June 29, 1983

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DMB 016

Docket No. 50-366

Mr. J. T. Beckham, Jr.
Vice President - Nuclear Generation
Georgia Power Company
P. O. Box 4545
Atlanta, Georgia 30302

Dear Mr. Beckham:

The Commission has issued the enclosed Amendment No. 33 to Facility Operating License No. NPF-5 for the Edwin I. Hatch Nuclear Plant, Unit 2. The amendment consists of changes to the Technical Specifications (TSs) in response to your application dated February 23, 1983, as supplemented April 19, 1983, and your application dated March 30, 1983 as supplemented May 10, 1983, May 20, 1983 and May 26, 1983.

This amendment modifies the TSs to provide additional and revised trip setpoints that reflect design modifications to reduce containment loads from plant transients. The TS changes (1) lower the opening and closing setpoints for actuation of four safety relief valves following initial actuation of any one of the four valves, and (2) lower the main steam isolation valve water level trip setpoint. The amendment also modifies the TSs to reflect changes to the core design for the third fuel reload of Unit No. 2.

A copy of the Safety Evaluation is also enclosed.

Sincerely,

"ORIGINAL SIGNED BY:"

George Rivenbark, Project Manager
Operating Reactors Branch #4
Division of Licensing

Enclosures:
1. Amendment No. 33 to NPF-5
2. Safety Evaluation

cc w/enclosures:
See next page

6/23

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OFFICE	ORB#4:DL	ORB#4:DL	ORB#4:DL	AD-OR:DL	OELD	AD RS DSI	
SURNAME	RIngram/dn	GRivenbark	JStolz	GLainas	Goddard	W. Houston	
DATE	6/3/83	6/3/83	6/3/83	6/20/83	6/7/83	6/22/83	

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UNITED STATES
LEAR REGULATORY COMMISSI
WASHINGTON, D. C. 20555

June 29, 1983

Docket No. 50-366

TAC 49989
51054

W. Burns - TS pages not telefaxed they are in accord with your submittals. George 6/30/83

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Vice President - Nuclear Generation
Georgia Power Company
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George Rivenbark, Project Manager
Operating Reactors Branch #4
Division of Licensing

Enclosures:
1. Amendment No. 33 to NPF-5
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Sholly - 5/18/83
98 FR 22389

cc w/enclosures:
See next page

-8307200357 2pp.

cc w/enclosure(s):

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

GEORGIA POWER COMPANY
OGLETHORPE POWER CORPORATION
MUNICIPAL ELECTRIC AUTHORITY OF GEORGIA
CITY OF DALTON, GEORGIA

DOCKET NO. 50-366

EDWIN I. HATCH NUCLEAR PLANT, UNIT NO. 2

AMENDMENT TO FACILITY OPERATING LICENSE

Amendment No. 33
License No. NPF-5

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

   A. The application for amendment by Georgia Power Company, et al., (the licensee) dated February 23, 1983, as supplemented April 19, 1983, and application dated March 30, 1983, as supplemented May 10, May 20, and May 26, 1983, comply 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 applications, 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. NPF-5 is hereby amended to read as follows:

   (2) Technical Specifications

   The Technical Specifications contained in Appendices A and B, as revised through Amendment No.33 , 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

John F. Stolz

John F. Stolz, Chief
Operating Reactors Branch #4
Division of Licensing

Attachment:
Changes to the Technical
Specifications

Date of Issuance: June 29, 1983

ATTACHMENT TO LICENSE AMENDMENT NO. 33

FACILITY OPERATING LICENSE NO. NPF-5

DOCKET NO. 50-366

Replace the following pages of the Appendix "A" Technical Specifications with the enclosed pages. The revised pages are identified by Amendment number and contain a vertical line indicating the area of change. The overleaf pages are provided to maintain document completeness.

Remove	Insert
3/4 2-1	3/4 2-1
3/4 2-2	3/4 2-2
3/4 2-3	3/4 2-3
3/4 2-4	3/4 2-4a
3/4 2-4A	3/4 2-4b
3/4 2-4B	3/4 2-4c
—	3/4 2-4d
—	3/4 2-4e
—	3/4 2-4f
3/4 2-6	3/4 2-6
—	3/4 2-7b
3/4 2-7b	3/4 2-7c
3/4 2-8	3/4 2-8
3/4 3-11	3/4 3-11
3/4 3-16	3/4 3-16
3/4 3-19	3/4 3/19
3/4 3-21	3/4 3-21

Remove	Insert
3/4 3-27	3/4 3-27
3/4 3-29	3/4 3-29
3/4 3-30	3/4 3-30
3/4 3-32	3/4 3-32
3/4 3-54	3/4 3-54
3/4 4-4	3/4 4-4
	3/4 4-4a
B 3/4 1-2	B 3/4 1-2
B 3/4 2-1	B 3/4 2-1
B 3/4 2-4	B 3/4 2-4
B 3/4 2-5	B 3/4 2-5
B 3/4 3-6	B 3/4 3-6
B 3/4 4-1	B 3/4 4-1
5-1	5-1

3/4.2 POWER DISTRIBUTION LIMITS

3/4.2.1 AVERAGE PLANAR LINEAR HEAT GENERATION RATE

LIMITING CONDITION FOR OPERATION

3.2.1 All AVERAGE PLANAR LINEAR HEAT GENERATION RATES (APLHGRs) for each type of fuel as a function of AVERAGE PLANAR EXPOSURE shall not exceed the limits shown in Figures 3.2.1-1 thru 3.2.1-8.

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

ACTION:

With an APLHGR exceeding the limits of Figures 3.2.1-1 thru 3.2.1-8, initiate corrective action within 15 minutes and continue corrective action so that APLHGR is within the limit within 2 hours or reduce THERMAL POWER to less than 25% of RATED THERMAL POWER within the next 4 hours.

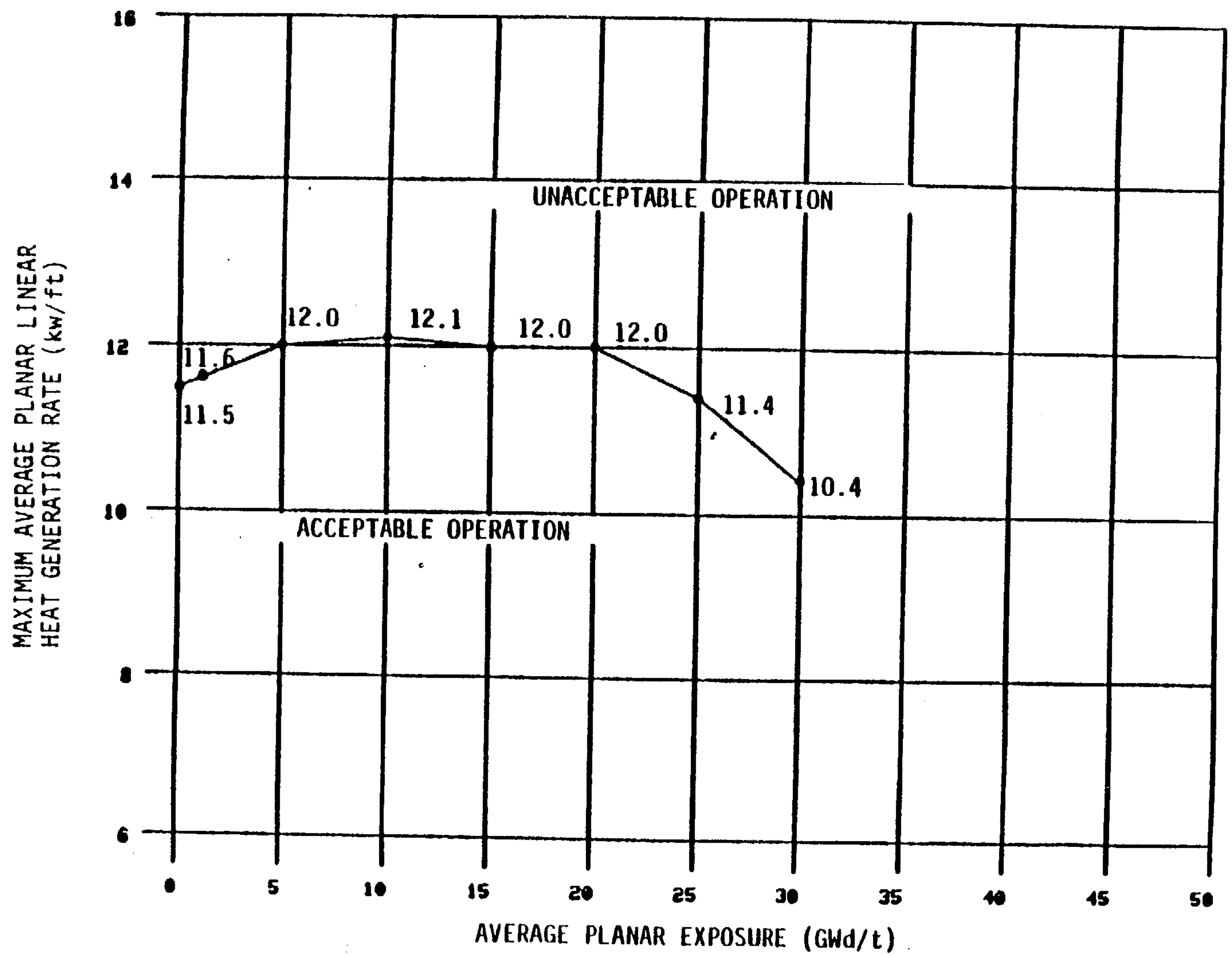
SURVEILLANCE REQUIREMENTS

4.2.1 All APLHGRs shall be verified to be equal to or less than the applicable limit determined from Figures 3.2.1-1 through 3.2.1-8:

a. At least once per 24 hours,

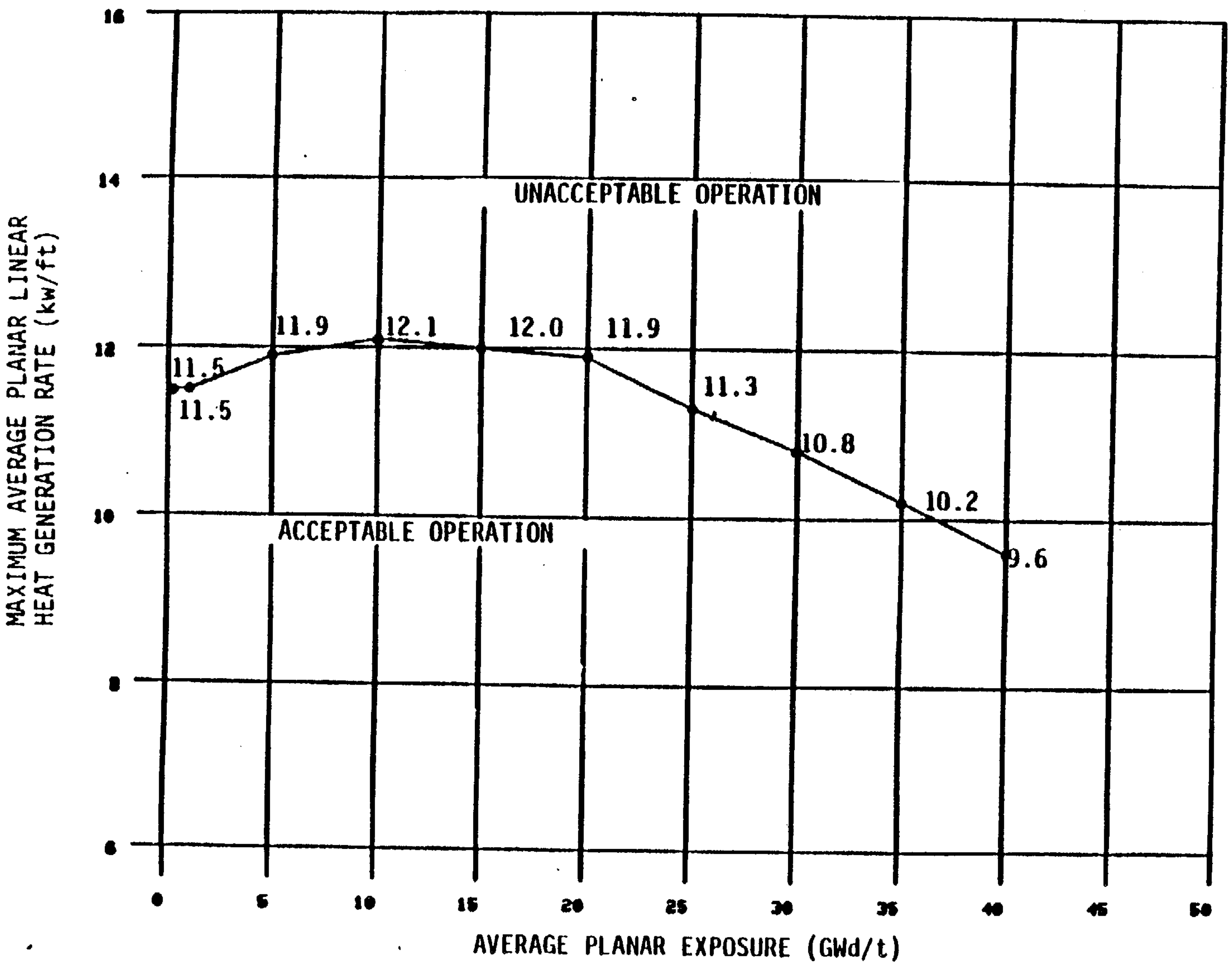
b. Whenever THERMAL POWER has been increased by at least 15% of RATED THERMAL POWER and steady state operating conditions have been established, and

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

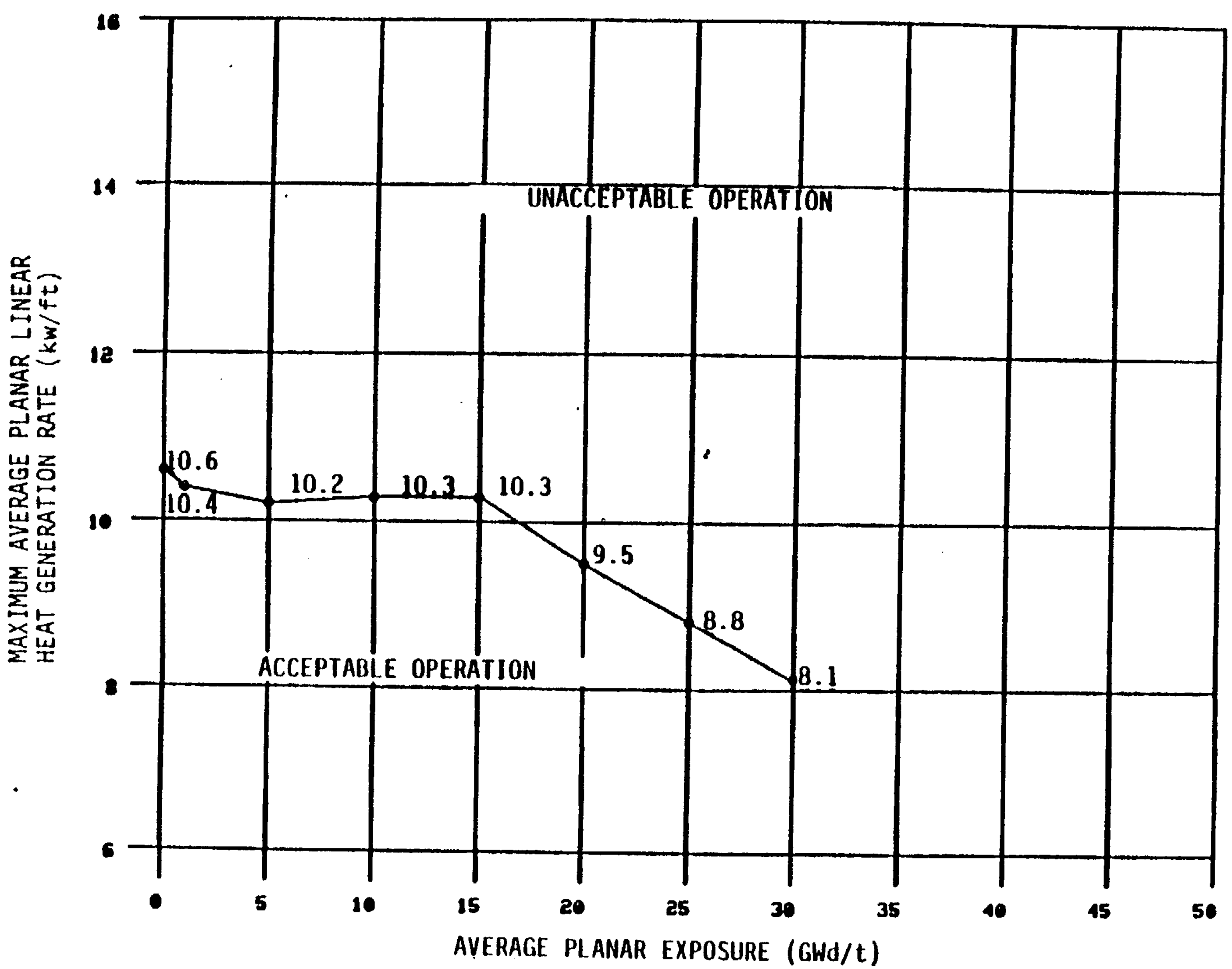


FUEL TYPE 8DIB175(8DRL183)
MAXIMUM AVERAGE PLANAR LINEAR HEAT GENERATION RATE (MAPLHGR) VERSUS AVERAGE PLANAR EXPOSURE

FIGURE 3.2.1-1

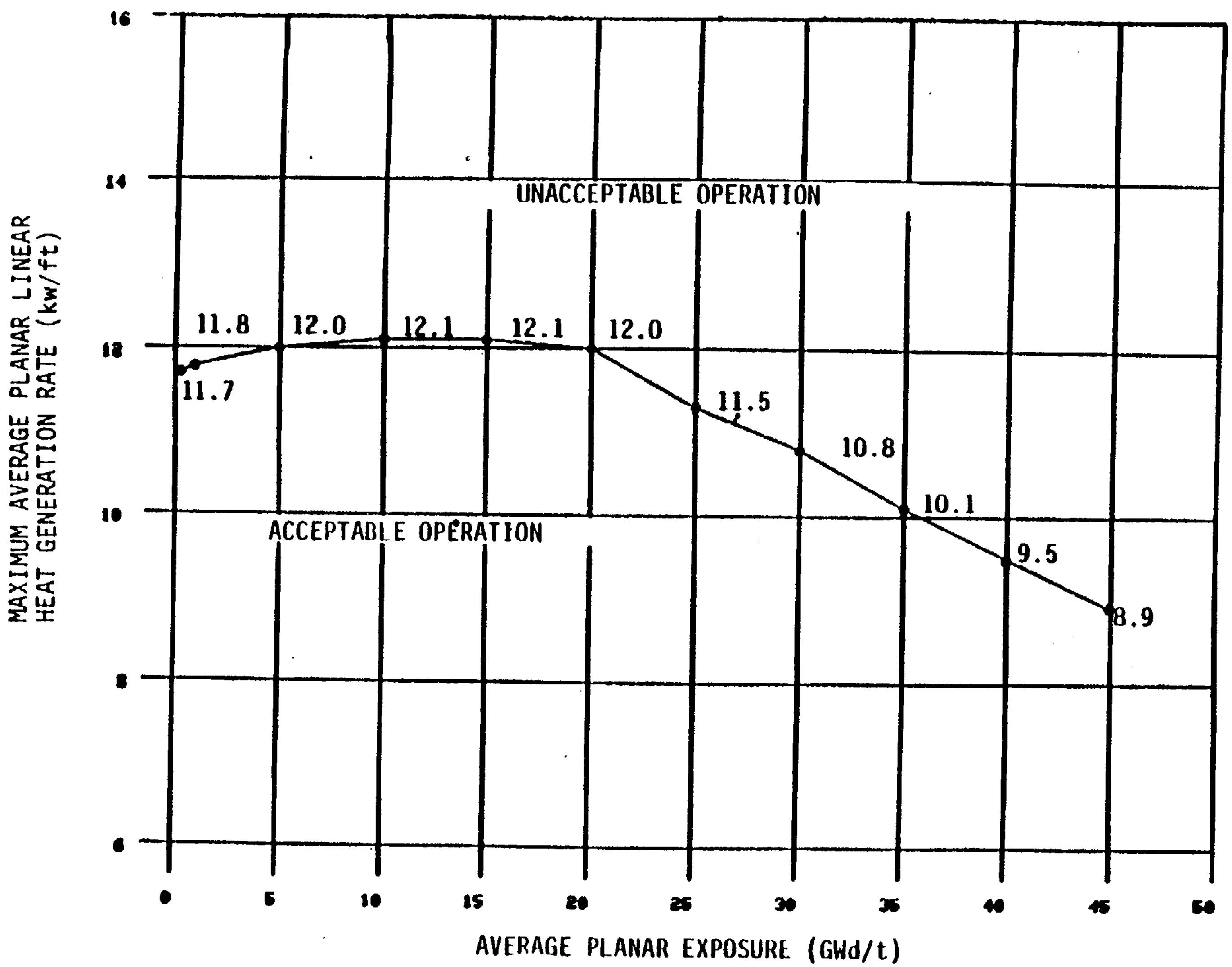


FUEL TYPE 8DIB221(8DRL233)
MAXIMUM AVERAGE PLANAR LINEAR HEAT GENERATION RATE (MAPLHGR) VERSUS AVERAGE PLANAR EXPOSURE
FIGURE 3.2.1-2

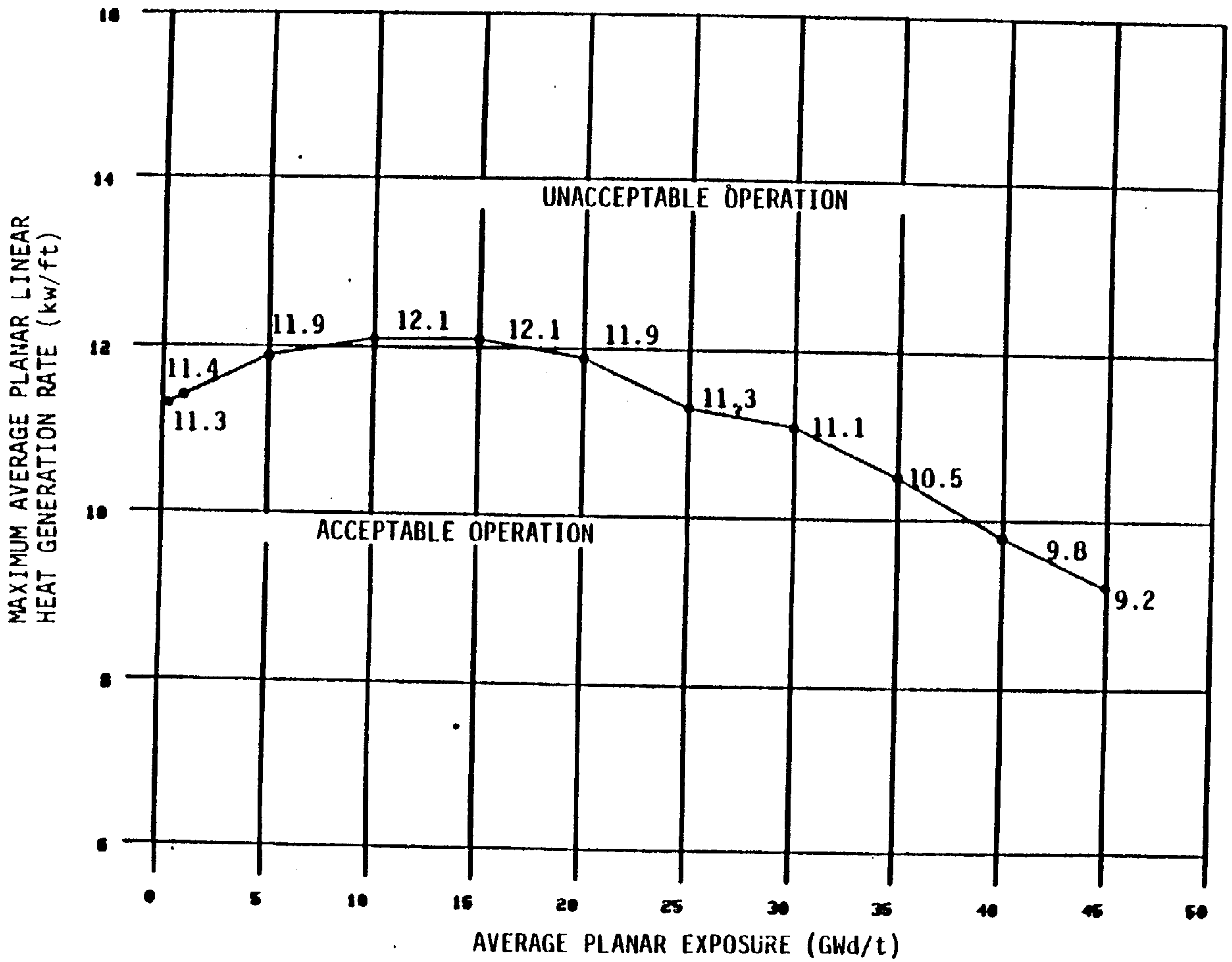


FUEL TYPE IE 711-00GD-100 MIL
MAXIMUM AVERAGE PLANAR LINEAR HEAT GENERATION RATE (MAPLHGR) VE[illegible] AVERAGE PLANAR EXPOSURE

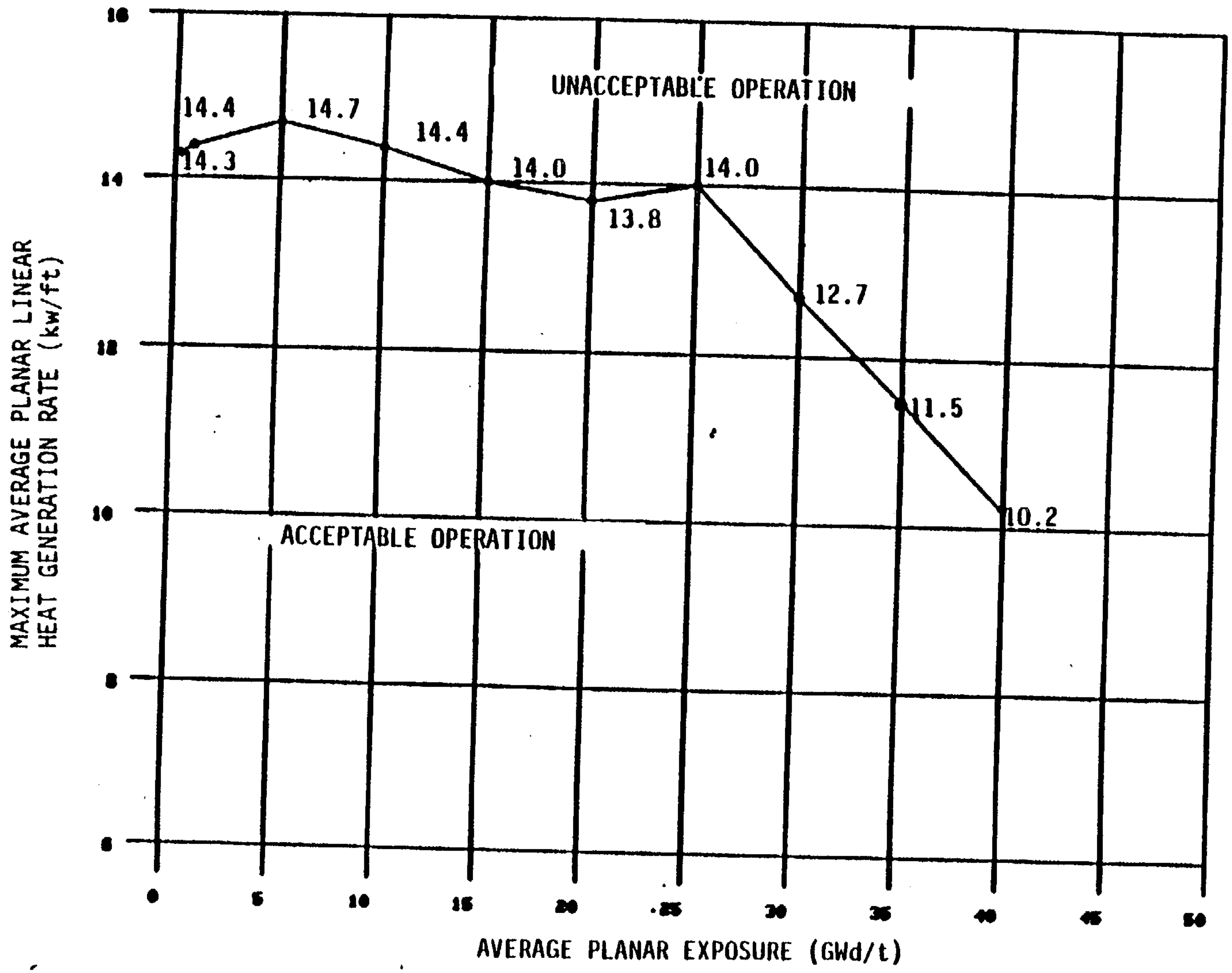
FIGURE 3.2.1-3



FUEL TYPE P8DRB284LA
MAXIMUM AVERAGE PLANAR LINEAR HEAT GENERATION RATE (MAPLHGR) VERSUS AVERAGE PLANAR EXPOSURE
FIGURE 3.2.1-4

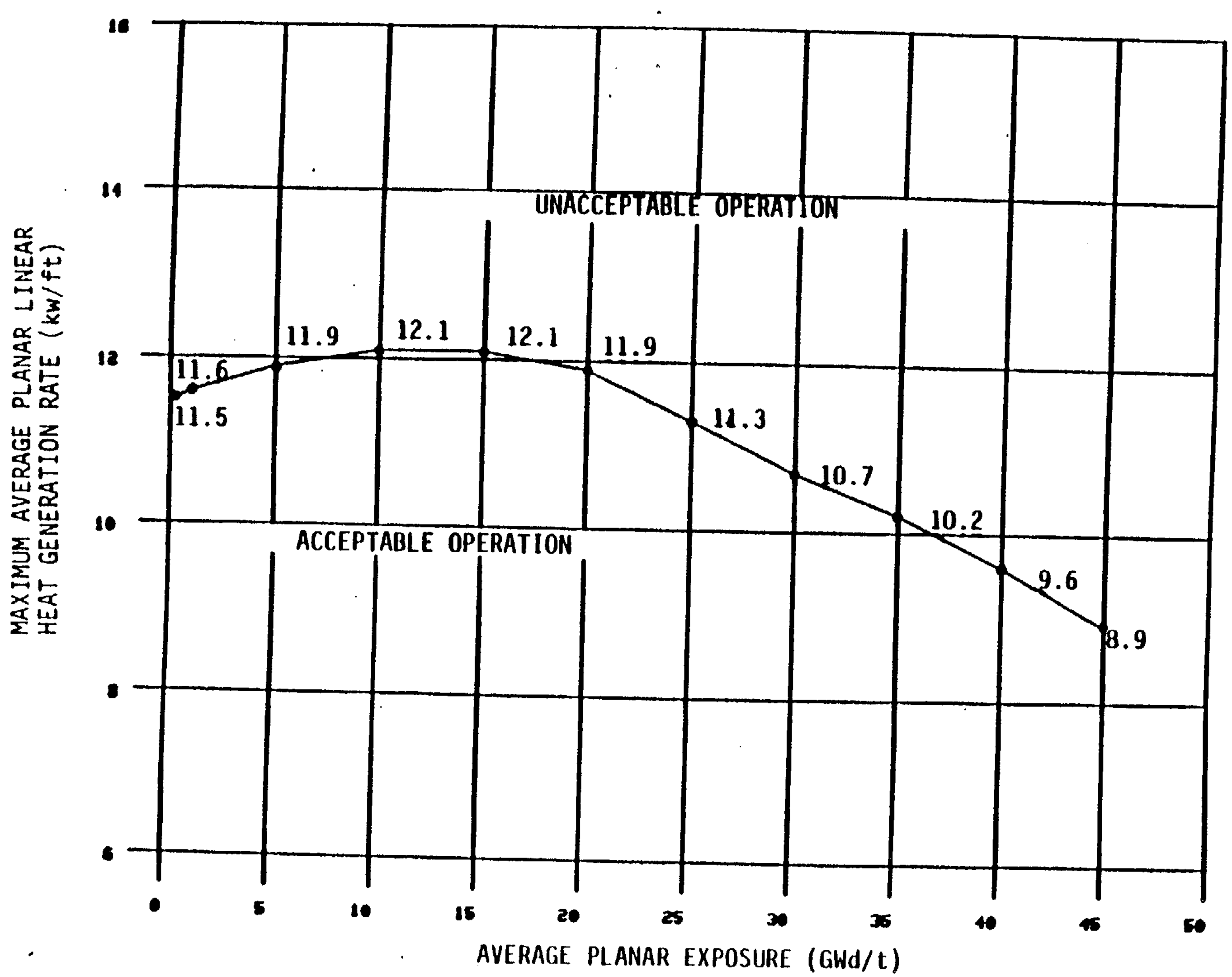


FUEL TYPE P8DRB283
MAXIMUM AVERAGE PLAN[illegible] INEAR HEAT GENERATION
RATE (MAPLHGR) VERSU[illegible] [illegible]ERAGE PLANAR EXPOSURE
FIGURE 3.2.1-5

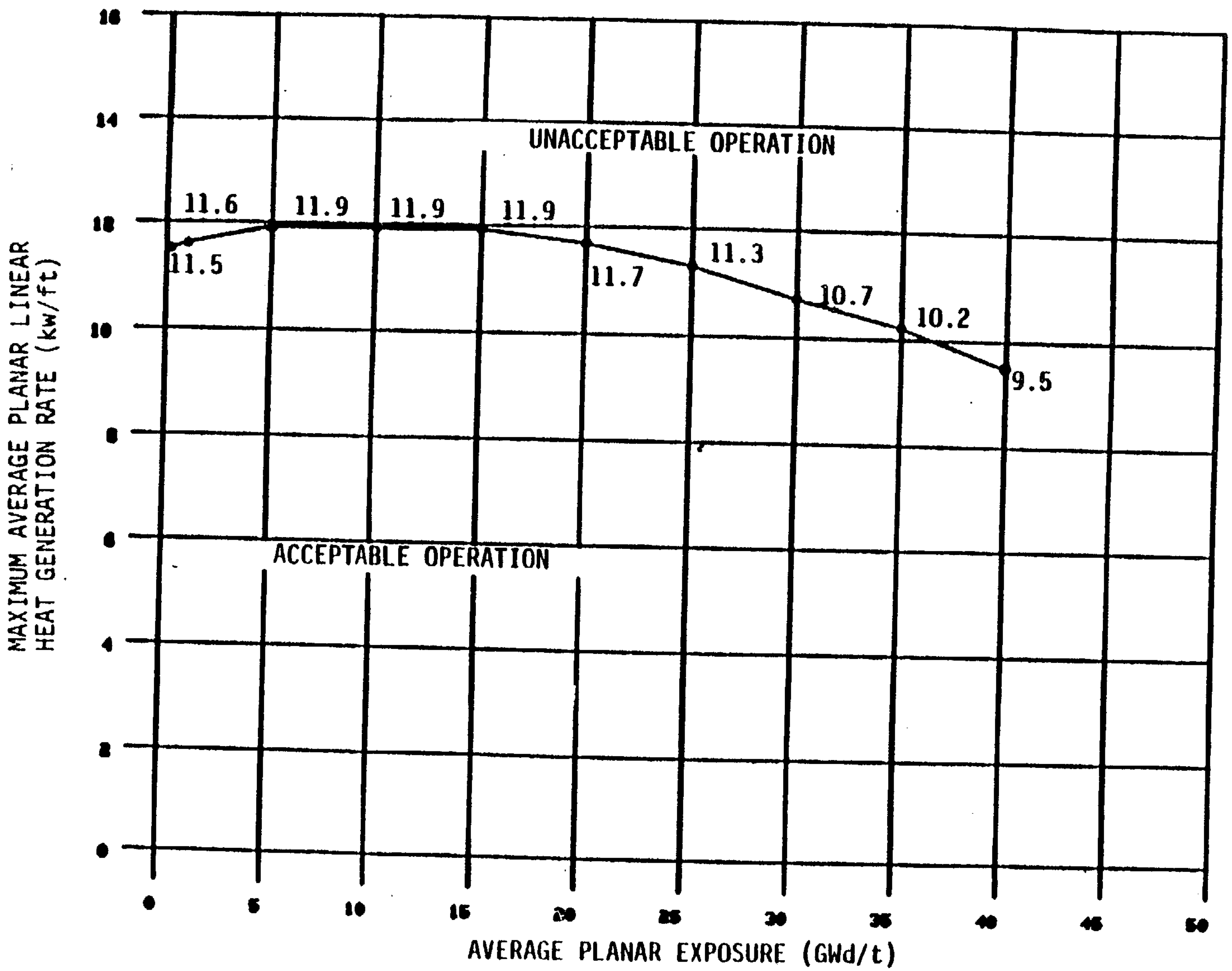


FUEL TYPE HATCH-1 I.C. 1,2,3 (7X7)
MAXIMUM AVERAGE PLANAR LINEAR HEAT GENERATION RATE (MAPLHGR) VERSUS AVERAGE PLANAR EXPOSURE

FIGURE 3.2.1-6



FUEL TYPE P8DRB265H
MAXIMUM AVERAGE PLANAR LINEAR HEAT GENERATION
RATE (MAPLHGR) VERSUS AVERAGE PLANAR EXPOSURE
FIGURE 3.2.1-7



FUEL TYPE 8DRB265H
MAXIMUM AVERAGE PLANAR LINEAR HEAT GENERATION RATE (MAPLHGR) VERSUS AVERAGE PLANAR EXPOSURE

FIGURE 3.2.1-8

3/4.2.2 APRM SETPOINTS

LIMITING CONDITION FOR OPERATION

3.2.2 The APRM flow referenced simulated thermal power scram trip setpoint (S) and control rod block trip setpoint (S_{RB}) shall be established* according to the following relationships:

$$S \leq (0.66W + 51\%)$$

$$S_{RB} \leq (0.66W + 42\%)$$

where: S and S_{RB} are in percent of RATED THERMAL POWER, and
W = Loop recirculation flow in percent of rated flow.

APPLICABILITY: CONDITION 1, when THERMAL POWER $\geq$ 25% of RATED THERMAL POWER.

ACTION:

With S or S_{RB} exceeding the allowable value, initiate corrective action within 15 minutes and continue corrective action so that S and S_{RB} are within the required limits* within 2 hours or reduce THERMAL POWER to less than 25% of RATED THERMAL POWER within the next 4 hours.

SURVEILLANCE REQUIREMENTS

4.2.2 The CMFLPD shall be determined and the APRM flow referenced simulated thermal power scram and control rod block trip setpoints or APRM readings adjusted, as required:

a. At least once per 24 hours,

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

c. Initially and at least once per 12 hours when the reactor is operating with a CMFLPD $\geq$ FRTP.

*With CORE MAXIMUM FRACTION OF LIMITING POWER DENSITY (CMFLPD) greater than the fraction of RATED THERMAL POWER (FRTP), $\frac{\text{THERMAL POWER}}{\text{RATED THERMAL POWER}}$, up to 95% of RATED THERMAL POWER, rather than adjusting the APRM setpoints, the APRM gain may be adjusted such that APRM readings are greater than or equal to 100% times CMFLPD, provided that the adjusted APRM reading does not exceed 100% of RATED THERMAL POWER and the required gain adjustment increment does not exceed 10% of RATED THERMAL POWER.

3/4.2.3 MINIMUM CRITICAL POWER RATIO

LIMITING CONDITION FOR OPERATION

3.2.3 The MINIMUM CRITICAL POWER RATIO (MCPR), as a function of average scram time and core flow, shall be equal to or greater than shown in Figure 3.2.3-1, Figure 3.2.3-2 or Figure 3.2.3-3 multiplied by the K_f shown in Figure 3.2.3-4, where:

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

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

$$\tau_B = 0.834 + 1.65 \left[\frac{N_1}{\sum_{i=1}^{n} N_i}\right]^{1/2} (0.059),$$

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

n = number of surveillance tests performed to date in cycle,
N_i = number of active control rods measured in the i^{th} surveillance test,
τ_i = average scram time to notch 36 of all rods measured in the i^{th} surveillance test, and
N_1 = total number of active rods measured in 4.1.3.2.a.

APPLICABILITY: CONDITION 1, when THERMAL POWER $\geq$ 25% RATED THERMAL POWER

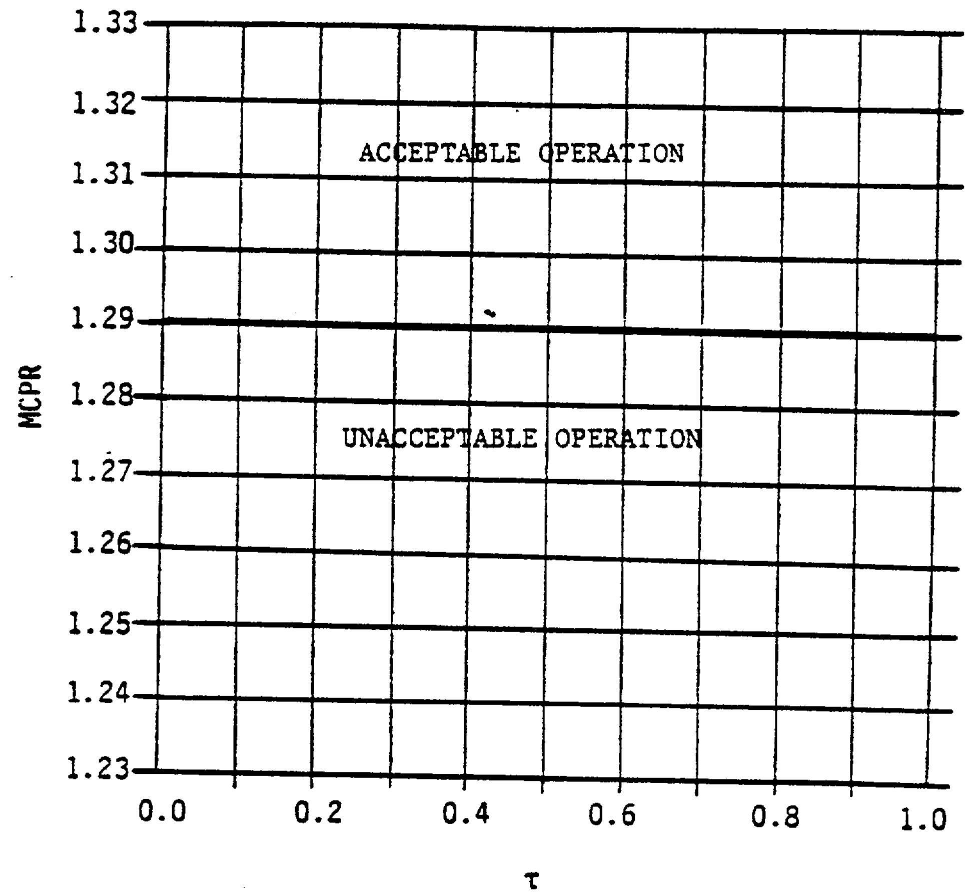
ACTION:

With MCPR less than the applicable limit determined from Figure 3.2.3-1, Figure 3.2.3-2 or Figure 3.2.3-3, initiate corrective action within 15 minutes and continue corrective action so that MCPR is equal to or greater than the applicable limit within 2 hours or reduce THERMAL POWER to less than 25% of RATED THERMAL POWER within the next 4 hours.

SURVEILLANCE REQUIREMENTS

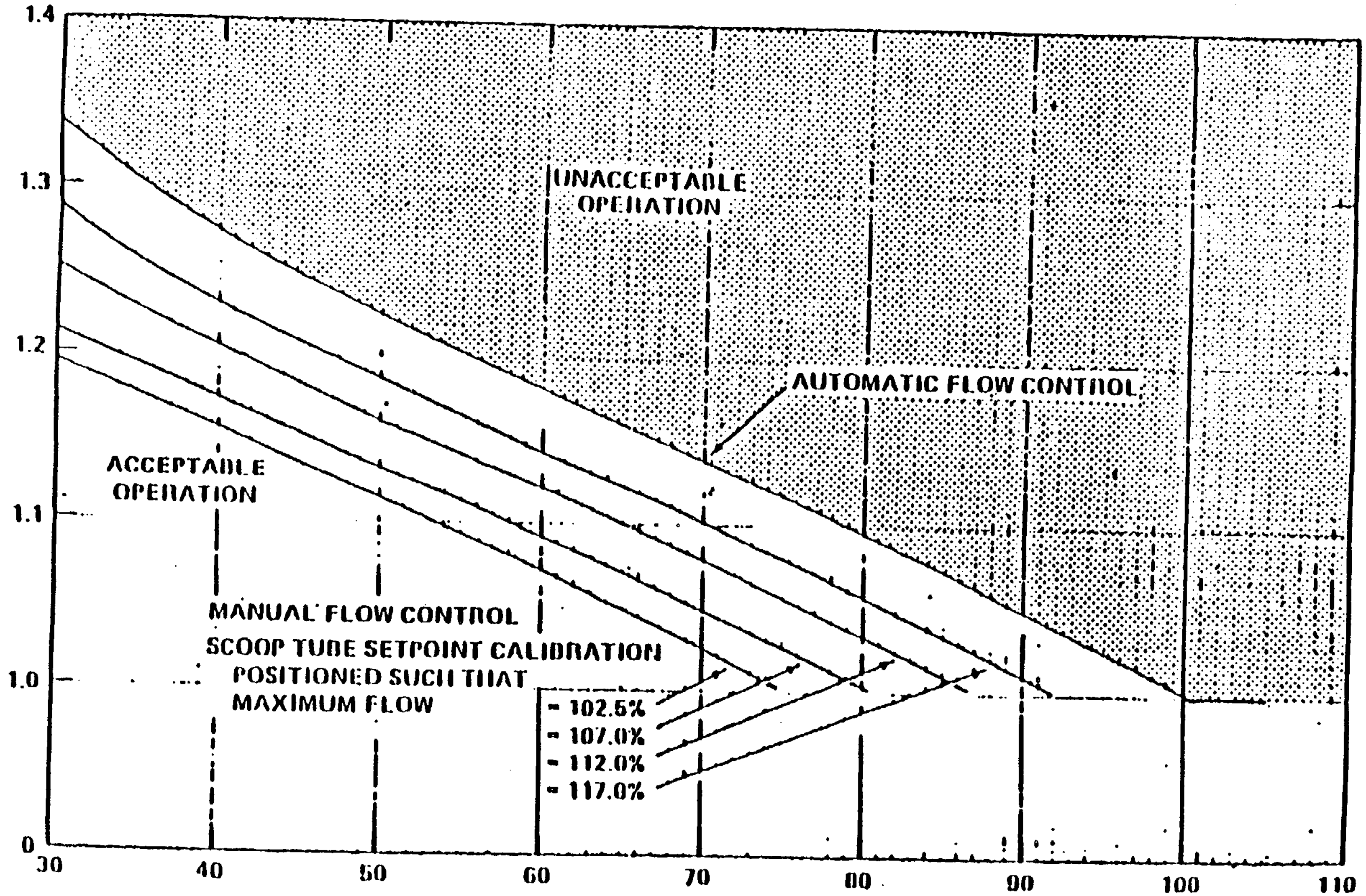
4.2.3 The MCPR limit at rated flow shall be determined for each type of fuel (8X8R, P8X8R, and 7X7) from Figures 3.2.3-1, 3.2.3-2, and 3.2.3-3 using:

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



MCPR LIMIT FOR 7X7 FUEL AT RATED FLOW

FIGURE 3.2.3-3



Core Flow, % of Rated Core Flow

K_F Factor

FIGURE 3- 4

POWER DISTRIBUTION LIMITS

3/4.2.4 LINEAR HEAT GENERATION RATE

LIMITING CONDITION FOR OPERATION

3.2.4 All LINEAR HEAT GENERATION RATES (LHGRs) shall not exceed 13.4 Kw/ft for 8X8R/P8X8R fuel or 18.0 Kw/ft for 7X7 fuel.

APPLICABILITY: CONDITION 1, when THERMAL POWER $\geq$25% of RATED THERMAL POWER.

ACTION:

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

SURVEILLANCE REQUIREMENTS

4.2.4 LHGRs shall be determined to be equal to or less than the limit;

a. At least once per 24 hours,

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

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

TABLE 3.3.2-1

ISOLATION ACTUATION INSTRUMENTATION

TRIP FUNCTION	VALVE GROUPS OPERATED BY SIGNAL(a)	MINIMUM NUMBER OPERABLE CHANNELS PER TRIP SYSTEM(b)(c)	APPLICABLE OPERATIONAL CONDITION	ACTION
1. PRIMARY CONTAINMENT ISOLATION				
a. Reactor Vessel Water Level				
1. Low (2B21-N017 A, B, C, D)	2, 5, 6, 10, 11, 12 *	2	1, 2, 3	20
2. Low-Low (2B21-N024 A, B and 2B21-N025 A, B)	#	2	1, 2, 3	20
3. Low-Low-Low (2B21-N024 A, B and 2B21-N025 A, B)	1	2	1, 2, 3	20
b. Drywell Pressure - High (2C71-N002 A, B, C, D)	2, 5, 6, 7, 10, 11, 12, #,*	2	1, 2, 3	20
c. Main Steam Line				
1. Radiation - High (2D11-K603 A, B, C, D)	1, 12, #, (d)	2	1, 2, 3	21
2. Pressure - Low (2B21-N015 A, B, C, D)	1	2	1	22
3. Flow - High (2B21-N006 A, B, C, D) (2B21-N007 A, B, C, D) (2B21-N008 A, B, C, D) (2B21-N009 A, B, C, D)	1, #	2/line	1, 2, 3	21
d. Main Steam Line Tunnel High Temperature - High (2B21-N010 A, B, C, D) (2B21-N011 A, B, C, D) (2B21-N012 A, B, C, D) (2B21-N013 A, B, C, D)	1	2/line(e)	1, 2, 3	21
e. Condenser Vacuum - Low (2B21-N056 A, B, C, D)	1	2	1, 2,(f), 3(f)	23
f. Turbine Building Area Temperature - High (2U61-R001, 2U61-R002, 2U61-R003, 2U61-R004)	1	2(e)	1, 2, 3	21

TABLE 3.3.2-1 (Continued)

ISOLATION ACTUATION INSTRUMENTATION

TRIP FUNCTION	VALVE GROUPS OPERATED BY SIGNAL(a)	MINIMUM NUMBER OPERABLE CHANNELS PER TRIP SYSTEM(b)(c)	APPLICABLE OPERATIONAL CONDITION	ACTION
2. SECONDARY CONTAINMENT ISOLATION				
a. Reactor Building Exhaust Radiation - High (2D11-K609 A, B, C, D)	6, 10, 12, *	2	1,2,3,5 and**	24
b. Drywell Pressure - High (2C71-N002 A, B, C, D)	2, 5, 6, 7, 10, 11, 12, #, *	2	1, 2, 3	24
c. Reactor Vessel Water Level - Low (2B21-N017 A, B, C, D)	2, 5, 6, 10, 11, 12, *	2	1, 2, 3	24
d. Refueling Floor Exhaust Radiation - High (2D11-K611 A, B, C, D)	6, 10, 12, #, *	2	1,2,3,5 and**	24
3. REACTOR WATER CLEANUP SYSTEM ISOLATION				
a. Δ Flow - High (2G31-N603 A, B)	5	1	1, 2, 3	25
b. Area Temperature - High (2G31-N600 A, B, C, D, E, F)	5	1	1, 2, 3	25
c. Area Ventilation Δ Temp. - High (2G31-N602 A, B, C, D, E, F)	5	1	1, 2, 3	25
d. SLCS Initiation (NA)	5(g)	NA	1, 2, 3	25
e. Reactor Vessel Water Level - Low (2B21-N017 A, B, C, D)	2, 5, 6, 10, 11, 12	2	1, 2, 3	25

TABLE 3.3.2-1 (Continued)

ISOLATION ACTUATION INSTRUMENTATION

ACTION

ACTION 20 - Be in at least HOT SHUTDOWN within 6 hours and in COLD SHUTDOWN within the next 30 hours.

ACTION 21 - Be in at least STARTUP with the main steam line isolation valves closed within 2 hours or be in at least HOT SHUTDOWN within 6 hours and in COLD SHUTDOWN within the next 30 hours.

ACTION 22 - Be in at least STARTUP within 2 hours.

ACTION 23 - Be in at least STARTUP with the Group 1 isolation valves closed within 2 hours or in at least HOT SHUTDOWN within 6 hours.

ACTION 24 - Establish SECONDARY CONTAINMENT INTEGRITY with the standby gas treatment system operating within one hour.

ACTION 25 - Isolate the reactor water cleanup system.

ACTION 26 - Close the affected system isolation valves and declare the affected system inoperable.

ACTION 27 - Verify power availability to the bus at least once per 12 hours or close the affected system isolation valves and declare the affected system inoperable.

ACTION 28 - Close the shutdown cooling supply and reactor vessel head spray isolation valves unless reactor steam dome pressure $\leq$ 135 psig.

NOTES

\# Actuates operation of the main control room environmental control system in the pressurization mode of operation.

* Actuates the standby gas treatment system.

** When handling irradiated fuel in the secondary containment.

a. See Specification 3.6.3.1, Table 3.6.3.1-1 for valves in each valve group.

b. A channel may be placed in an inoperable status for up to 2 hours for required surveillance without placing the trip system in the tripped condition provided at least one other OPERABLE channel in the same trip system is monitoring that parameter.

c. With a design providing only one channel per trip system, an inoperable channel need not be placed in the tripped condition where this would cause the Trip Function to occur. In these cases, the inoperable channel shall be restored to OPERABLE status within 2 hours or the ACTION required by Table 3.3.2-1 for that Trip Function shall be taken.

d. Trips the mechanical vacuum pumps.

e. A channel is OPERABLE if 2 of 4 instruments in that channel are OPERABLE.

f. May be bypassed with reactor steam pressure $\leq$ 1045 psig and all turbine stop valves closed.

g. Closes only RWCU outlet isolation valve 2G31-F004.

h. Alarm only.

i. Adjustable up to 60 minutes.

TABLE 3.3.2-2

ISOLATION ACTUATION INSTRUMENTATION SETPOINTS

TRIP FUNCTION	TRIP SETPOINT	ALLOWABLE VALUE
1. PRIMARY CONTAINMENT ISOLATION		
a. Reactor Vessel Water Level		
1. Low	$\geq$ 12.5 inches*	$\geq$ 12.5 inches*
2. Low Low	$\geq$ -38 inches*	$\geq$ -38 inches*
3. Low Low Low	$\geq$ -146.5 inches*	$\geq$ -146.5 inches*
b. Drywell Pressure - High	$\leq$ 2 psig	$\leq$ 2 psig
c. Main Steam Line		
1. Radiation - High	$\leq$ 3 x full power background	$\leq$ 3 x full power background
2. Pressure - Low	$\geq$ 825 psig	$\geq$ 825 psig
3. Flow - High	$\leq$ 140% of rated flow	$\leq$ 140% of rated flow
d. Main Steam Line Tunnel Temperature - High	$\leq$ 200°F	$\leq$ 200°F
e. Condenser Vacuum - Low	$\geq$ 7" Hg vacuum	$\geq$ 7" Hg vacuum
f. Turbine Building Area Temp.-High	$\leq$ 200°F	$\leq$ 200°F
2. SECONDARY CONTAINMENT ISOLATION		
a. Reactor Building Exhaust Radiation - High	$<$ 60 mr/hr**	$<$ 60 mr/hr**
b. Drywell Pressure - High	$<$ 2 psig	$<$ 2 psig
c. Reactor Vessel Water Level - Low	$>$ 12.5 inches*	$\geq$ 12.5 inches*
d. Refueling Floor Exhaust Radiation - High	$\leq$ 20 mr/hr**	$<$ 20 mr/hr**

* Bases Figure B 3/4 3-1.
**Initial setpoint. Final setpoint to be determined during startup testing.

TABLE 3.3.2-3

ISOLATION SYSTEM INSTRUMENTATION RESPONSE TIME

TRIP FUNCTION	RESPONSE TIME (Seconds)#
1. PRIMARY CONTAINMENT ISOLATION	
a. Reactor Vessel Water Level	
1. Low	≤ 13*
2. Low Low	≤ 13*
3. Low Low Low, except MSIVs	≤ 13*
b. Drywell Pressure - High	≤ 13*
c. Main Steam Line	
1. Radiation - High***	≤ 1.0**
2. Pressure - Low	≤ 13*
3. Flow - High	≤ 1.0**
4. Reactor Vessel Water Level - Low Low Low	≤ 1.0**
d. Main Steam Line Tunnel Temperature - High	≤ 13*
e. Condenser Vacuum - Low	NA
f. Turbine Building Area Temperature - High	NA
2. SECONDARY CONTAINMENT ISOLATION	
a. Reactor Building Exhaust Radiation - High***	≤ 13*
b. Drywell Pressure - High	≤ 13*
c. Reactor Vessel Water Level - Low	≤ 13*
d. Refueling Floor Exhaust Radiation - High***	≤ 13*

*The isolation actuation instrumentation response time shall be measured and recorded as a part of the ISOLATION SYSTEM RESPONSE TIME. Response time specified is diesel generator start delay time assumed in accident analysis.

**Isolation actuation instrumentation response time.

***Radiation detectors are exempt from response time testing. Response time shall be measured from detector output or the input of the first electronic component in the channel.

#Times to be added to valve movement times shown in Tables 3.6.3-1, 3.6.5.2-1 and 3.9.5.2-1 to obtain ISOLATION SYSTEM RESPONSE TIME for each valve.

TABLE 3.3.2-3 (Continued)

ISOLATION SYSTEM INSTRUMENTATION RESPONSE TIME

TRIP FUNCTION	RESPONSE TIME (Seconds)#
4. REACTOR WATER CLEANUP SYSTEM ISOLATION	
a. Δ Flow - High	$\leq$ 13*
b. Area Temperature - High	$\leq$ 13*
c. Area Ventilation Temperature ΔT - High	$\leq$ 13*
d. SLCS Initiation	NA
e. Reactor Vessel Water Level-Low Low	$\leq$ 13*
4. HIGH PRESSURE COOLANT INJECTION SYSTEM ISOLATION	
a. HPCI Steam Line Flow-High	$\leq$ 13*
b. HPCI Steam Supply Pressure - Low	$\leq$ 13*
c. HPCI Turbine Exhaust Diaphragm Pressure - High	NA
d. HPCI Equipment Room Temperature - High	NA
e. Suppression Pool Area Ambient Temp. - High	NA
f. Suppression Pool Area ΔT - High	NA
g. Suppression Pool Area Temp. Timer Relays	NA
h. Emergency Area Cooler Temperature - High	NA
i. Drywell Pressure - High	$\leq$ 13*
j. Logic Power Monitor	NA
5. REACTOR CORE ISOLATION COOLING SYSTEM ISOLATION	
a. RCIC Steam Line Flow - High	NA
b. RCIC Steam Supply Pressure - Low	NA
c. RCIC Turbine Exhaust Diaphragm Pressure - High	NA
d. Emergency Area Cooler Temperature - High	NA
e. Suppression Pool Area Ambient Temp. - High	NA
f. Suppression Pool Area ΔT - High	NA
g. Suppression Pool Area Temperature Timer Relays	NA
h. Drywell Pressure - High	$\leq$ 13*
i. Logic Power Monitor	NA
6. SHUTDOWN COOLING SYSTEM ISOLATION	
a. Reactor Vessel Water Level - Low	NA
b. Reactor Steam Dome Pressure - High	NA

TABLE 4.3.2-1

ISOLATION ACTUATION INSTRUMENTATION SURVEILLANCE REQUIREMENTS

TRIP FUNCTION	CHANNEL CHECK	CHANNEL FUNCTIONAL TEST	CHANNEL CALIBRATION	OPERATIONAL CONDITIONS IN WHICH SURVEILLANCE REQUIRED
1. PRIMARY CONTAINMENT ISOLATION				
a. Reactor Vessel Water Level				
1. Low	D	M	Q	1, 2, 3
2. Low Low	D	M	Q	1, 2, 3
3. Low Low Low	D	M	Q	1, 2, 3
b. Drywell Pressure - High	NA	M	Q	1, 2, 3
c. Main Steam Line				
1. Radiation - High	D	W(a)	R	1, 2, 3
2. Pressure - Low	NA	M	Q	1
3. Flow - High	D	M	Q	1, 2, 3
d. Main Steam Line Tunnel Temperature - High	NA	R	R	1, 2, 3
e. Condenser Vacuum - Low	NA	M	Q	1, 2#, 3#
f. Turbine Building Area Temp. - High	NA	M	R	1, 2, 3
2. SECONDARY CONTAINMENT ISOLATION				
a. Reactor Building Exhaust Radiation - High	D	M(a)	R	1, 2, 3, 5 and *
b. Drywell Pressure - High	NA	M	Q	1, 2, 3
c. Reactor Vessel Water Level - Low	D	M	Q	1, 2, 3
d. Refueling Floor Exhaust Radiation - High	D	M(a)	Q	1, 2, 3, 5 and *

*When handling irradiated fuel in the secondary containment.
#When reactor steam pressure $\geq$ 1045 psig and/or any turbine stop valve is open.
aInstrument alignment using a standard current source.

TABLE 4.3.2-1 (Continued)

ISOLATION ACTUATION INSTRUMENTATION SURVEILLANCE REQUIREMENTS

TRIP FUNCTION	CHANNEL CHECK	CHANNEL FUNCTIONAL TEST	CHANNEL CALIBRATION	OPERATIONAL CONDITIONS IN WHICH SURVEILLANCE REQUIRED
3. REACTOR WATER CLEANUP SYSTEM ISOLATION				
a. Δ Flow - High	D	M	R	1, 2, 3
b. Area Temperature - High	NA	M	R	1, 2, 3
c. Area Ventilation Δ Temperature - High	NA	M	R	1, 2, 3
d. SLCS Initiation	NA	R	NA	1, 2, 3
e. Reactor Vessel Water Level - Low	D	M	Q	1, 2, 3
4. HIGH PRESSURE COOLANT INJECTION SYSTEM ISOLATION				
a. HPCI Steam Line Flow-High	NA	M	Q	1, 2, 3
b. HPCI Steam Supply Pressure-Low	NA	M	Q	1, 2, 3
c. HPCI Turbine Exhaust Diaphragm Pressure-High	NA	M	Q	1, 2, 3
d. HPCI Equipment Room Temperature - High	NA	M	R	1, 2, 3
e. Suppression Pool Area Ambient Temp. - High	NA	M	R	1, 2, 3
f. Suppression Pool Area ΔT - High	NA	M	R	1, 2, 3
g. Suppression Pool Area Temp. Timer Relays	NA	SA	R	1, 2, 3
h. Emergency Area Cooler Temp. - High	NA	M	R	1, 2, 3
i. Drywell Pressure - High	NA	M	Q	1, 2, 3
j. Logic Power Monitor	NA	R	NA	1, 2, 3

TABLE 3.3.3-1 (Continued)

EMERGENCY CORE COOLING SYSTEM ACTUATION INSTRUMENTATION

TRIP FUNCTION	MINIMUM NUMBER OPERABLE CHANNELS PER TRIP SYSTEM	APPLICABLE OPERATIONAL CONDITIONS#
3. HIGH PRESSURE COOLANT INJECTION SYSTEM		
a. Reactor Vessel Water Level - Low Low (2B21-N031 A,B,C,D)	2	1, 2, 3
b. Drywell Pressure - High (2E11-N011 A,B,C,D)	2	1, 2, 3
c. Condensate Storage Tank Level-Low (2E41-N002, 2E41-N003)	2(b)(c)	1, 2, 3
d. Suppression Chamber Water Level-High (2E41-N015A,B)	2(b)(c)	1, 2, 3
e. Logic Power Monitor (2E41-K1)	1(a)	1, 2, 3
f. Reactor Vessel Water Level-High (2B21-N017 B,D)	2	1, 2, 3
4. AUTOMATIC DEPRESSURIZATION SYSTEM		
a. Drywell Pressure - High (2E11-N011A,B,C,D)	2	1, 2, 3
b. Reactor Vessel Water Level - Low Low Low (2B21-N031 A,B,C,D)	2	1, 2, 3
c. ADS Timer (2B21-K5A,B)	1	1, 2, 3
d. Reactor Vessel Water Level-Low (Permissive) (2B21-N042A,B)	1	1, 2, 3
e. Core Spray Pump Discharge Pressure - High (Permissive) (2E21-N008A,B; 2E21-N009A,B)	2	1, 2, 3
f. RHR (LPCI MODE) Pump Discharge Pressure - High (Permissive) (2E11-N016A,B,C,D; 2E11-N020A,B,C,D)	2/loop	1, 2, 3
g. Control Power Monitor (2B21-K1A,B)	1/bus(a)	1, 2, 3
5. LOW LOW SET S/RV SYSTEM		
a. Reactor Steam Dome Pressure - High (Permissive) (2B21-N620A,B,C,D)	2	1, 2, 3

(a) Alarm only. When inoperable, verify power availability to the bus at least once per 12 hours or declare the system inoperable.

(b) Provides signal to HPCI pump suction valves only.

(c) When either channel of the automatic transfer logic is inoperable, align HPCI pump suction to the suppression pool.

\# HPCI and ADS are not required to be OPERABLE with reactor steam dome pressure < 150 psig.

TABLE 3.3.3-2

EMERGENCY CORE COOLING SYSTEM ACTUATION INSTRUMENTATION SETPOINTS

TRIP FUNCTION	TRIP SETPOINT	ALLOWABLE VALUE
1. CORE SPRAY SYSTEM		
a. Reactor Vessel Water Level - Low Low Low	$\geq$-146.5 inches*	$\geq$-146.5 inches*
b. Drywell Pressure - High	$\leq$2 psig	$\leq$2 psig
c. Reactor Steam Dome Pressure - Low	$\leq$500 psig	$\leq$500 psig
d. Logic Power Monitor	NA	NA
2. LOW PRESSURE COOLANT INJECTION MODE OF RHR SYSTEM		
a. Drywell Pressure - High	$\leq$2 psig	$\leq$2 psig
b. Reactor Vessel Water Level - Low Low Low	$\geq$-146.5 inches*	$\geq$-146.5 inches*
c. Reactor Vessel Shroud Level - High	$\geq$-203.5 inches*	$\geq$-203.5 inches*
d. Reactor Steam Dome Pressure-Low	$\leq$500 psig	$\leq$500 psig
e. Reactor Steam Dome Pressure-Low	$\leq$335 psig	$\leq$335 psig
f. RHR Pump Start - Time Delay Relay		
1) Pumps A, B and D	10 $\pm$ 1 seconds	10 $\pm$ 1 seconds
2) Pump C	0.5 $\pm$ 0.5 seconds	0.5 $\pm$ 0.5 seconds
g. Logic Power Monitor	NA	NA

*See Bases Figure B 3/4 3-1.

TABLE 3.3.3-2 (Continued)

EMERGENCY CORE COOLING SYSTEM ACTUATION INSTRUMENTATION SETPOINTS

TRIP FUNCTION	TRIP SETPOINT	ALLOWABLE VALUE
3. HIGH PRESSURE COOLANT INJECTION SYSTEM		
a. Reactor Vessel Water Level - Low Low	≥ -38 inches*	≥ -38 inches*
b. Drywell Pressure-High	≤ 2 psig	≤ 2 psig
c. Condensate Storage Tank Level - Low	≥ 0 inches**	≥ 0 inches**
d. Suppression Chamber Water Level - High*	≤ 151 inches	≤ 151 inches
e. Logic Power Monitor	NA	NA
f. Reactor Vessel Water Level-High	≤ 58 inches	≤ 58 inches
4. AUTOMATIC DEPRESSURIZATION SYSTEM		
a. Drywell Pressure-High	≤ 2 psig	≤ 2 psig
b. Reactor Vessel Water Level - Low Low Low	≥ -146.5 inches*	≥ -146.5 inches*
c. ADS Timer	≤ 120 seconds	≤ 120 seconds
d. Reactor Vessel Water Level-Low	≥ 12.5 inches*	≥ 12.5 inches*
e. Core Spray Pump Discharge Pressure - High	≥ 100 psig	≥ 100 psig
f. RHR (LPCI MODE) Pump Discharge Pressure - High	≥ 100 psig	≥ 100 psig
g. Control Power Monitor	NA	NA
5. LOW LOW SET S/RV SYSTEM		
a. Reactor Steam Dome Pressure - High	≤ 1054 psig	≤ 1054 psig

* See Bases Figure B 3/4 3-1.
** Equivalent to 10,000 gallons of water in the CST.

TABLE 3.3.3-3

EMERGENCY CORE COOLING SYSTEM RESPONSE TIMES

ECCS	RESPONSE TIME (Seconds)
1. CORE SPRAY SYSTEM	$\leq$ 27
2. LOW PRESSURE COOLANT INJECTION MODE OF RHR SYSTEM	$\leq$ 40
3. HIGH PRESSURE COOLANT INJECTION SYSTEM	$\leq$ 30
4. AUTOMATIC DEPRESSURIZATION SYSTEM	NA
5. ARM LOW LOW SET SYSTEM	$\leq$ 1

TABLE 4.3.3-1

EMERGENCY CORE COOLING SYSTEM ACTUATION INSTRUMENTATION SURVEILLANCE REQUIREMENTS

TRIP FUNCTION	CHANNEL CHECK	CHANNEL FUNCTIONAL TEST	CHANNEL CALIBRATION	OPERATIONAL CONDITIONS IN WHICH SURVEILLANCE REQUIRED
1. CORE SPRAY SYSTEM				
a. Reactor Vessel Water Level - Low Low Low	D	M	Q	1, 2, 3, 4, 5
b. Drywell Pressure - High	NA	M	Q	1, 2, 3
c. Reactor Steam Dome Pressure - Low	NA	M	Q	1, 2, 3, 4, 5
d. Logic Power Monitor	NA	R	NA	1, 2, 3, 4, 5
2. LOW PRESSURE COOLANT INJECTION MODE OF RHR SYSTEM				
a. Drywell Pressure - High	NA	M	Q	1, 2, 3
b. Reactor Vessel Water Level - Low Low Low	D	M	Q	1, 2, 3, 4*, 5*
c. Reactor Vessel Shroud Level - High	D	M	Q	1, 2, 3, 4*, 5*
d. Reactor Steam Dome Pressure - Low	NA	M	Q	1, 2, 3, 4*, 5*
e. Reactor Steam Dome Pressure - Low	NA	M	Q	1, 2, 3, 4*, 5*
f. RHR Pump Start-Time Delay Relay	NA	NA	R	1, 2, 3, 4*, 5*
g. Logic Power Monitor	NA	R	NA	1, 2, 3, 4*, 5*

*Not applicable when two core spray system subsystems are OPERABLE per Specification 3.5.3.1.

TABLE 4.3.3-1 (Continued)

EMERGENCY CORE COOLING SYSTEM ACTUATION INSTRUMENTATION SURVEILLANCE REQUIREMENTS

TRIP FUNCTION	CHANNEL CHECK	CHANNEL FUNCTIONAL TEST	CHANNEL CALIBRATION	OPERATIONAL CONDITIONS IN WHICH SURVEILLANCE REQUIRED#
3. HIGH PRESSURE COOLANT INJECTION SYSTEM				
a. Reactor Vessel Water Level - Low Low	D	M	Q	1, 2, 3
b. Drywell Pressure-High	NA	M	Q	1, 2, 3
c. Condensate Storage Tank Level - Low	NA	M	Q	1, 2, 3
d. Suppression Chamber Water Level - High	NA	M	Q	1, 2, 3
e. Logic Power Monitor	NA	R	NA	1, 2, 3
f. Reactor Vessel Water Level-High	NA	M	Q	1, 2, 3
4. AUTOMATIC DEPRESSURIZATION SYSTEM				
a. Drywell Pressure-High	NA	M	Q	1, 2, 3
b. Reactor Vessel Water Level - Low Low Low	D	M	Q	1, 2, 3
c. ADS Timer	NA	NA	R	1, 2, 3
d. Reactor Vessel Water Level - Low	D	M	Q	1, 2, 3
e. Core Spray Pump Discharge Pressure - High	NA	M	Q	1, 2, 3
f. RHR (LPCI MODE) Pump Discharge Pressure - High	NA	M	Q	1, 2, 3
g. Control Power Monitor	NA	R	NA	1, 2, 3
5. LOW LOW SET S/RV SYSTEM				
a. Reactor Steam Dome Pressure - High	S	M	R	1, 2, 3

HPCI and ADS are not required to be OPERABLE with reactor steam dome pressure ≤ 150 psig.

POST-ACCIDENT MONITORING INSTRUMENTATION

LIMITING CONDITION FOR OPERATION

3.3.6.4 The post-accident monitoring instrumentation channels shown in Table 3.3.6.4-1 shall be OPERABLE.

APPLICABILITY: CONDITIONS 1 and 2.

ACTION:

a. With one or more of the above required post-accident monitoring channels inoperable, either restore the inoperable channel(s) to OPERABLE status within 30 days or be in at least HOT SHUTDOWN within the next 12 hours.

b. The provisions of Specification 3.0.4 are not applicable.

SURVEILLANCE REQUIREMENTS

4.3.6.4 Each of the above required post-accident monitoring instrumentation channels shall be demonstrated OPERABLE by performance of the CHANNEL CHECK and CHANNEL CALIBRATION operations at the frequencies shown in Table 4.3.6.4-1.

TABLE 3.3.6.4-1

POST-ACCIDENT MONITORING INSTRUMENTATION

INSTRUMENT	MINIMUM CHANNELS OPERABLE
1. Reactor Vessel Pressure (2C32-R605 A, B, C)	2
2. Reactor Vessel Water Level (2B21-R610, 2B21-R615)	2
3. Suppression Chamber Water Level (2T48-R622 A, B)	2
4. Suppression Chamber Water Temperature (2T47-R626, 2T47-R627)	2
5. Suppression Chamber Pressure (2T48-R608, 2T48-R609)	2
6. Drywell Pressure (2T48-R608, 2T48-R609)	2
7. Drywell Temperature (2T47-R626, 2T47-R627)	2
8. Post-LOCA Gamma Radiation (2D11-K622 A, B, C, D)	2
9. Drywell H_2 -O_2 Analyzer (2P33-R601 A, B)	2
10.a) Safety/Relief Valve Position Primary Indicator (2B21-N301 A-H and K-M)	*
b) Safety/Relief Valve Position Secondary Indicator (2B21-N004 A-H and K-M)	*

*If either the primary or secondary indication is inoperable, the torus temperature will be monitored at least once per shift to observe any unexplained temperature increases which might be indicative of an open SRV. With both the primary and secondary monitoring channels of an SRV inoperable, either verify that the S/RV is closed through monitoring the backup low low set logic position indicators (2B21-N302 A-H and K-M) at least once per shift or restore sufficient inoperable channels such that no more than one SRV has both primary and secondary channels inoperable within 7 days or be in at least hot shutdown within the next 12 hours.

REACTOR COOLANT SYSTEM

IDLE RECIRCULATION LOOP STARTUP

LIMITING CONDITION FOR OPERATION

3.4.1.3 An idle recirculation loop shall not be started unless the temperature differential between the reactor coolant within the dome and the bottom head drain is ≤ 145°F, and

a. The temperature differential between the reactor coolant within the idle loop to be started up and the coolant in the reactor pressure vessel is ≤ 50°F when both loops have been idle, or

b. The temperature differential between the reactor coolant within the idle and operating recirculation loops is ≤ 50°F when only one loop has been idle, and the operating loop flow rate is ≤ 50% of rated loop flow.

APPLICABILITY: CONDITIONS 1, 2, 3 and 4.

ACTION:

With temperature differences and/or flow rate exceeding the above limits, suspend startup of any idle recirculation loop.

SURVEILLANCE REQUIREMENTS

4.4.1.3 The temperature differential and flow rate shall be determined to be within the limit within 30 minutes prior to startup of an idle recirculation loop.

3/4.4.2 SAFETY/RELIEF VALVES

LIMITING CONDITION FOR OPERATION

3.4.2.1 The safety valve function of the following reactor coolant system safety/relief valves shall be OPERABLE with the mechanical lift settings within ± 1% of the indicated pressures*.

4 Safety-relief valves @ 1090 psig.
4 Safety-relief valves @ 1100 psig**.
3 Safety-relief valves @ 1110 psig**.

APPLICABILITY: CONDITIONS 1, 2 and 3.

ACTION:

a. For low-low set valves, take the action required by Specification 3.4.2.2. For ADS valves, take the action required by Specification 3.5.2.
b. With one or more safety/relief valves stuck open, place the reactor mode switch in the Shutdown position.
c. With one or more S/RV tailpipe pressure switches of an S/RV declared inoperable and the associated S/RV(s) otherwise indicated to be open, place the reactor mode switch in the shutdown position.
d. With one S/RV tailpipe pressure switch of an S/RV declared inoperable and the associated S/RV(s) otherwise indicated to be closed, plant operation may continue. Remove the function of that pressure switch from the low low set logic circuitry until the next COLD SHUTDOWN. Upon COLD SHUTDOWN, restore the pressure switch(s) to OPERABLE status before STARTUP.
e. With both S/RV tailpipe pressure switches of an S/RV declared inoperable and the associated S/RV(s) otherwise indicated to be closed, restore at least one inoperable switch to OPERABLE status within 14 days or be in at least HOT SHUTDOWN within the next 12 hours and in COLD SHUTDOWN within the following 24 hours.

SURVEILLANCE REQUIREMENTS

4.4.2.1 The tail-pipe pressure switches of each safety/relief valve shall be demonstrated OPERABLE by performance of:

a. CHANNEL FUNCTIONAL TEST:
   1. At least once per 31 days, except that all portions of the channel inside the primary containment may be excluded from the CHANNEL FUNCTIONAL TEST, and
   2. At each scheduled outage of greater than 72 hours during which entry is made into the primary containment, if not performed within the previous 31 days.
b. CHANNEL CALIBRATION and verifying the setpoint to be 85 psig, with an allowable tolerance of +15 psig and -5 psig, at least once per 18 months.

* The lift setting pressure shall correspond to ambient conditions of the valves at nominal operating temperature and pressure.

** Up to two inoperable valves may be replaced with spare OPERABLE valves with lower setpoints of 1090 and 1100 psig, respectively, until the next refueling outage.

REACTOR COOLANT SYSTEM

SAFETY/RELIEF VALVES LOW-LOW SET FUNCTION

LIMITING CONDITION FOR OPERATION

3.4.2.2 The relief valve function and the low-low set function of the following reactor coolant system safety/relief valves shall be OPERABLE with the following low-low set function lift settings:

Low Low Set Valve Function	Allowable Value (psig)* Open	Close
Low	≤ 1010	≤ 860
Medium Low	≤ 1025	≤ 875
Medium High	≤ 1040	≤ 890
High	≤ 1050	≤ 900

APPLICABILITY: OPERATIONAL CONDITIONS 1, 2 and 3

ACTION:

a. With the relief valve function and/or the low-low set function of one of the above required reactor coolant system safety/relief valves inoperable, restore the inoperable relief valve function and low-low set function to OPERABLE status within 14 days or be in at least HOT SHUTDOWN within the next 12 hours and in COLD SHUTDOWN within the following 24 hours.

b. With the relief valve function and/or the low-low set function of more than one of the above required reactor coolant system safety/relief valves inoperable, be in at least HOT SHUTDOWN within 12 hours and in COLD SHUTDOWN within the next 24 hours.

SURVEILLANCE REQUIREMENTS

4.4.2.2 The low-low set relief valve function and the low-low set function pressure actuation instrumentation shall be demonstrated OPERABLE by performance of a:

a. CHANNEL FUNCTIONAL TEST, including calibration of the trip unit and the dedicated high steam dome pressure channels**, at least once per month.

b. CHANNEL CALIBRATION, LOGIC SYSTEM FUNCTIONAL TEST and simulated automatic operation of the entire system at least once per refueling outage.

*The lift setting pressure of the valves is defined in subsection 3/4 3.4.2.1. The accuracy of the low-low set setpoints is defined to be the accuracy of the instrumentation controlling the setpoints of the low-low set valves.
**The setpoint for dedicated high steam dome pressure channels is less than or equal to 1054 psig.

BASES

3/4.1.1 SHUTDOWN MARGIN

A sufficient SHUTDOWN MARGIN ensures that (1) the reactor can be made subcritical from all operating conditions, (2) the reactivity transients associated with postulated accident conditions are controllable within acceptable limits, and (3) the reactor will be maintained sufficiently subcritical to preclude inadvertent criticality in the shutdown condition.

Since core reactivity values will vary through core life as a function of fuel depletion and poison burnup, the demonstration of SHUTDOWN MARGIN will be performed in the cold xenon-free condition and shall show the core to be subcritical by at least R + 0.28% ΔK or R + 0.38%ΔK, as appropriate. The value of R in units of %ΔK is the difference between the calculated value of maximum core reactivity during the operating cycle and the calculated beginning-of-life core reactivity. The value of R must be positive or zero and must be determined for each fuel loading cycle. Satisfaction of this limitation can be best demonstrated at the time of fuel loading but the margin must be determined anytime a control rod is incapable of insertion. This reactivity characteristic has been a basic assumption in the analysis of plant performance.

Two different values are supplied in the Limiting Condition for Operation to provide for the different methods of demonstration of the SHUTDOWN MARGIN. The highest worth rod may be determined analytically or by test. The SHUTDOWN MARGIN is demonstrated by an insequence control rod withdrawal at the beginning of life fuel cycle and, if necessary, at any future time in the cycle if the first demonstration indicates that the margin could be reduced as a function of exposure. Observation of subcriticality in this condition assures subcriticality with the most reactive control rod fully withdrawn.

3/4.1.2 REACTIVITY ANOMALIES

Since the SHUTDOWN MARGIN requirement for the reactor is small, a careful check on actual conditions to the predicted conditions is necessary, and the changes in reactivity can be inferred from these comparisons of rod patterns. Since the comparisons are easily done, frequent checks are not an imposition on normal operations. A 1% change is larger than is expected for normal operation so a change of this magnitude should be thoroughly evaluated. A change as large as 1% would not exceed the design conditions of the reactor and is on the safe side of the postulated transients.

3/4.1.3 CONTROL RODS

The specifications of this section ensure that (1) the minimum SHUTDOWN MARGIN is maintained, (2) the control rod insertion times are consistent with those used in the accident analysis, and (3) the potential effects of the rod drop accident are limited. The ACTION statements permit variations from the basic requirements, but at the same time impose more restrictive criteria for continued operation. A limitation on inoperable rods is set such that the resultant effect on total rod worth and scram shape will be kept to a minimum. The requirements for the various scramtime measurements ensure that any indication of systematic problems with rod drives will be investigated on a timely basis.

Damage within the control rod drive mechanism could be a generic problem; therefore, with a control rod immovable because of excessive friction or mechanical interference, operation of the reactor is limited to a time period which is reasonable to determine the cause of the inoperability and at the same time prevent operation with a large number of inoperable control rods.

Control rods that are inoperable for other reasons are permitted to be taken out of service provided that those in the nonfully-inserted position are consistent with the SHUTDOWN MARGIN requirements.

The number of control rods permitted to be inoperable could be more than the eight allowed by the specification, but the occurrence of eight inoperable rods could be indicative of a generic problem and the reactor must be shutdown for investigation and resolution of the problem.

The control rod system is analyzed to bring the reactor subcritical at a rate fast enough to prevent the MCPR from becoming less than 1.07 during the limiting power transient analyzed in Section 15 of the FSAR. This analysis shows that the negative reactivity rates resulting from the scram with the average response of all the drives as given in the specifications provide the required protection and MCPR remains greater than 1.07. The occurrence of scram times longer than those specified should be viewed as an indication of a systematic problem with the rod drives and therefore the surveillance interval is reduced in order to prevent operation of the reactor for long periods of time with a potentially serious problem.

Control rods with inoperable accumulators are declared inoperable and Specification 3.1.3.1 then applies. This prevents a pattern of inoperable accumulators that would result in less reactivity insertion on a scram

3/4.2 POWER DISTRIBUTION LIMITS

BASES

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

3/4.2.1 AVERAGE PLANAR LINEAR HEAT GENERATION RATE

This specification assures that the peak cladding temperature following the postulated design basis loss-of-coolant accident will not exceed the limit specified in 10 CFR 50, Appendix K.

The peak cladding temperature (PCT) following a postulated loss-of-coolant accident is primarily a function of the average heat generation rate of all the rods of a fuel assembly at any axial location and is dependent only secondarily on the rod-to-rod power distribution within an assembly. The peak clad temperature is calculated assuming an LHGR for the highest powered rod which is equal to or less than the design LHGR corrected for densification. This LHGR times 1.02 is used in the heatup code along with the exposure dependent steady state gap conductance and rod-to-rod local peaking factor. The Technical Specification APLHGR is this LHGR of the highest powered rod divided by its local peaking factor. The limiting value for APLHGR is shown in the figures in Technical Specification 3/4.2.1.

The calculational procedure used to establish the APLHGR shown in the figures in Technical Specification 3/4.2.1, is based on a loss-of-coolant accident analysis. The analysis was performed using General Electric (GE) calculational models which are consistent with the requirements of Appendix K to 10 CFR 50. A complete discussion of each code employed in the analysis is presented in Reference 1. Differences in this analysis compared to previous analyses performed with Reference 1 are: (1) the analysis assumes a fuel assembly planar power consistent with 102% of the MAPLHGR shown in the figures in Technical Specification 3/4.2.1; (2) fission product decay is computed assuming an energy release rate of 200 MEV/fission; (3) pool boiling is assumed after nucleate boiling is lost during the flow stagnation period; and (4) the effects of core spray entrainment and counter-current flow limitiation as described in Reference 2, are included in the reflooding calculations.

A list of the significant plant input parameters to the loss-of-coolant accident analysis presented in bases Table B 3.2.1-1.

Bases Table B 3.2.1-1

SIGNIFICANT INPUT PARAMETERS TO THE LOSS-OF-COOLANT ACCIDENT ANALYSIS FOR HATCH-UNIT 2

Plant Parameters:

Core Thermal Power .... 2531 Mwt which corresponds to 105% of license core power*

Vessel Steam Output .... 10.96×10^6 lbm/h which corresponds to 105% of rated steam flow

Vessel Steam Dome Pressure .... 1055 psia

Design Basis Recirculation Line Break Area For:

a. Large Breaks .... 4.0, 2.4, 2.0, 2.1 and 1.0 ft^2

b. Small Breaks .... 1.0, 0.9, 0.4 and 0.07 ft^2

Fuel Parameters:

FUEL TYPE	FUEL BUNDLE GEOMETRY	PEAK TECHNICAL SPECIFICATION LINEAR HEAT GENERATION RATE (kw/ft)	DESIGN AXIAL PEAKING FACTOR	INITIAL MINIMUM CRITICAL POWER RATIO
Initial Core	8 x 8	13.4	1.4	1.18

A more detailed list of input to each model and its source is presented in Section II of Reference 1 and subsection 6.3.3 of the FSAR.

*This power level meets the Appendix K requirement of 102%. The core heatup calculation assumes a bundle power consistent with operation of the highest powered rod at 102% of its Technical Specification linear heat generation rate limit.

3/4.2.2 APRM SETPOINTS

The fuel cladding integrity Safety Limits of Specification 2.1 were based on a power distribution which would yield the design LHGR at RATED THERMAL POWER. The scram setting and rod block functions of the APRM instruments or APRM readings must be adjusted to ensure that the MCPR does not become less than 1.0 in the degraded situation. The scram settings and rod block settings or APRM readings are adjusted in accordance with the formula in this specification when the combination of THERMAL POWER and CMFLPD indicates a higher peaked power distribution to ensure that an LHGR transient would not be increased in the degraded condition.

3/4.2.3 MINIMUM CRITICAL POWER RATIO

The required operating limit MCPRs at steady state operating conditions as specified in Specification 3.2.3 are derived from the established fuel cladding integrity Safety Limit MCPR of 1.07, and an analysis of abnormal operational transients. For any abnormal operating transient analysis evaluation with the initial condition of the reactor being at the steady state operating limit, it is required that the resulting MCPR does not decrease below the Safety Limit MCPR at any time during the transient assuming instrument trip setting as given in Specification 2.2.1.

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

The limiting transient which determines the required steady state MCPR limit is the load rejection trip with failure of the turbine bypass. This transient yields the largest Δ CPR. When added to the Safety Limit MCPR of 1.07 the required minimum operating limit MCPR of Specification 3.2.3 is obtained.

MINIMUM CRITICAL POWER RATIO (Continued)

The evaluation of a given transient begins with the system initial parameters shown in FSAR Table 15.1-6 that are input to a GE-core dynamic behavior transient computer program described in NEDO-10802(3). 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 NEDO-20566(1). The principal result of this evaluation is the reduction in MCPR caused by the transient.

The purpose of the K_f factor is to define operating limits at other than related flow conditions. At less than 100% of rated flow the required MCPR is the product of the operating limit MCPR and the K_f factor. Specifically, the K_f factor provides the required thermal margin to protect against a flow increase transient. The most limiting transient initiated from less than rated flow conditions is the recirculation pump speed up caused by a motor-generator speed control failure.

For operation in the automatic flow control mode, the K_f factors assure that the operating limit MCPR of Specification 3.2.3 will not be violated should the most limiting transient occur at less than rated flow. In the manual flow control mode, the K_f factors assure that the Safety Limit MCPR will not be violated should the most limiting transient occur at less than rated flow.

The K_f factor values shown in Figure 3.2.3-4 were developed generically and are applicable to all BWR/2, BWR/3 and BWR/4 reactors. The K_f factors wre derived using the flow control line corresponding to RATED THERMAL POWER at rated core flow.

For the manual flow control mode, the K_f factors were calculated such that the maximum flow rate, as limited by the pump scoop tube set point and the corresponding THERMAL POWER along the rated flow control line, the limiting bundle's relative power, was adjusted until the MCPR was slightly above the Safety Limit. Using this relative bundle power, the MCPRs were calculated at different points along the rated flow control line corresponding to different core flow. The ratio of the MCPR calculated at a given point of core flow, divided by the operating limit MCPR, determines the K_f.

MINIMUM CRITICAL POWER RATIC (Continued)

For operation in the automatic flow control mode, the same procedure was employed except the initial power distribution was established such that the MCPR was equal to the operating limit MCPR at RATED THERMAL POWER and ratec flow.

The K_f factors shown in Figure 3.2.3-4 are conservative for the General Electric Plant operation because the operating limit MCPRs of Specification 3.2.3 are greater than the original 1.20 operating limit MCPR used for the generic derivation of K_f.

At THERMAL POWER levels less than or equal to 25% of RATED THERMAL POWER, the reactor will be operating at minimum recirculation pump speed and the moderator void content will be very small. For all designated control rod patterns which may be employed at this point, operating plant experience indicated that the resulting MCPR value is in excess of requirements by a considerable margin. With this low void content, any inadvertent core flow increase would only place operation in a more conservative mode relative to MCPR. During initial startup testing of the plant, an MCPR evaluation will be made at 25% of RATED THERMAL POWER with minimum recirculation pump speed. The MCPR margin will thus be demonstrated such that future MCPR evaluation below this power level will be shown to be unnecessary. The daily requirement for calculating MCPR above 25% of RATED THERMAL POWER is sufficient since power distribution shifts are very slow when there have not been significant power or control rod changes. The requirement for calculating MCPR when a limiting control rod pattern is approached ensures that MCPR will be known following a change in THERMAL POWER or power shape, regardless of magnitude that could place operation at a thermal limit.

3/4.2.4 LINEAR HEAT GENERATION RATE

The LHGR specification assures that the linear heat generation rate in any rod is less than the design linear heat generation even if fuel pellet densification is postulated.

BASES

References:

1. General Electric Company Analytical Model for Loss-of-Coolant Analysis in Accordance with 10 CFR 50, Appendix K, NEDO-20566 (Draft), August 1974.

2. General Electric Refill Reflood Calculation (Supplement to SAFE Code Description) transmitted to USAEC by letter, G. L. Gyorey to V. Stello, Jr., dated December 20, 1974.

3. R. B. Linford, Analytical Methods of Plant Transient Evaluations for the GE BWR, February 1973 (NEDO-10802).

MONITORING INSTRUMENTATION (Continued)

FIRE DETECTION INSTRUMENTATION (Continued)

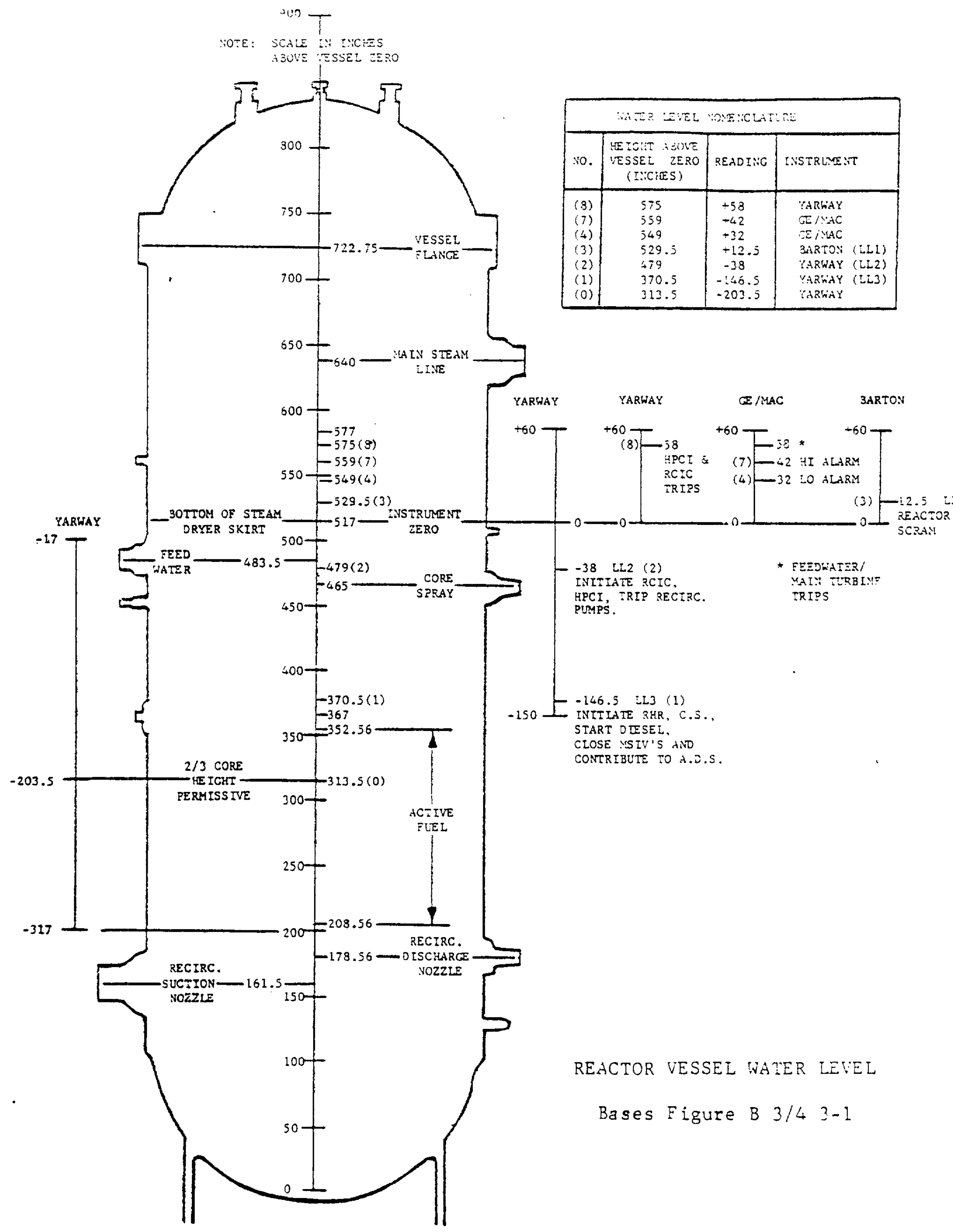
In the event that a portion of the fire detection instrumentation is inoperable, increasing the frequency of fire patrols in the affected areas is required to provide detection capability until the inoperable instrumentation is restored to OPERABILITY.

3/4.3.7 TURBINE OVERSPEED PROTECTION SYSTEM

This specification is provided to ensure that the turbine overspeed protection system instrumentation and the turbine speed control valves are OPERABLE and will protect the turbine from excessive overspeed. Protection from turbine excessive overspeed is required since excessive overspeed of the turbine could generate potentially damaging missiles which could impact and damage safety-related components, equipment or structures.

3/4.3.8 DEGRADED STATION VOLTAGE PROTECTION INSTRUMENTATION

The undervoltate relays shall automatically initite the disconnection of offsite power sources whenever the voltage setpoint and time delay limits have been exceeded. This action shall provide voltage protection for the emergency power systems by preventing sustained degraded voltage conditions due to the offsite power source and interaction between the offsite and onsite emergency power systems. The undervoltage relays have a time delay characteristic that provides protection against both a loss of voltage and degraded voltage condition and thus minimizes the effect of short duration disturbances without exceeding the maximum time delay, including margin, that is assumed in the FSAR accident analyses.



WATER LEVEL NOMENCLATURE			
NO.	HEIGHT ABOVE VESSEL ZERO (INCHES)	READING	INSTRUMENT
(8)	575	+58	YARWAY
(7)	559	+42	GE/MAC
(4)	549	+32	GE/MAC
(3)	529.5	+12.5	BARTON (LL1)
(2)	479	-38	YARWAY (LL2)
(1)	370.5	-146.5	YARWAY (LL3)
(0)	313.5	-203.5	YARWAY

REACTOR VESSEL WATER LEVEL

Bases Figure B 3/4 3-1

3/4.4 REACTOR COOLANT SYSTEM

BASES

3/4.4.1 RECIRCULATION SYSTEM

Operation for longer than 24 hours with a reactor core coolant recirculation loop inoperable is prohibited until an evaluation of the performance of the ECCS during one loop operation has been performed, evaluated and determined to be acceptable.

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

In order to prevent undue stress on the vessel nozzles and bottom head region the recirculation loop temperatures shall be within 50°F of each other prior to startup of an idle loop. Since the coolant in the bottom of the vessel is at a lower temperature than the water in the upper regions of the core, undue stress on the vessel would result if the temperature difference were greater than 145°F. The loop temperature must be within 50°F of the reactor pressure vessel coolant temperature to prevent thermal shock to the recirculation pump and recirculation nozzles.

3/4.4.2.1 SAFETY/RELIEF VALVES

The reactor coolant system safety valve function of the safety-relief valves operate to prevent the system from being pressurized above the Safety Limit of 1325 psig. The system is designed to meet the requirements of the ASME Boiler and Pressure Vessel Code, Section III, for the pressure vessel, and ANSI B31.1, 1975 Code, for the reactor coolant system piping. The capacity of the safety-relief valves is based on the full MSIV closure transient with failed trip scram, position switches, as described in Supplement 5.A of the FSAR, Section 5.A.6.

Demonstration of the safety-relief valve lift settings will occur only during shutdown and will be performed in accordance with the provisions of Section XI of the ASME Boiler and Pressure Vessel Code.

3/4.4.2.2 LOW-LOW SET SYSTEM

The low-low set (LLS) system lowers the opening and closing setpoints on four preselected safety/relief valves (S/RVs). The LLS system lowers the setpoints after any S/RV has opened at its normal steam pilot setpoint when a concurrent high reactor vessel steam dome pressure scram signal is present. The purpose of the LLS is to mitigate the induced high frequency loads on the containment and thrust loads on the SRV discharge line. The LLS system increases the amount of reactor depressurization during an S/RV blowdown because the lowered LLS setpoints keep the four selected LLS S/RVs open for a longer time. The high reactor vessel steam dome pressure signal for the LLS logic is provided by the exclusive analog trip channels. The purpose of installing special dedicated steam dome pressure channels is to maintain separation from the RPS high pressure scram functions.

3/4.4.3 REACTOR COOLANT SYSTEM LEAKAGE

3/4.4.3.1 LEAKAGE DETECTION SYSTEMS

The RCS leakage detection systems are provided to monitor and detect leakage from the reactor coolant pressure boundary.

3/4.4.3.2 OPERATIONAL LEAKAGE

The allowable leakage rates from the reactor coolant system have been based on the predicted and experimentally observed behavior of cracks in pipes. The normally expected background leakage due to equipment design and the detection capability of the instrumentation for determining system leakage was also considered. The evidence obtained from experiments suggests that for leakage somewhat greater than that specified for unidentified leakage the probability is small that the imperfection or crack associated with such leakage would grow rapidly. However, in all cases, if the leakage rates exceed the values specified or the leakage is located and known to be PRESSURE BOUNDARY LEAKAGE the reactor will be shutdown to allow further investigation and corrective action. Service sensitive reactor coolant system Type 304 and 316 austenitic stainless steel piping; i.e., those that are subject to high stress or that certain relatively stagnant, intermittent, or low flow fluids, requires additional surveillance and leakage limits.

3/4.4.4 CHEMISTRY

The water chemistry limits of the reactor coolant system are established to prevent damage to the reactor materials in contact with the coolant. Chloride limits are specified to prevent stress corrosion cracking of the stainless steel. The effect of chloride is not as great when the oxygen concentration in the coolant is low; thus the higher limit on chlorides is permitted during full power operation. During shutdown and refueling operations the temperature necessary for stress corrosion to occur is not present.

Conductivity measurements are required on a continuous basis since changes in this parameter are an indication of abnormal conditions. When the conductivity is within limits, the pH, chlorides and other impurities affecting conductivity must also be within their acceptable limits. With the conductivity meter inoperable, additional samples must be analyzed to ensure that the chlorides are not exceeding the limits.

The surveillance requirements provide adequate assurance that concentrations in excess of the limits will be detected in sufficient time to take corrective action.

5.0 DESIGN FEATURES

5.1 SITE

EXCLUSION AREA

5.1.1 The exclusion area shall be as shown in Figure 5.1.1-1.

LOW POPULATION ZONE

5.1.2 The low population zone coincides with the exclusion area and is also shown in Figure 5.1.1-1.

5.2 CONTAINMENT

CONFIGURATION

5.2.1 The primary containment is a steel structure composed of a series of vertical right cylinders and truncated cones which form a drywell. This drywell is attached to a suppression chamber through a series of vents. The suppression chamber is a steel pressure vessel in the shape of a torus. The primary containment has a total minimum free air volume of 255,978 cubic feet.

DESIGN TEMPERATURE AND PRESSURE

5.2.2 The primary containment is designed and shall be maintained for:

a. Maximum design internal pressure 56 psig.

b. Maximum allowable internal pressure 62 psig.

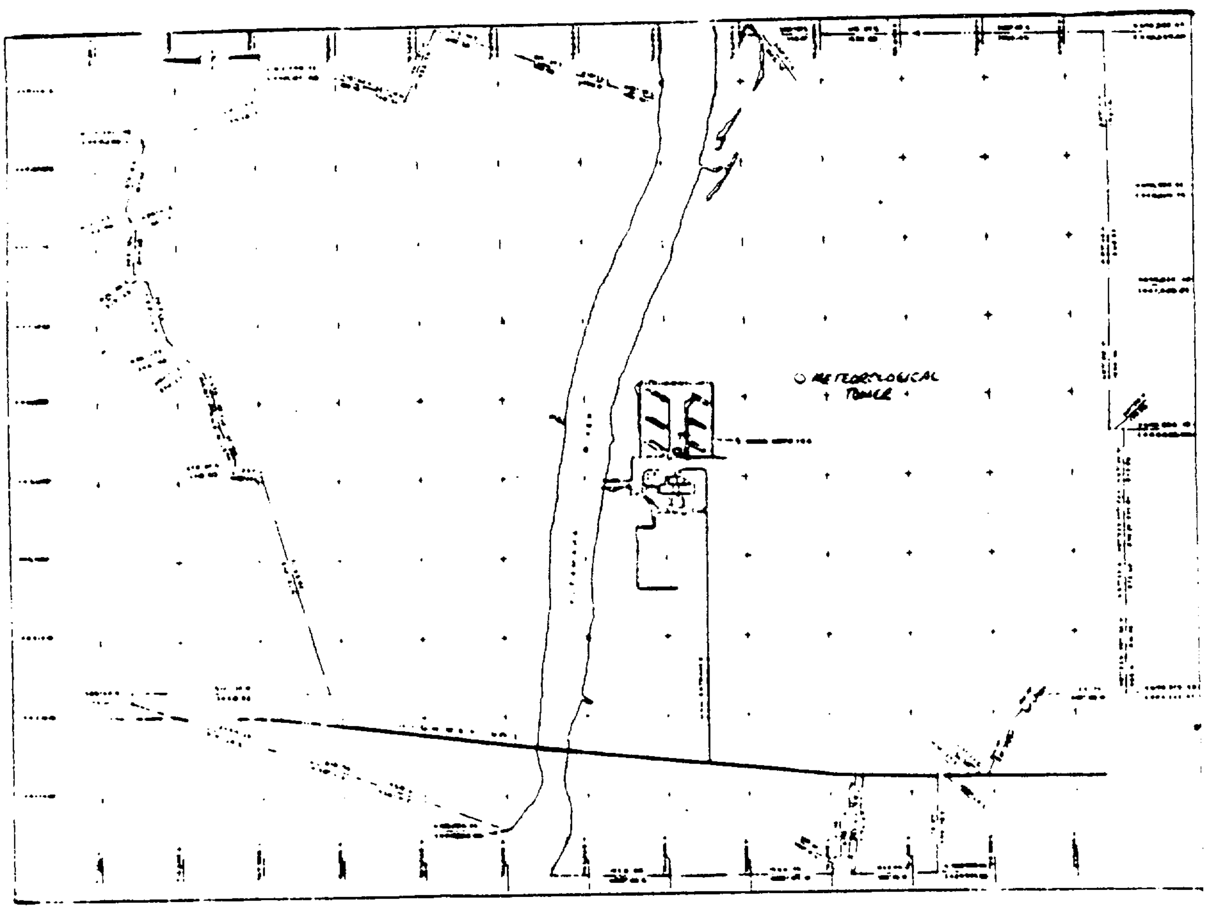
c. Maximum internal temperature 340°F.

d. Maximum external pressure 2 psig.

5.3 REACTOR CORE

FUEL ASSEMBLIES

5.3.1 The initial core shall contain 560 fuel assemblies with each fuel assembly containing 62 fuel rods and 2 water rods clad with Zircaloy -2. Each fuel rod shall have a nominal active fuel length of 150 inches and contain a maximum total weight of 3341 grams uranium. The initial core loading shall have a maximum average enrichment of 1.87 weight percent U-235. Reload fuel shall be similar in physical design to the initial core loading and shall have a maximum average enrichment of 2.90 weight percent U-235. 7X7 fuel containing 49 fuel rods and no water rods may also be inserted.



EXCLUSION AREA AND LOW POPULATION ZONE

FIGURE 5.1.1-1

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

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

SUPPORTING AMENDMENT NO. 33 TO FACILITY OPERATING LICENSE NO. NPF-5

GEORGIA POWER COMPANY
OGLETHORPE POWER CORPORATION
MUNICIPAL ELECTRIC AUTHORITY OF GEORGIA
CITY OF DALTON, GEORGIA

EDWIN I. HATCH NUCLEAR PLANT, UNIT NO. 2
DOCKET NO. 50-366

1.0 Introduction

By letter dated February 23, 1983 (Ref. 1), as supplemented by letter dated April 19, 1983 (Ref. 2), Georgia Power Company (the licensee) has proposed changes to the Technical Specifications (TSs) for Hatch Unit 2 related to design modifications that are being implemented at Hatch Unit 2 to reduce containment loads resulting from plant transients. These TS changes would 1) lower the opening and closing setpoints for actuation of four safety relief valves following initial actuation of any one of the four valves, and 2) lower the main steam isolation valve (MSIV) water level trip setpoint. The February 23, 1983 letter encloses a document entitled "Edwin I. Hatch Plant Unit 2, Docket No. 50-366, Proposed Plant Modifications - Low-Low Set Logic and Lowered MSIV Water Level" (Ref. 3) which provides a detailed description of the proposed changes, a Safety Evaluation of these changes, and proposed TS changes. This document also includes, as appendices, two General Electric Company Reports NEDE 22223 (Ref. 4) and NEDE 22224 (Ref. 5) in support of the proposals.

By letter dated March 30, 1983 (Ref. 6), as supplemented by letters dated May 10, 1983 (Ref. 7), May 20, 1983 (Ref. 8) and May 26, 1983 (Ref. 9), the licensee proposed TS changes that reflect changes to the core design for the third reload (Cycle 4) of Unit 2. The March 30, 1983 letter encloses General Electric Company document Y1003J01A57 (Ref. 10) and General Electric Company letters numbers LMQ:83-018 (Ref. 11) and LMQ: 83-022 (Ref. 12) in support of the proposed TS changes.

2.0 Evaluation

2.1 Low-Low Set Logic and Lowered MSIV Water Level System Response System Response

The low-low set (LLS) relief logic modification for BWRs with Mark 1 containments is designed to prevent multiple subsequent actuations of safety relief valves (SRVs) which might normally be expected during a transient following critical actuation of the SRVs. This in turn will reduce or prevent the discharge loads on the containment and suppression pool structures resulting from subsequent SRV actuations. The discharge loads from subsequent actuations tend to be higher due to the condensation of trapped steam in the safety relief valve discharge line (SRVDL) which results in a higher water leg in the SRVDL, and hence, larger thrust loads on subsequent actuations. In addition, the warmer steam air mixture in the SRVDL results in higher pressure air bubbles in the suppression pool, and therefore, increased torus loads on subsequent actuations.

The LLS for Hatch Unit 2 is an automatic SRV actuation system which, upon initiation, will assign preset opening and closing setpoints to four preselected SRVs. These setpoints are selected such that the LLS controlled SRVs will stay open longer, thus releasing more steam (energy) to the suppression pool, and hence more energy (and time) will be required for repressurization and subsequent SRV openings. The LLS increases the time between (or prevents) subsequent actuations sufficiently to allow the high water leg created from the initial SRV opening to return to (or fall below) its normal water level, thus, reducing thrust loads from subsequent actuations to within their design limits. In addition, since the LLS is designed to limit SRV subsequent actuations to one valve, torus loads will also be reduced.

The lower MSIV water level trip causes the MSIV closure actuation to be changed from a reactor water level 2 signal to a reactor water level 1 signal. This design modification maintains the main condenser availability for a longer time which allows more energy to be released to the main condenser and will result in a slower repressurization rate. The lower MSIV water level trip reduces isolations, SRV challenges and provides some benefit to SRV subsequent actuations.

The TS changes requested by the licensee reflect these logic modifications to 1) lower the opening and closing setpoints for actuation of the four selected SRVs following initial actuation of any one of the four and 2) to lower the MSIV water level trip from level 2 to level 1.

We have reviewed the licensee's submittal as discussed above and find that: 1) the design modifications are compatible with normal operations and other safety systems, 2) the licensee has demonstrated by analysis of limiting transients that the LLS will extend SRV subsequent actuation time sufficiently to clear the water column in the SRV discharge line, and 3) the design modifications will not adversely affect the plant performance or safety margins.

<u>LLS Circuitry Design</u>

The LLS circuitry consists of four redundant logic channels, each of which actuates one SRV. There are eleven SRVs at Hatch Unit 2, seven of which are actuated by the Automatic Depressurization System (ADS). The four non-ADS SRVs will be used for the LLS function. Each of the four LLS controlled SRVs will open when their respective solenoid becomes energized by the LLS logic. The LLS logic channels that actuate SRVs 2B21-F013B and 2B21-F013F, channels A and C respectively, are powered by 25 Vdc from division 1 Class 1E supply 2H11-P925. LLS logic channels B and D (SRVs 2B21-F013G and 2B21-F013D respectively) are supplied from division 2 Class 1E supply 2H11-P926.

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

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

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

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

All four LLS logic channels can be tested at power. Test status lights in the control room indicate when the arming logic relays and contacts have operated satisfactorily during testing. These test lights can also be used to verify proper operation of the seal-in and reset circuits. Each LLS channel provides annunciation in the control room upon loss of power. Test switches are provided to verify operability of this power monitor function. Additional test lights in the control room are used to verify operability of the trip units used in the LLS actuation/control logic during testing. The proposed Hatch Unit 2 TS changes associated with the LLS modification call for monthly channel functional tests of all reactor pressure instrument channels (used for both the arming logic permissive and SRV control/actuation). A channel functional test of all SRV discharge line pressure switches will also be performed monthly (portions of these channels inside the primary containment may be excluded from this test). Channel calibrations and LLS logic system functional tests will be performed during each refueling outage. This test frequency is consistent with the test interval for the ADS and is acceptable.

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

Based on our review as discussed above, we have determined that the LLS modification installed at Hatch Unit 2 is designed to perform its intended function given a single failure. In addition, no single failure in the electrical circuits could be found which would cause more than one SRV to stick open. The LLS is designed in accordance with the requirements of IEEE Standard 279-1971.

Conclusions

On the basis of our review and findings as discussed above, we conclude that the design modifications and the proposed TS changes related to the LLS relief logic and lowered MSIV water level trip are acceptable.

2.2 Cycle 4 Reload

The reload application involves (1) the replacement of 236 spent fuel assemblies with fresh P8x8R fuel assemblies and some previously irradiated 8x8R, (2) the analysis of safety considerations involved in the determination of Cycle 4 operating limits, and (3) the proposed modification of existing TSs and the addition of new TSs to reflect the changes made in the composition of the core for Cycle 4.

Fuel System Design

The Hatch Unit 2 Cycle 4 core will contain 560 fuel assemblies of which 236 will be changed during the current Cycle 4 outage.

About midway through Cycle 3 operation, fuel failures became evident from increasing coolant activity. General Electric Company (GE) believes that the failure mechanism was crud-induced localized corrosion (CILC) inasmuch as the Hatch Unit 2 Cycle 3 core contained batches of Zircaloy cladding believed to be susceptible to this type of degradation. To limit the number of failures, the licensee restricted the power level to 70% of rated power for the latter part of Cycle 3. With assistance from GE, the licensee anticipated (using a GE proprietary waterside-corrosion fuel-failure model) that the Hatch 2 failures would be in the initial-core assemblies and consequently structured the Cycle 4 fuel management scheme accordingly. However, the Cycle 4 outage sipping and visual examination revealed failures in 19 Reload-1 assemblies. Therefore, the licensee has recently revised (Ref. 7) the Cycle 4 fuel loading scheme that was originally described in the Supplemental Reload Licensing Submittal (Ref. 10). Since fuel failures should not be an expected occurrence during an operating cycle and since compliance with TS activity limits does not assure acceptable fuel performance, we asked the licensee to provide some assurance that additional CILC failures would not occur during Cycle 4. The licensee responded (Ref. 8) that all rods from the affected lots/ingot were either discharged or passed a visual and NDT examination and, therefore, that no CILC failures were expected in Cycle 4. We thus conclude that the licensee has provided reasonable assurance that Cycle 4 will be operated without a recurrence of CILC failures.

The 236 replacement assemblies to be installed for Cycle 4 operation will be fresh P8X8R assemblies and previously irradiated assemblies taken from Unit 1. The Unit 1 replacement assemblies will consist of reconstituted Reload-2 (i.e., 8X8R design) assemblies. The cycle 4 core composition is summarized in Table 1. The Cycle 4 fuel assemblies are thus standard designs that have been previously approved for application in the Hatch Units; consequently, the mechanical aspects of the reload fuel require no further NRC review.

Table 1

HATCH-2 CORE INVENTORY

Assembly Designation	Cycle Loaded	Number
8DRB221 (IC)	1	108
P8DRB284LA	2	96
P8DRB283	3	120
P8DRB265H (fresh)	4	132
8DRB265H (irradiated)	4	104
		560

The licensee's analysis of other considerations involved in the determination of Cycle 4 operating limits is presented in the reload safety analysis (Ref. 10). In all fuel-design-related areas except those separately identified, the reload report relies on the generic report, General Electric Standard Application for Reactor Fuel (Ref. 13). Reference 13 has been reviewed and approved by the NRC staff. We conclude that additional staff review of those portions of Reference 13concerning the standard fuel design is unnecessary for the Cycle 4 application.

The licensee's submittal provided both new and revised MAPLHGR limits. The new MAPLHGR limits are for the fresh fuel (P8X8R).

The revised MAPLHGR limits for all fuel types have been extended to accommodate exposures to as much as 45 GWd/ST. These limits were generated by methods previously approved (Ref. 14). Although the methodology used is generically applicable for the MAPLHGR limit determination, we believe that the effects of enhanced fission gas release in high-burnup (i.e., greater than 20 GWd/MTU) were not adequately considered in the fuel performance model. In response to this concern, GE requested (Refs. 15 and 16) that credit for approved, but unapplied, ECCS evaluation model changes and calculated peak cladding temperature margin be used to avoid MAPLHGR penalties at higher burnups. This proposal was found acceptable (Ref. 17) provided that certain plant-specific conditions were met. The licensee has stated (Ref. 7) that the GE proposal is applicable to the Hatch Unit 2 analysis with the exception of three exposure points where the calculated peak cladding temperatures for Hatch Unit 2 fuel exceed those assumed in the GE analysis. In those three cases, the peak cladding temperature for the Hatch Unit 2 fuel is less than 20°F larger than the limiting temperature in the generic analysis. The licensee has also stated that a temperature change of less than 20°F (per 10CFR50, Appendix K, Section II.1.b) is not considered significant. We accept this conclusion and conclude that the MAPLHGR Limits proposed for Cycle 4 operation of Hatch Unit 2 are acceptable.

Thermal and Hydraulic Design

A safety limit MCPR has been imposed to assure than 99.9 percent of the fuel rods in the core are not expected to experience boiling transition during normal and anticipated operational transients. As stated in Reference 13, the approved safety limit MCPR is 1.07. The safety limit MCPR of 1.07 is used for Hatch 2 Cycle 4 operation.

The most limiting events have been analyzed by the licensee to determine which event could potentially induce the largest reduction in the initial critical power ratio (ΔCPR). The ΔCPR values given in Section 9 of Reference 10 are plant specific values calculated by the methods including ODYN Methods. The calculated ΔCPRs are adjusted to reflect either Option A or Option B ΔCPRs by employing the conversion methods described in Reference 18. The MCPR values are determined by adding the adjusted ΔCPRs to the safety limit MCPR. Section 11 of Reference 10 presents both the cycle MCPR values of the pressurization and non-pressurization transients. The maximum cycle MCPR values (Options A and B) in Section 11 are specified as the operating limit MCPRs and incorporated into the TSs. For the MCPR limiting event, feedwater controller failure to maximum demand, the analysis has assumed operation of the high water level (Level 8) trip and the turbine bypass systems. We informed the licensee by the letter dated May 12, 1983 (Ref. 19) that since operation of the Level 8 trip and the turbine bypass system are assumed in the analysis, and neither of these systems have been demonstrated to be qualified to operate (i.e., not safety grade), TSs are required to assure their operability. The licensee's response (Ref. 9) disagreed with the need for these TSs and indicated that they would like to discuss this requirement further prior to committing to permanent TSs. However, they submitted proposed TSs for the Level 8 trip and the turbine bypass systems as we requested so that these TSs would be available to us for inclusion in the reload amendment if we did not allow additional time for discussion of the matter and required implemention of these TSs for the start of Cycle 4 operations. We believe that we should allow the licensee more time to present their arguments and discuss this request, and based on our review of the significance of these trip systems with regard to limiting transients during the fuel cycle, we have decided to defer implementation of these TSs for 60 days following startup in order to allow time for further discussion of this subject with the licensee.

Since the approved method was used to determine the operating limit MCPRs to avoid violation of the safety limit MCPR in the event of any anticipated transients, we conclude that these limits are acceptable.

The licensee has proposed the use of the MCPR limits currently in the Hatch Unit 2 TSs (Figures 3.2.3-1 and 3.2.3-2) for Cycle 4 operation at increased core flow up to 105% of rated flow. We find that the operating MCPRs based on the reload analyses for Cycle 4 (Ref. 10) are lower than the calculated values for the Cycle 3 core, and on this basis condlude that the use of the currently approved operating limit MCPRs in

Figures 3.2.3-1 and 3.2.3-2 of the TSs are acceptable for Cycle 4 operation.

The results of thermal-hydraulic analysis (Ref. 10) show that maximum reactor core stability decay ratio is about 0.83, which is same as the calculated value for the Cycle 3 core which has been previously approved. We therefore conclude that the thermal-hydraulic stability results are acceptable for Cycle 4 operation.

We find that approved thermal-hydraulic methods have been used and that results of analyses support the proposed limit MCPRs, which avoid violation of the safety limit MCPR for design transients.

Nuclear Design

The Hatch Unit 2 Cycle 4 reload will consist of 560 fuel bundles as shown in Table 1. The initial core loading had a maximum average enrichment of 1.87 w/o in U-235. The reload fuel is similar in physical design to the initial core load fuel, but it has a maximum average enrichment of 2.84 w/o in U-235. All fuel bundles consist of 62 fuel rods and 2 water rods. The active fuel length is 150 inches.

The shutdown margin of the new core meets the TS requirement that the core be at least .25%ΔK subcritical in the worst reactive condition when the highest worth control rod is fully withdrawn and all other rods are fully inserted. For Hatch Unit 2 Cycle 4, GE calculated that the k_{eff} under cold conditions and the strongest rod out is equal to .985 resulting in a shutdown margin of 3.3%ΔK.

The standby liquid control system is capable of bringing the reactor from full power to a cold shutdown condition assuming none of the withdrawn control rods is inserted. The 600 ppm boron concentration will bring the reactor subcritical to k_{eff} = .950 at 20°C xenon free conditions (Ref. 10).

Based on our review of the licensee's submittal (Ref. 6) and the plant specific analysis (Ref. 10), we have determined that the nuclear characteristics and the expected performance of the reload core for Hatch Unit 2 Cycle 4 are acceptable.

Conclusions

On the basis of our review of the reload safety analysis for Cycle 4 operation of Hatch Unit 2 including the proposed related changes to the Hatch Unit 2 TSs, as discussed above, we conclude that this core reload will not adversely affect the capability to operate Hatch Unit 2 safely during Cycle 4 operation and that the proposed related TS changes are acceptable.

3.0 Environmental Considerations

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.

4.0 Conclusion

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

References

1. Letter, Georgia Power Company to Director of Nuclear Reactor Regulation, U.S. Nuclear Regulatory Commission, dated February 23, 1983.

2. Letter, Georgia Power Company to Director of Nuclear Reactor Regulation, U.S. Nuclear Regulatory Commission, dated April 19, 1983.

3. Document entitled "Edwin I. Hatch Plant Unit 2, Docket 50-366, Proposed Plant Modifications - Low-Low Set Logic and MSIV Water Level", provided as an Enclosure to Reference 1.

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

5. General Electric Company Report "Low-Low Set Relief Logic System and Lower MSIV Water Level Trip for Edwin I. Hatch Nuclear Plants Units 1 and 2," NEDE 22224, December 1982. Proprietary.

6. Letter, Georgia Power Company to Director of Nuclear Reactor Regulation, U.S. Nuclear Regulatory Commission, dated March 30, 1983.

7. Letter, Georgia Power Company to Director of Nuclear Reactor Regulation, U.S. Nuclear Regulatory Commission, dated May 10, 1983.

8. Letter, Georgia Power Company to Director of Nuclear Reactor Regulation, U.S. Nuclear Regulatory Commission, dated May 20, 1983.

9. Letter, Georgia Power Company to Director of Nuclear Reactor Regulation, U.S. Nuclear Regulatory Commission dated May 26, 1983.

10. General Electric Company Document "Supplemental Reload Licensing Submittal for Edwin I. Hatch Nuclear Plant Unit 2 Reload 3 (Cycle 4), Y1003J01A57, January 1983.

11. Letter, General Electric Company, L.K. Mathews, Southern Company Services, dated February 22, 1983. Letter No. LMQ-83-018.

12. Letter, General Electric Company to R. D. Baker, Georgia Power Company dated February 24, 1983, Letter No. LMQ:83-022.

13. "General Electric Standard Application for Reactor Fuel", GE Report NEDE-240011-P-A-4, January 1982.

14. D. G. Eisenhut (NRC) letter to E. D. Fuller (GE), June 30, 1977.

15. R. E. Engel (GE) letter to T. A. Ippolito (NRC), May 6, 1981.

16. R. E. Engel (GE) letter to T. A. Ippolito (NRC), May 28, 1981.

17. L. S. Rubenstein (NRC) memorandum for T. Novak, "Extension of General Electric Emergency Core Cooling Systems Performance Limits," June 25, 1981.

18. "Qualification of the One-Dimensional Core Transient Model for Boiling Water Reactors," GE Report NEDE-24154-P, October 1978.

19. Letter, John Stolz (NRC) to J. Beckham, Jr., Georgia Power Company, dated May 12, 1983.

Dated: June 29, 1983

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