modified to require that the NRC is notified within one hour of the time the reactor is not in a controlled or expected condition of operation and (2) procedures required or were to be modified to require the establishment and maintenance of an open continuous communication channel with the NRC following such events.

The licensee, in its April 23, 1979 response, reported that it reviewed the Technical Specifications regarding "prompt notification" and 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, the licensee indicated it would work with the NRC to define those conditions where "the reactor is not in a controlled or expected condition or operation."

In its supplemental response dated August 3, 1979 the licensee stated that it will notify the NRC within one hour whenever it is determined that the reactor is not in a controlled or expected condition of operation. Furthermore, it will establish an open, continuous communication channel to the NRC using the recently established designated phone network.

We conclude that the licensee's response satisfies the intent of IE Bulletin 79-08, Item 9.

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.

The licensee's response was evaluated to determine if it described the means or systems available to remove hydrogen from the primary system as well as the treatment and control of hydrogen in the containment.

The licensee in its April 23, 1979 response stated that it reviewed its operating modes and procedures that address controlling significant amounts of hydrogen.

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-relief valves to the suppression pool. In addition, the reactor 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 concentration. In addition, there are other systems such as the containment atmospheric monitoring system and containment purge by means of the standby gas treatment system which can be used to assist in long-term hydrogen control.

We conclude that the licensee's response satisfies the intent of IE Bulletin 79-08, Item 10.

11. Propose changes, as required, to those technical specifications which must be modified as a result of your implementing the items above.

The licensee's response was evaluated to determine that a review of the Technical Specifications had been made to determine if any changes were required as a result of implementing Items 1 though 10 of IE Bulletin 79-08.

The licensee reported in its letter dated May 15, 1979 that its review has shown that no changes to the Technical Specifications are required. The licensee also noted that in its continuing review, should modifications to the Technical Specifications be required, they will be proposed in a timely manner.

We conclude that the licensee's response satisfies the intent of IE Bulletin 79-08, Item 11.

Conclusion

Based on our review of the information provided by the licensee to date, we conclude that the licensee has correctly interpreted IE Bulletin 79-08. The actions taken demonstrate the licensee's understanding of the concerns arising from the TMI-2 accident in reviewing their implementation on BSEP 1 & 2 operations, and provide added assurance for the protection of the public health and safety during the operation of BSEP 1 & 2.

References

1. IE Bulletin 79-05, dated April 1, 1979.

2. IE Bulletin 79-05A, dated April 5, 1979.

3. IE Bulletin 79-08, dated April 14, 1979.

4. CP&L letter, B. Furr to J. O'Reilly, dated April 23, 1979.

5. CP&L letter, B. Furr to J. O'Reilly, dated May 15, 1979.

6. NRC staff letter, T. Ippolito to J. Jones, dated July 20, 1979.

7. CP&L letter, E. Utley to J. O'Reilly, dated August 3, 1979.