Buchanan, NSIC

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

December 21, 1979

Docket Nos. 50-321
and 50-366

Mr. Charles F. Whitmer
Vice President - Engineering
Georgia Power Company
P. O. Box 4545
Atlanta, Georgia 30302

Dear Mr. Whitmer:

SUBJECT: NRC STAFF EVALUATION OF GEORGIA POWER COMPANY RESPONSES TO IE BULLETIN 79-08 FOR HATCH NUCLEAR PLANT UNITS NOS. 1 AND 2

We have completed our review of the information that you provided in your letters dated April 25 and May 9, 1979 in response to IE Bulletin 79-08 for the E. I. Hatch Nuclear Plants. We have also completed our review of the supplemental information that you provided in your letters of August 10, 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.

As you know, 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. For example, the Bulletins and Orders Task Force is conducting a generic review of operating boiling water reactor plants. Specific requirements for your facility that result from this and other TMI-2 investigations will be addressed to you in separate correspondence.

Sincerely,

Thomas A. Ippolito, Chief
Operating Reactors Branch #3
Division of Operating Reactors

Enclosure:
NRC Staff Evaluation

cc w/enclosure:
See next page

1701 219

8001080 035

cc:

G. F. Trowbridge, Esquire
Shaw, Pittman, Potts and Trowbridge
1800 M Street, N. W.
Washington, D. C. 20036

Ruble A. Thomas
Vice President
P. O. Box 2625
Southern Services, Inc.
Birmingham, Alabama 35202

Ozen Batum
P. O. Box 2625
Southern Services, Inc.
Birmingham, Alabama 35202

Mr. William Widner
Georgia Power Company
Power Generation Department
P. O. Box 4545
Atlanta, Georgia 30302

Mr. L. T. Gucwa
Georgia Power Company
Engineering Department
P. O. Box 4545
Atlanta, Georgia 30302

Appling County Public Library
Parker Street
Baxley, Georgia 31413

Mr. R. F. Rogers
U. S. Nuclear Regulatory Commission
P. O. Box 710
Baxley, Georgia 31513

EVALUATION OF LICENSEE'S RESPONSES

TO

IE BULLETIN 79-08

GEORGIA POWER COMPANY

EDWIN I. HATCH NUCLEAR PLANT, UNITS 1 AND 2

DOCKET NOS. 50-321 AND 50-366

Introduction

By letter dated April 14, 1979, we transmitted Office of Inspection and Enforcement (IE) Bulletin 79-08 to Georgia Power Company (GPC 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 25, 1979, GPC provided initial responses to Action Items 1 through 10 of IE Bulletin 79-08 for the Hatch Nuclear Plant, Units 1 and 2 (HNP 1 & 2). GPC supplemented this response by a letter dated May 9, 1979 to provide the response to Action Item 11 of IE Bulletin 79-08, and to revise and clarify Items 1 through 10.

The NRC staff review of the GPC responses led to the issuance of requests for additional information regarding the GPC 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 10, 1979, GPC responded to the staff's requests for additional information.

The GPC responses to IE Bulletin 79-08 provided the basis for our evaluation presented below. In addition, the actions taken by the licensee in response to the bulletin requirements and subsequent NRC requests were verified through inspections by IE inspectors.

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.

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 25, 1979 indicated that the required training had been completed and documented on training data sheets, except for three licensed personnel who were not available at the time the training was conducted. The licensee's revised response dated May 9, 1979 indicated that the required training had been completed and documented for all licensed personnel.

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 May 9, 1979 response identified nitrogen inerting make-up lines in Unit 1 and two-inch purge bypass lines in Units 1 and 2 that do not automatically isolate upon initiation of safety injection. The Unit 1 containment nitrogen inerting system does not automatically isolate so as to enable inerting the primary containment atmosphere with nitrogen during post-accident conditions. In a supplemental response dated August 10, 1979, the licensee committed to modify the design of the two-inch purge bypass lines in both units to provide for automatic isolation upon safety injection.

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's response dated May 9, 1979 described the automatic actions required when the main feedwater system is not operable. The reactor would automatically scram when the reactor vessel water level decreased to +12½ inches. Should the level continue to decrease to -38 inches, the high

pressure coolant injection (HPCI) system and the reactor core isolation cooling (RCIC) system will initiate and inject into the reactor vessel. We acknowledge the capability of these systems to provide the required heat removal action. The operator can manually secure the HPCI system when reactor vessel level is confirmed to be above the low level scram point (+12½ inches). When reactor water level is restored and stabilized at +37 inches (normal level), the operator can secure the RCIC system.

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 May 9, 1979 response listed the various control room instruments used for monitoring and recording vessel level. The indicated ranges of these instruments vary from 200 to 900 inches as measured from the bottom head drain. In addition to the control room instruments, the response listed several local instruments located in the reactor building, most of which do not require electrical power to operate. The indicated ranges of these instruments vary from 200 to 577 inches as measured from the bottom head drain. Manual initiation of safety systems can be based upon information from 19 separate indicators, 15 of which on Unit 1 and 14 of which on Unit 2 do not

require an external power source. Additional instrumentation which the operator can use to determine changes in reactor coolant inventory was addressed in a supplemental response dated August 10, 1979. Licensed personnel were instructed in the use of all available instrumentation.

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 May 9, 1979 response stated that HNP operating procedures caution the plant operators to ensure that the water level as well as the other plant parameters are normal before changing the system from its automatic function. The supplemental response dated August 10, 1979 stated that the operators are instructed to not rely upon vessel level indication alone for manual action. These instructions are provided by the training department and documented attendance is required.

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 May 9, 1979 described the review of safety-related valve positioning requirements. Complete valve line-up checks are performed for safety-related valves prior to startup from an extended outage. (Both plants were in cold shutdown when the initial response to the bulletin was prepared.) Administrative controls governing normal operation, surveillance testing, and maintenance were described. The supplemental response dated August 10, 1979 confirmed that valve position and locked valve status are documented on data sheets.

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 May 9, 1979 response, the licensee reported that potentially radioactive gases are transferred from containment through the drywell vent and purge system. For Unit 1, the valves are part of the containment atmospheric dilution (CAD) system and are opened by procedure under continuous operator supervision. The operator is required to close the valves upon receipt of an isolation signal. On Unit 2, the normal vent valves are used and they will automatically shut upon receipt of a reactor building or refueling area high radiation signal, or a containment isolation signal.

In the supplemental response dated August 10, 1979, the licensee committed to install plant modifications to isolate the valves that could inadvertently transfer radioactive gases from the containment upon reset of an isolation signal. This modification is to be completed during the first outage of sufficient duration subsequent to receipt of engineering and materials.

In the May 9, 1979 response, the licensee reported that the radwaste sump subsystem is provided to transfer potentially radioactive liquids out of the primary containment. The pump discharge valves are automatically closed by the containment isolation signals; however, they could re-open upon isolation

signal reset, thus setting the stage for an inadvertent release. Administrative controls are presently in place to require an operator to close these valves prior to resetting. In addition, an additional reset for these radwaste isolation valves will be installed at the next outage of sufficient duration subsequent to receipt of engineering and materials. The additional reset will preclude an inadvertent release path from being established. The annunciator response procedure requires the operator to manually close these valves on reactor building high radiation or refueling area high radiation (as well as containment isolation).

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 May 9, 1979 indicated that operability of redundant safety-related systems was verified by inspection of clearance sheets and tags, verbal communications, and visual examination of the system inoperability status board. Some revisions to maintenance and test procedures were found to be necesary to assure provisions were adequate to verify operability of safety-related systems when they are returned to service following maintenance or testing. The licensee committed to complete the necessary procedural modifications prior to startup of the applicable unit.

The supplemental response dated August 10, 1979 described the methods used to assure 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.

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 reported in its May 9, 1979 response that the Technical Specifications and plant procedures required NRC notification within 24 hours. However, a commitment was made to modify the plant emergency procedures by July 1, 1979 to require plant personnel to establish and maintain an open continuous communication channel with the NRC within one hour after determining that an emergency condition exists. The supplemental response dated August 10, 1979 indicated that the communication channel would be used where it is apparent that immediate NRC attention is necessary. The licensee confirmed its intent to notify the NRC within one hour whenever the reactor is operating in an uncontrolled or unexpected condition by telephone on November 9, 1979.

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 May 9, 1979 response stated that it reviewed plant procedures which 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. Several discrepancies were identified and corrected in the post-accident venting procedure for HNP-2 which uses two 100-percent capacity hydrogen recombiner systems designed to control and recombine the hydrogen buildup within containment. The HNP-1 containment is inerted with nitrogen and would employ the containment nitrogen inerting system to control the post-accident hydrogen buildup within containment. Both units are equipped with the normal vent and relief lines to control hydrogen buildup in the primary system.

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 9, 1979 that its review has shown that no changes to the Technical Specifications are required. The licensee stated in subsequent telephone conversations, that 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 HNP 1 & 2 operations, and provide added assurance for the protection of the public health and safety during the operation of HNP 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. GPC letter, W. Widner to J. O'Reilly, dated April 25, 1979.

5. GPC letter, W. Widner to J. O'Reilly, dated May 9, 1979.

6. NRC staff letter, T. Ippolito to C. Whitmer, dated July 20, 1979.

7. GPC letter, R. Kelly to T. Ippolito, dated August 10, 1979.