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

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

SUPPORTING AMENDMENT NO. 39 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

The Georgia Power Company (GPC or the licensee) has requested certain changes to the Technical Specifications of the Hatch Nuclear Plant, Unit 2, in order to reload and operate the unit for Cycle 5 (Reload 4). The correspondence transmitting the requests and the subjects dealt with are as follows:

Initial Licensee Requests	Supplementary Submittals	Subjects
1. Request to Change TS-Reload 4 (NED-84-192) dated April 3, 1984		1a. MAPLHGHR for New Fuel Bundle
		1b. OLMCPR Increase
		1c. Hybrid I Control Rods
		1d. Fuel Loading Around SRM Detectors
2. Proposed TS Changes to Support ATTS Installation (NED-84-017) dated January 23, 1984	2a. Response to Verbal Questions (NED-84-281) dated June 7, 1984	2. ATTS Installation
Revision to Request for TS Changes to Support ATTS Installation (NED-84-017) April 3, 1984	2b. Revised Responses (NED-84-321) dated June 14, 1984	
	2c. Additional Clarification (NED-84-326) dated June 15, 1984	

Initial Licensee Requests	Supplementary Submittals	Subjects
3. Proposed TS Changes for ARTS Improvements (NED-84-030) dated February 6, 1984	3a. Additional Information on ARTS (NED-84-186) dated April 3, 1984	3. ARTS Improvements
	3b. Confirmation of Telephone Conversation- ARTS (NED-84-336) dated June 20, 1984	
	3c. ARTS Improvements dated June 27, 1984	

A brief description of each subject follows.

1.1 Addition of MAPLHGR Curve for New Fuel Bundle

This change is requested in connection with the core reloading of Unit 2 to allow for introduction of a new fuel type. The licensee proposes to add a curve of Maximum Average Planar Linear Heat Generation Rate (MAPLHGR) vs. Planar Exposure for the fuel based on the Unit 2 Loss of Coolant Accident (LOCA) response.

1.2 Operating Limit MCPR Increase

This change would increase the Operating Limit Minimum Critical Power Ratio (OLMCPR) for the fuel used in Cycle 5 and subsequent cycles. Approval of this change would permit licensing of subsequent cycles under 10 CFR 50.59.

1.3 Use of Hybrid I Control Rods

This change would allow operation of Hatch 2 with the new Hybrid I control rod assemblies to take advantage of improvements expected from the design which has been reviewed and approved by the NRC staff.

1.4 Fuel Loading Around SRM Detectors

This change would ensure achievement of the required minimum count rate in the Source Range Monitor (SRM) detectors.

1.5 ATTS Installation

This change would permit operation with the newly installed Analog Transmitter Trip System (ATTS) which takes the place of the mechanical type digital sensor switches originally used in the Reactor Protection System (RPS) and the Emergency Core Cooling System (ECCS). 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, and to improve sensor intelligence, reliability and testing procedures. We have reviewed and found acceptable the ATTS on a generic basis with the provision that plant specific information would also have to be reviewed (Reference 1, Bibliography).

1.6 The ARTS Improvement Program

This set of changes is required to implement the Average Power Range Monitor (APRM)/Rod Block Monitor (RBM)/Technical Specification (ARTS) Improvement Program. The primary goals of the ARTS program are to:

1. Replace the APRM scram and rod block trip setdown requirements with more meaningful limits to reduce the need for manual setpoint adjustments and to allow more direct thermal limits administration;
2. Change the RBM hardware using up-to-date electronics;
3. Change the Local Power Range Monitor (LPRM) input assignments to improve response of the RBM;
4. Revise the trip logic and signal normalization procedures for the RBM;
5. Introduce new requirements for power and flow dependent MAPLHGR and MCPR limits; and
6. Revise the Technical Specifications to be consistent with the changes.

The extended load line limit analysis provides the basis for changing the slope of the flow bias algorithm and for the revised APRM rod block line.

2.0 DESCRIPTION AND EVALUATION

2.1 Addition of MAPLHGR Curve for New Fuel Bundle

Addition of a new fuel bundle (type P8DRB284H) requires that an additional curve of MAPLHGR as a function of burnup be added to the Technical Specifications. This curve was obtained by methods described in GESTAR II (Reference 2) which have been approved for use in obtaining MAPLHGR values for extended burnup. We conclude that the MAPLHGR values for the new fuel bundle are acceptable.

2.2 Operating Limit MCPR Increase

The full power, full flow OLMCPR is being revised to bound required values for Cycle 5 and beyond in order to permit licensing these cycles under 10 CFR 50.59. The proposed revisions would increase the Option B (T = 0) limits to 1.29 and the Option A limits to 1.37. The limits are currently 1.26 and 1.32, respectively, for type 8x8R fuel and 1.27 and 1.35 for P8x8R fuel. These changes are conservative and are therefore acceptable.

Cycle specific analyses using the approved methods of GESTAR II (Reference 2) will be performed to confirm that the limits bound the requirements for each cycle.

2.3 Use of Hybrid I Control Rods

The description of the control rod assemblies is being revised to permit the replacement of the standard control rod assemblies with the General Electric Hybrid I Control Rod (HICR) assemblies. The use of these control rods in BWRs has been reviewed and approved by the NRC staff (Safety Evaluation letter dated August 22, 1983, Reference 3), and we conclude that their use is acceptable in Hatch Unit 2.

The details of the design and materials will not be included in the revised Technical Specifications. Since descriptions of the standard blades exist in the FSAR and of the HICR blades in approved Topical Report NEDE-22290-A (Reference 4), and the safety design criteria which control rods must meet are contained in the FSAR and in other Technical Specifications, we conclude that this is acceptable.

2.4 Loading of Fuel Assemblies Around SRM Detectors

The Technical Specifications require that a count rate of 3 counts per second (CPS) be present in Source Range Monitor Channels when loading fuel into the core. A spiral loading technique is to be used for Reload 4. In order to achieve the 3 CPS count rate, it is necessary to load irradiated fuel around the SRM detectors. Present Technical Specifications permit the loading of two assemblies and the revised specifications would permit as many as four. Since the present outage has been longer than usual, two assemblies may not be sufficient to provide the required count rate.

Since 16 or more fuel assemblies are required to achieve criticality and the k-effective of an uncontrolled 2x2 array of maximum reactivity assemblies is less than 0.95, we conclude that no criticality problems exist with the proposed configurations. Further, the same assemblies that were present around the SRM detectors during unloading will be returned there. Since these configurations were sub-critical at that time, they will also be so when reloaded. Hence, loading of up to four assemblies around the SRM detectors is acceptable.

2.5 ATTS Installation

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 comprising 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 convertors, 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, 300 VAC or ±5.0V (24 VDC); and 4) radiated RFEMI fields, 0.5 to 100 MHZ, 5 V 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 IE/non-Class IE separation is carried through up to the trip relay. The annunicator 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 annunicator wiring and the class 1E wiring. This does not pose a problem because the annunicator circuitry is a low energy circuit. The annunicators interrogate contacts in the ATTS with a 140 V dc signal that is currently limited to a maximum of 1 ma by the annunicator 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., load 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/under voltage 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 8.3.2.1.1.a 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 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.

In addition to the ATTS modification as 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 addition, prevent unnecessary plant transients by using less conservation setpoints or delete certain isolation, actuation and permissive sensors. These revisions are as follows.

1. Reactor Water Low Low (Level 2) Trip Setpoint Modifications

The proposed Technical Specification trip setpoint/allowable value for the reactor vessel water level 2 signal is >-55 in. Reactor vessel water level 2 is for the initiation of high-pressure coolant injection (HPCI) and reactor core isolation cooling (RCIC) and the recirculation pump trip. The proposed analytical limit of -58 in. was selected by the licensee to provide the best flexibility and protective margin for the plant. The ECCS calculations are insensitive to the variation in HPCI actuation water level so that a lower water level for level 2 has no significant effect on the ECCS system performance. In addition, this proposed change will have no effect on the (MAPLHGR) limit. The requirements of 10 CFR 100 will still be met.

2. Deletion of High Drywell Pressure Signal for Residual Heat Removal, RPV Head Spray Valves, and Reactor Water Cleanup System Isolation

High drywell pressure has been used as a signal to isolate reactor water cleanup (RWCU) and the shutdown cooling mode of RHR. Small steam leaks in the drywell can cause a high drywell signal which would prohibit an acceptable normal shutdown procedure by preventing operation of the RHR and RWCU systems during the shutdown cooling mode. To resolve this operational concern, the high drywell pressure signal would be deleted from the isolation logic for the RHR shutdown cooling suction and discharge valves, as well as the reactor pressure vessel (RPV) head spray isolation valves and RWCU isolation valves.

The use of high drywell pressure as an isolation signal has little effect in preventing coolant losses due to an RHR or RWCU pipe break inside the drywell since the inboard isolation valves are located as close as possible to the drywell wall. Such pipe breaks do not present a site boundary dose problem since the leaked fluid and associated radioactivity are completely retained within the primary containment boundary. The high drywell pressure signal for the RPV head spray isolation valves will also be deleted. Since the RPV head spray valves are used as part of the shutdown cooling procedures, this change is consistent with above mentioned proposed RHR (shutdown cooling mode) system modification.

This change does not affect the Appendix K calculation results presented in the FSAR. Therefore, the requirements of 10 CFR 100 will still be met. In addition, this modification has been implemented and accepted by the staff on other BWR/4s.

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. Deletion of Ambient Temperature Loops in Leak Detection System

Typically, the leak detection system uses ambient and differential temperatures to detect the small high-temperature leaks. In the earlier design, the ambient temperature trip was provided by an independent temperature element and trip device, and the differential temperature trip was provided by two independent temperature elements and a Δ T trip device. By using the ATTS, the ambient temperature trip may be obtained from one leg of the differential temperature trip. With this arrangement, the sensitivity of leak detection may be changed slightly, dependent on heating, ventilation, and air-conditioning design; but it will not defeat the intended function of the system. This arrangement is suitable for the small rooms containing leak detection temperature monitoring as part of the isolation logic because only large rooms, such as the turbine building, need the spatial location of sensors to adequately protect the room against leaks. This scheme allows the deletion of several unnecessary temperature loops in the RWCU system.

The RWCU temperature and differential temperature sensors sense the temperature in the two pump rooms and the heat exchanger room. Each room has a redundant set of temperature instrumentation that provides input to the RWCU isolation logic. By using the hot leg of the differential temperature sensor for the high ambient trip, several devices may be deleted without any loss of protective function.

For these modifications, single-failure criteria will be maintained.

The proposed Technical Specification revisions will reference the trip unit loop from which the ambient temperature trip is taken in place of the existing ambient temperature trip instrument. Included in these proposed changes are new surveillance frequencies which correspond with the surveillance requirements of the ATTS.

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

The original design of Plant Hatch has the high drywell pressure signals for the ECCS coming from eight sensing devices. For example, E11-N011A, B, C, D (existing MPL numbers) provide signals to RHR, core spray, and HPCI; E11-N010A, B, C, D (existing MPL numbers) provide signals to ADS. This configuration is inconsistent with the inputs for the reactor water levels 1 and 2 trips which are provided by only four sensing devices, namely B21-N031A, B, C, D (existing MPL numbers). To make drywell pressure sensor configuration consistent with that for the water levels 1 and 2 sensors, drywell pressure sensors E11-N010A, B, C, D may be used to provide signals for all four systems of the ECCS and still maintain single-failure criteria. Plant safety margin is not being reduced since the level of redundancy to serve a 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.

It is proposed that the surveillance frequencies be modified to those of instruments E11-N010A, B, C, D. It should be noted that the surveilllance frequencies of both E11-N010A, B, C, D, and E11-N011A, B, C, D are the same in the existing Unit 2 Technical Specifications.

6. Trip Setpoint/Allowable Value Setpoint Modifications

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:

- o Main steamline flow-high (B21-N686A, B, C, D through B21-N689A, B, C, D)
- o Main steamline tunnel temperature-high (B21-N623A, B, C, D through B21-N626A, B, C, D)

- Reactor vessel steam dome pressure-high (B21-N678A, B, C, D)
- Reactor vessel water level-level 3 (B21-N680A, B, C, D)
- Reactor vessel steam dome pressure-low(B31-N679A, D)
- Reactor vessel water level-level 1 (B21-N681A, B, C, D)
- Drywell pressure-high (C71-N650A, B, C, D)
- RWCU room ambient temperature-high (G31-N622A, D, E, H, J, M)
- RWCU area differential temperature-high (G31-N663A, D, E, H, J, M) (G31-N661A, D, E, H, J, M)

7. 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 enclosing 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.

8. Elimination of the Reactor Pressure Permissive to the Bypass of the MSIV Closure Signal Due to Low Condenser Vacuum

The licensee proposed to delete the reactor steam dome pressure permissive which prevents the group 1 isolation valves signal from being bypassed on a low condenser vacuum isolation.

The manual bypass is provided to facilitate the following operations:

A. The bypass allows cold shutdown testing of the main steamline isolation logic and allows stroking the MSIVs open and closed for maintenance even though there is no condenser vacuum.

B. The bypass allows the MSIVs to be opened so seal steam and ejector steam can be available at the turbine and condensor, thereby allowing restart of the reactor from a hot pressurized condition. Attempting to establish condenser vacuum without seal steam from the hot condition by the mechanical vacuum pump may damage the turbine shaft seals.

Thus, the manual bypass of the MSIV closure is performed only when the reactor is not operating at full power. In addition, the manual bypass, which is annunciated in the control room, has the following three permissive conditions:

A. When the keylocked manual switches located on the back cabinets housing the MSIV logic are in the bypass position. One keylocked switch is in each isolation logic string.

B. When the turbine stop valves are less than 90 percent full open, the four independent contacts of the turbine stop valve position switch sensor relay of the RPS will trip.

C. When the reactor is below 1045 psig.

Of these three permissives on the manual bypass of the MSIV closure, only the reactor pressure permissive (item C above) does not have a safety function. This is also the only permissive that is proposed for deletion. With the setpoint at 1045 psig, which is the same setpoint for the high reactor pressure scram, the manual bypass can be performed at any operating reactor pressure provided the other two permissives are cleared. When the manual bypass is activated, plant protection is provided by those two other permissives by the normal scram and isolation signals, e.g., turbine stop valve position, low reactor water level, high steam flow, high steam tunnel temperature, and turbine building temperature, and by the annunciators in the control room. Eliminating the reactor pressure permissive does not affect the existing plant protection in any way.

We have reviewed the acceptability of these proposed Technical Specification revisions. 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 (revisions (1) and (3) as discussed on pages 7 and 8 of this evaluation), the analytical limits used in the setpoint calculations were the original analytical limits used in the Hatch Safety Analysis. For the analytical limits that were revised, the licensee stated that, in no case with these new limits do the FSAR analyzed transients or accidents 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 Unit 2 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 and allows a range of values that the trip unit may vary. 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 range + 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 was 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.

In conclusion, we have previously reviewed (Reference 1) the use of the ATTS and found that, provided certain interface requirements were satisfied, the system is acceptable. 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 2.

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.

2.6 The ARTS Improvement Program

Each of the components of the improvement program is discussed below:

2.6.1 Extended Load Line Limit Analysis

The effect on transient analysis and core stability were examined for the extended load line limit operation which permits higher powers for low flow conditions by changing the slope of the APRM rod block line. The effect is to allow operation at 100 percent power for greater than 87 percent flow and to increase the permitted power at 40 percent flow by about 5 percent to 63.2 percent.

2.6.1.1 Transient and Accident Analyses

The transient and accident analyses described in the evaluation of the ARTS program below have all assumed operation with the extended load line limit. The changes in core behavior caused by the extended operating range have thus been accounted for in the revised analyses.

2.6.1.2 Thermal-Hydraulic Stability

The results of the thermal-hydraulic analysis (NEDO-30260, Reference 5) show that the core has the smallest stability margin for the power/flow map at the point where the extrapolated rod block line intercepts the natural circulation line and the corresponding maximum decay ratios are 0.91 and 0.93 for Unit 1 and Unit 2, respectively. In addition, the licensee has committed that operating procedures for both Units, prior to Cycle 4 startup for Unit 2, will implement the recommendations of General Electric Company Service Information Letter (SIL) #380 regarding precautionary monitoring of local and average power instrumentation to avoid unstable operation at low flow and will provide for insertion of control rods to or below the 80% rod line in event of pump trip leading to single loop operation. Since (1) the calculated maximum decay ratios are less than that of some of the operating plants (for example, Peach Bottom Units 2 and 3 have the maximum decay ratio of 0.98), (2) there will be added margin to the core stability because the Technical Specifications prohibit natural circulation as a normal operating mode, and (3) since the licensee is implementing operating procedures to assure thermal-hydraulic stability while operating at low flow or with a single recirculation loop in service, we have concluded that the thermal-hydraulic stability results are acceptable for extension of the load line limits for Unit 1, Cycle 7 and Unit 2, Cycle 4 operations.

2.6.2 APRM System Improvements

Each APRM channel consists of a number of LPRMs which are chosen in such a way that the channel output is proportional to core power. The APRM signals are compared to a fixed scram trip (at 120% full power) and to a flow biased rod withdrawal block trip. In addition, the APRM signals are passed through a filter having a time constant of approximately six seconds to form the simulated thermal power monitor (STPM). The STPM output is then compared to a flow biased scram trip.

Current Hatch Technical Specifications require that the flow biased APRM setpoints be lowered (set down) if the core maximum fraction of limiting power density (CMFLPD) exceeds the fraction of rated power (FRP). This may be accomplished by increasing the APRM channel gain.

If CMFLPD exceeds FRP and the core power is raised to its full value, the operating limit value for MAPLHGR or MCPR would be exceeded and the assumptions used in the plant transient analyses violated.

In the proposed APRM system, the setdown requirement would be removed. It would be replaced by power and flow dependent MAPLHGR and MCPR limits. Analyses have been performed to obtain the multipliers to be applied to the full power values of MAPLHGR and MCPR in order to prevent violation of safety criteria during transients and accidents. The LOCA and limiting transients were reanalyzed.

2.6.2.1 Loss-of-Coolant Accident

Previous analyses of the LOCA at less than rated flow have assumed operation under the proposed flow bias APRM rod block line (0.58 W + 50) with the APRM setdown in effect. The analyses showed that no revision of the MAPLHGR limits was required for low flow. However, if the setdown factor (FRP/CMFLPD) is not applied, the evaluation shows that a factor of 0.86 must be applied to the MAPLHGR operating limits when core flow is below 61 percent of rated flow. Accordingly, a Technical Specification curve including this factor, along with others, as described below, is constructed for the Hatch Plants. Because the LOCA analyses were performed with previously used and approved methods, we find them to be acceptable.

2.6.2.2 Transients

In order to restore safety margins which might be reduced when the APRM setdown is removed, the limiting transients were reanalyzed assuming the absence of this feature. The analyses assumed operation within the proposed extended power/flow domain with flows up to 105 percent of rated flow. Analyses of the transient events were made as a function of initial power and flow and the results used to determine multipliers to be applied to full power-full flow values of MCPR and MAPLHGR. The power dependence was most sensitive at full flow and the feedwater controller failure was the transient showing the largest sensitivity. This event was then used to construct a curve of MCPR multiplier, K_p, and MAPLHGR multiplier, $MAPFAC_p$, as a function of core power. Conservative curves were drawn in order to bound future cycles.

Flow dependence of MCPR and MAPLHGR was determined from analyses of flow runout events in which the core flow is ramped rapidly upward to the maximum value permitted by the setting of the recirculation pump scoop. The flow multipliers, K_f and $MAPFAC_f$, are thus a function of the initial flow and the maximum flow and a family of curves is drawn. The multipliers are chosen so that a flow runout to the maximum flow will not result in a violation of MCPR or LHGR safety limits. The $MAPFAC_f$ curves are combined with the results of the LOCA analysis described above and the combined family of curves is used in the Technical Specifications. For inclusion in the Technical Specifications, the K_f curve family is transposed to a $MCPR_f$ family by assuming a value of 1.2 for the full flow MCPR. This is the lowest value that may be used for the Hatch Units (constrained by the ECCS analyses).

The discussion immediately above applies to the power range from 30 to 100 percent of full power. Below 30 percent of full power the turbine stop and control valve scrams are bypassed and the analyses do not apply. Below 25 percent of full power no MCPR and MAPLHGR limits are defined. In the interval between 25 and 30 percent of full power, flow dependent effects are taken into account by having two power dependent curves - one for flows greater than 50 percent of rated and one for lower flows. Analyses are then performed to obtain limiting MCPR and MAPLHGR values in these domains.

Approved methods were used to perform the analyses described above except for those used for the loss of feedwater heater event. For that event the trend analysis was performed by a code approved for other purposes. However this event is not limiting and safety analyses for the event are done by approved methods. We find this acceptable.

We conclude that deletion of the APRM setdown requirement is acceptable when it is replaced by the power and flow dependent operating limits described above.

2.6.3 Rod Block Monitor System Improvements

The Rod Block Monitor (RBM) System is used to prevent violation of fuel thermal-hydraulic limits in the event of inadvertent continuous withdrawal of a control rod. When a rod is selected for withdrawal, the surrounding LPRM strings are selected. Their response to the withdrawal is monitored, and a withdrawal block is initiated by the RBM if that response exceeds certain limits. These limits are selected so that no violation of fuel limits occurs. The RBM has two independent channels either of which will initiate a rod block if tripped.

The proposed Rod Block Monitor improvements include:

1. Re-ordering of the assignment of LPRM detectors to the two RBM channels in order to increase instrument sensitivity and provide more uniformity of response between the two channels.

2. Changing the baseline normalization of the RBM from an APRM channel to a fixed signal in order to reduce the number of unnecessary rod blocks.

3. Replacing the flow-biased trip setpoints with fixed power-dependent trip setpoints, and

4. Elimination of the resettable trips in order to make operation simpler.

In addition, the electronics hardware has been updated to increase the reliability of operation.

The change in LPRM assignments is described in the licensee submittal and a comparison of the RBM channel responses to those of the current design made. The revised design shows similar responses for the two channels each of which has a response similar to that of the most responsive channel in the current design.

A block diagram of the revised RBM system is presented and a discussion of the electronics changes given in Appendix A to NEDC-30474-P (Reference 6). We conclude that sufficient information is given in the report to permit the conclusion that the proposed revisions to the RBM system design are acceptable. The electronics changes are discussed separately below.

2.6.3.1 Reanalysis of Rod Withdrawal Error Event

The revisions of the RBM system necessitate the re-evaluation of the Rod Withdrawal Error Event. The present deterministic, bounding, cycle specific analysis is replaced with a statistical analysis valid for application to all Hatch cores using GE fuel up to type P8x8R inclusive. A data base calculated from actual plant operating states was created which covers the spectrum of plant sizes and power densities. The data base construction began with the selection of operating states at near full power which had low MCPRs and/or high MAPLHGRs in bundles near deeply inserted control rods. The rod configurations were then adjusted to bring the MCPR values to approximately 1.20. Thirty-nine such configurations were chosen. In order to investigate power and flow dependence, the rod configuration in 26 of the above cases was held constant, the flow was reduced to 40 percent of rated and xenon allowed to equilibrate. Finally, for the 26 cases, the flow was held constant at 40 percent and the rod pattern altered to yield 40 percent power with no xenon. For each of the 91 cases described above 100 rod withdrawal error (RWE) analyses were performed assuming a random distribution of starting points for the error rod (and thus initial MCPR values, $MCPR_I$,) and random failures of the LPRMs which provide inputs to the Rod Block Monitor. All cases which did not result in a rod block were rejected from the data base unless the rod started from the fully inserted position. A 15 percent random failure rate was assigned to each LPRM. Experience has shown this value to be high.

The Rod Block Monitor response was generated as a function of error rod position for each RWE. The currently used and approved methods were employed in the analyses. The results were tabulated as error rod position vs assumed Rod Block Monitor setting. These results were then transformed into values of normalized MCPR change ($^{\Delta MCPR}/_{MCPR_I}$) and the mean and standard deviation of the distribution for each set of 100 RWE analyses were determined for each RBM setting. These data were then combined to obtain a mean and standard deviation for the entire data base at each power/flow state for each RBM channel at each assumed RBM setting.

A plot of the required initial MCPR value ($MCPR_I$) as a function of Rod Block Monitor trip setting is constructed. The required value of $MCPR_I$ is that which assures that 95 percent of the rod withdrawal errors which are initiated from it do not violate the MCPR safety limit (1.07) with a 95 percent confidence level.

The final step is the selection of suitable setpoints for the Rod Block Monitor. These are chosen so that the rod withdrawal event is not limiting. At any power level the required operating limit MCPR for this event is not greater than that required for other transients as described in Section 2.6.2 above. A value of 1.20 at full power/full flow is dictated by ECCS considerations. In keeping with the three trip settings of the present system, the power range from 25 percent to full power is divided into three intervals with a constant setpoint in each interval. For Hatch the intervals are 30-65, 65-85, and 85-100 percent of full power. The analytic setpoints for the intervals are respectively, 118, 112, and 108 percent of the reference signal.

The effect of the absence of LPRM strings for certain rods near the periphery of the core has been analyzed and it was shown that the setpoints described above are adequate to mitigate the consequences of the rod withdrawal error on the periphery of the core.

A downscale trip at about 94 percent of the reference signal also inhibits rod withdrawal.

The analyses described above assumed unfiltered LPRM signal inputs to the RBM. However provision is made in the instrument for a filter having a time constant of up to 0.55 seconds. Use of such a filter would necessitate the reduction of the setpoints given above by an amount which depends on the time constant chosen. Analyses were performed to determine the required adjustments and the applicable values are given in NEDC-30474-P (Reference 5). If anything other than no filtering is chosen, the maximum time constant is recommended. In addition, a delay occurs between the time when the input signal reaches the setpoint and the imposition of the rod block. A value of 2.0 seconds was assumed for this delay and no greater value may be permitted. This value is incorporated into the Technical Specifications.

In order to confirm the use of a 15 percent failure probability in the statistical analysis, a sensitivity study was performed in which failure rates up to 30 percent were assumed. Increasing the failure rate to the higher value had a negligible effect on the results.

The Rod Block Monitor is currently required to be operable when core power is greater than some low power setpoint (25-30 percent of full power). Additional surveillance is required if the core has a "limiting control rod pattern" - defined to be a pattern which causes the core to be at the operating limit on MCPR, APLHGR or LHGR. Strictly speaking however, the RBM is only required if the complete withdrawal of any single rod in the core would violate safety limits. Analyses have been performed using the data base described above to obtain operating limit MCPR values above which no rod withdrawal error could lead to violation of the limits. Two values are defined - one for power levels greater than 90 percent full power and one for levels from 25 to 90 percent full power. If the plant is operating at or below these limits, it is on a "limiting control rod pattern" and the RBM is required to be operable. It may be bypassed when operating above these limits.

2.6.3.2 Electrical Instrumentation and Control

The RBM system is designed to automatically detect and block control rod withdrawal that could violate Technical Specification safety limits during a single control rod withdrawal error (RWE) transient. It is assumed that the core is operated in compliance with plant Technical Specifications before the RWE event. There are two RBM channels, either of which can initiate a rod block (i.e., prevent control rod withdrawal). The RBM channels are powered from the Reactor Protection System (RPS) buses (RBM channel A is powered from RPS bus A, and RBM channel B is powered from RPS bus B). Although the RBM system is not safety related, separation is provided between the RBM channels to allow for single failures, and to allow one channel to be bypassed if necessary. RBM channel bypass is accomplished via a single three position bypass switch such that only one RBM channel can be bypassed at a time. Both RBM channels are operable when the switch is placed in the center (normal) position. Both local and remote indication of an RBM channel bypass are provided via indicator lights. The licensee has stated that implementation of the ARTS program will not compromise the redundancy provided between the RBM channels, and that isolation will be maintained between the RBM system and safety related circuits. The RBM output functions (i.e., recorders located on the reactor operator's console, local meters, trip units, and the on-line computer) will remain unchanged, although in some cases the signals used for these functions have been modified. The hardware changes involved in the ARTS modification include new model printed circuit (PC) cards, relays, relay sockets, mounting hardware, and wiring.

Upon selecting a control rod for movement, each RBM channel automatically computes the average of all assigned (and unbypassed) local power range monitor channels. The average signal is then filtered (to reduce signal noise), delayed (to allow the signal to reach its maximum/equilibrium value), and then amplified to read the same as a fixed reference signal. This process (referred to as the RBM null sequence) is reinitiated each time a new rod is selected for movement. Control rod motion is blocked during the null sequence. Each RBM channel then compares the calibrated (nulled) signal to an automatically selected preset rod block alarm/trip level (one of three power biased upscale trip levels is selected dependent upon the current reactor power level). The trip level is selected based on the magnitude of a reference APRM. If the local neutron flux level increases to the upscale trip level, further control rod withdrawal is blocked, thus limiting the change (increase) in local power. Thus, the ARTS modification to the RBM trip logic replaces the standard RBM flow biased (recirculation flow) trip feature with power (neutron flux level) biased trips. This modification will be implemented by changes to PC card electronics (averaging cards, null sequence cards, RBM setpoint cards, and quad trip cards).

It should be noted that an adjustable time delay (t_{d2} 1 to 50 seconds ± 0.5 seconds) has been added to delay the calibrated (nulled) average local neutron flux signal to the RBM trip logic. The purpose of this delay is to allow minimum rod movements despite abnormally high signal noise not removed by filtering. This delay is typically set at a value of 1 to 2 seconds. The design of the control rod drive system is for a normal speed of 3 inches per second ± 0.6 inches per second. The licensee has analyses that show the delay is short enough to limit rod movement well below that which could cause a thermal limits violation. However, if this time delay is set above the minimum value, it is considered a bypass of the associated RBM channel since the analyses did not consider time delays in excess of the minimum value. The licensee has stated that testing and calibration of the time delay will be performed at each refueling as part of the RBM system calibration procedure. The licensee's supporting document, Reference 6, indicates that setting of t_{d2} above the minimum could be used as a means for bypassing the RBM. We have taken exception to this provision, and the licensee has stated that RBM channel bypass will be affected using only the RBM bypass switch. We consider this to be acceptable procedure.

Other RBM trip functions include too few LPRM inputs (either inoperative or bypassed), downscale (RBM signal abnormally low), and instrument inoperative (e.g., calibrate-operate switch not in the operate position and RBM equipment interlocks such as module removed and failure to null to the reference signal). The licensee has stated that the response time and accuracy (including setpoint drift) of the new RBM circuitry either equals or exceeds that of the existing design. All rod blocks are alarmed. The upscale rod block alarm can only be reset by activating a reset switch or selecting another rod for movement. Locally mounted color coded status lights are provided to indicate the type of rod block (upscale-amber, instrument inspective and downscale-white).

The RBM system is required to be operable whenever a limiting rod pattern exists. A limiting rod pattern exists when any control rod in the core would result in violation of the safety limit MCPR if it were fully withdrawn. During operation with a limiting rod pattern, both RBM channels should be operable. If only one RBM channel is operable, an instrument functional test of the operable (unbypassed) channel must be performed prior to withdrawal of any control rods. If the inoperable channel is not restored within 24 hours, then all control rod withdrawal shall be blocked. If both RBM channels are inoperable, then all control rod withdrawal shall be blocked within an hour. We find the Hatch Technical Specification requirements for RBM system operability and the associated LCOs to be acceptable. It should be noted that the operators are responsible for determining whether a limiting rod pattern exists (and therefore, for determining RBM system operability requirements) prior to control rod withdrawal in accordance with plant operating procedures. We have found this to be acceptable. The APRM and RBM instrument surveillance requirements (i.e., instrument functional tests and calibrations) have not changed as a result of implementation of the ARTS improvement program.

2.6.4 Technical Specification Changes

Implementation of the hardware changes and revised analyses described above requires changes in the Hatch Nuclear Plant, Units 1 and 2, Technical Specifications. These changes are discussed below:

2.6.4.1 APRM Specification Changes

The requirement for the setdown of the trip setpoint is deleted from the specification and the setdown factor (Fraction of Rated Power divided by Core Maximum Fraction of Limiting Power Density) is removed from the equation for the trip setpoint. The slope and intercept of the APRM flow biased rod block line and of the APRM/STPM flow biased scram are altered to permit operation within the domain defined by the extended load line limit analysis.

2.6.4.2 Rod Block Monitor Technical Specifications

The RBM flow biased trip equation is replaced by power dependent setpoint definitions and incorporate RBM filter and time delay setpoints. Current operability requirements are replaced by the new ones including the revised definition of the limiting control rod pattern.

2.6.4.3 Thermal-Hydraulic Operating Limit Specifications

The following changes are required in the Power Distribution Limit Specifications:

1. A curve of MCPR multiplier, K_p, as a function of power must be added.

2. The K_f family of curves must be replaced with curves of $MCPR_f$ as a function of flow.

3. The MCPR Technical Specification must be altered to define the manner in which the two curves are combined with the full power, full flow value of the operating limit MCPR to obtain the power/flow dependent limit.

4. Power and flow dependent multiplier factors ($MAPFAC_p$ and $MAPFAC_f$) must be added and the MAPLHGR Technical Specification must be altered to define the manner in which the two curves are combined with the full power/full flow MAPLHGR curves to obtain the power and flow dependent MAPLHGR limits.

5. The bases for the various Technical Specifications must be modified to account for the altered Technical Specifications.

2.6.5 Conclusions

Based on our review, which is described above, we conclude that the proposed ARTS Improvement Program is acceptable for use in Hatch Nuclear Plant, Units 1 and 2. We further conclude that the supporting document, NEDC-30474-P (Reference 6), may be used as a reference to describe the program and its analyses for Hatch 1 and 2, and to describe the methods used in applications of this program to other reactors. This conclusion is based on the following:

1. The analysis methods used for the safety analyses presented in the report are those which have been previously used and approved for reload safety analyses.

2. The revised operating limits and procedures do not result in reductions to safety margins relative to current values. In general, margins are increased.

3. The revised operating procedures are simpler to follow which tends to increase operating safety.

4. Implementation of this design complies with the requirements of Section 7.7 (Control Systems) of the Standard Review Plan (NUREG-0800), and therefore, is acceptable. The separation provided between redundant RBM channels and the isolation provided between the RBM system and safety related circuits have not been compromised as a result of the ARTS modification.

3.0 ENVIRONMENTAL CONSIDERATION

The reload and ATTS portions of this amendment involve changes in the installation or use of a facility component located within the restricted area as defined in 10 CFR Part 20. We have determined that they involve 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 these items involve no significant hazards consideration and there has been no public comment on such finding. Accordingly, these items of the amendment meet 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 these items of the amendment.

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

Dated: July 13, 1984

The following NRC personnel have contributed to this Safety Evaluation: Jerry Mauck, Marty Virgilio, Rick Kendall, Walter Brooks, and S. Sun.

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