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

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

SUPPORTING AMENDMENT NO. 121 TO FACILITY OPERATING LICENSE

NO. DPR-57

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

EDWIN I. HATCH NUCLEAR PLANT, UNIT NO. 1

DOCKET NO. 50-321

1.0 INTRODUCTION

By letter dated July 24, 1985 (NED-85-528), Georgia Power Company (GPC) proposed an amendment to the Hatch Nuclear Plant Unit 1 (HNP-1) Technical Specifications. Numerous change items are identified in the submittal that support the installation of the analog transmitter trip system (ATTS). The installation of the ATTS was previously reviewed and approved by the NRC in Amendment 103 to the HNP-1 Operating License.

The ATTS is a new design for portions of the system instrumentation of the Reactor Protective Systems (RPS) of Boiling Water Reactor (BWRs). It was developed by the General Electric Company (GE) and is being supplied as original equipment in later built BWRs (e.g., BWR 6). The design was adapted to the HNP-1 as a backfit. GE developed the ATTS to offset operating disadvantages of the digital sensor switches of the original safety system instrumentation. The principal objective of the ATTS is to improve sensor intelligence and reliability while enhancing testing procedures.

2.0 EVALUATION

Nomenclature Changes to the Technical Specifications

The ATTS modification replaces pressure, level, and temperature digital switches in the RPS with analog/trip unit combinations. The digital switches are identified in the instrument description of the current Technical Specification. The licensee proposes to change the instruments listed in the Technical Specification to reflect the installation of the new ATTS. This change is acceptable to the staff.

Modification of the Surveillance Frequency

The licensee proposes that the surveillance frequencies for the ATTS equipment be changed from that listed in the existing Technical Specification (for equipment replaced by the ATTS). The licensee proposes the following surveillance frequencies:

Once per shift for channel check
Once per month for channel functional test
Once per operating cycle for channel calibration

The channel check once per shift is a new requirement for the ATTS equipment. Such a test was not applicable for the mechanical switches replaced by the ATTS. The addition of this requirement is, therefore, a change toward the more conservative surveillance for the ATTS equipment.

The channel functional test required each month is either as conservative or more conservative than required by the existing Technical Specifications.

The channel calibration frequency of once per operating cycle is less conservative than the present HNP-1 requirement for calibrations which in most cases is once every 3 months. However, currently approved Technical Specifications for BWR6's, which utilize the ATTS system, provide channel calibrations based on a frequency of once per operating cycle. Once per operating cycle channel calibration frequencies have been previously approved for other ATTS instrumentation for Hatch Unit 1 (e.g., HNP-1 Operating License Amendment 103).

The GE report NEDE-22154-1 (the supporting GE document for the installation of the ATTS in Hatch Unit 1) recommends transmitter calibration once per operating cycle when the reactor is out of service for refueling. It also states that the operating cycle time is dependent on the reload fuel design which can be between 12 and 18 months.

The primary factor in setting the calibration surveillance frequency is the drift of the transmitters and trip units. The total loop accuracy and the total loop drift are added to obtain the trip setpoint. Setpoint drift is the only value that is extrapolated in the licensee's setpoint methodology. In many cases, the manufacturer's specifications only provide drift values for 6 to 12 month intervals. These values were extrapolated linearly to provide 18 to 24 month drift values for use in the Hatch setpoint calculations.

The licensee intends to evaluate the performance of the ATTS against the manufacturer's specifications and, if necessary, propose modifications to the surveillance frequencies specified in the Technical Specifications. The Definition of Surveillance Frequency states that the operating cycle interval as pertaining to instrument and electrical surveillance shall never exceed 15 months. Therefore, the proposed requirement for calibration once per operating cycle is a requirement to calibrate once per operating cycle or once per 15 months, whichever is the shorter interval. Current Standard Technical Specifications require calibration once per 18 months.

Based on the above information, the staff finds that Technical Specification changes requiring channel checks of the ATTS equipment once per shift and channel functional tests once per month of the ATTS equipment and channel calibration of the ATTS equipment once per operating cycle not to exceed 15 months are acceptable.

The reactor vessel water level, shroud water level and reactor pressure post-accident monitoring instruments all receive input from ATTS instruments. Because of this, the existing recorders and indicators are being replaced with qualified class 1E devices compatible with ATTS. Two new recorders are also being added.

The licensee proposed that the calibration frequency be changed for these instruments to once every operating cycle except that the recorders be calibrated once per 12 months. The manufacturer's recommended calibration for the recorders is once per 12 months for the recorders and once per 5 years for the indicators.

The staff finds that calibration of the recorders each 12 months is acceptable. The proposed revision of other calibration frequencies is to once each operating cycle (which as previously discussed in the section on modification of surveillance frequency cannot exceed 15 months). The staff finds that calibration of this equipment once per operating cycle not to exceed 15 months is acceptable.

Deletion of Drywell Pressure Sensors E11-N011A, B, C, D

The original design of HNP-1 has the high drywell pressure signals for ECCS coming from eight sensing devices. For example, E11-N011A, B, C, D (existing MPL numbers) provide signals to reactor heat removal (RHR), core spray (CS), and high pressure core injection (HPCI) systems; E11-N010A, B, C, D (existing MPL numbers) provide signals to the automatic depressurization system (ADS). This configuration is inconsistent with the inputs for reactor water levels 1 and 2 which are provided by only four sensing devices, namely B21-N031A, B, C, D (existing MPL numbers). The licensee proposes to make drywell pressure sensor configuration consistent with water levels 1 and 2 sensors, by using drywell pressure sensors E11-N010A, B, C, D to provide signals for all four systems of ECCS. This change will still satisfy the single-failure criterion. The reliability of the drywell pressure trip logic for ECCS will not be affected adversely by this change. Plant safety margin is not being reduced since the level of sensor redundancy for each trip function is maintained.

This change deletes instruments E11-N011A, B, C, D and transfers their associated trip function to instruments E11-N010A, B, C, D. Since these instruments (E11-N010A, B, C, D) are being incorporated into the ATTS modification, the instrument number was changed to E11-N694A, B, C, D.

The proposed Technical Specifications revision changes the Remarks column of tables 3.2-4, 3.2-5, and 3.2-6 to include all the functions of drywell sensors E11-N694A, B, C, D. Based on the above information, the staff finds this modification and the proposed Technical Specification change, as discussed above is acceptable.

RCIC Turbine Exhaust Pressure Trip Setpoint Modifications

The proposed trip setpoint value for the reactor core isolation cooling (RCIC) turbine exhaust pressure is 45 psig compared to 25 psig in the current Technical Specifications. The objective of this change is to increase the RCIC system availability for some small and intermediate break LOCAs. Justification for the proposed change is presented in General Electric Company report NEDC-30136 which was provided with the submittal.

In Hatch Unit 1, as in other BWR plants, the exhaust line pressure signal is used to trip the RCIC turbine since it could indicate line blockage. The high pressure signal from the space between the two rupture diaphragms in the exhaust line is used to initiate closure of the isolation valves in the RCIC turbine steam supply line. For Hatch Unit 1, both pressure signals are included in the Technical Specifications. However, only the diaphragm pressure signal is included in the Standard Technical Specifications and the Technical Specifications of other BWR/4 plants that were checked during the review.

The justification for the proposed increase in NEDC-30136 included considerations of the beneficial effects of increased availability as well as drawbacks such as increase in offsite and onsite doses resulting from the higher leak rates from the turbine gland seals and from governor and stop valve stems at the higher exhaust pressures. The proposed increase in setpoint would not affect normal system operation or the consequences of large LOCAs (for the LOCA, rapid system blowdown would prevent RCIC system operations). However, the proposed increase could be beneficial for some small and intermediate break LOCAs involving significant increases in containment pressure when the RCIC system could provide an alternate source of makeup water and prevent fuel damage. For these scenarios, the proposed increase permits longer system operation before the system is tripped by this signal as the result of the increase in containment pressure. The higher doses for these scenarios due to operation at higher exhaust line pressures are well below 10CFR20 limits.

On the basis of our review of the justification in NEDC-30136 we conclude that the proposed increase in setpoint results in a positive contribution to reactor safety and is acceptable.

Trip Setpoint/Allowable Values For Rosemount Transmitters

The proposed change involves revision of the trip setpoint/allowable values for reactor vessel levels 1, 2 and 3, shroud water level and reactor vessel steam dome pressure low instruments. It was stated that a) the trip setpoint/allowable values were calculated with methodology approved by the staff in the safety evaluation included with License Amendment 103 and b) the proposed values are more conservative with respect to the analytical limits then the present values. We requested and reviewed the General Electric Company justifications for the analytical limits.

The reactor water level 1 signal initiates closure of the MSIVs, startup of the Core Spray Spray system and the RHR system in the LPCI mode, and is one of the ADS initiation signals. The analytical limit of -152.5 (equivalent to 12 inches above the top of the active fuel) is the value used in the Appendix K calculations (ref.1).

The reactor vessel level 2 signal initiates HPCI, RCIC and the SBGT systems. It is also used for initiating isolation of secondary containment and partially isolating primary containment. The previous analytical limit used in the Appendix K calculations was -38 inches. The General Electric Company conducted a sensitivity study to evaluate the effect of changing this value and found that the use of a new value of -58 inches for level 2 had "no effect on the analyzed accident/transient consequences" (ref. 1). Hence, a new value of -58 inches was proposed in the submittal to reduce unnecessary challenges to the HPCI, RCIC and isolation systems.

The reactor vessel water level 3 provides one of the ADS initiation signals and partially isolates primary containment. The analytical limit of +1.5 inches is the value used in the Appendix K calculations (ref. 1).

The reactor shroud water level 0 signal is used as an interlock to prevent diversion of LPCI flow to containment spray. In reference 1 it is stated that the analytical limit of -211 inches meets the specification of Section 7.4.3.5.4 of the Hatch Unit 1 FSAR that this interlock be set at a value no lower than two-thirds core height.

The reactor vessel steam dome pressure signal for the recirculation pump discharge valves has an analytical limit of 300 psig to conform to the Appendix K calculation assumptions. The valves are assumed to close within 33 seconds after the pressure drops to this value (ref. 1).

On the basis of our review of the proposed setpoints and the justification provided in reference 1, we conclude that the proposed changes are acceptable.

Reactor Vessel Steam Dome Pressure Permissive Modifications for CS and LPCI Injection Valves

The CS and LPCI injection valves have both upper bound and lower bound limits for the reactor vessel steam dome pressure permissive signals. The upper bound limit helps provide overpressurization protection for these low pressure systems and has a value of 500 psig in the current Technical Specifications. The lower limit is the pressure at which the valves are assumed to start opening in the LOCA analyses. In this submittal, it is proposed to eliminate the upper limit. This proposal was made because the difference between the current values of the upper and lower bounds was too small to permit meeting the limits with the current setpoint methodology and the specifications of the new pressure transmitters. We have reviewed the proposed change and justification and conclude that the current upper bound limit, which helps prevent overpressurization of the CS and RHR systems, should be retained. This retention would require a small decrease in the lower limit with a resulting small increase in the calculated peak clad temperature (PCT) for the limiting break. We understand that the licensee is evaluating the expected small increase in PCT and will present the results in another submittal.

Miscellaneous Trip Setpoint/Allowable Value Modifications

Modifications to the trip setpoint allowable values were proposed for 25 RPS and ECCS trip functions. These are listed in Table 1. In reference 2 and the present submittal, the licensee, in response to staff questions, stated that a) unless noted as such in the submittal, the analytical limits used in the setpoint calculations were the original limits used in the Hatch Unit 1 safety analyses, b) any changes to the limits had been justified by a safety evaluation and c) "in no case with these new limits do the FSAR analyzed transients or accidents exceed the safety limits which are specified in the Plant Hatch Technical Specifications".

The bases for the analytical limits were audited and discussions of particular limits associated with the RHR, CS, RCIC and HPCI systems were held with the licensee. We conclude that the proposed miscellaneous modifications are acceptable.

3.0 ENVIRONMENTAL CONSIDERATION

An Environmental Assessment and Finding of No Significant Impact has been issued for this 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 these amendments will not be inimical to the common defense and security or to the health and safety of the public.

5.0 REFERENCES

1. Letter from L. T. Gucwa, Georgia Power, to Director of NRR, "Response to Staff Questions on Proposed Technical Specification Changes for ATTS," December 7, 1985.

2. Letter from L. T. Gucwa, Georgia Power, to J. F. Stolz, NRC, June 7, 1984.

Dated: January 17, 1986

Principal Contributors: J. Mauck and C. Graves

Attachment:
Table 1

TABLE 1
Miscellaneous Setpoint Modifications

ECCS Trip Function

Drywell pressure - high

RHR pump discharge pressure - high

RHR pump flow - low

Core spray pump discharge pressure - high

Core spray pump discharge flow - low

HPCI steam supply pressure - low

HPCI pump discharge flow - high, low

HPCI pump suction pressure - low

HPCI turbine exhaust diaphragm pressure - high

Suppression chamber water level - high

HPCI turbine exhaust pressure - high

HPCI emergency area cooler ambient temperature - high

RCIC pump discharge flow - high, low

RCIC pump suction pressure - low

RCIC pump suction pressure - low

RCIC steam supply pressure - low

RCIC turbine exhaust diaphragem pressure - high

RCIC steam line differential pressure - high

Suppression chamber ambient temperature - high

Suppression chamber differential temperature - high

RCIC emergency area cooler ambient temperature - high

RPS Trip Function

Main steam line flow - high

Main steam line tunnel temperature - high

Reactor vessel steam dome pressure - low permissive

Drywell pressure - high

RWCU area temperature - high

RWCU area ventilation differential temperature - high

U.S. NUCLEAR REGULATORY COMMISSION

GEORGIA POWER COMPANY, ET AL

NOTICE OF ISSUANCE OF AMENDMENT TO

FACILITY OPERATING LICENSE

The U. S. Nuclear Regulatory Commission (the Commission) has issued Amendment No. 121 to Facility Operating License No. DPR-57, issued to Georgia Power Company, Oglethorpe Power Corporation, Municipal Electric Authority for Georgia, City of Dalton, Georgia (the licensees), which revised the Technical Specifications (TSs) for operation of the Edwin I. Hatch Nuclear Plant, Unit No. 1 (the facility) located in Appling County, Georgia. The amendment is effective as of the date of its issuance and shall be implemented within 30 days.

This amendment revises the TSs for Hatch Unit 1 to support the installation of the analog transmitter trip system (ATTS). It includes changes to the surveillance frequencies and trip setpoints associated with the ATTS equipment.

The application for the amendment complies with the standards and requirements of the Atomic Energy Act of 1954, as amended (the Act), and the Commission's rules and regulations. The Commission has made appropriate findings as required by the Act and the Commission's rules and regulations in 10 CFR Chapter I, which are set forth in the license amendment.
Notice of Consideration of Issuance of Amendment and Opportunity for Prior Hearing in connection with this action was published in the FEDERAL REGISTER on August 26, 1985 (50FR 34559). No request for a hearing or petition for leave to intervene was filed following this notice.

Also, in connection with this action, the Commission prepared an Environmental Assessment and Finding of No Significant Impact which was published in the FEDERAL REGISTER on January 9, 1986 (51 FR 1051).

For further details with respect to this action, see (1) the application for amendment dated July 24, 1985, (2) Amendment No. 121 to License No. DPR-57, and (3) the Commission's related Safety Evaluation. All of these items are available for public inspection at the Commission's Public Document Room, 1717 H. Street, N.W., Washington, D.C. 20555, and at the Appling County Public Library, 301 City Hall Drive, Baxley, Georgia 31513. A copy of items (2) and (3) may be obtained upon request addressed to the U.S. Nuclear Regulatory Commission, Washington, D.C 20555, Attention: Director, Division of Licensing.

Dated at Bethesda, Maryland, this 17th day of January 1986.

FOR THE NUCLEAR REGULATORY COMMISSION

Daniel R. Muller

Daniel R. Muller, Director
Project Directorate #2
Division of BWR Licensing