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

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

SUPPORTING AMENDMENT NO. 40 TO FACILITY OPERATING LICENSE NO. NPF-5

GEORGIA POWER COMPANY
OGLETHORPE POWER CORPORATION
MUNICIPAL ELECTRIC AUTHORITY OF GEORGIA
CITY OF DALTON, GEORGIA

EDWIN I. HATCH NUCLEAR PLANT, UNIT NO. 2

DOCKET NO. 50-366

1.0 Introduction

On May 4, 1984, the licensee discovered a discrepancy between containment isolation valve actuation setpoints specified in the technical specifications and the installed actuation setpoints. The discrepancy was identified for 10 containment isolation valves in the RHR and core spray systems.

The valves and their respective systems are:

2E11-F011 A, B	RHR Heat Exchanger Drain Isolation Valves
2E11-F026 A, B	RHR Heat Exchanger Drain Isolation Valves
2E11-F016 A, B	Drywell Spray Isolation Valves
2E11-F028 A, B	Torus Spray/Suppression Pool Cooling Isolation Valves
2E21-F015 A, B	Core Spray Full Flow Test Line Isolation Valves

Each of these valves are normally closed and receive automatic confirmatory isolation signals to close. The technical specifications list each of these valves as receiving a Group 2 isolation signal. Group 2 isolation signals are actuated by either high drywell pressure or low reactor vessel water level. However, the as-built auto-closure setpoint for these valves was discovered to be either a high drywell pressure or a low-low-low reactor vessel water level.

Hatch Unit 2 is currently completing a refueling outage and is scheduled for plant restart on August 22, 1984. Upon identifying the discrepancy between the technical specifications and the as-built plant, the licensee declared the 10 isolation valves to be inoperable. Technical Specification 3.6.3 (Primary Containment Isolation Valves) would permit plant restart with the isolation valves in question declared inoperable as long as they were closed and deactivated. This action, however, would prevent both trains of the RHR system from operating in the safety-related suppression pool cooling mode. Technical Specification 3.6.2 (Suppression Pool Cooling) requires both trains of suppression pool cooling to be operable for plant startup. The current Technical Specification 3.6.3, which allows startup and continued operation with these containment isolation valves deactivated, was intended to be applicable to the containment isolation valves in general and does not reflect the importance of an individual valve's

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The Georgia Power Company's letters of August 6, 10, 14 and 16, 1984, requested a change to Technical Specification 3.6.3 which would result in the valves being returned to operable status and permit plant restart. Specifically, the licensee requested that the Technical Specifications be revised to identify the 10 valves in question as isolating upon either a high drywell pressure signal or a triple low reactor vessel level signal. This is the current as-built configuration. The licensee has concluded that the proposed change is primarily administrative and that it is consistent with the FSAR and the original engineering safety analysis.

2.0 EVALUATION

The licensee's basis for the proposed changes include:

1. The ECCS analysis is unaffected by the proposed changes. This is because the auto-closure verification signal assumed in both the FSAR and the architectural engineer's design analysis remains unchanged at the triple low level signal.

2. FSAR LOCA analyses show that, for a spectrum of postulated break sizes, containment isolation and ECCS actuation will be activated by a high drywell pressure signal before either a low or triple low reactor vessel water level signal is actuated. Therefore, the containment isolation function remains unchanged.

3. The proposed technical specification change is consistent with corresponding isolation signals of other recently licensed BWR facilities.

The proposed Technical Specification changes do not result in a hardware change but rather an administrative change so that the Technical Specification will reflect both the analyzed and as-built plant design. The staff's former analysis was on the basis of the actual configuration of the plant, and not the words of the Technical Specifications which are here changed.

We concur with the licensee's conclusion in that the proposed changes will not affect the ECCS performance or the isolation function. All 10 of the valves in question are normally closed and receive a verification signal to close at the same time as LPCI and core spray system actuation.

The licensee has examined other containment isolation valves receiving a Group 2 isolation signal and has verified that all affected valves have been identified.

With regard to the safety significance of isolating on a low versus triple low reactor vessel water level, the staff concludes that this is insignificant. The staff has accepted both low and triple low reactor vessel water level signals as isolation signals for corresponding valves on recently licensed BWR facilities. In addition, as pointed out by the licensee, high drywell pressure (2.0 psig) is predicted to occur prior to either of the water level setpoints.

Emergency Circumstances

GPC (the licensee's) plant personnel identified on May 4, 1984 an apparent discrepancy between the installed actuation setpoints for the 10 valves in question and the actuation setpoints referenced in the Technical Specifications. GPC requested its Architect/Engineer (A/E) to investigate and evaluate this identified discrepancy. On August 2, 1984, GPC received its A/E report which resulted in the licensee submitting its emergency amendment request of August 6, 1984 in order to avoid delay in Unit 2 startup.

Prior to requesting a Technical Specification change, the licensee investigated plant hardware design changes and modifications which could resolve the discrepancy. This option was estimated to result in a 22 week delay in startup due to time constraints in procuring certain hardware components. This request was received without sufficient time to permit prior notice and opportunity for public comment.

Final No Significant Hazards Consideration Determination

The Commission's regulations in 10 CFR 50.92 state that the Commission may make a final determination that a license amendment involves no significant hazards consideraitons if operation of the facility in accordance with the amendment would not:

(1) Involve a significant increase in the probability or consequences of an accident previously evaluated; or

(2) Create the possibility of a new or different kind of accident from any accident previously evaluated; or

(3) Involve a significant reduction in a margin of safety.

The information in this SE provides the basis for evaluating this license amendment against these criteria. Since the requested change does not affect the original design basis, plant operating conditions, the physical status of the plant, and dose consequences of potential accidents, the staff concludes that:

(1) Operation of the facility in accordance with the amendment would not significantly increase the probability or consequences of an accident previously evaluated.

(2) Operation of the facility in accordance with the amendment would not create the possibility of a new or different kind of accident from any accident previously evaluated.

(3) Operation of the facility in accordance with the amendment would not involve a significant reduction in a margin of safety.

Accordingly, we conclude that the amemdment to Facility Operating License NPF-5 requiring that certain containment isolation valves be automatically closed upon receipt of a low-low-low reactor water level signal involves no significant hazards considerations.

State Consultation

In accordance with the Commission's regulations, consultation was held with the State of Georgia by telephone. The State expressed no concern either from the standpoint of safety or of our no significant hazards consideration determination in view of the fact the changes makes the Technical Specifications consistent with the original signal design specifications and with the as-built systems.

3.0 ENVIRONMENTAL CONSIDERATIONS

The amendment involves a change in the installation or use of a facility component located within the restricted area as defined in 10 CFR Part 20 and a change to a surveillance requirement. The staff has determined that the amendment involves no significant increase in the amounts, and no significant change in the types, of any effluents that may be released offsite, and that there is no significant increase in individual or cumulative occupational radiation exposure. The Commission has made a final no significant hazards consideration finding with respect to this amendment. Accordingly, the amendment meets the eligibility criteria for categorical exclusion set forth in 10 CFR 51.22(c)(9). Pursuant to 10 CFR 51.22(b), no environmental impact statement or environmental assessment need be prepared in connection with the issuance of the amendment.

4.0 CONCLUSION

We have concluded, based on the considerations discussed above, that: (1) there is reasonable assurance that the health and safety of the public will not be endangered by operation in the proposed manner, and (2) such activities will be conducted in compliance with the Commission's regulations, and the issuance of the amendment will not be inimical to the common defense and security or to the health and safety of the public.

Dated: August 22, 1984

Principal Contributor: D. Pickett and E. Butcher, Jr.