December 29, 1994

Mr. John J. Barton
Vice President and Director
GPU Nuclear Corporation
Oyster Creek Nuclear Generating Station
Post Office Box 388
Forked River, NJ 08731

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Docket File	PMcKee
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	JFRogge, RI

SUBJECT: ISSUANCE OF AMENDMENT (TAC NO. M81944)

Dear Mr. Barton:

The Commission has issued the enclosed Amendment No. 176 to Facility Operating License No. DPR-16 for the Oyster Creek Nuclear Generating Station, in response to your application dated October 9, 1991, as supplemented March 9, and April 27, 1994.

The amendment establishes additional requirements for the availability of Local Power Range Monitors (LPRMs) associated with the Average Power Range Monitoring (APRM) system. These additional requirements further restrict the allowable number of out-of-service LPRM/APRM detectors in order to ensure a sufficient response to regional thermal hydraulic oscillations in the reactor core to prevent violation of the Minimum Critical Power Ratio (MCPR) safety limit. The amendment also identifies a lower bound MCPR operating limit for each cycle as identified in the Core Operating Limits Report. This limit shall be greater than or equal to 1.47.

A copy of the related Safety Evaluation is also enclosed. The notice of issuance will be included in the Commission's biweekly Federal Register notice.

Sincerely,

Original signed by Ronald W. Hernan

for: Alexander W. Dromerick, Senior Project Manager
Project Directorate I-4
Division of Reactor Projects - I/II
Office of Nuclear Reactor Regulation

Docket No. 50-219

Enclosures: 1. Amendment No. 176 to DPR-16
2. Safety Evaluation

cc w/encls: See next page

CP-1

DOCUMENT NAME: G:\DROMERIC\M81944.AMD *See previous concurrence

OFFICE	LA:PDI-4		PM:PDI-4		D:PDI-4		OGC*			
NAME	SNorris		ADromerick:cn		PMcKee		See con. copy			
DATE	12/27/94		12/29/94		12/29/94		11/2/94			

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OFFICE	LA:PDI-4	PM:PDI-4	D:PDI-4	OGC*	
NAME	SNorris	ADromerick:	PMcKee	See con. copy	
DATE	12/21/94	12/29/94	12/29/94	11/2/94	

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UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D.C. 20555-0001

December 29, 1994

Mr. John J. Barton
Vice President and Director
GPU Nuclear Corporation
Oyster Creek Nuclear Generating Station
Post Office Box 388
Forked River, NJ 08731

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Dear Mr. Barton:

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The amendment establishes additional requirements for the availability of Local Power Range Monitors (LPRMs) associated with the Average Power Range Monitoring (APRM) system. These additional requirements further restrict the allowable number of out-of-service LPRM/APRM detectors in order to ensure a sufficient response to regional thermal hydraulic oscillations in the reactor core to prevent violation of the Minimum Critical Power Ratio (MCPR) safety limit. The amendment also identifies a lower bound MCPR operating limit for each cycle as identified in the Core Operating Limits Report. This limit shall be greater than or equal to 1.47.

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Ronald W. Hernan for

Alexander W. Dromerick, Senior Project Manager
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Office of Nuclear Reactor Regulation

Docket No. 50-219

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cc w/encls: See next page

cc:

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UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D.C. 20555-0001

GPU NUCLEAR CORPORATION

AND

JERSEY CENTRAL POWER & LIGHT COMPANY

DOCKET NO. 50-219

OYSTER CREEK NUCLEAR GENERATING STATION

AMENDMENT TO FACILITY OPERATING LICENSE

Amendment No. 176
License No. DPR-16

1. The Nuclear Regulatory Commission (the Commission) has found that:

   A. The application for amendment by GPU Nuclear Corporation, et al. (the licensee), dated October 9, 1991, as supplemented March 9, April 27, and December 15, 1994, complies with the standards and requirements of the Atomic Energy Act of 1954, as amended (the Act), and the Commission's rules and regulations set forth in 10 CFR Chapter I;

   B. The facility will operate in conformity with the application, the provisions of the Act, and the rules and regulations of the Commission;

   C. There is reasonable assurance (i) that the activities authorized by this amendment can be conducted without endangering the health and safety of the public, and (ii) that such activities will be conducted in compliance with the Commission's regulations;

   D. The issuance of this amendment will not be inimical to the common defense and security or to the health and safety of the public; and

   E. The issuance of this amendment is in accordance with 10 CFR Part 51 of the Commission's regulations and all applicable requirements have been satisfied.

2. Accordingly, the license is amended by changes to the Technical Specifications as indicated in the attachment to this license amendment, and paragraph 2.C.(2) of Facility Operating License No. DPR-16 is hereby amended to read as follows:

   (2) Technical Specifications

   The Technical Specifications contained in Appendices A and B, as revised through Amendment No. 176 , are hereby incorporated in the license. GPU Nuclear Corporation shall operate the facility in accordance with the Technical Specifications.

3. This license amendment is effective as of the date of issuance, to be implemented within 30 days of issuance.

FOR THE NUCLEAR REGULATORY COMMISSION

Ronald W. Hernan for

Phillip F. McKee, Director
Project Directorate I-4
Division of Reactor Projects - I/II
Office of Nuclear Reactor Regulation

Attachment: Changes to the Technical Specifications

Date of Issuance: December 29, 1994

ATTACHMENT TO LICENSE AMENDMENT NO. 176

FACILITY OPERATING LICENSE NO. DPR-16

DOCKET NO. 50-219

Replace the following pages of the Appendix A Technical Specifications with the enclosed pages as indicated. The revised pages are identified by amendment number and contain vertical lines indicating the areas of change.

Remove	Insert
3.1-1	3.1-1
3.1-2	3.1-2
3.1-3	3.1-3
3.1-4	3.1-4
3.1-5	3.1-5
3.1-6	3.1-6
3.1-7	3.1-7
3.10-2	3.10-2
3.10-3	3.10-3

3.1 PROTECTIVE INSTRUMENTATION

Applicability: Applies to the operating status of plant instrumentation which performs a protective function.

Objective: To assure the OPERABILITY of protective instrumentation.

Specifications: A. The following operating requirements for plant protective instrumentation are given in Table 3.1.1:

1. The reactor mode in which a specified function must be OPERABLE including allowable bypass conditions.

2. The minimum number of OPERABLE instrument channels per OPERABLE trip system.

3. The trip settings which initiate automatic protective action.

4. The action required when the limiting conditions for operation are not satisfied.

B. 1. Failure of four chambers assigned to any one APRM shall make the APRM inoperable.

2. Failure of two chambers from one radial core location in any one APRM shall make that APRM inoperable.

3. Except during the performance of Technical Specification required LPRM/APRM surveillance, reactor power shall be reduced below the 80% rod line or the corresponding RPS trip system shall be placed in the tripped condition, whenever all three of the following conditions exist:

    1. Reactor power is greater than 35%

    -and-

    2. More than one LPRM detector is bypassed or failed in the A level or the B level assigned to a single APRM channel

    -and-

    3. The diagonally opposite quadrant contains a single APRM channel with more than one bypassed or failed LPRM detector on the same axial level as the bypassed or failed detectors specified in (2) above.

c. Any two (2) LPRM assemblies which are input to the APRM system and are separated in distance by less than three (3) times the control rod pitch may not contain a combination of more than three (3) inoperable detectors (i.e., APRM channel failed or bypassed, or LPRM detectors failed or bypassed) out of the four (4) detectors located in either the A and B, or the C and D levels.

2. A Travelling In-Core Probe (TIP) chamber may be used as an APRM input to meet the criteria of 3.1.B and 3.1.C.1, provided the TIP is positioned in close proximity to one of the failed LPRM's. If the criteria of 3.1.B.2 or 3.1.C.1 cannot be met, POWER OPERATION may continue at up to rated power level provided a control rod withdrawal block is OPERATING or at power levels less than 61% of rated power until the TIP can be connected, positioned and satisfactorily tested, as long as Specification 3.1.B.1 and Table 3.1.1 are satisfied.

Bases: The plant protection system automatically initiates protective functions to prevent exceeding established limits. In addition, other protective instrumentation is provided to initiate action which mitigates the consequences of accidents or terminates operator control. This specification provides the limiting conditions for operation necessary to preserve the effectiveness of these instrument systems.

Table 3.1.1 defines, for each function, the minimum number of OPERABLE instrument channels for an OPERABLE trip system for the various functions specified. There are usually two trip systems required or available for each function. The specified limiting conditions for operation apply for the indicated modes of operation. When the specified limiting condition cannot be met, the specified Actions Required shall be undertaken promptly to modify plant operation to the condition indicated in a normal manner. Conditions under which the specified plant instrumentation may be out-of-service are also defined in Table 3.1.1.

Except as noted in Table 3.1.1 an inoperable trip system will be placed in the tripped condition. A tripped trip system is considered OPERATING since by virtue of being tripped it is performing its required function. All sensors in the untripped trip system must be OPERABLE, except as follows:

1. The high temperature sensor system in the main steam line tunnel has eight sensors in each protection logic channel. This multiplicity of sensors serving a duplicate function permits this system to operate for twenty month nominal intervals without calibration. Thus, if one of the temperature sensors causes a trip in one of the two trip systems, there are several cross checks that would verify if this were a real one. If not, this sensor could be removed from service. However, a minimum of two of eight are required to be OPERABLE and only one of the two is required to accomplish a trip in a single trip system.

2. One APRM of the four in each trip system may be bypassed without tripping the trip system if core protection is maintained. Core protection is maintained by the remaining three APRM's in each trip system as discussed in Section 7.5.1.8.7 of the Updated FSAR.

3. One IRM channel in each of the two trip systems may be bypassed without compromising the effectiveness of the system. There are few possible sources of rapid reactivity input to the system in the low power low flow condition. Effects of increasing pressure at zero or low void content are minor, cold water from sources available during startup is not much colder than that already in the system, temperature coefficients are small, and control rod patterns are constrained to be uniform by operating procedures backed up by the rod worth minimizer. Worth of individual rods is very low in a uniform rod pattern. Thus, of all possible sources of reactivity input, uniform control rod withdrawal is the most probable cause of significant power rise. Because the flux distribution associated with uniform rod withdrawals does not involve high local peaks, and because several rods must be moved to change power by a significant percentage of rated, the rate of power rise is very slow. Generally the heat flux is in near equilibrium with the fission rate. In an assumed uniform rod withdrawal approach to the scram level, the rate of power rise is no more than five percent of rated per minute, and three OPERABLE IRM instruments in each trip system would be more than adequate to assure a scram before the power could exceed the safety limit. In many cases, if properly located, a single OPERABLE IRM channel in each trip system would suffice.

4. When required for surveillance testing, a channel is made inoperable. In order to be able to test its trip function to the final actuating device of its trip system, the trip system cannot already be tripped by some other means such as a mode switch, interlock, or manual trip. Therefore, there will be times during the test that the channel is inoperable but not tripped. For a two channel trip system, this means that full reliance is being placed on the channel that is not being tested. A channel may be placed in an inoperable status for up to 6 hours for required surveillance without placing the trip system in the tripped condition provided at least one OPERABLE channel in the same trip system is monitoring that parameter.

5. Allowed outage times (AOT) to permit restoration of inoperable instrumentation to OPERABLE status are provided in Table 3.1.1. AOTs vary depending on type of function and the number of inoperable channels per function. If an inoperable channel cannot be restored to OPERABLE status within the AOT, the channel or the associated trip system must be placed in the tripped condition. Placing the inoperable channel in trip (or the associated trip system in trip) conservatively compensates for the inoperability and allows operation to continue. Alternatively, if it is not desired to place the channel (or trip system) in trip (e.g., as in the case where placing the inoperable channel in trip would result in a full scram) the Action Required must be taken.

AOTs discusse n 4 (6 hours for surveillance) id 5 (repair AOTs in Table 3.1.1, Notes nn, oo and pp) above have been determined in accordance with References 1 through 6 except for instrumentation in Table 3.1.1, Sections M and N. Note kk has been provided to specify a 2 hour surveillance AOT for those instruments.

Bypasses of inputs to a trip system other than the IRM and APRM bypasses are provided for meeting operational requirements listed in the notes in Table 3.1.1. Note 'a' allows the "high water level in scram discharge volume" scram trip to be bypassed in the refuel mode. In order to reset the safety system after a scram condition, it is necessary to drain the scram discharge volume to clear this scram input condition. (This condition usually follows any scram, no matter what the initial cause might have been.) In order to do this, this particular scram function can be bypassed only in the refuel position. Since all of the control rods are completely inserted following a scram, it is permissible to bypass this condition because a control rod block prevents withdrawal as long as the switch is in the bypass condition for this function.

The manual scram associated with moving the mode switch to shutdown is used merely to provide a mechanism whereby the reactor protection system scram logic channels and the reactor manual control system can be energized. The ability to reset a scram twenty (20) seconds after going into the SHUTDOWN MODE provides the beneficial function of relieving scram pressure from the control rod drives which will increase their expected lifetime.

To permit plant operation to generate adequate steam and pressure to establish turbine seals and condenser vacuum at relatively low reactor power, the main condenser vacuum trip is bypassed until 600 psig. This bypass also applies to the main steam isolation valves for the same reason.

The action required when the minimum instrument logic conditions are not met is chosen so as to bring plant operation promptly to such a condition that the particular protection instrument is not required; or the plant is placed in the protection or safe condition that the instrument initiates. This is accomplished in a normal manner without subjecting the plant to abnormal operations conditions. The action and out-of-service requirements apply to all instrumentation within a particular function, e.g., if the requirements on any one of the ten scram functions cannot be met then control rods shall be inserted.

The trip level settings not specified in Specification 2.3 have been included in this specification. The bases for these settings are discussed below.

The high drywell pressure trip setting is $\leq$ 3.5 psig. This trip will scram the reactor, initiate core spray, initiate primary containment isolation, initiate automatic depressurization in conjunction with low-low-low-reactor water level, initiate the standby gas treatment system and isolate the reactor building. The scram function shuts the core down during the loss-of-coolant accidents. A steam leak of about 15 gpm and a liquid leak of about 35 gpm from the primary system will cause drywell pressure to reach the scram point; and, therefore, the scram provides protection for breaks greater than the above.

High drywell pressure provides a second means of initiating the core spray to mitigate the consequences of loss-of-coolant accident. Its trip setting of $\leq$3.5 psig initiates the core spray in time to provide adequate core cooling. The break size coverage of high drywell pressure was discussed above. Low-low water level and high drywell pressure in addition to initiating core spray also causes isolation valve closure. These settings are adequate to cause isolation to minimize the offsite dose within required limits.

It is permissible to make the drywell pressure instrument channels inoperable during performance of the integrated primary containment leakage rate test provided the reactor is in the COLD SHUTDOWN condition. The reason for this is that the Engineered Safety Features, which are effective in case of a LOCA under these conditions, will still be effective because they will be activated (when the Engineered Safety Features system is required as identified in the technical specification of the system) by low-low reactor water level.*

The scram discharge volume has two separate instrument volumes utilized to detect water accumulation. The high water level is based on the design that the water in the SDIV's, as detected by either set of level instruments, shall not be allowed to exceed 29.0 gallons; thereby, permitting 137 control rods to scram. To provide further margin, an accumulation of not more than 14.0 gallons of water, as detected by either instrument volume, will result in a rod block and an alarm. The accumulation of not more than 7.0 gallons of water, as detected in either instrument volume will result in an alarm.

Detailed analyses of transients have shown that sufficient protection is provided by other scrams below 45% power to permit bypassing of the turbine trip and generator load rejection scrams. However, for operational convenience, 40% of rated power has been chosen as the setpoint below which these trips are bypassed. This setpoint is coincident with bypass valve capacity.

A low condenser vacuum scram trip of 20 inches Hg has been provided to protect the main condenser in the event that vacuum is lost. A loss of condenser vacuum would cause the turbine stop valves to close, resulting in a turbine trip transient.

The low condenser vacuum trip provides a reliable backup to the turbine trip. Thus, if there is a failure of the turbine trip on low vacuum, the reactor would automatically scram at 20 inches Hg. The condenser is capable of receiving bypass steam until 7 inches Hg vacuum thereby mitigating the transient and providing a margin.

The settings to isolate the isolation condenser in the event of a break in the steam or condensate lines are based on the predicted maximum flows that these systems would experience during operation, thus permitting operation while affording protection in the event of a break. The settings correspond to a flow rate of less than three times the normal flow rate of $3.2X10^{5}$ lb/hr. Upon initiation of the alternate shutdown panel, this function is bypassed to prevent spurious isolation due to fire induced circuit faults.

The setting c ten times the stack release lin for isolation of the air-ejector offgas line is to permit the operator to perform normal, immediate remedial action if the stack limit is exceeded. The time necessary for this action would be extremely short when considering the annual averaging which is allowed under 10 CFR 20.106, and, therefore, would produce insignificant effects on doses to the public.

Four radiation monitors are provided which initiate isolation of the reactor building and operation of the standby gas treatment system. Two monitors are located in the ventilation ducts, one is located in the area of the refueling pool and one is located in the reactor vessel head storage area. The trip logic is basically a 1 out of 4 system. Any upscale trip will cause the desired action. Trip settings of 17 mr/hr in the duct and 100 mr/hr on the refueling floor are based upon initiating standby gas treatment system so as not to exceed allowed dose rates of 10 CFR 20 at the nearest site boundary.

The SRM upscale of 5×10^5 CPS initiates a rod block so that the chamber can be relocated to a lower flux area to maintain SRM capability as power is increased to the IRM range. Full scale reading is 1×10^6 CPS. This rod block is bypassed in IRM Ranges 8 and higher since a level of 5×10^5 CPS is reached and the SRM chamber is at its fully withdrawn position.

The SRM downscale rod block of 100 CPS prevents the instrument chamber from being withdrawn too far from the core during the period that it is required to monitor the neutron flux. This downscale rod block is also bypassed in IRM Ranges 8 and higher. It is not required at this power level since good indication exists in the Intermediate Range and the SRM will be reading approximately 5×10^5 CPS when using IRM Ranges 8 and higher.

The IRM downscale rod block in conjunction with the chamber full-in position and range switch setting, provides a rod block to assure that the IRM is in its most sensitive condition before startup. If the two latter conditions are satisfied, control rod withdrawal may commence even if the IRM is not reading at least 5%. However, after a substantial neutron flux is obtained, the rod block setting prevents the chamber from being withdrawn to an insensitive area of the core.

The APRM downscale setting of $\geq$ 2/150 full scale is provided in the RUN MODE to prevent control rod withdrawal without adequate neutron monitoring.

High flow in the main steamline is set at 120% of rated flow. At this setting the isolation valves close and in the event of a steam line break limit the loss of inventory so that fuel clad perforation does not occur. The 120% flow would correspond to the thermal power so this would either indicate a line break or too high a power.

Temperature sensors are provided in the steam line tunnel to provide for closure of the main steamline isolation valves should a break or leak occur in this area of the plant. The trip is set at 50°F above ambient temperature at rated power. This setting will cause isolation to occur for main steamline breaks which result in a flow of a few pounds per minute or greater. Isolation occurs soon enough to meet the criterion of no clad perforation.

The low-low- water level trip point is set 4'8" above the top of the active fuel and will prevent spurious operation of the automatic relief system. The trip point established will initiate the automatic depressurization system in time to provide adequate core cooling.

Specification 3.1.B.1 defines the minimum number of APRM channel inputs required to permit accurate average core power monitoring. Specification 3.1.B.3 defines APRM channel input operability requirements in order to ensure a sufficient APRM response to regional power oscillations. Specifications 3.1.B.2 and 3.1.C.1 further define the distribution of the OPERABLE chambers to provide monitoring of local power changes that might be caused by a single rod withdrawal. Any nearby, OPERABLE LPRM chamber can provide the required input for average core monitoring. A Travelling Incore Probe or Probes can be used temporarily to provide APRM input(s) until LPRM replacement is possible. Since APRM rod block protection is not required below 61% of rated power, as discussed in Section 2.3, Limiting Safety System Settings, operation may continue below 61% as long as Specification 3.1.B.1 and the requirements of Table 3.1.1 are met. In order to maintain reliability of core monitoring in that quadrant where an APRM is inoperable, it is permitted to remove the OPERABLE APRM from service for calibration and/or test provided that the same core protection is maintained by alternate means.

In the rare event that Travelling In-core Probes (TIPs) are used to meet the requirements 3.1.B or 3.1.C, the licensee may perform an analysis of substitute LPRM inputs to the APRM system using spare (non-APRM input) LPRM detectors and change the APRM system as permitted by 10 CFR 50.59.

Under assumed loss-of-coolant accident conditions and certain loss of offsite power conditions with no assumed loss-of-coolant accident, it is inadvisable to allow the simultaneous starting of emergency core cooling and heavy load auxiliary systems in order to minimize the voltage drop across the emergency buses and to protect against a potential diesel generator overload. The diesel generator load sequence time delay relays provide this protective function and are set accordingly. The repetitive accuracy rating of the timer mechanism as well as parametric analyses to evaluate the maximum acceptable tolerances for the diesel loading sequence timers were considered in the establishment of the appropriate load sequencing.

Manual actuation can be accomplished by the operator and is considered appropriate only when the automatic load sequencing has been completed. This will prevent simultaneous starting of heavy load auxiliary systems and protect against the potential for diesel generator overload.

Also, the Reactor Building Closed Cooling Water and Service Water pump circuit breakers will trip whenever a loss-of-coolant accident condition exists. This is justified by Amendment 42 of the Licensing Application which determined that these pumps were not required during this accident condition.

C. Minimum Critical Power Ratio (MCPR)

During steady state power operation the MINIMUM CRITICAL POWER RATIO (MCPR) shall be equal to or greater than the MCPR limit as specified in the COLR.

The MCPR limit for each cycle as identified in the COLR shall be greater than or equal to 1.47.

When APRM status changes due to instrument failure (APRM or LPRM input failure), the MCPR requirement for the degraded condition shall be met within a time interval of eight (8) hours, provided that the control rod block is placed in operation during this interval.

For core flows other than rated, the nominal value for MCPR shall be increased by a factor of k_f, where k_f is as shown in the COLR.

If at any time during power operation it is determined by normal surveillance that the limiting value for MCPR is being exceeded for reasons other than instrument failure, action shall be initiated to restore operation to within the prescribed limits. If the steady state MCPR is not returned to within the prescribed limits within two [2] hours, action shall be initiated to bring the reactor to the cold shutdown condition within 36 hours. During this period, surveillance and corresponding action shall continue until reactor operation is within the prescribed limit at which time power operation may be continued.

Bases:

The Specification for average planar LHGR assures that the peak cladding temperature following the postulated design basis loss-of-coolant accident will not exceed the 2200°F limit specified in 10 CFR 50.46. The analytical methods and assumptions used in evaluating the fuel design limits are presented in FSAR Chapter 4.

LOCA analyses are performed for each fuel design at selected exposure points to determined APLHGR limits that meet the PCT and maximum oxidation limits of 10 CFR 50.46. The analysis is performed using GE calculational models which are consistent with the requirements of 10 CFR 50, Appendix K.

The PCT following a postulated LOCA is primarily a function of the average heat generation rate of all the rods of a fuel assembly at any axial location and is not strongly influenced by the rod to rod power distribution within an assembly. Since expected location variations in power distribution within a fuel assembly affect the calculated peak clad temperature by less than ± 20°F relative to the peak temperature for a typical fuel design, the limit on the average linear heat generation rate is sufficient to assure that calculated temperatures are below the limits specified in 10 CFR 50.46.

The maximum average planar LHGR limits for the various fuel types currently being used are provided in the COLR. The MAPLHGR limits for both five-loop and four-loop operation with the idle loop unisolated are shown. Four-loop operation with the idle loop isolated (suction, discharge and discharge bypass valves closed) requires that a MAPLHGR multiplier of 0.98 be applied to all fuel types. Additional requirements for isolated loop operation are given in Specification 3.3.F.2.

Fuel design evaluations are performed to demonstrate that the cladding 1% plastic strain and other fuel design limits are not exceeded during anticipated operational occurrences for operation with LHGRs up to the operating limit LHGR.

The analytical methods and assumptions used in evaluating the anticipated operational occurrences to establish the operating limit MCPR are presented in the FSAR, Chapters 4, 6 and 15 and in Technical Specification 6.9.1.f. To assure that the Safety Limit MCPR is not exceeded during any moderate frequency transient event, limiting transients have been analyzed to determine the largest reduction in Critical Power Ratio (CPR). The types of transients evaluated are pressurization, positive reactivity insertion and coolant temperature decrease. The operational MCPR limit is selected to provide margin to accommodate transients and uncertainties in monitoring the core operating state, manufacturing, and in the critical power correlation itself. This limit is derived by addition of the CPR for the most limiting transient to the safety limit MCPR designated in Specification 2.1.

A lower bound of 1.47 has been established for the operating limit MCPR value to provide sufficient margin to the MCPR safety limit in the event of reactor thermal-hydraulic instability. The 1.47 limit will be considered against the minimum operating CPR limit based on reload transient and accident analysis. The higher of core stability or reactor transient and accident determined MCPR will be used to determine the cycle operating limit.

The APRM response is used to predict when the rod block occurs in the analysis of the rod withdrawal error transient. The transient rod position at the rod block and corresponding MCPR can be determined. The MCPR has been evaluated for different APRM responses which would result from changes in the APRM status as a consequence of bypassed APRM channel and/or failed/bypassed LPRM inputs. The steady state MCPR required to protect the minimum transient CPR for the worst case APRM status condition (APRM Status 1) is determined in the rod withdrawal error transient analysis. The steady state MCPR values for APRM status conditions 1, 2, and 3 will be evaluated each cycle. For those cycles where the rod withdrawal error transient is not the most severe transient the MCPR Value for APRM status conditions 1, 2, and 3 will be the same and be equal to the limiting transient MCPR value.

UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D.C. 20555-0001

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

RELATED TO AMENDMENT NO. 176

TO FACILITY OPERATING LICENSE NO. DPR-16

GPU NUCLEAR CORPORATION AND

JERSEY CENTRAL POWER & LIGHT COMPANY

OYSTER CREEK NUCLEAR GENERATING STATION

DOCKET NO. 50-219

1.0 INTRODUCTION

By letter dated October 9, 1991, as supplemented March 9, April 27, and December 15, 1994, "Technical Specification Change Request No. 191" (Ref. 1), GPU Nuclear Corporation (GPUN/the licensee) requested changes to the Oyster Creek Nuclear Generating Station (OCNGS) technical specifications (TS). These changes provide (1) additional requirements for availability of local power range monitors (LPRM) associated with average power range monitors (APRM) and (2) a lower bound for the minimum critical power ratio (MCPR) limiting condition for operation (LCO). The changes are intended to ensure a suitable APRM response to core-wide or regional thermal-hydraulic power oscillations. Accompanying the proposed changes was the GPUN Topical Report No. 068, Rev. 2, "Licensing Basis for Oyster Creek Long-Term Solution to Reactor Instability," which discusses the OCNGS long-term solution (LTS) to core instability issues. The March 9, and April 27, 1994 letters provided clarifying information that did not change the initial proposed no significant hazards consideration determination.

The NRC staff and the BWR Owners Group (BWROG) have been working since 1988 to develop LTS for instability events. The BWROG has developed several LTS design concepts which cover the range of BWR types. The LTS-II concept, in particular, was developed for the BWR-2 class, which includes the OCNGS. It takes advantage of the BWR-2 reactor core quadrant based, flow-biased, APRM protection system to provide an appropriate scram signal for either a core wide or regional power oscillation event. The BWROG generic LTSs (including Option II) are described in the topical reports of References 2 and 3, and the staff evaluation of these reports for the generic aspects of the proposed systems is given in Ref. 4. That evaluation concluded, for aspects relevant to LTS-II that (1) the methodology used to evaluate the protection provided by LTS-II is acceptable, (2) LTS-II is acceptable for BWR-2s (with plant-specific implementation analysis to show the APRM scram provides sufficient protection for out of phase oscillations to avoid exceeding core power ratio (CPR) safety limits), (3) since protection is not provided for single fuel assembly channel instability, the stability of lead use assemblies in a core reload should be reviewed to ensure single channel instability will not occur, and (4) the recirculation drive flow channel should comply with the requirements of appropriate standards. The indicated plant specific aspects are addressed in this evaluation.

The NRC contractor, Oak Ridge National Laboratory (ORNL), assisted the staff in reviewing the stability related material submitted by GPUN. ORNL has provided a technical evaluation report (TER) that is included with this evaluation as Attachment 1. Also included as Attachment 2 is the staff review of the recirculation flow electronics upgrade proposed by GPUN to satisfy the intent of item (4) above from the staff LTS generic review.

2.0 EVALUATION

The GPUN topical report TR No. 068 describes (1) the OCNGS BWR-2 quadrant based, flow biased neutron flux scram, APRM system, (2) the APRM response to power oscillations, (3) design criteria relative to oscillation detection and response, (4) procedural actions such as avoidance of the region of potential instability on the power-flow map, and (5) supporting analyses. The supporting analyses are plant specific calculations of examples of core wide and regional oscillations, sufficient (1) to determine requirements for MCPR operating limits, in order to avoid exceeding the MCPR safety limit should oscillations occur, and (2) LPRM/APRM inoperable limits, to ensure acceptable determination of power distribution relevant to oscillation detection. These calculations provide the basis for the proposed TS changes.

This material was part of the review by ORNL discussed in the attached TER. The staff review agrees with and adopts the conclusions and basis for the conclusions presented in the TER. These conclusions are, in brief (1) LTS-II is applicable to the OCNGS and (2) the solution implementation satisfies the LTS criteria and the General Design Criteria 10 and 12. Also adopted are two reservations indicated in the TER. They are (1) that since LTS-II does not provide automatic protection for single fuel assembly channel instability, reload fuel assemblies, including lead use assemblies should be placed in the core only if it has been demonstrated by analysis, using an approved methodology, that the limiting channel stability decay ratios are equal to or better than for fuel designs, other than lead test assemblies, in industry service at the time of this review, and (2) the approval of the OCNGS submittal should not imply general approval of Figure 4.1 of TR No. 068.

In a letter of March 9, 1994 (Ref. 5) GPUN presented information on a modification to the recirculation flow monitoring electronics. This change and the submittal was provided (in part) to satisfy the intent of the conclusion by the staff in the safety evaluation for the generic LTS report (Ref. 4) concerning recirculation flow requirements as indicated in item (4) of the discussion of the generic evaluation above. This information has been reviewed by the staff and is discussed in detail in Attachment 2. The conclusions from the review, which are adopted as part of this evaluation, states that the modifications, which are part of the safety-related class 1-E plant protection system, meet staff acceptance criteria for such instrumentation, including independence and environmental and seismic qualification and are acceptable for use in connection with the OCNGS LTS-II.

As discussed above and in both of the Attachments, GPUN has proposed TS changes to ensure (1) the MCPR safety limit is not exceeded if oscillations occur and (2) there are a sufficient number and distribution of LPRMs to detect and act on oscillations. TS 3.1.B is augmented to require that power must be below the 80 percent rod line or the relevant trip system placed in a tripped condition when specified conditions for reactor power and bypassed A and B level LPRMs are exceeded. TS 3.10.C is augmented to provide a minimum operating limit MCPR of 1.47. The staff review of the analyses carried out to support these changes concludes that the analyses and the changes are acceptable.

The staff has reviewed the licensing basis submitted by GPUN for the LTS selected for OCNGS, and adopts the recommendations described in the attached reviews. GPUN has presented reactor specific information to augment the generic information in References 2 and 3, proposed changes to the TS and an upgrade of the protection system instrumentation, in order to adopt the BWROG LTS-II for detection and suppression of thermal-hydraulic instability power oscillations. The staff review finds the changes to the TS and protection system to be acceptable. There are two conditions to the acceptance:

(1) Fuel assemblies, including lead use assemblies, should be used in the OCNGS core only if analysis, by approved methodology, demonstrates that limiting channel stability decay ratios are equal to or better than for fuel designs, other than lead test assemblies, in industry service at the time of this review. In a letter dated December 15, 1994, the licensee committed to this condition.

(2) Approval of the GPUN submittal should not imply general approval of GPUN TR No. 068 Figure 4.1.

3.0 STATE CONSULTATION

In accordance with the Commission's regulations, the New Jersey State official was notified of the proposed issuance of the amendment. The State official had no comments.

4.0 ENVIRONMENTAL CONSIDERATION

The amendment changes a requirement with respect to installation or use of a facility component located within the restricted area as defined in 10 CFR Part 20. The NRC staff has determined that the amendment involves no significant increase in the amounts, and no significant change in the types, of any effluents that may be released offsite, and that there is no significant increase in individual or cumulative occupational radiation exposure. The Commission has previously issued a proposed finding that the amendment involves no significant hazards consideration, and there has been no public comment on such finding (56 FR 57697). Accordingly, the amendment meets the eligibility criteria for categorical exclusion set forth in 10 CFR 51.22(c)(9). Pursuant to 10 CFR 51.22(b) no environmental impact statement or environmental assessment need be prepared in connection with the issuance of the amendment.

5.0 CONCLUSION

The Commission has 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, (2) such activities will be conducted in compliance with the Commission's regulations, and (3) the issuance of the amendment will not be inimical to the common defense and security or to the health and safety of the public.

6.0 REFERENCES

1. Letter and attachment from J. Barton, GPUN, to U.S. NRC, dated October 9, 1991, "Oyster Creek Nuclear Generating Station, Technical Specification Change Request No. 191."

2. NEDO-31960, "BWR Owners Group, Long-Term Stability Solutions Licensing Methodology," May 1991.

3. NEDO-31960, Supplement 1, "BWR Owners Group Long-Term Stability Solutions Licensing Methodology," March 1992.

4. Letter and enclosure from A. Thadani, NRC to L. England, BWR Owners Group, dated July 12, 1993, "Acceptance for Referencing of Topical Reports NEDO-31960 and NEDO-31960 Supplement 1, "BWR Owners Group Long-Term Stability Solutions Licensing Methodology."

5. Letter and enclosure from R. Keaten, GPUN, to U.S. NRC, (Document), dated March 9, 1994, OCNGS TSCR No. 191 - Additional Information.

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Date: December 29, 1994

Attachments: 1. TER by ORNL
2. Safety Evaluation

ORNL/NRC/LTR-92/14

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Attachment 1

SUMMARY: CONCLUSIONS AND RECOMMENDATIONS

This report documents our review of the Technical Specification Change request No. 191[1] and the associated GPU Technical Report No. 068/R2,[2] "Licensing Basis for Oyster Creek Long Term Solution to Reactor Instability," which deal with technical specification modifications to satisfy the "Long Term Solution" requirements for the stability issue in Oyster Creek. The main conclusions of our review are two:

(1) A Long Term Solution of "Type II" is applicable to Oyster Creek because of its quadrant APRM and flow biased scram system.

(2) The solution implementation, as defined in the proposed Tech Spec changes, appears to satisfy the main criteria of a Long Term Solution by providing a viable detect and suppress function that will guarantee, in the case of an instability, a very small likelihood of core damage without the need of operator intervention. Therefore General Design Criteria (GDC) 12 is satisfied by Oyster Creek even if unstable power oscillations were to develop.

Based on this review, we recommend that the technical specification change No. 191 be approved, and that it be the basis for Oyster Creek conformance with the Long Term Solution requirements caused by the stability issue.

BASIS FOR REVIEW CONCLUSIONS

Based on the analyses in GPU TR-068/R2,[2] GPU has demonstrated that either core-wide (in-phase) or regional (out-of-phase) oscillations are likely to be detected by the existing flow-biased, quadrant APRM (average power range monitor) scram system in Oyster Creek without exceeding specified acceptable fuel design limits (SAFDLs). To guarantee this margin to SAFDLs, Oyster Creek requires that the operating limit MCPR be greater than 1.47, and that no more than one LPRM (local power range monitor) per quadrant at level A be out-of-service. These restrictions have been incorporated to Oyster Creek technical specifications (request No. 191[1]).

For these calculations, GPU postulated "credible" oscillation contours that were superimposed on the initial 3-D power distribution. The "credible" oscillation contours included core-wide and "first-order side-to-side" regional oscillations. Based on previous BWROG (Boiling Water Reactor Owners' Group) analyses,[3,4] GPU concluded that higher order regional oscillations and single channel thermohydraulic oscillations are highly unlikely. We have concurred with this BWROG position in ref. 5.

The APRM response was simulated for several conditions based on the above contours and a random number of failed LPRM sensors. Based on these analyses, GPU determined that as long as the operating limit CPR is greater than 1.47 and at least one LPRM at level A is operable per quadrant, the postulated oscillations did not result in SAFDLs violation.

RESERVATIONS

Although we agree with most of the technical basis in GPU Technical Report No. 068/R2, we have a series of reservations with respect to this implementation that do not invalidate the previous conclusions but are worth mentioning:

(1) Long Term Solutions of Type II (the one implemented in Oyster Creek) do not provide automatic protection in the event of single channel thermohydraulic instabilities. However, as argued convincingly by the BWR Owners Group, the likelihood of single channel thermohydraulic instability without first triggering a core-wide or out-of-phase instability is very small. Nevertheless, since automatic protection is not provided, we might want to restrict Oyster Creek to load only fuel elements that satisfy the recent NRC stability criteria of being at least as stable as existing fuels. This restriction would apply specially to lead use assemblies (LUAs) which are the ones that could lead to a loading a one or two unstable channels (if a really unstable LUA were to be used). In other words, it might be prudent to ask Oyster Creek to analyze the channel stability of any LUA's that they may want to load.

(2) The Justification report No. TR-068/R2 uses Fig. 4.1 that has been discredited by the BWROG. In particular, BWROG has decided to modify Solution Type III so that the use of a figure like Fig. 4.1 is unnecessary because they feel (and rightly so) that it is difficult if not impossible to generate a universal correlation for ΔCPR versus oscillation amplitude that applies to many fuel types and operating conditions. Fig. 4.1 is used in report TR-068/R2 to justify that the margins for the in-phase instability mode are very large. Certainly, the in-phase mode margins are larger that for the out-of-phase; so that if the solution is acceptable for the out-of-phase mode, it should be acceptable for the in-phase mode. In this respect, the use of Fig 4.1 is only as an example or an approximate justification given the large margins available in this case. We should be careful, however, that our approval of Oyster Creek's submittal with Fig. 4.1 in it is not interpreted as an implicit approval of this figure. In particular, the use of this general figure for Solution I-D (for small cores with inlet orifices) would probably not be acceptable unless the same type of margins are shown.

REFERENCES

1. Oyster Creek Nuclear Generating Station, *Technical Specification Change Request No. 191*, Docket No. 50-219, October 9, 1991.

2. GPU Technical Report No. 068/R2, *Licensing Basis for Oyster Creek Long Term Solution to Reactor Instability*, Rev 2. August 1991

3. General Electric Company, *BWR Owners' Group Long-Term Stability Solutions Licensing Methodology*, NEDO-31960, May 1991.

4. General Electric Company, *BWR Owners' Group Long-Term Stability Solutions Licensing Methodology*, NEDO-31960 Supplement 1, March 1992.

5. Jose March-Leuba, *Licensing Basis for Long-Term Solutions to BWR Stability Proposed by the BWR Owners' Group,* Oak Ridge National Laboratory ORNL/NRC/LTR-92/15, August 1992

UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D.C. 20555-0001

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

OYSTER CREEK TECHNICAL SPECIFICATION REQUEST - CORE STABILITY

REPLACEMENT OF EXISTING RECIRCULATION FLOW MONITORING ELECTRONICS

GPU NUCLEAR CORPORATION

OYSTER CREEK NUCLEAR GENERATING STATION

DOCKET NO. 50-219

1.0 INTRODUCTION

By letter dated October 9, 1991, as supplemented April 27, 1994, GPU Nuclear Corporation (GPUN/the licensee) requested an amendment to Facility Operating License No. DPR-16 to change the Oyster Creek Nuclear Generating Station (OCNGS) technical specifications (TSs) to establish additional requirements for the availability of Local Power Range Monitors (LPRMs) associated with the Average Power Range Monitor (APRM) system. The purpose of this change is to restrict the allowable number of out-of-service LPRM/APRM detectors to ensure the ability to detect and suppress power oscillations prior to exceeding the Minimum Critical Power Ratio (MCPR) safety limit. This request also identifies a lower bound MCPR operating limit for each cycle as identified in the Core Operating Limits Report (COLR).

Additionally, by letter dated March 9, 1994, the licensee submitted their planned modification to utilize Foxboro Specification 200 electronics in place of the existing recirculation flow monitoring electronics. The flow electronics are used to bias the APRM setpoint for the reactor power instability trip.

2.0 DISCUSSION

The APRM system consists of electronic equipment that averages the output signals from selected incore LPRM amplifiers and develops an output signal representative of the rated core thermal power. These LPRM signals are grouped together such that the resulting APRM signals provide coverage of expected power oscillations. The trip units associated with the APRM system actuate an automatic protective action when APRM signals exceed preset flow-biased values.

The APRM system consists of eight independent channels - two channels per core quadrant. Channels 1 through 4 are associated with Reactor Protection System (RPS) #1, and channels 5 through 8 are associated with RPS #2. Each core quadrant is monitored by two APRM channels, each of which is associated with a different RPS. The two APRM channels in a given core quadrant utilize the same four LPRM detector strings with RPS #1 APRM channel receiving inputs from A and C LPRM detectors, and RPS #2 APRM channel receiving inputs from B and D LPRM detectors. Each APRM channel normally averages the inputs of eight LPRM detectors.

The quadrant-based APRM system provides automatic reactor protection by generating a flow-biased reactor trip signal based on the output of the individual APRM channels. At least one APRM channel in each of the two RPS must have a high flux or inoperative trip condition to produce a full reactor scram.

As a result of aging, the existing recirculation (recirc) flow monitoring electronics for determining flow biasing, have exhibited poor drift characteristics, calibration problems, and a lack of spare parts. These problems have reduced the reliable operating life of the electronics to less than 24 months. Consequently, the licensee proposes a modification to replace the existing electronics with state-of-the-art hardware manufactured by Foxboro. This replacement includes 10 new flow transmitters (2 in each of the 5 recirc loops), and new electronics in the associated control room panels. Control room equipment being replaced includes the 2 transmitter power supplies, square root converters for each flow transmitter, 4 summers, and the APRMs (2 flow converters and 2 power supplies).

Each division of the new recirc flow electronics converts the differential pressure (dp) in the five recirc loop venturis into an equivalent flow using a square root function. Flow signals from Division 1 are used by the plant computer; signals from Division 2 are indicated on a control room panel.

The total flow in each division is calculated from the sum of the five division flows. The Division 1 total flow signal is provided to:

1) the flow recorder on Panel 3F,
2) the Division 2 flow converter (through an isolator) for comparison with the Division 2 total flow signal, and
3) the APRM 1, 2, 3 and 4 trip bias units.

The Division 2 total flow signal is provided to:

1) the total flow indicator on Panel 4F,
2) the Division 1 flow converter (through an isolator) for comparison with the Division 1 total flow signal, and
3) the APRM 5, 6, 7 and 8 trip bias units.

The electronics in each division provide the following trip functions:

1) <u>Upscale Trip</u> - This half scram function is designed to initiate on high flow. This trip also results in a rod block, illumination of the UPSCALE and INOP lights on the flow converter module, and actuation of the APRM FLO BIAS OFF NORMAL annunciator alarm. The following conditions initiate an Upscale Trip:

- Total flow ≥114% ± 1% rated flow,
- Loss of power (trip is initiated but status lights do not illuminate due to power loss).

2) Comparator Trip - This rod block function is designed to detect a mismatch between divisions. This trip also results in the illumination of the COMP and INOP lights on the flow converter module, and actuation of the APRM FLO BIAS OFF NORMAL annunciator alarm. The following conditions initiate a Comparator Trip:

- Flow mismatch between divisions ≥10% ± 1% rated flow,
- Loss of power (trip is initiated but status lights do not illuminate due to power loss).

3) Inop Trip - This scram function results in the same actions as the Upscale trip with the exception of the trip status indication. If power is lost, the Inop light is the only indicator that illuminates due to an Inop trip. An Inop Trip is initiated when the total flow voltage signal is below the 2.5v zero-flow level. This is an indication of a power supply or module failure.

The Upscale and Comparator Trips reset automatically when flow conditions return to normal. The status of the trips indicated at the flow converter module must be manually reset. The above trip functions can be tested using internal calibration signals.

Each division of the Foxboro electronics consists of two nests. Each nest contains an individual nest power supply. A power supply failure in the lower nest results in a fail safe Upscale and Comparator trip when the relays deenergize. A total flow voltage signal (2.5 - 12.5V) from the upper nest is monitored by an alarm module in the lower nest.

The enhanced system uses sensors that have the square root function incorporated into the flow transmitters, which reduces the number of modules required in the control room. The new transmitters also allow simplified calibration of the control room electronics.

Test blocks have been added to the Foxboro electronics where needed to facilitate surveillance and calibration using external test signals. Additionally, the total flow signal between the divisions is provided through isolators to ensure separation between the two RPS divisions.

3.0 EVALUATION

The staff's evaluation of the requested changes is discussed in this section. The requested changes consist of TS changes that address operability of the LPRM/APRM system, and the previously described upgrade of the recirculation flow electronics that will improve the reliability of the flow biased APRM trip function.

3.1 Technical Specification Changes

The possibility of power oscillations caused by thermal-hydraulic instabilities in BWRs and the consequences of such events are addressed in Generic Letter 86-02, "Long-Term Solutions to Thermal-Hydraulic Instabilities in Boiling Water Reactors," which requested licensees to examine each core reload and to impose operating limitations, as appropriate, to ensure compliance with General Design Criteria (GDC) 10 and 12. GDC 10 requires that the reactor core be designed with appropriate margin to assure that specified fuel design limits will not be exceeded during any condition of normal operation, including the effects of anticipated operational occurrences. GDC 12 requires assurance that power oscillations that can result in conditions that exceed specified acceptable fuel design limits are either not possible or can be reliably and readily detected and suppressed.

As a result of core-wide oscillations at a Boiling Water Reactor (BWR) in the U.S., the NRC staff questioned the adequacy of previous BWR core stability analyses and the ability of existing systems to detect and suppress large magnitude oscillations prior to violation of fuel design limits. More recent analyses performed by General Electric have demonstrated that for large magnitude oscillations, the potential exists for violation of the safety limit MCPR. In response to this concern, GPUN performed plant-specific analyses to assess the capability of the OCNGS APRM system to respond to power oscillations.

The licensee requested TS changes to improve the availability of the LPRM/APRM detectors to detect and suppress power oscillations prior to exceeding core fuel design limits. The changes provide a more stringent requirement on the availability of the LPRMs associated with the APRM system by restricting the allowable number of out-of-service LPRM/APRM monitors on the A and B levels. Additionally, the changes provide for a minimum operating limit critical power ratio to address core stability concerns.

Section 3.1.B.1 of the existing OCNGS TS defines the minimum number of APRM channel inputs required to permit accurate average core power monitoring. Specifications 3.1.B.2 and 3.1.C.1 further define the distribution of the operable chambers to provide monitoring of local power changes that might be caused by a single rod withdrawal. TS Section 3.10.C identifies requirements associated with the MCPR during steady state power operations.

The licensee determined that the APRM channel response is more sensitive to the availability of the A and B level LPRM detectors (bottom half of the core) than to the C and D level LPRM detectors (top half of the core). APRM channel response is significantly improved if it can be assumed that at least one channel (per RPS system) responding to regional oscillations has no more than one A level (or B level) detector out of service. The licensee's TS change request for TS Sections 3.1.B.1, 3.1.B.2 and 3.1.C.1 places additional and more restrictive operability requirements on the number of allowed out-of-service LPRM/APRM detectors on the A and B levels. The licensee states that this will ensure the availability of LPRM/APRM detectors in order to provide a sufficient response to global as well as regional oscillations to prevent violation of the MCPR limit.

The proposed change to TS Section 3.10.C places a lower bound on the MCPR for each cycle, as identified in the COLR. The lower bound limit is required to provide sufficient margin to ensure the MCPR limit is not exceeded during global and regional power oscillations. The new proposed limit shall be greater than or equal to 1.47.

The proposed TS changes requested by the licensee address the requirements of GDC 10 and 12, and the guidelines of GL 86-02 for detection and suppression of power oscillation and are, therefore, acceptable.

3.2 Enhanced Recirculation Flow Electronics

The enhanced APRM electronics are part of the safety-related Class 1E plant protection system. Therefore, the staff reviewed the system design against the applicable GDC and IEEE Standards for safety-related instrumentation and control systems as indicated in the Standard Review Plan (SRP), NUREG-0800.

To ensure that the enhanced APRM system will perform its intended function(s) under accident conditions, the staff reviewed the equipment for (1) independence, (2) environmental qualification, (3) seismic qualification and (4) maintenance and testing.

3.2.1 Independence

The staff reviewed the equipment design using the criteria of IEEE Standard 384-1981, "IEEE Standard Criteria for Independence of Class 1E Equipment and Circuits." The safety-related components receive power from dedicated Class 1E power supplies, and interface with non-Class 1E equipment through qualified isolation devices. The staff determined that the methods of isolating the Class 1E components from the non-Class 1E components are consistent with IEEE Standard 384-1981 and are, therefore, acceptable.

3.2.2 Environmental Qualification

The licensee used IEEE Standard 323-1974, "Standard for Qualifying Class 1E Equipment for Nuclear Power Generating Stations" to qualify the new APRM equipment to the same environmental limits specified for the equipment being replaced. The new equipment was designed for use in areas that have a mild environment. The equipment is located in an environmentally qualified zone that is considered a mild environment during normal operations. The equipment is not required to operate during conditions that produce a harsh environment.

The new electronics are qualified for 104°F (which includes an estimated 19°F temperature rise in the cabinet), 14.7 psia, 100% relative humidity, and $7.9X10^3$ Rad. These conditions bound the environmental conditions for the locations in which the equipment is to operate. Consequently, the staff finds that the environmental qualification of the equipment meets the intent of IEEE Standard 323-1974 and is, therefore, acceptable.

3.2.3 Seismic Qualification

The APRM modification replaces the existing flow transmitters and control room electronics with equipment that the licensee has committed to qualify as seismic Category 1. Supports and mounting of this equipment will be in accordance with the Seismic Qualification Utilities Group (SQUG) Generic Implementation Procedure (GIP) for seismic verification of nuclear plant equipment. The staff finds the seismic qualification to be acceptable.

3.2.4 Periodic Maintenance and Site Acceptance Testing

The control room electronics for the Foxboro system do not require changes in access or space allocations. Access to this equipment requires removal of a transparent cover plate that serves to protect the equipment. Routine preventative maintenance activities will be performed in accordance with the licensee's maintenance procedures, which have been prepared in accordance with the manufacturer's recommendations.

The new recirc flow monitoring system will undergo site acceptance testing to verify operability. This will include simulation of inputs and verification of outputs. The testing will include the following:

1) Verification of proper flow signals (individual and total loop flow) based on actual or simulated flow input signals,

2) verification of proper flow indication (flow indicators, recorder, and plant computer),

3) verification of proper trip functions at the appropriate setpoints and upon loss of power or downscale indication (Inop), and

4) proper operation of the flow biased rod block and scram functions within the APRMs.

The staff finds the scope of the periodic maintenance and site acceptance test activities to be in accordance with the guidelines of the Standard Review Plan and, therefore, acceptable.

4.0 CONCLUSION

Based on the above, the staff concludes that the design changes related to the upgrade of the existing LPRM/APRM equipment, and the associated changes to the TSs to address core power oscillations meet the requirements of GDC 10 and 12 for control of core power oscillations, and the criteria of IEEE 384, and 323 and Seismic Qualification Utility Group (SQUG) procedures for independence, environmental qualification and seismic qualification and are, therefore, acceptable.

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