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Docket No. 50-321

Georgia Power Company
Oglethorpe Electric Membership Corporation
Municipal Electric Association of Georgia
City of Dalton, Georgia
ATTN: Mr. Charles F. Whitmer
Vice President - Engineering
Georgia Power Company
Atlanta, Georgia 30302

Gentlemen:

The Commission has issued the enclosed Amendment No. 52 to Facility Operating License No. DPR-57 for the Edwin I. Hatch Nuclear Plant Unit No. 1. The amendment consists of changes to the Technical Specifications in response to your request of August 26, 1977 as supplemented December 1, 1977, January 5, February 22 and March 8 and 16, 1978.

The amendment modifies the Technical Specifications to: (1) permit operation of the facility during Cycle 3 with up to 168 improved two water rod 8x8R reload fuel bundles, designed and fabricated by the General Electric Company (GE) and having an average enrichment of 2.65 wt/% U-235, and (2) revise the maximum average planar linear heat generation rates (MAPLHGR's) as determined by the reevaluation of the ECCS performance.

Copies of the Safety Evaluation and Notice of Issuance are also enclosed.

Sincerely,

George Lear, Chief
Operating Reactors Branch #3
Division of Operating Reactors

Enclosures:
1. Amendment No. 52 to DPR-57
2. Safety Evaluation
3. Notice

Subject to condition in note

GD

cc w/enclosures:
see next page

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DATE	4/4/78	4/4/78	4/7/78	4/11/78	4/11/78

Georgia Power Company
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Municipal Electric Association of Georgia
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UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D. C. 20555

GEORGIA POWER COMPANY
OGLETHORPE ELECTRIC MEMBERSHIP CORPORATION
MUNICIPAL ELECTRIC ASSOCIATION 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. 52
License No. DPR-57

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

   A. The application for amendment by Georgia Power Company, et al, (the licensee) dated August 26, 1977, as supplemented December 1, 1977, January 3, January 5, February 22 and March 8 and 16, 1978, 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-57 is hereby amended to read as follows:

   (2) Technical Specifications

   The Technical Specifications contained in Appendices A and B, as revised through Amendment No. 52, 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 the date of its issuance.

FOR THE NUCLEAR REGULATORY COMMISSION

Brian K. Grimes, Assistant Director
for Engineering and Projects
Division of Operating Reactors

Attachment:
Changes to the Technical
Specifications

Date of Issuance: April 11, 1978

ATTACHMENT TO LICENSE AMENDMENT NO. 52

FACILITY OPERATING LICENSE NO. DPR-57

DOCKET NO. 50-321

Revise Appendix A as follows:

Remove Pages	Insert Pages
1.1-1	1.1-1
1.1-2	1.1-2
1.1-8	1.1-8
1.1-12	1.1-12
1.1-13	1.1-13
1.1-14	1.1-14
1.2-3	1.2-3
1.2-5	1.2-5
3.2-11	3.2-11
3.2-49*	3.2-49*
3.2-50	3.2-50
3.2-52	3.2-52
3.2-55	3.2-55
3.2-56*	3.2-56*
3.2-61	3.2-61
3.2-63	3.2-63
3.2-64	3.2-64
3.3-5	3.3-5
3.3-10	3.3-10
3.3-15	3.3-15
3.6-20	3.6-20
3.11-1	3.11-1
3.11-2	3.11-2
3.11-4	3.11-4
Figure 3.11-1 (Sheets 1-3)	Figure 3.11-1 (Sheets 1, 2)
5.0-1	5.0-1

*overleaf, no changes on this page

1.1 FUEL CLADDING INTEGRITY

Applicability

The Safety Limits established to preserve the fuel cladding integrity apply to those variables which monitor the fuel thermal behavior.

Objective

The objective of the Safety Limits is to establish limits below which the integrity of the fuel cladding is preserved.

Specifications

A. Reactor Pressure > 800 psia and Core Flow > 10% of Rated

The existence of a minimum critical power ratio (MCPR) less than 1.07 shall constitute violation of the fuel cladding integrity safety limit.

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

When the reactor pressure is ≤ 800 psia or core flow is less than 10% of rated, the core thermal power shall not exceed 25% of rated thermal power.

C. Power Transient

To ensure that the Safety Limit established in Specification 1.1.A and 1.1.B is not exceeded, each required scram shall be initiated by its expected scram signal. The Safety Limit shall be assumed to be exceeded when scram is accomplished by a means other than the expected scram signal.

2.1 FUEL CLADDING INTEGRITY

Applicability

The Limiting Safety System Settings apply to trip settings of the instruments and devices which are provided to prevent the fuel cladding integrity Safety Limits from being exceeded.

Objective

The objective of the Limiting Safety System Settings is to define the level of the process variables at which automatic protective action is initiated to prevent the fuel cladding integrity Safety Limits from being exceeded.

Specifications

A. Trip Settings

The limiting safety system trip settings shall be as specified below:

1. Neutron Flux Trip Settings

a. IRM High High Flux Scram Trip Setting
The IRM flux scram trip setting shall be ≤ 120/125 of full scale.

b. APRM Flux Scram Trip Setting (Refuel or Start & Hot Standby Mode)
When the Mode Switch is in the REFUEL or START & HOT STANDBY position, the APRM flux scram trip setting shall be ≤ 15/125 of full scale (i.e., ≤ 15% of rated thermal power).

c. APRM High High Flux Scram Trip Setting (Run Mode)
When the Mode Switch is in the RUN position, the APRM flux scram trip setting shall be:

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

Whenever the reactor is in the Hot or Cold Shutdown Condition with irradiated fuel in the reactor vessel, the water level shall be > 378 inches above vessel invert when fuel is seated in the core.

2.1.A.1.c. APRM High High Flux Scram Trip Setting (Run Mode) (Continued)

$$S \leq 0.66\ W + 54\%$$

where:

S = Setting in percent of rated thermal power (2436 MWt)

W = Loop recirculation flow rate in percent of rated (rated loop recirculation flow rate equals 34.2×10^6 lb/hr)

In the event of operation with a maximum total peaking factor (MTPF) greater than the design value, the setting shall be modified as follows:

$$S \leq (0.66\ W + 54\%)\ \frac{A}{MTPF}$$

where:

MTPF = The value of the existing maximum total peaking factor

A = 2.60 for 7x7 fuel
2.42 for 8x8 fuel

For no combination of loop recirculation flow rate and core thermal power shall the APRM flux scram trip setting be allowed to exceed 120% of rated thermal power.

Surveillance requirements for MTPF are given in Specification 4.1.B.

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 BWR Thermal Analysis Basis (GETAB) Data, Correlation and Design Application", NEDO 10958 and NEDE 10958.

2. "Edwin I. Hatch Nuclear Plant Unit 1 Channel Inspection and Safety Analysis with Bypass Flow Holes Plugged", NEDO-21124, November, 1975.

3. General Electric "Process Computer Performance Evaluation Accuracy", NEDO-20340, and Amendment 1, NEDO-20340-1, dated June, 1974 and December, 1974, respectively.

2.1.A.1.a. IRM Flux Scram Trip Setting (Continued)

tism was taken in this analysis by assuming that the IRM channel closest to the withdrawn rod is bypassed. The results of this analysis show that the reactor is scrammed and peak power limited to one percent of rated power, thus maintaining MCPR above 1.07. Based on the above analysis, the IRM provides protection against local control rod withdrawal errors and continues withdrawal of control rods in sequence and provides backup protection for the APRM.

b. APRM Flux Scram Trip Setting (Refuel or Start & Hot Standby Mode)

For operation in the startup mode while the reactor is at low pressure, the APRM scram setting of 15 percent of rated power provides adequate thermal margin between the setpoint and the safety limit, 25 percent of rated. The margin is adequate to accomodate anticipated maneuvers associated with power plant startup. 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 and the Rod Sequence Control System. 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 power, 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 5 percent of rated power per minute, and the APRM system would be more than adequate to assure a scram before the power could exceed the safety limit. The 15 percent APRM scram remains active until the mode switch is placed in the RUN position. This switch occurs when reactor pressure is greater than 825 psig.

c. APRM High High Flux Scram Trip Setting (Run Mode)

The average power range monitoring (APRM) system, which is calibrated using heat balance data taken during steady state conditions, reads in percent of rated thermal power (2436 MWt). Because fission chambers provide the basic input signals, the APRM system responds directly to average neutron flux. During transients, the instantaneous rate of heat transfer from the fuel (reactor thermal power) is less than the instantaneous neutron flux due to the time constant of the fuel. Therefore, during abnormal operational transients, the thermal power of the fuel will be less than that indicated by the neutron flux at the scram setting. Analyses demonstrate that with a 120 percent scram trip setting, none of the abnormal operational transients analyzed violate the fuel Safety Limit and there is a substantial margin from fuel damage. Therefore, the use of flow referenced scram trip provides even additional margin.

2.1.A.1.c. APRM High High Flux Scram Trip Setting (Run Mode) (Continued)

An increase in the APRM scram trip setting would decrease the margin present before the fuel cladding integrity Safety Limit is reached. The APRM scram trip setting was determined by an analysis of margins required to provide a reasonable range for maneuvering during operation. Reducing this operating margin would increase the frequency of spurious scrams which have an adverse effect on reactor safety because of the resulting thermal stresses. Thus, the APRM scram trip setting was selected because it provides adequate margin for the fuel cladding integrity Safety Limit yet allows operating margin that reduces the possibility of unnecessary scrams.

The scram trip setting must be adjusted to ensure that the LHGR transient peak is not increased for any combination of MTPF and reactor core thermal power. The scram setting is adjusted in accordance with the formula in Specification 2.1.A.1.c., when the maximum total peaking factor is greater than 2.60 for 7x7 fuel and 2.42 for 8x8 fuel.

Analyses of the limiting transients show that no scram adjustment is required to assure MCPR > 1.07 when the transient is initiated from the operating MCPR limit.

d APRM Rod Block Trip Setting

Reactor power level may be varied by moving control rods or by varying the recirculation flow rate. The APRM system provides a control rod block to prevent rod withdrawal beyond a given point at constant recirculation flow rate, and thus to protect against the condition of a MCPR less than 1.07. This rod block trip setting, which is automatically varied with recirculation loop flow rate, prevents an increase in the reactor power level to excessive values due to control rod withdrawal. The flow variable trip setting provides substantial margin from fuel damage, assuming a steady-state operation at the trip setting, over the entire recirculation flow range. The margin to the Safety Limit increases as the flow decreases for the specified trip setting versus flow relationship; therefore the worst case MCPR which would occur during a steady-state operation is at 108% of rated thermal power because of the APRM rod block trip setting. The actual power distribution in the core is established by specified control rod sequences and is monitored continuously by the in-core LPRM system. As with the APRM scram trip setting, the APRM rod block trip setting is adjusted downward if the maximum total peaking factor exceeds 2.60 for 7x7 fuel and 2.42 for 8x8 fuel, thus preserving the APRM rod block safety margin.

2. Reactor Water Low Level Scram Trip Setting (LL1)

The trip setting for low level scram is above the bottom of the separator skirt. This level is > 14 feet above the top of the active fuel. This level has been used in transient analyses dealing with coolant inventory decrease. The results reported in FSAR Section 14.3 show that a scram at this level adequately protects the fuel and the pressure barrier. The scram trip setting is approximately 33 inches below the normal operating range and is thus adequate to avoid spurious scrams.

2.1.A.3. Turbine Stop Valve Closure Scram Trip Settings

The turbine stop valve closure scram trip anticipates the pressure, neutron flux and heat flux increase that could result from rapid closure of the turbine stop valves. With a scram trip setting of ≤ 10 percent of valve closure from full open, the resultant increase in surface heat flux is limited such that MCPR remains above 1.07 during the worst case transient that assumes the turbine bypass is closed. This scram is bypassed when turbine steam flow is below 30% of rated, as measured by turbine first stage pressure.

4. Turbine Control Valve Fast Closure Scram Trip Setting

This turbine control valve fast closure scram anticipates the pressure, neutron flux, and heat flux increase that could result from fast closure of the turbine control valves due to load rejection exceeding the capability of the turbine bypass. The Reactor Protection System initiates a scram when fast closure of the control valves is initiated by the fast acting solenoid valves. This is achieved by the action of the fast acting solenoid valves in rapidly reducing hydraulic control oil pressure at the main turbine control valve actuator disc dump valves. This loss of pressure is sensed by 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 and no more severe than for the stop valve. This scram is bypassed when turbine steam flow is below 30% of rated, as measured by turbine first stage pressure.

5. Main Steam Line Isolation Valve Closure Scram Trip Setting

The main steam line isolation valve closure scram occurs within 10% of valve movement from the fully open position and thus anticipates the neutron flux and pressure scrams which remain as available backup protection. This scram function is bypassed automatically when the reactor pressure is below 1045 psig and the Mode Switch is not in the RUN position.

6. Main Steam Line Isolation Valve Closure on Low Pressure

The low pressure isolation of the main steam lines at 825 psig was provided to protect against rapid reactor depressurization and the resulting rapid cooldown of the vessel, which might result from a pressure regulator failure causing inadvertent opening of the control and/or bypass valves.

1.2 REACTOR COOLANT SYSTEM INTEGRITY

The reactor coolant system integrity is an important barrier in the prevention of uncontrolled release of fission products. It is essential that the integrity of this system be protected by establishing a pressure limit to be observed for all operating conditions and whenever there is irradiated fuel in the reactor vessel.

A. Reactor Vessel Steam Dome Pressure

1. When Irradiated Fuel is in the Reactor

The pressure Safety Limit of 1325 psig as measured by the reactor vessel steam dome pressure indicator is equivalent to 1375 psig at the lowest elevation of the reactor coolant system. The 1375 psig value is derived from the design pressure of the reactor pressure vessel (1250 psig) and coolant system piping (suction piping: 1150 psig; discharge piping: 1350 psig). The pressure Safety Limit was chosen as the lower pressure resulting from the pressure transients permitted by the applicable design codes: ASME Boiler and Pressure Vessel Code, Section III for the pressure vessel and USASI B31.1 Code for the reactor coolant system piping. The ASME Boiler and Pressure Vessel Code permits pressure transients up to 10% over design pressure (110% x 1250 = 1375 psig), and the USASI Code permits pressure transients up to 20% over the design pressure (120% x 1150 = 1380 psig; 120% x 1350 = 1620 psig).

The pressure relief system (relief/safety valves) has been sized to meet the overpressure protection criteria of the ASME Boiler and Pressure Vessel Code, Section III, Nuclear Vessels.

The details of the overpressure protection analysis showing compliance with the ASME Boiler and Pressure Vessel Code, Section III, Nuclear Vessels is provided in the FSAR, Appendix M, Summary Technical Report of Reactor Vessel Overpressure Protection. To determine the required steamflow capacity, a parametric study was performed assuming the plant was operating at the turbine generator design condition of 105 percent rated steam flow (10.6 x 10^6 pounds per hour) with a vessel dome pressure of 1020 psig, at a reactor thermal power of 2537 Mw, and the reactor experiences the worst pressurization transient. The analysis of the worst overpressure transient, a 3 second closure of all main steam line isolation valves neglecting the direct scram (valve position scram) results in a maximum vessel pressure (bottom) of 1234 psig if a neutron flux scram is assumed. In addition, the same event was analyzed to determine the number of installed valves which would limit pressure to below the code limit. The results of this analysis show that the eleven installed relief/safety valves are adequate even if assuming the backup neutron flux scram, and provide 141 psi margin.

Load rejection from high power without bypass is the most severe transient resulting directly in a nuclear system pressure increase, assuming the turbine trip scram. This event is presented in Reference 5. The analysis shows that the peak pressure in the bottom of the vessel is limited to 1192 psig. Peak steam line pressure is 1152 psig, showing adequate protection for this abnormal operational transient.

1.2.B. References

1. ASME Boiler and Pressure Vessel Code Section III

2. USASI Piping Code, Section B31.1

3. FSAR Section 4.2, Reactor Vessel and Appurtenances Mechanical Design

4. FSAR Section 14.3, Analysis of Abnormal Operation Transients

5. General Electric Boiling Water Reactor Supplemental Reload Licensing Amendment for the Edwin I. Hatch Nuclear Plant Unit 1 Reload 2, NEDO-24078, November, 1977.

TABLE 3.2-5

INSTRUMENTATION WHICH INITIATES OR CONTROLS THE LPCI MODE OF RHR

Ref. No. (a)	Instrument	Trip Condition Nomenclature	Required Operable Channels per Trip System (b)	Trip Setting	Remarks
1	Reactor Water Level (Yarway)				
		Low Low Low (LL3)	2	≥ - 146.5 inches	Initiates LPCI mode of RHR
2	Drywell	High	2	≤ 2 psig	Initiates LPCI mode of RHR
3	Reactor Pressure	High (Shutdown Cooling Mode)	1	≤ 135 psig	With primary containment isolation signal, closes RHR (LPCI) inboard motor operated injection valves
		Low	2	≤ 335 psig	Permissive to close Recirculation Discharge Valve and Bypass Valve
		Low	2	≤ 500 psig	Permissive to open LPCI injection valves
4	Reactor Water Level (Shroud Level Indicator)		1	> 313.5 inches	Acts as permissive to divert some LPCI flow to containment spray
5	LPCI Cross Connect Valve Open Annunciator	N/A	1	Valve not closed	Initiates annunciator when valve is not closed

Notes for Table 4.2-11

a. The column entitled "Ref. No." is inly for convenience so that a one-to-one relationship can be established between items in Table 4.2-11 and items in Table 3.2-11.

b. Instrument checks are not required when the instruments are not required to be operable or are tripped. However, if instrument checks are missed, they shall be performed prior to returning the instrument to an operable status.

c. Calibrations are not required when the instruments are not required to be operable or are tripped. However, if calibrations are missed, they shall be performed prior to returning the instrument to an operable status.

3.2 PROTECTIVE INSTRUMENTATION

In addition to the Reactor Protection System (RPS) instrumentation which initiates a reactor scram, protective instrumentation has been provided which initiates action to mitigate the consequences of accidents which are beyond the operators ability to control, or terminates operator errors before they result in serious consequences. This set of Specifications provides the limiting conditions for operation of the instrumentation:
(a) which initiates reactor vessel and primary containment isolation,
(b) which initiates or controls the core and containment cooling systems,
(c) which initiates control rod blocks, (d) which initiates protective action,
(e) which monitors leakage into the drywell and (f) which provides surveillance information. The objectives of these specifications are (i) to assure the effectiveness of the protective instrumentation when required by preserving its capability to tolerate a single failure of any component of such systems even during periods when portions of such systems are out of service for maintenance, and (ii) to prescribe the trip settings required to assure adequate performance. When necessary, one channel may be made inoperable for brief intervals to conduct required functional tests and calibrations.

A. Instrumentation Which Initiates Reactor Vessel and Primary Containment Isolation (Table 3.2-1)

Isolation valves are installed in those lines which penetrate the primary containment and must be isolated during a loss of coolant accident so that the radiation dose limits are not exceeded during an accident condition. Actuation of these valves is initiated by protective instrumentation shown in Table 3.2-1 which senses the conditions for which isolation is required. Such instrumentation must be available whenever primary containment integrity is required. The objective is to isolate the primary containment so that the guidelines of 10 CFR 100 are not exceeded during an accident. The events when isolation is required are discussed in Appendix G of the FSAR. The instrumentation which initiates primary system isolation is connected in a dual bus arrangement.

1. Reactor Water Level

a. Reactor Water Level low (LL1) (Narrow Range)

The reactor water level instrumentation is set to trip when reactor water level is approximately 14 feet above the top of the active fuel. This level is referred to as LL1 in the Technical Specifications and corresponds to a reading of 12.5 inches on the Narrow Range Scale. This trip initiates Group 2 and 5 isolation but does not trip the recirculation pumps.

b. Reactor Water Level Low Low (LL2) (Yarway)

The reactor water level instrumentation is set to trip when reactor water level is approximately 10 feet above the top of the active fuel. This level is referred to as LL2 in the Technical Specifications and corresponds to a reading of -38 inches on the Yarway. This trip initiates Group 1 isolation and trips the recirculation pumps.

3.2.A.7. Main Steam Line Tunnel Temperature High (Continued)

with the resultant small release of radioactivity, gives isolation before the guidelines of 10 CFR 100 are exceeded.

8. Reactor Water Cleanup System Differential Flow High

Gross leakage (pipe break) from the reactor water cleanup system is detected by measuring the difference of flow entering and leaving the system. The set point is low enough to ensure prompt isolation of the cleanup system in the event of such a break but, not so low that spurious isolation can occur due to normal system flow fluctuations and instrument noise. Time delay relays are used to prevent the isolation signal which might be generated frofm the initial flow surge when the cleanup system is started or when operational system adjustments are made which produce short term transients.

9. Reactor Water Cleanup Equipment Room Temperature High and

10. Reactor Water Cleanup Equipment Room Differential Temperature High

Leakage in the high temperature process flow of the reactor water cleanup system external to the primary containment will be detected by temperature sensing elements. Temperature sensors are located in the inlet and outlet ventilation ducts to measure the temperature difference. Local ambient temperature sensors are located in the compartment containing equipment and piping for this system. An alarm in the main control room will be set to annunciate a temperature rise corresponding to a leakage within the identified limit. In addition to annunciation, a high cleanup room temperature will actuate automatic isolation of the cleanup system.

11. Condenser Vacuum Low

The Bases for Condenser Vacuum Low are discussed in The Bases for Specification 2.1.A.7.

B. Instrumentation Which Initiates or Controls HPCI (Table 3.2-2)

1. Reactor Water Level Low Low (LL2) (Yarway)

The reactor water level instrumentation setpoint which initiates HPCI is $\geq$ -38 inches on the Yarway. This level is approximately 10 feet above the top of the active fuel and in the Technical Specifications is referred to as LL2. The reactor vessel low water level setting for HPCI system initiation is selected high enough above the active fuel to start the HPCI system in time both to prevent excessive fuel clad temperatures and to prevent more than a small fraction of the core from reaching the temperature at which gross fuel failure occurs. The water level setting is far enough below normal levels that spurious HPCI system startups are avoided.

2. Drywell Pressure High

The drywell pressure instrumentation setpoint which initiates HPCI is $\leq$ 2 psig. High drywell pressure is indicative of a failure of the nuclear system process barrier. This pressure is selected to be as low as possible without inducing spurious HPCI system startups. This instrumentation serves as a backup to the water level instrumentation described above.

3.2.B.14 Suppression Chamber Area Differential Air Temperature High

As for the HPCI equipment room differential temperature, and for the same reason, a differential air temperature between the inlet and outlet ducts which ventilate the suppression chamber area will also initiate a timer to isolate the HPCI turbine steam line.

15. Condensate Storage Tank Level Low

The CST is the preferred source of suction for HPCI. In order to provide an adequate water supply, an indication of low level in the CST automatically switches the suction to the suppression chamber. A trip setting of 0 inches corresponds to 10,000 gallons of water remaining in the tank.

16. Suppression Chamber Water Level High

A high water level in the suppression chamber automatically switches HPCI suction to the suppression chamber from the CST.

17. HPCI Logic Power Failure Monitor

The HPCI Logic Power Failure Monitor monitors the availability of power to the logic system. In the event of loss of availability of power to the logic system, an alarm is annunciated in the control room.

C. Instrumentation Which Initiates or Controls RCIC (Table 3.2-3)

1. Reactor Water Level Low Low (LL2) (Yarway)

The reactor water level instrumentation setpoint which initiates RCIC is $\geq$ -38 inches on the Yarway. This level is approximately 10 feet above the top of the active fuel and is referred to as LL2. This setpoint insures that RCIC is started in time to preclude conditions which lead to inadequate core cooling.

2. RCIC Turbine Overspeed

The RCIC turbine is automatically shutdown by tripping the RCIC turbine stop valve closed when the 125% speed at rated flow setpoint on the mechanical governor is reached. Turbine overspeed is indicative of a condition which threatens the physical integrity of the system. An electrical tachometer trip setpoint of 110% also will trip the RCIC turbine stop valve closed.

3. RCIC Turbine Exhaust Pressure High

When RCIC turbine exhaust pressure reaches the setpoint ($\leq$ 25 psig), the RCIC turbine is automatically shutdown by tripping the RCIC turbine stop valve closed. RCIC turbine exhaust high pressure is indicative of a condition which threatens the physical integrity of the exhaust line.

4. RCIC Pump Suction Pressure Low

One pressure switch is used to detect low RCIC system pump suction pressure and is set to trip the RCIC turbine at $\leq$ 15 inches of mercury vacuum. This setpoint is chosen to prevent pump damage by cavitation.

3.2.C.5. Reactor Water Level High (Narrow Range)

A reactor water level of +58 inches on the Narrow Range scale is indicative that the RCIC system has performed satisfactorily in providing make-up water to the reactor vessel. The reactor vessel high water level setting which trips the RCIC turbine is near the top of the steam separators and is sufficient to prevent gross moisture carryover to the RCIC turbine. Two level switches trip to initiate an RCIC turbine shutdown.

6. RCIC System Flow

To prevent damage by overheating at reduced RCIC system pump flow, a pump discharge minimum flow bypass is provided. The bypass is controlled by an automatic, D. C. motor-operated valve. A high flow signal from a flow meter downstream of the pump on the main RCIC line will cause the bypass valve to close. Two signals are required to open the valve: An RCIC pump discharge pressure switch high pressure signal must be received to act as a permissive to open the bypass valve in the presence of a low flow signal from the flow switch.

Note:

Because the steam supply line to the RCIC turbine is part of the nuclear system process barrier, the following conditions (7 - 13) automatically isolate this line, causing shutdown of the RCIC system turbine.

7. RCIC Equipment Room Temperature High

High ambient temperature in the RCIC equipment room near the emergency area cooler could indicate a break in the RCIC system turbine steam line. The automatic closure of the RCIC steam line valves prevents the excessive loss of reactor coolant and the release of significant amounts of radioactive material from the nuclear system process barrier. The high temperature setting of 90 F + ambient was selected to be far enough above anticipated normal RCIC system operational levels to avoid spurious isolation but low enough to provide timely detection of an RCIC turbine steam line break. The high temperature trip initiates a timer which isolates the RCIC turbine steam line if the temperature is not reduced below the setpoint.

8. RCIC Equipment Room Differential Temperature High

A high differential temperature of 50 F between the RCIC equipment room vent air inlet and outlet ducts could also indicate an RCIC turbine steam line break and will also initiate the timer to isolate the RCIC turbine steam line.

9. RCIC Steam Line Pressure Low

Low pressure in the RCIC Steam Line could indicate a break in the RCIC steam line. Therefore, the RCIC steam line isolation valves are automatically closed. The steam line low pressure function is provided so that in the event a gross rupture of the RCIC steam line occurred upstream from the high flow sensing location, thus negating the high flow indicating function, isolation would be effected on low pressure. The iso-

3.2.E.3. Reactor Pressure Low (Continued)

jection valves. The valves do not open, however, until reactor pressure falls below the discharge head of LPCI.

4. Reactor Water Level (Shroud Level Indicator)

A reactor water level ≥313.5 inches above vessel invert is indicative that LPCI has made progress in reflooding the core. A simultaneous high drywell pressure trip indicates the need for containment cooling. The ≥ 313.5 inch setpoint acts as a permissive for manual diversion for some of the LPCI flow to containment spray.

5. LPCI Cross Connect Valve Open Annunciator

With the modified LPCI arrangement, the cross connect valve status was changed from normally open to normally closed. Inadvertent opening of this valve could negate the LPCI system injection when needed. The annunciator will alrm when the LPCI cross connect valve is not fully closed.

6. RHR (LPCI) Pump Discharge Pressure Interlocks

A pressure ≥ 100 psig on the RHR pump discharge indicates that the pump has started successfully. The setpoint provides a permissive signal to ADS which allows ADS initiation if other requirements are met.

7. RHR (LPCI) Pump Flow (Δp Switch) Low

A flow switch is provided downsteam of each RHR pump to indicate the condition of each pump. To protect the pumps from overheating at low flow rates a minimum flow bypass line, which routes water from the pump discharge to the suppression chamber, is provided for each pair of pumps. A single motor-operated valve controls the condition of each bypass line. The minimum flow bypass valve automatically opens upon sensing low flow in the discharge lines from both pumps of the associated pump pair. The valve automatically closes whenever the flow from either of the associated main system pumps is above the low flow setting.

8. RHR (LPCI) Pump Start Timers

If normal AC power is available, four pumps automatically start without delay. If normal AC power is not available, one pump starts without delay as soon as power becomes available from the standby sources. The other three pumps start after a 10-second delay. The timer provides correct sequencing of the loads to the diesel generator.

3.2.F.5. Core Spray Pump Discharge Flow

A flow switch is provided downstream of each core spray pump to indicate the condition of each pump. To protect the pumps from overheating at low flow rates a minimum flow bypass line, which routes water from the pump discharge to the suppression chamber, is provided. A single motor-operated valve controls the condition of each bypass line. The minimum flow bypass valve automatically opens upon sensing low flow in the discharge line. The valve automatically closes whenever the flow is above the low flow setting.

6. Core Spray Pump Discharge Interlock

A pressure $\geq$ 100 psig on the core spray pump discharge indicates that the pump has started successfully. The setpoint provides a permissive signal to ADS which allows ADS initiation if other requirements are met.

7. Core Spray Logic Power Failure Monitor

The Core Spray Logic Power Failure Monitor monitors the availability of power to the logic system. In the event of loss of availability of power to the logic system, an alarm is annunciated in the control room.

G. Neutron Monitoring Instrumentation Which Initiates Control Rod Blocks (Table 3.2-7)

These control rod block functions are provided to prevent excessive control rod withdrawal so that MCPR does not decrease to 1.07. The trip logic for this function is 1 out of n: e.g., any trip on one of six APRM's, eight IRM's or four SRM's will result in a rod block.

The minimum instrument channel requirements assure sufficient instrumentation to assure that the single failure criteria is met.

1. SRM

a. Inoperative

This rod block assures that no control rod is withdrawn during low neutron flux level operations unless proper neutron monitoring capability is available, in that all SRM channels are in service or properly bypassed.

b. Not Fully Inserted

Any source range monitor not fully inserted into the core when the SRM count rate level is below the retract permit level will cause a rod block. This assures that no control rod is withdrawn unless all SRM detectors are properly inserted when they must be relied upon to provide the operator with a knowledge of the neutron flux level.

c. Downscale

This rod block assures that no control rod is withdrawn unless the SRM count rate is above the minimum prescribed for low neutron flux level monitoring.

3.2.G.1.d. Upscale

This rod block assures that no control is withdrawn unless the SRM detectors are properly retracted during reactor startup. This setting is selected at the upper end of the range over which the SRM is designed to detect and measure neutron flux.

2. IRM

The trip logic for this function is 1 out of 8; any trip on one of the eight IRM's will result in a rod block. The IRM rod block function provides local as well as gross core protection.

a. Inoperative

This rod block assures that no control rod is withdrawn unless the IRM's are in service.

b. Not Fully Inserted (Refuel and Start & Hot Standby Mode)

This rod block assures that no control rod is withdrawn during low neutron flux level operations unless proper neutron monitoring capability is available in that all IRM detectors are properly located.

c. Downscale

A downscale indication of $\leq$ 5/125 full scale on an IRM is an indication that the instrument has failed or the instrument is not sensitive enough. In either case, the instrument will not respond to changes in control rod motion and thus, control rod motion is prevented. The downscale trip is set at $\geq$ 5/125 full scale. This rod block trip is bypassed when the IRM is on the range 1.

d. High Flux

If the IRM channels are in the worst condition of allowed bypass, the scaling arrangement is such that for unbypassed IRM channels a rod block signal is generated before the detected neutron flux has increased by more than a factor of 10.

3. APRM

The trip logic for this function is 1 out of 6; any trip on one of the six APRM's will result in a rod block. The APRM rod block function provides gross core protection; i.e., limits the gross core power increase from withdrawal of control rods in the normal withdrawal sequence. The trips are set so that MCPR is maintained greater than 1.07 under normal operating conditions.

a. Inoperative

This rod block assures that no control rod is withdrawn unless the APRM's are in service.

3.3.F. Operation with a Limiting Control Rod Pattern

During operation with a Limiting Control Rod Pattern, either:

1. Both RBM channels shall be operable, or

2. Control rod withdrawal shall be blocked, or

3. The operating power level shall be limited so that the MCPR will remain above 1.07 assuming a single error that results in complete withdrawal of any single operable control rod.

4.3.F. Operation with a Limiting Control Rod Pattern

During operation when a Limiting Control Rod Pattern exists, an instrument functional test of the RBM shall be performed prior to withdrawal of the designated rod(s) and daily thereafter.

G. Limiting the Worth of a Control Rod Below 20% Rated Thermal Power

1. Rod Worth Minimizer (RWM)

Whenever the reactor is in the Start & Hot Standby or Run Mode below 20% rated thermal power, the Rod Worth Minimizer shall be operable or a second licensed operator shall verify that the operator at the reactor console is following the control rod program.

G. Limiting the Worth of a Control Rod Below 20% 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 Rod Worth Minimizer to properly fulfill its function shall be verified by the following checks.

a. The correctness of the control rod 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.

3.3.C. Control Rod Drive System

1. Control Rod Drive Coupling Integrity

Limiting Conditions for Operation:

Operability of the control rod drive system requires that the drive be coupled to the control rod. In the analysis of control rod drop accidents it has been assumed that one control rod drive coupling has lost its integrity. To assure that not more than one coupling could be in this condition, it is required that either a drive is coupled to the control rod or the drive is fully inserted and disarmed electrically. This requirement serves to maintain operation within the envelope of conditions by the plant safety analyses.

Surveillance Requirements

Observation of a response from the nuclear instrumentation during an attempt to withdraw a control rod provides an indication that the rod is following the drive. The overtravel position feature provides a positive check on the coupling integrity, for only an uncoupled drive can reach the overtravel position.

2. Scram Insertion Times

Limiting Conditions for Operation:

The control rod drive system is designed to bring the reactor subcritical at a rate fast enough to prevent excessive fuel damage. Analysis of the limiting transient shows that the negative reactivity rates resulting from the scram with the average response of all the drives as given in the specification provide the required protection and MCPR remains greater than 1.07. The limit on the number and pattern of rods permitted to have long scram times is specified to assure that the effect of rods of long scram times are minimized in regard to reactivity insertion rate. Grouping of long scram time rods is prevented by not permitting more than one slow rod in any four rod array. The minimum amount of reactivity to be inserted during a scram is controlled by permitting no operable control rod to have a scram insertion time for 90% insertion greater than 7 seconds.

3.3.F. Operation with a Limiting Control Rod Pattern

Surveillance Requirements:

A limiting control rod pattern is a pattern which results in the core being on a thermal hydraulic limit; i.e., operating on a limiting value for APLHGR, LHGR, or MCPR as defined in Specifications 3.11.A, B, and C. During use of such patterns, it is judged that testing of the RBM system prior to withdrawal of such rods to assure its operability will assure that improper withdrawal does not occur. It is normally the responsibility of the Reactor Engineer to identify these limiting patterns and the designated rods either when the patterns are initially established or as they develop due to the occurrence of inoperable control rods in other than limiting patterns.

G. Limiting the Worth of a Control Rod Below 20% Rated Thermal Power

1. Rod Worth Minimizer (RWM)

Limiting Conditions for Operation:

The Rod Worth Minimizer (RWM) and the Rod Sequence Control System (RSCS) restrict withdrawals and insertions of control rods to pre-specified 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, 14.4.2, and Appendix P of the FSAR, and NEDO-24040.

3.6.G. Reactor Coolant Leakage (Continued)

would grow rapidly. However, the establishment of allowable unidentified leakage greater than that given in Specification 3.6.G on the basis of the data presently available would be premature because of uncertainties associated with the data. For leakage of the order of 5 gpm, as specified in Specification 3.6.G, the experimental and analytical data suggest a reasonable margin of safety that such leakage magnitude would not result from a crack approaching the critical size for rapid propagation (Reference FSAR, Question 10.4.2). Leakage less than the magnitude specified can be detected reasonably in a manner of a few hours utilizing the available leakage detection scheme, and if the origin cannot be determined in a reasonably short time the plant shall be shut down to allow further investigation and corrective action. The total leakage rate consists of all leakage, identified and unidentified which flows to the drywell floor drain and equipment drain sump. The capacity of the drywell floor sump pumps is 100 gpm and the capacity of the drywell equipment sump pumps is also 100 gpm. Removal of 25 gpm from either of these sumps can be accomplished with considerable margin.

H. Relief/Safety Valves

The pressure relief system (relief/safety valves) has been sized to meet the overpressure protection criteria of the ASME Boiler and Pressure Vessel Code, Section III, Nuclear Vessels.

The details of the overpressure protection analysis showing compliance with ASME, Section XII is provided in the FSAR, Appendix M, Summary Technical Report of Reactor Vessel Overpressure Protection. To determine the required steamflow capacity, a parametric study was performed assuming the plant was operating at the turbine-generator design condition of 105 percent rated steam flow (10.6×10^6 pounds per hour) with a vessel dome pressure of 1020 psig, at a reactor thermal power of 2537 Mw, and the reactor experiences the worst pressurization transient. The reanalysis for Reload-2 (NEDO-24078) of the worst overpressure transient, a 3 second closure of all main steam line isolation valves neglecting the direct scram (valve position scram) results in a maximum vessel pressure of 1234 psig if a neutron flux scram is assumed. In addition, the same event was analyzed to determine the number of installed valves which would limit pressure to below the code limit. The results of this analysis show that the eleven installed relief/safety valves are adequate, even if assuming the backup neutron flux scram and provide 141 psi margin.

Generator load rejection from high power without bypass is the most severe transient resulting directly in a nuclear system pressure increase, assuming the turbine trip scram. This event is presented in NEDO-24040. The analysis shows that the peak pressure in the bottom of the vessel is limited to 1192 psig. Peak steam line pressure is 1152 psig, showing adequate protection for this worst abnormal operational transient.

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 each type of fuel as a function of average planar exposure shall not exceed the limiting value shown in Figure 3.11-1, sheets 1 and 2. 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 APLHGR 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.

B. Linear Heat Generation Rate (LHGR)

During power operation, the LHGR as a function of core height shall not exceed the limiting value shown in Figure 3.11-2, sheets 1 and 2. If at any time during 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

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

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

LHGR 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.

C. Minimum Critical Power Ratio (MCPR)

The MCPR limit is specified as a function of fuel average exposure. From BOC3 to 1000 MWD/t before EOC3 the MCPR limit is 1.21 for 7x7, 1.23 for 8x8 and 1.27 for 8x8R fuels. From 1000 MWD/t before EOC3 to EOC3 the MCPR MCPR limit is 1.23 for 7x7 and 1.29 for 8x8 & 8x8R fuels. During power operation, MCPR shall be as above at rated power and flow. 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. For core flows other than rated the MCPR shall be K_f times the MCPR value applicable above, where K_f is as shown in Figure 3.11-3.

D. Reporting Requirements

If any of the limiting values identified in Specifications 3.11.A., B., or C. are exceeded, a Reportable Occurrence report shall be submitted.

If the corrective action is taken, as described, a thirty-day written report will meet the requirements of this specification.

4.11.C. Minimum Critical Power Ratio (MCPR)

MCPR shall be determined daily during reactor power operation at $\geq$ 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.

3.11.B. Linear Heat Generation Rate (LHGR)

This specification assures that the linear heat generation rate in any rod is less than the design linear heat generation if fuel pellet densification is postulated. The power spike penalty specified is based on the analysis presented in Section 3.2.1 of Reference 4 and References 5 and 6, and assumes a linearly increasing variation in axial gaps between core bottom and top, and assures with a 95% confidence, that no more than one fuel rod exceeds the design linear heat generation rate due to power spiking. The LHGR as a function of core height shall be checked daily during reactor operation at $\geq$ 25% power to determine if fuel burnup, or control rod movement has caused changes in power distribution. For LHGR to be a limiting value below 25% rated thermal power, the MTPF would have to be greater than 10 which is precluded by a considerable margin when employing any permissible control rod pattern.

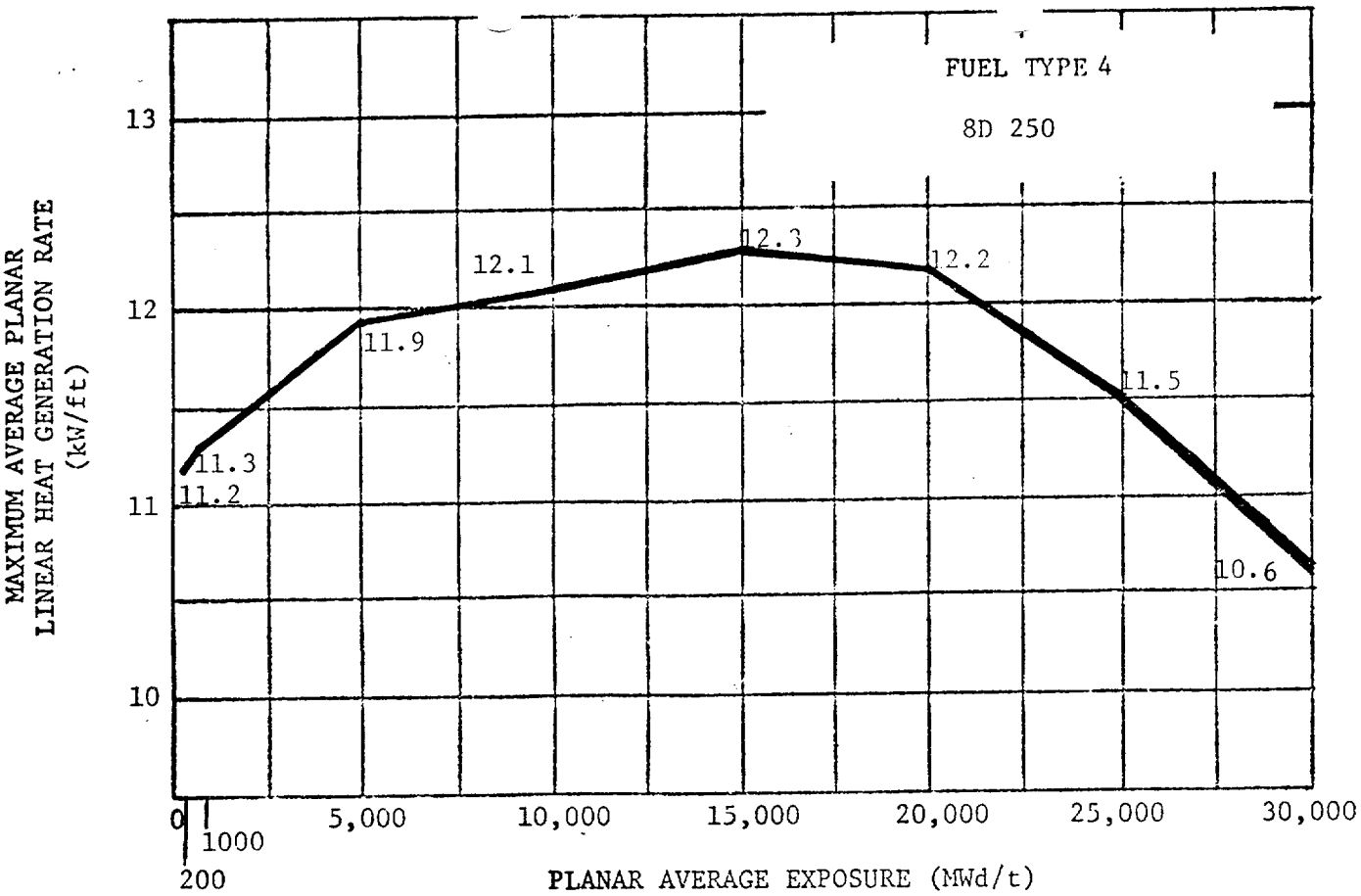
C. Minimum Critical Power Ratio (MCPR)

The required operating limit MCPR as specified in Specification 3.11.C is derived from the established fuel cladding integrity Safety Limit MCPR of 1.07 and an analysis of abnormal operational transients presented in Reference 7.

Various transient events will reduce the MCPR below the operating MCPR. To assure that the fuel cladding integrity safety limit (MCPR of 1.07) 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.

The evaluation of a given transient begins with the system initial parameters shown in Table 6-2 of Reference 9 that are input to a GE core dynamic behavior transient computer program described in Reference 8. Also, the void reactivity coefficients that were input to the transient calculational procedure are based on a new method of calculation termed NEV which provides a better agreement between the calculated and plant instrument power distributions. The outputs of this program along with the initial MCPR form the input for further analyses of the thermally limiting bundle with the single channel transient thermal hydraulic SCAT code described in Reference 1. The principal result of this evaluation is the reduction in MCPR caused by the transient.

From BOC3 to 1000 MWD/t before EOC3 the most limiting transient is the loss of 100°F feedwater heating for 7x7 fuel with a Δ CPR of 0.13. The most limiting transient for 8x8 fuels from BOC3 to 1000 MWD/t before EOC3 is the load rejection without bypass, resulting in a Δ CPR of 0.16. The most limiting transient from 1000 MWD/t before EOC3 to EOC3 is the load rejection without bypass, resulting in a Δ CPR of 0.16 for 7x7 fuel and 0.22 for 8x8 fuels. The worst fuel loading error (FLE), consisting of placing a fresh 8x8R bundle, misoriented in its 8x8R cell location, results in a MCPR of 1.03 from an initial MCPR of 1.23. A fresh 8x8R in an exposed 7x7 location results in a 1.06 MCPR when starting from an initial MCPR of 1.20. Therefore, the MCPRs specified in 3.11.C are based on the results of the most severe abnormal operational transient and the FLE.



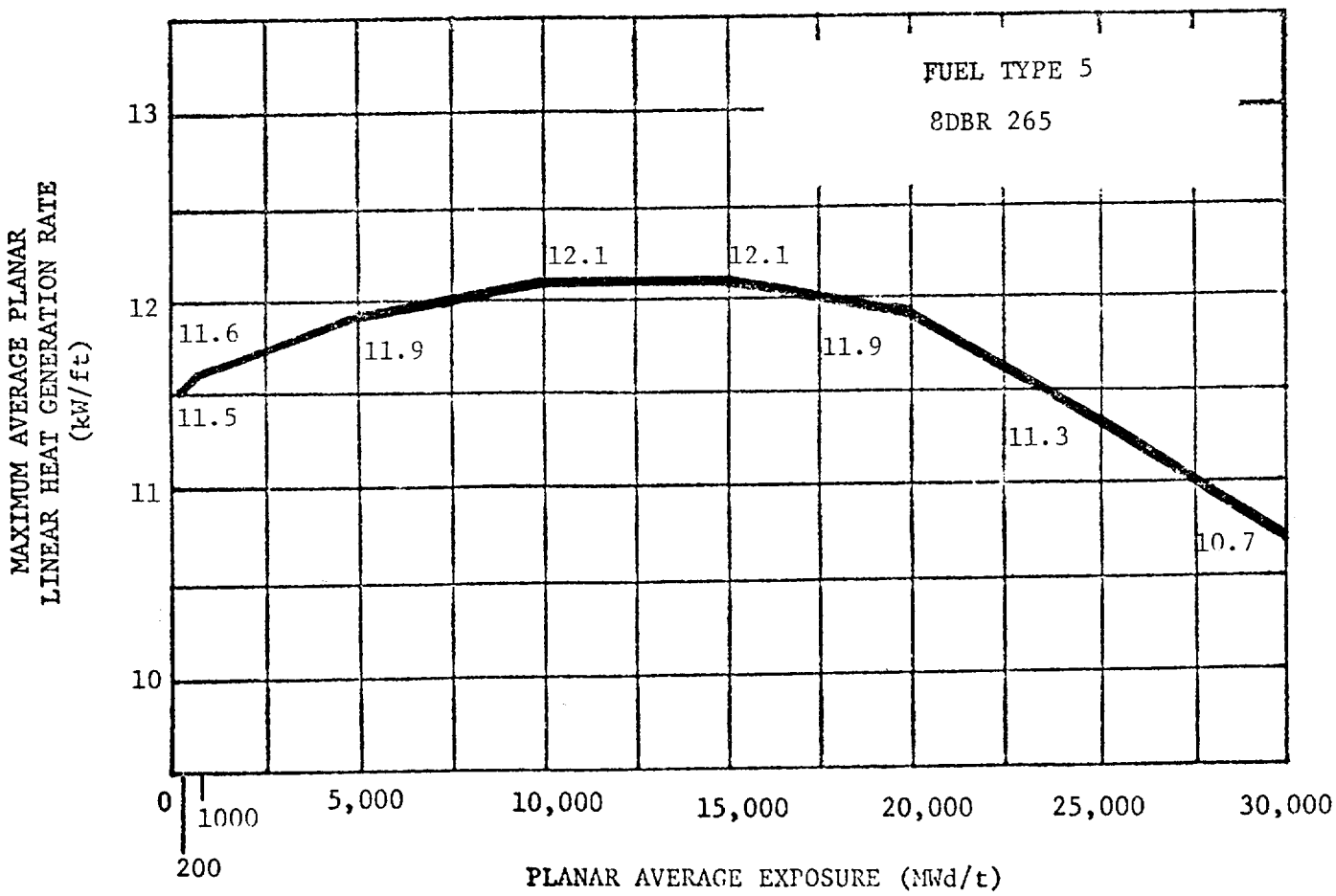


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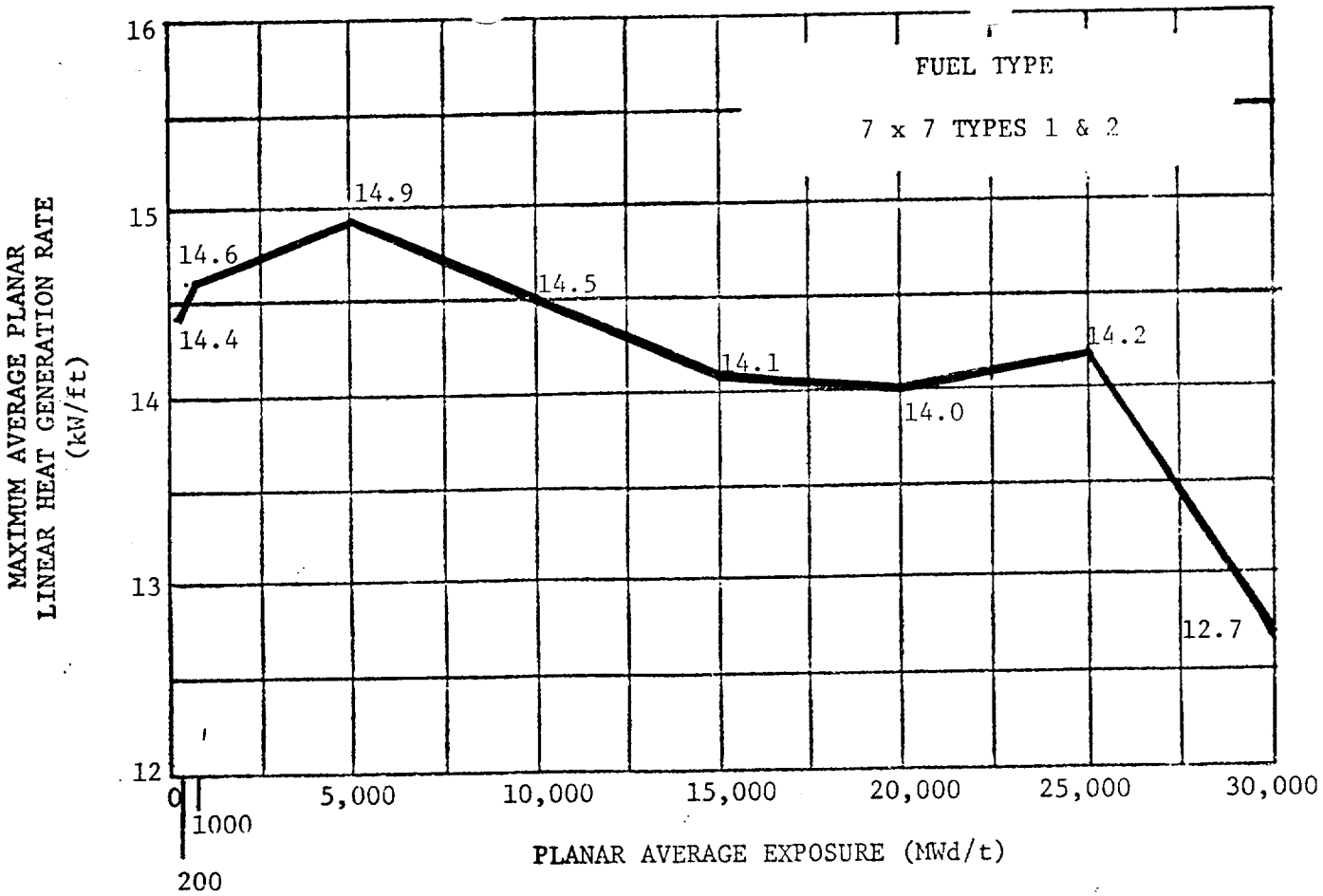
FIGURE 3.11-1 (SHEET 2)



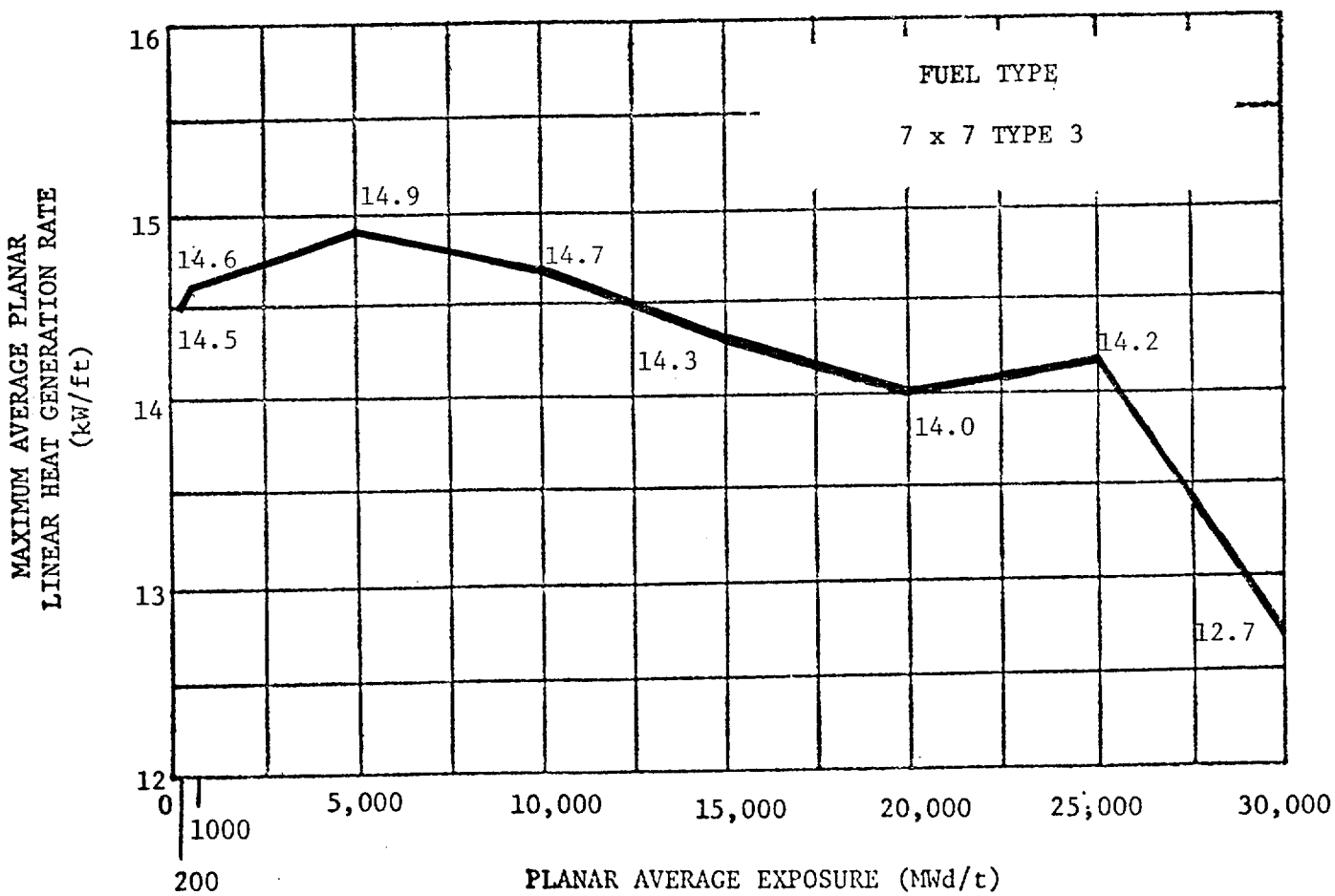


FIGURE 3.11-1 (SHEET 1)

5.0 MAJOR DESIGN FEATURES

A. Site

Edwin I. Hatch Nuclear Plant Unit No. 1 is located on a site of about 2244 acres, which is owned by Georgia Power Company, on the south side of the Altamaha River in Appling County near Baxley, Georgia. The Universal Transverse Mercator Coordinates of the center of the reactor building are: Zone 17R LF 372,935.2m E and 3,533,765.2m N.

B. Reactor Core

1. Fuel Assemblies
The core shall consist of not more than 560 fuel assemblies of the licensed combination of 7x7 bundles which contain 49 fuel rods and 8x8 fuel bundles which contain 62 or 63 fuel rods each.

2. Control Rods
The reactor shall contain 137 cruciform-shaped control rods. The control material shall be boron carbide powder (B_4C) compacted to approximately 70% of its theoretical density.

C. Reactor Vessel

The reactor vessel is described in Table 4.2-2 of the FSAR. The applicable design specifications shall be as listed in Table 4.2-1 of the FSAR.

D. Containment

1. Primary Containment
The principal design parameters and characteristics of the primary containment shall be as given in Table 5.2-1 of the FSAR.

2. Secondary Containment
The secondary containment shall be as described in Section 5.3.3.1 of the FSAR and the applicable codes shall be as given in Section 12.4.4 of the FSAR.

3. Primary Containment Penetrations
Penetrations to the primary containment and piping passing through such penetrations shall be designed in accordance with standards set forth in Section 5.2.3.4 of the FSAR.

E. Fuel Storage

1. Spent Fuel
All arrangements of fuel in the spent fuel storage racks shall be maintained in a subcritical configuration having a k_{eff} not greater than 0.90 for normal conditions and a k_{eff} not greater than 0.95 for abnormal conditions.

2. New Fuel

The new fuel storage vault shall be such that the k_{eff} dry shall not be greater than 0.90 and the k_{eff} flooded shall not be greater than 0.95.

EDWIN I. HATCH NUCLEAR PLANT UNIT NO. 1

RELOAD 2

SAFETY EVALUATION REPORT

EDWIN I. HATCH NUCLEAR PLANT UNIT NO. 1
RELOAD 2
SAFETY EVALUATION REPORT

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

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

SUPPORTING AMENDMENT NO. 52 TO FACILITY OPERATING LICENSE NO. DPR-57

GEORGIA POWER COMPANY

OGLETHORPE ELECTRIC MEMBERSHIP CORPORATION

MUNICIPAL ELECTRIC ASSOCIATION OF GEORGIA

CITY OF DALTON, GEORGIA

EDWIN I. HATCH NUCLEAR PLANT UNIT NO. 1

DOCKET NO. 50-321

1.0 Introduction

By letter dated August 26, 1977 and supplemented and amended by letters dated December 1, 1977, January 5, 1978, February 22, 1978, March 16, 1978, and March 8, 1978, (References 2 through 9), Georgia Power Company (the licensee) requested an amendment to Facility Operating License No. DPR-57. By letter dated January 3, 1978, Georgia Power Company submitted a reevaluation of the Emergency Core Cooling System (ECCS) performance in compliance with our Order for Modification of license dated March 11, 1977. The amendment would modify the Technical Specifications for the Edwin I. Hatch Nuclear Plant, Unit No. 1 (Hatch-1), to: (1) permit operation of the facility during Cycle 3 with up to 168 improved two water rod 8x8R reload fuel bundles, designed and fabricated by the General Electric Company (GE) and having an average enrichment of 2.65 wt/% U-235, and (2) revise the maximum average planar linear heat generation rates (MAPLHGR's) as determined by the reevaluation of the ECCS performance. This licensing action was noticed in the FEDERAL REGISTER on December 28, 1977 (42 FR 64749).

2.0 Background

Georgia Power Company's proposed reload of Hatch Unit No. 1 with 168 GE 8x8R (retrofit) fuel bundles for Cycle 3 (Reload 2) represents the first application of the new General Electric two water rod fuel bundle design on a batch reload basis for an operating BWR. Hatch-1 Reload 1, previously reviewed and approved[10] by the staff,

incorporated 92 single water rod 8x8 fuel bundles as replacement for an equal number of 7x7 fuel bundles discharged from the initial core. The Reload 2 fuel design for Hatch-1 represents a slight modification to GE's previous single water rod 8x8 reload fuel assembly design, currently in operation in 14 domestic BWR's including Hatch-1. The retrofit 8x8 fuel design is essentially identical to the BWR/6 Fuel Design[11] and the Hatch Unit No. 2 initial core fuel design which has already been accepted[12] by the staff for first cycle operation.

The replacement fuel for Reload 2 of Hatch-1 is not the first performance experience for the new two water rod fuel bundle. Four retrofit 8x8 demonstration assemblies[13] have operated successfully in a BWR/4 for at least one cycle[14]. In addition, one lead 8x8R assembly containing several pressurized fuel rods is presently in operation in a second BWR/4[15].

The documentation submitted in support of the proposed reload includes: (1) the GE BWR Reload 2 licensing application for Hatch-1[1] which contains the related fuel design information and a description of the plant unique reload analyses performed, including the analytical methods employed, (2) a supplemental reload licensing submittal[2], which presents the results of the plant unique safety analyses performed for the second reload (except for LOCA analysis results), (3) the Hatch-1 Loss of Coolant Accident analysis results for the new and exposed fuel[4], (4) other supplemental information[6,7,8], and (5) the proposed Technical Specification changes[3,5,9].

3.1 Mechanical Design Evaluation

The licensee has considered[1,2] the adequacy of the thermal-mechanical, structural and chemical design of the reload retrofit 8x8 fuel assembly for all modes of operation of the Hatch-1 plant, including the effects of steady-state and normal operating transients, abnormal operating transients and postulated accident conditions. Our evaluation of the adequacy of the fuel bundle design, as reported in the mechanical design evaluation provided by the licensee, is contained in the following subsections.

3.1.1 Fuel Mechanical Design Description

The Reload 2 assembly design for Hatch-1 is a modified version of the General Electric 8x8 fuel assembly design currently in operation in 14 domestic BWR's. The Hatch-1 reload fuel design is very nearly the same as that described in the BWR/6 Fuel Design and Hatch Unit No. 2 initial core fuel designs,[11] reviewed by the staff for first cycle operation[12]. For identification purposes, the Reload 2 fuel design will be referred to as the "retrofit 8x8," "two water rod 8x8," or simply "8x8R," while the older 8x8 fuel design will be referred to as the "standard 8x8," "one water rod 8x8," or simply "8x8."

For comparison purposes, fuel assembly design parameters for the two fuel types (and the 7x7 design) are given in Table 3.1 herein. Except for the second water rod and the use of natural uranium at the fuel column ends, the design features of the retrofit 8x8 fuel assemblies are the same as those found in the standard 8x8 fuel assemblies currently operating in numerous BWR's. The 8x8 assemblies have exhibited satisfactory performance to-date[16].

As seen in Table 3.1, the 8x8 fuel bundle contains 63 fuel rods and one water rod whereas the 8x8R bundle utilizes 62 fuel rods and two water rods. The two water rods in the 8x8R assembly have a slightly larger diameter than the single water rod used in the 8x8 assembly. The two larger water rods permit improved axial and local power flattening in the 8x8R fuel assembly, compared with both the 7x7 assembly and single water rod 8x8 assembly.

TABLE 3-1

COMPARISON OF FUEL ASSEMBLY DESIGN PARAMETERS

Design Parameter	Fuel Type 7x7	8x8	8x8R
Fueled Rods/Assembly	49	63	62
Active Fuel Length (in.)	144	144	150*
Rod-to-Rod Pitch (in.)	0.738	0.640	0.640
Water/Fuel Ratio (cold)	2.53	2.60	2.75
Cladding O.D. (in.)	0.563	0.493	0.483

Cladding Thickness (in.)	0.037	0.034	0.032
Thickness/Diameter Ratio	0.0657	0.0689	0.0662
Fuel Pellet O.D. (in.)	0.477	0.416	0.410
Pellet/Clad Diametral Gap (mils)	12	9	9
Maximum Linear Heat Generation Rate (Kw/ft)	18.5	13.4	13.4

*Includes 6 inches of natural UO_2 at bottom and top of fuel column

The water rods are capped, hollow, Zircaloy tubes, with small flow holes at the top and bottom ends, to permit controlled coolant flow within the interior of the tubes. One of the water rods axially positions the seven Zircaloy-4 fuel assembly spacer grids. The fuel column of the 8x8R fuel assembly is 6 inches longer than the 144-inch stack length associated with the 8x8 fuel assemblies used for Reload 1. Additionally, several U-235 enrichments are used within each reload fuel assembly to aid in reducing the local power peaking. Gadolinium, a burnable poison, is also used to supplement the rod-to-rod enrichment pattern in the fuel bundle. That is, selected interior fuel rods contain uniformly distributed gadolinium in the form of gadolinia-urania pellets for local power shaping early in life. Gadolinium-bearing fuel rods were first incorporated as a regular design feature of the initial core of Quad Cities Units No. 1 and 2, starting in 1971 and 1972, respectively. Moreover, since 1965, a substantial number of test and production gadolinia-urania rods have been successfully irradiated to appreciable exposures(17).

The combined effects of the additional water rod, longer fuel column, smaller fuel rod diameter, radial enrichment zoning and rods with gadolinia-bearing fuel pellets result in increased operating margins (in more of the fuel rods in the bundle) with respect to the linear power density design limit and maximum fuel temperatures.

The reload 8x8R fuel assemblies also incorporate finger springs, fastened to the lower tie plate, to control coolant flow through the lower tie plate-to-channel bypass flow path. In addition, the Hatch-1 reload assemblies will have two alternate path flow holes drilled in the lower tie plate orifice nozzle.

3.1.2 Materials Properties

The retrofit 8x8 fuel assembly components are fabricated with Zircaloy-2, Zircaloy-4, Type 304 stainless steel, Inconel X and ceramic uranium dioxide and gadolinia. These materials are the same as those used for the design of the standard 8x8 and 7x7 fuel assemblies. A substantial number of reactor-years of operating experience has been accumulated with these materials under BWR core environmental conditions. This experience has shown these materials to be compatible with the BWR environment and to retain their functional capability during reactor operations during the design life of the fuel.

References 1 and 7 provide the materials properties used in the safety analyses associated with the mechanical design of the reload 8x8 fuel bundle. The various properties are the same as those used for the mechanical design of the standard 8x8 fuel assembly.

A 1% plastic strain limit is used as a safety limit for the Zircaloy-2 fuel rod cladding. Below this safety limit, perforation of the cladding, due to overstraining, is not expected to occur. The empirical basis for this stra limit is an estimate of the strain at which an internally pressurized tube reaches plastic instability. GE bases this limit on strain capability of irradiated Zircaloy cladding segments, from fuel rods operated in several BWR's[18]. A 1% cladding plastic strain limit historically has been specified by GE as a fuel integrity safety limit for fuel consequences associated with abnormal operational transients.

We have reviewed the basis for materials properties used in the mechanical design analyses of the retrofit 8x8 fuel assembly and find them to be acceptable.

3.1.3 Fuel Rod Thermal-Mechanical Design

The thermal-mechanical evaluations of the retrofit 8x8 fuel rods are based on a maximum steady-state operating linear heat generation rate (LHGR) of 13.4 Kw/ft. The elastic stress limits for the fuel rod mechanical design, during normal and abnormal operating reactor conditions are based on the stress categories presented in the ASME Boiler and Pressure Vessel Code, Section III. The cladding is also designed to be free-standing during the fuel design lifetime. A fatigue analysis, based on Miner's linear cumulative damage rule, was performed to assure that the cladding will not fail as a result of cumulative fatigue damage. In addition, for abnormal operational transients, a value of 1% plastic strain, discussed in Section 3.1.2, is established as the safety limit, below which damage

due to cladding plastic deformation is not expected to occur. A thermal-mechanical evaluation is performed to determine the equivalent local linear heat generation rate, which is established as the fuel cladding integrity safety limit for abnormal operating transient conditions. Pellet cladding interaction, waterlogging, fretting-corrosion, hydriding and lateral deflection have also been considered in the fuel rod mechanical design. We have reviewed the information provided by the licensee in the above thermal-mechanical design areas. Our evaluation is reported herein.

Cladding Stress Analysis

The elastic stress limits for the fuel rod cladding utilize the Tresca maximum shear stress theory to calculate stress intensities, which are then compared with the stress intensity limits given in Table 2-6 of Reference 19. The maximum shear stress theory for combined stress, as well as the stress limits, were also used by GE for the design of the 7x7 and 8x8 fuel rods. The results of the cladding stress analyses, using the stress models appearing in Table 2-8 of Reference 19, show that the calculated maximum stress intensities are all well within the applicable stress intensity limits, during all normal and abnormal operating conditions. The analyses include load cycles derived from power changes such as those occurring during start-up and shutdown, for both hot and cold conditions. Daily load changes and overpower conditions are also included in the stress evaluations. The stress evaluations incorporate the effects of fuel densification power spiking to substantiate the 13.4 Kw/ft design limit LHGR. On the basis of the information provided by the licensee, including actual BWR operating experience with 7x7, 8x8, and 8x8R fuel assemblies, designed to the above stress intensity limits, the staff finds the fuel rod cladding stress analysis results to be acceptable.

Cladding Collapse Analysis

Cladding collapse potential has been assessed as part of the overall thermal-mechanical design evaluation of the retrofit 8x8R fuel rods. A collapse analysis was performed using the generic methods described in the SAFE-COLAPS Model[20]. This model has been previously approved[21] by the staff. The limiting collapse criteria assumes an instantaneous increase of 160 psi in the hot full power reactor core pressure, due to a turbine trip with bypass failure. This event can occur at any time during the life of the fuel assembly. For Hatch-1, the maximum pressure increase, for the most severe pressurization transient (load rejection with bypass failure) during Cycle 3, is less than 160 psi. Thus, the generic analysis is conservative. Additionally, the analysis includes the effect of fuel densification power spiking on cladding temperature. Finally, cladding

collapse has never been observed in operating BWR fuel rods. The staff, therefore, finds the cladding collapse analysis results to be acceptable for Hatch-1 during Cycle 3.

Fatigue Analysis

The fatigue analysis uses Miner's linear cumulative damage rule[22]. The fuel rod location GE considers subject to the greatest fatigue damage is the fuel rod clad tube-to-end plug weld juncture. The cyclic loads considered in the analysis are coolant pressure and thermal gradients as described in Tables 2-12 and 2-13 of Reference 19. The cyclic loads are reported by GE to be representative of a four-year residence time, at maximum thermal gradients corresponding to beginning of life conditions.

The staff considers the fatigue damage limit, as described by GE, to be adequate[23]. Moreover, the results of the fatigue analysis, using the stress models appearing in Table 2-8 of Reference 19, show that the cumulative fatigue damage is well within the fatigue damage limit.

Fuel Cladding Integrity Safety Limit LHGR

In order to avoid fuel rod rupture, due to excessive cladding strain caused by fuel pellet expansion, GE has established a cladding plastic diametral strain limit of 1%. Using the previously accepted methods for calculating cladding strains, exposure-dependent linear heat generation rates (LHGR's), corresponding to 1% cladding plastic diametral strain were determined by General Electric. The corresponding LHGR's for the UO_2 fuel rods are approximately 25, 23 and 20 Kw/ft at 0, 20,000 and 40,000 Mwd/t, respectively. However, because urania-gadolinia fuel material has a lower thermal conductivity and melting temperature than urania fuel, the LHGR's corresponding to 1% plastic strain for the gadolinia bearing fuel rods in the 8x8R fuel assemblies are lower than the above values.

For the urania-gadolinia fuel rods having the maximum gadolinia loading concentration, the calculated LHGR's corresponding to 1% plastic strain are not less than 22.0, 20.5, and 17.5 Kw/ft for 0, 20,000 and 40,000 Mwd/t, respectively. The above LHGR's, for the maximum gadolinia concentration fuel rods, are thus established as the exposure-dependent fuel cladding integrity safety limit LHGR's for both 8x8 and 8x8R fuel rods. Fuel rods with peak pellet LHGR's below the safety limit LHGR are not expected to exhibit cladding failure due to overstraining, during the most severe abnormal operational transient event.

The adverse effects of fuel densification power spiking have not, however, been directly considered in the establishment of the above LHGR's. Thus, the staff requires that the maximum calculated LHGR's (for each fuel type) for the most severe transient event, be augmented by an amount equal to the densification power spike penalty before comparison with the above limits. On this basis, the above LHGR's are acceptable fuel cladding integrity safety limits for the consequences associated with abnormal operational transients such that no fuel damage is calculated to occur if the limit is not violated. Transient results are presented in Section 3.4.

Waterlogging

Another area of continuing generic review, which is addressed adequately in the Hatch-1 reload submittal, is the potential and consequences of operating with waterlogged fuel rods. We have reviewed the safety aspects of waterlogging failures that could result from pellet cladding interaction (PCI). A survey of the available information, which includes: (1) test results from SPERT and NSRR in Japan and (2) observations of waterlogging failures in commercial reactors, indicates that rupture of a waterlogged fuel rod should not result in failure propagation or significant fuel assembly damage that would affect coolability of the fuel rod assembly. Thus, we agree that the evaluation of waterlogging failures, as presented in the Hatch-1 submittal, is correct and that cladding stress design limits will not be exceeded.

Fretting-Corrosion Wear

Fretting-corrosion wear, due to flow induced fuel rod vibration against the spacer contacts has been considered in the fuel assembly design. The fuel rod vibration and support characteristics of the retrofit 8x8 fuel design are very similar to

the 7x7 and standard 8x8 fuel design. Moreover, the 8x8R fuel assembly will operate in the same Hatch-1 core environment as the 7x7 and 8x8 assemblies. Fuel rod vibration experiments and years of actual reactor operating experience[25] has provided substantial confidence in the adequacy of the BWR fuel design relative to fretting-corrosion wear behavior. Moreover, actual operating experience with lead 8x8R fuel assemblies[14] has shown the fuel to perform adequately relative to fretting wear. In view of the similarity of the 8x8R fuel design to the older GE BWR fuel designs together with the operating conditions to be associated with the 8x8R reload assemblies, the staff finds that the fretting-corrosion wear potential of the reload fuel assemblies to be acceptably low.

Lateral Deflection

Fuel rod lateral deflection, or bowing, has been investigated by GE and considered in the 8x8R fuel assembly design. The deflection limits on the magnitude of fuel rod bowing are based on: (1) cladding stress limits and (2) rod-to-rod and rod-to-channel clearance limits. Thermal-hydraulic tests[26] have demonstrated that a minimum clearance of .060 inches (design clearance is 0.157 inches) is sufficient to ensure a very low probability of local rod overheating caused by a critical heat flux condition. In the GE fuel assembly surveillance programs, more than 2400 peripheral fuel rods have been examined by borescopic techniques. GE studies[26,27,28] show: (1) no observable gross bowing in the standard 8x8 design, (2) very low frequency of minor bowing, (3) calculated deflections within the design limit, and (4) no significant DNB problem at small rod-to-rod and rod-to-channel clearances, based on thermal- hydraulic testing. In view of the above, the staff agrees that there does not appear to be a significant safety concern relative to potential fuel rod lateral deflection, associated with the 8x8R fuel assembly design.

Pellet Cladding Interaction

Pellet cladding interaction (PCI) is addressed in the Hatch-1 reload submittal. Since 1972, General Electric has made changes in the fuel assembly design and has recommended changes in the mode of reactor operation to reduce the incidence of PCI fuel failures. To minimize the potential for pellet ridging, a shorter,

chamfered pellet with no dishing will be used. The 8x8R design also includes a higher annealing temperature for the Zircaloy-2 cladding, to achieve improved uniformity of mechanical properties. In addition to these design changes, General Electric continues to recommend specific operating procedures identified as Preconditioning Interim Operating Management Recommendations (PCIOMR's). Under these procedures, the Hatch-1 reload fuel will be preconditioned for subsequent full power operation and power cycling by first being taken to full power at a slow ramp rate. On this basis and the thermal-mechanical stress and strain evaluations performed for the reload fuel rod design, the staff agrees that the 8x8R fuel rod design will exhibit adequate performance relative to PCI type fuel failure.

Fuel Densification

The Hatch-1 Reload 2 submittal(3) references the GE densification analysis(29) previously approved(30) by the staff. The effects of fuel densification on the fuel rod are to increase the stored energy, increase the linear thermal output and increase the probability of local power spikes from axial gaps.

The primary effects of densification on the fuel rod mechanical design are manifested in the calculation for fuel/cladding gap conductance, cladding collapse time and fuel duty (stress and fatigue evaluations). The approved analytical model incorporates time-dependent gap closure and cladding creepdown for the calculation of gap conductance. The cladding collapse time calculation also includes the effect of local gaps on cladding temperature. Finally, cladding collapse has not been observed in BWR fuels.

More recent densification analyses submitted by GE(31) and approved by the staff(32) have addressed the effects of increased densification in gadolinia-urania fuels. The stored energy effects of increased densification in gadolinia-urania fuels are offset by the significantly lower LHGR in the gadolinia bearing fuel rods compared to the non-gadolinia bearing fuel rods in the bundle. With regard to densification power spiking effects, GE has shown that the offsetting effects of excess thermal expansion and axial heat transfer, not previously taken credit for, more than offsets the adverse spiking effects associated with gadolinia. Thus, the staff finds that fuel densification has been acceptably accounted

for in the mechanical design of the retrofit 8x8 fuel assemblies. Fuel densification effects on transients and accident consequences are addressed separately in Sections 3.4 and 3.5 herein.

Fission Gas Release

The staff has questioned the validity of the fission gas release predictions in vendor thermal-mechanical performance codes, including GEGAP-III[24], at burnups in excess of 20,000 Mwd/t. By letter[33] dated January 18, 1978, the NRC requested that GE revise their fuel performance model to account for burnup enhanced gas release and submit the revised model for staff review within one year. The staff intends to request all licensees to provide a schedule for incorporating burnup enhanced fission gas release into their safety analysis. For Cycle 3, the fresh reload fuel will not achieve burnups at which fission gas release enhancement occurs. Thus, the effect of enhanced fission gas release on safety analyses is not a concern for the 8x8R fuel for Cycle 3. Our concern, relative to the exposed fuel, is being handled generically, as described above.

Operating Experience

The standard 8x8 fuel design is currently in operation in 14 BWR's and a substantial number of fuel bundles (>250) are in their third irradiation cycle. A detailed post-irradiation examination has been performed at Monticello on lead 8x8 test assemblies at the end of their first two cycles and indicates satisfactory performance[24]. Four lead test assemblies of the 8x8R fuel design began operation in Peach Bottom Unit No. 2, in March 1976. These four assemblies were extensively visually examined at the end of one cycle in mid-1977. The examination results have demonstrated that the 8x8R assemblies and channels are in excellent condition for continued operation.[14] These assemblies are presently operating satisfactorily in their second cycle of operation. In addition, one lead 8x8R assembly, containing several pressurized fuel rods, is presently in its first cycle of operation at Peach Bottom Unit No. 3[15].

3.1.4 Fuel Assembly Structural Design

The reload 8x8 fuel assembly is designed to withstand the predicted thermal, pressure and mechanical interaction loadings occurring during handling, startup, normal operation and abnormal operational transients without impairment of functional capability. The fuel assembly is designed

to sustain predicted loadings from an operating basis earthquake. Also, the design-analysis of the fuel assembly shows that the functional capabilities will not be exceeded as a result of a safe shutdown earthquake. The ability of the 8x8R assembly and its components to meet these capabilities is evaluated by (1) analyses based on classical methods and the ASME Boiler and Pressure Vessel Code which are compared against acceptance criteria (design ratios) and (2) testing programs.

The adequacy of the fuel assembly structure during normal operations and normal operating transients is based principally on stress limits and stress formulations which are consistent with the requirements of the ASME Boiler and Pressure Vessel Code Section III. Based on our review of the analysis results provided by the licensees and actual reactor operating experience for the 7x7, 8x8 and lead 8x8R fuel assemblies we find the 8x8R to be structurally adequate for normal operating conditions for the Hatch-1 plant.

The adequacy of the fuel assembly structural design during abnormal operating transients principally relate to the fuel cladding integrity safety limit LHGR and cladding collapse potential. These are evaluated in Section 3.1.3 herein.

The question of the adequacy of the 8x8R fuel assembly structual design during handling, and combined earthquake and LOCA loading conditions is currently being reviewed generically by the staff via topical reports submitted by GE. At present, we have not identified a significant safety concern, however, for this licensing action we have reviewed the capability of the fuel assembly to withstand the control rod drop accident, pipe breaks inside and outside of containment, the fuel handling accident and the recirculation pump seizure accident. These are addressed separately in Section 3.5.

3.2 Nuclear Design Evaluation

Our evaluation of the Cycle 3 core nuclear design for Hatch-1 consisted of two parts. The first part consisted of a review of the adequacy of the reference neutronics methods, for the analysis of the reload retrofit 8x8 fuel assembly and the Cycle 3 mixed core configuration. The second part addressed the acceptability of the calculated fuel assembly and core nuclear characteristics, applicable to Hatch-1, during the third cycle of operation.

3.2.1 Nuclear Design Methods

The staff has reviewed and evaluated the information presented[1,6] on the nuclear analysis methods. The basic calculational procedures used for generating neutron cross sections are part of General Electric's Lattice Physics Model.[35,36] In this model the many-group fast and resonance energy cross sections are computed by a GAM-type program. The fast groups are treated by integral collision probabilities to account for geometrical effects in fast fission. Resonance energy cross sections are calculated by using the intermediate resonance approximation, with energy and position-dependent Dancoff factors included. The thermal cross sections are computed by a THERMOS-type program. The model accounts for the spatially varying thermal spectrum throughout the fuel bundle. These calculations are performed for an extensive combination of parameters including fuel enrichment and distribution, fuel and moderator temperature, burnup, voids, void history, the presence or absence of adjacent control rods and gadolinia concentration and distribution in the fuel rods. As part of the Lattice Physics Model, three-group two-dimensional XY diffusion calculations for one or four fuel bundles are performed. In this way, local fuel rod powers can be calculated as well as single bundle or four bundle (with or without a control rod present) average cross sections. The General Electric Company has submitted two licensing topical reports[35,36] which describe in detail and verify the adequacy of the procedures outlined above. The staff has reviewed these reports and has concluded[37] that the methods satisfy its requirements for core physics methods. These methods are considered acceptable for the Cycle 3 Hatch-1 core, incorporating the retrofit 8x8 fuel bundles.

The single or four bundle averaged neutron cross sections which are obtained from the Lattice Physics Model are used in either two or three-dimensional calculations. Two-dimensional XY calculations are usually performed in three-groups at a given axial location to obtain gross power distribution, reactivities and average three-group neutron cross sections for use in one-dimensional axial calculations. The three-dimensional diffusion calculations use one energy group and can couple neutron and thermal-hydraulic phenomena. These three-dimensional calculations are performed using 24 axial nodes and 1 radial node per fuel bundle resulting in about 14,000 to 20,000 spatial nodes. This three-dimensional calculation provides power distributions, void distributions, control rod positions, reactivities, eigenvalues, and average cross sections for use in the one-dimensional

axial calculations. The three-dimensional calculations have been described and verified in two licensing topical reports[38,39] which were submitted by the General Electric Company. The staff has reviewed these reports and has reached the same conclusions[37] as those reached for the Lattice Physics Methods reports. These methods are also considered acceptable for the Hatch-1 Cycle 3 core incorporating the retrofit 8x8 fuel bundles.

The one-dimensional calculation referred to above is a space-time diffusion calculation which is coupled to a single channel thermal-hydraulic model. This axial calculation is used to generate the scram reactivity function for various core operating states. This one-dimensional space-time code has been compared by the General Electric Company with results obtained using the industry standard code, WIGLE, and shown to be conservative. Our consultant, Brookhaven National Laboratories, has performed an extensive study[40] of BWR scram reactivity behavior and has concluded that the end of cycle, all rods out configuration, represents the limiting condition for BWR scram system effectiveness. Thus, we conclude that the method and assumptions used by General Electric to obtain the scram reactivity curves are acceptable.

The Doppler, moderator void and moderator temperature reactivity coefficients are generated in a rudimentary manner from data obtained from the Lattice Physics Model. The effective delayed neutron fraction and the prompt neutron lifetime are computed using the one-dimensional space-time code. The power coefficient is obtained by appropriately combining the moderator void, Doppler, and moderator-temperature reactivity coefficients.

The General Electric Company has submitted a licensing topical report[41] describing the methods used for the generation of void and Doppler reactivity feedback for application to BWR design. The staff is currently reviewing this report. Based on our review to-date we find the methods used by General Electric to be acceptable for the reasons which follow. Comparisons between calculated and measured local and gross power distributions have been presented by the General Electric Company in two topical reports[36,39]. Predicted local (intra-bundle) power distributions were compared to data obtained from critical

experiments and from gamma scans performed on operating plants. Gross radial and axial power distributions obtained from operating plants have been compared with values predicted by the BWR Simulator code. These comparisons have yielded values for calculational uncertainties to be applied to power distributions. Comparisons have also been made of calculated values of cold, xenon-free reactivity and hot operating reactivity of a number of operating reactors as a function of cycle exposure. These comparisons have been used to establish shutdown reactivity requirements.

The staff has reviewed the two topical reports and found them acceptable[37] for reference in licensing actions.

3.2.2 Nuclear Characteristics

Introduction

The reference core design[2] for Hatch-1 Cycle 3 utilizes 168 fresh retrofit 8x8 reload fuel bundles with a bundle average U-235 enrichment of 2.65 wt %, to replace 168 exposed 7x7 fuel assemblies from the initial core. The Hatch-1 core contains a total of 560 fuel bundles. Thus about 30% of the fuel bundles are being replaced for this reload. The loading pattern for Cycle 3 results in a symmetrical scatter loading of the Reload 2 assemblies within the core

The reload 8x8R fuel assemblies have a total active fueled length of 150 inches. This compares with a 144-inch long pellet column incorporated in the design of the initial core 7x7 fuel bundles and the first reload 8x8 bundles. The top six inches and bottom six inches of the fuel column of the retrofit fuel assembly consists of fuel pellets with natural uranium enrichment. The remaining central 138 inches contain fuel pellets of 2.82 wt % U-235. This arrangement results in a bundle average enrichment of 2.65 wt % U-235 and a lattice average enrichment (used for nuclear analysis) of 2.82 wt % U-235. The reload bundles will also incorporate several fuel rods containing enriched gadolinia as a burnable poison for local power shaping early in the cycle.

The retrofit 8x8 fuel bundles incorporate two unfueled water rods symmetrically placed on either side of the lattice diagonal. This compares with a single water rod in the Reload 1 standard 8x8 fuel bundle design. Each of the two water rods is also slightly larger than the water rod used in the 8x8 bundles. In addition, the fuel rod outside diameter (and pellet diameter) has been decreased by 10 (and 6 mils). The effect of the two larger water rods together with the smaller pellet diameter has resulted in a decrease in the fuel to water ratio.

Power Distribution

The limits on power distribution for this reload of Hatch-1 are determined by specified acceptable fuel design limits (SAFDL) and by the accident and transient analyses. These limits are reflected in the Technical Specifications as limits on the linear heat generation rate (LHGR), average planar linear heat generation rate (APLHGR), critical power ratio (CPR) and the total peaking factor (TPF). The criterion used for the review of power distributions is that these limits are assured during normal operation. For Hatch-1 this criterion is met by monitoring the gross radial and axial power distributions and by pre-calculating the local power distributions.

The licensee has conservatively calculated the local power distributions and local peaking factors, by the methods described in Section 3.2.1, over the range of exposure for Cycle 3 for the 7x7, 8x8, and 8x8R designs. The retrofit 8x8 fuel design will have a lower operating local peaking factor than would a fresh standard 8x8 bundle, but not necessarily as low as an exposed 8x8 bundle of the older design. The licensee has conservatively accounted for this by assuming a local peaking factor of 1.17 which bounds both 8x8 designs for the entire third cycle. The 7x7 assemblies were assumed to have a local peaking factor of 1.27. These local peaking factors are acceptable.

Gross power distributions (radial and axial) will be monitored by periodic TIP scans, which will be kept updated between scans by means of the LPRM detectors. This basic method is unchanged by the reload and new fuel design. The reload fuel will have a different void and axial power distribution than the older designs, due to the additional liquid water contained in the two larger water rods and the natural uranium at the fuel column ends. The calculational method used to transform detector signal to flux and power, evaluated in Reference 42, tracks ^{235}U, ^{239}Pu and ^{240}Pu in six inch segments for each assembly. An iterative technique is used to obtain self-consistent axial power and void distributions. This calculational method will measure the axial power distribution in individual bundles of the new and older designs in an acceptable manner.

Although the gross, radial and axial distributions will change due to the change in void feedback (and radial self-flattening), this is not a major effect. Since incore methods are used to monitor APLHGR, LHGR, CPR, AND TPF, the changes in gross distribution are not of safety significance.

Reactivity Coefficients

Limits on reactivity coefficients are set by the transient and accident analyses, stability analysis and General Design Criterion 11, which requires that, in the power operating range, the net effect of the prompt inherent nuclear feedback characteristics tend to compensate for rapid increases in reactivity. In the Hatch-1 core, the coolant is nearly isothermal at power operating conditions, and the only significant independent coefficients of reactivity are the Doppler coefficient and the void coefficient. During startup, there is also a moderator temperature coefficient.

Because the Doppler coefficient is least negative for unexposed (plutonium-free) fuel, the "worst case," least negative condition for the Doppler coefficient is at BOC. Because the new 8x8 fuel

design has a different water to fuel ratio, void distribution, Dancoff factor and pin self-shielding, the behavior of the Doppler coefficient as a function of void and exposure is somewhat different than that of the standard 8x8 fuel design. The overall value remains negative under all conditions, thus meeting the requirements of GDC 11. The licensee has considered these effects on the Doppler coefficients, in the accident and transient analyses, in an acceptable manner. This is discussed further in Sections 3.4 and 3.5 herein.

The void coefficient is the dominant reactivity feedback coefficient. It will always be strongly negative, under all conditions encountered during Cycle 3 reactor operation. The accident and transient analyses place lower, as well as upper limits on the algebraic value of the void coefficient, depending on whether power increase or decrease events are being considered. The effect of the extra water rod in the 8x8R fuel design is to reduce the absolute magnitude of the void coefficient. The licensee has calculated the void coefficient for the individual fuel types, and for the entire core, as a function of exposure, and has selected a most negative bounding value in a given exposure interval for use in the transient analyses. The calculated void coefficient is increased in absolute magnitude by the application of a 1.25 design conservatism factor when used for core wide transient analyses. This is an acceptable, conservative approach.

The licensee has not calculated a moderator temperature coefficient for this cycle. This coefficient can become slightly positive during certain conditions. However, this coefficient is important only during startup and shutdown, is very slowly acting, and is overshadowed by the Doppler coefficient. Because of this, and because no credit is taken for the moderator temperature coefficient in the safety analyses, the coefficient has no direct safety significance. Therefore, the staff finds it acceptable to exclude the moderator temperature coefficient from the safety analysis.

Shutdown Capability

Shutdown margin, reactivity control systems, and scram reactivity fall under General Design Criteria 20 through 29. When applied to this reload, these Criteria reduce to the following requirements:

- The control rods must be capable of rendering the core sub-critical in a cold, xenon-free condition, at any time in the cycle, with the highest worth control rod stuck out of the core.

- The shutdown margin and scram reactivity curve must be consistent with the accident and transient analyses.

- The Standby Liquid Control System (SLCS) must be capable of rendering the core subcritical, in a cold, xenon-free condition, with the control rods at their minimum position, at any time in the cycle.

The retrofit 8x8 fuel bundles incorporate the use of small amounts of gadolinia as a bundle poison. With burnable poison in the reload core, fuel reactivity initially decreases, as samarium builds in, then increases to a peak as the burnable poison burns out, then finally decreases monotonically until EOC, as fissile nuclide depletion becomes dominant. Thus, the point of maximum core reactivity is generally not at BOC, but occurs later in the cycle. This burnable poison depletion effect causes some control cells to increase in worth, while others may decrease, thus causing the location of the strongest rod to change. The licensee has calculated the effective multiplication factor in a core configuration with the strongest control rod out, under a cold, xenon-free condition. This calculation gives shutdown margin directly. The calculations were done for various exposures during the cycle, and a search for the strongest control rod was done at each exposure. To ensure conservatism, the minimum expected exposure for the previous cycle was assumed in the depletion calculations. The information presented in Reference 2 shows for Cycle 3, the minimum shutdown margin is 1.1% ΔK. Therefore, the shutdown margin of the reconstituted core meets the Technical Specification requirement that the core be at least 0.38% ΔK subcritical.

The licensee has calculated the scram reactivity versus time (scram curve) for both EOC-1000 MWd/t and EOC conditions. This scram curve, with a 0.80 multiplier for model uncertainty and error allowance, is used in the transient analyses. Rod insertion times assumed in the calculation were the slowest allowed by the

Hatch-1 Technical Specifications. These conditions are conservative for earlier exposures, because of the decrease in rod density as the EOC condition is approached. That is, at EOC, there are less partially inserted rods which insert reactivity more quickly than the fully withdrawn rods. The power distribution used in the calculation at each exposure is based upon the Haling axial power and exposure distributions. Since actual EOC power distributions are generally more bottom-peaked than the Haling distribution prediction, this calculation is considered conservative. Therefore, the calculation of the scram curve is acceptable, and the second of the three requirements is satisfied.

The licensee has calculated the multiplication factor and shutdown margin for a 600 ppm sodium pentaborate concentration in the coolant corresponding to the Technical Specifications basis for the SLCS. Calculations were for an exposure corresponding to the maximum fuel reactivity, with the core in a cold, xenon-free state. Additionally, control rods were assumed to be out of the core. The results show the SLCS capable of rendering the core sub-critical by at least .033 ΔK for these conditions. The third requirement is satisfied for the alternate shutdown system. Thus General Design Criteria 20 through 29 are satisfied.

Control Rod Patterns and Reactivity Worths

The limits on control rod worth originate in the accident and transient analyses. Additionally, it is required that reactivity additions resulting from a single control rod notch should not result in a period which the operator cannot safely control. However, the maximum worth of one notch has never been excessive in the power range for an operating BWR.

The rod drop accident requires limits on dropped rod worth during startup, and the inadvertent rod withdrawal transient requires limits on individual rod worth during power operation. During startup, the maximum dropped rod worth is limited by limiting the possible control rod withdrawal patterns. The patterns are enforced by the Rod Sequence Control System (RSCS). This pattern restriction is independent of fuel type, and remains acceptable.

During power operation, the voided condition of the moderator greatly reduces the worth of a dropped rod, and the rod drop accident consequences are not limiting. Therefore, above 20% power, the RSCS is automatically disengaged and there is no safety-related systems to control rod patterns. Further discussion may be found in the evaluation of the Control Rod Drop Accident appearing in Section 3.5.4 herein.

The limits on rod worth resulting from the analysis of the rod withdrawal error transient are enforced by means of the Rod Block Monitor System (RBM). When a control rod is selected for movement, the nearest LPRM detectors are automatically monitored, and a rod block is effected when the local power increase reaches the RBM setpoint. Thus, the RBM restricts the control rod worth through the local power coefficient, rather than via control rod patterns. This system is also independent of fuel type, and remains acceptable. Further discussion is provided in the evaluation of the Rod Withdrawal Error transient in Section 3.4 herein.

Reactivity of Fuel in Storage

The Technical Specification requirement for the storage of fuel in the Hatch-1 spent fuel storage pool is that the effective multiplication factor, K_{eff}, of the fuel, as stored in the fuel storage racks, is less than 0.90 for normal storage conditions. This requirement is met if the uncontrolled infinite lattice multiplication factor, K_∞ , of a 8x8R fuel bundle in the reactor core configuration is less than or equal to 1.30. The 8DRB265 reload fuel bundle, at the peak reactivity point, at 65°C, has a maximum K_∞ of 1.184 for the enriched UO_2 fuel zone and 0.8810 for the naturally enriched UO_2 at the ends of the fuel column. Thus, the reload fuel meets the Technical Specification fuel storage subcriticality requirements.

3.3 Thermal and Hydraulic Design Evaluation

Our review of the thermal-hydraulic design for Cycle 3 of Hatch-1 consisted of two parts. The first part addressed the applicability of the referenced and described thermal and hydraulic models and methods(1,6), for the analysis of the Hatch-1 Cycle 3 core. Since the reconstituted core incorporates three different fuel bundle types, i.e., 7x7, 8x8 and 8x8R, the applicability of the thermal and hydraulic methods to the new retrofit 8x8 fuel bundle design was reviewed, along with a review of the adequacy of the methods for mixed cores. The second part consisted of a review of the thermal-hydraulic analysis results. The results for Hatch-1 included the statistical determination of a new fuel cladding integrity safety limit MCPR for the reconstituted core and the channel and reactor core stability decay ratios.

3.3.1 Steady-State Hydraulic Methods

The core steady-state hydraulic analysis is performed to establish flow, pressure, enthalpy, void, and quality distributions within the core. This analysis also establishes initial reactor coolant conditions for reactor physics calculations and the analysis of anticipated operational transients. The hydraulic model of the reactor core includes descriptions of the orifices, lower tie plates, fuel rods, fuel assembly spacers, upper tie plates, fuel channels and core bypass flow paths. The core steady-state hydraulic model is composed of separate effects models, which simulate various pressure loss characteristics, and composite models, which simulate the channel-by-channel and core bypass flow paths.

The separate effects hydraulic models of the core and channel components consider frictional, local, elevation, and acceleration hydraulic pressure loss characteristics. The frictional characteristics of the core components are modeled by use of the single phase frictional pressure drop equation with a two-phase friction multiplier. The use of this equation and multiplier requires correlations for the friction factor, f, and two-phase multiplier, ϕ^2_{TPF}. GE has correlated these multipliers, on a best-estimate basis, to a significant amount of multi-rod geometry data[43], that are representative of modern BWR fuel bundles. The largest collection of these data was acquired from the ATLAS loop during development testing for the GEXL correlation. The data for these correlations cover the range of BWR conditions. On this basis, the use of these correlations is appropriate.

The local pressure drop characteristics have been established in a manner similar to the formulation used for the frictional pressure drop characteristics. It differs to the extent that a local pressure loss coefficient is substituted for the product of friction factor and characteristic length-to-diameter ratio. This is a common hydraulic analysis procedure. This modeling has also been verified[44] experimentally throughout the

range of conditions by the ATLAS loop tests for the GEXL correlation[43]. This modeling technique is used to simulate the pressure losses of the orifice, lower tie plate, spacer, upper tie plate, and lower tie plate bypass flow holes.

The acceleration pressure drop has two components, i.e., area change and density variation. The area change is modeled similar to the local pressure drop. Since an area change is generally treated in this manner, this modeling approach is acceptable. The density variation uses the same formulation as the elevation pressure drop characteristic, except that it accounts for density variations along the fluid channel. This is also a standard hydraulic analysis practice, and is acceptable.

These separate effects hydraulic characteristics are utilized to simulate the hydraulic conditions through the orifices, lower tie plates, fuel rods, water rods, fuel rod spacers, upper tie plate and fuel channel. The core bypass flow paths have been modeled from experimental[45] results and verified by analytical techniques. These tests were previously reviewed and were found to be acceptable for this use.[46]

The above separate effects hydraulic models, which simulate reactor core component pressure losses and flow paths, permit a composite model of a single fuel channel to be simulated. The fuel channel is then categorized into a fuel "channel type." In order to reduce the number of nodes in the analysis, the fuel channels are grouped by "channel type" and modeled as a single typical channel of that type. Thus, the flow distribution of a particular fuel channel is assumed to be the same as the typical channel for that fuel channel type.

A channel type is classified by five characteristics: (1) orificing type (central or peripheral), (2) fuel geometry (7x7, 8x8, or 8x8R), (3) relative bundle power (high power or average), (4) lower tie plate type (drilled or undrilled), and (5) bypass type (finger springs or no finger springs).

with regard to the core relative bundle power distribution, sensitivity studies show[44] that classification by high power and average power density channels adequately models the core flow distribution. This is due to the fact that average channel characteristics are dominant in establishing the core pressure drop. Therefore, categorization as a function of channel power density need not be broken down into additional sub-channels. The other characteristics completely cover the range of channel type possibilities.

In order to perform channel type categorizations, each fuel channel must have the same pressure drop across its length. This is a major assumption of the steady-state hydraulic analysis. This has been shown to be valid by flow distribution and pressure drop measurements in several operating BWR's[47,48,49]. These tests further show that the pressure drop across any fuel channel or bypass flow path in the core is the same as for any other fuel channel or bypass flow path in the core. The above referenced documents have been previously accepted[50] for justification of this assumption.

The steady-state hydraulic analysis uses a digital computer code to calculate the hydraulic characteristics of the core. The code utilizes a trial and error iteration for flow rate, pressure drop, enthalpy, quality, and void distribution for each channel type. It equates the total plenum-to-plenum differential pressure across each flow path, and matches the sum of the flows to the total core flow. Comparison[44] of analytical predictions to tests performed in the ATLAS test facility as a function of pressure drop, mass flux, and bundle power show reasonably good agreement, i.e., <6% error for the range of interest. This qualifies the calculational technique and modeling for the steady-state hydraulic analysis methods for reactor pressures greater than 800 psia and core flow greater than 10%.

3.3.2.1 Fuel Cladding Integrity Safety Limit MCPR

General Design Criterion 10 requires that the reactor core be designed with appropriate margin, to assure that specified acceptable fuel design limits are not exceeded during any condition of normal operation, including the effects of abnormal operational transients. In order to avoid fuel damage caused by overheating of the cladding, transients are limited such that more than 99.9% of the fuel rods would be expected

to avoid boiling transition during a transient event. This design basis has been previously accepted[51] for initial and reload core applications in connection with the staff's review of the General Electric Thermal Analysis Basis (GETAB) method[52]. This design basis can also be stated in terms of a statistically determined Minimum Critical Power Ratio (MCPR) safety limit. The GETAB statistical analysis procedure, including codes, correlations and analytical procedures has also been previously reviewed and approved by the staff in connection with MCPR safety limits established for initial cores and reload core applications. Our review, therefore, centered upon evaluation of the adequacy of the described statistical analysis procedures for the Hatch-1 reload core, which contains three fuel types (7x7, 8x8, and 8x8R) as well as a review of the key inputs to the statistical analysis.

The nominal values of the plant process variables (e.g., core flow, dome pressure) used in the GETAB statistical analysis, are shown in Table A-2 of Reference 3. The values shown in the table correspond to the same previously approved generic 251/764 core selected for the GETAB statistical analysis, for operating BWR's which have reloaded with the standard 8x8 fuel assemblies. Substitution of the retrofit 8x8 reload fuel assemblies in the statistical analysis does not alter our previous conclusion on the acceptability of the generic core process variable parameter values selected.

The generic uncertainties associated with the core process variables, fuel bundle power determination, CHF correlation and fuel assembly manufacturing tolerances, used in the statistical analysis, appear in Table A-1 of Reference 3. The uncertainties are the same or more conservative than those shown in the GETAB report[52]. The only uncertainties in the table which are potentially reload or fuel-dependent are for TIP Readings, R-Factor, GEXL Correlation and Channel Flow Area uncertainties. The standard deviation for the TIP Readings uncertainty is 8.7% whereas the GETAB report uses a 6.3% uncertainty. The latter uncertainty is appropriate for an initial core. The uncertainty increase in TIP uncertainty to 8.7% is a consequence of the increase in the uncertainty in the bundle power measurement of a reload (exposed) core. This uncertainty is also considered to be adequate for the Hatch-1 retrofit 8x8 fuel assemblies. Table A-1 gives an R-Factor uncertainty of 1.6%, which is the same

as that used for 8x8 reloads. The R-Factor uncertainty is derived from the uncertainty in the local power peaking distribution calculation. The addition of a second water rod in the retrofit fuel design is not expected to increase the uncertainty in the power distribution calculation, based on the approved neutronics methods. The 3.0% Channel Flow Area uncertainty, shown in the table, accounts for manufacturing and operationally induced variations in the free flow area within the assembly. Although the effective channel flow area for the 8x8R assembly is slightly different than for the 8x8 assembly, the manufacturing tolerances are the same. Thus, a channel flow area uncertainty of 3.0% (which is the same as the 8x8 assembly) is acceptable.

A value of 1.038 was selected for the nominal value for the retrofit 8x8 R-Factor. This compares with 1.098 and 1.100, for the 7x7 and 8x8 assembly R-Factors, used in connection with the first Hatch-1 reload. Reload 1 utilized the single water rod 8x8 fuel design. The core wide bundle histogram, used in the new GETAB statistical analysis for this reload of Hatch-1, appears in Figure A-2 of Reference 3. The CPR histogram is different from the histogram previously used in the statistical analyses of BWR 2/3/4 D-Lattice 8x8 reload cores. The new histogram indicates fewer bundles at and near the MCPR safety limit. The licensee was requested to provide additional justification to support the new retrofit 8x8 R-Factor and CPR histogram which were used in the analysis.

The additional information[7] submitted by the licensee states that the lower bundle R-Factor results from the flatter local power distribution of the 8x8R fuel design. A flatter power distribution also gives rise to a more adverse rod-by-rod critical heat flux (CHF) probability distribution and thus is more conservative relative to the number of rods calculated to be in boiling transition when the hot bundle is placed on the thermal MCPR (safety) limit. The CPR histogram used in the calculation corresponds to an all 8x8R (equilibrium cycle) reload 251/764 core. This yields the flattest bundle CPR histogram compared to non-equilibrium cycles. This also results in an adverse CHF accounting when compared to the actual or expected CPR histogram for the Hatch-1 core during Cycle 3. The staff concludes that the R-Factor and bundle CPR distribution selected for the GETAB statistical analysis are appropriately conservative for Cycle 3.

The derived MCPR safety limit for Cycle 3, using the approved GETAB statistical methods and the inputs discussed above, is 1.07. This is an increase of .01 from the 1.06 safety limit applicable during Cycle 2 and increase of .02 from the 1.05 safety limit applicable during Cycle 1. On the basis of the evaluation above, the staff finds the calculated 1.07 safety limit MCPR to be acceptable for Hatch-1 during Cycle 3.

3.3.2.2 Thermal-Hydraulic Stability

A Cycle 3 thermal-hydraulic stability analysis, using the analytical methods discussed in Reference 1, was presented by the licensee for Hatch-1. The results show that the 7x7, 8x8 and 8x8R channel hydrodynamic stability decay ratios at the least stable reactor operating state are substantially below the 1.0 Ultimate Performance Limit decay ratio proposed by GE in Reference 1. The least stable reactor operating state for Hatch-1 corresponds to the intercept point of the 105% rod line and the natural circulation curve appearing in the plant's power flow map. The licensee has also submitted the results of the Cycle 3 reactor core thermal-hydraulic stability analysis for the least stable operating state. The results of this analysis show that the reactor core stability decay ratio is also well within the 1.0 Ultimate Performance Limit decay ratio proposed by GE.

The staff has expressed generic concerns regarding reactor core thermal-hydraulic stability at the least stable reactor condition. This condition could be reached during an operational transient from high power if the plant were to sustain a trip of both recirculation pumps without a reactor trip. The concerns are motivated by increasing decay ratios as equilibrium fuel cycles are approached and as reload fuel designs change. The staff concerns relate to both the consequences of operating at a decay ratio of 1.0 and the capability of the analytical methods to accurately predict decay ratios.

The General Electric Company is addressing these staff concerns through meetings, topical reports and a stability test program. The participants of the on-going stability test program include GE and the licensee of a large BWR/4. Although a final

test report has not as yet been received by the staff, it is expected that the test results will aid considerably in resolving the staff concerns. As an interim measure, the staff, for Cycle 2, imposed a requirement on Hatch-1 which restricted planned operation in the natural circulation mode. This restriction will continue during cycle 3 and will provide a significant increase in the reactor core stability margins. On the basis of the foregoing, the staff considers the thermal-hydraulic stability of Hatch-1 during Cycle 3 to be acceptable.

3.4 Abnormal Operational Transients Evaluation

Abnormal operational transients are plant system conditions, caused by a single operator error or a single equipment malfunction, which are expected to occur one or more times during the life of the nuclear plant unit. Safety (SAFDL) limits applicable for transients include the fuel cladding integrity safety limit MCPR for the 8x8R reload core and the fuel cladding integrity 1% plastic cladding strain (LHGR) safety limit for the fresh and exposed fuel designs and the reactor coolant pressure boundary (RCPB) pressure safety limit.

Our evaluation of a 1.07 safety limit MCPR for Hatch-1 during Cycle 3 is provided in Section 3.3.2.1. The LHGR safety limit for the retrofit 8x8, standard 8x8 and 7x7 fuel rod types is evaluated in Section 3.1.3, herein. With regard to the RCPB pressure safety limit, the maximum reactor coolant pressure achieved during the most severe abnormal operational transient is bounded by the limiting overpressurization event (MSIV closure with an indirect high flux scram) evaluated in Section 3.6 herein. The reactor vessel pressure safety limit applicable to Hatch-1 during abnormal operational transients is that permitted by the ASME Boiler and Pressure Vessel Code Section III, Class 1, which permits pressure transients up to 10% above the reactor vessel design pressure. Since the design pressure of the RCPB is 1250 psig, the pressure safety limit for both abnormal operational transients and the limiting overpressurization event is 1375 psig.

We have reviewed both the methods used for simulating the fuel, core and plant system performance during transients, along with the acceptability of the calculated transient results relative to the above safety limits. Our review of the transient methods was limited to an evaluation of the applicability and adequacy of the described and referenced transient codes, correlations and analytical procedures for the Hatch-1, Cycle 3 core and the new retrofit 8x8 fuel design. Our evaluation of the transient methods is reported in Section 3.4.1. The results of the transient analyses for Cycle 3 of Hatch-1 is evaluated in Section 3.4.2.

3.4.1 Transient Analysis Methods

3.4.1.1 Transient Analysis Methods for Local Events - Rod Withdrawal Error

The control rod withdrawal error is an abnormal operational transient which effects only a limited number of fuel assemblies in the core. The local and radial peaking factors increase substantially in the fuel assemblies in the immediate vicinity of the withdrawn control rod. Thus, this transient is of safety concern, with regard to potential fuel rod overheating (MCPR) and clad overstraining (1% plastic strain). Since the rate and magnitude of the gross core power increase from this event is low, the reactor pressure increase is not large enough to be of concern relative to the RCPB pressure safety limit.

The method used to calculate the consequences of this transient involves a series of steady-state calculations. The simulated core is assumed to be at its most reactive exposure, with no xenon or samarium present. The rod pattern is chosen such that the maximum worth control rod is fully inserted and the laterally adjacent or diagonally adjacent bundles are at their thermal operating limits. A series of steady-state calculations is then performed for succeeding positions of the worst case control rod using the BWR Simulator Code which calculates the response actions of the Rod Block Monitor (assuming the most adverse detector failure allowed by the Technical Specifications). The results are then used to select a setpoint for the Rod Block Monitor such that the two fuel integrity safety limits are not violated.

This procedure of using a series of steady-state calculations to approximate the transients' behavior is the standard analysis method for all GE BWR reloads. Past analyses and reviews have shown that, even at the maximum control rod drive withdrawal speed and rod worth, the rate of power increase is small, and thus a quasi-static approximation (in the power range) is valid. Because the new 8x8R fuel rod has a faster thermal time constant than the older types, and because the codes assume homogenized bundles, both the quasi-static procedure and the codes remain acceptable.

3.4.1.2 Transient Methods for Core Wide Events

Abnormal operational transients which effect the entire core are of safety concern only with regard to fuel rod overheating (CPR) and RCPB overpressurization considerations. Local (intra-assembly) peaking factors during core wide transients remain relatively low and essentially unchanged from normal operating values. Thus, local LHGR's do not closely approach the safety limit LHGR during such occurrences and are not a safety concern for initial or reload cores.

GETAB-SCAT Code Analysis

GE uses a framework of codes for predicting the hot bundle transient critical power ratio during core wide transient events. This framework has been consistently used by GE for initial and reload core licensing applications.

GETAB Transient Analysis

The central code in the GETAB transient analysis is the SCAT code[53], which incorporates the GEXL correlation[52] for predicting the change in bundle critical power ratio (CPR) during the transient. The SCAT code has been previously reviewed and approved by the staff in connection with transient CPR calculations of 7x7 and 8x8 bundles for ECCS Appendix K analyses[65]. The code is also considered to be acceptable for transient analysis applications. The two water rod 8x8R bundle geometry input for the SCAT code analysis does not represent a significant difference from fuel designs previously approved for analyses with the code (i.e., 7x7 or single water rod 8x8). The longer heated length (150 inches vs. 144 inches) and fuel rod diameter changes do not represent calculational difficulty, thus the 8x8R fuel element design is considered to be within the analysis capability of the code to yield conservative estimates of CPR.

The critical bundle power correlation used in the SCAT code analysis is the GEXL correlation. As discussed in Section 3.3.2.1, the GEXL correlation, employing the previously approved R-Factor formulation(51), results in non-conservative predictions of experimental CPR data for certain 8x8R local peaking factor distributions. However, these distributions are not expected to occur during the first operating cycle of the retrofit 8x8 assemblies. Thus, the use of the GEXL correlation for Cycle 3 of Hatch-1 is acceptable. Additional data should be submitted to the staff for review, to justify the conservatism of the GEXL correlation for the second and subsequent cycles of operation of the retrofit 8x8 bundles.

Geometrical differences between the 8x8 and 8x8R fuel designs which can affect the bundle critical power calculation include the heated length, L, and thermal diameter, D_0. The licensee was requested to provide additional information which would justify the acceptability of a single GETAB transient analysis for the two fuel designs for a given core wide transient event. The sensitivity results presented(7) show that there is a ΔCPR difference of approximately 0.001 between the two fuel geometries. Thus, we find it acceptable to perform a single GETAB transient analysis for both fuel types for a particular core wide event.

The effect of fuel densification on SCAT bundle critical power calculations has been considered. GE has presented analyses of the effect of densification power spikes on bundle critical power. These analyses utilized an "Integral Concept"(54). The Integral Concept is widely used and considered to be an acceptable method for quantification of boiling transition correlations. The Integral Concept also requires an empirical base. This base has been found to conservatively represent BWR conditions by comparison with an independently established procedure(55). GE has additionally demonstrated the effect of densification on R-factor and has concluded that the effect is insignificant. Based on the analyses presented, we find that the effects of fuel densification have been appropriately considered in the bundle CPR calculations.

GE develops the SCAT code initial conditions and transient history inputs from the nuclear analysis, core hydraulic analysis and plant system transient analysis. The Hatch-1 inputs which do not vary from cycle to cycle appear in Table 5-3 of Reference 1. The remaining GETAB transient inputs were calculated for Reload 2 for each fuel type. The initial hot bundle flow for each fuel type is determined by the models and methods described in Section 4 of References 1 and 6. These methods are evaluated in Section 3.3.1 herein. The initial integral bundle power and local pin powers are determined by the GE BWR Simulator Code and Lattice Physics Methods, respectively. These codes and methods are evaluated in Section 3.2.1 herein.

Plant System Transient Analysis

GE develops the balance of the required input data for the GETAB transient (SCAT) code analysis from the output of the plant system transient (REDY)[56,57,58] code analysis. The plant system transient results required for each AOT event analyzed by the SCAT code consist of normalized core flow vs. time, reactor core pressure vs. time and core (hot bundle) nuclear power vs. time. These REDY code results are input into the GETAB analysis without modification (no conservatism factors applied to the output). Since safety analysis consequences (i.e., CPR, pressure increase) must be conservatively calculated, this would be an acceptable procedure provided the unmodified REDY code output is already adequately conservative, or provides for an overall adequately conservative CPR methodology. In this regard, the REDY code and related methods are currently under staff review and evaluation in connection with the conservatism afforded by transient predictions.

The REDY code by design is a best-estimate code. GE believes that adequate conservatism exists in the code predictions of plant system transient performance, by way of the conservatism factors applied to key nuclear (core) transient inputs. As seen

in Table 5-2 of Reference 1, GE applies "design conservatism factors" (DCF's) of 0.95, 1.25 and 0.80 to the nominal Doppler, void and scram reactivities predicted by the nuclear analysis. These factors contribute to the currently used licensing basis analysis methods, and are intended to account for non-conservatisms and uncertainties associated with the calculation of the nuclear input parameters and the plant transient analysis models and methods.

Staff concern for the adequacy of the plant system transient methods has been raised by the apparently non-conservative predictions of transient tests recently conducted at a large BWR/4 reactor. The tests involved three end-of-cycle manual turbine trips, initiated from intermediate power levels with the direct (turbine stop valve position switch) reactor trip intentionally disabled. This required the reactor to trip on the indirect (high neutron flux) scram. Several key transient test parameters were underpredicted, even when the present licensing basis plant transient methods (REDY code and DCF's) were employed.

GE has evaluated(59) the differences between the turbine trip test conditions and the licensing basis event (turbine trip without bypass with a direct reactor scram) using a normalized REDY code model as well as a more detailed transient code model. The GE evaluation indicates that a degree of conservatism is available when using the licensing basis methods to predict the consequences of the licensing basis event. The staff agrees with this conclusion. The staff, in the interim, while reviewing another plant system transient code proposed by GE, has concluded that the present plant transient methods adequately predict the consequences of the limiting (licensing basis) core wide events(59).

Several of the plant system transient code models derive their input values from the fuel mechanical design. For example, the multi-noded thermal-hydraulic and heat transfer relationships utilize the fuel rod (fuel and clad) diameters and fuel

column length as inputs. These parameters can, therefore, effect the dynamic behavior of the core via fuel thermal time constant and axial void sweep effects. When a substantial fraction of the core is composed of a mixture of fuel designs, the proper selection of the input values for fuel modeling must be carefully considered. The plant system transient code models heat transfer with a single fuel element representing the entire core. The staff has reviewed GE's analytical procedure for treating these fuel related inputs for the plant transient analysis of mixed cores such as Hatch-1 Cycle 3.

GE's current procedure requires the single fuel element to be the "dominant" fuel type (except for fuel clad gap conductances) rather than a "weighted average." For Hatch-1 this would result in the modeling of a 7x7 fuel element, since the 7x7 bundle is the dominant fuel type during Cycle 3. The 7x7 fuel element has a significantly slower fuel time constant compared with the 8x8 or 8x8R fuel element. Fuel time constant sensitivity studies with the REDY code[56] indicates that a faster fuel time constant results in more severe fuel consequences (e.g., peak heat flux). Thus, the present procedure may be somewhat non-conservative for mixed cores such as Hatch-1. The staff is continuing to evaluate this GE analytical procedure for transient performance of mixed cores. Since the limiting transient event (which develops the operating limit MCPR's) for Hatch-1 during Cycle 3 is not a core wide event (see Fuel Loading Error, Section 3.5.3), our concern pertaining to fuel element modeling for plant transient methods is not a significant concern for Cycle 3 of Hatch-1. The staff is continuing its review of the current GE procedure on a generic basis, however.

GE also uses the REDY code predictions for evaluating conformance with the criteria relating to overpressurization of the reactor coolant system. REDY code simulations of the aforementioned transient tests (using the licensing basis DCF's) demonstrate that the peak transient pressure is consistently overpredicted by the code. The staff has considered the differences between the nature of the turbine trip tests and licensing basis pressurization events (i.e., turbine-generator trip without bypass with direct scram and Main Steam Isolation Valve Closure with indirect high flux scram) and concludes that the code can be expected to also overpredict the peak

transient pressure due to the licensing basis event. The use of the REDY code is, therefore, considered acceptable for RCS overpressurization evaluations for Hatch-1.

The Hatch-1 REDY code input data, relating to pressure relief system characteristics, which do not vary from cycle to cycle appear in Table 5-1 of Reference 1. These characteristics are acceptable.

Exposure-Dependent Operating Limits

The severity of abnormal operational transients is worst at the end of the cycle, primarily because the EOC, all rods out scram gives the least effective scram reactivity response. Operating limits, e.g., MCPR, relief may be obtained by analyzing the transients at other interim points in the cycle and administrating the resulting limits on an "exposure-dependent" basis.

The analytical procedure used by GE for developing exposure-dependent operating limits consists of analyzing transient events (which rely on reactor scram for protection) at selected mid-cycle exposures in addition to the end-of-cycle "worst-case" analysis. Because the scram reactivity function monotonically degrades with cycle burnup towards end-of-cycle, the operating limit determined for a given exposure E , can be conservatively administrated in the exposure interval E, where $E_{i-1} < E < E_i$, and E_{i-1} is the next earlier exposure point analyzed.

The staff considers the exposure-dependent operating limits concept to be reasonable. Furthermore, since the codes, correlations and analytical procedures are the same as those used by GE for determining operating limits for an EOC worst case analysis our approvals and concerns relative to these methods are the same as those stated in Section 3.4.1.2 herein.

Additionally, the staff has raised questions about the conservative implementation of the calculated limits which would appear in the Hatch-1 Technical Specifications. The exposure-dependent limits are referenced from the end-of-cycle, i.e., EOC minus 1000 Mwd/t. The "true" cycle (exposure) length is not precisely predictable, and hence is not exactly known,

until the end-of-cycle, all rods out condition is actually attained. This is due to uncertainties which exist in the burnup of the previous cycle, actual vs. projected (idealized) rod patterns and modeling uncertainties. Thus, the end of cycle exposure would not be precisely known during the cycle, nor would an exposure corresponding to EOC minus 1000 Mwd/t.

GE has informed the staff that a reanalysis of the projected cycle length is performed early into the operational cycle. This procedure permits previous cycle exposure history uncertainties to be eliminated from the cycle length predictions. Using this procedure and the standard nuclear methods evaluated in Section 3.2.1, gives rise to errors in these predictions of approximately 150 Mwd/t. This is considered to be a small error in comparison to the actual measured cycle lengths.

An error (delay) of this magnitude, in the implementation of an intermediate exposure point operating limit would give rise to an insignificant non-conservatism in the predicted consequences (e.g., MCPR), even for the transient event having the greatest sensitivity (CPR) to time in cycle. In view of the analysis methods and procedure used by GE for calculating cycle exposure and exposure-dependent limits, the staff considers the exposure-dependent operating limits methods to be acceptable for Hatch-1 during Cycle 3 and that the resulting Technical Specifications requirements can be adequately implemented.

3.4.2 Transient Analysis Results

References 2 and 8 provide the results of the reanalysis of the most severe abnormal operational transients for Cycle 3 of Hatch-1. The types of abnormal operational transients analyzed were reactor pressure increase, feedwater temperature decrease, feedwater flow increase and local positive reactivity insertion events. The methods used in the analysis of the limiting transients applicable to Hatch-1 are described in Reference 1. Our evaluation of these methods is provided in Section 3.4.1. The licensee has elected to analyze the transient events on an exposure-dependent basis in order to provide greater operating margins and flexibility during the cycle. The acceptability of the exposure-dependent transient methods is also

provided in Section 3.4.1 herein. Our evaluation of the transient analysis results for Hatch-1 Reload 2, relative to the MCPR safety limit, LHGR safety limit and RCPB safety limit is provided in the following subsections.

3.4.2.1 Transients Effecting the Entire Core

Load Rejection Without Bypass

The load rejection without bypass transient produces the most severe reactor isolation. The reactor pressure increase due to fast closure of the turbine control valves causes a significant decrease in the core void fraction which in turn induces a positive core reactivity insertion, resulting in a rapid and substantial rise in the core neutron flux. The transient is terminated by a reactor trip initiated by fast closure position switches on the turbine control valves.

The analysis of this transient was performed at exposures corresponding to 1000 Mwd/t before EOC-3 and EOC-3. Since the severity of this event (reactor pressure increase and bundle CPR decrease) increases with burnup, the former analysis provides conservative results for reactor operation from BOC-3 to 1000 Mwd/t before EOC-3, while the latter provides conservative results for operation from 1000 Mwd/t before the end of cycle to the end of Cycle 3. The analyses were performed assuming an initial reactor thermal power level correpsonding to 104% of the licensed limit, which is considered to be adequately conservative. The analysis results provided in Section 9 of Reference 8, show that at the most limiting (EOC) condition, a 180 psi margin exists between the peak transient pressure and the 1375 psig RCPB safety limit.

The load rejection without bypass event also results in a significant reduction in MCPR from the operating value. This is caused by the combined effects of the rapid and substantial increase in the neutron flux, which results in a significant increase in the fuel rod surface heat flux together with the substantial increase in reactor pressure. The reanalysis results presented in Section 9 of Reference 8 shows that the reduction in operating MCPR is more severe at EOC-3 than at 1000 MWd/t before EOC-3. The reduction in operating MCPR at EOC is 0.16 for the 7x7 fuel, 0.22 for the 8x8 fuel and 0.22 for the 8x8R fuel. At 1000 Mwd/t prior to EOC-3 the reduction is 0.10

and 0.16 for the 7x7 and the two 8x8 fuel types, respectively. Comparing these results with the other transient events affecting the entire core shows that, except for the loss of 100°F feedwater heating capability event occurring at 1000 Mwd/t prior to EOC-3, the load rejection with bypass failure is the most severe core wide transient for Hatch-1 during Cycle 3.

Other Core Wide Transients

The other core wide transients analyzed for Cycle 3 were feedwater controller failure (maximum demand) and loss of 100°F feedwater heating (LFWH) capability. The event descriptions for these transients are given in Reference 1.

A comparison of these events with the load rejection without bypass, shows that the reactor pressure increase associated with these two transients is less severe than the pressure increase for the load rejection without bypass. In connection with CPR effects, the loss of 100°F feedwater heating capability transient results in a larger reduction in MCPR for the 7x7 fuel type from BOC-3 to 1000 Mwd/t before EOC-3, when compared with the CPR for the load rejection event. The 7x7 CPR during this exposure interval is 0.13 for the LFWH and 0.10 for the load rejection event. The CPR's for the load rejection without bypass event are more severe for all fuel types at all other exposures.

3.4.2.2 Rod Withdrawal Error

The rod withdrawal error (RWE) transient can occur when the reactor operator makes a procedural error and attempts to withdraw the maximum worth control rod to its fully withdrawn position. The attendant local power increase in the fuel assemblies in the vicinity of the withdrawn control rod causes a reduction in the bundle CPR's in addition to an increase in the fuel rod local LHGR's. The information provided in Reference 1 indicates that the local power range monitor subsystem (LPRM's) will detect and alarm a high local power condition. However, even if the reactor operator ignores the LPRM alarm, References 1 and 3 indicate that the rod block monitor subsystem (set at 105% of rated power at 100% core flow) will terminate the RWE transient with the control rod only 4.5 feet withdrawn. This will limit the maximum reduction in the critical power ratio to 0.12 for the effected 7x7 assemblies, 0.11 for the 8x8 assemblies, and 0.15 for the 8x8R assemblies.

A RBM rod block occurring at 105% power and full core flow results in peak linear heat generation rates of 15.2 Kw/ft, 13.3 Kw/ft and 13.3 Kw/ft for the effected 7x7, 8x8 and 8x8R assemblies, respectively. These calculated LHGR's are below the safety limit LHGR's for 7x7 and 8x8 fuels even when the effects of densification spiking are included and are therefore acceptable to the staff.

3.4.3 MCPR Operating Limits for Rated Conditions

Abnormal operating transients, as discussed in the previous section, will reduce fuel bundle critical power ratios from steady-state operating values. In order to assure that the 1.07 fuel cladding integrity safety limit MCPR is not violated during the most severe transient, the most limiting transients have been reanalyzed for Cycle 3 on an exposure-dependent basis, to determine which transient event results in the largest decrease in critical power ratio (i.e., CPR). The most limiting abnormal operational transient which can occur at any time during Cycle 3 is the load rejection without bypass, with the exception of the LFWH event, which is more severe for the 7x7 fuel type between BOC-3 and 1000 Mwd/t before EOC-3. A summary of the calculated fuel type dependent CPR's, for the exposure increments analyzed,(8) is as follows:

TABLE 3.1

	Exposure Interval	
Fuel Type	BOC-3 to EOC-3 Minus 1000 Mwd/t	EOC-3 Minus 1000 Mwd/t to EOC-3
7x7	0.13	0.16
8x8	0.16	0.22
8x8R	0.16	0.22

Addition of the above CPR's to the safety limit MCPR, would normally provide the minimum operating limit MCPR, for each fuel type (and exposure increment) required to avoid violation of the safety limit, should the most limiting transient occur. The licensee has therefore proposed the following MCPR operating limits:

TABLE 3.2

Fuel Type	Exposure Interval: BOC-3 to EOC-3 Minus 1000 Mwd/t	Exposure Interval: EOC-3 Minus 1000 Mwd/t to EOC-3
7x7	1.20	1.23
8x8	1.23	1.29
8x8R	1.23	1.29

However, the licensee reports in the amended reload supplement[8] that the worst case fuel loading error (FLE), consisting of a fresh 8x8R bundle misoriented in its correct 8x8R cell location, results in a MCPR of 1.03 when starting from an initial MCPR of 1.23. Furthermore, the licensee reports[7] that placing a fresh 8x8R in an exposed 7x7 location results in a 1.06 MCPR when starting from an initial MCPR of 1.20. Finally, no violation of the safety limit MCPR occurs when a fresh 8x8R assembly is placed into a standard 8x8 cell location.

As discussed in Section 3.5.3, the staff has the fuel loading error under generic review. Until the issues raised in connection with this event are resolved, the staff, in the interim, requires that the operating limit MCPR's proposed by the licensee be sufficiently increased for all fuel types and exposures to adequately account for the possibility of a fuel loading error such that the safety limit MCPR is not violated. Thus, based on the analysis of both the most severe abnormal operational transients and the fuel loading error, we require that the operating limit MCPR's for Hatch-1 during Cycle 3 be as follows:

TABLE 3.3

Fuel Type	Exposure Interval: BOC-3 to EOC-3 Minus 1000 Mwd/t	Exposure Interval: EOC-3 Minus 1000 Mwd/t to EOC-3
7x7	1.21	1.23
8x8	1.23	1.29
8x8R	1.27	1.29

The adjusted MCPR operating limits appearing in Table 3.3 have been discussed with the licensee. The licensee has agreed to increase the Cycle 3 MCPR operating limits to the values shown. These limits are acceptable to the staff.

3.4.4 MCPR Operating Limits for Less than Rated Flow

To assure that the 1.07 safety limit MCPR is not violated for the limiting flow increase transient (recirculation pump speed control failure) starting from less than rated flow conditions, the licensee will operate Hatch-1 in conformance with the limiting conditions for operation as stated in paragraph 3.5-K of the Technical Specifications. This requires that for core flow rates less than rated flow, the licensee shall maintain the MCPR above the minimum operating values. The minimum MCPR values for less than rated flow are equal to the MCPR for rated flow multiplied by the respective K_f reactor values appearing in Figure 3.5-2 of the Technical Specifications. The K_f factor curves were generically derived and assure that for the most limiting (flow increase) transient, occurring from less than rated core flow, the actual MCPR will not exceed the safety limit MCPR of 1.07. The K_f curves were generically derived[58] and are applicable for all fuel types present in the Hatch-1 Cycle 3 core.

Application of the above stated K_f factors, for reduced flow conditions, results in calculated consequences for the limiting anticipated flow increase transients which do not exceed the thermal limits of the fuel.

Thus, we conclude the analyses and the operating limits, based upon the use of the General Electric Thermal Analysis Basis[52], have been conservatively applied to Hatch-1, Reload 2, and are acceptable.

3.5 Accident Analysis Evaluation

3.5.1 Loss of Coolant Accident

ECCS Appendix K Model Applicability

Because of the physical differences between the standard 8x8 (and 7x7) and the retrofit 8x8 fuel designs described in Section 3.1.1, we reviewed the acceptability of continued application of the previously approved,[65] unchanged, ECCS-LOCA models to the new fuel. Our review and evaluation of GE's responses[60] to our request for justification of such continued application follow.

The staff agrees with the following assertions made by GE:

- All parameters of the new 8x8R fuel, such as hydraulic diameter, pressure, flow, power, and temperature are within the range of data used in developing the GEXL correlation in the ECCS-LOCA models to determine time-to-DNB for the retrofit fuel is acceptable. Also, the R-Factors used in this (LOCA) application of GEXL result from a conservative and therefore acceptable initialization procedure.

- Slightly higher PCT's are calculated for the new fuel (compared to the standard 8x8 fuel at the same MAPLHGR). This is due to the small change in fuel dimensions (resulting in reduced surface area) and a shift in local power peaking toward the center of the bundle. These effects are properly included in the models, so continued application of the models in the new PCT-MAPLHGR range is acceptable.

- GE has previously stated that substantial changes in rod dimensions, spacing, linear heat generation rate, and lattice design do not significantly affect spray cooling heat transfer coefficients. We agree with GE that the changes from the standard 8x8 fuel design to the two water rod retrofit fuel design will not affect the overall conservatism and acceptability of the spray cooling coefficients assumed for the new fuel in the ECCS model inputs.

- The radiative heat transfer model used in CHASTE code was written to handle calculations with various size rods, including rods of unequal radii. Hence it is capable of calculating radiative heat transfer for the new fuel design, and its application for that purpose is acceptable.

- The data base used to develop the swelling and rupture model covered the range of internal pressures and temperatures expected for the new retrofit 8x8 fuel. The swelling and rupture model is therefore equally acceptable to both the standard and retrofit 8x8 fuel designs.

- The data base used to develop the gap conductivity model included the range of temperature, internal pressure, and gap size applicability to the retrofit fuel design. Application of the gap conductivity model to the new fuel is therefore acceptable.

- It has been known by the staff that GE's method of initializing gap conductivity (as a function of assumed fuel rod linear power level) could possibly be more conservative than it is. However, GE has shown that this initialization method, when applied to the retrofit 8x8 fuel, is slightly more conservative than when it is applied to the standard 8x8 fuel design, where its application has previously been accepted. The initialization method therefore is also acceptable for use with the new retrofit fuel.

- The retrofit 8x8 fuel has a more uniform axial power profile and a six-inch longer active fuel length. These factors make it possible that the plane of maximum PCT could shift to a higher elevation (power is lower above the core midplane, but loss-of-nucleate boiling occurs earlier for the new fuel compared with the old fuel). However, the application of the model to Hatch-1 includes a calculation to demonstrate that such a shift has not occurred[61]. Continued use of that calculation provision for Hatch-1 will ensure that the application of the model to the new fuel will be at the axial plane producing the highest PCT, and will therefore be acceptable.

For the reasons stated above, we conclude the continued application of the present GE ECCS-LOCA ("Appendix K") models to the 8x8 retrofit fuel is acceptable for Hatch-1.

ECCS Appendix K LOCA Analysis Results

On December 27, 1974, the Atomic Energy Commission issued an Order for Modification of License implementing the requirements of 10 CFR 50.46, "Acceptance Criteria and Emergency Core Cooling Systems for Light Water Nuclear Power Reactors." One of the requirements of the Order was that prior to any license amendment authorizing core reloading "...the licensee shall submit a re-evaluation of ECCS performance calculated in accordance with an acceptable evaluation model which conforms to the provisions of 10 CFR 50.46." The Order also required that the evaluation shall be accompanied by such proposed changes in Technical Specifications or license amendments as may be necessary to implement the evaluation results and assumptions.

In December of 1976, the NRC staff was informed that certain input errors and computer code errors had been made in the evaluations that were provided under the requirements described above. An Order was issued by the Nuclear Regulatory Commission to the Georgia Power Company on March 11, 1977, requiring that corrected "revised calculations fully conforming to the requirements of 10 CFR 50.46 are to be provided for (Hatch-1) as soon as possible." Such corrected analyses were provided for the previous core and the reloaded (Cycle 3) core in Reference 4. The revised calculations included corrections of all of the input errors and all computer code errors. The corrected analyses were performed using a calculational model which contains several model changes approved by the staff in its Safety Evaluation issued April 12, 1977[65].

The analyses submitted in Reference 4 provides all information requested regarding number of breaks to be analyzed, documentation to be provided, etc., for the new analyses. The ECCS-LOCA analysis for Hatch-1 references the "lead plant" (James A. Fitzpatrick Nuclear Power Plant) analysis for BWR/4 plants with the low-pressure-coolant-injection system modification[62].

The break spectrum (i.e., PCT vs. break size) for the lead plant showed that the particular break producing the highest PCT for the lead plant was a recirculation pump discharge line break having an area approximately 80% as large as the largest discharge line break[62]. The break spectrum for Hatch-1 showed

that this same size and location break also produces the highest PCT for Hatch-1[4]. The SER for the lead plant[63] justifies the "80%" discharge break size and location as the limiting break for that plant. Since Hatch-1 is identical in size and reactor type to the lead plant, and since the same size and location break is limiting for both plants, the discussions presented in the lead plant SER are directly applicable to Hatch-1 and are not repeated here.

We therefore conclude, for the reasons stated in the lead plant SER[63], that the most limiting break for Hatch-1 is the discharge line break with 80% of the largest discharge line break's area. That break was used to generate the proposed revised MAPLHGR limits.

Thus, based on our review of the applicability of the ECCS-LOCA models to the 8x8R fuel type, the corrected analyses[4] and the lead plant analysis, we conclude that Hatch-1 will be in conformance with all requirements of 10 CFR 50.46 and Appendix K to 10 CFR 50.46 when operated in accordance with the MAPLHGR versus Planar Average Exposure curves given in Figure 3.11-1, sheets 1 and 2, of the proposed Technical Specifications[5].

3.5.2 Steamline Break Accident

The radiological consequences of a postulated steamline break outside of the primary containment are dependent on the amount of primary coolant lost during the accident and the concentration of the radioactivity in the coolant. The amount of coolant lost is primarily a function of plant system parameters, which would be insignificantly changed by introduction of the 8x8R fuel assemblies into the core. The concentration of radioactivity in the coolant is limited by the Hatch-1 plant Technical Specifications and is also unchanged for this reload. Therefore, the previously calculated radiological consequences of a postulated steamline break accident at Hatch-1 are unaffected by the use of the 8x8R fuel assemblies.

3.5.3 Fuel Loading Error

References 2 and 8 give the results of the fuel loading error analysis for Hatch-1 during Cycle 3. The most severe fuel loading error event consists of a misoriented, fresh 8x8R fuel bundle. The information in Reference 8 indicates that this worst case fuel loading error, were it to

occur, would result in a minimum critical power ratio (MCPR) of 1.03 in the misoriented fuel bundle during steady-state full power operating conditions. Fuel bundles adjacent to the misloaded fuel-assembly would be negligibly affected by the misoriented bundle. The calculated MCPR of 1.03 in the misloaded bundle violates the 1.07 fuel cladding integrity safety limit MCPR. Furthermore, the licensee reports[7] that placing a fresh 8x8R in an exposed 7x7 location results in a 1.06 MCPR when starting from an initial MCPR of 1.20. Finally, no violation of the safety limit MCPR occurs when a fresh 8x8R assembly is placed into a standard 8x8 cell location.

The fuel loading error event is being generically reviewed by the staff and a generic resolution is anticipated. Our ongoing review includes an evaluation of the adequacy of proposed new fuel loading error methods, event probabilities resulting from improved core loading control procedures, in addition to acceptable consequences for the fuel loading error event. Until these evaluations are complete we require that the licensee increase the MCPR operating limits to values which will assure that, during normal operation, the safety limit MCPR will not be violated. Some adjustment in the 7x7 and 8x8R exposure-dependent operating limit MCPR's proposed in Section 11 of Reference 8 is necessary to meet this requirement. For Cycle 3, the MCPR operating limits shown in Table 3.3 herein, will assure that the most severe fuel loading error will not cause a violation of the safety limit MCPR.

3.5.4 Control Rod Drop Accident

The postulated control rod drop accident assumes that a control rod has been fully inserted and becomes stuck in this position. The control rod drive is assumed to be uncoupled and withdrawn. The rod subsequently becomes free and rapidly falls out of the core onto the withdrawn drive coupling. The amount of reactivity represented by this event is introduced into the reactor core at a rate consistent with the maximum control rod drop velocity.

There are two criteria which must be satisfied in the analysis of the control rod drop accident:

- Reactivity excursions must not result in a fuel enthalpy greater than 280 cal/g at any axial pellet location in any fuel rod. This limit assures that dispersal of fuel into the reactor coolant will not occur.

- The maximum reactor pressure during any portion of the accident must be less than the value that will cause reactor system stresses to exceed the emergency condition stress limits defined in the ASME code.

The analysis of the control rod drop accident was performed by General Electric on a generic (bounding) basis and presented in Reference 1. The bundle cross sections, developed by the lattice calculations (discussed in Section 3.2.1) for the rod drop excursion model, are homogenized. As a result, the rod drop excursion model does not recognize the difference between 7x7, 8x8 or 8x8R fuel. Therefore, the calculational model used in the generic analysis remains acceptable for the new fuel design. The evaluation of the control rod drop accident thus consists of ensuring that the appropriate parameters of the new core are bounded by the input parameter values used in the generic analysis.

The generic analysis assumes the slowest scram allowed by the Technical Specifications (and assumes that the dropped rod does not scram), the most rapid credible rod drop velocity, and the smallest (i.e., high exposure) value for delayed neutron fraction. The remaining parameters of interest include the Doppler feedback, the scram reactivity, and the accident reactivity characteristics.

We have reviewed the bounding calculations presented in Reference 1 with regard to the 280 cal/g limit and find them to be acceptable for reference, provided the key input parameters for the Hatch-1 Reload 2 application fall conservatively within the assumed bounding analysis values. The key parameters are Doppler coefficient, scram reactivity function and accident reactivity function.

Our review of the key Cycle 3 control rod drop accident inputs for Hatch-1 indicates that they are not all within the bounding analysis inputs presented in Reference 1. The actual Cycle 3 Doppler coefficient and scram reactivity shape function for both hot and cold conditions both conservatively fall within the values assumed in the bounding analysis. The accident reactivity shape function for the cold condition however does not. Thus, it may be concluded that for a control rod drop accident starting from hot shutdown conditions, the peak fuel pellet enthalpy will be below the design limit of 280 cal/g calculated with the bounding analysis inputs. With regard to the control rod drop accident during cold conditions, the licensee has performed a plant specific analysis for Hatch-1, Cycle 3. The plant specific analysis was performed using an actual cold Doppler coefficient of reactivity corresponding to the beginning of cycle, which is most limiting for this accident since the Doppler coefficient is least negative at BOC.

The results of the control rod drop accident analysis for the cold condition shows that the positive reactivity insertion rate of dropped rod is compensated sufficiently by the negative Doppler and scram reactivity effects to limit the peak pellet enthalpy to a maximum of 217 cal/g. Thus, it is concluded that a control rod drop accident occurring during Cycle 3 from any in-sequence control rod movement will result in a peak pellet enthalpy which is below the design limit of 280 cal/g and therefore has acceptable fuel consequences.

The licensee was requested to provide an overpressurization analysis which demonstrates that the maximum reactor coolant pressure boundary pressure occurring during a control rod drop accident would not cause applicable ASME stress limits to be exceeded. The results of this analysis was not available prior to the completion of our review. Preliminary results[6] indicate that the pressure transient is relatively mild, i.e., 15 psi. The staff will continue to review this aspect of the control rod drop accident on a generic basis.

3.5.5 Fuel Handling Accident

The refueling accident has been generically reanalyzed(1) to determine the radiological consequences for the 8x8R fuel assembly. The analysis assumes (1) the fuel assembly is dropped from the maximum height (maximum potential energy) allowed by the fuel handling equipment, (2) none of the kinetic energy is viscously dissipated as the assembly falls through the water covering the core and (3) none of the kinetic energy is absorbed by the fuel material (UO_2) in the assembly. Using energy methods to predict cladding failures, it is shown that a total of 125 8x8R fuel rods fail during the accident. This compares with 111 rods for a 7x7 core. There would be no difference in failed rods for an 8x8 core. The evaluation also conservatively assumes that the fractional plenum activity in the 8x8R rod is the same as for a 7x7 rod. In actuality an 8x8R rod would have substantially lower gap activity as compared to a 7x7 rod as a result of the significantly lower linear heat generation rate (fuel temperatures) applicable to the new fuel bundle design. Comparing the average activity per 8x8R fuel rod to the average activity per 7x7 rod together with the number of failed rods for each bundle type (125 vs 111), it is shown by the licensee that the 8x8R fuel bundle results in a relative activity release of only 88% of the activity released for a 7x7 core. Thus, of the total activity available for release above the core, the fission product activity component attributable to the fuel is less for the 8x8R fuel than for the 7x7 fuel. The FSAR analyses of the 7x7 core for Hatch-1 showed fuel handling accident dose consequences which were appropriately well within the guidelines set forth in 10 CFR 100. Thus, we conclude that the dose consequences of the fuel handling accident, associated with the 8x8R fuel assembly for Hatch-1, are also well within 10 CFR 100 guidelines and are acceptable.

3.5.6 Recirculation Pump Seizure Accident

The analysis of the single pump seizure event shows that it is relatively mild with regard to radiological consequences, plant system behavior and fuel performance when compared to a large LOCA. For both accidents recirculation flow rapidly terminates. In the case of the LOCA, the forced recirculation flow disruption is more rapid and severe than the pump seizure event. Furthermore, the loss of coolant accident results in core uncovery

with subsequent rapid and substantial temperature rise of the fuel cladding. The pump seizure accident also does not result in as rapid and core pressure drop as does the LOCA. The combination of higher peak cladding temperature and lower RCS pressure during a LOCA event results in greater cladding perforation potential for the LOCA than the pump seizure event. The staff agrees that the potential adverse effects on the fuel of a pump seizure accident are conservatively bounded by a LOCA. Additionally, the LOCA results in the removal of reactor coolant pressure boundary as a barrier to the release of fission products outside of containment. The single pump seizure does not result in the loss of this barrier. Therefore, it may also be concluded that the radiological consequences associated with the LOCA conservatively bounds the radiological consequences of the pump seizure event. Since the radiological consequences of the LOCA as described in the Hatch-1 FSAR were shown to be acceptable, the consequences of the pump seizure accident are also considered to be acceptable.

3.6 Overpressurization Analysis

The licensee presented the results of an overpressurization analysis[2] to demonstrate that margin exists to the ASME code allowable reactor vessel pressure limit. This limit, as discussed previously in Section 3.4, is 110% of the vessel design pressure and corresponds to a pressure of 1375 psig. The methods[1] used for this analysis are evaluated in Section 3.4.1.2 herein. The transient event analyzed was the rapid closure of all main steam isolation valves (MSIV) with an indirect reactor trip on high neutron flux. The analysis was performed assuming an initial core thermal power level corresponding to 104% of the license limit. In addition, the analysis conservatively utilized the end-of-cycle scram reactivity insertion rate curve, with void and Doppler reactivity coefficients applicable for this reload. Moreover, no credit was taken for the relief function of the 11 dual action safety/relief valves installed on the main steam lines. All valves were assumed operative in the analysis. The result of the analysis shows that the peak pressure at the bottom of the reactor vessel is 1234 psig.

Furthermore, generic analyses[1] applied to Hatch-1 show that the failure of one of the safety/relief valves would cause the maximum vessel pressure to increase by no more than 20 psi. Thus, the peak transient pressure at the vessel bottom for the MSIV closure overpressurization event from full power with flux scram, no relief function of the safety/relief valves and one failed safety/relief valve is calculated to be less than 1254 psig. This results in an adequate margin to the 1375 psig ASME code allowable pressure limit and is thus acceptable to the staff.

4.0 Physics Startup Testing

As part of our evaluation of Reload 2 of Hatch-1, we reviewed the physics startup test program which will be conducted by the licensee at the beginning of Cycle 3. The test program description is provided in Reference 1. Based on our review of the information[1,64] provided by the licensee, the staff finds that the physics startup tests together with the tests required to assure compliance with the Technical Specifications, provide an acceptable physics startup test program.

5.0 Technical Specification Changes

The proposed revisions[3,5,9] to the Hatch-1 Technical Specifications for Cycle 3 operation include changes to the MCPR safety limit, the MCPR operating limits and the MAPLHGR vs. planar average exposure curves. The bases for the Technical Specification changes are documented in the reload submittals provided by the licensee. The bases for the proposed revisions have been evaluated by the staff and are discussed in Section 3.0.

As discussed in Section 3.3.2.1 of this evaluation, the MCPR safety limit has been increased from 1.06 to 1.07 for Cycle 3. This is to accommodate the combined effects of the flatter intra-assembly power peaking distribution associated with the retrofit 8x8 reload fuel assembly and the revised core relative bundle power histogram (distribution) associated with a reload 8x8R cycle approaching equilibrium conditions. Based on our review of the information submitted by the licensee, the staff finds the 1.07 safety limit MCPR proposed for Hatch-1 during Cycle 3 to be acceptable.

The Hatch-1 Technical Specification revisions for Cycle 3 also address a change in the MCPR operating limits for the 7x7, 8x8 and 8x8R fuel types, based on the reload safety analysis results presented in References 2 and 8. As discussed in Sections 3.4.3 and 3.5.3 herein, the proposed operating limit MCPR's must be increased in come cases to assure that the 1.07 safety limit MCPR is not violated in the event of a fuel loading error. The Technical Specification operating limit MCPR's for each fuel type and exposure interval selected by the licensee must, therefore, be those appearing n Table 3.3 of this evaluation. The adjusted MCPR operating limits have been discussed with and accepted by the licensee.

The licensee has also proposed changes and additions(5) to the MAPLHGR vs. planar average exposure curves currently appearing in the Hatch-1 Technical Specifications. Our evaluation of the proposed curves is discussed in Section 3.5.1. The MAPLHGR changes for the 7x7 and standard 8x8 fuel types reflect the results of revised LOCA calculations, performed to correct all of the GE input errors, made in connection with the previous LOCA analysis. An additional MAPLHGR curve, based on corrected inputs for the reload 8x8R fuel is also provided. Based on our evaluation of the information provided, the staff finds the proposed new MAPLHGR vs. planar average exposure curves appearing in Figure 3.11-1 of Reference 4 to be acceptable.

The analysis contained in the licensee's reload submittal are based on the same water levels and setpoints as those which currently exist in the Hatch Unit No. 1 Technical Specifications. Since the 8x8R reload fuel bundles have greater active fuel length, the licensee proposed changes to those specifications which refer to the top of the active fuel and to 2/3 core height. These limiting conditions for operation and their associated bases have been changed to reference either the vessel invert or vessel zero. In addition reference to the approximate height above the top of the active fuel has been reduced six inches to reflect the change in core height affected by the reload fuel bundles. Since these changes are editorial in nature and do not revise any water-level setpoints, we find them acceptable as proposed.

6.0 Environmental Consideration

We have determined that the amendment does not authorize a change in effluent types or total amounts nor an increase in power level and will not result in any significant environmental impact. Having made this determination, we have further concluded that the amendment involves an action which is insignificant from the standpoint of environmental impact and, pursuant to 10 CFR 51.5(d)(4), that an environmental impact statement or negative declaration and environmental impact appraisal need not be prepared in connection with the issuance of this amendment.

7.0 Conclusion

We have concluded, based on the considerations discussed above, that: (1) there is reasonable assurance that the health and safety of the public will not be endangered by operation in the proposed manner, and (2) such activities will be conducted in compliance with the Commission's regulations and the issuance of this amendment will not be inimical to the common defense and security or to the health and safety of the public.

Dated: April 11, 1978

7.0 References

1. "General Electric Boiling Water Reactor Reload-2 Licensing Application for Edwin I. Hatch Nuclear Plant Unit 1," NEDO-24040, July 1977, submitted as an Enclosure of Georgia Power Company letter (C. Whitmer) to NRC (V. Stello) dated August 26, 1977.

2. "Supplemental Reload Licensing Submittal for Edwin I. Hatch Nuclear Plant Unit 1 Reload 2," NEDO-24078, November 1977, submitted as Enclosure 2 of Georgia Power Company letter (C. Whitmer) to NRC (V. Stello) dated December 1, 1977.

3. Georgia Power Company letter (C. Whitmer) to NRC (V. Stello) dated December 1, 1977, Enclosure 1, Proposed Changes to Technical Specifications.

4. "Loss of Coolant Accident Analysis Report for Edwin I. Hatch Nuclear Plant Unit No. 1," NEDO-24086, December 1977, submitted as Enclosure 2 of Georgia Power Company letter (C. Whitmer) to NRC (V. Stello) dated Januray 5, 1978.

5. Georgia Power Company letter (C. Whitmer) to NRC (V. Stello) dated January 3, 1978, Enclosure 1, Proposed Change to Technical Specifications.

6. Georgia Power Company letter (C. Whitmer) to NRC (V. Stello) dated February 22, 1978.

7. Georgia Power Company letter (C. Whitmer) to NRC dated March 16, 1978.

8. "Supplemental Reload Licensing Submittal for Edwin I. Hatch Nuclear Plant Unit 1, Reload 2," NEDO-24078, Revision 1, February 1978, submitted as Enclosure 2 to Georgia Power Company letter (C. Whitmer) to NRC dated March 8, 1978.

9. Georgia Power Company letter (C. Whitmer) to NRC dated March 8, 1978, Enclosure 1, Modifications to Proposed Changes to Technical Specifications.

10. Safety Evaluation Report Supporting Amendment No. 42 to Facility Operating License DPR-57 dated May 1977.

11. "BWR/6 Fuel Design", General Electric Report, NEDE-20948P, dated December 1975.

12. "Report to the Advisory Committee on Reactor Safeguards in the Matter of Georgia Power Company et al on Edwin I. Hatch Unit 2", dated January 4, 1978.

13. "Lead Test Assembly Supplemental Information for Reload 1 Licensing Submittal for Peach Bottom Atomic Power Station, Unit 2", General Electric Report, NEDO-21172, Revision 1, Supplement 1, March 1976.

14. Philadelphia Electric Company letter (M. Cooney) to NRC (G. Lear), July 5, 1977.

15. "Pressurized Test Assembly Supplemental Information for Reload-1 Licensing Amendment for Peach Bottom Atomic Power Station Unit No. 3", NEDO-21363-1, November 1976.

16. "Experience with BWR Fuel Through December 1976", NEDE-21660P, July 1977.

17. G. C. Potts, "Urania-Gadolinia Nuclear Fuel Physical and Irradiation Characteristics and Material Properties", General Electric Report, NEDE-20943, January 1977.

18. H. F. Williamson and D. C. Ditmore, "Experience with BWR Fuel Through September 1971", NEDO-10505, May 1972.

19. "Generic Reload Fuel Application", General Electric Report, NEDE-24011-P, May 1977.

20. "Creep Collapse Analysis of BWR Fuel Using SAFE-COLAPS Model", NEDE-20606PA, August 1976.

21. NRC letter (W. Butler) to General Electric (I. Stuart), April 2, 1975.

22. M. A. Miner, "Cumulative Damage in Fatique", Journal of Applied Mechanics 12, Transaction of the ASME, 67, 1945.

23. Status Report on the Licensing Topical Report, "General Electric Boiling Water Reactor Generic Reload Application for 8x8 Fuel", NEDO-20360, Revision 1, and Supplement 1, dated April 1975.

24. "GEGAP-III: A Model for the Prediction of Pellet-Cladding Thermal Conductance in BWR Fuel Rods", General Electric Report, NEDC-20181, November 1973.

25. "Technical Report on the General Electric Company 8x8 Fuel Assembly" by the Regulatory Stafff, February 5, 1974.

26. General Electric letter (G. Sherwood) to NRC (D. Eisenhut), "NRC Questions on Rod Bowing", March 29, 1977.

27. "BWR/6 Fuel Design, Amendment No. 1", NEDE-20948-1P, November 1976.

28. "BWR/4 and BWR/5 Fuel Design, Amendment No. 1", NEDE-20944-1, January 1977.

29. USAEC letter (V. Moore) to General Electric (I.S. Mitchell), "Modified GE Model for Fuel Densification", Docket 50-321, March 22, 1974.

30. "Technical Report on Densification of General Electric Reactor Fuels, Supplement 1, by the USAEC Regulatory Staff, December 1973.

31. "General Electric Densification Program Status", NEDE-21282-P, Revision 1, April 1977.

32. NRC letter (O. Parr) to General Electric (G. Sherwood), "General Electric Densification Program Status", dated January 3, 1978.

33. NRC letter (D. Ross) to General Electric (G. Sherwood), dated January 18, 1978.

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60. Letter from General Electric (R. Engle) to NRC (D. Eisenhut), "Applicability of GE-LOCA Models to 8x8 Two Water Rod Fuel", dated Feburary 10, 1978.

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62. Power Authority of the State of New York letter (G. Berry) to NRC (R. Reid), dated July 29, 1977.

63. NRC letter (R. Reid) to PASNY (G. Berry), dated September 16, 1977.

64. Georgia Power Company letter (C. Whitmer) to NRC (V. Stello), "Edwin I. Hatch Nuclear Plant Unit 1 Reload Submittal - Response to Questions" dated April 11, 1977.

65. NRC letter (Goller) to General Electric Company (Sherwood) transmitting Safety Evaluation, April 12, 1977.

UNITED STATES NUCLEAR REGULATORY COMMISSION

DOCKET NO. 50-321

GEORGIA POWER COMPANY, ET AL

NOTICE OF ISSUANCE OF AMENDMENT TO FACILITY OPERATING LICENSE

The U. S. Nuclear Regulatory Commission (the Commission) has issued Amendment No. 52 to Facility Operating License No. DPR-57 issued to Georgia Power Company, Oglethorpe Electric Membership Corporation, Municipal Electric Association of Georgia and City of Dalton, Georgia, which revised Technical Specifications for operation of the Edwin I. Hatch Nuclear Plant, Unit No. 1, located in Appling County, Georgia. The amendment is effective as of its date of issuance.

The amendment modifies the Technical Specifications to: (1) permit operation of the facility during Cycle 3 with up to 168 improved two water rod 8x8R reload fuel bundles, designed and fabricated by the General Electric Company (GE) and having an average enrichment of 2.65 wt/% U-235, and (2) revise the maximum average planar linear heat generation rates (MAPLHGR's) as determined by the reevaluation of the ECCS performance.

The application for the amendment complies with the standards and requirements of the Atomic Energy Act of 1954, as amended (the Act), and the Commission's rules and regulations. The Commission has made appropriate findings as required by the Act and the Commission's

rules and regulations in 10 CFR Chapter I, which are set forth in the license amendment. Notice of Proposed Issuance of Amendment to Facility Operating License in connection with this action was published in the FEDERAL REGISTER on December 28, 1977 (42 FR 64749). No request for a hearing or petition for leave to intervene was filed following notice of the proposed action.

The Commission has determined that the issuance of this amendment will not result in any significant environmental impact and that pursuant to 10 CFR §51.5(d)(4) an environmental impact statement or negative declaration and environmental impact appraisal need not be prepared in connection with issuance of this amendment.

For further details with respect to this action, see (1) the application for amendment dated August 26, 1977 as supplemented December 1, 1977, January 3, January 5, February 22 and March 8 and 16, 1978, (2) Amendment No. 52 to License No. DPR-57 and (3) The Commission's related Safety Evaluation. All of these items are available for public inspection at the Commission's Public Document Room, 1717 H Street, N. W. Washington, D. C. and at the Appling County Public Library, Parker Street, Baxley, Georgia 31513. A copy of items (2) and (3) may be obtained upon request addressed to the U. S. Nuclear Regulatory Commission, Washington, D. C. 20555, Attention: Director, Division of Operating Reactors.

Dated at Bethesda, Maryland this 11 day of April 1978.

FOR THE NUCLEAR REGULATORY COMMISSION

George Lear

George Lear, Chief
Operating Reactors Branch #3
Division of Operating Reactors