LICENSE AUTHORITY FILE COPY

November 6, 1986

DO NOT REMOVE

Posted
Amdt. 67
to NPF-5

Docket No. 50-366

DISTRIBUTION
Docket File
NRC PDR
LPDR
DVassallo
RBernero
OGC-Beth
OPA
LHarmon
ACRS-10
TBarnhart-4
EJordan
BGrimes
WJones
Plant File
SECY
JPartlow
SNorris
GRivenbark
NThompson
EButcher

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. 67 to Facility Operating License No. NPF-5, for the Edwin I. Hatch Nuclear Plant, Unit No. 2. The amendment consists of changes to the Technical Specifications in response to your application dated July 18, 1986.

The amendment (1) revises allowable values to provide for the use of Rosemount, as well as Barton, transmitters for certain instrumentation channels associated with the Analog Transmitter Trip System (ATTS); (2) provides administrative clarifications; (3) revises allowable values for instruments which actuate on high drywell pressure; and (4) lowers the core spray and residual heat removal low pressure coolant injection low reactor pressure injection permissive setpoints to allow for increased flexibility in the use of Rosemount transmitters for this trip function.

A copy of the Safety Evaluation is also enclosed. Notice of Issuance will be included in the Commission's Bi-Weekly Federal Register Notice.

Sincerely,

Original signed by

George W. Rivenbark, Project Manager
BWR Project Directorate #2
Division of BWR Licensing

Enclosures:
1. Amendment No. 67 to NPF-5
2. Safety Evaluation

cc w/enclosures:
See next page

BWR:PD#2	BWR:PD#2	OGC	BWR:PD#2
SNorris	GRivenbark		DMuller
11/ /86	10/27/86	10/31/86	11/6/86

Mr. J. T. Beckham, Jr.
Georgia Power Company

Edwin I. Hatch Nuclear Plant,
Units Nos. 1 and 2

cc:

Bruce W. Chruchill, Esquire
Shaw, Pittman, Potts & Trowbridge
2300 N Street, N.W.
Washington, D.C. 20037

Mr. L. T. Gucwa
Engineering Department
Georgia Power Company
Post Office Box 4545
Atlanta, Georgia 30302

Mr. H. C. Nix, Jr., General Manager
Edwin I. Hatch Nuclear Plant
Georgia Power Company
Post Office Box 442
Baxley, Georgia 31513

Mr. Louis B. Long
Southern Company Services, Inc.
Post Office Box 2625
Birmingham, Alabama 35202

Resident Inspector
U.S. Nuclear Regulatory Commission
Route 1, Post Office Box 279
Baxley, Georgia 31513

Regional Administrator, Region II
U.S. Nuclear Regulatory Commission,
101 Marietta Street, Suite 2900
Atlanta, Georgia 30303

Mr. Charles H. Badger
Office of Planning and Budget
Room 610
270 Washington Street, S.W.
Atlanta, Georgia 30334

Mr. J. Leonard Ledbetter, Commissioner
Department of Natural Resources
270 Washington Street, N.W.
Atlanta, Georgia 30334

Chairman
Appling County Commissioners
County Courthouse
Baxley, Georgia 31513

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. 67
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 July 18, 1986 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. NPF-5 is hereby amended to read as follows:

ATTACHMENT TO LICENSE AMENDMENT NO.67

FACILITY OPERATING LICENSE NO. NPF-5

DOCKET NO. 50-366

17 Nov 86

Talked w/S Norris about this

Replace the following pages of the Appendix "A" Technical Specifications with the enclosed pages. The revised pages are identified by Amendment number and contain vertical lines indicating the area of change. The overleaf pages are provided for convenience.

Said She'd look into it

T

Remove	Insert
2-4	2-4
3/4 3-2	3/4 3-2
3/4 3-16	3/4 3-16
3/4 3-17	3/4-3-17
3/4 3-18	3/4 3-18
3/4 3-26	3/4 3-26
3/4 3-28	3/4 3-28
3/4 3-29	3/4 3-29
3/4 3-31	3/4 3-31
3/4 3-35	3/4 3-35
B 3/4 3-6	B 3/4 3-6

TABLE 2.2.1-1

REACTOR PROTECTION SYSTEM INSTRUMENTATION SETPOINTS

FUNCTIONAL UNIT	TRIP SETPOINT	ALLOWABLE VALUES
1. Intermediate Range Monitor, Neutron Flux-High (2C51-K601 A,B,C,D,E,F,G,H)	≤ 120/125 divisions of full scale	≤ 120/125 divisions of full scale
2. Average Power Range Monitor: (2C51-K605 A,B,C,D,E,F)		
a. Neutron Flux-Upscale, 15%	≤ 15/125 divisions of full scale	≤ 20/125 divisions of full scale
b. Flow Referenced Simulated Thermal Power-Upscale	≤ (0.58 W + 59%), with a maximum ≤ 113.5% of RATED THERMAL POWER	≤ (0.58 W + 62%), with a maximum ≤ 115.5% of RATED THERMAL POWER
c. Fixed Neutron Flux-Upscale, 118%	≤ 118% of RATED THERMAL POWER	≤ 120% of RATED THERMAL POWER
3. Reactor Vessel Steam Dome Pressure - High (2B21-N678 A,B,C,D)	≤ 1054 psig	≤ 1054 psig
4. Reactor Vessel Water Level - Low (Level 3) (2B21-N680 A,B,C,D)	≥ 10 inches above instrument zero*	≥ 10 inches above instrument zero*
5. Main Steam Line Isolation Valve - Closure (NA)	≤ 10% closed	≤ 10% closed
6. Main Steam Line Radiation - High (2D11-K603A,B,C,D)	≤ 3 x full power background	≤ 3 x full power background
7. Drywell Pressure - High (2C71-N650A,B,C,D)	≤ 1.92 psig	≤ 1.92 psig

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

(2) <u>Technical Specifications</u>

The Technical Specifications contained in Appendices A and B, as revised through Amenment No. 67, are hereby incorporated in the license. The licensee shall operate the facility in accordance with the Technical Specifications.

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

FOR THE NUCLEAR REGULATORY COMMISSION

George W. Rivenbark for

Daniel R. Muller, Director
BWR Project Directorate #2
Division of BWR Licensing

Attachment:
Changes to the Technical
Specifications

Date of Issuance: November 6, 1986

TABLE 3.3.1-1

REACTOR PROTECTION SYSTEM INSTRUMENTATION

FUNCTIONAL UNIT	APPLICABLE OPERATIONAL CONDITIONS	MINIMUM NUMBER OPERABLE CHANNELS PER TRIP SYSTEM(a)	ACTION
1. Intermediate Range Monitors: (2C51-K601, A, B, C, D, E, F, G, H)			
a. Neutron Flux - High	2[e], 5[b]	3	1
	3, 4	2	2
b. Inoperative	2,5[b]	3	1
	3, 4	2	2
2. Average Power Range Monitor: (2C51-K605 A, B, C, D, E, F)			
a. Neutron Flux - Upscale, 15%	2, 5	2	1
b. Flow Referenced Simulated Thermal Power - Upscale	1	2	3
c. Fixed Neutron Flux - Upscale, 118%	1	2	3
d. Inoperative	1, 2, 5	2	4
e. Downscale	1	2	3
f. LPRM	1, 2, 5	(d)	NA
3. Reactor Vessel Steam Dome Pressure - High (2B21-N678 A, B, C, D)	1, 2[e]	(J, 2B21-N045 A, B, C, D)	5
4. Reactor Vessel Water Level - Low (Level 3) (2B21-N680 A, B, C, D)	1, 2	(J, 2B21-N024A, B 2B21-N025A, B)	5
5. Main Steam Line Isolation Valve - Closure (NA)	1[f]	4	3
6. Main Steam Line Radiation - High (2D11-K603 A, B, C, D)	1, 2[e]	2	6
7. Drywell Pressure - High (2C71-N650 A, B, C, D)	1, 2[g]	2	5

2.2 LIMITING SAFETY SYSTEM SETTINGS

REACTOR PROTECTION SYSTEM INSTRUMENTATION SETPOINTS

2.2.1 The reactor protection system instrumentation setpoints shall be set consistent with the Trip Setpoint values shown in Table 2.2.1-1.

APPLICABILITY: As shown for each channel in Table 3.3.1-1.

ACTION:

With a reactor protection system instrumentation setpoint less conservative than the value shown in the Allowable Values column of Table 2.2.1-1, declare the channel inoperable and apply the applicable ACTION statement requirement of Specification 3.3.1 until the channel is restored to OPERABLE status with its trip setpoint adjusted consistent with the Trip Setpoint value.

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 (Level 3)	≥ 10 inches*	≥ 10 inches*
2. Low Low (Level 2)	≥ -47 inches*	≥ -47 inches*
3. Low Low Low (Level 1)	≥ -113 inches*	≥ -113 inches*
b. Drywell Pressure - High	≤ 1.92 psig	≤ 1.92 psig
c. Main Steam Line		
1. Radiation - High	≤ 3 x full power background	≤ 3 x full power background
2. Pressure - Low	≥ 825 psig	≥ 825 psig
3. Flow - High	≤ 138% rated flow	≤ 138% rated flow
d. Main Steam Line Tunnel Temperature - High	≤ 194°F	≤ 194°F
e. Condenser Vacuum - Low	≥ 7" Hg vacuum	≥ 7" Hg vacuum
f. Turbine Building Area Temp.-High	≤ 200°F	≤ 200°F
2. SECONDARY CONTAINMENT ISOLATION		
a. Reactor Building Exhaust Radiation - High	≤ 60 mr/hr	≤ 60 mr/hr
b. Drywell Pressure - High	≤ 1.92 psig	≤ 1.92 psig
c. Reactor Vessel Water Level - Low Low (Level 2)	≥ -47 inches*	≥ -47 inches*
d. Refueling Floor Exhaust Radiation - High	≤ 20 mr/hr	≤ 20 mr/hr

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

3/4.3 INSTRUMENTATION

3/4.3.1 REACTOR PROTECTION SYSTEM INSTRUMENTATION

LIMITING CONDITION FOR OPERATION

3.3.1 As a minimum, the reactor protection system instrumentation channels shown in Table 3.3.1-1 shall be OPERABLE with the REACTOR PROTECTION SYSTEM RESPONSE TIME as shown in Table 3.3.1-2. Set points and interlocks are given in Table 2.2.1-1.

APPLICABILITY: As shown in Table 3.3.1-1.

ACTION:

a. With the requirements for the minimum number of OPERABLE channels not satisfied for one trip system, place at least one inoperable channel in the tripped condition within one hour.

b. With the requirements for the minimum number of OPERABLE channels not satisfied for both trip systems, place at least one inoperable channel in at least one trip system* in the tripped condition within one hour and take the ACTION required by Table 3.3.1-1.

c. The provisions of Specification 3.0.3 are not applicable in OPERATIONAL CONDITION 5.

SURVEILLANCE REQUIREMENTS

4.3.1.1 Each reactor protection system instrumentation channel shall be demonstrated OPERABLE by the performance of the CHANNEL CHECK, CHANNEL FUNCTION TEST and CHANNEL CALIBRATION operations during the OPERATIONAL CONDITIONS and at the frequencies shown in Table 4.3.1-1.

4.3.1.2 LOGIC SYSTEM FUNCTIONAL TESTS and simulated automatic operation of all channels shall be performed at least once per 18 months and shall include calibration of time delay relays and timers necessary for proper functioning of the trip system.

4.3.1.3 The REACTOR PROTECTION SYSTEM RESPONSE TIME of each reactor trip function of Table 3.3.1-2 shall be demonstrated to be within its limit at least once per 18 months. Each test shall include at least one logic train such that both logic trains are tested at least once per 36 months and one channel per function such that all channels are tested at least once every N times 18 months where N is the total number of redundant channels in a specific reactor trip function.

*If both channels are inoperable in one trip system, select at least one inoperable channel in that trip system to place in the tripped conditions, except when this could cause the Trip Function to occur.

TABLE 3.3.2-2 (Continued)

ISOLATION ACTUATION INSTRUMENTATION SETPOINTS

TRIP FUNCTION	TRIP SETPOINT	ALLOWABLE VALUE
5. REACTOR CORE ISOLATION COOLING SYSTEM ISOLATION		
a. RCIC Steam Line Flow - High	≤ 307% of rated flow	≤ 307% of rated flow
b. RCIC Steam Supply Pressure - Low	≥ 60 psig	≥ 60 psig
c. RCIC Turbine Exhaust Diaphragm Pressure - High	≤ 20 psig	≤ 20 psig
d. Emergency Area Cooler Temperature-High	≤ 169°F	≤ 169°F
e. Suppression Pool Area Ambient Temperature High	≤ 169°F	≤ 169°F
f. Suppression Pool Area ΔT - High	≤ 42°F	≤ 42°F
g. Suppression Pool Area Temperature Timer Relays	NA	NA
h. Drywell Pressure - High	≤ 1.92 psig	≤ 1.92 psig
i. Logic Power Monitor	NA	NA
6. SHUTDOWN COOLING SYSTEM ISOLATION		
a. Reactor Vessel Water Level - Low (Level 3)	≥ 10 inches*	≥ 10 inches*
b. Reactor Steam Dome Pressure - High	≤ 145 psig	≤ 145 psig

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

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 $\leq$ 145 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, Table 3.6.3-1 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 all turbine stop valves closed.

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

h. Alarm only.

i. Adjustable up to 60 minutes.

TABLE 3.3.3-1

EMERGENCY CORE COOLING SYSTEM ACTUATION INSTRUMENTATION

TRIP FUNCTION	MINIMUM NUMBER OPERABLE CHANNELS PER TRIP SYSTEM	APPLICABLE OPERATIONAL CONDITIONS
1. CORE SPRAY SYSTEM		
a. Reactor Vessel Water Level - Low Low Low (Level 1) (2B21-N691A,B,C,D)	2	1,2,3,4,5
b. Drywell Pressure - High (2E11-N694 A,B,C,D)	2	1,2,3
c. Reactor Steam Dome Pressure - Low (Injection Permissive) (2B21-N690A,B,C,D)	2	1,2,3,4,5
d. Logic Power Monitor (2E21-K1A,B)	1/bus[(a)]	1,2,3,4,5
2. LOW PRESSURE COOLANT INJECTION MODE OF RHR SYSTEM		
a. Drywell Pressure - High (2E11-N694A,B,C,D)	2	1,2,3
b. Reactor Vessel Water Level - Low Low Low (Level 1) (2B21-N691A,B,C,D)	2	1,2,3,4*,5*
c. Reactor Vessel Shroud Level (Level 0) (Drywell Spray Permissive) (2B21-N685A, B)	1	1,2,3,4*,5*
d. Reactor Steam Dome Pressure - Low (Injection Permissive) (2B21-N690A,B,C,D)	2	1,2,3,4*,5*
e. Reactor Steam Dome Pressure - Low (Recirc. Discharge Valve Permissive) (2B21-N641B,C and 2B21-N690E,F)	2	1,2,3,4*,5*
f. RHR Pump Start - Time Delay Relay 1) Pump A (2E11-K70A, 2E11-K125B) 2) Pump B (2E11-K70B, 2E11-K125A) 3) Pump C (2E11-K75B) 4) Pump D (2E11-K75A, 2E11-K126)	1/pump	1,2,3,4*,5*
g. Logic Power Monitor (2E11-K1A,B)	1/bus[(a)]	1,2,3,4*,5*

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

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

TABLE 3.3.2-2 (Continued)

ISOLATION ACTUATION INSTRUMENTATION SETPOINTS

TRIP FUNCTION	TRIP SETPOINT	ALLOWABLE VALUE
3. REACTOR WATER CLEANUP SYSTEM ISOLATION		
a. Δ Flow - High	≤ 79 gpm	≤ 79 gpm
b. Area Temperature-High	≤ 124°F	≤ 124°F
c. Area Ventilation Δ Temperature - High	≤ 67°F	≤ 67°F
d. SLCS Initiation	NA	NA
e. Reactor Vessel Water Level-Low Low (Level 2)	≥ -47 inches*	≥ -47 inches*
4. HIGH PRESSURE COOLANT INJECTION SYSTEM ISOLATION		
a. HPCI Steam Line Flow-High	≤ 303% of rated flow	≤ 303% of rated flow
b. HPCI Steam Supply Pressure - Low	≥ 100 psig	≥ 100 psig
c. HPCI Turbine Exhaust Diaphragm Pressure-High	≤ 20 psig	≤ 20 psig
d. HPCI Pipe Penetration Room Temperature - High	≤ 169°F	≤ 169°F
e. Suppression Pool Area Ambient Temperature-High	≤ 169°F	≤ 169°F
f. Suppression Pool Area ΔT - High	< 42°F	< 42°F
g. Suppression Pool Area Temperature Timer Relays	NA	NA
h. Emergency Area Cooler Temperature - High	≤ 169°F	≤ 169°F
i. Drywell Pressure - High	≤ 1.92 psig	≤ 1.92 psig
j. Logic Power Bus Monitors	NA	NA

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

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 (Level 1)	≥ -113 inches*	≥ -113 inches*
b. Drywell Pressure - High	≤ 1.92 psig	≤ 1.92 psig
c. Reactor Steam Dome Pressure - Low	≥ 390 psig**	≥ 390 psig**
d. Logic Power Monitor	NA	NA
2. LOW PRESSURE COOLANT INJECTION MODE OF RHR SYSTEM		
a. Drywell Pressure - High	≤ 1.92 psig	≤ 1.92 psig
b. Reactor Vessel Water Level - Low Low Low (Level 1)	≥ -113 inches*	≥ -113 inches*
c. Reactor Vessel Shroud Level (Level 0) - High	≥ -202 inches*	≥ -202 inches*
d. Reactor Steam Dome Pressure-Low	≥ 390 psig**	≥ 390 psig**
e. Reactor Steam Dome Pressure-Low	≥ 335 psig	≥ 335 psig
f. RHR Pump Start - Time Delay Relay		
1) Pumps A, B and D	10 ± 1 seconds	10 ± 1 seconds
2) Pump C	0.5 ± 0.5 seconds	0.5 ± 0.5 seconds
g. Logic Power Monitor	NA	NA

*See Bases Figure B 3/4 3-1.
**This trip function shall be less than or equal to 500 psig.

SURVEILLANCE REQUIREMENTS (Continued)

4.3.3.3 The ECCS RESPONSE TIME of each ECCS function shown in Table 3.3.3-3 shall be demonstrated to be within the limit at least once per 18 months. Each test shall include at least one logic train such that both logic trains are tested at least once per 36 months and one channel per function such that all channels are tested at least once every N times 18 months where N is the total number of redundant channels in a specific ECCS function.

TABLE 3.3.3-3

EMERGENCY CORE COOLING SYSTEM RESPONSE TIMES

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

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 (Level 2) (2B21-N692 A,B,C,D)	2	1, 2, 3
b. Drywell Pressure - High (2E11-N694 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-N662B,D)	2(b) (c)	1, 2, 3
e. Logic Power Monitor (2E41-K1)	1(a)	1, 2, 3
f. Reactor Vessel Water Level-High (Level 8) (2B21-N693 B,D)	2	1, 2, 3
4. AUTOMATIC DEPRESSURIZATION SYSTEM		
a. Drywell Pressure - High (Permissive) (2E11-N694A,B,C,D)	2	1, 2, 3
b. Reactor Vessel Water Level - Low Low Low (Level 1) (2B21-N691 A,B,C,D)	2	1, 2, 3
c. ADS Timer (2B21-K752 A, B)	1	1, 2, 3
d. ADS Low Water Level Actuation Timer (2B21-K754A,B; 2B21-K756A,B)	2	1, 2, 3
e. Reactor Vessel Water Level-Low (Level 3) (Permissive) (2B21-N695A,B)	1	1, 2, 3
f. Core Spray Pump Discharge Pressure - High (Permissive) (2E21-N655A,B; 2E21-N652A,B)	2	1, 2, 3
g. RHR (LPCI MODE) Pump Discharge Pressure - High (Permissive) (2E11-N655A,B,C,D; 2E11-N656A,B,C,D)	2/loop	1, 2, 3
h. 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 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 (Level 1)	S	M	R	1, 2, 3, 4, 5
b. Drywell Pressure - High	S	M	R	1, 2, 3
c. Reactor Steam Dome Pressure - Low	S	M	R	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	S	M	R	1, 2, 3
b. Reactor Vessel Water Level - Low Low Low (Level 1)	S	M	R	1, 2, 3, 4*, 5*
c. Reactor Vessel Shroud Level (Level 0)	S	M	R	1, 2, 3, 4*, 5*
d. Reactor Steam Dome Pressure - Low	S	M	R	1, 2, 3, 4*, 5*
e. Reactor Steam Dome Pressure - Low	S	M	R	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 subsystems are OPERABLE per Specification 3.5.3.1.

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 (Level 2)	≥ -47 inches*	≥ -47 inches*
b. Drywell Pressure-High	≤ 1.92 psig	≤ 1.92 psig
c. Condensate Storage Tank Level - Low	≥ 0 inches**	≥ 0 inches**
d. Suppression Chamber Water Level - High	≤ 154.2 inches***	≤ 154.2 inches***
e. Logic Power Monitor	NA	NA
f. Reactor Vessel Water Level-High (Level 8)*	≤ 56.5 inches	≤ 56.5 inches
4. AUTOMATIC DEPRESSURIZATION SYSTEM		
a. Drywell Pressure-High	≤ 1.92 psig	≤ 1.92 psig
b. Reactor Vessel Water Level - Low Low Low (Level 1)	≥ -113 inches*	≥ -113 inches*
c. ADS Timer	≤ 120 seconds	≤ 120 seconds
d. ADS Low Water Level Actuation Timer	≤ 13 minutes	≤ 13 minutes
e. Reactor Vessel Water Level - Low (Level 3)	≥ 10 inches*	≥ 10 inches*
f. Core Spray Pump Discharge Pressure - High	≥ 137 psig	≥ 137 psig
g. RHR (LPCI MODE) Pump Discharge Pressure - High	≥ 112 psig	≥ 112 psig
h. 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.
*** Measured above torus invert.

TABLE 4.3.4-1

REACTOR CORE ISOLATION COOLING SYSTEM ACTUATION INSTRUMENTATION SURVEILLANCE REQUIREMENTS

FUNCTIONAL UNITS	CHANNEL CHECK	CHANNEL FUNCTIONAL TEST	CHANNEL CALIBRATION
a. Reactor Vessel Water Level-Low Low (Level 2)	S	M	R
b. Condensate Storage Tank Level- Low	NA	M	Q
c. Suppression Pool Water Level- High	NA	M	Q

TABLE 4.[illegible]-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 (Level 2)	S	M	R	1, 2, 3
b. Drywell Pressure-High	S	M	R	1, 2, 3
c. Condensate Storage Tank Level - Low	NA	M	Q	1, 2, 3
d. Suppression Chamber Water Level - High	S	M	R	1, 2, 3
e. Logic Power Monitor	NA	R	NA	1, 2, 3
f. Reactor Vessel Water Level - High (Level 8)	S	M	R	1, 2, 3
4. AUTOMATIC DEPRESSURIZATION SYSTEM				
a. Drywell Pressure - High	S	M	R	1, 2, 3
b. Reactor Vessel Water Level - Low Low Low (Level 1)	S	M	R	1, 2, 3
c. ADS Timer	NA	NA	R	1, 2, 3
d. ADS Low Water Level Actuation Timer	NA	NA	R	1, 2, 3
e. Reactor Vessel Water Level - Low (Level 3)	S	M	R	1, 2, 3
f. Core Spray Pump Discharge Pressure - High	S	M	R	1, 2, 3
g. RHR (LPCI MODE) Pump Discharge Pressure - High	S	M	R	1, 2, 3
h. 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.

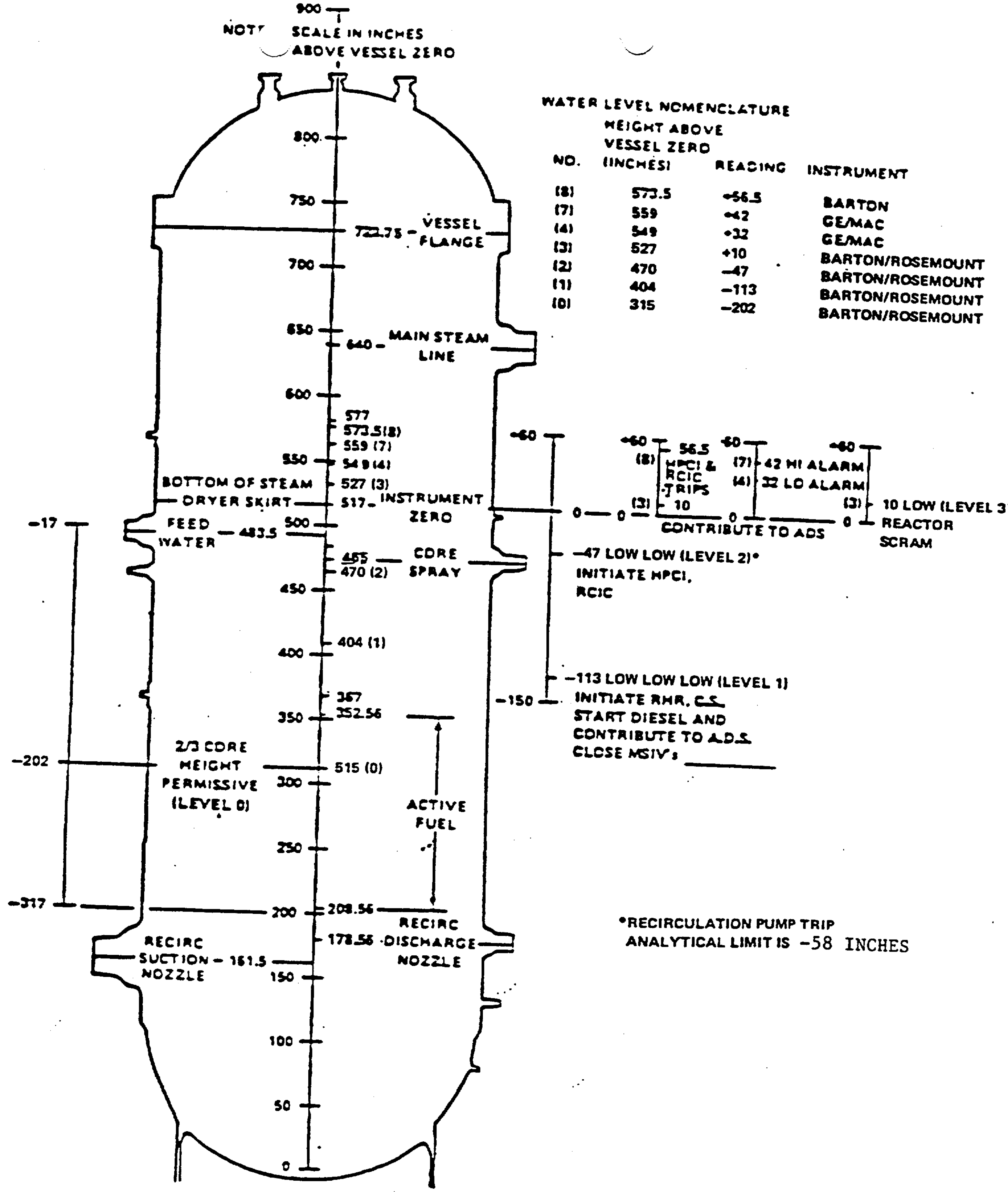
TABLE 3.3.4-2

REACTOR CORE ISOLATION COOLING SYSTEM ACTUATION INSTRUMENTATION

FUNCTIONAL UNITS	TRIP SETPOINT	ALLOWABLE VALUE
a. Reactor Vessel Water Level - Low Low (Level 2)	≥ -47 inches*	≥ -47 inches*
b. Condensate Storage Tank Level - Low	≥ 0 inches**	≥ 0 inches**
c. Suppression Pool Water Level - High	≤ 151 inches	≤ 151 inches

*See Bases Figure B 3/4 3-1

**Equivalent to 10,000 gallons of water in the CST.



WATER LEVEL NOMENCLATURE

NO.	HEIGHT ABOVE VESSEL ZERO (INCHES)	READING	INSTRUMENT
(8)	573.5	+56.5	BARTON
(7)	559	+42	GE/MAC
(4)	549	+32	GE/MAC
(3)	527	+10	BARTON/ROSEMOUNT
(2)	470	-47	BARTON/ROSEMOUNT
(1)	404	-113	BARTON/ROSEMOUNT
(0)	315	-202	BARTON/ROSEMOUNT

BASES FIGURE B 3/4 3-1
REACTOR VESSEL WATER LEVELS

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

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

SUPPORT AMENDMENT NO. 67 TO

FACILITY OPERATING LICENSE NO. NPF-5

GEORGIA POWER COMPANY
OGETHORPE 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 July 18, 1986 (Reference 1), the Georgia Power Company (GPC) proposed changes to the Hatch Plant Unit 2 Technical Specifications that would (1) revise allowable values to provide for the use of Rosemount, as well as Barton, transmitters for certain instrumentation channels associated with the Analog Transmitter Trip System (ATTS); (2) provide certain administrative clarifications; (3) revise allowable values for instruments which actuate on high drywell pressure; and (4) lower the core spray and residual heat removal low pressure coolant injection low reactor pressure injection permissive setpoints to allow for increased flexibility in the use of Rosemount transmitters for this trip function. Additional information was provided by letters dated September 26, 1986 (Reference 2) and October 15, 1986 (Reference 3) in response to staff requests.

2.0 EVALUATION

The original design of the ATTS used Barton Models 763 and 764 transmitters. GPC has decided to replace the Barton transmitters with Rosemount Models 1153 and 1154 transmitters, because they offer increased operational and maintenance flexibility. Because the accuracy characteristics of those transmitters differ, the Technical Specifications allowable values need to be revised in order to accommodate the new transmitter types. In particular, the following changes were proposed in change 1:

Trip Function	Present Allowable Value	Proposed Allowable Value
Reactor Vessel Water Level 1	-121.5 inches	-113 inches
Reactor Vessel Water Level 2	-55 inches	-47 inches
Reactor Vessel Water Level 3	8.5 inches	10 inches
Reactor Shroud Water Level 0	-207 inches	-202 inches
Reactor Vessel Steam Dome Pressure Low	325 psig	335 psig
HPCI Steam Line High Flow	307%	303%
RCIC Steam Line High Flow	307%	312%

Since there is no change of analytical limits, there will be no effect on the staff approved safety analyses as a result of the proposed change 1.

The proposed change 2 provides administrative clarifications or corrections to the Technical Specifications as follows:

A. Correct the parts number for the ATTS recirculation pump trip instruments appearing in Table 3.3.1-1. Correct parts numbers 2B21-N024 A, B and 2B21-N025A, B would replace the presently listed 2B21-N681 A, B, C, D.

B. The reactor shroud water level trip (2B21-N685 A, B), as listed in Table 3.3.3-1, is not a high-level trip. Thus, the word "high" is deleted from the table.

C. The suppression chamber water level high trip function (2E41-N662A, B) used an arbitrary zero point to derive the allowable value. The new allowable value (154.2 inches versus 33.2 inches) is derived from the torus invert. The actual level is unchanged.

D. The suppression pool area differential temperature high system allowable value is proposed to be changed from 42.5°F to 42°F for simplicity.

Since this change does not impact any plant operation, there will be no effect on the staff approved safety analyses as a result of the proposed change 2.

During the forthcoming refueling outage, the presently installed Barton Model 746 transmitters which provide the high drywell pressure trip function, would be replaced with Rosemount Model 1154 transmitters. This modification necessitates a revision of the Technical Specifications allowable value for the high drywell pressure trip function. It was proposed to revise the allowable values to bound the use of either Barton or Rosemount transmitters. The calculated allowable value of 1.70 psig, however, was unacceptably low from an operation's standpoint such that spurious trips during drywell inerting would likely occur. A new allowable value was calculated using a smaller range Rosemount 1154 transmitter. The proposed allowable value for this transmitter type (1.92 psig) was developed using the 2.0 psig analytical limit and the criteria of Regulatory Guide 1.105. Since there is no change of analytical limits, there will be no effect on the staff approved safety analyses as a result of the proposed change 3.

Change 4 proposes to lower the allowable value for the RHR-LPCI and core spray low reactor pressure permissives, which open the injection valves, from 422 psig to 390 psig. This change is required to allow for increased flexibility in the use of Rosemount transmitters for this function. In order to justify this change, an equivalent relaxation to the corresponding analytical limit needs to be provided. General Electric analyzed the proposed analytical limit for its effects on fuel thermal limits and LOCA limits in MDE-263-1185 (Appendix in Reference 1).

As to the fuel thermal limits, i.e. minimum critical power ratio (MCPR), transients governing these limits occur at relatively high pressure, far above the operating ranges of the RHR-LPCI and core spray low reactor pressure permissive. Therefore, the fuel thermal limits are not impacted by this proposed change 4.

As to the LOCA limits, the core maximum average planar linear heat generation rate (MAPLHGR) limits were determined by analyzing the postulated LOCAs in accordance with 10 CFR 50. General Electric's evaluation (Appendix in Reference 1) showed that the proposed change 4 to the low pressure injection permissive would not have a significant effect on peak clad temperature (PCT) for the limiting design basis accident. Because PCTs for certain fuel types and exposures are presently at or near the limiting value 2200°F, further clarification was requested by the staff as to whether the proposed change would result in the calculated limiting PCT remaining at or below 2200°F when calculated according to approved Appendix K models. This verification was provided by General Electric to GPC that the calculated MAPLHGR limits for Hatch-2 would not change and the calculated limiting PCT would remain at or below 2200°F (Reference 2).

GPC referenced the utilization of a setpoint methodology for the subject modifications and stated that the methodology is consistent with that previously reviewed and accepted by the staff during its evaluation of setpoint modifications (Amendment 39 to Hatch, Unit 2 Operating License) related to the implementation of the original ATTS design for Hatch, Unit 2. The staff based its acceptance of the original ATTS setpoint modification on the review of information submitted in a June 7, 1984 letter from GPC. The subject letter contained (1) specific responses to NRC requests for information related to the setpoint methodology program and (2) information prepared by General Electric (GE) which addresses the specific setpoint calculation methodology utilized for Hatch, Unit 2. The staff's evaluation of this information was provided in the Safety Evaluation for Amendment 39 and was enclosed with the amendment in the staff's July 13, 1986 letter to Mr. J. T. Beckham, Jr., Georgia Power Company.

GPC has verified by letter dated October 15, 1986 that the information provided in the June 7, 1984 letter (with additional clarification information) remains valid in support of the latest request for ATTS setpoint changes. The staff has reviewed the clarification information and finds it to be acceptable.

Based on the above discussion, we conclude that the proposed Technical Specification changes are acceptable.

3.0 ENVIRONMENTAL CONSIDERATIONS

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

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 the amendment will not be inimical to the common defense and security or to the health and safety of the public.

Principal Contributor: D. Yue and R. Stevens

Date: November 6, 1986

REFERENCES

1. Letter from J. T. Beckham, Jr., Georgia Power Company, to D. Muller, NRC, July 18, 1986.

2. Letter from L. T. Gucwa, Georgia Power Company, to D. Muller, NRC, September 26, 1986.

3. Letter from L. T. Gucwa, Georgia Power Company to D. Muller, NRC, October 15, 1986.