UNITED STATES
NUCLEAR REGULATORY COMMISSION
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SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

SUPPORTING AMENDMENT NO. 103 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 application dated September 5, 1984, and supplemented October 18, 1984, Georgia Power Company (the licensee) requested an amendment to Appendix A of Facility Operating License No. DPR-57 for Edwin I. Hatch Nuclear Plant, Unit No. 1. The changes to the Limiting Safety System Settings, the Limiting Conditions for Operation, the Surveillance Requirements and their supporting bases reflect the following plant modifications: 1) incorporation of the low-low set (LLS) logic into the pressure relief system; 2) changing the MSIV water level trip from Level 2 to Level 1; 3) deletion of the high drywell pressure signal from the isolation logic for the RHR shutdown cooling suction valves and vessel head spray valves; and 4) elimination of the MSIV closure scram in the startup mode. In addition, the licensee requested a change in the low water Level 2 setpoint, the HPCI high steam flow isolation setpoint, the RHR shutdown cooling safety limit, and the steam dome pressure permissive for recirculation discharge valve closure.

The first of these modifications was originally developed by the General Electric Company (GE) and involved installing a new design for safety systems instrumentation in the RPS (i.e., Reactor Trip System (RTS) and Engineered Safety Features (ESF)) of Boiling Water Reactors (BWR). The new design, referred to as the Analog Transmitter Trip System (ATTS), is being supplied as original equipment in later built BWRs (e.g., BWR 6), and the design is adaptable to operating BWRs as a retrofit. The ATTS essentially replaces pressure, level and temperature digital switches with analog sensor/trip unit combinations, which provide continual monitoring of critical parameters in addition to performing basic logic trip operations. GE developed 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.

GE presented ATTS to the NRC staff under topical reports NEDO-21617 (April 1977) and NEDO-21617-1 (January 1978). The staff reviewed on a generic basis and found acceptable ATTS as indicated in its letter to GE dated June 27, 1978. GE presented the ATTS for Edwin I. Hatch Nuclear Plant Units 1 and 2 under topical report NEDE 22154-1, Revision 1 (July 1983).

In the staff's evaluation of the generic design for the ATTS, certain plant specific information is identified and requested to be submitted by those licensees who will be implementing the new design. This information will allow the staff to determine whether the interfaces between the ATTS and other systems in the plant are acceptable and whether the ATTS has been qualified in accordance with the service conditions that it will be exposed to in the plant. Particular information required from the licensees will be environmental qualification of equipment and divisional separation of redundant hardware to be installed in the plant.

The second modification submitted was the low-low set (LLS) relief logic for BWRs with Mark I containments. This modification is designed to prevent multiple subsequent actuations of safety relief valves (SRVs) which might normally be expected during a transient. This in turn will reduce or prevent the discharge loads on the containment and suppression pool structures resulting from subsequent SRV actuations. The discharge loads from subsequent actuations tend to be higher due to the condensation of trapped steam in the safety relief valve discharge line (SRVDL) which results in a high water leg in the SRVDL, and hence, larger thrust loads on subsequent actuations. In addition, the warmer steam air mixture in the SRVDL results in higher pressure air bubbles in the suppression pool, and therefore, increased torus loads on subsequent actuations.

The LLS is an automatic SRV actuation system which, upon initiation, will assign preset opening and closing setpoints to four preselected SRVs. These setpoints are selected such that the LLS controlled SRVs will stay open longer, thus releasing more steam (energy) to the suppression pool, and hence more energy (and time) will be required for repressurization and subsequent SRV openings. The LLS increases the time between (or prevents) subsequent actuations sufficiently to allow the high water leg created from the initial SRV opening to return to (or fall below) its normal water level, thus, reducing thrust loads from subsequent actuations to within their design limits. In addition, since the LLS is designed to limit SRV subsequent actuations to one valve, torus loads will also be reduced.

2.0 EVALUATION

2.1 Systems Aspects

1. Low-Low Set Relief System

The LLS relief logic system employs four non-ADS SRVs to reduce subsequent actuations of SRVs during pressurization transients. This modification was proposed as part of the Mark I containment program to reduce containment loads from subsequent SRV actuations. The licensee has demonstrated by analysis of limiting transients (ref. 1 and 2) that the LLS with setpoints as proposed in the technical specifications, will extend the time between SRV actuations so as to allow the water column in the SRV discharge lines to drain. Clearing of the water column is sufficient to assure that subsequent discharge loadings will be within the containment design limits. This is acceptable to us.

2. MSIV Water Level Trip

The current technical specifications require MSIV closure on a Level 2 water level signal. The lowering of the trip setpoint to Level 1 is intended to reduce reactor isolations and therefore challenges to the SRVs. This modification was also proposed as part of the Mark I containment program. References 1 and 2 indicate that this change will not adversely impact plant performance or safety margins. This is acceptable to us.

3. Deletion of the High Pressure Signal From the RHR Isolation Logic

The current specifications require the RHR suction valves and vessel head spray valves to close upon receipt of a high drywell pressure or Level 3 water level signal. The purpose for the isolation is to permit realignment of the RHR system to the low pressure coolant injection (LPCI) mode of operation. A high drywell or Level 3 signal is presumed to indicate a break in the primary pressure boundary.

We have reviewed the conditions in which the high drywell signal may initiate isolation of the RHR system. We conclude that its role is insignificant as follows: The RHR operates in the shutdown cooling mode only when reactor pressure is less than 135 psig. For a normal cooldown at 100°F/hr, the reactor would be shutdown at least 90 minutes before initiation of the RHR system. If a large break event occurred, there would be only a short time between receipt of the high drywell pressure and the Level 3 signals. Since Level 3 is more than 14 feet above the top of the active fuel and the decay heat rate is relatively low and effectively constant for short transient time periods, no impact on accident consequences is expected should the high drywell signal be absent. Small break events would proceed more slowly than large break events. Even if isolation is delayed until Level 3, slow level changes are unlikely to result in any fuel uncovery. In addition, the decay heat is low enough at this time that a single loop of core spray (two are available) is sufficient to maintain core cooling. The licensee's proposal is, therefore, acceptable.

4. Elimination of the MSIV Closure Scram in Startup

The current technical specifications require an automatic scram on MSIV closure when the reactor is in the startup/hot standby mode and pressure is above 1045 psig. The high pressure scram however, which is operable in all modes of operation, has a setpoint of 1045 psig. Therefore, the reactor will always scram above 1045 psig regardless of MSIV position or the mode switch position. Since the high pressure scram signal is safety grade, we conclude that elimination of the MSIV closure scram in startup is acceptable.

5. Reactor Vessel Water Level - Low-Low (Level 2)

The licensee has proposed a technical specification limit of ≥-55 in. for the Low-Low (Level 2) setting. This is based on the analytical limit of -58 in. used in the loss-of-coolant accident analyses and the setpoint methodology of Regulatory Guide 1.105.

The loss-of-coolant accident analyses using a low-low (Level 2) analytical limit of -58 inches result in calculated peak cladding temperatures of less than the 10 CFR 50.46 limit of 2200°F. The analytical limit is, therefore, acceptable. The use of Regulatory Guide 1.105 is also acceptable as indicated in Standard Review Plan Section 7.1.

6. HPCI High Steam Flow Isolation

The licensee has proposed changing the allowable value for the high HPCI steam flow isolation signal from 300% to 303% of rated flow. The purpose of this signal is to isolate the HPCI steamline in the event of a break in the steam supply line.

The new setpoint value of 303% results from an analytical limit of 307% and the setpoint methodology of Regulatory Guide 1.105. Accident consequences assuming the higher analytical limit remain bounded by main steamline breaks in the current FSAR analyses. The change is, therefore, acceptable to us.

7. Residual Heat Removal Shutdown Cooling Safety Limit Modification

The existing specifications (1.2.A.2) list a safety limit of 135 psig for the reactor steam dome pressure whenever operating the RHR system in the shutdown cooling mode. The licensee has stated that this value is incorrect and that analyses indicate a safety limit of 162 psig will prevent overpressurization of the heat exchangers. The specification will be corrected to show 162 psig for the safety limit and $\leq$ 135 psig as the permissive for initiating shutdown cooling. This is acceptable to us.

8. Pressure Permissive For Recirculation Discharge Valve Closure

The existing specifications list a value of $\leq$ 335 psig for this permissive. The safety analyses assume that the valves receive a permissive to close at 300 psig. Therefore, the specification needs a sign change to assure that the valve permissive is satisfied within analysis assumptions. The licensee has recalculated the setpoint and allowable values using Regulatory Guide 1.105 and an analytical limit of 300 psig. The nominal setpoint is 360 psig with an allowable value of $\geq$ 325 psig. This is acceptable to us.

2.2 Instrument and Controls Aspects

Analog/Transmitter Trip System

The ATTS, as stated above, is a replacement for the mechanical type digital sensor switches. The existing logic arrangement will not be affected. The ATTS and the trip relays provide the input intelligence for the plant process parameters to the system logics for the RPS and the ECCS, including the reactor core isolation cooling (RCIC) system. The proposed instrument modifications are intended to: 1) reduce primary sensor element drift; 2) reduce the frequency of setpoint drift occurrences; 3) provide indication for each primary sensor which will verify operability of the sensor; 4) reduce the time RPS logic must be in half scram condition to functionally test or calibrate a Safety Trip; 5) reduce the functional test and calibration frequency for the primary sensor and facilitate calibration of the primary sensor when the reactor is shutdown for refueling; 6) reduce the likelihood of instrument valving errors; 7) reduce the potential for instrument testing related scrams; and 8) replace devices that are required to mitigate a LOCA and high energy line break with environmentally qualified hardware.

The analog trip system hardware is used to process inputs into the ECCS, RPS, and RCIC. All of the trip unit card files and power supplies for ECCS and RCIC are contained within two sets of Division 1 and 2 cabinets. These devices operate with logic in the energize-to-actuate mode, using the 125 VAC station emergency battery for their power source. Similarly, the RPS contains its own power supply and trip unit card files within four separate independent cabinets, one for each RPS division. These devices operate with logic in deenergized-to-actuate mode using the 120 VAC power from RPS motor generator sets. Since the dual channel design (with two trip systems) of the RPS is not being altered, the operation of the trip system remains the same. The automatic and manual initiation and protective action of essential systems remain unchanged.

The service environments applicable to each item of hardware compromising the ATTS are specified in the Product Qualification Program Requirements Document (22A7011). The cabinet mounted equipment consists of the trip unit hardware, trip unit calibration module, card file, trip relay, voltage converters, and miscellaneous hardware. The reactor building mounted equipment consists of the pressure and differential pressure transmitters which are mounted locally either on the structures or instrument racks; the sealed sensor differential pressure transmitters which are locally mounted on customer supplied supports in the reactor building; and the RTD temperature sensors which are also locally mounted on supports in the reactor building. The methods used to demonstrate the qualification program of the ATTS at Hatch Units 1 and 2 included type testing and/or analysis. In type testing, the equipment tested was aged and subjected to all applicable environmental influences to provide assurance that all such equipment would be able to perform the intended functions for the required minimum operating time. Qualification by analysis included the construction of valid mathematical models of the equipment to be qualified, verification of the mathematical models by test, and quantitative analysis of the mathematical models to demonstrate that the product performance characteristics met or exceeded the equipment design requirements.

Inductive or capacitive coupled electro-magnetic interaction (EMI) from radiated electromagnetic fields are limited only to near-fields because the distance from the interfering source is usually less than $\lambda 2/\pi$, where λ is the wave-length of the interference signal. The following type of EMI susceptibility tests were conducted on the ATTS: 1) conducted EMI transients; 100 to 500 KHZ, 300 VAC peak-to-peak or ±5.0V (24 VDC); 2) conducted RFEMI, 0.5 to 100 MHZ, 5 V peak-to-peak; 3) radiated transient EMI fields, 100 to 500 KHZ, 5 V peak-to-peak; and 4) radiated RFEMI fields 0.5 to 100 MHZ, 5V peak-to-peak.

Only conducted tests were performed on the converter input leads and relay leads. Only radiated tests were done on the transmitter, RTD, and auxiliary analog output leads since there are no associated branch connections. The EMI tests with the transmitters showed that EMI requirements could only be met with the addition of an EMI filter on each transmitter lead. An EMI filter assembly was designed for use with the transmitters and has become part of the ATTS design. The filter assembly was qualified by analysis to the same environmental requirements as the transmitters. Therefore, we concluded that the ATTS is qualified for operation in its present EMI environment.

The licensee has stated that the wiring for the ATTS design conforms to the recommendations of Regulatory Guide 1.75 to the maximum practical extent. Divisional separation is maintained within the cabinet. Class 1E/non-Class 1E separation is carried through up to the annunciation trip relay. The annunciator trip relays provide separation between 1E and non-1E circuits (i.e., separation is via the contact to coil separation within the relay).

Within the cabinets, the minimum separation distance is 6 in. up to the relay. Within the relay, limitations exist related to the distance from the contact to the coil. This design prevents maintaining complete physical separation between the annunciation wiring and the class 1E wiring. This does not pose a problem because the annunciator circuitry is a low energy circuit. The annunciators interrogate contacts in the ATTS with a 140 V DC signal that is current limited to the maximum of 1 ma by the annunciator input resistance.

In another area, the licensee indicated that there were non-class 1E loads powered from class 1E buses with a circuit breaker as the only separation device. This is an acceptable means of separation consistent with the original design basis of the plant. We examined the new hardware associated with the addition of the ATTS with respect to susceptibility to failures (i.e., voltage variations, hot shorts, open circuits) caused by non-class 1E loads. The licensee stated that the additional hardware associated with the ATTS is no more susceptible to these failures than the hardware it replaced. Therefore, no new failure modes have been introduced in the Hatch design, and the original licensing basis of the plant with respect to the application of isolation breakers has not been modified. We conclude that this portion of the design is enveloped by the original design basis of the plant and is acceptable.

Previous instruction manuals (4471-1 Rev. A) have contained a warning regarding operating at a low ATTS power supply voltage because if certain conditions exist (e.g., lead length, wire diameter, temperature), a low supply voltage at the transmitter may cause it to operate improperly and a desired trip may not occur. Thus, we requested information to address a concern that an undervoltage condition could exist that would incapacitate the trip functions of all the effected ATTS units.

The licensee responded stating that the purpose of the maximum lead length requirement is to assure sufficient voltage out of the trip unit to drive the transmitter. Calculations by General Electric indicate that lead lengths as long as 3820 feet are acceptable using 16 gauge wire. The maximum length of cable used in the Hatch ATTS design is 1800 feet, utilizing 16 gauge wire.

In addition, the licensee stated that the RPS portion of the ATTS is supplied, as is the remainder of the RPS, from the RPS motor-generator (MG) set which has a class 1E electrical protection assembly (EPA) that is installed between each RPS bus and each power source. This protects each RPS bus against a sustained over/undervoltage or underfrequency condition. Each EPA consists of a circuit breaker with a trip coil driven by logic circuitry that senses line voltage and frequency and trips the circuit breaker open on conditions of overvoltage, undervoltage or underfrequency. The system itself is a fail-safe system. Therefore, with a loss of power, all instruments go to their safety position.

The ECCS portion of the ATTS is powered off the plant batteries. The class 1E batteries are divisionalized and supplied by chargers that are powered off the emergency buses. The batteries are sized per FSAR Section 8.5.3 for two hours continuous duty without the chargers. The power supply for the ECCS portion of ATTS is consistent with the original design basis of the plant. Undervoltage on ECCS portions of the ATTS is protected via the protective design features included in the battery and charger that provide power to ECCS. The minimum voltage that the batteries would ever supply based on the FSAR requirement is 105 VDC. The ATTS has voltage converters which operate from 105 to 140 VDC on the input and provide a nominal output of 25 VDC. We find this to be acceptable since the ATTS is designed to operate with a minimum voltage at 23.5 VDC.

The operability of the trip unit and auxiliary relays is verified by periodic functional testing using special test equipment supplied as part of the ATTS. Operability of the transmitters is verified by periodic comparison of the redundant indicators on the master trip units which monitor the same parameter. Gross transmitter failure is detected by special monitoring circuits. The licensee stated that the high/low gross failure setpoints are to be set at values of 35 ±0.5 and .5 ±0.5 ma respectively. These values are provided to indicate a short-circuit and open circuit. Therefore, the setpoint values can be varied significantly outside the saturation range of the transmitter and still provide adequate protection.

Low-Low Setpoints

The LLS circuitry consists of four redundant logic channels, each of which actuates one SRV. There are eleven SRVs at Hatch Unit 1, seven of which are actuated by the Automatic Depressurization System (ADS). The four non-ADS SRVs will be used for the LLS function. Each of the four LLS controlled SRVs will open when their respective solenoid becomes energized by the LLS logic. The LLS logic channels that actuate SRVs B21-F013H and B21-F013G, channels A and C respectively, are powered by 25 Vdc from division 1 Class 1E supply H11-P925. LLS logic channels B and D (SRVs B21-F013A and B21-F013C respectively) are supplied from division 2 Class 1E supply H11-P926.

In order for a LLS channel to energize its solenoid, both an arming logic and an initiation logic must be satisfied. The arming logic is satisfied when any SRV has opened and reactor pressure has exceeded the high pressure scram setpoint (this setpoint is selected above the reactor protection system high reactor pressure scram setting to assure that a scram has occurred). Four separate reactor pressure instrument channels (one for each LLS channel), each consisting of a transmitter and associated trip unit, have been added to provide this reactor high pressure permissive function in the LLS arming logic. Each transmitter and trip unit are powered from the same division as their corresponding logic channels.

Once the arming logic for any LLS channel is satisfied, it is sealed in and annunciated in the control room, and remains sealed in until manually reset by the operator. In addition, the arming logic in either LLS channel of the same division will seal in the arming logic in the remaining LLS channel of that division provided the reactor high pressure permissive in that channel is satisfied.

Initial SRV actuations are detected by two sets of pressure switches located in the SRV discharge lines. Each discharge line contains one pressure switch powered from division 1 and the other from division 2. Contacts from these switches are used in the arming logic of the corresponding divisional LLS logics. These pressure switches are set above the normal pressure expected in the discharge line (85 psig).

Once armed the LLS actuation/control logic uses newly added reactor pressure instrumentation to control the LLS SRV solenoids, thus opening and closing these SRVs at their assigned LLS setpoints. The actuation/control logic remains in effect as long as the arming logic is sealed in. The added instrumentation consists of one transmitter and an associated trip unit for each of the four LLS logic channels. In addition, a second trip unit associated with the transmitter providing the arming logic pressure permissive for each LLS channel has been added and is used in the actuation/control logic for that channel. Both trip units providing control for a given SRV have the same setpoints such that they actuate simultaneously. This arrangement prevents single failures within the transmitter and trip unit portion of the LLS circuitry from causing a spurious SRV opening once the arming logic is satisfied, and from causing a SRV to remain open after reactor pressure has decreased to the reclose setpoint. The added transmitters and trip units are powered from the same division as their corresponding logic channels.

All four LLS logic channels can be tested at power. Test status lights in the control room indicate when the arming logic relays and contacts have operated satisfactorily during testing. These test lights can also be used to verify proper operation of the seal-in and reset circuits. Each LLS channel provides annunciation in the control room upon loss of power. Test switches are provided to verify operability of this power monitor function. Additional test lights in the control room are used to verify operability of the trip units used in the LLS actuation/control logic during testing.

The LLS circuitry contains no channel or operating bypasses. The circuitry added for the LLS function is located in the control room and is separated in accordance with IEEE 384-1974. The components of the LLS system (including power supplies) are classified as Class 1E. The LLS will remain operable in the event of loss of offsite power. LLS components located inside the drywell are qualified for the environmental conditions associated with a small break LOCA.

Applicable Technical Specification Revisions

In addition to the ATTS and LLS modifications discussed above, the licensee provided information regarding several proposed Technical Specification revisions. The purposes of these proposed revisions are to utilize the benefits of the ATTS and LLS additions, prevent unnecessary plant transients by using less conservation setpoints or delete certain isolation, actuation and permissive sensors. These Technical Specification revisions that are included in this portion of the evaluation are as follows:

1. Low-Low Set Logic Design Modification

The proposed Hatch Unit 1 Technical Specification changes associated with the LLS modification call for monthly channel functional tests of all reactor pressure instrument channels (used for both the arming logic permissive and SRV control/actuation). A channel functional test of all SRV discharge line pressure switches will also be performed monthly (portions of these channels inside the primary containment may be excluded from this test). Channel calibrations and LLS logic system functional tests will be performed during each refueling outage. This test frequency is consistent with the test interval for the ADS and is acceptable to the staff.

2. Reactor Vessel Water Level - High (Level 8) Trip Instrumentation Modifications

After HPCI and RCIC have activated, this trip function prevents the water level in the reactor vessel from reaching the height of the main steam outlet. Its intended protective function is accomplished by tripping the HPCI steam turbine and closing the HPCI and RCIC steam supply valves when the water level in the reactor vessel reaches the level 8 setting. The trip function is to protect the HPCI and RCIC steam turbine system from potential damage.

This trip function is currently assigned to B21-N017A, B, C, D which controls the RPS reactor vessel water level 3 instrumentation. To separate the RPS and ECCS functions, the ATTS design assigns the reactor vessel water level 8 trip function to ECCS instrumentation. The functions of the level 8 trip remain the same.

The analytical limit for this function is 59.5 in. The licensee stated that the trip setpoint/allowable value of $\leq$56.5 in. was developed using the criteria of Regulatory Guide 1.105, and the designated trip setpoint for the plant will take into consideration setpoint drift.

3. Lowered Water Level Trip Setpoint for Isolation of Reactor Water Cleanup System and Secondary Containment, and Starting of Standby Gas Treatment System (SGTS)

Reactor scram from normal power levels (above 50 percent of rated) usually results in a reactor vessel water level transient due to void collapse that causes isolation of the RWCU system at reactor water level 3.

The result is typically the dropping of the cleanup filter cake, added radwaste processing, loss of ability to remove water from the reactor vessel immediately after scram, and other undesirable operational problems. These results adversely affect plant availability and operability. By lowering the isolation setpoint to reactor water level 2, these problems may be resolved without any adverse safety impact. The lowering of the level trip for isolation of RWCU from reactor water level 3 to reactor water level 2 will not have any adverse effect on plant transient and accident analyses. For any reactor pressure coolant boundary line breaks inside the primary containment, the LOCA design basis accident (DBA) analysis shows that the ECCS is capable of mitigating all break sizes including and up to the recirculation line break. For a RWCU line break outside the primary containment, the break detection is provided by the high differential temperature rather than by water level variation.

By lowering the SGTS actuation and secondary containment isolation from reactor water level 3 to reactor water level 2, a potential for spurious trips is reduced. The ECCS analysis design basis assumes that the SGTS will initiate at the same time as the ECCS which initiates at reactor water level 2.

This modification has been implemented and accepted by the staff on other BWR 4s. The requirements of 10 CFR 100 will still be met.

4. Safety/Relief Valve Position Indicator Modifications

During qualification testing, it was shown that the variable setting of the S/RV pressure switches could not hold their setpoints. To eliminate this problem, new switches were designed which do not have the variable setpoint capability. The switches are set at the factory with a fixed setpoint of 85 psig. Also, an additional clarification of the technical specifications was required to reflect the fact that the secondary S/RV indicator is a recorder with a range of 0-600°F that receives input from a temperature element and not a temperature element with the above range.

Since these changes only clarify the technical specifications and do not revise any setpoints, the staff considers this modification and clarification acceptable.

5. Trip Function Identification Modifications

Several of the trip function descriptions were revised to correspond with the Hatch-2 Technical Specifications and in some cases with the Standard Technical Specifications. Since these modifications are editorial in nature, the staff considers that they are acceptable.

6. <u>Trip Setpoint/Allowable Value Setpoint Modifications</u>

The instruments to be incorporated into the ATTS possess less drift and greater accuracy than the existing instruments in use at Plant Hatch. Therefore, new calculations were performed to determine the setpoint value for each instrument. The Plant Hatch analytical limits were used (were applicable) to develop the allowable values and trip setpoints. The values that are proposed to be inserted into the Technical Specifications are the calculated allowable values. The setpoints used at Plant Hatch will take into consideration instrument drift and will be developed from the allowable values. The proposed Technical Specification revisions include modification of the trip setpoint/allowable values for the following instruments:

- Reactor vessel steam dome pressure-high (B21-N678A, B, C, D)
- Reactor vessel water level-level 3 (B21-N680A, B, C, D)
- Reactor vessel water level-level 1 (B21-N681A, B, C, D)
- Reactor vessel water level-level 1 (B21-691A, B, C, D)
- Reactor vessel steam dome pressure-low (B21-N690A, B, C, D, E, F and B21-N641 B,C)
- Reactor vessel water level-level 3 (B21-N695A, B)

We have reviewed the acceptability of the proposed Technical Specification revisions discussed above. We questioned the licensee regarding the basis for the analytical limits used in the safety analysis. The licensee stated that the analytical limits are the values used as inputs to the safety analysis in the FSAR. For Hatch, the analytical limits were selected to prevent violation of the applicable safety limits. For example, the analytical limit for the level 1 reactor water level trip satisfies the peak cladding temperature of 2200°F in the Hatch Appendix K LOCA analyses. Unless otherwise noted (Reference 5), the analytical limits used in the Hatch Safety Analysis. For the analytical limit that was revised (Reference 5), the licensee stated that with this new limit, the FSAR analyzed transients or accidents do not exceed the safety limits which are specified in the Hatch Technical Specifications. The conservatisms in the Hatch design basis computer codes were not used in place of the analytical limit for the starting value of the calculations.

The allowable value was obtained by either adding or subtracting (whichever was conservative) the loop accuracy from the analytical limit. Loop accuracy was determined by utilizing the square root of the sum of the squares of the transmitter accuracy, trip unit accuracy and calibration accuracy. These accuracies are treated as independent variables between the analytical limit and allowable value. The trip setpoint was calculated by adding or subtracting (whichever was used to obtain the allowable value) the loop drift and the leave alone range from the allowable value.

Each of these terms is a function of other parameters; for instance, the transmitter accuracy reflects transmitter performance with regard to the transmitter basic reference accuracy, transmitter temperature specifications, power supply specifications and static pressure specifications. The licensee stated that these parameters envelope the Hatch design requirements. Drift of the trip units will be monitored on a monthly basis and drift of the transmitters will be monitored on an operating cycle basis using plant procedures. 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 transmitter and trip unit drifts are treated as independent variables between the allowable value and trip setpoint. The total loop accuracy and the total loop drift (dependent variables) are directly 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 or 12 month intervals. These values were extrapolated linearly to provide 18 and 24 month drift values for use in the Hatch setpoint calculations.

An additional variable called the leave-alone band was added (treated as a dependent variable) between the allowable value and trip setpoint. This band is set at ±0.25 percent of the trip unit range. A setpoint adjustment is not required when the trip unit setting is within this ±0.25 percent range. If the trip unit is out of the range from the setpoint on a monthly calibration functional test, the operator resets the trip unit trip setpoint within the 0.25 percent range. Currently, if the trip unit is outside the ±0.60% (sum of leave alone + trip unit drift), a deficiency report will be generated internally at Hatch by the licensee.

The calibration accuracy leads to the only possible component of error caused by a man-machine interface. To counter this error, the licensee has installed a requirement that calibration be performed with instruments of 0.25 percent or better accuracy. This value was assumed in the setpoint calculations.

The trip setpoint milliamp value is read directly from the calibration unit. The calibration unit locks in the trip setpoint value and presents a digital display. During channel calibration, the readings are taken with a digital voltmeter. Sufficient stability of these readouts is presented such that the human ability to read the display presents insignificant errors in the overall results of the setpoint calculations.

We questioned the licensee regarding the effects of a harsh environment on the resulting setpoint. The licensee stated that the two areas explicitly considered in the harsh environment effects were radiation and temperature compensation. These were considered as independent effects. The reasoning that they are independent effects is that temperature peaks relatively early in a LOCA event while significant radiation integrated doses occur later. As a result of a GE evaluation for Barton transmitters, it was determined that radiation effects were not a significant effect in the setpoint calculations. Therefore, the setpoint calculations did not explicitly consider radiation as a parameter. An evaluation was performed which allowed exclusion of the radiation effect also for those trip functions where Rosemount transmitters are to be installed. Humidity was not an explicit parameter in the setpoint calculations. The testing program for the transmitters included exposure to a steam environment during the DBE/post-DBE testing phases. Therefore, the effects of humidity are accounted for in the temperature compensation factor. In addition, post-accident harsh environment pressure effects on the ATTS accuracies were also evaluated. This evaluation has shown that this environmental factor has a negligible effect on setpoint drift or instrument error.

The final consideration of environmental effects on setpoints is presently an ongoing study which is being performed by the utilities as a part of the equipment qualification program. The findings of the staff review of this study will be factored into the setpoint methodology for the Hatch Plant.

We have reviewed the acceptability of the proposed Technical Specification revisions and have concluded that the proposed Technical Specification revisions permit the operation of the facility in a manner that is consistent with the licensing basis and accident analysis. Therefore, we find that, with the provisions of the generic review noted above, the Technical Specification revisions related to the ATTS are acceptable.

3.0 SUMMARY

In summary, we have previously reviewed the use of the ATTS and found that, provided certain interface requirements were satisfied, the system is acceptable (letter of approval, dated June 27, 1978, is a part of General Electric Topical Report NEDO-21617-A dated December, 1978). Based on our review of the documentation submitted by the licensee, we conclude that the modifications proposed satisfy the constraints of our prior approval and also satisfy the requirements of the applicable General Design Criterion and Regulatory Guides. In addition, based on the data submitted, we conclude that:

1) The reliability, accuracy, and response time of the replacement instrumentation are better than that of the existing instrumentation.

2) The separation criteria of the original plant design is unchanged or improved in some areas. Separation is provided by locating equipment on separate racks and panels and by running cable in separated cable trays or conduits. The power supply used for an instrument channel is dependent on that channel's divisional assignment.

3) No new single failure events have been created. Therefore, no single failure will result in any action not previously evaluated in the FSAR.

4) All new equipment has been tested or analyzed to assure that the design environmental conditions and the design basis seismic requirements are met.

5) Means are provided to test the trip units periodically by injecting a signal and observing the trip output. Operability of the analog loop is verified by instrument checks.

6) Proposed Technical Specification revisions permit the operation of the facility in a manner that is consistent with the licensing basis and accident analysis for Hatch.

Based on our review, we have also determined that the LLS modification installed at Hatch Unit 1 is designed to perform its intended function given a single failure. In addition, no single failure in the electrical circuits could be found which would cause more than one SRV to stick open. The LLS is designed in accordance with the requirements of IEEE Standard 279-1971 and therefore, is acceptable.

Therefore, we conclude that the modifications of the RPS, ECCS and RCIC as discussed above are acceptable. It is further concluded that the applicable, revised Technical Specification pages are acceptable.

4.0 ENVIRONMENTAL CONSIDERATIONS

This 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. We have determined that this 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 previously issued a proposed finding that this amendment involves no significant hazards consideration and there has been no public comment on such finding. Accordingly, this 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 this amendment.

5.0 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 this amendment will not be inimical to the common defense and security or to the health and safety of the public.

Dated: December 7, 1984

Principal Contributors: J. Mauch and T. Collins

5.0 REFERENCES

(1) NEDE-22224, December 1982, General Electric Company, "Low-Low-Set Relief Logic System and Lower MSIV Water Level Trip for Edwin I. Hatch Nuclear Power Plant Units 1 & 2."

(2) NEDE-22223, September 1982, General Electric Company, "Low-Low-Set Logic and Lower MSIV Water Level Trip for BWRs with Mark I Containment."

(3) Letter from J. T. Beckham to J. F. Stolz, "Request For Technical Specification Changes to Support Analog Transmitter Trip System Installation," dated September 5, 1984.

(4) Letter from L. T. Gucwa to J. F. Stolz, "Revision of Request for Technical Specification Changes to Support Analog Transmitter Trip System Installation," dated October 18, 1984.

(5) Letter from L. T. Gucwa to J. F. Stolz, "Revision of Previous HPCI Pressure Trip Technical Specification Change Proposal for ATTS," dated November 6, 1984.

(6) Letter dated June 7, 1984, from L. T. Gucwa to John F. Stolz.

(7) Letter dated June 14, 1984, from L. T. Gucwa to John F. Stolz.

(8) Letter dated June 15, 1984, from L. T. Gucwa to John F. Stolz.

(9) NEDE-22154-1, Revision 1 dated July, 1983, "Analog Trip System For Engineered Safeguard Sensor Trip Inputs - Edwin I. Hatch Nuclear Plant Units 1 and 2."

(10) NEDO-21617-1 dated January, 1978, "Analog Transmitter/Trip Unit System For Engineered Safeguard Sensor Trip Input."