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

GEORGIA POWER COMPANY

OGLETHORPE POWER CORPORATION

MUNICIPAL ELECTRIC AUTHORITY OF GEORGIA

CITY OF DALTON, GEORGIA

DOCKET NO. 50-321

EDWIN I. HATCH NUCLEAR PLANT, UNIT NO. 1

AMENDMENT TO FACILITY OPERATING LICENSE

Amendment No. 168
License No. DPR-57

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

   A. The application for amendment to the Edwin I. Hatch Nuclear Plant, Unit 1 (the facility) Facility Operating License No. DPR-57 filed by Georgia Power Company, acting for itself, Oglethorpe Power Corporation, Municipal Electric Authority of Georgia, and City of Dalton, Georgia (the licensee) dated June 22, 1989, as amended July 31, 1989, and October 4, 1989, 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 set forth in 10 CFR Chapter I;

   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-57 is hereby amended to read as follows:

   Technical Specifications

   The Technical Specifications contained in Appendices A and B, as revised through Amendment No. 168, are hereby incorporated in the license. The licensee shall operate the facility in accordance with the Technical Specifications.

3. This license amendment is effective as of its date of issuance and shall be implemented within 60 days of issuance.

FOR THE NUCLEAR REGULATORY COMMISSION

Darl Hood

for David B. Matthews, Director
Project Directorate II-3
Division of Reactor Projects - I/II
Office of Nuclear Reactor Regulation

Attachment:
Changes to the Technical Specifications

Date of Issuance: December 29, 1989

ATTACHMENT TO LICENSE AMENDMENT NO. 168

FACILITY OPERATING LICENSE NO. DPR-57

DOCKET NO. 50-321

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

Remove Pages	Insert Pages
iii	iii
x	x
xi	--
1.0-7	1.0-7
1.1-6	1.1-6
1.1-7	1.1-7
1.1-8	1.1-8
3.3-5	3.3-5
3.3-6	3.3-6
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3.6-22	3.6-22
3.11-1	3.11-1
3.11-1a	3.11-1a
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3.11-3	3.11-3
3.11-4	3.11-4
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Figure 3.11-1, Sheets 1 through 8	--
Figure 3.11-3	--
Figure 3.11-4	--
Figure 3.11-6	--
6-15d	6-15d

LIST OF FIGURES

1.1 FUEL CLADDING-INTEGRITY

A. Fuel Cladding Integrity Limit at Reactor Pressure > 800 psia and Core Flow > 10% of Rated

The fuel cladding integrity Safety Limit is set such that no fuel damage is calculated to occur if the limit is not violated. Since the parameters which result in fuel damage are not directly observable during reactor operation the thermal and hydraulic conditions resulting in a departure from nucleate boiling have been used to mark the beginning of the region where fuel damage could occur. Although it is recognized that a departure from nucleate boiling would not necessarily result in damage to BWR fuel rods, the critical power at which boiling transition is calculated to occur has been adopted as a convenient limit. However, the uncertainties in monitoring the core operating state and in the procedures used to calculate the critical power result in an uncertainty in the value of the critical power. Therefore the fuel cladding integrity safety limit is defined as the critical power ratio in the limiting fuel assembly for which more than 99.9% of the fuel rods in the core are expected to avoid boiling transition considering the power distribution within the core and all uncertainties.

The MCPR Safety Limit is determined using a model that combines all of the uncertainties in operating parameters and the procedures used to calculate critical power. The probability of the occurrence of boiling transition is determined using an NRC-approved critical power correlation. This MCPR Safety Limit is increased for single-loop operation over the comparable two-loop value (Reference 2). Details of the fuel cladding integrity Safety Limit calculation are presented in Reference 1.

MM. Minimum Critical Power Ratio (MCPR) - Minimum Critical Power Ratio (MCPR) is the value of the critical power ratio associated with the most limiting assembly in the reactor core. Critical Power Ratio (CPR) is the ratio of that power in a fuel assembly, which is calculated to cause some point in the assembly to experience boiling transition, to the actual assembly operation power.

NN. Trip System - A trip system means an arrangement of instrument channel trip signals and auxiliary equipment required to initiate action to accomplish a protective function. A trip system may require one or more instrument channel trip signals related to one or more plant parameters in order to initiate trip system action. Initiation of protective action may require the tripping of a single trip system or the coincident tripping of two trip systems.

OO. (Deleted)

PP. Core Operating Limits Report - The Core Operating Limits Report is the unit-specific document that provides core operating limits for the current operating reload cycle. These cycle-specific core operating limits shall be determined for each reload cycle in accordance with Specification 6.9.1.11. Plant operation within these operating limits is addressed in individual specifications.

QQ. Channel Calibration - A Channel Calibration is the adjustment, as necessary, of the channel output such that it responds with the necessary range and accuracy to known values of the parameter which the channel monitors. The Channel Calibration shall encompass the entire channel including the sensor and alarm and/or trip functions, and shall include the Channel Functional Test. The Channel Calibration may be performed by any series of sequential, overlapping or total channel steps such that the entire channel is calibrated.

RR. Channel Functional Test - A Channel Functional Test shall be:

a. Analog channels - the injection of a simulated signal into the channel as close to the primary sensor as practicable to verify operability including alarm and/or trip functions.

b. Bistable Channels - the injection of a simulated signal into the channel sensor to verify operability including alarm and/or trip functions.

SS. Fraction of Limiting Power Density (FLPD) - the ratio of the linear heat generation rate (LHGR) existing at a given location to the design LHGR for the bundle type.

TT. Core Maximum Fraction of Limiting Power Density (CMFLPD) - the CMFLPD is the highest value existing in the core of the FLPD.

1.1.B. Core Thermal Power Limit (Reactor Pressure < 800 psia)

At pressures below 800 psia, the core elevation pressure drop (0 power, 0 flow) is greater than 4.56 psi. At low powers and flows this pressure differential is maintained in the bypass region of the core. Since the pressure drop in the bypass region is essentially all elevation head, the core pressure drop at low powers and flows will always be greater than 4.56 psi. Analyses show that with a flow of $28x10^3$ lbs/hr bundle flow, bundle pressure drop is nearly independent of bundle power and has a value of 3.5 psi. Thus, the bundle flow with a 4.56 psi driving head will be greater than $28x10^3$ lbs/hr. Full scale ATLAS test data taken at pressures from 14.7 psia to 800 psia indicate that the fuel assembly critical power at this flow is approximately 3.35 MWt. With the design peaking factors this corresponds to a core thermal power of more than 50%. Thus, a core thermal power limit of 25% for reactor pressures below 800 psia is conservative.

C. Power Transient

Plant safety analyses have shown that the scrams caused by exceeding any safety system setting will assure that the Safety Limit of 1.1.A or 1.1.B will not be exceeded. Scram times are checked periodically to assure the insertion times are adequate. The thermal power transient resulting when a scram is accomplished other than by the expected scram signal (e.g., scram from neutron flux following closure of the main turbine stop valves) does not necessarily cause fuel damage. However, for this specification a Safety Limit violation will be assumed when a scram is only accomplished by means of a backup feature of the plant design. The concept of not approaching a Safety Limit provided scram signals are operable is supported by the extensive plant safety analysis.

D. Reactor Water Level (Hot or Cold Shutdown Condition)

For the fuel in the core during periods when the reactor is shutdown, consideration must be given to water level requirements due to the effect of decay heat. If the water level should drop below the top of the fuel during this time, the ability to remove decay heat is reduced. This reduction in cooling capability could lead to elevated cladding temperatures and clad perforation in the event that the water level became less than two-thirds of the core height. The Safety Limit has been established at 378 inches above vessel invert to provide a point which can be monitored and also provide adequate margin.

E. References

1. "General Electric Standard Application for Reactor Fuel (Supplement for United States)," NEDE-24011-P-A.

2. "Edwin I. Hatch Nuclear Plant Units 1 and 2 Single-Loop Operation," NEDO-24205, August 1979.

3.3.F. Operation with a Limiting Control Rod Pattern (for Rod Withdrawal Error, RWE)

A Limiting Rod Pattern for RWE exists when the MCPR is less than the value provided in the Core Operating Limits Report.

During operation with a Limiting Control Rod Pattern for RWE and when core thermal power is ≥ 30%, either:

1. Both rod block monitor (RBM) channels shall be operable, or
2. If only one RBM channel is operable, control rod withdrawal shall be blocked within 24 hours, or
3. If neither RBM channel is operable, control rod withdrawal shall be blocked.

G. Limiting the Worth of a Control Rod Below 10% Rated Thermal Power

1. Rod Worth Minimizer (RWM)

Whenever the reactor is in the Start & Hot Standby or Run Mode below 10% rated thermal power, the RWM shall be operable or a second licensed operator shall verify that the operator at the reactor console is following the control rod program.

4.3.F. Operation with a Limiting Control Rod Pattern (for Rod Withdrawal Error, RWE)

During operation when a Limiting Control Rod Pattern for RWE exists and only one RBM channel is operable, an instrument functional test of the RBM shall be performed prior to withdrawal of the control rod(s). A Limiting Rod Pattern for RWE is defined by Specification 3.3.F.

G. Limiting the Worth of a Control Rod Below 10% Rated Thermal Power

1. Rod Worth Minimizer (RWM)

Prior to the start of control rod withdrawal at startup, and as soon as automatic initiation of the RWM occurs during rod insertion while shutting down, the capability of the RWM to properly fulfill its function shall be verified by the following checks.

a. The correctness of the Banked Position Withdrawal Sequence input to the RWM computer shall be verified.
b. The RWM computer on line diagnostic test shall be successfully performed.
c. Proper annunciation of the selection error of at least one out-of-sequence control rod in each fully inserted group shall be verified.
d. The rod block function of the RWM shall be verified by withdrawing or inserting an out-of-sequence control rod no more than to the block point.

LIMITING CONDITIONS FOR OPERATION

3.3.G.2. Rod Sequence Control System (RSCS)

a. Operability

When the reactor is in the Start and Hot Standby or Run Mode below 10% rated thermal power and control rod movement is within the group notch mode after 50% of the control rods have been withdrawn, the Rod Sequence Control System shall be operable except when performing the RWM surveillance tests.

b. Failed Position Switch

Control rods with a failed "Full-in" or "Full-out" position switch may be bypassed in the Rod Sequence Control System if the actual rod position is known. These rods shall be moved in sequence to their correct positions (full in on insertion or full out on withdrawal).

SURVEILLANCE REQUIREMENTS

2. Rod Sequence Control System (RSCS)

a. Operability

As soon as the group notch mode is entered during each reactor startup and as soon as automatic initiation of the RSCS occurs during rod insertion while shutting down, the capability of the Rod Sequence Control System to properly fulfill its function shall be verified by attempting to select and move an inhibited control rod.

When the control rod movement is within the group notch mode and as soon as automatic initiation of the RSCS occurs during rod insertion while shutting down, the operability of the notching restriction shall be demonstrated by attempting to move a control rod more than one notch in the first programmed rod group.

b. Failed Position Switch

A second licensed operator shall verify the conformance to Specification 3.3.G.2.b before a rod may be bypassed in the Rod Sequence Control System.

3.3.G.2.c. Shutdown Margin/Scram Time Testing

In order to perform the required shutdown margin demonstrations subsequent to any fuel loading operations, or to perform control rod drive scram and/or friction testing as specified in Surveillance Requirement 4.3.C.2 and the initial startup test program, the relaxation of the following RSCS restraints is permitted. The sequence restraints imposed on control rod groups A_{12}, A_{34}, B_{12}, or B_{34} after 50% of the control rods have been withdrawn may be removed for the test period by means of the individual rod position bypass switches.

4.3.G.2.c. Shutdown Margin/Scram Time Testing

Prior to control rod withdrawal for startup, verify the conformance to Specification 3.3.G.2.b. before a rod may be bypassed in the RSCS. The requirements to allow use of the individual rod position bypass switches within rod groups A_{12}, A_{34}, B_{12}, or B_{34} of the RSCS during shutdown margin, scram time or friction testing are:

(1) RWM operable as per Specification 3.3.G.1.

(2) After the bypassing of the rods in the RSCS groups A_{12}, A_{34}, B_{12}, or B_{34} for test purposes, it shall be demonstrated that movement of the rods in the 50% density to 10% of rated thermal power range is blocked or limited to the single notch mode of withdrawal.

(3) A second licensed operator shall verify the conformance to procedures and this Specification.

H. Shutdown Requirements

If Specifications 3.3.A through 3.3.G are not met, an orderly shutdown shall be initiated and the reactor placed in the Cold Shutdown Condition within 24 hours.

B. Control Rods

Limiting Conditions for Operation:

Specification 3.3.B.1 requires that a rod which cannot be moved with drive pressure be taken out of service by being disarmed electrically. To disarm the drive electrically, four amphenol type plug connectors are removed from the drive insert and withdrawal solenoids rendering the rod incapable of withdrawal. This procedure is equivalent to valving out the drive and is preferred because, in this condition, drive water cools and minimizes crud accumulation in the drive. Electrical disarming does not eliminate position indication. If the rod is fully inserted and disarmed electrically, it is in a safe position of maximum contribution to shutdown reactivity. If it is disarmed electrically in a non-fully inserted position, that position shall be consistent with the shutdown reactivity limitation stated in Specification 3.3.A. This assures that the core can be shutdown at all times with the remaining control rods, assuming the highest worth operable control rod does not insert. An allowable pattern for control rods disarmed electrically, which shall meet this Specification, will be determined and made available to the operator. Also if damage within the control rod drive mechanism and in particular, cracks in drive internal housing, cannot be ruled out, then a generic problem affecting a number of drives cannot be ruled out. Circumferential cracks resulting from stress assisted intergranular corrosion have occurred in the collet housing of drives at several BWRs. This type of cracking could occur in a number of drives and if the cracks propagated until severance of the collet housing occurred, scram could be prevented in the affected rods. Limiting the period of operation with a potentially severed collet housing will assure that the reactor will not be operated with a large number of rods with failed collet housings.

Surveillance Requirements:

The weekly control rod exercise test serves as a periodic check against deterioration of the control rod system and also verifies the ability of the control rod drive to scram, since, if a rod can be moved with drive pressure, it will scram because of higher pressure applied during scram. The frequency of exercising the control rods under the conditions of three or more inoperable rods provides even further assurance of the reliability of the remaining control rods.

3.3.F. Operation with a Limiting Control Rod Pattern (for Rod Withdrawal Error, RWE)

Surveillance Requirements:

A limiting control rod pattern for RWE is a pattern which, due to unrestricted withdrawal of any single control rod, could result in violation of the MCPR Safety Limit. Specification 3.3.F. defines a limiting control rod pattern for RWE. During use of such patterns when both RBM channels are not operable, it is judged that testing of the RBM system prior to withdrawal of control rods to assure its operability will assure that improper withdrawal does not occur. Reference NEDC-30474-P (Ref. 17) for more information.

G. Limiting the Worth of a Control Rod Below 10% Rated Thermal Power

1. Rod Worth Minimizer (RWM)

Limiting Conditions for Operation:

The RWM and the Rod Sequence Control System (RSCS) restrict withdrawals and insertions of control rods to prespecified sequences. All patterns associated with these sequences have the characteristics that, assuming the worst single deviation from the sequence, the drop of any control rod from the fully inserted position to the position of the control rod drive would not cause the reactor to sustain a power excursion resulting in any pellet average enthalpy in excess of 280 calories per gram. An enthalpy of 280 calories per gram is well below the level at which rapid fuel dispersal could occur (i.e., 425 calories per gram). Primary system damage in this accident is not possible unless a significant amount of fuel is rapidly dispersed. Reference Sections 3.6.5.4, 3.6.6, 7.14.5.3, and 14.4.2, and Appendix P of the FSAR, and NEDO-24040.

3.3.G.1. Rod Worth Minimizer (RWM) (Continued)

In performing the function described above, the RWM and RSCS are not required to impose any restrictions at core power levels in excess of 10% of rated. Material in the cited references shows that it is impossible to reach 280 calories per gram in the event of a control rod drop occurring at power greater than 10%, regardless of the rod pattern. This is true for all normal and abnormal patterns including those which maximize the individual control rod worth.

At power levels below 10% of rated, abnormal control rod patterns could produce rod worths high enough to be of concern relative to the 280 calorie per gram rod drop limit. In this range of RWM and the RSCS constrain the control rod sequences and patterns to those which involve only acceptable rod worths.

The RWM and the RSCS provide automatic supervision to assure that out of sequence control rods will not be withdrawn or inserted; i.e., it limits operator deviations from planned withdrawal sequences. They serve as a backup to procedural control of control rod sequences, which limit the maximum reactivity worth of control rods. In the event that the RWM is out of service, when required, a second licensed operator or other qualified technical plant employee whose qualifications have been reviewed by the AEC can manually fulfill the control rod pattern conformance functions of this system.

The functions of the RWM and RSCS make it unnecessary to specify a license limit on rod worth to preclude unacceptable consequences in the event of a control rod drop. At low powers, below 10%, these devices force adherence to acceptable rod patterns. Above 10% of rated power, the consequences of a rod drop event without RWM or RSCS are acceptable. Power level for automatic cutout of the RSCS function is sensed by first stage turbine pressure. Power level for automatic cutout of the RWM function is sensed by feedwater and steam flow and is set to be consistent with the RSCS setting.

Surveillance Requirements:

Functional testing of the RWM prior to the start of control rod withdrawal at startup, and prior to attaining 10% of rated thermal power during rod insertion while shutting down, will ensure reliable operation and minimize the probability of the rod drop accident.

2. Rod Sequence Control System (RSCS)

a. Operability

Limiting Conditions for Operation:

See bases for Technical Specification 3.3.G.1. Rod Worth Minimizer.

3.3.G.2.c. Shutdown Margin/Scram Time Testing (Continued)

tion simulation switches provided in the RSCS for such purposes. During the scram time testing, reactor conditions will be such that the reactor rod pattern will be in RSCS B group. All A_{12} and A_{34} rods will be fully withdrawn, alternatively the rod pattern will be in RSCS group A and all B_{12} and B_{34} rods will be fully withdrawn. To test A_{34} rods, it will be necessary to simulate all withdrawn B rods as being at the full-in position, and for testing A_{12} rods, all A_{34} and all withdrawn B rods as being at the full-in position. The simulation of already withdrawn control rods in the 100% to 50% rod density range (A_{12} and A_{34} or alternatively B_{12} and B_{34}) as being full-in to perform the individual rod test does not violate the intent of the RSCS since; (a) the single notch mode of rod withdrawal for rods in the 50% density to 10% of rated thermal power range will remain in effect until that power level has been achieved and the test procedure will require that this be verified; (b) no B group rods can be selected either for withdrawal or insertion during the time that an A_{12} or A_{34} rod is fully inserted or is simulated as being in the fully inserted position (similarly for the A group rods when the B sequence is chosen for startup and (c) all rod position simulation switch operations will be verified by a second independent check.

H. Shutdown Requirements

Should circumstances be such that the Limiting Conditions for Operation as stated in Specifications 3.3.A. through 3.3.G. cannot be met, an orderly shutdown shall be initiated and the reactor placed in the Cold Shutdown Condition within 24 hours.

I. Scram Discharge Volume Vent and Drain Valves

The scram discharge volume vent and drain valves are required to be OPERABLE so that the scram discharge volume will be available when needed to accept discharge water from the control rods during a reactor scram and will isolate the reactor coolant system from the containment when required.

J. References

1. FSAR Section 3.4, Reactivity Control Mechanical Design
2. FSAR Section 3.5.2, Safety Design Bases
3. FSAR Section 3.5.4, Safety Evaluation
4. FSAR Section 3.5, Control Rod Drive Housing Supports

3.6.I. Jet Pumps (Continued)

A nozzle-riser system failure could also generate the coincident failure of a jet pump body; however, the converse is not true. The lack of any substantial stress in the jet pump body makes failure impossible without an initial nozzle riser system failure.

One of the acceptable procedures for jet pump surveillance, identified in NUREG/CR-3025, Reference 2, was adopted for Hatch Unit 1. The surveillance is performed to verify that neither of the following conditions occur:

(a) The Recirculation Pump Flow/Speed Ratio deviates by more than 5% from the normal range, or

(b) The Jet Pump Loop Flow/Speed Ratio deviates by more than 5% from the normal range.

If either criterion is failed, then the procedure calls for comparing either the individual jet pump flow or individual jet pump diffuser to lower plenum differential pressures to the criteria of the Limiting Conditions for Operation (LCO). If the LCO criteria are not satisfied and pump speed is less than 60% of rated, it may be necessary to increase pump speed to above 60% of rated and repeat the measurements before declaring a jet pump inoperable. In this case, it is recommended that close monitoring and increased recirculation pump speed should be performed only if the criteria are exceeded by an amount to be determined from previous plant operating experience.

3.6.J. Recirculation System

Operation with a single reactor coolant system recirculation pump is allowed, provided that adjustments to the flow referenced scram and APRM rod block setpoints, MCPR cladding integrity Safety Limit, MCPR Operating Limit, and MAPLHGR limit are made. An evaluation of the performance of the ECCS with single-loop operation has been performed and determined to be acceptable, Reference 4. Based on this Reference, a factor is applied to reduce the APLHGR limits during single-loop operation. To account for increased uncertainties in the total core flow and TIP readings when operating with a single recirculation loop, an increase is applied to the MCPR cladding integrity Safety Limit and MCPR Operating Limit over the comparable two-loop values. The flow referenced simulated thermal power scram and rod block setpoints for single-recirculation-loop operation is reduced by the amount of mΔW, where m is the flow reference slope for the rod block monitor and ΔW is the largest difference between two-loop and single-loop effective drive flow when the active loop indicated flow is the same. This adjustment is necessary to preserve the original relationship between the rod block and actual effective drive flow.

When restarting an idle pump, the discharge valve of the idle loop is required to remain closed until the speed of the faster pump is below 50% of its rated speed to provide assurance that when going from one- to two-loop operations, excessive vibration of the jet pump risers will not occur.

The possibility of experiencing limit cycle oscillations during single-loop operation is precluded by restricting the core flow to greater than or equal to 45% of rated core flow when core power is greater than the 80% rod line. This requirement is based on General Electric's recommendations contained in SIL 380, Revision 1, which defines the region where the limit cycle oscillations are more likely to occur.

3.11. **FUEL RODS**

Applicability

The Limiting Conditions for Operation associated with the fuel rods apply to those parameters which monitor the fuel rod operating conditions.

Objective

The Objective of the Limiting Conditions for Operation is to assure the performance of the fuel rods.

Specifications

A. Average Planar Linear Heat Generation Rate (APLHGR)

During power operation, the APLHGR for all core locations shall not exceed the appropriate APLHGR limit provided in the Core Operating Limits Report. If at any time during operation it is determined by normal surveillance that the limiting value for APLHGR is being exceeded, action shall be initiated within 15 minutes to restore operation to within the prescribed limits. If the ALPHGR is not returned to within the prescribed limits within two (2) hours, then reduce reactor power to less than 25% of rated thermal power within the next four (4) hours. If the limiting condition for operation is restored prior to expiration of the specified time interval, then further progression to less than 25% of rated thermal power is not required.

4.11. **FUEL RODS**

Applicability

The Surveillance Requirements apply to the parameters which monitor the fuel rod operating conditions.

Objective

The Objective of the Surveillance Requirements is to specify the type and frequency of surveillance to be applied to the fuel rods.

Specifications

A. Average Planar Linear Heat Generation Rate (APLHGR)

The APLHGR for each type of fuel as a function of average planar exposure shall be determined daily during reactor operation at $\geq$ 25% rated thermal power.

B. Linear Heat Generation Rate (LHGR)

The LHGR as function of core height shall be checked daily during reactor operation at $\geq$ 25% rated thermal power.

LIMITING CONDITIONS FOR OPERATION		SURVEILLANCE REQUIREMENTS	
3.11.B.	<u>Linear Heat Generation Rate (LHGR)</u>	4.11.B.	<u>Linear Heat Generation Rate (LHGR)</u>
	During power operation, the LHGR shall not exceed the limiting value provided in the Core Operating Limits Report. If at any time during		The LHGR shall be checked daily during reactor operation at ≥ 25% rated thermal power.

3.11.B. Linear Heat Generation Rate (LHGR) (Continued)

operation it is determined by normal surveillance that the limiting value for LHGR is being exceeded, action shall be initiated within 15 minutes to restore operation to within the prescribed limits. If the LHGR is not returned to within the prescribed limits within 2 hours, then reduce reactor power to less than 25% of rated thermal power within the next 4 hours. If the limiting condition for operation is restored prior to expiration of the specified time interval, then further progression to less than 25% of rated thermal power is not required.

C. Minimum Critical Power Ratio (MCPR)

The minimum critical power ratio (MCPR) shall be equal to or greater than the operating limit MCPR (OLMCPR) provided in the CORE OPERATING LIMITS REPORT.

If at any time during operation it is determined by normal surveillance that the limiting value for MCPR is being exceeded, action shall be initiated within 15 minutes 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, then reduce reactor power to less than 25% of rated thermal power within the next four(4) hours. If the Limiting Condition for Operation is restored prior to expiration of the specified time interval, then further progression to less than 25% of rated thermal power is not required.

4.11.C.1. Minimum Critical Power Ratio (MCPR)

MCPR shall be determined to be equal to or greater than the applicable limit, daily during reactor power operation at ≥ 25% rated thermal power and following any change in power level or distribution that would cause operation with a limiting control rod pattern as described in the bases for Specification 3.3.F.

4.11.C.2. Minimum Critical Power Ratio Limit

The MCPR limit at rated flow and rated power shall be determined, as provided in the CORE OPERATING LIMITS REPORT, using:

a. τ=1.0 prior to initial scram time measurements for the cycle, performed in accordance with Specification 4.3.C.2.a.

or

b. τ is determined from scram time measurements performed in accordance with Specification 4.3.C.2.

The determination of the limit must be completed within 72 hours of the conclusion of each scram time surveillance test required by Specification 4.3.C.2.

3.11. FUEL RODS

A. Average Planar Linear Heat Generation Rate (APLHGR)

This specification assures that the peak cladding temperature following the postulated design basis loss-of-coolant accident (LOCA) will not exceed the limit specified in 10 CFR 50.46 even considering the postulated effects of fuel pellet densification.

The peak cladding temperature following a postulated loss-of-coolant accident is primarily a function of the average heat generation rate of all the rods of a fuel assembly at any axial location and is only dependent secondarily on the rod to rod power distribution within an assembly. Since expected local 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 conform to 10 CFR 50.46. The limiting value for APLHGR is provided in the Core Operating Limits Report.

The calculational procedure used to establish the APLHGR limits is based on a LOCA analysis. The analysis was performed using General Electric (GE) calculational models which are consistent with the requirements of Appendix K to 10 CFR 50. The LOCA analysis was performed utilizing the new improved calculational model, SAFER/GESTR-LOCA. The analysis demonstrated that loss-of-coolant concerns do not limit the operation of the fuel since margin to the 2200°F limit was demonstrated (Reference 9). Therefore, the APLHGR limits are derived to assure that the fuel thermal-mechanical design criteria are met.

A flow dependent correction factor is applied to the rated conditions APLHGR to assure that the 2200°F PCT limit is complied with during LOCA initiated from less than rated core flow. In addition, other power and flow dependent corrections are applied to the rated conditions APLHGR limits to assure that the fuel thermal-mechanical design criteria are met during abnormal transients initiated from off-rated conditions for two-loop and single-loop operations, References 2 and 8. For single-loop operation, a multiplicative factor is applied to the rated conditions APLHGR limit for all fuel bundles when core power exceeds a specified value. The power and flow-dependent correction factors, and the limiting values for APLHGR for each fuel type used in a particular cycle are specified in the Core Operating Limits Report.

3.11.B. Linear Heat Generation Rate (LHGR)

This specification assures that the LHGR in any rod is less than the design linear heat generation if fuel pellet densification is postulated. For LHGR to be a limiting value below 25-percent rated thermal power, the ratio of peak LHGR to core average LHGR would have to be greater than 9.6, which is precluded by a considerable margin when employing any permissible control rod pattern.

C. Minimum Critical Power Ratio (MCPR)

The required operating limit MCPR as specified in Specification 3.11.C is derived from the established fuel cladding integrity Safety Limit MCPR and an analysis of abnormal operational transients presented in Reference 1.

Various transient events will reduce the MCPR below the operating MCPR. To assure that the fuel cladding integrity Safety Limit is not violated during anticipated abnormal operational transients, the most limiting transients have been analyzed to determine which one results in the largest reduction in critical power ratio (Δ MCPR). Addition of the largest Δ MCPR to the Safety Limit MCPR gives the minimum operating limit MCPR to avoid violation of the Safety Limit should the most limiting transient occur. The type of transients evaluated were loss of flow, increase in pressure and power, positive reactivity insertion, and coolant temperature decrease.

Details of how MCPR evaluations are performed, the methods used, and how the MCPR limit is adjusted for operation at less than rated power and flow conditions used for single-loop operation are given in Reference 1 and in the Core Operating Limits Report.

3.11.C. **Minimum Critical Power Ratio (MCPR)** (Continued)

According to the CORE OPERATING LIMITS REPORT the 100% power, 100% flow operating limit MCPR (OLMCPR) depends on the average scram time, τ, of the control rods, where:

$$t = 0 \text{ or } \frac{\tau_{ave} - \tau_B}{\tau_A - \tau_B}\text{, whichever is greater}$$

where: τ_A = 1.096 sec (Specification 3.3.C.2.a, scram time limit to notch 36)

$$\tau_B = \mu + 1.65 \left[\frac{N_1}{\sum_{i=1}^{n} N_i}\right]^{1/2} \sigma \quad \text{[Reference 7]}$$

where: μ = 0.822 sec (mean scram time used in the transient analysis)

σ = .018 sec (standard deviation of μ)

$$\tau_{ave} = \left[\frac{\sum_{i=1}^{n} N_i \tau_i}{\sum_{i=1}^{n} N_i}\right]$$

where: n = number of surveillance tests performed to date in the cycle

N_i = number of active control rods measured in the ith surveillance test

τ_i = average scram time to notch 36 of all rods in the ith surveillance test

N_1 = total number of active rods measured in 4.3.C.2.a

e. Type of container, e.g., LSA, type A, type B, large quantity.

f. Solidification agent, e.g., cement.

The Radioactive Effluent Release Report shall include (on a quarterly basis) unplanned releases from the site to unrestricted areas of radioactive materials in gaseous and liquid effluents that were in excess of 1 Ci, excluding dissolved and entrained gases and tritium for liquid effluents, or those in excess of 150 Ci of noble gases or 0.02 Ci of radioiodines for gaseous releases.

The Radioactive Effluent Release Report shall include any changes to the PROCESS CONTROL PROGRAM and to the OFFSITE DOSE CALCULATION MANUAL made during the reporting period.

MONTHLY OPERATING REPORT

6.9.1.10. Routine reports of operating statistics and shutdown experience shall be submitted on a monthly basis to the Director, Office of Management and Program Analysis, U. S. Nuclear Regulatory Commission, Washington, D. C. 20555, with a copy to the Regional Office of Inspection and Enforcement no later than the 15th of each month following the calendar month covered by the report.

CORE OPERATING LIMITS REPORT

6.9.1.11.a. Core operating limits shall be established and documented in the CORE OPERATING LIMITS REPORT before each reload cycle or any remaining part of a reload cycle for the following:

(1) Operation with a Limiting Control Rod Pattern (for Rod Withdrawal Error, RWE) for Specification 3.3.F,

(2) The Average Planar Linear Heat Generation Rate (APLHGR) for Specification 3.11.A,

(3) The Linear Heat Generation Rate (LHGR) for Specification 3.11.B, and

(4) The Minimum Critical Power Ratio (MCPR) for Specifications 3.3.F and 3.11.C and Surveillance Requirement 4.11.C.

b. The analytical methods used to determine the core operating limits shall be those previously reviewed and approved by the NRC in the following documents.

(1) NEDE-24011-P-A, "General Electric Standard Application for Reactor Fuel," (applicable amendment specified in the CORE OPERATING LIMITS REPORT).

(2) "Safety Evaluation by the Office of Nuclear Reactor Regulation Supporting Amendment No. 157 to Facility Operating License DPR-57," dated September 12, 1988.

c. The core operating limits shall be determined so that all applicable limits (e.g., fuel thermal-mechanical limits, core thermal-hydraulic limits, ECCS limits, nuclear limits such as shutdown margin, and transient and accident analysis limits) of the safety analysis are met.

d. The CORE OPERATING LIMITS REPORT, including any mid-cycle revisions or supplements thereto, shall be provided upon issuance, for each reload cycle, to the NRC Document Control Desk with copies to the Regional Administrator and Resident Inspector.

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

GEORGIA POWER COMPANY

OGLETHORPE POWER CORPORATION

MUNICIPAL ELECTRIC AUTHORITY OF GEORGIA

CITY OF DALTON, GEORGIA

DOCKET NO. 50-366

EDWIN I. HATCH NUCLEAR PLANT, UNIT NO. 2

AMENDMENT TO FACILITY OPERATING LICENSE

Amendment No. 106
License No. NPF-5

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

   A. The application for amendment to the Edwin I. Hatch Nuclear Plant, Unit 2 (the facility) Facility Operating License No. NPF-5 filed by Georgia Power Company, acting for itself, Oglethorpe Power Corporation, Municipal Electric Authority of Georgia, and City of Dalton, Georgia (the licensee) dated June 22, 1989, as amended July 31, 1989, and October 4, 1989, 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 set forth in 10 CFR Chapter I;

   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. NPF-5 is hereby amended to read as follows:

   Technical Specifications

   The Technical Specifications contained in Appendices A and B, as revised through Amendment No. 106, are hereby incorporated in the license. The licensee shall operate the facility in accordance with the Technical Specifications.

3. This license amendment is effective as of its date of issuance and shall be implemented within 60 days of issuance.

FOR THE NUCLEAR REGULATORY COMMISSION

Darl Hood

for David B. Matthews, Director
Project Directorate II-3
Division of Reactor Projects - I/II
Office of Nuclear Reactor Regulation

Attachment:
Changes to the Technical Specifications

Date of Issuance: December 29, 1989

ATTACHMENT TO LICENSE AMENDMENT NO. 106

FACILITY OPERATING LICENSE NO. NPF-5

DOCKET NO. 50-366

Replace the following pages of the Appendix A Technical Specifications with the enclosed pages. The revised pages are identified by amendment number and contain vertical lines indicating the areas of change. Corresponding overleaf pages are provided to maintain document completeness.

Remove Pages	Insert Pages
I	I
XVI	XVI
1-2	1-2
B2-1	B2-1
B2-2	B2-2
B2-3	B2-3
B2-4	B2-4
B2-5	B2-5
B2-13	B2-13
3/4 1-4	3/4 1-4
3/4 1-14	3/4 1-14
3/4 1-15	3/4 1-15
3/4 1-16	3/4 1-16
3/4 1-17	3/4 1-17
3/4 2-1	3/4 2-1
3/4 2-2	3/4 2-2
3/4 2-3	3/4 2-3
3/4 2-4	3/4 2-4
3/4 2-4a	---
3/4 2-4b	---
3/4 2-4c	---
3/4 2-4d	---
3/4 2-4e - 3/4 2-4i	---
3/4 2-4j	---
3/4 2-4k	---
3/4 2-6	3/4 2-6
3/4 2-7	3/4 2-7
3/4 2-7a	3/4 2-7a
3/4 2-7b	3/4 2-7b - 3/4 2-7d
3/4 2-7c	---
3/4 2-7d	---
3/4 2-8	3/4 2-8
3/4 3-39	3/4 3-39
3/4 3-42	3/4 3-42
3/4 10-2	3/4 10-2
B 3/4 1-3	B 3/4 1-3
B 3/4 2-1	B 3/4 2-1
B 3/4 2-2	B 3/4 2-2
B 3/4 2-3	B 3/4 2-3
B 3/4 2-4	B 3/4 2-4
B 3/4 2-6	B 3/4 2-6
B 3/4 4-1	B 3/4 4-1
6-14d	6-14d

INDEX

DEFINITIONS

SECTION

1.0 DEFINITIONS

SECTION

1.0 DEFINITIONS (Continued)

1.0 DEFINITIONS

The following terms are defined so that uniform interpretation of these specifications may be achieved. The defined terms appear in capitalized type and shall be applicable throughout these Technical Specifications.

ACTION

ACTIONS shall be those additional requirements specified as corollary statements to each specification and shall be part of the specifications.

AVERAGE PLANAR EXPOSURE

The AVERAGE PLANAR EXPOSURE shall be applicable to a specific planar height and is equal to the sum of the exposure of all the fuel rods in the specified bundle at the specified height divided by the number of fuel rods in the fuel bundle.

AVERAGE PLANAR LINEAR HEAT GENERATION RATE

The AVERAGE PLANAR LINEAR HEAT GENERATION RATE (APLHGR) shall be applicable to a specific planar height and is equal to the sum of the LINEAR HEAT GENERATION RATES for all the fuel rods in the specified bundle at the specified height divided by the number of fuel rods in the fuel bundle.

CHANNEL CALIBRATION

A CHANNEL CALIBRATION shall be the adjustment, as necessary, of the channel output such that it responds with the necessary range and accuracy to known values of the parameter which the channel monitors. The CHANNEL CALIBRATION shall encompass the entire channel including the sensor and alarm and/or trip functions, and shall include the CHANNEL FUNCTIONAL TEST. The CHANNEL CALIBRATION may be performed by any series of sequential, overlapping or total channel steps such that the entire channel is calibrated.

CHANNEL CHECK

A CHANNEL CHECK shall be the qualitative assessment of channel behavior during operation by observation. This determination shall include, where possible, comparison of the channel indication and/or status with other indications and/or status derived from independent instrument channels measuring the same parameter.

ADMINISTRATIVE CONTROLS

CHANNEL FUNCTIONAL TEST

A CHANNEL FUNCTIONAL TEST shall be:

a. Analog channels - the injection of a simulated signal into the channel as close to the primary sensor as practicable to verify OPERABILITY including alarm and/or trip functions and channel failure trips.

b. Bistable channels - the injection of a simulated signal into the channel sensor to verify OPERABILITY including alarm and/or trip functions.

CORE ALTERATION

CORE ALTERATION shall be the addition, removal, relocation or movement of fuel, sources, incore instruments or reactivity controls within the reactor pressure vessel with the vessel head removed and fuel in the vessel. Suspension of CORE ALTERATIONS shall not preclude completion of the movement of a component to a safe conservative position.

CORE MAXIMUM FRACTION OF LIMITING POWER DENSITY

The CORE MAXIMUM FRACTION OF LIMITING POWER DENSITY (CMFLPD) shall be the largest FLPD which exists in the core for a given operating condition.

CORE OPERATING LIMITS REPORT

The CORE OPERATING LIMITS REPORT shall be the unit-specific document that provides core operating limits for the current operating reload cycle. These cycle-specific core operating limits shall be determined for each reload cycle in accordance with Specification 6.9.1.11. Plant operation within these operating limits is addressed in individual specifications.

CRITICAL POWER RATIO

The CRITICAL POWER RATIO (CPR) shall be the ratio of that power in the assembly which is calculated by application of an NRC-approved critical power correlation to cause some point in the assembly to experience boiling transition, divided by the actual assembly operating power.

$\bar{E}$-AVERAGE DISINTEGRATION ENERGY

$\bar{E}$ shall be the average, weighted in proportion to the concentration of each radionuclide in the reactor coolant at the time of sampling, of the sum of the average beta and gamma energies per disintegration, in MeV, for isotopes with half lives greater than 15 minutes making up at least 95% of the total non-iodine activity in the coolant.

2.0 The fuel cladding, reactor pressure vessel and primary system piping are the principal barriers to the release of radioactive materials to the environs. Safety Limits are established to protect the integrity of these barriers during normal plant operations and anticipated transients. The fuel cladding integrity Safety Limit is set such that no fuel damage is calculated to occur if the limit is not violated. Because fuel damage is not directly observable, a step-back approach is used to establish a Safety Limit such that the MCPR is not less than 1.04 for two-loop operation and 1.05 for single-loop operation. These limits represent a conservative margin relative to the conditions required to maintain fuel cladding integrity. The fuel cladding is one of the physical barriers which separate the radioactive materials from the environs. The integrity of this cladding barrier is related to its relative freedom from perforations or cracking. Although some corrosion or use related cracking may occur during the life of the cladding, fission product migration from this source is incrementally cumulative and continuously measurable. Fuel cladding perforations, however, can result from thermal stresses which occur from reactor operation significantly above design conditions and the Limiting Safety System Settings. While fission product migration from cladding perforation is just as measurable as that from use related cracking, the thermally caused cladding perforations signal a threshold beyond which still greater thermal stresses may cause gross rather than incremental cladding deterioration. Therefore, the fuel cladding Safety Limit is defined with a margin to the conditions which would produce onset of transition boiling, MCPR of 1.0. These conditions represent a significant departure from the condition intended by design for planned operation.

The evaluations which justify normal operation, abnormal transient, and accident analyses for two-loop operation are discussed in detail in Reference 1. Evaluation for single-loop operation demonstrates that two-loop transient and accident analyses are more limiting than single-loop, Reference 2.

2.1.1 THERMAL POWER (Low Pressure or Low Flow)

The use of the NRC-approved transition boiling correlation is not valid for all critical power calculations at pressures below 785 psig or core flows less than 10% of rated flow. Therefore, the fuel cladding integrity Safety Limit is established by other means. This is done by establishing a limiting condition on core THERMAL POWER with the following basis. Since the pressure drop in the bypass region is essentially all elevation head, the core pressure drop at low power and flows will always be greater than 4.5 psi. Analyses show that with a bundle flow of 28×10^3 lbs/hr, bundle pressure drop is nearly independent of bundle power and has a value of 3.5 psi. Thus, the bundle flow with a 4.5 psi driving head will be greater than 28×10^3 lbs/hr. Full scale ATLAS test data taken at pressures from 14.7 psia to 800 psia indicate that the fuel assembly critical power at this flow is approximately 3.35 Mwt. With the design peaking factors, this corresponds to a THERMAL POWER of more than 50% of RATED THERMAL POWER. Thus, a THERMAL POWER limit of 25% of RATED THERMAL POWER for reactor pressure below 785 psig is conservative.

2.1.2 THERMAL POWER (High Pressure and High Flow)

The fuel cladding integrity Safety Limit is set such that no fuel damage is calculated to occur if the limit is not violated. Since the parameters which result in fuel damage are not directly observable during reactor operation, the thermal and hydraulic conditions resulting in a departure from nucleate boiling have been used to mark the beginning of the region where fuel damage could occur. Although it is recognized that a departure from nucleate boiling would not necessarily result in damage to BWR fuel rods, the critical power at which boiling transition is calculated to occur has been adopted as a convenient limit. However, the uncertainties in monitoring the core operating state and in the procedures used to calculate the critical power result in an uncertainty in the value of the critical power. Therefore, the fuel cladding integrity Safety Limit is defined as the CPR in the limiting fuel assembly for which more than 99.9% of the fuel rods in the core are expected to avoid boiling transition considering the power distribution within the core and all uncertainties.

The MCPR Safety Limit is determined using a model that combines all of the uncertainties in operating parameters and the procedures used to calculate critical power. The probability of the occurrence of boiling transition is determined using an NRC-approved critical power correlation. Details of the fuel cladding integrity Safety Limit calculation are presented in Reference 1.

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REACTOR PROTECTION SYSTEM INSTRUMENTATION SETPOINTS (Continued)

Turbine Control Valve Fast Closure, Trip Oil Pressure-Low (Continued)

pressure switches whose contacts form the one-out-of-two-twice logic input to the Reactor Protection System. This trip setting, a nominally 50% greater closure time and a different valve characteristic from that of the turbine stop valve, combine to produce transients very similar to that for the stop valve. No significant change in MCPR occurs. Relevant transient analyses are discussed in Section 15 of the Final Safety Analysis Report. This scram is bypassed when turbine steam flow is below that corresponding to 30% of RATED THERMAL POWER, as measured by turbine first stage pressure.

11. Reactor Mode Switch In Shutdown Position

The reactor mode switch Shutdown position trip is a redundant channel to the automatic protective instrumentation channels and provides additional manual reactor trip capability.

12. Manual Scram

The Manual Scram is a redundant channel to the automatic protective instrumentation channels and provides manual reactor trip capability.

2.3 REFERENCES

1. "General Electric Standard Application for Reactor Fuel (Supplement for United States)," NEDO-24011-P-A.

2. "Edwin I. Hatch Nuclear Plant Units 1 and 2 Single-Loop Operation," NEDO-24205, August 1979.

3/4.1.3 CONTROL RODS

CONTROL ROD OPERABILITY

LIMITING CONDITION FOR OPERATION

3.1.3.1 All withdrawn control rods shall be OPERABLE.

APPLICABILITY: CONDITIONS 1 and 2.

ACTION:

a. With one withdrawn control rod declared inoperable due to being immovable as a result of excessive friction or mechanical interference or known to be untrippable, restore the inoperable control rod to OPERABLE status within 48 hours or be in at least HOT SHUTDOWN within the next 12 hours.

b. With no more than 8 withdrawn control rods declared inoperable, the provisions of Specification 3.0.4 are not applicable and operation may continue, provided that within 2 hours;

   1. The insertion capability of each inoperable withdrawn control rod is demonstrated by inserting the control rod at least one notch* by drive water pressure within the normal operating range and then either the directional control valves are electrically disarmed or the withdraw isolation valve is closed, or

   2. The inoperable control rod is fully inserted and either the directional control valves are electrically disarmed or the withdraw isolation valve is closed, and

   3. Each inoperable withdrawn control rod is separated from all other inoperable withdrawn control rods by at least 2 OPERABLE control rods in all directions;

   Otherwise, be in at least HOT SHUTDOWN within 12 hours.

*The inoperable control rod may then be withdrawn to a position no further withdrawn than its position when found to be inoperable.

REACTIVITY CONTROL SYSTEMS

SURVEILLANCE REQUIREMENTS

4.1.3.1 All withdrawn control rods that do not have their directional control valves electrically disarmed or their withdraw isolation valve closed shall be demonstrated OPERABLE by moving each control rod at least one notch;

a. At least once per 7 days when above 30% of RATED THERMAL POWER, or

b. At least once per 24 hours when above 30% of RATED THERMAL POWER and three or more control rods are immovable.

3/4.1.4 CONTROL ROD PROGRAM CONTROLS

ROD WORTH MINIMIZER

LIMITING CONDITION FOR OPERATION

3.1.4.1 The Rod Worth Minimizer (RWM) shall be OPERABLE.

APPLICABILITY: CONDITIONS 1 and 2*, when THERMAL POWER is less than 10% of RATED THERMAL POWER.

ACTION:

With the RWM inoperable, the provisions of Specification 3.0.4 are not applicable, operation may continue and control rod movement is permitted provided that a second licensed operator or other qualified member of the technical staff is present at the reactor control console and verifies compliance with the prescribed control rod pattern.

SURVEILLANCE REQUIREMENTS

4.1.4.1 The RWM shall be demonstrated OPERABLE:

a. In CONDITION 2 prior to withdrawal of control rods for the purpose of making the reactor critical, and in CONDITION 1 when the RWM is initiated during control rod insertion when reducing THERMAL POWER, by:

   1. Verifying proper annunciation of the selection error of at least one out-of-sequence control rod, and

   2. Verifying the rod block function of the RWM by moving an out-of-sequence control rod.

b. By verifying that the Banked Position Withdrawal Sequence input to the RWM computer is correct following any loading of the sequence program into the computer.

*Entry into OPERATIONAL CONDITION 2 and withdrawal of selected control rods is permitted for the purpose of determining the OPERABILITY of the RWM prior to withdrawal of control rods for the purpose of bringing the reactor to criticality.

REACTIVITY CONTROL SYSTEMS

ROD SEQUENCE CONTROL SYSTEM

LIMITING CONDITION FOR OPERATION

3.1.4.2 The Rod Sequence Control System (RSCS) shall be OPERABLE.

APPLICABILITY: CONDITIONS 1* and 2*#, when THERMAL POWER is less than 10% of RATED THERMAL POWER and control rod movement is within the group notch mode after 50% of the control rods have been withdrawn.

ACTION:

With the RSCS inoperable control rod movement shall not be permitted, except by a scram.

SURVEILLANCE REQUIREMENTS

4.1.4.2 The RSCS shall be demonstrated OPERABLE by:

a. Selecting and attempting to move an inhibited control rod:

1. As soon as the group notch mode is entered during each reactor startup, and

2. As soon as the rod inhibit mode is automatically initiated during control rod insertion.

*See Special Test Exception 3.10.2.

#Entry into CONDITION 2 and withdrawal of selected control rods is permitted for the purpose of determining the OPERABILITY of the RSCS prior to withdrawal of control rods for the purpose of bringing the reactor to criticality.

SURVEILLANCE REQUIREMENTS (Continued)

---

b. Attempting to move a control rod more than one notch as soon as the group notch mode is automatically initiated during control rod:

   1. Withdrawal each reactor startup, and

   2. Insertion.

c. Performance of the comparator check of the group notch circuits prior to control rod;

   1. Movement within the group notch mode during each reactor startup, and

   2. Insertion to reduce THERMAL POWER to less than 10% of RATED THERMAL POWER.

REACTIVITY CONTROL SYSTEMS

ROD BLOCK MONITOR

LIMITING CONDITION FOR OPERATION

3.1.4.3 Both Rod Block Monitor (RBM) channels shall be OPERABLE.

APPLICABILITY: CONDITION 1, when THERMAL POWER is greater than or equal to 30% of RATED THERMAL POWER and when the MCPR is less than the value provided in the CORE OPERATING LIMITS REPORT.

ACTION:

a. With one RBM channel inoperable, POWER OPERATION may continue provided that the inoperable RBM channel is restored to OPERABLE status within 24 hours; otherwise, trip at least one rod block monitor channel within the next hour.

b. With both RBM channels inoperable, trip at least one rod block monitor channel within one hour.

SURVEILLANCE REQUIREMENTS

4.1.4.3 a. With both RBM channels OPERABLE, surveillance requirements are given in Specification 4.3.5.

b. With one RBM channel INOPERABLE, the other channel shall be demonstrated OPERABLE by performance of a CHANNEL FUNCTIONAL TEST prior to withdrawal of control rods.

3/4.2 POWER DISTRIBUTION LIMITS

3/4.2.1 AVERAGE PLANAR LINEAR HEAT GENERATION RATE

LIMITING CONDITION FOR OPERATION

3.2.1 ALL AVERAGE PLANAR LINEAR HEAT GENERATION RATES (APLHGRs) shall be equal to or less than their applicable APLHGR limits provided in the CORE OPERATING LIMITS REPORT.

APPLICABILITY: CONDITION 1, when THERMAL POWER ≥ 25% of RATED THERMAL POWER.

ACTION:

With an APLHGR exceeding its applicable limit provided in the CORE OPERATING LIMITS REPORT, initiate corrective action within 15 minutes and continue corrective action so that the APLHGR meets 3.2.1 within 2 hours or reduce THERMAL POWER to less than 25% of RATED THERMAL POWER within the next 4 hours.

SURVEILLANCE REQUIREMENTS

4.2.1 All APLHGRs shall be verified to be equal to or less than their applicable APLHGR limits provided in the CORE OPERATING LIMITS REPORT:

a. At least once per 24 hours,

b. Whenever THERMAL POWER has been increased by at least 15% of RATED THERMAL POWER and steady state operating conditions have been established, and

c. Initially and at least once per 12 hours when the reactor is operating with a LIMITING CONTROL ROD PATTERN for APLHGR.

Figure 3.2.1-1 (Deleted)

Figure 3.2.1-2 (Deleted)

Figure 3.2.1-3 (Deleted)

Figure 3.2.1-4 (Deleted)

Figure 3.2.1-5 (Deleted)

Figure 3.2.1-6 (Deleted)

Figure 3.2.1-7 (Deleted)

Figure 3.2.1-8 (Deleted)

Figure 3.2.1-9 (Deleted)

Figure 3.2.1-10 (Deleted)

Figure 3.2.1-11 (Deleted)

Figure 3.2.1-12 (Deleted)

Figure 3.2.1-13 (Deleted)

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3/4.2.3 MINIMUM CRITICAL POWER RATIO

LIMITING CONDITION FOR OPERATION

3.2.3 ALL MINIMUM CRITICAL POWER RATIOS (MCPRs) shall be equal to or greater than their applicable MCPR operating limits provided in the CORE OPERATING LIMITS REPORT.

APPLICABILITY: CONDITION 1, when THERMAL POWER ≥ 25% RATED THERMAL POWER

ACTION:

With MCPR less than the applicable operating limit provided in the CORE OPERATING LIMITS REPORT, initiate corrective action within 15 minutes and continue corrective action so that MCPR is equal to or greater than the applicable limit within 2 hours or reduce THERMAL POWER to less than or equal to 25% of RATED THERMAL POWER within the next 4 hours.

SURVEILLANCE REQUIREMENTS

4.2.3.1 The MCPR operating limits shall be determined, as provided in the CORE OPERATING LIMITS REPORT, using:

a. τ = 1.0 prior to the initial scram time measurements for the cycle performed in accordance with Specification 4.1.3.2.a, or

b. τ is determined from scram time measurements performed in accordance with Specification 4.1.3.2. The determination of the limit must be completed within 72 hours of the conclusion of each scram time surveillance test required by Specification 4.1.3.2.

4.2.3.2 All MCPRs shall be determined to be equal to or greater than the applicable limits:

a. At least once per 24 hours,

b. Whenever THERMAL POWER has been increased by at least 15% of RATED THERMAL POWER and steady state operating conditions have been established, and

c. Initially and at least once per 12 hours when the reactor is operating with a LIMITING CONTROL ROD PATTERN for MCPR.

Figure 3.2.3-1 (Deleted)

Figure 3.2.3-2 (Deleted)

Figure 3.2.3-3 (Deleted)

Figure 3.2.3-4 (Deleted)

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3/4.2.4 LINEAR HEAT GENERATION RATE

LIMITING CONDITION FOR OPERATION

3.2.4 ALL LINEAR HEAT GENERATION RATES (LHGRs) shall not exceed their applicable LHGR limits provided in the CORE OPERATING LIMITS REPORT.

APPLICABILITY: CONDITION 1, when THERMAL POWER ≥25% of RATED THERMAL POWER.

ACTION:

With the LHGR of any fuel rod exceeding its LHGR limit provided in the CORE OPERATING LIMITS REPORT, initiate corrective action within 15 minutes and continue corrective action so that the LHGR is within the limit within 2 hours or reduce THERMAL POWER to less than 25% of RATED THERMAL POWER within the next 4 hours.

SURVEILLANCE REQUIREMENTS

4.2.4 All LHGRs shall be determined to be equal to or less than their applicable LHGR limits provided in the CORE OPERATING LIMITS REPORT:

a. At least once per 24 hours,

b. When THERMAL POWER has been increased by at least 15% of RATED THERMAL POWER and steady state operating conditions have been established, and

c. Initially and at least once per 12 hours when the reactor is operating on a LIMITING CONTROL ROD PATTERN FOR LHGR.

CONTROL ROD WITHDRAWAL BLOCK INSTRUMENTATION

NOTE

a. When the limiting condition defined in section 3.1.4.3 exists.

b. This function is bypassed if detector is reading > 100 cps or the IRM channels are on range 3 or higher.

c. This function is bypassed when the associated IRM channels are on range 8 or higher.

d. A total of 6 IRM instruments must be OPERABLE.

e. This function is bypassed when the IRM channels are on range 1.

f. With any control rod withdrawn. Not applicable to control rods removed per Specification 3.9.11.1 or 3.9.11.2.

TABLE 4.3.5-1

CONTROL ROD WITHDRAWAL BLOCK INSTRUMENTATION SURVEILLANCE REQUIREMENTS

TRIP FUNCTION	CHANNEL CHECK	CHANNEL FUNCTIONAL TEST	CHANNEL CALIBRATION[(a)]	OPERATIONAL CONDITIONS IN WHICH SURVEILLANCE REQUIRED
1. APRM				
a. Flow Referenced Simulated Thermal Power- Upscale	NA	S/U[(b)],M	R	1
b. Inoperative	NA	S/U[(b)],M	NA	1, 2, 5
c. Downscale	NA	S/U[(b)],M	R	1
d. Neutron Flux - High, 12%	NA	S/U[(b)],M	R	2, 5
2. ROD BLOCK MONITOR				
a. Upscale	NA	S/U[(b)], M	R	1[(d)]
b. Inoperative	NA	S/U[(b)], M	NA	1[(d)]
c. Downscale	NA	S/U[(b)], M	R	1[(d)]
3. SOURCE RANGE MONITORS				
a. Detector not full in	NA	S/U[(b)],W	NA	2, 5
b.	NA	S/U[(b)],W	R	2, 5
c.	NA	S/U[(b)],W	NA	2, 5
d.	NA	S/U[(b)],W	R	2, 5
4. II				
a.	NA	S/U[(b)],W[(c)]	NA	2, 5
b. Upscale	NA	S/U[(b)],W[(c)]	R	2, 5
c. Inoperative	NA	S/U[(b)],W[(c)]	NA	2, 5
d. Downscale	NA	S/U[(b)],W[(c)]	R	2, 5
5. SCRAM DISCHARGE VOLUME				
a. Water Level-High	NA	Q	R	1, 2, 5[(e)]

TABLE 4.3.5-1 (Continued)

CONTROL ROD WITHDRAWAL BLOCK INSTRUMENTATION SURVEILLANCE REQUIREMENTS

NOTES:

a. Neutron detectors may be excluded from CHANNEL CALIBRATION.

b. Within 24 hours prior to startup, if not performed within the previous 7 days.

c. When changing from CONDITION 1 to CONDITION 2, perform the required surveillance within 12 hours after entering CONDITION 2.

d. When THERMAL POWER exceeds 30% of RATED THERMAL POWER. The additional surveillance defined in Specification 4.1.4.3 will be required when the Limiting Condition defined in Specification 3.1.4.3 exists.

e. With any control rod withdrawn. Not applicable to control rods removed per Specification 3.9.11.1 or 3.9.11.2.

3/4.10.1 PRIMARY CONTAINMENT INTEGRITY

LIMITING CONDITION FOR OPERATION

3.10.1 The provisions of Specifications 3.6.1.1 and 3.6.1.3 may be suspended to permit the reactor pressure vessel closure head and drywell head to be removed and the air lock doors to be open during low power PHYSICS TESTS with THERMAL POWER < 1% of RATED THERMAL POWER and reactor coolant temperature < 212°F.

APPLICABILITY: CONDITION 2, during low power PHYSICS TESTS.

ACTION:

With THERMAL POWER exceeding 1% of RATED THERMAL POWER or with the reactor coolant temperature ≥ 212°F, immediately actuate the manual scram.

SURVEILLANCE REQUIREMENTS

4.10.1 The THERMAL POWER and reactor coolant temperature shall be verified to be within the limits at least once per hour.

SPECIAL TEST EXCEPTIONS

3/4.10.2 ROD SEQUENCE CONTROL SYSTEM

LIMITING CONDITION FOR OPERATION

3.10.2 The sequence constraints imposed on control rod groups A_{12}, A_{34}, B_{12} and B_{34} by the Rod Sequence Control System per Specification 3.1.4.2 may be suspended by means of the individual rod position bypass switches, provided that at least the requirements of Specification 3.1.3.1 and 3.1.4.1 are satisfied, for the following tests:

a. Shutdown margin demonstrations, Specification 4.1.1,

b. Control rod scram, Specification 4.1.3.2a,

c. Control rod friction measurements, and

d. Startup Test Program, with the THERMAL POWER < 10% of RATED THERMAL POWER.

APPLICABILITY: CONDITIONS 1 and 2.

ACTION:

With the requirements of the above specification not satisfied, verify that the RSCS is OPERABLE per Specification 3.1.4.2.

SURVEILLANCE REQUIREMENTS

4.10.2 When the sequence constraints of control rod groups A_{12}, A_{34}, B_{12} and B_{34} are bypassed, verify;

a. That the RWM is OPERABLE per Specification 3.1.4.1,

b. That movement of the control rods from 50% ROD DENSITY to 10% of RATED THERMAL POWER is blocked or limited to the single notch mode, and

c. Conformance with this specification and procedures by a second licensed operator or other qualified member of the technical staff.

CONTROL RODS (Continued)

than has been analyzed even though control rods with inoperable accumulators may still be inserted with normal drive water pressure. Operability of the accumulator ensures that there is a means available to insert the control rods even under the most unfavorable depressurization of the reactors.

Control rod coupling integrity is required to ensure compliance with the analysis of the rod drop accident in the FSAR. The overtravel position feature provides the only positive means of determining that a rod is properly coupled and therefore this check must be performed prior to achieving criticality after each refueling. The subsequent check is performed as a backup to the initial demonstration.

In order to ensure that the control rod patterns can be followed and therefore that other parameters are within their limits, the control rod position indication system must be OPERABLE.

The control rod housing support restricts the outward movement of a control rod to less than 3 inches in the event of a housing failure. The amount of rod reactivity which could be added by this small amount of rod withdrawal is less than a normal withdrawal increment and will not contribute to any damage to the primary coolant system. The support is not required when there is no pressure to act as a driving force to rapidly eject a drive housing.

The required surveillance intervals are adequate to determine that the rods are OPERABLE and not so frequent as to cause excessive wear on the system components.

3/4.1.4 CONTROL ROD PROGRAM CONTROLS

Control rod withdrawal and insertion sequences are established to assure that the maximum insequence individual control rod or control rod segments which are withdrawn at any time during the fuel cycle could not be worth enough to cause the peak fuel enthalpy for any postulated control rod accident to exceed 280 cal/gm. The specified sequences are characterized by homogeneous, scattered patterns of control rod withdrawal. When THERMAL POWER is ≥ 10% of RATED THERMAL POWER, there is no possible rod worth which, if dropped at the design rate of the velocity limiter, could result in a peak enthalpy of 280 cal/gm. Thus, requiring the RWM to be OPERABLE below 10% of RATED THERMAL POWER and the RSCS to be OPERABLE from 50% control rod density to 10% of RATED THERMAL POWER provides adequate control.

The specifications of this section assure that the peak cladding temperature following the postulated design basis loss-of-coolant accident will not exceed the 2200°F limit specified in the Final Acceptance Criteria (FAC) issued in June 1971 considering the postulated effects of fuel pellet densification. These specifications also assure that fuel design margins are maintained during abnormal transients.

3/4.2.1 AVERAGE PLANAR LINEAR HEAT GENERATION RATE

This specification assures that the peak cladding temperature following the postulated design basis loss-of-coolant accident will not exceed the limit specified in 10 CFR 50.46.

The peak cladding temperature (PCT) following a postulated loss-of-coolant accident is primarily a function of the average heat generation rate of all the rods of a fuel assembly at any axial location and is dependent only secondarily on the rod-to-rod power distribution within an assembly. The peak clad temperature is calculated assuming an LHGR for the highest powered rod which is equal to the design LHGR for that fuel type.

The calculational procedure used to establish the APLHGR limits for Technical Specification 3/4.2.1 is based on a loss-of-coolant accident analysis. The analysis was performed using General Electric (GE) calculational models which are consistent with the requirements of Appendix K to 10 CFR 50. The Loss-of-Coolant Accident (LOCA) analysis was performed utilizing the new improved calculational model, SAFER/GESTR-LOCA. The analysis demonstrated that loss-of-coolant concerns do not limit the operation of the fuel since margin to the 2200°F limit was demonstrated for all of these fuel types (Reference 4). Therefore, the APLHGR limits for the fuel types shown in the CORE OPERATING LIMITS REPORT are based on the fuel thermal-mechanical design criteria.

A flow dependent correction factor is applied to the rated conditions APLHGR to assure that the 2200°F PCT limit is complied with during a LOCA initiated from less than rated core flow. In addition, other power and flow dependent corrections are applied to the rated conditions APLHGR limit to assure that the fuel thermal-mechanical design criteria are preserved during abnormal transients initiated from off-rated conditions. For single-loop operation, a multiplicative factor is applied to the rated conditions APLHGR limit for all fuel bundles when core power exceeds a specified value. The power and flow-dependent correction factors and the limiting values for APLHGR for each fuel type used in a particular cycle are specified in the CORE OPERATING LIMITS REPORT.

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3/4.2.2 APRM SETPOINTS

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3/4.2.3 MINIMUM CRITICAL POWER RATIO

The required operating limit MCPRs at steady state operating conditions as specified in Specification 3.2.3 are derived from the established fuel cladding integrity Safety Limit MCPR of 1.04 for two-loop operation and 1.05 for single-loop operation, and an analysis of abnormal operational transients (Reference 1). For any abnormal operating transient analysis evaluation with the initial condition of the reactor being at the steady state operating limit (specified in the CORE OPERATING LIMITS REPORT), it is required that the resulting MCPR does not decrease below the Safety Limit MCPR at any time during the transient assuming instrument trip setting as given in Specification 2.2.1.

To assure that the fuel cladding integrity Safety Limits are not violated during any anticipated abnormal operational transient, the most limiting transients have been analyzed to determine which results in the largest reduction in CRITICAL POWER RATIO (CPR). The type of transients evaluated were loss of flow, increase in pressure and power, positive reactivity insertion, and coolant temperature decrease.

Details of how evaluations are performed, the methods used, and how the MCPR limit is adjusted for operation at less than rated power and flow conditions are given in Reference 1 and in the CORE OPERATING LIMITS REPORTS.

MINIMUM CRITICAL POWER RATIO (Continued)

As depicted in the CORE OPERATING LIMITS REPORT the 100% power, 100% flow operating limit MCPR (OLMCPR) depends on the average scram time, τ, of the control rods, where:

$$\tau = 0 \text{ or } \frac{\tau_{ave} - \tau_B}{\tau_A - \tau_B}\text{, whichever is greater}$$

where: τ_A = 1.096 sec (Specification 3.1.3.3, scram time limit to notch 36)

$$\tau_B = \mu + 1.65\left[\frac{N_1}{\sum_{i=1}^{n} N_i}\right]^{1/2}\sigma$$

where: μ = 0.822 sec (mean scram time used in the transient analysis)

σ = .018 sec (standard deviation of μ)

$$\tau_{ave} = \frac{\sum_{i=1}^{n} N_i \tau_i}{\sum_{i=1}^{n} N_i}$$

where: n = number of surveillance tests performed to date in the cycle

N_i = number of active control rods measured in the ith surveillance test

τ_i = average scram time to notch 36 of all rods in the ith surveillance test

N_1 = total number of active rods measured in 4.1.3.2.a.

References:

1. "General Electric Standard Application for Reactor Fuel (Supplement for United States)," NEDE-24011-P-A.

3/4.4 REACTOR COOLANT SYSTEM

BASES

3/4.4.1.1 RECIRCULATION SYSTEM

Operation with a reactor coolant recirculation loop inoperable is allowed, provided that adjustments to the flow referenced scram and APRM rod block setpoints, MCPR cladding integrity Safety Limit, MCPR Operating Limit, and MAPLHGR limit are made. An evaluation of the performance of the ECCS with single-loop operation has been performed and determined to be acceptable, Reference 1. The adjustments to the APLHGR and the MCPR limits that are required for single-loop operation are provided in the CORE OPERATING LIMITS REPORT. The flow referenced simulated thermal power setpoint for single-loop operation is reduced by the amount of mΔW, where m is the flow reference slope for the rod block monitor and ΔW is the largest difference between two-loop and single-loop effective drive flow when the active loop indicated flow is the same. This adjustment is necessary to preserve the original relationship between the scram trip and actual drive flow.

The possibility of experiencing limit cycle oscillations during single-loop operation is precluded by restricting the core flow to greater than or equal to 45% of rated when core thermal power is greater than the 80% rod line. This requirement is based on General Electric's recommendations contained in SIL-380, Revision 1, which defines the region where the limit cycle oscillations are more likely to occur.

3/4.4.1.2 JET PUMPS

An inoperable jet pump is not, in itself, a sufficient reason to declare a recirculation loop inoperable, but it does present a hazard in case of a design basis Loss-of-Coolant Accident by increasing the blowdown area and eliminating the capability of reflooding the core; thus, the requirement for shutdown of the facility with a jet pump inoperable is necessary.

One of the acceptable procedures for jet pump failure surveillance identified in NUREG/CR-3052, Reference 2, was adopted for Hatch Unit 2. The surveillance is performed to verify that neither of the following conditions occur:

(a) The recirculation pump flow/speed ratio deviates by more than 5% from the normal range, or

(b) The jet pump loop flow/speed ratio deviates by more than 5% from the normal range.

If either criterion is failed, then the procedure calls for comparing either the individual jet pump flows or the individual jet pump diffuser to lower plenum differential pressures to the criteria of the Limiting Conditions for Operation (LCO). If the LCO is not satisfied and pump speed is less than

e. Type of container, e.g., LSA, type A, type B, large quantity

f. Solidification agent, e.g., cement.

The Radioactive Effluent Release Report shall include (on a quarterly basis) unplanned releases from the site to unrestricted areas of radioactive materials in gaseous and liquid effluents that were in excess of 1 Ci, excluding dissolved and entrained gases and tritium for liquid effluents, or those in excess of 150 Ci of noble gases or 0.02 Ci of radioiodines for gaseous releases.

The Radioactive Effluent Release Report shall include any changes to the PROCESS CONTROL PROGRAM and to the OFFSITE DOSE CALCULATION MANUAL made during the reporting period.

MONTHLY OPERATING REPORT

6.9.1.10 Routine reports of operating statistics and shutdown experience shall be submitted on a monthly basis to the Director, Office of Management and Program Analysis, U. S. Nuclear Regulatory Commission, Washington, D. C. 20555, with a copy to the Regional Office of Inspection and Enforcement no later than the 15th of each month following the calendar month covered by the report.

CORE OPERATING LIMITS REPORT

6.9.1.11.a. Core operating limits shall be established and documented in the CORE OPERATING LIMITS REPORT before each reload cycle or any remaining part of a reload cycle for the following:

(1) Control Rod Program Controls - Rod Block Monitor for Specification 3.1.4.3,

(2) The Average Planar Linear Heat Generation Rate for Specification 3.2.1 and Surveillance Requirement 4.2.1,

(3) The Minimum Critical Power Ratio for Specifications 3.1.4.3 and 3.2.3 and Surveillance Requirement 4.2.3, and

(4) The Linear Heat Generation Rate for Specification 3.2.4 and Surveillance Requirement 4.2.4.

b. The analytical methods used to determine the core operating limits shall be those previously reviewed and approved by the NRC in the following documents.

(1) NEDE-24011-P-A, "General Electric Standard Application for Reactor Fuel," (applicable amendment specified in the CORE OPERATING LIMITS REPORT).

(2) "Safety Evaluation by the Office of Nuclear Reactor Regulation Supporting Amendment Nos. 151 and 89 to Facility Operating Licenses DPR-57 and NPF-5," dated January 22, 1988.

c. The core operating limits shall be determined so that all applicable limits (e.g., fuel thermal-mechanical limits, core thermal-hydraulic limits, ECCS limits, nuclear limits such as shutdown margin, and transient and accident analysis limits) of the safety analysis are met.

d. The CORE OPERATING LIMITS REPORT, including any mid-cycle revisions or supplements thereto, shall be provided upon issuance, for each reload cycle, to the NRC Document Control Desk with copies to the Regional Administrator and Resident Inspector.