DMB 016

January 4, 1984

Dockets Nos. 50-321
and 50-366

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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 Amendments Nos. 97 and 34 to Facility Operating Licenses Nos. DPR-57 and NPF-5, respectively, for the Edwin I. Hatch Nuclear Plant, Units Nos. 1 and 2. The amendments consist of changes to the Technical Specifications (TSs) in response to your application dated February 26, 1981, as supplemented and modified by your submittals dated October 1, 1981, September 19, 1983, October 3, 1983, December 14, 1983, and December 20, 1983.

The amendments revise the TSs for both Hatch Units 1 and 2 to: 1) add a Limiting Condition for Operation (LCO) and surveillance requirements for the Scram Discharge Volume (SDV) vent and drain valves, and 2) add the new diverse SDV high water level scram instrumentation (thermal level sensors), including trip setpoint, LCO, Action Statement and surveillance requirements, to the Reactor Protection System Instrumentation Tables.

The amendments also revise the TSs for Hatch Unit 1 to: 1) add the SDV high water level trip instrumentation, including trip setpoint, LCO, Action Statement and surveillance requirements, to the Control Rod Block Instrumentation Tables, and 2) change the required frequency for functional testing of the SDV high water level reactor scram instrumentation from once per three months to once per month.

The amendments do not include the SDV vent and drain valve surveillance closure time requirement. Your late revision, dated December 14, 1983, changed your September 19, 1983, proposed closure time of 30 seconds for each SDV vent and drain valve to 60 seconds for the inboard vent and drain valves and 120 seconds for the outboard vent and drain valves. This proposal deviates from the closure time acceptability guidelines of 30 seconds previously provided by the staff. Accordingly, the proposed closure times will require further evaluation by the NRC staff before they can be approved as Technical Specification requirements in a subsequent amendment. In the interim, the staff has concluded that there is reasonable assurance of safe operation of the plants based on the implementation of the short-term corrective measures noted in the June 24, 1983 Order; and the long-term corrective measures noted in the enclosed Safety Evaluations.

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PDR ADOCK 05000321
P PDR

Copies of the Safety Evaluations are also enclosed enclosed. Notice of Issuance will be included in the Commission's Monthly Notice.

Sincerely,
ORIGINAL SIGNED BY
JOHN F. STOLZ
John F. Stolz, Chief
Operating Reactors Branch #4
Division of Licensing

Enclosures:
1. Amendments Nos. 97 and 34
2. Safety Evaluations w/enclosed Technical Evaluation Reports

cc w/enclosures: See next page

1/4/84					
ORB#4:DL	ORB#4:DL*	C-ORB#4:DL	AD:OR:DL	OELD	
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12/30/83	12/ /83	12/30/83	1/4/84	12/ /83	Sinte concurred as changed by telecon with Hunger R 12/3/83

* See previous concurrence attached.

have not included a closure time requirement in these amendments. We will provide TS requirements for closure time of these SDV vent and drain valves in a subsequent amendment when we have completed our review of your December 14, 1983 proposal.

Copies of the Safety Evaluations are also enclosed enclosed. Notice of Issuance will be included in the Commission's Monthly Notice.

Sincerely,

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

Enclosures:
1. Amendments Nos. and
2. Safety Evaluations w/enclosed Technical Evaluation Reports

cc w/enclosures: See next page

ORB#4:DL	ORB#4:DL	C-ORB#4:DL	AD:OR:DL	OELD
RIngram	GRivenbark;cf	JStolz	GLainas	
12/21/83	12/21/83	12/ /83	12/ /83	12/29/83

Hatch 1/2
Georgia Power Company

50-321/366

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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- 321

EDWIN I. HATCH NUCLEAR PLANT, UNIT NO. 1

AMENDMENT TO FACILITY OPERATING LICENSE

Amendment No. 97
License No. DPR-57

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

   A. The application for amendment by Georgia Power Company, et al., (the licensee) dated February 26, 1981, as supplemented October 1, 1981, September 19, 1983, October 3, 1983, December 14, 1983, and December 20, 1983, complies with the standards and requirements of the Atomic Energy Act of 1954, as amended (the Act), and the Commission's rules and regulations set forth in 10 CFR Chapter I;

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

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

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

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

2. Accordingly, the license is amended by changes to the Technical Specifications as indicated in the attachment to this license amendment and paragraph 2.C.(2) of Facility Operating License No. DPR-57is hereby amended to read as follows:

   Technical Specifications

   The Technical Specifications contained in Appendices A and B, as revised through Amendment No.97 , 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: January 4, 1984

ATTACHMENT TO LICENSE AMENDMENT NO. 97

FACILITY OPERATING LICENSE NO. DPR-57

DOCKET NO. 50-321

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

Remove Pages	Insert Pages
3.1-4	3.1-4
3.1-7	3.1-7
3.2-16	3.2-16
3.2-17	3.2-17
3.2-40	3.2-40
----	3.3-7a
3.3-18	3.3-18
----	3.3-18a

Table 3.1-1 (Continued)

Scram Number (a)	Source of Scram Trip Signal	Operable Channels Required Per Trip System (b)	Scram Trip Setting	Source of Scram Signal is Required To Be Operable Except as Indicated Below
5	High Drywell Pressure	2	≤2 psig	Not required to be operable when primary containment integrity is not required. May be bypassed when necessary during purging for containment inerting or deinerting.
6	Reactor Water Low Level (LL1) (Narrow Range)	2	≥12.5 inches	
7	Scram Discharge Volume High High Level			Permissible to bypass (initiates control rod block) in order to reset RPS when the Mode Switch is in the REFUEL or SHUTDOWN position.
	a. Float Switches	2	≤71 gallons	
	b. Thermal Level Sensors	2	≤71 gallons	
8	APRM Flow Referenced Neutron Flux	2	S ≤ 0.66W+54% (Not to exceed 117%) Tech Spec 2.1.A.1.c	
	Fixed High Neutron Flux	2	S ≤120% Power Tech Spec 2.1.A.1.c	
	Inoperative	2	Not Applicable	An APRM is inoperable if there are less than two LPRM inputs per level or there are less than 11 LPRM inputs to the APRM channel

Table 4.1-1

Reactor Protection System (RPS) Instrumentation Functional Test, Functional Test Minimum Frequency, and Calibraton Minimum Frequency

Scram Number (a)	Source of Scram Trip Signal	Group (b)	Instrument Functional Test Minimum Frequency (c)	Instrument Calibration Minimum Frequency
1	Mode Switch in SHUTDOWN	A	Once/Operating Cycle	Not Applicable
2	Manual Scram	A	Every 3 months	Not Applicable
3	IRM High High Flux	C	Once/Week during refueling and Within 24 hours of Startup (e)	Once/Week
	Inoperative	C	Once/Week during refueling and within 24 hours of Startup (e)	Once/Week
4	High Reactor Pressure	A	Once/Month (f)	Every 3 months
5	High Drywell Pressure	A	Once/Month (f)	Every 3 months
6	Reactor Water Low Level (LL1)	A	Once/Month (f) (g)	Every 3 months
7	Scram Discharge Volume High High Level			
	a. Float Switches	A	Once/Month (f)	(h)
	b. Thermal Level Sensors	B	Once/Month (f)	Once/Operating Cycle
8	APRM Fixed High Flux	B	Once/Week (e)	Twice/Week
	Inoperable	B	Once/Week (e)	Twice/Week
	Downscale	B	Once/Week (e)	Twice/Week
	Flow Reference	B	Once/Week (f)	Once/Operating Cycle
	15% Flux	C	Within 24 Hours of Startup (e)	Once/Week

Amendment No. 32, 42, 87, 92, 97

Table 3.2-7 (Continued)

Ref. No. (a)	Instrument	Trip Condition Nomenclature	Required Operable Channels per Trip System (b)	Trip Setting	Remarks
3	APRM	Downscale	2 (e)	$\geq$ 3/125 of full scale	Not required while performing low power physics test at atmospheric pressure during or after refueling at power levels not to exceed 5 MWt.
		12% Flux	2 (e)	$\leq$ 12/125 of full scale	This function is bypassed when the Mode Switch is placed in the RUN position.
		High Flux	2 (e)	$\leq$ 0.66W + 42%	W is the loop recirculation flow rate in percent of rated. Trip level setting is in percent of rated power. Not required while performing low power physics tests at atmospheric pressure during or after refueling at power levels not to exceed 5 MWt.
4	RBM	Inoperative	1 (e)(f) (g)(h)	Not Applicable	Inoperative trip produced by switch not in operate, circuit boards not in curcuit, fails to null, less than required number of LPRM inputs for rod selected.
		Downscale	1 (e)(f) (g)(h)	$\geq$ 3/125 of full scale	
		High Flux	1 (e)(f)	$\leq$ 0.66W + 41% Not to exceed 107%	W is the loop recirculation flow rate in percent of rated. Trip level setting is in percent of rated thermal power.
5	Scram Discharge Volume	High Water Level	1 (i)	$\leq$ 18 gallons	

Notes for Table 3.2-7

a. The column entitled "Ref. No." is only for convenience so that a one-to-one relationship can be established between items in Table 3.2-7 and items in Table 4.2-7.

b. For the START & HOT STANDBY position of the Mode Switch, there shall be two operable or tripped systems for each potential trip condition. If the requirements established by the column cannot be met for one of the two trip systems, the condition may exist for up to seven days provided that during that time the operable system is functionally tested immediately and daily thereafter; if this condition lasts longer than seven days, the system shall be tripped. If the requirements established by this column cannot be met for both trip systems, the systems shall be tripped.

c. One of the four SRM inputs may be bypassed.

d. The SRM and IRM blocks need not be operable in the Run Mode. This function is bypassed when the Mode Switch is placed in the RUN position.

e. The APRM and RBM rod blocks need not be Operable in the Start and Hot Standby Mode (Except 12% APRM Rod Block)

f. Not required while performing low power physics tests at atmospheric pressure during or after refueling at power levels not to exceed 5 MWt.

g. This trip bypassed when reactor power is $\leq$30%.

h. One channel of the RBM may be inoperative or bypassed if this condition does not persist longer than 24 hours in a 30 day period.

i. This trip is Operable in Power Operation and Hot Standby Mode, and Refuel Mode when any control rod is withdrawn. Not applicable to control rods removed per specification 3.10.E.

Notes for Table 4.2-6 (Cont'd)

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

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

d. Initially once per month or according to Figure 4.1-1 with an interval of not less than one month nor more than three months. The compilation of instrument failure rate data may include data obtained from other BWR's for which the same design instrument operates in an environment similar to that of HNP-1. The failure rate must be reviewed and approved by the AEC prior to any change in the once-a-month frequency.

Logic system functional tests and simulated automatic actuation shall be performed once each operating cycle for the following:

1. Core Spray Subsystem

The logic system functional test shall include a calibration of time delay relays and timers necessary for proper functioning of the trip systems.

Table 4.2-7

Check, Functional Test, and Calibration Minimum Frequency for Neutron Monitoring Instrumentation Which Initiates Control Rod Blocks

Ref. No. (a)	Instrument	Instrument Check Minimum Frequency (b)	Instrument Functional Test Minimum Frequency (c)	Instrument Calibration Minimum Frequency (d)
1	Source Range Monitors (SRM)	Once/day	(f)	(g)
2	Intermediate Range Monitors (IRM)	Once/day	(f)	(g)
3	Average Power Range Monitors (APRM)	Once/Day	(e)	Every 3 months
4	Rod Block Monitor (RBM)	Once/day	(e)	Every 3 months
5	Scram Discharge Volume	N/A	Every 3 months	Once/Operating Cycle

Notes for Table 4.2-7

a. The column titled "Ref. No." is only for convenience so that a one-to-one relationship can be established between items in Table 4.2-7 and items in Table 3.2-7.

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

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

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

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

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

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

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

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

H. Shutdown Requirements

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

3.3.I Scram Discharge Volume Vent and Drain Valves

During reactor power operation, all scram discharge volume vent and drain valves shall be operable. If this specification cannot be met, an orderly shutdown shall be initiated and the reactor shall be placed in the Hot Shutdown Condition within 12 hours.

4.3.I Scram Discharge Volume Vent and Drain Valves

1. Valve Position Verification

Each scram discharge volume vent and drain valve shall be verified to be in the open position* at least once per 31 days.

2. Valve Testing

a. Each scram discharge volume vent and drain valve shall be operated through at least one complete cycle of full travel at least once per 92 days.

b. At least once per 18 months, it shall be verified that the scram discharge volume vent and drain valves:

1. Close within (time limit to be provided in subsequent amendment) seconds after receipt of a signal for control rods to scram, and

2. Open when the scram scram signal is reset.

*These valves may be closed intermittently for testing under administrative controls.

3.3 REACTIVITY CONTROL

A. Core Reactivity Margin

Limiting Conditions for Operation:

The requirements for the Control Rod Drive System have been identified by evaluating the need for reactivity control via control rod movement over the full spectrum of plant conditions and events. As discussed in Section 3.6.4.1 of the FSAR, the control rod system design is intended to provide sufficient control of core reactivity so that the core could be made subcritical with the highest worth rod fully withdrawn. This reactivity characteristic has been a basic assumption in the analysis of plant performance.

Surveillance Requirements:

The core reactivity limitation is a restriction to be applied principally to the design of new fuel which may be loaded in the core or into a particular refueling pattern. Satisfaction of this limitation can only be demonstrated at the time of loading and should be such that it will apply to the entire subsequent operating cycle. The generalized form is that the reactivity of the core loading will be limited so the core can be made subcritical by at least R + 0.38% Δk at the time of the test, with the analytically determined highest worth control rod fully withdrawn and all other rods capable of insertion fully inserted. The value of R in % Δk is the amount by which the core reactivity, at any time in the operating cycle, is calculated to be greater than at the time of the check; i.e., the initial loading. R must be a positive quantity or zero. A core containing burnable neutron absorbers can have a reactivity characteristic which, related to the most reactive (cold shutdown) condition, increases with core lifetime, goes through a maximum and then decreases thereafter. A new value of R must be determined for each fuel cycle.

The 0.38% Δk in the expression R + 0.38% Δk is provided as a finite, demonstrable, subcriticality margin. This margin is demonstrated by full withdrawal of the analytically determined, highest worth rod and the additional withdrawal of an analytically selected rod or rods to positions calculated to insert at least R + 0.38% Δk in reactivity.

In determining the "analytically highest worth" rod, it is assumed that every fuel assembly of the same type has identical material properties. In the actual core, however, the control cell material properties vary within allowed manufacturing tolerances, and the highest worth rod is determined by a combination of the control cell geometry and local k_∞. Therefore, an additional margin is included in the shutdown margin test to account for the fact that the rod used for the demonstration (the "analytically highest worth") is not necessarily the highest worth rod in the core. Studies have been made which compare experimental criticals with calculated criticals. These studies have shown that actual criticals can be predicted within a given tolerance band. For gadolinia cores the additional margin required due to control cell material manufacturing tolerances and calculational uncertainties has experimentally been determined to be 0.38% Δk. When this additional margin is demonstrated, it assures that the reactivity control requirement

3.3.G.2.a. Operability

Surveillance Requirements:

The RSCS can be functionally tested prior to control rod withdrawal for reactor startup. By selecting, for example, A_{12} and attempting to withdraw, by one notch, a rod or all rods in each other group, it can be determined that the A_{12} group is exclusive. By bypassing to full out all A_{12} rods, selecting A_{34} and attempting to withdraw, by one notch, a rod or all rods in group B, the A_{34} group is determined exclusive. The same procedure can be repeated for the B groups. After 50% of the control rods have been withdrawn (e.g., groups A_{12} and A_{34}), it is demonstrated that the continuous withdrawal mode for the control drives is inhibited.

This demonstration is made by attempting to withdraw a control rod more than one notch in the first programmed rod group subsequent to reaching the 50% rod density point. This restriction to the notching mode of operation for control rod withdrawal is automatically removed when the reactor reaches the automatic initiation setpoint.

During reactor shutdown, similar surveillance checks shall be made with regard to rod group availability as soon as automatic initiation of the RSCS occurs and subsequently at appropriate stages of the control rod insertion.

b. Failed Position Switch

Limiting Conditions for Operation:

In the event that a control rod has a failed "Full-in" or "Full-out" position switch while that rod is in one of the initial groups comprising the first 50% of control rods withdrawn, it may be bypassed in the Rod Sequence Control System if its position is otherwise known. It is a safer and more desirable condition for such rods to occupy their proper positions in the control rod patterns during reactor startup or shutdown.

Surveillance Requirements:

Having a second licensed operator verify the actual rod position prior to bypassing a rod in the Rod Sequence Control System provides assurance that Specification 3.3.G.2.b is met.

c. Shutdown Margin/Scram Time Testing

After initial fuel loading and subsequent refuelings when operating above 950 psig all control rods shall be scram tested within the constraints imposed by the RSCS and before the 40% power level is reached. To maintain the required reactor pressure conditions the individually scrammed or inserted rod should be withdrawn to its original position immediately following testing of each rod. In order to select and withdraw the scrammed or inserted insequence control rod (also to select and insert a fully withdrawn insequence rod in case of friction testing) it will be necessary to simulate all the insequence withdrawn rods of the succeeding RSCS groups as being at full in position by utilizing the individual rod posi-

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

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

H. Shutdown Requirements

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

I. Scram Discharge Volume Vent and Drain Valves

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

J. References

1. FSAR Section 3.4, Reactivity Control Mechanical Design

2. FSAR Section 3.5.2, Safety Design Bases

3. FSAR Section 3.5.4, Safety Evaluation

4. FSAR Section 3.5, Control Rod Drive Housing Supports

J. References (Continued)

5. FSAR Section 14.4.3, Loss-of-Coolant Accident

6. FSAR Section 14.4.2, Control Rod Drop Accident

7. FSAR Appendix G, Plant Nuclear Safety Operational Analysis

8. C. J. Paone, R. C. Stirn, J. A. Woolley, "Rod Drop Accident Analysis for Large Boiling Water Reactors", NEDO-10527 Class I, March, 1972.

9. R. C. Strin, C. J. Paone, R. M. Young, "Rod Drop Accident Analysis For Large Boiling Water Reactors Addendum No. 1 Multiple Enrichment Cores With Axial Gadolinium:, Supplement 1 NEDO-10527 Class I, July 1972.

10. FSAR Section 3.6.5.4, Control Rod Worth

11. FSAR Section 3.6.6, Nuclear Evaluations

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. 34
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 26, 1981, as supplemented October 1, 1981, September 19, 1983, October 3, 1983, December 14, 1983, and December 20, 1983, 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:

   Technical Specifications

   The Technical Specifications contained in Appendices A and B, as revised through Amendment No. 34, are hereby incorporated in the license. The licensee shall operate the facility in accordance with the Technical Specifications.

3. This license amendment becomes effective within 30 days after 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: January 4, 1984

ATTACHMENT TO LICENSE AMENDMENT NO. 34

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 Pages	Insert Pages
2-5	2-5
3/4 3-3	3/4 3-3
3/4 3-8	3/4 3-8
-----	3/4 1-21
-----	B3/4 1-5

TABLE 2.2.1-1 (Continued)

REACTOR PROTECTION SYSTEM INSTRUMENTATION SETPOINTS

FUNCTIONAL UNIT	TRIP SETPOINT	ALLOWABLE VALUES
8. Scram Discharge Volume Water Level - High (2C11-N013A,B,C,D and 2C11-N060A,B,C,D)	$\leq$ 57.15 gallons	$\leq$ 57.15 gallons
9. Turbine Stop Valve - Closure (NA)	$\leq$ 10% closed	$\leq$ 10% closed
10. Turbine Control Valve Fast Closure, Trip Oil Pressure-Low (2C71-N005A,B,C,D)	$\geq$ 600 psig	$\geq$ 600 psig
11. Reactor Mode Switch in Shutdown Position (NA)	NA	NA
12. Manual Scram (NA)	NA	NA

TABLE 3.3.1-1 (Continued)

REACTOR PROTECTION SYSTEM INSTRUMENTATION

FUNCTIONAL UNIT	APPLICABLE OPERATIONAL CONDITIONS	MINIMUM NUMBER OPERABLE CHANNELS PER TRIP SYSTEM(a)	ACTION
8. Scram Discharge Volume Water			
Level - High (2C11-N013A,B,C,D)	1, 2, 5(h)	2	4
Level - High (2C11-N060A,B,C,D)	1, 2, 5	2	4
9. Turbine Stop Valve - Closure (NA)	1(i)	4(k)	7
10. Turbine Control Valve Fast Closure, Trip Oil Pressure - Low (2C71-N005A,B,C,D)	1(i)	2(k)	7
11. Reactor Mode Switch in Shutdown Position (NA)	1, 2, 3, 4, 5	1	8
12. Manual Scram (NA)	1, 2, 3, 4, 5	1	9

TABLE 3.3.1-1 (Continued)

REACTOR PROTECTION SYSTEM INSTRUMENTATION

ACTION

ACTION 1 - In OPERATIONAL CONDITION 2, be in at least HOT SHUTDOWN within 6 hours.

In OPERATIONAL CONDITION 5, suspend all operations involving CORE ALTERATIONS or postive reactivity changes and fully insert all insertable control rods within one hour.

ACTION 2 - Lock the reactor mode switch in the Shutdown position within one hour.

ACTION 3 - Be in at least STARTUP within 2 hours.

ACTION 4 - In OPERATIONAL CONDITION 1 or 2, be in at least HOT SHUTDOWN within 6 hours.

In OPERATIONAL CONDITION 5, suspend all operations involving CORE ALTERATIONS or positive reactivity changes and fully insert all insertable control rods within one hour.

ACTION 5 - Be in at least HOT SHUTDOWN within 6 hours.

ACTION 6 - Be in STARTUP with the main steam line isolation valves closed within 2 hours or in at least HOT SHUTDOWN within 6 hours.

ACTION 7 - Initiate a reduction in THERMAL POWER within 15 minutes and be at less than 30% of RATED THERMAL POWER within 2 hours.

ACTION 8 - In OPERATIONAL CONDITION 1 or 2, be in at least HOT SHUTDOWN within 6 hours.

In OPERATIONAL CONDITION 3 or 4, immediately and at least once per 12 hours verify that all control rods are fully inserted.

In OPERATIONAL CONDITION 5, suspend all operations involving CORE ALTERATIONS or positive reactivity changes and fully insert all insertable control rods within one hour.

TABLE 4.3.1-1

REACTOR PROTECTION SYSTEM INSTRUMENTATION SURVEILLANCE REQUIREMENTS

FUNCTIONAL UNIT	CHANNEL CHECK	CHANNEL FUNCTIONAL TEST	CHANNEL CALIBRATION(a)	OPERATIONAL CONDITIONS IN WHICH SURVEILLANCE REQUIRED
1. Intermediate Range Monitors:				
a. Neutron Flux - High	D	S/U(b)(c)	R	2
	D	W	R	3, 4, 5
b. Inoperative	NA	W	NA	2, 3, 4, 5
2. Average Power Range Monitor:				
a. Neutron Flux - Upscale, 15%	S	S/U(b)(c), W(d)	S/U(b), W(d)	2
	S	W	W	5
b. Flow Referenced Simulated Thermal Power - Upscale	S	S/U(b), W	W(e)(f), SA	1
c. Fixed Neutron Flux - Upscale, 118%	S	S/U(b), W	W(e), SA	1
d. Inoperative	NA	W	NA	1, 2, 5
e. Downscale	NA	W	NA	1
f. LPRM	D	NA	(g)	1, 2, 5
3. Reactor Vessel Steam Dome Pressure - High	NA	M	Q	1, 2
4. Reactor Vessel Water Level - Low	D	M	Q	1, 2
5. Main Steam Line Isolation Valve - Closure	NA	M	R(h)	1
6. Main Steam Line Radiation - High	D	W(i)	R(j)	1, 2
7. Drywell Pressure - High	NA	M	Q	1, 2
8. Scram Discharge Volume Water Level - High	NA	M	R(h)	1, 2, 5

TABLE 4.3.1-1 (Continued)

REACTOR PROTECTION SYSTEM INSTRUMENTATION SURVEILLANCE REQUIREMENTS

FUNCTIONAL UNIT	CHANNEL CHECK	CHANNEL FUNCTIONAL TEST	CHANNEL CALIBRATION	OPERATIONAL CONDITIONS IN WHICH SURVEILLANCE REQUIRED
9. Turbine Stop Valve - Closure	NA	M	R(h)	1
10. Turbine Control Valve Fast Closure, Trip Oil Pressure - Low	NA	M	R	1
11. Reactor Mode Switch in Shutdown Position	NA	R	NA	1, 2, 3, 4, 5
12. Manual Scram	NA	M	NA	1, 2, 3, 4, 5

a. Neutron detectors may be excluded from CHANNEL CALIBRATION.

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

c. The APRM, IRM and SRM channels shall be compared for overlap during each startup, if not performed within the previous 7 days.

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

e. This calibration shall consist of the adjustment of the APRM channel to conform to the power values calculated by a heat balance during CONDITION 1 when THERMAL POWER _ 25% of RATED THERMAL POWER. Adjust the APRM channel if the absolute difference _ 2%. Any APRM channel gain adjustment made in compliance with Specification 3.2.2 shall not be included in determining the absolute difference.

f. This calibration shall consist of the adjustment of the APRM flow referenced simulated thermal power channel to conform to a calibrated flow signal.

g. The LPRM's shall be calibrated at least once per 1000 effective full power hours (EFPH) using the TIP system.

h. Physical inspection and actuation of switches for instruments 2C11-N013A,B,C,D

i. Instrument alignment using a standard current source.

j. ration using a standard radiation source.

3/4.1.6 SCRAM DISCHARGE VOLUME VENT AND DRAIN VALVES

LIMITING CONDITION FOR OPERATION

3.1.6.1 All scram discharge volume vent and drain valves shall be OPERABLE.

APPLICABILITY: Conditions 1 and 2.

ACTION: With any scram discharge volume vent or drain valve inoperable, be in at least HOT SHUTDOWN within 12 hours.

SURVEILLANCE REQUIREMENTS

4.1.6.1 The scram discharge volume vent and drain valves shall be demonstrated OPERABLE by:

a. At least once per 31 days verifying each valve to be open*.

b. At least once per 92 days cycling each valve through at least one complete cycle of travel.

c. At least once per 18 months, by verifying that the drain and vent valves:

   1. Close within (closure time to be provided in a subsequent amendment) seconds after receipt of a signal for control rods to scram, and

   2. Open when the scram signal is reset.

*These valves may be closed intermittently for testing under administrative controls.

BASES

3/4.1.6 SCRAM DISCHARGE VOLUME VENT AND DRAIN VALVES

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

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

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

SUPPORTING AMENDMENT NO. 97 TO FACILITY OPERATING LICENSE NO. DPR-57

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

EDWIN I. HATCH NUCLEAR PLANT, UNIT NO. 1

DOCKET NO. 50-321

Introduction

As a result of events involving common cause failures of Scram Discharge Volume (SDV) limit switches and SDV drain valve operability, the NRC staff issued IE Bulletin 80-14 on June 12, 1980. In addition, the NRC staff sent a letter dated July 7, 1980, to all operating BWR licensees requesting that they propose Technical Specification (TS) changes to provide surveillance requirements for SDV vent and drain valves and Limiting Conditions for Operation (LCO)/surveillance requirements on SDV limit switches. Model TSs were enclosed with this letter to provide guidance to licensees for preparation of the requested submittals.

Evaluation

The enclosed report (TER-C5506-73) was prepared for us by Franklin Research Center (FRC) as part of a technical assistance contract program. Their report provides their Technical Evaluation of the compliance of Georgia Power Company's (the licensee) submittal with NRC provided criteria.

FRC has concluded that the licensee's response does not meet the explicit requirements of paragraph 3.3-6 and Table 3.3.6-1 of the NRC staff's Model TSs. However, the FRC report concludes that technical bases are defined on page 50 of the staff's "Generic Safety Evaluation Report BWR Scram Discharge System," December 1, 1980, for this departure from the explicit requirements of the Model TSs. We conclude that these technical bases justify a deviation from the explicit requirements of the Model TSs.

The licensee proposed in its February 26, 1981, submittal to list the SDV vent and drain valves as containment isolation valves and to perform only the surveillance required for containment isolation valves. The FRC report notes that it found this proposed surveillance to be unacceptable. It also notes that in discussions on this subject with FRC, the licensee orally agreed to revise its proposed surveillance requirements to meet the Model TS requirements for surveillance of SDV vent and drain valves.

The FRC report also notes that the licensee orally agreed to modify existing TS requirements for performing the Reactor Protection System water level-high Channel Functional Test from once per 3 months to once per month. On the basis of these two oral agreements to revise the TSs, the FRC report concludes that the licensee's proposal meets the Model TS requirements with respect to these two items and is acceptable.

The current Unit 1 TSs do not include requirements for SDV high water rod block trip setpoint or surveillance requirements. By letter dated October 1, 1981, the licensee stated that it would propose an amendment to the Unit 1 license to incorporate requirements similar to those contained in the existing Unit 2 TSs. The FRC report concludes that this proposal is acceptable.

Thus, FRC has concluded that the licensee's proposed TSs (as revised by subsequent discussion with the licensee) meet our criteria.

Subsequently, by Order dated June 24, 1983, the Commission required that the licensee 1) install the long term BWR scram discharge modifications in conformance with the staff's December 1, 1980 Generic SER on Scram Discharge Systems before December 31, 1983 and 2) submit TS changes required for operation with the modified system at least 3 months prior to the required implementation date.

By letter dated September 19, 1983, the licensee submitted proposed TS revisions in accordance with its previous oral and written commitments, as discussed above and in response to the June 24, 1983 Order. We have reviewed these proposed revisions and find that they are consistent with these previous commitments that provided the bases for acceptance in the FRC report and conclude that they are acceptable. The licensee also proposed to add an LCO for the operability of SDV vent and drain valves. This LCO would require the plant to be placed in Hot Shutdown in 12 hours if any SDV vent or drain valve is inoperable. We find this proposed LCO is consistent with the NRC staff Model TSs and conclude that it is acceptable.

By letter dated October 3, 1983, the licensee has withdrawn its February 26, 1981, submittal that requested that the SDV vent and drain valves be listed in containment isolation valve tables and be required to meet only containment isolation valve surveillance requirements. These surveillance requirements are now provided by the licensee's proposed TSs submitted in its September 19, 1983, letter as discussed above.

In its September 19, 1983, submittal, the licensee has also proposed to add TS requirements, including trip setpoint, LCO, Action Statement and surveillance requirements, for the new diverse SDV high water level scram instrumentation (Thermal Level Sensors) to the Reactor Protection System instrumentation tables. The new instruments have been given the same requirements as the original level switches which were found acceptable in the FRC report. We conclude that this proposed addition is acceptable.

By letter dated December 14, 1983, the licensee proposed to change the 30 second closure time requirement for the SDV vent and drain valves as proposed in its September 19, 1983 submittal to 60 seconds for the inboard vent and drain valves and 120 seconds for the outboard vent and drain valves. This proposal deviates from the acceptability guidelines of 30 seconds closure time provided by the staff in its Generic Safety Evaluation Report on Scram Discharge Systems dated December 1, 1980. The staff is currently reviewing the licensee's justification for this latest proposed change. In the interim, the staff has concluded that there is reasonable assurance of safe operation of the plants based on implementation of the short-term corrective measures noted in the June 24, 1983 Order; and the long-term corrective measures noted herein.

Based upon our review of the contractor's report and discussions with the reviewer and on our review, as discussed above, of the licensee's subsequent submittals augmenting the information reviewed by the contractor, we conclude that except for the SDV vent and drain valve closure time requirements as discussed above, the licensee's proposed TSs satisfy our requirements for surveillance of SDV vent and drain valves and for LCOs and surveillance requirements for SDV limit switches. Consequently, we find the licensee's proposed TSs, except for the SDV vent and drain valve closure times, acceptable.

Environmental Consideration

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

Conclusion

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

Dated: January 4, 1984

The following NRC personnel contributed to this Safety Evaluation: Ken Eccleston and George Rivenbark.

TECHNICAL EVALUATION REPORT

BWR SCRAM DISCHARGE VOLUME LONG-TERM MODIFICATIONS

GEORGIA POWER COMPANY

EDWIN I. HATCH NUCLEAR PLANT UNIT 1

NRC DOCKET NO. 50-321

NRC TAC NO. 42219

NRC CONTRACT NO. NRC-03-81-130

FRC PROJECT C5506

FRC ASSIGNMENT 2

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FOREWORD

This Technical Evaluation Report was prepared by Franklin Research Center under a contract with the U.S. Nuclear Regulatory Commission (Office of Nuclear Reactor Regulation, Division of Operating Reactors) for technical assistance in support of NRC operating reactor licensing actions. The technical evaluation was conducted in accordance with criteria established by the NRC.

SUMMARY

This technical evaluation report reviews and evaluates proposed Phase 1 changes in the Edwin I. Hatch Nuclear Plant Unit 1 Technical Specifications for scram discharge volume (SDV) long-term modifications regarding surveillance requirements for SDV vent and drain valves and the limiting condition for operation (LCO)/surveillance requirements for reactor protection system and control rod withdrawal block SDV limit switches. Conclusions were based on the degree of compliance of the Licensee's submittal with criteria from the Nuclear Regulatory Commission (NRC) staff's Model Technical Specifications.

The proposed placement of the SDV drain and vent valves in the tables of power-operated isolation valves (see revised pages 3.7-18a and 3.7-20 of the Hatch Nuclear Plant Unit 1 Technical Specifications) in order to apply isolation valve surveillance requirements to them is not acceptable. However, the Licensee's agreement to revise proposed specifications changes to require verifying each valve to be open at least once per 31 days and cycling each valve at least one complete cycle of full travel at least once per 92 days meets the NRC staff's Model Technical Specifications requirements of paragraphs 4.1.3.1.1a and 4.1.3.1.1b and is acceptable.

The Licensee's agreement to revise the original provisions of the Hatch Nuclear Plant Unit 1 Technical Specifications given on page 3.1-8, Table 4.1-1 in regard to performing the reactor protection system SDV water level-high Channel Functional Test monthly instead once per 3 months is acceptable. It meets the NRC staff's Model Technical Specifications requirements of paragraph 4.3.1.1 and Table 4.3.1.1-1.

The proposed amendment of the Hatch Nuclear Plant Unit 1 Technical Specifications to incorporate surveillance requirements for control rod withdrawal block SDV limit switches similar to those contained in the Unit 2 Technical Specifications Tables 3.3.5-1, 3.3.5-2, and 4.3.5-1 is acceptable.

Table 5-1 on pages 22 and 23 summarizes the evaluation results.

1. INTRODUCTION

1.1 PURPOSE OF THE TECHNICAL EVALUATION

The purpose of this technical evaluation report (TER) is to review and evaluate the proposed changes in the Technical Specifications of the Hatch Nuclear Plant Unit 1 boiling water reactor (BWR) in regard to "BWR Scram Discharge Volume Long Term Modification," specifically:

- o surveillance requirements for scram discharge volume (SDV) vent and drain valves
- o limiting condition for operation (LCO)/surveillance requirements for the reactor protection system limit switches
- o LCO/surveillance requirements for the control rod withdrawal block SDV limit switches.

The evaluation used criteria proposed by the NRC staff in Model Technical Specifications (see Appendix A of this report). This effort is directed toward the NRC objective of increasing the reliability of installed BWR scram discharge volume systems, the need for which was made apparent by events described below.

1.2 GENERIC ISSUE BACKGROUND

On June 13, 1979, while the reactor at Hatch Unit 1 was in the refuel mode, two SDV high level switches had been modified, tested, and found inoperable. The remaining switches were operable. Inspection of each inoperable level switch revealed a bent float rod binding against the side of the float chamber.

On October 19, 1979, Brunswick Unit 1 reported that water hammer due to slow closure of the SDV drain valve during a reactor scram damaged several pipe supports on the SDV drain line. Drain valve closure time was approximately 5 minutes because of a faulty solenoid controlling the air supply to the valve. After repair, to avoid probable damage from a scram, the unit was started with the SDV vent and drain valves closed except for periodic draining. During this mode of operation, the reactor scrammed due to a high water level in the

SDV system without prior actuation of either the high level alarm or rod block switch. Inspection revealed that the float ball on the rod block switch was bent, making the switches inoperable. The water hammer was reported to be the cause of these level switch failures.

As a result of these events involving common-cause failures of SDV limit switches and SDV drain valve operability, the NRC issued IE Bulletin 80-14, "Degradation of BWR Scram Discharge Volume Capability," on June 12, 1980 [1]. In addition, to strengthen the provisions of this bulletin and to ensure that the scram system would continue to work during reactor operation, the NRC sent a letter dated July 7, 1980 [2] to all operating BWR licensees requesting that they propose Technical Specifications changes to provide surveillance requirements for reactor protection system and control rod block SDV limit switches. The letter also contained the NRC staff's Model Technical Specifications to be used as a guide by licensees in preparing their submittals.

Meanwhile, during a routine shutdown of the Browns Ferry Unit 3 reactor on June 28, 1980, 76 of 185 control rods failed to insert fully. Full insertion required two additional manual scrams and an automatic scram for a total elapsed time of approximately 15 minutes between the first scram initiation and the complete insertion of all the rods. On July 3, 1980, in response to both this event and the previous events at Hatch Unit 1 and Brunswick Unit 1, the NRC issued (in addition to the earlier IE Bulletin 80-14) IE Bulletin 80-17 followed by five supplements. These initiated short-term and long-term programs described in "Generic Safety Evaluation Report BWR Scram Discharge System," NRC staff, December 1, 1980 [9] and "Staff Report and Evaluation of Supplement 4 to IE Bulletin 80-17 (Continuous Monitoring Systems)" [10].

Analysis and evaluation of the Browns Ferry Unit 3 and other SDV system events convinced the NRC staff that SDV systems in all BWRs should be modified to assure long-term SDV reliability. Improvements were needed in three major areas: SDV-IV hydraulic coupling, level instrumentation, and system isolation. To achieve these objectives, an Office of Nuclear Reactor Regulation (NRR) task force and a subgroup of the BWR Owners Group developed Revised Scram Discharge

System Design and Safety Criteria for use in establishing acceptable SDV systems modifications [9]. Also, an NRC letter dated October 1, 1980 requested all operating BWR licensees to reevaluate installed SDV systems and modify them as necessary to comply with the revised criteria.

In Reference 9, the SDV-IV hydraulic coupling at the Big Rock Point, Brunswick Units 1 and 2, Duane Arnold, and Hatch Units 1 and 2 BWRs was judged acceptable. The remaining BWRs will require modification to meet the revised SDV-IV hydraulic coupling criteria, and all operating BWRs may require modification to meet the revised instrumentation and isolation criteria. The changes in Technical Specifications associated with this effort will be carried out in two phases:

Phase 1 - Improvements in surveillance for vent and drain valves and instrument volume level switches.

Phase 2 - Improvements required as a result of long-term modifications made to comply with revised design and performance criteria.

This TER is a review and evaluation of Technical Specifications changes proposed for Phase 1.

1.3 PLANT-SPECIFIC BACKGROUND

The July 7, 1980 NRC letter [2] not only requested all BWR licensees to amend their facilities' Technical Specifications with respect to control rod drive SDV capability, but enclosed the NRC staff's proposed Model Technical Specifications (see Appendix A of this TER) as a guide for the licensees in preparing the requested submittals and as a source of criteria for an FRC technical evaluation of the submittals. In this TER, FRC has reviewed and evaluated the Technical Specifications changes for the Hatch Nuclear Plant Unit 1 as proposed in letters dated February 26 and October 1, 1981 (see Appendices B and C, respectively) by the Licensee, the Georgia Power Company (GPC), in regard to "BWR Scram Discharge Volume (SDV) Long-Term Modifications" and, specifically, the surveillance requirements for SDV vent and drain valves and the limiting condition for operation (LCO)/surveillance requirements for the reactor protection system and control rod withdrawal block SDV limit switches. FRC assessed the adequacy with which the GPC information documented compliance of the proposed Technical Specifications changes with the NRC staff's Model Technical Specifications.

2. REVIEW CRITERIA

The criteria established by the NRC staff's Model Technical Specifications involving surveillance requirements of the main SDV components and instrumentation cover three areas of concern:

- o surveillance requirements for SDV vent and drain valves
- o LCO/surveillance requirements for reactor protection system SDV limit switches
- o LCO/surveillance requirements for control rod block SDV limit switches.

2.1 SURVEILLANCE REQUIREMENTS FOR SDV DRAIN AND VENT VALVES

The surveillance criteria of the NRC staff's Model Technical Specifications for SDV drain and vent valves are:

"4.1.3.1.1 - The scram discharge volume drain and vent valves shall be demonstrated OPERABLE by:

a. Verifying each valve to be open* at least once per 31 days, and
b. Cycling each valve at least one complete cycle of full travel at least once per 92 days.

*These valves may be closed intermittently for testing under administrative controls."

The Model Technical Specifications require testing the drain and vent valves, checking at least once in every 31 days that each valve is fully open during normal operation, and cycling each valve at least one complete cycle of full travel under adminstrative controls at least once per 92 days.

Full opening of each valve during normal operation indicates that there is no degradation in the control air system and its components that control the air pressure to the pneumatic actuators of the drain and vent valves. Cycling each valve checks whether the valve opens fully and whether its movement is smooth, jerky, or oscillatory.

During normal operation, the drain and vent valves stay in the open position for very long periods. A silt of particulates such as metal chips and flakes, various fibers, lint, sand, and weld slag from the water or air may accumulate at moving parts of the valves and temporarily "freeze" them. A strong breakout force may be needed to overcome this temporary "freeze," producing a violent jerk which may induce a severe water hammer if it occurs during a scram or a scram resetting. Periodic cycling of the drain and vent valves is the best method to clear the effects of particulate silting, thus promoting smooth opening and closing and more reliable valve operation. Also, in case of improper valve operation, cycling can indicate whether excessive pressure transients may be generated during and after a reactor scram which might damage the SDV piping system and cause a loss of system integrity or function.

2.2 LCO/SURVEILLANCE REQUIREMENTS FOR REACTOR PROTECTION SYSTEM SDV LIMIT SWITCHES

The paragraphs of the NRC staff's Model Technical Specifications pertinent to LCO/surveillance requirements for reactor protection system SDV limit switches are:

> "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.

Table 3.3.1-1. Reactor Protection System Instrumentation

Functional Unit	Applicable Operational Conditions	Minimum Operable Channels Per Trip System (a)	Action
8. Scram Discharge Volume Water Level-High	1,2,5 (h)	2	4

Table 3.3.1-2. Reactor Protection System Response Times

Functional Unit	Response Time (Seconds)
8. Scram Discharge Volume Water Level-High	NA"

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

Table 4.3.1.1-1. Reactor Protection System Instrumentation Surveillance Requirements

Functional Unit	Channel Check	Channel Functional Test	Channel Calibration	Operational Conditions in Which Surveillance Required
8. Scram Discharge Volume Water Level-High	NA	M	R	1,2,5

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

(h) With any control rod withdrawn. Not applicable to control rods removed per Specification 3.9.10.1 or 3.9.10.2

Action 4: In OPERATIONAL CONDITION 1 or 2, be in at least HOT SHUTDOWN within 6 hours.

In OPERATIONAL CONDITION 5, suspend all operations involving CORE ALTERATIONS* and fully insert all insertable control rods within one hour.

*Except movement of IRM, SRM or special movable detectors, or replacement of LPRM strings provided SRM instrumentation is OPERABLE per Specification 3.9.2."

Paragraph 3.3.1 and Table 3.3.1-1 of the Model Technical Specifications require the functional unit of SDV water level-high to have at least 2 operable channels containing 2 limit switches per trip system, for a total of 4 operable channels containing 4 limit switches per 2 trip systems for the reactor protection system which automatically initiates a scram. The technical objective of these requirements is to provide 1-out-of-2-taken-twice logic for the reactor protection system. The response time of the reactor protection system for the functional unit of SDV water level-high should be measured and kept available (it is not given in Table 3.3.1-2).

Paragraph 4.3.1.1 and Table 4.3.1.1-1 give reactor protection system instrumentation surveillance requirements for the functional unit of SDV water level-high. Each reactor protection system instrumentation channel containing a limit switch should be shown to be operable by the Channel Functional Test monthly and Channel Calibration at each refueling outage.

2.3 LCO/SURVEILLANCE REQUIREMENTS FOR CONTROL ROD WITHDRAWAL BLOCK SDV LIMIT SWITCHES

The NRC staff's Model Technical Specifications specify the following LCO/surveillance requirements for control rod withdrawal block SDV limit switches:

"3.3.6 - The control rod withdrawal block instrumentation channel shown in Table 3.3.6-1 shall be OPERABLE with trip setpoints set consistent with the values shown in the Trip Setpoint column of Table 3.3.6-2.

Table 3.3.6-1. Control Rod Withdrawal Block Instrumentation

Trip Function	Minimum Operable Channels Per Trip Function	Applicable Operational Conditions	Action
5. Scram Discharge Volume			
a. Water level-high	2	1, 2, 5**	62
b. Scram trip bypassed	1	(1, 2, 5**)	62

ACTION 62: With the number of OPERABLE channels less than required by the minimum OPERABLE channels per Trip Function requirement, place the inoperable channel in the tripped condition within one hour.

**With more than one control rod withdrawn. Not applicable to control rods removed per Specification 3.9.10.1 or 3.9.10.2.

Table 3.3.6-2. Control Rod Withdrawal Block Instrumentation Setpoints

Trip Function	Trip Setpoint	Allowable Value
5. Scram Discharge Volume		
a. Water level-high	To be specified	NA
b. Scram trip bypassed	NA	NA"

"4.3.6 - Each of the above control rod withdrawal block trip systems and instrumentation channels shall be demonstrated OPERABLE by the performance of the CHANNEL CHECK, CHANNEL FUNCTIONAL TEST and CHANNEL CALIBRATION operations for the OPERATIONAL CONDITIONS and at the frequencies shown in Table 4.3.6-1.

Table 4.3.6-1. Control Rod Withdrawal Block Instrumentation Surveillance Requirements

Trip Function	Channel Check	Channel Functional Test	Channel Calibration	Operational Conditions in Which Surveillance Required
5. Scram Discharge Volume				
a. Water Level-High	NA	Q	R	1, 2, 5**
b. Scram Trip Bypassed	NA	M	NA	(1, 2, 5**)

**With more than one control rod withdrawn. Not applicable to control rods removed per Specification 3.9.10.1 or 3.9.10.2."

Paragraph 3.3.6 and Table 3.3.6-1 of the Model Technical Specifications require the control rod withdrawal block instrumentation to have at least 2 operable channels containing 2 limit switches for SDV water level-high and 1 operable channel containing 1 limit switch for SDV scram trip bypassed. The technical objective of these requirements is to have at least one channel containing one limit switch available to monitor the SDV water level when the other channel with a limit switch is being tested or undergoing maintenance. The trip setpoint for control rod withdrawal block instrumentation monitoring

SDV water level-high should be specified as indicated in Table 3.3.6-2. The trip function prevents further withdrawal of any control rod when the control rod block SDV limit switches indicate water level-high.

Paragraph 4.3.6 and Table 4.3.6-1 require that each control rod withdrawal block instrumentation channel containing a limit switch be shown to be operable by the Channel Functional Test once per 3 months for SDV water level-high, by the Channel Functional Test once per month for SDV scram trip bypassed, and by Channel Calibration at each refueling outage for SDV water level-high.

The Surveillance Criteria of the BWR Owners Subgroup given in Appendix A, "Long-Term Evaluation of Scram Discharge System," of "Generic Safety Evaluation Report BWR Scram Discharge System," written by the NRC staff and issued on December 1, 1980, are:

1. Vent and drain valves shall be periodically tested.
2. Verifying and level detection instrumentation shall be periodically tested in place.
3. The operability of the entire system as an integrated whole shall be demonstrated periodically and during each operating cycle, by demonstrating scram instrument response and valve function at pressure and temperature at approximately 50% control rod density.

Analysis of the above criteria indicates that the NRC staff's Model Technical Specifications requirements, the acceptance criteria for the present TER, fully cover the BWR Owners Subgroup Surveillance Criteria 1 and 2 and partially cover Criterion 3.

3. METHOD OF EVALUATION

The GPC submittal for the Hatch Nuclear Plant Unit 1 was evaluated in two stages, initial and final.

During the initial evaluation, only the NRC staff's Model Technical Specifications requirements were used to determine if:

- o the Licensee's submittal was responsive to the July 7, 1980 NRC request for proposed Technical Specifications changes involving the surveillance requirements of the SDV vent and drain valves, LCO/surveillance requirements for reactor protection system SDV limit switches, and LCO/surveillance requiremenmts for control rod block SDV limit switches
- o the submitted information was sufficient to permit a detailed technical evaluation.

During the final evaluation, in addition to the NRC staff's Model Technical Specifications requirements, background material in References 1 through 10, pertinent sections of "Georgia Power Company Edwin I. Hatch Nuclear Plant Unit 1 Safety Analysis Report," and Hatch Nuclear Plant Unit 1 Technical Specifications were studied to determine the technical bases for the design of SDV main components and instrumentation. Subsequently, the Licensee's response was compared directly to the requirements of the NRC staff's Model Technical Specifications. The findings of the final evaluation are presented in Section 4 of this report.

The initial evaluation concluded that the Licensee's submittal was responsive to the NRC's request of July 7, 1980, but did not contain sufficient information to permit preparation of a TER. A request for additional information (RFI) was sent to GPC by the NRC on September 1, 1981. Thus, this TER is based on the Licensee's initial submittal (see Appendix B) and the Licensee's response to the RFI, dated October 1, 1981 (see Appendix C).

4. TECHNICAL EVALUATION

4.1 SURVEILLANCE REQUIREMENTS FOR SDV DRAIN AND VENT VALVES

NRC STAFF'S MODEL TECHNICAL SPECIFICATIONS

Paragraph 4.1.3.1.1 requires demonstrating that the SDV drain and vent valves are operable by:

a. verifying each valve to be open at least once per 31 days (valves may be closed intermittently for testing under administrative controls), and

b. cycling each valve at least one complete cycle of full travel at least once per 92 days.

LICENSEE RESPONSE

The Licensee proposed to revise pages 3.7-18a and 3.7-20 of the Hatch Nuclear Plant Unit 1 Technical Specifications. The revised page 3.7-18a contains Table 3.7-1 with the information given below.

"Table 3.7-1. Primary Containment Isolation Valves

Isolation Group (g)	Valve Identification	Number of Power Operated Valves Inside	Number of Power Operated Valves Outside	Maximum Operating Time (sec)	Normal Position (a)	Action On Initiating Signal (a)
(f)	Scram Discharge Vent Valves (C11-F010A, C11-F010B)		2	60	0	GC
(f)	Scram Discharge Drain Valve (C11-F011)		1	60	0	GC

(g)	Valve Identification	Number of Valves Inside	Number of Valves Outside	(g)	Normal Position	(g)
	Scram Discharge Volume Relief Valve (C11-F012)		1		C"	

From revised page 3.7-20: "Notes to Table 3.7-1 (Concluded)
(f) Valves receive isolation signal on any scram
(g) Not applicable"

In response to the RFI, the Licensee provided the following statement:

"Item 1

The model Technical Specifications contained in your July 7, 1980, letter placed the scram discharge volume vent and drain valves in section 3/4.1.3.1 of the model Technical Specifications; 'Control Rod Operability.' Item 1 of the FRC request asked for a reference to the section of the Technical Specifications where the requested change is incorporated.

Our February 26, 1981, letter proposed that these valves be placed in the tables of power operated containment isolation valves instead of the 'Control Rod Operability' section. These valves do not affect control rod operability at Plant Hatch. The plant unique geometry of this system at Plant Hatch allows free communication between the scram level switches and the scram discharge volume (SDV). Thus, the level switches, not the vent and drain valves, protect the scram function, and in a sense control rod operability, by providing assurance that the SDV is empty. The vent and drain valves are important, however, insofar as they provide a containment pressure boundary during the time that a scram is sealed-in. Forthis reason we have chosen to place the valves in the tables of containment isolation valves. The surveillance requirements are therefore different than those proposed by the model Technical Specifications in order to be consistent with the requirements for other comparable containment isolation valves."

The Licensee agreed to revise the proposed specifications changes to require:

a. verifying each valve to be open at least once per 31 days (valves may be closed intermittently for testing under administrative controls), and

b. cycling each valve at least one complete cycle of full travel at least once per 92 days.

FRC EVALUATION

The proposed placement of the SDV drain and vent valves in the tables of power-operated isolation valves (see revised pages 3.7-18a and 3.7-20 of the Hatch Nuclear Plant Unit 1 Technical Specifications) in order to apply isolation valve surveillance requirements to them is not acceptable. However, the Licensee's agreement to revise proposed specifications changes to require verifying each valve to be open at least once per 31 days and cycling each valve at least one complete cycle of full travel at least once per 92 days meets the NRC staff's Model Technical Specifications requirements of paragraphs 4.1.3.1.1a and 4.1.3.1.1b and is acceptable.

4.2 LCO/SURVEILLANCE REQUIREMENTS FOR REACTOR PROTECTION SYSTEM SDV LIMIT SWITCHES

NRC STAFF'S MODEL TECHNICAL SPECIFICATIONS

Paragraph 3.3.1 and Table 3.3.1-1 require the functional unit of SDV water level-high to have at least 2 operable channels containing 2 limit switches per trip system, for a total of 4 operable channels containing 4 limit switches per 2 trip systems for the reactor protection system which automatically initiates scram.

Paragraph 3.3.1 and Table 3.3.1-2 concern the response time of the reactor protection system for the functional unit of SDV water level-high which should be specified for each BWR (it is not specified in the table). Paragraph 4.3.1.1 and Table 4.3.1.1-1 require that each reactor protection system instrumentation channel containing a limit switch be shown to be operable for the functional unit of SDV water level-high by the Channel Functional Test monthly and Channel Calibration at each refueling outage. The applicable operational conditions for these requirements are startup, run, and refuel.

LICENSEE RESPONSE

The Licensee provided the following information in answer to the RFI:

"As indicated in our October 10, 1980, letter the scram level switches are currently covered by Technical Specifications on each unit. For Unit 1, please refer to Specifications 3.1 and 4.1, Tables 3.1-1 and 4.1-1, item 7. For Unit 2, the appropriate reference is Specification 3/4.3.1, tables 3.3.1-1 and 4.3.1-1, item 8. The instrument functional test frequency for Unit 1 is once every three months as initially approved by the Commission on issuance of the Unit 1 Operating License. We have not proposed to modify this specification."

Page 3.1-4 of the Hatch Nuclear Plant Unit 1 Technical Specifications addresses the NRC staff's Model Technical Specifications requirements of paragraph 3.3.1 and Table 3.3.1-1, giving the following information in Table

3.1-1 (Cont.), Reactor Protection System (RPS) Instrumentation Requirements, for "Source of Scram Trip Signal Scram Discharge Volume High High Level":

"1. Scram Number(a): 7

2. Operable Channels Required Per Trip System(b): 2

3. Operable Trip Setting: ≤ 71 gallons

4. Source of Scram Signal is Required to be Operable Except as Indicated Below: Permissible to bypass (initiates control rod block) in order to reset RPS when the Mode Switch is in the Refuel or Shutdown position."

Notes for Table 3.1-1:

"a. The column entitled 'Scram Number' is for convenience so that a one-to-one relationship can be established between items in Table 3.1-1 and items in Table 4.1-1.

b. There shall be two operable or tripped trip systems for each potential scram signal. If the number of operable channels cannot be met for one of the trip systems, that trip system shall be tripped. However, one trip signal channel of a trip system may be inoperable for up to two (2) hours during periods of required surveillance testing without tripping the associated trip system, provided that the other remaining channel(s) monitoring that parameter within that trip system is (are) operable."

Page 3.3-12 of the Hatch Nuclear Plant Unit 1 Technical Specifications gives the reactor protection system response time as follows:

"In the analytical treatment of the transients, 390 milliseconds are allowed between a neutron sensor reaching the scram point and start of negative reactivity insertion. This is adequate and conservative when compared to the typically observe time delay of about 270 milliseconds. Approximately 70 milliseconds after neutron flux reaches the trip point, the pilot scram valve solenoid power supply voltage goes to zero and approximately 200 milliseconds later control rod motion begins. The 200 milliseconds are included in the allowable scram insertion times specified in specification 3.3.C."

This covers the requirements of paragraph 3.3.1 and Table 3.3.1-2 of the NRC staff's Model Technical Specifications.

Pages 3.1-7, 3.1-8, and 3.1-9 of the Hatch Nuclear Power Plant Unit 1 Technical Specifications address the NRC staff's Model Technical Specifications requirements of paragraph 4.3.1.1 and Table 4.3.1.1-1, providing Table 4.1-1, Reactor Protection System (RPS) Instrumentation Functional Test, Functional

Test Minimum Frequency, and Calibration Minimum Frequency, with the following information for "Source of Scram Trip Signal Scram Discharge Volume High High Level":

"1. Scram Number(a): 7

2. Group(b): A

3. Instrument Functional Test Minimum Frequency(c): Every 3 months

4. Instrument Calibration Minimum Frequency: (h)"

Notes for Table 4.1-1:

"a. See Note a for Table 3.1-1.

b. The definition of ... Group A: On-off sensors that provide a scram trip signal.

c. Functional tests are not required when the systems are not required to be operable or are tripped. However, if functional tests are missed they shall be performed prior to returning the system to an operable status.

h. Physical inspection and actuation of these position switches will be performed once per operating cycle."

The Licensee agreed to revise the original provisions of the Hatch Nuclear Plant Unit 1 Technical Specifications given on page 3.1-8, Table 4.1-1, in regard to performing the reactor protection system SDV water level-high Channel Functional Test monthly instead once per 3 months.

FRC EVALUATION

The Licensee's response to the NRC staff's Model Technical Specifications requirements of paragraph 3.3.1 and Table 3.3.1-1 is acceptable. The Hatch Nuclear Plant Unit 1 reactor protection system SDV water level-high instrumentation consists of 2 operable channels containing 2 limit switches per trip system, for a total of 4 operable channels containing 4 limit switches per 2 trip systems, making 1-out-of-2-taken-twice logic. The original page 3.1-4 with Table 3.1-1 also specifies $\leq$ 71 gal as a trip setting for scram initiation, which is acceptable.

The reactor protection system response time of 390 milliseconds specified on original page 3.3-12 of the Hatch Nuclear Plant Unit 1 Technical Specifications addresses the requirements of paragraph 3.3.1 and Table 3.3.1-2 and is acceptable.

The Licensee's agreement to revise the original provisions of the Hatch Nuclear Plant Unit 1 Technical Specifications given in page 3.1-8, Table 4.1-1, in regard to performing the reactor protection system SDV water level-high Channel Functional Test monthly instead of once per 3 months meets the NRC staff's Model Technical Specifications requirements of paragraph 4.3.1.1 and Table 4.3.1.1-1 and is acceptable.

4.3 LCO/SURVEILLANCE REQUIREMENTS FOR CONTROL ROD WITHDRAWAL BLOCK SDV LIMIT SWITCHES

NRC STAFF'S MODEL TECHNICAL SPECIFICATIONS

Paragraph 3.3.6 and Table 3.3.6-1 require the control rod withdrawal block instrumentation to have at least 2 operable channels containing 2 limit switches for SDV water level-high, and 1 operable channel containing 1 limit switch for SDV trip bypassed. Paragraph 3.3.6 also requires specifying the trip setpoint for control rod withdrawal block instrumentation monitoring SDV water level-high as indicated in Table 3.3.6-2.

Paragraph 4.3.6 and Table 4.3.6-1 require that each control rod withdrawal block instrumentation channel containing a limit switch be shown to be operable by the Channel Functional Test once per 3 months for SDV water level-high, by the Channel Functional Test once per month for SDV scram trip bypassed, and by Channel Calibration at each refueling outage for SDV water level-high.

LICENSEE RESPONSE

The Licensee responded to the RFI as follows:

"The SDV rod block setpoint and surveillance requirements are specified in Unit 2 Technical Specification Section 3.3.5 and Tables 3.3.5-1, 3.3.5-2 and 4.3.5-1. In reviewing the Technical Specifications for our February 26, 1981 submittal, the absence of a comparable specification in the Unit 1 Technical Specifications was not noted. We agreee that it is appropriate to specify the limits and surveillance requirements for the

SDV rod block alarm switch and will propose an amendment to the Unit 1 license to incorporate requirements similar to those contained in our Unit 2 Specifications refesenced above."

As seen from the above statement, the present Hatch Nuclear Plant Unit 1 Technical Specifications do not contain any LCO/surveillance requirements for control rod withdrawal block SDV limit switches. The Licensee proposes to amend the Hatch Nuclear Plant Unit 1 Technical Specifications to incorporate surveillance requirements similar to those contained in the Unit 2 Technical Specifications given in Section 3.3.5 and Tables 3.3.5-1, 3.3.5-2, and 4.3.5-1. The information contained in these tables will be evaluated in this TER for the Hatch Nuclear Plant Unit 1.

The information provided in Table 3.3.5-1, Control Rod Withdrawal Block Instrumentation, is as follows for "Trip Function Scram Discharge Volume Water Level-High":

"1. Minimum Number of Operable Channels per Trip Function: 1

2. Applicable Operational Conditions: 1, 2, 5 (f)"

Note:

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

Table 3.3.5-2, Control Rod Withdrawal Block Instrumentation Setpoints, contains the following information for "Trip Function Scram Discharge Volume Water Level-High":

"1. Trip Setpoint: $\leq$ 36.2 gallons

2. Allowable Value: $\leq$ 36.2 gallons"

The contents of Table 3.3.5-1 and 3.3.5-2 address the NRC staff's Model Technical Specifications requirements of paragraph 3.3.6 and Table 3.3.6-1. Table 4.3.5-1, Control Rod Withdrawal Block Instrumentation Surveillance Requirements, addresses the NRC staff's Model Technical Specifications requirements of paragraph 4.3.6 and Table 4.3.6-1, providing the following information for "Trip Function Scram Discharge Volume Water Level-High":

"1. Channel Check: NA

2. Channel Functional Test: Q (quarterly)

3. Channel Calibration (a): R (each refueling)

4. Operational Conditions in Which Surveillance Required: 1, 2, 5(e)"

Notes:

"a. Neutron detectors may be excluded from CHANNEL CALIBRATION.
e. With any control rod withdrawn. Not applicable to control rods removed per Specification 3.9.11.1 or 3.9.11.2."

FRC EVALUATION

The existing Hatch Unit 1 scram discharge system has six level switches on the scram discharge volume (see FSAR, page 3-43) set at three different water levels to guard against operation of the reactor without sufficient free volume present in the scram discharge headers to receive the scram discharge water in the event of a scram. At the first (lowest) level, one level switch initiates an alarm for operator action. At the second level, with the setpoint of ≤36.2 gallons (see page 3/43-40, Table 3.3.5-2, of the Hatch Unit 2 Technical Specifications), one level switch initiates a rod withdrawal block to prevent further withdrawal of any control rod. At the third (highest) level, with the setpoint of ≤71 gallons (see page 3.1-4, Table 3.1-1, of the Hatch Unit 1 Technical Specifications), the four level switches (two for each reactor protection system trip system) initiate a scram to shut down the reactor while sufficient free volume is available to receive the scram discharge water. Reference 9, page 50, defines Design Criterion 9 ("Instrumentation shall be provided to aid the operator in the detection of water accumulation in the instrumented volume(s) prior to scram initiation"), gives the technical basis for "Long-Term Evaluation of Scram Discharge System," and defines acceptable compliance ("The present alarm and rod block instrumentation meets this criterion given adequate hydraulic coupling with the SDV headers"). The Hatch scram discharge system has adequate hydraulic coupling between scram discharge headers and instrumented volume. Thus, the present alarm and rod block instrumentation is also acceptable.

In Hatch Unit 1, "Scram Discharge Volume Scram Trips" cannot be bypassed while the reactor is in operational conditions of startup and run (see FSAR page 7-17), and operational condition "refuel with more than one control rod withdrawn" is not applicable, since interlocks are provided which prevent the withdrawal of more than one control rod with the mode switch in the refuel position. Thus, the NRC staff's Model Technical Specifications requirements of paragraph 3.3.6 with Table 3.3.6-1 and paragraph 4.3.6 with Table 4.3.6-1 are not applicable to Hatch Unit 1 for "Trip Function 5.b, SDV Scram Trip Bypassed."

The proposed trip setpoint of $\leq$ 36.2 gallons for control rod withdrawal block instrumentation channel is acceptable. The Licensee's proposed amendment of the Hatch Nuclear Plant Unit 1 Technical Specifications to incorporate surveillance requirements similar to those contained in the Unit 2 Technical Specifications of Table 4.3.5-1 is acceptable. It prescribes the Channel Functional Test of each control rod withdrawal block instrumentation channel containing a limit switch quarterly and Channel Calibration each refueling for SDV water level-high.

5. CONCLUSIONS

Table 5-1 summarizes the results of the final review and evaluation of the Hatch Nuclear Plant Unit 1 Phase 1 proposed Technical Specifications changes for SDV long-term modification in regard to surveillance requirements for SDV vent and drain valves and LCO/surveillance requirements for reactor protection system and control rod block SDV limit switches. The following conclusions were made:

- o The proposed placement of the SDV drain and vent valves in the tables of power-operated isolation valves (see revised pages 3.7-18a and 3.7-20 of the Hatch Nuclear Plant Unit 1 Technical Specifications) in order to apply isolation valve surveillance requirements to them is not acceptable. However, the Licensee's agreement to revise proposed specifications changes to require verifying each valve to be open at least once per 31 days and cycling each valve at least one complete cycle of full travel at least once per 92 days meets the NRC staff's Model Technical Specifications requirements of paragraphs 4.1.3.1.1a and 4.1.3.1.1b and is acceptable.

- o The Licensee's agreement to revise the original provisions of the Hatch Nuclear Plant Unit 1 Technical Specifications given on page 3.1-8, Table 4.1-1, in regard to performing the reactor protection system SDV water level-high Channel Functional Test monthly instead once per 3 months meets the NRC staff's Model Technical Specifications requirements of paragraph 4.3.1.1 and Table 4.3.1.1-1 and is acceptable.

- o The proposed amendment of the Hatch Nuclear Plant Unit 1 Technical Specifications to incorporate surveillance requirements for control rod withdrawal block SDV limit switches similar to those contained in the Unit 2 Technical Specifications Tables 3.3.5-1, 3.3.5-2, and 4.3.5-1 is acceptable.

Table 5-1. Evaluation of Proposed Phase 1 Technical Specifications Changes for Scram Discharge Volume Long-Term Modifications Hatch Nuclear Plant Unit 1

	Technical Specifications		
Surveillance Requirements	NRC Staff Model (Paragraph)	Proposed by Licensee	Evaluation
SDV DRAIN AND VENT VALVES			
Verify each valve open	Once per 31 days (4.1.3.1.1a)	Once per 31 days (p. 3.7-13*)	Acceptable
Cycle each valve one complete cycle	Once per 92 days (4.1.3.1.1b)	Once per 92 days (p. 3.7-13*)	Acceptable
REACTOR PROTECTION SYSTEM SDV LIMIT SWITCHES			
Minimum operable channels per trip system	2 (3.3.1, Table 3.3.1-1)	2 (p. 3.1-4, Table 3.1-1)	Acceptable
SDV water level-high response time	NA (3.3.1, Table 3.3.1-2)	0.390 sec. max. 0.270 sec. test. (p. 3.3-12)	Acceptable
SDV water level-high			
Channel functional test	Monthly (4.3.1.1, Table 4.3.1.1-1)	Monthly (p. 3.1-7, 3.1-8,* 3.1-9)	Acceptable
Channel calibration	Each refueling (4.3.1.1, Table 4.3.1.1-1)	Once per operating cycle (p. 3.1-7, 3.1-8, 3.1-9)	Acceptable

*The Licensee agreed to revise pp. 3.7-13 and 3.1.8.

Table 5-1 (Cont.)

Surveillance Requirements	Technical Specifications: NRC Staff Model (Paragraph)	Technical Specifications: Proposed by Licensee	Evaluation
CONTROL ROD BLOCK SDV LIMIT SWITCHES			
Minimum operable channels per trip function			
SDV water level-high	2 (3.3.6, Table 3.3.6-1)	1 (As for Unit 2, Table 3.3.5-1)	Acceptable*
SDV scram trip bypassed	1 (3.3.6, Table 3.3.6-1)	NA (As for Unit 2, Table 3.3.5-1)	Acceptable*
SDV water level-high			
Trip setpoint	NA (3.3.6, Table 3.3.6-2)	<36.2 gallons (As for Unit 2, Table 3.3.5-2)	Acceptable
Channel functional test	Quarterly (4.3.6, Table 4.3.6-1)	Quarterly (As for Unit 2, Table 4.3.5-1)	Acceptable
Channel calibration	Each refueling (4.3.6, Table 4.3.6-1)	Each refueling (As for Unit 2, Table 4.3.5-1)	Acceptable
SDV scram trip bypassed			
Channel functional test	Monthly (4.3.6, Table 4.3.6-1)	NA	Acceptable*

* See Reference 9, p. 50, and p. 20 of this TER.

6. REFERENCES

1. IE Bulletin 80-14, "Degradation of BWR Scram Discharge Volume Capacity"
   NRC, Office of Inspection and Enforcement, June 12, 1980

2. D. G. Eisenhut (NRR), letter "To All Operating Boiling Water Reactors (BWRs)" with enclosure, "Model Technical Specifications"
   July 7, 1980

3. IE Bulletin 80-17, "Failure of 76 of 185 Control Rods to Fully Insert During a Scram at a BWR"
   NRC, Office of Inspection and Enforcement, July 3, 1980

4. IE Bulletin 80-17, Supplement 1, "Failure of 76 of 185 Control Rods to Fully Insert During a Scram at a BWR"
   NRC, Office of Inspection and Enforcement, July 18, 1980

5. IE Bulletin 80-17, Supplement 2, "Failures Revealed by Testing Subsequent to Failure of Control Rods to Insert During a Scram at a BWR"
   NRC, Office of Inspection and Enforcement, July 22, 1980

6. IE Bulletin 80-17, Supplement 3, "Failure of Control Rods to Insert During a Scram at a BWR"
   NRC, Office of Inspection and Enforcement, August 22, 1980

7. IE Bulletin 80-17, Supplement 4, "Failure of Control Rods to Insert During a Scram at a BWR"
   NRC, Office of Inspection and Enforcement, December 18, 1980

8. IE Bulletin 80-17, Supplement 5, "Failure of Control Rods to Insert During a Scram at a BWR"
   NRC, Office of Inspection and Enforcement, February 13, 1981

9. P. S. Check (NRR), memorandum with enclosure, "Generic Safety Evaluation Report BWR Scram Discharge System"
   December 1, 1980

10. P. S. Check (NRR), memorandum with enclosure, "Staff Report and Evaluation of Supplement 4 to IE Bulletin 80-17"
    June 10, 1981

APPENDIX A

NRC STAFF'S MODEL TECHNICAL SPECIFICATIONS*

* Note: Applicable changes are marked by vertical lines in the margins.

REACTIVITY CONTROL SYSTEMS

LIMITING CONDITION FOR OPERATION (Continued)

ACTION (Continued)

2. If the inoperable control rod(s) is inserted, within one hour disarm the associated directional control valves either:

   a) Electrically, or

   b) Hydraulically by closing the drive water and exhaust water isolation valves.

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

c. With more than 8 control rods inoperable, be in at least HOT SHUTDOWN within 12 hours.

SURVEILLANCE REQUIREMENTS

4.1.3.1.1 The scram discharge volume drain and vent valves shall be demonstrated OPERABLE by:

a. Verifying each valve to be open* at least once per 31 days and

b. Cycling each valve through at least one complete cycle of full travel at least once per 92 days.

4.1.3.1.2 When above the preset power level of the RWM and RSCS, all withdrawn control rods not required to have their directional control valves disarmed electrically or hydraulically shall be demonstrated OPERABLE by moving each control rod at least one notch:

a. At least once per 7 days, and

b. At least once per 24 hours when any control rod is immovable as a result of excessive friction or mechanical interference.

4.1.3.1.3 All control rods shall be demonstrated OPERABLE by performance of Surveillance Requirements 4.1.3.2, 4.1.3.4, 4.1.3.5, 4.1.3.6 and 4.1.3.7.

*These valves may be closed intermittently for testing under administrative controls.

REACTIVITY CONTROL SYSTEMS

CONTROL ROD MAXIMUM SCRAM INSERTION TIMES

LIMITING CONDITION FOR OPERATION

3.1.3.2 The maximum scram insertion time of each control rod from the fully withdrawn position to notch position (6), based on de-energization of the scram pilot valve solenoids as time zero, shall not exceed (7.0) seconds.

APPLICABILITY: OPERATIONAL CONDITIONS 1 and 2.

ACTION:

With the maximum scram insertion time of one or more control rods exceeding (7.0) seconds:

a. Declare the control rod(s) with the slow insertion time inoperable, and

b. Perform the Surveillance Requirements of Specification 4.1.3.2.c at least once per 60 days when operation is continued with three or more control rods with maximum scram insertion times in excess of (7.0) seconds, or

c. Be in at least HOT SHUTDOWN within 12 hours.

SURVEILLANCE REQUIREMENTS

4.1.3.2 The maximum scram insertion time of the control rods shall be demonstrated through measurement with reactor coolant pressure greater than or equal to 950 psig and, during single control rod scram time tests, the control rod drive pumps isolated from the accumulators:

a. For all control rods prior to THERMAL POWER exceeding 40% of RATED THERMAL POWER following CORE ALTERATIONS or after a reactor shutdown that is greater than 120 days,

b. For specifically affected individual control rods following maintenance on or modification to the control rod or control rod drive system which could affect the scram insertion time of those specific control rods, and

c. For 10% of the control rods, on a rotating basis, at least once per 120 days of operation.

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.

APPLICABILITY: As shown in Table 3.3.1-1.

ACTION:

a. With the number of OPERABLE channels less than required by the Minimum OPERABLE Channels per Trip System requirement for one trip system, place at least one inoperable channel in the tripped condition within one hour.

b. With the number of OPERABLE channels less than required by the Minimum OPERABLE Channels per Trip System requirement 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 FUNCTIONAL TEST and CHANNEL CALIBRATION operations for the OPERATIONAL CONDITIONS and at the frequencies shown in Table 4.3.1.1-1.

4.3.1.2 LOGIC SYSTEM FUNCTIONAL TESTS and simulated automatic operation of all channels shall be performed at least once per 18 months.

4.3.1.3 The REACTOR PROTECTION SYSTEM RESPONSE TIME of each reactor trip function shown in 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 condition, except when this would cause the Trip Function to occur.

TABLE 3.3.1-1 (Continued)

REACTOR PROTECTION SYSTEM INSTRUMENTATION

FUNCTIONAL UNIT	APPLICABLE OPERATIONAL CONDITIONS	MINIMUM OPERABLE CHANNELS PER TRIP SYSTEM (a)	ACTION
8. Scram Discharge Volume Water Level - High	1, 2, 5[h]	2	4
9. Turbine Stop Valve - Closure	1[i]	4[j]	7
10. Turbine Control Valve Fast Closure, Trip Oil Pressure - Low	1[i]	2[j]	7
11. Reactor Mode Switch in Shutdown Position	1, 2, 3, 4, 5	1	8
12. Manual Scram	1, 2, 3, 4, 5	1	9

TABLE 3.3.1-1 (Continued)

REACTOR PROTECTION SYSTEM INSTRUMENTATION

ACTION

ACTION 1 - In OPERATIONAL CONDITION 2, be in at least HOT SHUTDOWN within 6 hours.

In OPERATIONAL CONDITION 5, suspend all operations involving CORE ALTERATIONS* and fully insert all insertable control rods within one hour.

ACTION 2 - Lock the reactor mode switch in the Shutdown position within one hour.

ACTION 3 - Be in at least STARTUP within 2 hours.

ACTION 4 - In OPERATIONAL CONDITION 1 or 2, be in at least HOT SHUTDOWN within 6 hours.

In OPERATIONAL CONDITION 5, suspend all operations involving CORE ALTERATIONS* and fully insert all insertable control rods within one hour.

ACTION 5 - Be in at least HOT SHUTDOWN within 6 hours.

ACTION 6 - Be in STARTUP with the main steam line isolation valves closed within 2 hours or in at least HOT SHUTDOWN within 6 hours.

ACTION 7 - Initiate a reduction in THERMAL POWER within 15 minutes and reduce turbine first stage pressure to < (250) psig, equivalent to THERMAL POWER less than (30)% of RATED THERMAL POWER, within 2 hours..

ACTION 8 - In OPERATIONAL CONDITION 1 or 2, be in at least HOT SHUTDOWN within 6 hours.

In OPERATIONAL CONDITION 3 or 4, verify all insertable control rods to be fully inserted within one hour.

In OPERATIONAL CONDITION 5, suspend all operations involving CORE ALTERATIONS* and fully insert all insertable control rods within one hour.

ACTION 9 In OPERATIONAL CONDITION 1 or 2, be in at least HOT SHUTDOWN within 6 hours.

In OPERATIONAL CONDITION 3 or 4, lock the reactor mode switch in the Shutdown position within one hour.

In OPERATIONAL CONDITION 5, suspend all operations involving CORE ALTERATIONS* and fully insert all insertable control rods within one hour.

*Except movement of IRM, SRM or special movable detectors, or replacement of LPRM strings provided SRM instrumentation is OPERABLE per Specification 3.9.2.

TABLE 3.3.1-1 (Continued)

REACTOR PROTECTION SYSTEM INSTRUMENTATION

TABLE NOTATIONS

(a) 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 OPERABLE channel in the same trip system is monitoring that parameter.

b) The "shorting links" shall be removed from the RPS circuitry prior to and during the time any control rod is withdrawn* and shutdown margin demonstrations performed per Specification 3.10.3.

(c) An APRM channel is inoperable if there are less than 2 LPRM inputs per level or less than (11) LPRM inputs to an APRM channel.

(d) These functions are not required to be OPERABLE when the reactor pressure vessel head is unbolted or removed per Specification 3.10.1.

(e) This function shall be automatically bypassed when the reactor mode switch is not in the Run position.

(f) This function is not required to be OPERABLE when PRIMARY CONTAINMENT INTEGRITY is not required.

(g) Also actuates the standby gas treatment system.

(h) With any control rod withdrawn. Not applicable to control rods removed per Specification 3.9.10.1 or 3.9.10.2.

(i) These functions are automatically bypassed when turbine first stage pressure is < (250) psig, equivalent to THERMAL POWER less than (30)% of RATED THERMAL POWER.

(j) Also actuates the EOC-RPT system.

*Not required for control rods removed per Specification 3.9.10.1 or 3.9.10.2.

TABLE 3.3.1-2

REACTOR PROTECTION SYSTEM RESPONSE TIMES

FUNCTIONAL UNIT	RESPONSE TIME (Seconds)
1. Intermediate Range Monitors:	
a. Neutron Flux - Upscale	NA
b. Inoperative	NA
2. Average Power Range Monitor*:	
a. Neutron Flux - Upscale, (15)%	NA
b. Flow Biased Simulated Thermal Power - Upscale	$\leq$ (0.09)**
c. Fixed Neutron Flux - Upscale, (118)%	$\leq$ (0.09)
d. Inoperative	NA
e. LPRM	NA
3. Reactor Vessel Steam Dome Pressure - High	$\leq$ (0.55)
4. Reactor Vessel Water Level - Low, Level 3	$\leq$ (1.05)
5. Main Steam Line Isolation Valve - Closure	$\leq$ (0.06)
6. Main Steam Line Radiation - High	NA
7. Primary Containment Pressure - High	NA
8. Scram Discharge Volume Water Level - High	NA
9. Turbine Stop Valve - Closure	$\leq$ (0.06)
10. Turbine Control Valve Fast Closure, Trip Oil Pressure - Low	$\leq$ (0.08)#
11. Reactor Mode Switch in Shutdown Position	NA
12. Manual Scram	NA

*Neutron detectors are exempt from response time testing. Response time shall be measured from the detector output or from the input of the first electronic component in the channel. (This provision is not applicable to Construction Permits docketed after January 1, 1978. See Regulatory Guide 1.18, November 1977.)

**Not including simulated thermal power time constant.

#Measured from start of turbine control valve fast closure.

TABLE 4.3.1.1-1 (Continued)

REACTOR PROTECTION SYSTEM INSTRUMENTATION SURVEILLANCE REQUIREMENTS

FUNCTIONAL UNIT	CHANNEL CHECK	CHANNEL FUNCTIONAL TEST	CHANNEL CALIBRATION	OPERATIONAL CONDITIONS IN WHICH SURVEILLANCE REQUIRED
8. Scram Discharge Volume Water Level - High	NA	M	R	1, 2, 5
9. Turbine Stop Valve - Closure	NA	M	R	1
10. Turbine Control Valve Fast Closure Trip Oil Pressure - Low	NA	M	Q	1
11. Reactor Mode Switch in Shutdown Position	NA	R	NA	1, 2, 3, 4, 5
12. Manual Scram	NA	M	NA	1, 2, 3, 4, 5

]

(a) Neutron detectors may be excluded from CHANNEL CALIBRATION.
(b) Within 24 hours prior to startup, if not performed within the previous 7 days.
(c) The IRM and SRM channels shall be determined to overlap for at least (   ) decades during each startup and the IRM and APRM channels shall be determined to overlap for at least (   ) decades during each controlled shutdown, if not performed within the previous 7 days.
(d) This calibration shall consist of the adjustment of the APRM channel to conform to the power values calculated by a heat balance during OPERATIONAL CONDITION 1 when THERMAL POWER $\geq$ 25% of RATED THERMAL POWER. Adjust the APRM channel if the absolute difference greater than 2%. Any APRM channel gain adjustment made in compliance with Specification 3.2.2 shall not be included in determining the absolute difference.
(e) This calibration shall consist of the adjustment of the APRM readout to conform to a calibrated flow signal.
(f) The LPRMs shall be calibrated at least once per 1000 effective full power hours (EFPH) using the TIP system.

INSTRUMENTATION

3/4.3.6 CONTROL ROD WITHDRAWAL BLOCK INSTRUMENTATION

LIMITING CONDITION FOR OPERATION

3.3.6. The control rod withdrawal block instrumentation channels shown in Table 3.3.6-1 shall be OPERABLE with their trip setpoints set consistent with the values shown in the Trip Setpoint column of Table 3.3.6-2.

APPLICABILITY: As shown in Table 3.3.6-1.

ACTION:

a. With a control rod withdrawal block instrumentation channel trip setpoint less conservative than the value shown in the Allowable Values column of Table 3.3.6-2, declare the channel inoperable until the channel is restored to OPERABLE status with its trip setpoint adjusted consistent with the Trip Setpoint value.

b. With the number of OPERABLE channels less than required by the Minimum OPERABLE Channels per Trip Function requirement, take the ACTION required by Table 3.3.6-1.

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

SURVEILLANCE REQUIREMENTS

4.3.6 Each of the above required control rod withdrawal block trip systems and instrumentation channels shall be demonstrated OPERABLE by the performance of the CHANNEL CHECK, CHANNEL FUNCTIONAL TEST and CHANNEL CALIBRATION operations for the OPERATIONAL CONDITIONS and at the frequencies shown in Table 4.3.6-1.

TABLE 3.3.6-1

CONTROL ROD WITHDRAWAL BLOCK INSTRUMENTATION

TRIP FUNCTION	MINIMUM OPERABLE CHANNELS PER TRIP FUNCTION	APPLICABLE OPERATIONAL CONDITIONS	ACTION
1. ROD BLOCK MONITOR(a)			
a. Upscale	2	1*	60
b. Inoperative	2	1*	60
c. Downscale	2	1*	60
2. APRM			
a. Flow Biased Simulated Thermal Power - Upscale	4	1	61
b. Inoperative	4	1, 2, 5	61
c. Downscale	4	1	61
d. Neutron Flux - Upscale, Startup	4	2, 5	61
3. SOURCE RANGE MONITORS			
a. Detector not full in(b)	3	2	61
	2	5	61
b. Upscale(c)	3	2	61
	2	5	61
c. Inoperative(c)	3	2	61
	2	5	61
d. Downscale(d)	3	2	61
	2	5	61
4. INTERMEDIATE RANGE MONITORS			
a. Detector not full in (e)	6	2, 5	61
b. Upscale	6	2, 5	61
c. Inoperative	6	2, 5	61
d. Downscale(e)	6	2, 5	61
5. SCRAM DISCHARGE VOLUME			
a. Water Level-High	2	1, 2, 5**	62
b. Scram Trip Bypassed	1	(1, 2, 5**)	62
6. REACTOR COOLANT SYSTEM RECIRCULATION FLOW			
a. Upscale	2	1	62
b. Inoperative	2	1	62
c. (Comparator) (Downscale)	2	1	62

TABLE 3.3.6-1 (Continued)

CONTROL ROD WITHDRAWAL BLOCK INSTRUMENTATION

ACTION

ACTION 60 - Take the ACTION required by Specification 3.1.4.3.

ACTION 61 - With the number of OPERABLE Channels:

a. One less than required by the Minimum OPERABLE Channels per Trip Function requirement, restore the inoperable channel to OPERABLE status within 7 days or place the inoperable channel in the tripped condition within the next hour.

b. Two or more less than required by the Minimum OPERABLE Channels per Trip Function requirement, place at least one inoperable channel in the tripped condition within one hour.

ACTION 62 - With the number of OPERABLE channels less than required by the Minimum OPERABLE Channels per Trip Function requirement, place the inoperable channel in the tripped condition within one hour.

NOTES

* With THERMAL POWER ≥ (20)% of RATED THERMAL POWER.

** With more than one control rod withdrawn. Not applicable to control rods removed per Specification 3.9.10.1 or 3.9.10.2.

a. The RBM shall be automatically bypassed when a peripheral control rod is selected.

b. This function shall be automatically bypassed if detector count rate is > 100 cps or the IRM channels are on range (2) or higher.

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

d. This function shall be automatically bypassed when the IRM channels are on range 3 or higher.

e. This function shall be automatically bypassed when the IRM channels are on range 1.

Franklin Research Center
A Division of The Franklin Institute

TABLE 3.3.6-2

CONTROL ROD WITHDRAWAL BLOCK INSTRUMENTATION SETPOINTS

TRIP FUNCTION	TRIP SETPOINT	ALLOWABLE VALUE
1. ROD BLOCK MONITOR		
a. Upscale	< 0.66 W + (40)%	< 0.66 W + (43)%
b. Inoperative	NA	NA
c. Downscale	≥ (5)% of RATED THERMAL POWER	≥ (3)% of RATED THERMAL POWER
2. APRM		
a. Flow Biased Simulated Thermal Power - Upscale	< 0.66 W + (42)%*	< 0.66 W + (45)%*
b. Inoperative	NA	NA
c. Downscale	≥ (5)% of RATED THERMAL POWER	≥ (3)% of RATED THERMAL POWER
d. Neutron Flux - Upscale Startup	≤ (12)% of RATED THERMAL POWER	≤ (14)% of RATED THERMAL POWER
3. SOURCE RANGE MONITORS		
a. Detector not full in	NA	NA
b. Upscale	< (2 x 10^5) cps	< (5 x 10^5) cps
c. Inoperative	NA	NA
d. Downscale	≥ (3) cps	≥ (2) cps
4. INTERMEDIATE RANGE MONITORS		
a. Detector not full in	NA	NA
b. Upscale	< (100/125) of full scale	< (110/125) of full scale
c. Inoperative	NA	NA
d. Downscale	≥ (5/125) of full scale	≥ (3/125) of full scale
5. SCRAM DISCHARGE VOLUME		
a. Water Level High	To be specified	NA
b. Scram Trip Bypassed	NA	NA
6. REACTOR COOLANT SYSTEM RECIRCULATION FLOW		
a. Upscale	< (__/__) of full scale	< (__/__) of full scale
b. Inoperative	NA	NA
c. (Comparator) (Downscale)	≤ (10)% flow deviation	≤ (__)% flow deviation

*The Average Power Range Monitor rod block function is varied as a function of recirculation loop flow (W). The trip setting of this function must be maintained in accordance with Specification 3.2.2.

TABLE 4.3.6-1

CONTROL ROD WITHDRAWAL BLOCK INSTRUMENTATION SURVEILLANCE REQUIREMENTS

TRIP FUNCTION	CHANNEL CHECK	CHANNEL FUNCTIONAL TEST	CHANNEL CALIBRATION[a]	OPERATIONAL CONDITIONS IN WHICH SURVEILLANCE REQUIRED
1. ROD BLOCK MONITOR				
a. Upscale	NA	S/U(b), M	Q	1*
b. Inoperative	NA	S/U(b), M	NA	1*
c. Downscale	NA	S/U(b), M	Q	1*
2. APRM				
a. Flow Biased Simulated Thermal Power - Upscale	NA	S/U(b), M	Q	1
b. Inoperative	NA	S/U(b), M	NA	1, 2, 5
c. Downscale	NA	S/U(b), M	Q	1
d. Neutron Flux - Upscale, Startup	NA	S/U(b), M	Q	2, 5
3. SOURCE RANGE MONITORS				
a. Detector not full in	NA	S/U(b), W(c)	NA	2, 5
b. Upscale	NA	S/U(b), W(c)	Q	2, 5
c. Inoperative	NA	S/U(b), W(c)	NA	2, 5
d. Downscale	NA	S/U(b), W(c)	Q	2, 5
4. INTERMEDIATE RANGE MONITORS				
a. Detector not full in	NA	S/U(b), W(c)	NA	2, 5
b. Upscale	NA	S/U(b), W(c)	Q	2, 5
c. Inoperative	NA	S/U(b), W(c)	NA	2, 5
d. Downscale	NA	S/U(b), W(c)	Q	2, 5
5. SCRAM DISCHARGE VOLUME				
a. Water Level-High	NA	Q	R	1, 2, 5**
b. Scram Trip Bypassed	NA	M	NA	(1, 2, 5**)
6. REACTOR COOLANT SYSTEM RECIRCULATION FLOW				
a. Upscale	NA	S/U(b), M	Q	1
b. Inoperative	NA	S/U(b), M	NA	1
c. (Comparator) (Downscale)	NA	S/U(b), M	Q	1

TABLE 4.3.6-1 (Continued)

CONTROL ROD WITHDRAWAL BLOCK INSTRUMENTATION SURVEILLANCE REQUIREMENTS

NOTES:

a. Neutron detectors may be excluded from CHANNEL CALIBRATION.

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

c. When making an unscheduled change from OPERATIONAL CONDITION 1 to OPERATIONAL CONDITION 2, perform the required surveillance within 12 hours after entering OPERATIONAL CONDITION 2.

* With THERMAL POWER $\geq$ (20)% of RATED THERMAL POWER.

** With any control rod withdrawn. Not applicable to control rods removed per Specification 3.9.10.1 or 3.9.10.2.

APPENDIX B

GEORGIA POWER COMPANY LETTER OF FEBRUARY 26, 1981

AND

SUBMITTAL WITH PROPOSED TECHNICAL SPECIFICATIONS CHANGES

FOR

EDWIN I. HATCH NUCLEAR PLANT UNITS 1 AND 2

Georgia Power

February 26, 1981

W. A. Widner
[illegible] Manager
[illegible]

U. S. Nuclear Regulatory Commission
Office of Nuclear Reactor Regulation
Washington, D. C. 20555

NRC DOCKETS 50-321, 50-366
OPERATING LICENSES DPR-57, NPF-5
EDWIN I. HATCH NUCLEAR PLANT UNITS
BWR SCRAM SYSTEM DESIGN

Gentlemen:

In accordance with the provisions of 10 CFR 50.90, as required by 10 CFR 50.59(c)(1), Georgia Power Company hereby proposes amendments to Operating Licenses NPF-5 and DPR-57. The proposed amendment would be to incorporate revised Technical Specifications in response to your July 7, 1980, letter. The proposed Technical Specifications will strengthen the provisions for assuring continued operability of the control rod drive system during reactor operation by providing surveillance requirements on the scram discharge volume vent and drain valves.

In addition, the pressure relief valves are added to the existing tables of containment isolation valves and included in the normal surveillance requirements.

The proposed changes in no way alter system design or operation, and thus do not create the possibility of a new accident or malfunction of equipment, nor does it increase the probability of previously analyzed accidents or malfunctions. Margins of safety are increased by the addition of periodic surveillance on these valves which become containment isolation valves during the time period following a scram and before the scram is reset.

The Plant Review Board and the Safety Review Board have reviewed the proposed changes to the Technical Specifications and the basis, stated above, for the proposed changes, and have concluded that they do not involve an unreviewed safety question.

Accordingly, we therefore request your review and approval of the proposed changes to the Technical Specifications as shown in the attachments.

Very truly yours,

W. A. Widner

RDB/mb
Attachments

Sworn to and subscribed before me this 26th day of February, 1981.

Notary Public
[illegible] at Large
[illegible]

xc: R. F. Rogers, III
[illegible]

8103100[illegible]

ATTACHMENT 1

NRC DOCKETS 50-321, 50-366
OPERATING LICENSES DPR-57, NPF-5
EDWIN I. HATCH NUCLEAR PLANT UNITS 1, 2
PROPOSED CHANGES TO TECHNICAL SPECIFICATIONS

Pursuant to 10 CFR 170.12 (c), Georgia Power Company has evaluated the attached proposed amendment to Operating Licenses DPR-57 and NPF-5 and has determined that:

a) The proposed amendment does not require the evaluation of a new Safety Analysis Report or rewrite of the facility license;

b) The proposed amendment does not contain several complex issues, does not involve ACRS review, and does not require an environmental impact statement;

c) The proposed amendment does not involve a complex issue, an environmental issue or more than one safety issue;

d) The proposed amendment does involve a single safety issue, namely, the addition of scram discharge volume vent valves, drain valves, and pressure relief valves to the existing tables of containment isolation valves.

e) The proposed change is therefore a Class III amendment for one unit and a Class I amendment for the other unit.

ATTACHMENT 2
NRC DOCKET 50-366
OPERATING LICENSE NPF-5
EDWIN I. HATCH NUCLEAR PLANT UNIT 2
PROPOSED CHANGE TO TECHNICAL SPECIFICATIONS

The proposed change to Technical Specifications (Appendix A to Operating License NPF-5) would be incorporated as follows:

Remove Page	Insert Page
3/4 6-23	3/4 6-23
3/4 6-32	3/4 6-32

TABLE 3.6.3-1 (Continued)

PRIMARY CONTAINMENT ISOLATION VALVES

VALVE FUNCTION AND NUMBER	VALVE GROUP[a]	ISOLATION TIME (Seconds)
A. Automatic Isolation Valves (Continued)		
25. Traversing Incore Probe Isolation Valve Ball Valves	(b)	NA
26. Vacuum Relief Isolation Valves		
2T48-F309	6	5
2T48-F324	6	5
27. Scram Discharge Volume vent valves		
2C11-F010A	(c)	60
2C11-F010B	(c)	60
28. Scram Discharge Volume drain valve		
2C11-F011	(c)	60

(a) See Specification 3.3.2, Table 3.3.2-1, for isolation signals that operate each valve group.

(b) Closes upon withdrawal of TIP. TIP automatic withdrawal is actuated by either low reactor vessel water level or high drywell pressure.

(c) Isolates on receipt of any scram signal.

TABLE 3.6.3-1 (Continued)

PRIMARY CONTAINMENT ISOLATION VALVES

VALVE FUNCTION AND NUMBER

C. OTHER ISOLATION VALVES (Continued)

25. HPCI exhaust drain isolation valves
    2E41-F022
    2E41-F040

26. RHR relief valve discharge isolation valves
    2E11-F055 A, B(i)
    RV(j)
    RV(j)
    2T49-F009 A, B

27. Core spray test line isolation valves
    2E21-F036 A, B
    2E21-F044 A, B

28. Control air to vacuum breakers isolation valve
    Solenoid valve

29. Scram discharge volume relief valve
    2C11-F012(i)

(i) Pressure relief valve.

(j) Thermal relief valve.

ATTACHMENT 3
NRC DOCKET 50-321
OPERATING LICENSE DPR-57
EDWIN I. HATCH NUCLEAR PLANT UNIT 1
PROPOSED CHANGE TO TECHNICAL SPECIFICATIONS

The proposed change to Technical Specifications (Appendix A to Operating License DPR-57 would be incorporated as follows:

Remove Page	Insert Page
3.7 13a	3.7 13a
3.7 20	3.7 20*

TABLE 3.7-1

PRIMARY CONTAINMENT ISOLATION VALVES

Isolation Group (g)	Valve Identification	Number of Power Operated Valves Inside	Number of Power Operated Valves Outside	Maximum Operating Time (sec)	Normal Position (a)	Action on Initiating Signal (a)
(f)	Scram Discharge Volume vent valves (C11-F010A, C11-F010B)		2	60	O	GC
(f)	Scram Discharge Volume drain valve (C11-F011)		1	60	O	GC

(g)	Valve Identification	Number of Valves Inside	Number of Valves Outside	(g)	Normal Position	(g)
	Scram Discharge Volume relief valve (C11-F012)		1		C	

Notes to Table 3.7-1 (Cont'd)

(f) Valves receive isolation signal on any scram.

(g) Not applicable

3.7-20

APPENDIX C

GEORGIA POWER COMPANY LETTER OF OCTOBER 1, 1981

WITH

RESPONSE TO RFI FOR

EDWIN I. HATCH NUCLEAR PLANT UNITS 1 AND 2

Georgia Power Company
333 Piedmont Avenue
Atlanta, Georgia 30308
Telephone 404 526-7020

Mailing Address:
Post Office Box 4545
Atlanta, Georgia 30302

October 1, 1981

Georgia Power
the southern electric system

J. T. Beckham, Jr.
Vice President and General Manager
Nuclear Generation

Director of Nuclear Reactor Regulation
U. S. Nuclear Regulatory Commission
Washington, D. C. 20555

NRC DOCKETS 50-321, 50-366
OPERATING LICENSES DPR-57, NPF-5
EDWIN I. HATCH NUCLEAR PLANT UNITS 1, 2
RESPONSE TO FRANKLIN RESEARCH CENTER REQUEST
CONCERNING SCRAM DISCHARGE SYSTEM TECHNICAL SPECIFICATIONS

GENTLEMEN:

Your letter dated September 1, 1981, conveyed to Georgia Power Company a request for additional information from Franklin Research Center (FRC) concerning our February 26, 1981, submittal of proposed modifications to the Technical Specifications regarding the scram discharge volume and associated instruments. The following information is supplied in response to the FRC request:

Item 1

The model Technical Specifications contained in your July 7, 1980, letter placed the scram discharge volume vent and drain valves in section 3/4.1.3.1 of the model Technical Specifications; "Control Rod Operability". Item 1 of the FRC request asked for a reference to the section of the Technical Specifications where the requested change is incorporated.

Our February 26, 1981, letter proposed that these valves be placed in the tables of power operated containment isolation valves instead of the "Control Rod Operability" section. These valves do not affect control rod operability at Plant Hatch. The plant unique geometry of this system at Plant Hatch allows free communication between the scram level switches and the scram discharge volume (SDV). Thus, the level switches, not the vent and drain valves, protect the scram function, and in a sense control rod operability, by providing assurance that the SDV is empty. The vent and drain valves are important, however, insofar as they provide a containment pressure boundary during the time that a scram is sealed-in. For this reason we have chosen to place the valves in the tables of containment isolation valves. The surveillance requirements are therefore different than those proposed by the model Technical Specifications in order to be consistent with the requirements for other comparable containment isolation valves.

A001 s/0

Georgia Power

Director of Nuclear Reactor Regulation
U. S. Nuclear Regulatory Commission
October 1, 1981
Page Two

Items 2 and 3

As indicated in our October 10, 1980, letter the scram level switches are currently covered by Technical Specifications on each unit. For Unit 1, please refer to Specifications 3.1 and 4.1, Tables 3.1-1 and 4.1-1, item 7. For Unit 2, the appropriate reference is Specification 3/4.3.1, tables 3.3.1-1 and 4.3.1-1, item 8. The instrument functional test frequency for Unit 1 is once every three months as initially approved by the Commission on issuance of the Unit 1 Operating License. We have not proposed to modify this specification.

Items 4, 5 and 6

The SDV rod block setpoint and surveillance requirements are specified in Unit 2 Technical Specification Section 3.3.5 and Tables 3.3.5-1, 3.3.5-2 and 4.3.5-1. In reviewing the Technical Specifications for our February 26, 1981 submittal, the absence of a comparable specification in the Unit 1 Technical Specifications was not noted. We agree that it is appropriate to specify the limits and surveillance requirements for the SDV rod block alarm switch and will propose an amendment to the Unit 1 license to incorporate requirements similar to those contained in our Unit 2 Specifications referenced above.

Very truly yours,

W. A. Widner

J. T. Beckham, Jr.

RDB/mb

xc: M. Manry
R. F. Rogers, III

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

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

SUPPORTING AMENDMENT NO. 34 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

Introduction

As a result of events involving common cause failures of Scram Discharge Volume (SDV) limit switches and SDV drain valve operability, the NRC staff issued IE Bulletin 80-14 on June 12, 1980. In addition, the NRC staff sent a letter dated July 7, 1980, to all operating BWR licensees requesting that they propose Technical Specification (TS) changes to provide surveillance requirements for SDV vent and drain valves and Limiting Conditions for Operation (LCO)/surveillance requirements on SDV limit switches. Model TSs were enclosed with this letter to provide guidance to licensees for preparation of the requested submittals.

Evaluation

The enclosed report (TER-C5506-78) was prepared for us by Franklin Research Center (FRC) as part of a technical assistance contract program. Their report provides their Technical Evaluation of the compliance of Georgia Power Company's (the licensee) submittal with NRC provided criteria.

FRC has concluded that the licensee's response does not meet the explicit requirements of paragraph 3.3-6 and Table 3.3.6-1 of the NRC staff's Model TSs. However, the FRC report concludes that technical bases are defined on page 50 of the staff's "Generic Safety Evaluation Report BWR Scram Discharge System," December 1, 1980, for this departure from the explicit requirements of the Model TSs. We conclude that these technical bases justify a deviation from the explicit requirements of the Model TSs.

The licensee proposed in its February 26, 1981, submittal to list the SDV vent and drain valves as containment isolation valves and to perform only the surveillance required for containment isolation valves. The FRC report notes that it found this proposed surveillance to be unacceptable. It also notes that in discussions on this subject with FRC, the licensee orally agreed to revise its proposed surveillance requirements to meet the NRC staff Model TS requirements for surveillance of these SDV vent and drain valves.

Thus, FRC has concluded that the licensee's proposed TSs (as revised by subsequent discussion with the licensee) meet our criteria.

Subsequently, by Order dated June 24, 1983, the Commission required that the licensee 1) install the long term BWR scram discharge modifications in conformance with the staff's December 1, 1980 Generic SER on Scram Discharge Systems before December 31, 1983 and 2) submit TS changes required for operation with the modified system at least 3 months prior to the required implementation date.

By letter dated September 19, 1983, the licensee submitted proposed TS revisions in accordance with its previous oral commitment, as discussed above. We have reviewed these proposed revisions and find that they are consistent with this previous commitment that provided the bases for acceptance in the FRC report and conclude that they are acceptable. The licensee also proposed to add an LCO for the operability of SDV vent and drain valves. This LCO would require the plant to be placed in Hot Shutdown in 12 hours if any SDV vent or drain valve is inoperable. We find this proposed LCO is consistent with the NRC staff Model TSs and conclude that it is acceptable.

By letter dated October 3, 1983, the licensee has withdrawn its February 26, 1981, submittal that requested that the SDV vent and drain valves be listed in containment isolation valve tables and be required to meet only containment isolation valve surveillance requirements. These surveillance requirements are now provided by the licensee's proposed TSs submitted in its September 19, 1983, letter as discussed above.

In its September 19, 1983, submittal, the licensee has also proposed to add TS requirements, including trip setpoint, LCO, Action Statement and surveillance requirements, for the new diverse SDV high water level scram instrumentation (Thermal Level Sensors) to the Reactor Protection System instrumentation tables. The new instruments have been given the same requirements as the original level switches which were found acceptable in the FRC report. We conclude that this proposed addition is acceptable.

By letter dated December 14, 1983, the licensee proposed to change the 30 second closure time requirement for the SDV vent and drain valves as proposed in its September 19, 1983 submittal to 60 seconds for the inboard vent and drain valves and 120 seconds for the outboard vent and drain valves. This proposal deviates from the acceptability guidelines of 30 seconds closure time provided by the staff in its Generic Safety Evaluation Report on Scram Discharge Systems dated December 1, 1980. The staff is currently reviewing the licensee's justification for this latest proposed change. In the interim, the staff has concluded that there is reasonable assurance of safe operation of the plants based on implementation of the short-term corrective measures noted in the June 24, 1983 Order; and the long-term corrective measures noted herein.

Based upon our review of the contractor's report and discussions with the reviewer and on our review, as discussed above, of the licensee's subsequent submittals augmenting the information reviewed by the contractor, we conclude that except for the SDV vent and drain valve closure requirements as discussed above, the licensee's proposed TSs satisfy our requirements for surveillance of SDV vent and drain valves and for LCOs and surveillance requirements for SDV limit switches. Consequently, we find the licensee's proposed TSs, except for the SDV vent and drain valve closure times, acceptable.

Environmental Consideration

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

Conclusion

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

Dated: January 4, 1984

The following NRC personnel contributed to this Safety Evaluation:
Ken Eccleston and George Rivenbark

TECHNICAL EVALUATION REPORT

BWR SCRAM DISCHARGE VOLUME LONG-TERM MODIFICATIONS

GEORGIA POWER COMPANY

EDWIN I. HATCH NUCLEAR PLANT UNIT 2

NRC DOCKET NO. 50-366	FRC PROJECT C5506
NRC TAC NO. 42218	FRC ASSIGNMENT 2
NRC CONTRACT NO. NRC-03-81-130	FRC TASK 78

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FOREWORD

This Technical Evaluation Report was prepared by Franklin Research Center under a contract with the U.S. Nuclear Regulatory Commission (Office of Nuclear Reactor Regulation, Division of Operating Reactors) for technical assistance in support of NRC operating reactor licensing actions. The technical evaluation was conducted in accordance with criteria established by the NRC.

SUMMARY

This technical evaluation report reviews and evaluates proposed Phase 1 changes in the Hatch Nuclear Plant Unit 2 Technical Specifications for scram discharge volume (SDV) long-term modifications regarding surveillance requirements for SDV vent and drain valves and the limiting condition for operation (LCO)/surveillance requirements for reactor protection system and control rod withdrawal block SDV limit switches. Conclusions were based on the degree of compliance of the Licensee's submittal with criteria from the Nuclear Regulatory Commission (NRC) staff's Model Technical Specifications.

The proposed placement of the SDV drain and vent valves in the tables of power-operated isolation valves (see revised pages 3/4 6-23 and 3/4 6-32 of the Hatch Nuclear Plant Unit 2 Technical Specifications) in order to apply isolation valve surveillance requirements to them is not acceptable. However, the Licensee's agreement to revise proposed specifications changes to require verifying each valve to be open at least once per 31 days and cycling each valve at least one complete cycle of full travel at least once per 92 days meets the NRC staff's Model Technical Specifications requirements of paragraphs 4.1.3.1a and 4.1.3.1b, and is acceptable.

The remaining surveillance requirements are met by pages 3/4 1-5, 3/4 1-6, 3/4 1-7, 3/4 3-4, 3/4 3-7, 3/4 3-38, and 3/4 3-41 of the Hatch Nuclear Plant Unit 2 Technical Specifications without any revision. Table 5-1 on pages 21 and 22 of this report summarizes the evaluation results.

1. INTRODUCTION

1.1 PURPOSE OF THE TECHNICAL EVALUATION

The purpose of this technical evaluation report (TER) is to review and evaluate the proposed changes in the Technical Specifications of the Hatch Nuclear Plant Unit 2 boiling water reactor (BWR) in regard to "BWR Scram Discharge Volume Long Term Modification," specifically:

- o surveillance requirements for scram discharge volume (SDV) vent and drain valves
- o limiting condition for operation (LCO)/surveillance requirements for the reactor protection system limit switches
- o LCO/surveillance requirements for the control rod withdrawal block SDV limit switches.

The evaluation used criteria proposed by the NRC staff in Model Technical Specifications (see Appendix A of this report). This effort is directed toward the NRC objective of increasing the reliability of installed BWR scram discharge volume systems, the need for which was made apparent by events described below.

1.2 GENERIC ISSUE BACKGROUND

On June 13, 1979, while the reactor at Hatch Unit 1 was in the refuel mode, two SDV high level switches had been modified, tested, and found inoperable. The remaining switches were operable. Inspection of each inoperable level switch revealed a bent float rod binding against the side of the float chamber.

On October 19, 1979, Brunswick Unit 1 reported that water hammer due to slow closure of the SDV drain valve during a reactor scram damaged several pipe supports on the SDV drain line. Drain valve closure time was approximately 5 minutes because of a faulty solenoid controlling the air supply to the valve. After repair, to avoid probable damage from a scram, the unit was started with the SDV vent and drain valves closed except for periodic draining. During this mode of operation, the reactor scrammed due to a high water level in the

SDV system without prior actuation of either the high level alarm or rod block switch. Inspection revealed that the float ball on the rod block switch was bent, making the switches inoperable. The water hammer was reported to be the cause of these level switch failures.

As a result of these events involving common-cause failures of SDV limit switches and SDV drain valve operability, the NRC issued IE Bulletin 80-14, "Degradation of BWR Scram Discharge Volume Capability," on June 12, 1980 [1]. In addition, to strengthen the provisions of this bulletin and to ensure that the scram system would continue to work during reactor operation, the NRC sent a letter dated July 7, 1980 [2] to all operating BWR licensees requesting that they propose Technical Specifications changes to provide surveillance requirements for reactor protection system and control rod block SDV limit switches. The letter also contained the NRC staff's Model Technical Specifications to be used as a guide by licensees in preparing their submittals.

Meanwhile, during a routine shutdown of the Browns Ferry Unit 3 reactor on June 28, 1980, 76 of 185 control rods failed to insert fully. Full insertion required two additional manual scrams and an automatic scram for a total elapsed time of approximately 15 minutes between the first scram initiation and the complete insertion of all the rods. On July 3, 1980, in response to both this event and the previous events at Hatch Unit 1 and Brunswick Unit 1, the NRC issued (in addition to the earlier IE Bulletin 80-14) IE Bulletin 80-17 followed by five supplements. These initiated short-term and long-term programs described in "Generic Safety Evaluation Report BWR Scram Discharge System," NRC Staff, December 1, 1980 [9] and "Staff Report and Evaluation of Supplement 4 to IE Bulletin 80-17 (Continuous Monitoring Systems)" [10].

Analysis and evaluation of the Browns Ferry Unit 3 and other SDV system events convinced the NRC staff that SDV systems in all BWRs should be modified to assure long-term SDV reliability. Improvements were needed in three major areas: SDV-IV hydraulic coupling, level instrumentation, and system isolation. To achieve these objectives, an Office of Nuclear Reactor Regulation (NRR) task force and a subgroup of the BWR Owners Group developed Revised Scram Discharge System Design and Safety Criteria for use in establishing acceptable SDV systems modifications [9]. Also, an NRC letter dated October 1, 1980 requested all operating BWR licensees to reevaluate installed SDV systems and modify them as necessary to comply with the revised criteria.

In Reference 9, the SDV-IV hydraulic coupling at the Big Rock Point, Brunswick Units 1 and 2, Duane Arnold, and Hatch Units 1 and 2 BWRs was judged acceptable. The remaining BWRs will require modification to meet the revised SDV-IV hydraulic coupling criteria, and all operating BWRs may require modification to meet the revised instrumentation and isolation criteria. The changes in Technical Specifications associated with this effort will be carried out in two phases:

Phase 1 - Improvements in surveillance for vent and drain valves and instrument volume level switches.

Phase 2 - Improvements required as a result of long-term modifications made to comply with revised design and performance criteria.

This TER is a review and evaluation of Technical Specifications changes proposed for Phase 1.

1.3 PLANT-SPECIFIC BACKGROUND

The July 7, 1980 NRC letter [2] not only requested all BWR licensees to amend their facilities' Technical Specifications with respect to control rod drive SDV capability, but enclosed the NRC staff's proposed Model Technical Specifications (see Appendix A of this TER) as a guide for the licensees in preparing the requested submittals and as a source of criteria for an FRC technical evaluation of the submittals. In this TER, FRC has reviewed and evaluated Technical Specifications changes for the Hatch Nuclear Plant Unit 2 as proposed in letters dated February 26, 1981 and October 1, 1981 (see Appendices B and C, respectively) by the Licensee, the Georgia Power Company (GPC), in regard to "BWR Scram Discharge Volume (SDV) Long-Term Modifications" and, specifically, in regard to the surveillance requirements for SDV vent and drain valves and the limiting condition for operation (LCO)/ surveillance requirements for the reactor protection system and control rod withdrawal block SDV limit switches. FRC assessed the adequacy with which the GPC information documented compliance of the proposed Technical Specifications changes with the NRC staff's Model Technical Specifications.

2. REVIEW CRITERIA

The criteria established by the NRC staff's Model Technical Specifications involving surveillance requirements of the main SDV components and instrumentation cover three areas of concern:

- o surveillance requirements for SDV vent and drain valves
- o LCO/surveillance requirements for reactor protection system SDV limit switches
- o LCO/surveillance requirements for control rod block SDV limit switches.

2.1 SURVEILLANCE REQUIREMENTS FOR SDV DRAIN AND VENT VALVES

The surveillance criteria of the NRC staff's Model Technical Specification for SDV drain and vent valves are:

> "4.1.3.1.1 - The scram discharge volume drain and vent valves shall be demonstrated OPERABLE by:
>
> a. Verifying each valve to be open* at least once per 31 days, and
> b. Cycling each valve at least one complete cycle of full travel at least once per 92 days.
>
> *These valves may be closed intermittently for testing under administrative controls."

The Model Technical Specifications require testing the drain and vent valves, checking at least once every 31 days that each valve is fully open during normal operation, and cycling each valve at least one complete cycle of full travel under administrative controls at least once per 92 days.

Full opening of each valve during normal operation indicates that there is no degradation in the control air system and its components that control the air pressure to the pneumatic actuators of the drain and vent valves. Cycling each valve checks whether the valve opens fully and whether its movement is smooth, jerky, or oscillatory.

During normal operation, the drain and vent valves stay in the open position for very long periods. A silt of particulates such as metal chips and flakes, various fibers, lint, sand, and weld slag from the water or air may accumulate at moving parts of the valves and temporarily "freeze" them. A strong breakout force may be needed to overcome this temporary "freeze," producing a violent jerk which may induce a severe water hammer if it occurs during a scram or a scram resetting. Periodic cycling of the drain and vent valves is the best method to clear the effects of particulate silting, thus promoting smooth opening and closing and more reliable valve operation. Also, in case of improper valve operation, cycling can indicate whether excessive pressure transients may be generated during and after a reactor scram which might damage the SDV piping system and cause a loss of system integrity or function.

2.2 LCO/SURVEILLANCE REQUIREMENTS FOR REACTOR PROTECTION SYSTEM SDV LIMIT SWITCHES

The paragraphs of the NRC staff's Model Technical Specifications pertinent to LCO/surveillance requirements for reactor protection system SDV limit switches are:

> "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.

Table 3.3.1-1. Reactor Protection System Instrumentation

Functional Unit	Applicable Operational Conditions	Minimum Operable Channels Per Trip System (a)	Action
8. Scram Discharge Volume Water Level-High	1,2,5 (h)	2	4

Table 3.3.1-2. Reactor Protection System Response Times

Functional Unit	Response Time (Seconds)
8. Scram Discharge Volume Water Level-High	NA"

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

Table 4.3.1.1-1. Reactor Protection System Instrumentation Surveillance Requirements

Functional Unit	Channel Check	Channel Functional Test	Channel Calibration	Operational Conditions in Which Surveillance Required
8. Scram Discharge Volume Water Level-High	NA	M	R	1,2,5

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

(h) With any control rod withdrawn. Not applicable to control rods removed per Specification 3.9.10.1 or 3.9.10.2

Action 4: In OPERATIONAL CONDITION 1 or 2, be in at least HOT SHUTDOWN within 6 hours.

In OPERATIONAL CONDITION 5, suspend all operations involving CORE ALTERATIONS* and fully insert all insertable control rods within one hour.

*Except movement of IRM, SRM or special movable detectors, or replacement of LPRM strings provided SRM instrumentation is OPERABLE per Specification 3.9.2."

Paragraph 3.3.1 and Table 3.3.1-1 of the Model Technical Specifications require the functional unit of SDV water level-high to have at least 2 operable channels containing 2 limit switches per trip system, a total of 4 operable channels containing 4 limit switches per 2 trip systems for the reactor protection system which automatically initiates a scram. The technical objective of these requirements is to provide 1-out-of-2-taken-twice logic for the reactor protection system. The response time of the reactor protection system for the functional unit of SDV water level-high should be measured and kept available (it is not given in Table 3.3.1-2).

Paragraph 4.3.1.1 and Table 4.3.1.1-1 give reactor protection system instrumentation surveillance requirements for the functional unit of SDV water level-high. Each reactor protection system instrumentation channel containing a limit switch should be shown to be operable by the Channel Functional Test monthly and Channel Calibration at each refueling outage.

2.3 LCO/SURVEILLANCE REQUIREMENTS FOR CONTROL ROD WITHDRAWAL BLOCK SDV LIMIT SWITCHES

The NRC staff's Model Technical Specifications specify the following LCO/ surveillance requirements for control rod withdrawal block SDV limit switches:

"3.3.6 - The control rod withdrawal block instrumentation channel shown in Table 3.3.6-1 shall be OPERABLE with trip setpoints set consistent with the values shown in the Trip Setpoint column of Table 3.3.6-2.

Table 3.3.6-1. Control Rod Withdrawal Block Instrumentation

Trip Function	Minimum Operable Channels Per Trip Function	Applicable Operational Conditions	Action
5. Scram Discharge Volume			
a. Water level-high	2	1, 2, 5**	62
b. Scram trip bypassed	1	(1, 2, 5**)	62

ACTION 62: With the number of OPERABLE channels less than required by the minimum OPERABLE channels per Trip Function requirement, place the inoperable channel in the tripped condition within one hour.

**With more than one control rod withdrawn. Not applicable to control rods removed per Specification 3.9.10.1 or 3.9.10.2.

Table 3.3.6-2 Control Rod Withdrawal Block Instrumentation Setpoints

Trip Function	Trip Setpoint	Allowable Value
5. Scram Discharge Volume		
a. Water level-high	To be specified	NA
b. Scram trip bypassed	NA	NA"

"4.3.6 - Each of the above control rod withdrawal block trip systems and instrumentation channels shall be demonstrated OPERABLE by the performance of the CHANNEL CHECK, CHANNEL FUNCTIONAL TEST and CHANNEL CALIBRATION operations for the OPERATIONAL CONDITIONS and at the frequencies shown in Table 4.3.6-1.

Table 4.3.6-1. Control Rod Withdrawal Block Instrumentation Surveillance Requirements

Trip Function	Channel Check	Channel Functional Test	Channel Calibration	Operational Conditions in Which Surveillance Required
5. Scram Discharge Volume				
a. Water Level-High	NA	Q	R	1, 2, 5**
b. Scram Trip Bypassed	NA	M	NA	(1, 2, 5**)

**With more than one control rod withdrawn. Not applicable to control rods removed per Specification 3.9.10.1 or 3.9.10.2."

Paragraph 3.3.6 and Table 3.3.6-1 of the Model Technical Specifications require the control rod withdrawal block instrumentation to have at least 2 operable channels containing 2 limit switches for SDV water level-high and 1 operable channel containing 1 limit switch for SDV scram trip bypassed. The technical objective of these requirements is to have at least one channel containing one limit switch available to monitor the SDV water level when the other channel with a limit switch is being tested or undergoing maintenance. The trip setpoint for control rod withdrawal block instrumentation monitoring

SDV water level-high should be specified as indicated in Table 3.3.6-2. The trip function prevents further withdrawal of any control rod when the control rod block SDV limit switches indicate water level-high.

Paragraph 4.3.6 and Table 4.3.6-1 require that each control rod withdrawal block instrumentation channel containing a limit switch be shown to be operable by the Channel Functional Test once per 3 months for SDV water level-high, by the Channel Functional Test once per month for SDV scram trip bypassed, and by Channel Calibration at each refueling outage for SDV water level-high.

The Surveillance Criteria of the BWR Owners Subgroup given in Appendix A, "Long-Term Evaluation of Scram Discharge System," of "Generic Safety Evaluation Report BWR Scram Discharge System," written by the NRC staff and issued on December 1, 1980, are:

1. Vent and drain valves shall be periodically tested.
2. Verifying and level detection instrumentation shall be periodically tested in place.
3. The operability of the entire system as an integrated whole shall be demonstrated periodically and during each operating cycle, by demonstrating scram instrument response and valve function at pressure and temperature at approximately 50% control rod density.

Analysis of the above criteria indicates that the NRC staff's Model Technical Specifications requirements, the acceptance criteria for the present TER, fully cover the BWR Owners Subgroup Surveillance Criteria 1 and 2 and partially cover Criterion 3.

3. METHOD OF EVALUATION

The GPC submittal for the Hatch Nuclear Plant Unit 2 was evaluated in two stages, initial and final.

During the initial evaluation, only the NRC staff's Model Technical Specifications requirements were used to determine if:

- o the Licensee's submittal was responsive to the July 7, 1980 NRC request for proposed Technical Specifications changes involving the surveillance requirements of the SDV vent and drain valves, LCO/surveillance requirements for reactor protection system SDV limit switches, and LCO/surveillance requiremenmts for control rod block SDV limit switches

- o the submitted information was sufficient to permit a detailed technical evaluation.

During the final evaluation, in addition to the NRC staff's Model Technical Specifications requirements, background material in References 1 through 10, pertinent sections of "Georgia Power Company Hatch Nuclear Plant Unit 2 Safety Analysis Report" and Hatch Unit 2 Technical Specifications were studied to determine the technical bases for the design of the SDV main components and instrumentation. Subsequently, the Licensee's response was compared directly to the requirements of the NRC staff's Model Technical Specifications. The findings of the final evaluation are presented in Section 4 of this report.

The initial evaluation concluded that the Licensee's submittal was responsive to the NRC request of July 7, 1980, but certain information was not available. A request for additional information (RFI) was sent to GPC by the NRC on September 1, 1981. Thus, this TER is based on the initial submittal and the Licensee's response to the RFI, dated October 1, 1981 (see Appendix C).

4. TECHNICAL EVALUATION

4.1 SURVEILLANCE REQUIREMENTS FOR SDV DRAIN AND VENT VALVES

NRC STAFF'S MODEL TECHNICAL SPECIFICATIONS

Paragraph 4.1.3.1.1 requires demonstrating that the SDV drain and vent valves are operable by:

a. verifying each valve to be open at least once per 31 days (valves may be closed intermittently for testing under administrative controls)

b. cycling each valve at least one complete cycle of full travel at least once per 92 days.

LICENSEE RESPONSE

The Licensee proposed to revise the Hatch Nuclear Plant Unit 2 Technical Specifications pages 3/4 6-23 and 3/4 6-32 which contain Table 3.6.3-1 (Continued) with information given below.

"Table 3.6.3-1 (Continued)
Primary Containment Isolation Valves

Valve Function and Number	Valve Group	Isolation Time (Seconds)
A. Automatic Isolation Valves		
27. Scram Discharge Volume Vent Valves		
2C11-F010A	(c)	60
2C11-F010B	(c)	60
28. Scram Discharge Volume Drain Valve		
2C11-F011	(c)	60
C. Other Isolation Valves		
29. Scram Discharge Volume Relief Valve 2C11-F012 (i)"		

Notes: "(c) Isolates on receipt of any scram signal
(i) Pressure relief valve"

In response to the RFI, the Licensee provided the following statement:

"Item 1

The model Technical Specifications contained in your July 7, 1980, letter placed the scram discharge volume vent and drain valves in section 3/4.1.3.1 of the model Technical Specifications; 'Control Rod Operability.' Item 1 of the FRC request asked for a reference to the section of the Technical Specifications where the request change is incorporated.

Our February 26, 1981, letter proposed that these valves be placed in the tables of power operated containment isolation valves instead of the 'Control Rod Operability' section. These valves do not affect control rod operability at Plant Hatch. The plant unique geometry of this system at Plant Hatch allows free communication between the scram level switches and the scram discharge volume (SDV). Thus, the level switches, not the vent and drain valves, protect the scram function, and in a sense control rod operability, by providing assurance that the SDV is empty. The vent and drain valves are important, however, insofar as they provide a containment pressure boundary during the time that a scram is sealed-in. For this reason we have chosen to place the valves in the tables of containment isolation valves. The surveillance requirements are therefore different than those proposed by the model Technical Specifications in order to be consistent with the requirements for other comparable containment isolation valves."

The Licensee agreed to revise proposed specifications changes to require:

a. verifying each valve to be open at least once per 31 days (valves may be closed intermittently for testing under administrative controls), and

b. cycling each valve at least one complete cycle of full travel at least once per 92 days.

FRC EVALUATION

The proposed placement of the SDV drain and vent valves in the tables of power-operated isolation valves (see revised pages 3/4 6-23 and 3/4 6-32 of the Hatch Nuclear Plant Unit 2 Technical Specifications) in order to apply isolation valve surveillance requirements to them is not acceptable. However, the Licensee's agreement to revise proposed specifications changes to require verifying each valve to be open at least once per 31 days and cycling each valve at least one complete cycle of full travel at least once per 92 days meets the NRC staff's Model Technical Specifications requirements of paragraphs 4.1.3.1.1a and 4.1.3.1.1b, and is acceptable.

4.2 LCO/SURVEILLANCE REQUIREMENTS FOR REACTOR PROTECTION SYSTEM SDV LIMIT SWITCHES

NRC STAFF'S MODEL TECHNICAL SPECIFICATIONS

Paragraph 3.3.1 and Table 3.3.1-1 require the functional unit of SDV water level-high to have at least 2 operable channels containing 2 limit switches per trip system, a total of 4 operable channels containing 4 limit switches per 2 trip systems for the reactor protection system which automatically initiates scram.

Paragraph 3.3.1 and Table 3.3.1-2 concern the response time of the reactor protection system for the functional unit of SDV water level-high which should be specified for each BWR (it is not specified in the table). Paragraph 4.3.1.1 and Table 4.3.1.1-1 require that each reactor protection system instrumentation channel containing a limit switch be shown to be operable for the functional unit of SDV water level-high by the Channel Functional Test monthly and Channel Calibration at each refueling outage. The applicable operational conditions for these requirements are Startup, Run, and Refuel.

LICENSEE RESPONSE

The Licensee provided the following information in answer to the RFI:

"As indicated in our October 10, 1980, letter the scram level switches are currently covered by Technical Specifications on each unit. For Unit 1, please refer to Specifications 3.1 and 4.1, Tables 3.1-1 and 4.1-1, item 7. For Unit 2, the appropriate reference is Specification 3/4.3.1, tables 3.3.1-1 and 4.3.1-1, item 8. The instrument functional test frequency for Unit 1 is once every three months as initially approved by the Commission on issuance of the Unit 1 Operating License. We have not proposed to modify this specification."

Page 3/4 3-3 of the Hatch Nuclear Plant Unit 2 Technical Specifications addresses the NRC staff's Model Technical Specifications requirements of paragraph 3.3.1 and Table 3.3.1-1, giving the following information in Table 3.3.1-1 (continued), Reactor Protection System Instrumentation, for "Functional Unit 8 Scram Discharge Volume Water Level-High":

"1. Applicable Operational Conditions: 1, 2, 5(h)

2. Minimum Number Operable Channels Per Trip System (a): 2

3. Action: 4

Action 4 - In operational condition 1 or 2, be in at least HOT SHUTDOWN within 6 hours.

TABLE NOTATIONS:

a. 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 OPERABLE channel in the same trip system is monitoring that parameter.

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

The requirements of the NRC staff's Model Technical Specifications of paragraph 3.3.1 and Table 3.3.1-2 are covered in the Hatch Nuclear Plant Unit 2 Technical Specifications, Sections 3/4.1.3.2, 3/4.1.3.3 and 3/4.1.3.4 which specify control rod maximum scram insertion times, control rod average scram insertion times, and four control rod group scram insertion times, respectively.

Page 3/4 3-7 of the Hatch Nuclear Plant Unit 2 Technical Specifications addresses the NRC staff's Model Technical Specifications requirements of paragraph 4.3.1.1 and Table 4.3.1.1-1, providing Table 4.3.1-1, Reactor Protection System Instrumentation Surveillance Requirements, with the following information for "Functional Unit 8 Scram Discharge Volume Water Level-High":

"1. Channel Check: NA

2. Channel Functional Test: M (monthly)

3. Channel Calibration (a): R(h) (each refueling)

4. Operational Conditions in Which Surveillance Required: 1, 2, 5"

Notes (from page 3/4 3-8):

"a. Neutron detectors may be excluded from channel calibration
h. Physical inspection and actuation of switches."

FRC EVALUATION

The Licensee's response to the NRC staff's Model Technical Specifications requirements of paragraph 3.3.1 and Table 3.3.1-1 is acceptable. The Hatch Nuclear Plant Unit 2 reactor protection system SDV water level-high instrumentation consists of 2 operable channels containing 2 limit switches per trip system, for a total of 4 operable channels containing 4 limit switches per 2 trip systems, making 1-out-of-2-taken-twice logic.

Although the Hatch Nuclear Plant Unit 2 Technical Specifications do not specify directly the reactor protection system SDV water level-high response time as required in the NRC staff's Model Technical Specifications, paragraph 3.3.1 and Table 3.3.1-2, they have requirements for scram time tests, which include the required response time (see Sections 3/4.1.3.2, 3/4.1.3.3, and 3/4.1.3.4). This approach is acceptable, since the reactor protection system SDV water level-high response time can be deduced from the scram time test.

The original provisions of the Hatch Nuclear Plant Unit 2 Technical Specifications given in page 3/4 3-7, Table 4.3.1-1, in regard to reactor protection system SDV water level-high Channel Functional Test and Channel Calibration are acceptable. They meet the NRC staff's Model Technical Specifications requirements of paragraph 4.3.1.1 and Table 4.3.1.1-1, which require the Channel Functional Test monthly and Channel Calibration each refueling outage.

4.3 LCO/SURVEILLANCE REQUIREMENTS FOR CONTROL ROD WITHDRAWAL BLOCK SDV LIMIT SWITCHES

NRC STAFF'S MODEL TECHNICAL SPECIFICATIONS

Paragraph 3.3.6 and Table 3.3.6-1 require the control rod withdrawal block instrumentation to have at least 2 operable channels containing 2 limit switches for SDV water level-high, and 1 operable channel containing 1 limit switch for SDV trip bypassed. Paragraph 3.3.6 also requires specifying the trip setpoint for control rod withdrawal block instrumentation monitoring SDV water level-high as indicated in Table 3.3.6-2.

Paragraph 4.3.6 and Table 4.3.6-1 require that each control rod withdrawal block instrumentation channel containing a limit switch be shown to be operable by the Channel Functional Test once per 3 months for SDV water level-high, by the Channel Functional Test once per month for SDV scram trip bypassed, and by Channel Calibration at each refueling outage for SDV water level-high.

LICENSEE RESPONSE

The Licensee responded as follows:

"The SDV rod block setpoint and surveillance requirements are specified in Unit 2 Technical Specification Section 3.3.5 and Tables 3.3.5-1, 3.3.5-2 and 4.3.5-1. In reviewing the Technical Specifications for our February 26, 1981 submittal, the absence of a comparable specification in the Unit 1 Technical Specifications was not noted. We agree that it is appropriate to specify the limits and surveillance requirements for the SDV rod block alarm switch and will propose an amendent to the Unit 1 license to incorporate requirements similar to those contained in our Unit 2 Specifications referenced above."

The information provided in Table 3.3.5-1, Control Rod Withdrawal Block Instrumentation, is as follows for "Trip Function 5. Scram Discharge Volume a. Water Level-High":

"1. Minimum Number of Operable Channels per Trip Function: 1

2. Applicable Operational Conditions: 1, 2, 5 (f)"

Note

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

Table 3.3.5-2, Control Rod Withdrawal Block Instrumentation Setpoints, contains the following information for "Trip Function 5. Scram Discharge volume a. Water Level-High":

"1. Trip Setpoint: $\leq$ 36.2 gallons

2. Allowable Value: $\leq$ 36.2 gallons"

The contents of Table 3.3.5-1 and 3.3.5-2 address the NRC staff's Model Technical Specifications requirements of paragraph 3.3.6 and Table 3.3.6-1. Table 4.3.5-1, Control Rod Withdrawal Block Instrumentation Surveillance

Requirements, addresses the NRC staff's Model Technical Specifications requirements of paragraph 4.3.6 and Table 4.3.6-1, providing the following information for "Trip Function 5. Scram Discharge Volume a. Water Level-High":

"1. Channel Check: NA

2. Channel Functional Test: Q (quarterly)

3. Channel Calibration (a): R (each refueling)

4. Operational Conditions in Which Surveillance Required: 1, 2, 5(e)"

NOTES:

"a. Neutron detectors may be excluded from CHANNEL CALIBRATION.
e. With any control rod withdrawn. Not applicable to control rods removed per Specification 3.9.11.1 or 3.9.11.2."

FRC EVALUATION

The existing Hatch Nuclear Plant Unit 2 discharge system has six level switches on the scram discharge volume (see FSAR, page 4.2-48) set at three different water levels to guard against operation of the reactor without sufficient free volume present in the scram discharge headers to receive the scram discharge water in the event of a scram. At the first (lowest) level, one level switch initiates an alarm for operator action. At the second level, with the setpoint of $\leq$ 36.2 gallons (see page 3/43-40, Table 3.3.5-2 of the Hatch Unit 2 Technical Specifications), one level switch initiates a rod withdrawal block to prevent further withdrawal of any control rod. At the third (highest) level, with the setpoint of 50+6 or 50-1 gallons (see the Hatch Nuclear Plant Unit 2 FSAR, Table 7.2-1), the four level switches (two for each reactor protection system trip system) initiate a scram to shut down the reactor while sufficient free volume is available to receive the scram discharge water. Reference 9, page 50, defines Design Criterion 9 ("Instrumentation shall be provided to aid the operator in the detection of water accumulation in the instrumented volume(s) prior to scram initiation"), gives the technical basis for "Long-Term Evaluation of Scram Discharge System," and defines acceptable compliance ("The present alarm and rod block instrumentation meets this criterion given adequate hydraulic coupling with the SDV headers"). The Hatch Nuclear Plant Unit 2 has adequate hydraulic coupling between scram discharge headers and instrumented volume. Thus, the present alarm and rod block instrumentation is also acceptable.

In the Hatch Nuclear Plant Unit 2, "Scram Discharge Volume Scram Trips" cannot be bypassed while the reactor is in operational conditions of startup and run (see FSAR page 7.2-10) and operational condition "refuel with more than one control rod withdrawn" is not applicable, since interlocks are provided which prevent the withdrawal of more than one control rod with the mode switch in the refuel position. Thus, the NRC staff's Model Technical Specifications requirements of paragraph 3.3.6 with Table 4.3.6-1 and paragraph 4.3.6 with Table 4.3.6-1 are not applicable to the Hatch Nuclear Plant Unit 2 for "Trip Function 5.b, SDV Scram Trip Bypassed."

The proposed trip setpoint of $\leq$ 36.2 gallons for control rod withdrawal block instrumentation channel is acceptable. The provision of Table 4.3.5-1 for control rod withdrawal block instrumentation surveillance requirements of the Hatch Nuclear Plant Unit 2 Technical Specifications meets the NRC staff's Model Technical Specifications requirements of paragraph 4.3.6 and Table 4.3.6-1. It prescribes the Channel Functional Test of each control rod withdrawal block instrumentation channel containing a limit switch quarterly and channel calibration each refueling for SDV water level-high.

5. CONCLUSIONS

Table 5-1 summarizes the results of the final review and evaluation of the Hatch Nuclear Plant Unit 2 proposed Phase 1 Technical Specifications changes for SDV long-term modification in regard to surveillance requirements for SDV vent and drain valves and LCO/surveillance requirements for reactor protection system and control rod block SDV limit switches. The following conclusions were made:

- o The proposed placement of the SDV drain and vent valves in the tables of power-operated isolation valves (see revised pages 3/4 6-23 and 3/4 6-32 of the Hatch Nuclear Plant Unit 2 Technical Specifications) in order to apply isolation valve surveillance requirements to them is not acceptable. However, the Licensee's agreement to revise proposed specifications changes to require verifying each valve to be open at least once per 31 days and cycling each valve at least one complete cycle of full travel at least once per 92 days meets the NRC staff's Model Technical Specifications requirements of paragraphs 4.1.3.1.1a and 4.1.3.1.1b, and is acceptable.

- o The remaining surveillance requirements are met by pages 3/4 1-5, 3/4 1-6, 3/4 1-7, 3/4 3-4, 3/4 3-7, 3/4 3-38, and 3/4 3-41 of the Hatch Nuclear Plant Unit 2 Technical Specifications without any revision.

Table 5-1. Evaluation of Phase 1 Proposed Technical Specifications Changes for Scram Discharge Volume Long-Term Modifications Hatch Nuclear Plant Unit 2

	Technical Specifications		
Surveillance Requirements	NRC Staff Model (Paragraph)	Proposed by Licensee	Evaluation
SDV DRAIN AND VENT VALVES			
Verify each valve open	Once per 31 days (4.1.3.1.1a)	Once per 31 days (p. 3/4 6-16*)	Acceptable
Cycle each valve one complete cycle	Once per 92 days (4.1.3.1.1b)	Once per 92 days (p. 3/4 6-16*)	Acceptable
REACTOR PROTECTION SYSTEM SDV LIMIT SWITCHES			
Minimum operable channels per trip system	2 (3.3.1, Table 3.3.1-1)	2 (p. 3/4 3-3, Table 3.3.1-1)	Acceptable
SDV water level-high response time	NA (3.3.1, Table 3.3.1-2)	(p. 3/4 1-5, 3/4 1-6, and 3/4 1-7)	Acceptable
SDV water level-high			
Channel functional test	Monthly (4.3.1.1, Table 4.3.1.1-1)	Monthly (p. 3/4 3-7, Table 4.3.1-1)	Acceptable
Channel calibration	Each refueling (4.3.1.1, Table 4.3.1.1-1)	Each refueling (p. 3/4 3-7, Table 4.3.1-1)	Acceptable

*The Licensee agreed to revise page 3/4 6-16.

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Table 5-1 (Cont.)

Surveillance Requirements	Technical Specifications: NRC Staff Model (Paragraph)	Technical Specifications: Proposed by Licensee	Evaluation
CONTROL ROD BLOCK SDV LIMIT SWITCHES			
Minimum operable channels per trip function			
SDV water level-high	2 (3.3.6, Table 3.3.6-1)	1 (p. 3/4 3-38, Table 3.3.5-1)	Acceptable*
SDV scram trip bypassed	1 (3.3.6, Table 3.3.6-1)	NA (p. 3/4 3-38, Table 3.3.5-1)	Acceptable*
SDV water level-high		3-40	
Trip setpoint	NA (3.3.6, Table 3.3.6-2)	$\leq$ 36.2 gallons (p. 3/4 3-4[illegible], Table 3.3.5-2)	Acceptable
Channel functional test	Quarterly (4.3.6, Table 4.3.6-1)	Quarterly (p. 3/4 3-41, Table 4.3.5-1)	Acceptable
Channel calibration	Each refueling (4.3.6, Table 4.3.6-1)	Each refueling (p. 3/4 3-41, Table 4.3.5-1)	Acceptable
SDV scram trip bypassed			
Channel functional test	Monthly (4.3.6, Table 4.3.6-1)	NA	Acceptable*

* See Reference 9, p. 50, and pp. 18 and 19 of this TER.

6. REFERENCES

1. IE Bulletin 80-14, "Degradation of BWR Scram Discharge Volume Capacity"
   NRC, Office of Inspection and Enforcement, June 12, 1980

2. D. G. Eisenhut (NRR), letter "To All Operating Boiling Water Reactors (BWRs)" with enclosure, "Model Technical Specifications"
   July 7, 1980

3. IE Bulletin 80-17, "Failure of 76 of 185 Control Rods to Fully Insert During a Scram at a BWR"
   NRC, Office of Inspection and Enforcement, July 3, 1980

4. IE Bulletin 80-17, Supplement 1, "Failure of 76 of 185 Control Rods to Fully Insert During a Scram at a BWR"
   NRC, Office of Inspection and Enforcement, July 18, 1980

5. IE Bulletin 80-17, Supplement 2, "Failures Revealed by Testing Subsequent to Failure of Control Rods to Insert During a Scram at a BWR"
   NRC, Office of Inspection and Enforcement, July 22, 1980

6. IE Bulletin 80-17, Supplement 3, "Failure of Control Rods to Insert During a Scram at a BWR"
   NRC, Office of Inspection and Enforcement, August 22, 1980

7. IE Bulletin 80-17, Supplement 4, "Failure of Control Rods to Insert During a Scram at a BWR"
   NRC, Office of Inspection and Enforcement, December 18, 1980

8. IE Bulletin 80-17, Supplement 5, "Failure of Control Rods to Insert During a Scram at a BWR"
   NRC, Office of Inspection and Enforcement, February 13, 1981

9. P. S. Check (NRR), memorandum with enclosure, "Generic Safety Evaluation Report BWR Scram Discharge System"
   December 1, 1980

10. P. S. Check (NRR), memorandum with enclosure, "Staff Report and Evaluation of Supplement 4 to IE Bulletin 80-17"
    June 10, 1981

APPENDIX A

NRC STAFF'S MODEL TECHNICAL SPECIFICATIONS*

* Note: Applicable changes are marked by vertical lines in the margins.

REACTIVITY CONTROL SYSTEMS

LIMITING CONDITION FOR OPERATION (Continued)

ACTION (Continued)

2. If the inoperable control rod(s) is inserted, within one hour disarm the associated directional control valves either:

   a) Electrically, or

   b) Hydraulically by closing the drive water and exhaust water isolation valves.

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

c. With more than 8 control rods inoperable, be in at least HOT SHUTDOWN within 12 hours.

SURVEILLANCE REQUIREMENTS

4.1.3.1.1 The scram discharge volume drain and vent valves shall be demonstrated OPERABLE by:

a. Verifying each valve to be open* at least once per 31 days and

b. Cycling each valve through at least one complete cycle of full travel at least once per 92 days.

4.1.3.1.2 When above the preset power level of the RWM and RSCS, all withdrawn control rods not required to have their directional control valves disarmed electrically or hydraulically shall be demonstrated OPERABLE by moving each control rod at least one notch:

a. At least once per 7 days, and

b. At least once per 24 hours when any control rod is immovable as a result of excessive friction or mechanical interference.

4.1.3.1.3 All control rods shall be demonstrated OPERABLE by performance of Surveillance Requirements 4.1.3.2, 4.1.3.4, 4.1.3.5, 4.1.3.6 and 4.1.3.7.

*These valves may be closed intermittently for testing under administrative controls.

REACTIVITY CONTROL SYSTEMS

CONTROL ROD MAXIMUM SCRAM INSERTION TIMES

LIMITING CONDITION FOR OPERATION

3.1.3.2 The maximum scram insertion time of each control rod from the fully withdrawn position to notch position (6), based on de-energization of the scram pilot valve solenoids as time zero, shall not exceed (7.0) seconds.

APPLICABILITY: OPERATIONAL CONDITIONS 1 and 2.

ACTION:

With the maximum scram insertion time of one or more control rods exceeding (7.0) seconds:

a. Declare the control rod(s) with the slow insertion time inoperable, and

b. Perform the Surveillance Requirements of Specification 4.1.3.2.c at least once per 60 days when operation is continued with three or more control rods with maximum scram insertion times in excess of (7.0) seconds, or

c. Be in at least HOT SHUTDOWN within 12 hours.

SURVEILLANCE REQUIREMENTS

4.1.3.2 The maximum scram insertion time of the control rods shall be demonstrated through measurement with reactor coolant pressure greater than or equal to 950 psig and, during single control rod scram time tests, the control rod drive pumps isolated from the accumulators:

a. For all control rods prior to THERMAL POWER exceeding 40% of RATED THERMAL POWER following CORE ALTERATIONS or after a reactor shutdown that is greater than 120 days,

b. For specifically affected individual control rods following maintenance on or modification to the control rod or control rod drive system which could affect the scram insertion time of those specific control rods, and

c. For 10% of the control rods, on a rotating basis, at least once per 120 days of operation.

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.

APPLICABILITY: As shown in Table 3.3.1-1.

ACTION:

a. With the number of OPERABLE channels less than required by the Minimum OPERABLE Channels per Trip System requirement for one trip system, place at least one inoperable channel in the tripped condition within one hour.

b. With the number of OPERABLE channels less than required by the Minimum OPERABLE Channels per Trip System requirement 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 FUNCTIONAL TEST and CHANNEL CALIBRATION operations for the OPERATIONAL CONDITIONS and at the frequencies shown in Table 4.3.1.1-1.

4.3.1.2 LOGIC SYSTEM FUNCTIONAL TESTS and simulated automatic operation of all channels shall be performed at least once per 18 months.

4.3.1.3 The REACTOR PROTECTION SYSTEM RESPONSE TIME of each reactor trip function shown in 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.

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

TABLE 3.3.1-1 (Continued)

REACTOR PROTECTION SYSTEM INSTRUMENTATION

FUNCTIONAL UNIT	APPLICABLE OPERATIONAL CONDITIONS	MINIMUM OPERABLE CHANNELS PER TRIP SYSTEM (a)	ACTION
8. Scram Discharge Volume Water Level - High	1, 2, 5(h)	2	4
9. Turbine Stop Valve - Closure	1(i)	4(j)	7
10. Turbine Control Valve Fast Closure, Trip Oil Pressure - Low	1(i)	2(j)	7
11. Reactor Mode Switch in Shutdown Position	1, 2, 3, 4, 5	1	8
12. Manual Scram	1, 2, 3, 4, 5	1	9

TABLE 3.3.1-1 (Continued)

REACTOR PROTECTION SYSTEM INSTRUMENTATION

ACTION

ACTION 1 - In OPERATIONAL CONDITION 2, be in at least HOT SHUTDOWN within 6 hours.

In OPERATIONAL CONDITION 5, suspend all operations involving CORE ALTERATIONS* and fully insert all insertable control rods within one hour.

ACTION 2 - Lock the reactor mode switch in the Shutdown position within one hour.

ACTION 3 - Be in at least STARTUP within 2 hours.

ACTION 4 - In OPERATIONAL CONDITION 1 or 2, be in at least HOT SHUTDOWN within 6 hours.

In OPERATIONAL CONDITION 5, suspend all operations involving CORE ALTERATIONS* and fully insert all insertable control rods within one hour.

ACTION 5 - Be in at least HOT SHUTDOWN within 6 hours.

ACTION 6 - Be in STARTUP with the main steam line isolation valves closed within 2 hours or in at least HOT SHUTDOWN within 6 hours.

ACTION 7 - Initiate a reduction in THERMAL POWER within 15 minutes and reduce turbine first stage pressure to < (250) psig, equivalent to THERMAL POWER less than (30)% of RATED THERMAL POWER, within 2 hours.

ACTION 8 - In OPERATIONAL CONDITION 1 or 2, be in at least HOT SHUTDOWN within 6 hours.

In OPERATIONAL CONDITION 3 or 4, verify all insertable control rods to be fully inserted within one hour.

In OPERATIONAL CONDITION 5, suspend all operations involving CORE ALTERATIONS* and fully insert all insertable control rods within one hour.

ACTION 9 In OPERATIONAL CONDITION 1 or 2, be in at least HOT SHUTDOWN within 6 hours.

In OPERATIONAL CONDITION 3 or 4, lock the reactor mode switch in the Shutdown position within one hour.

In OPERATIONAL CONDITION 5, suspend all operations involving CORE ALTERATIONS* and fully insert all insertable control rods within one hour.

*Except movement of IRM, SRM or special movable detectors, or replacement of LPRM strings provided SRM instrumentation is OPERABLE per Specification 3.9.2.

TABLE 3.3.1-1 (Continued)

REACTOR PROTECTION SYSTEM INSTRUMENTATION

TABLE NOTATIONS

(a) 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 OPERABLE channel in the same trip system is monitoring that parameter.

b) The "shorting links" shall be removed from the RPS circuitry prior to and during the time any control rod is withdrawn* and shutdown margin demonstrations performed per Specification 3.10.3.

(c) An APRM channel is inoperable if there are less than 2 LPRM inputs per level or less than (11) LPRM inputs to an APRM channel.

(d) These functions are not required to be OPERABLE when the reactor pressure vessel head is unbolted or removed per Specification 3.10.1.

(e) This function shall be automatically bypassed when the reactor mode switch is not in the Run position.

(f) This function is not required to be OPERABLE when PRIMARY CONTAINMENT INTEGRITY is not required.

(g) Also actuates the standby gas treatment system.

(h) With any control rod withdrawn. Not applicable to control rods removed per Specification 3.9.10.1 or 3.9.10.2.

(i) These functions are automatically bypassed when turbine first stage pressure is $\leq$ (250) psig, equivalent to THERMAL POWER less than (30)% of RATED THERMAL POWER.

(j) Also actuates the EOC-RPT system.

*Not required for control rods removed per Specification 3.9.10.1 or 3.9.10.2.

TABLE 3.3.1-2

REACTOR PROTECTION SYSTEM RESPONSE TIMES

FUNCTIONAL UNIT	RESPONSE TIME (Seconds)
1. Intermediate Range Monitors:	
a. Neutron Flux - Upscale	NA
b. Inoperative	NA
2. Average Power Range Monitor*:	
a. Neutron Flux - Upscale, (15)%	NA
b. Flow Biased Simulated Thermal Power - Upscale	≤ (0.09)**
c. Fixed Neutron Flux - Upscale, (118)%	≤ (0.09)
d. Inoperative	NA
e. LPRM	NA
3. Reactor Vessel Steam Dome Pressure - High	≤ (0.55)
4. Reactor Vessel Water Level - Low, Level 3	≤ (1.05)
5. Main Steam Line Isolation Valve - Closure	≤ (0.06)
6. Main Steam Line Radiation - High	NA
7. Primary Containment Pressure - High	NA
8. Scram Discharge Volume Water Level - High	NA
9. Turbine Stop Valve - Closure	≤ (0.06)
10. Turbine Control Valve Fast Closure, Trip Oil Pressure - Low	≤ (0.08)#
11. Reactor Mode Switch in Shutdown Position	NA
12. Manual Scram	NA

*Neutron detectors are exempt from response time testing. Response time shall be measured from the detector output or from the input of the first electronic component in the channel. (This provision is not applicable to Construction Permits docketed after January 1, 1978. See Regulatory Guide 1.18, November 1977.)
**Not including simulated thermal power time constant.
#Measured from start of turbine control valve fast closure.

TABLE 4.3.1.1-1 (Continued)

REACTOR PROTECTION SYSTEM INSTRUMENTATION SURVEILLANCE REQUIREMENTS

FUNCTIONAL UNIT	CHANNEL CHECK	CHANNEL FUNCTIONAL TEST	CHANNEL CALIBRATION	OPERATIONAL CONDITIONS IN WHICH SURVEILLANCE REQUIRED
8. Scram Discharge Volume Water Level - High	NA	M	R	1, 2, 5
9. Turbine Stop Valve - Closure	NA	M	R	1
10. Turbine Control Valve Fast Closure Trip Oil Pressure - Low	NA	M	Q	1
11. Reactor Mode Switch in Shutdown Position	NA	R	NA	1, 2, 3, 4, 5
12. Manual Scram	NA	M	NA	1, 2, 3, 4, 5

(a) Neutron detectors may be excluded from CHANNEL CALIBRATION.
(b) Within 24 hours prior to startup, if not performed within the previous 7 days.
(c) The IRM and SRM channels shall be determined to overlap for at least ( ) decades during each startup and the IRM and APRM channels shall be determined to overlap for at least ( ) decades during each controlled shutdown, if not performed within the previous 7 days.
(d) This calibration shall consist of the adjustment of the APRM channel to conform to the power values calculated by a heat balance during OPERATIONAL CONDITION 1 when THERMAL POWER ≥ 25% of RATED THERMAL POWER. Adjust the APRM channel if the absolute difference greater than 2%. Any APRM channel gain adjustment made in compliance with Specification 3.2.2 shall not be included in determining the absolute difference.
(e) This calibration shall consist of the adjustment of the APRM readout to conform to a calibrated flow signal.
(f) The LPRMs shall be calibrated at least once per 1000 effective full power hours (EFPH) using the TIP system.

INSTRUMENTATION

3/4.3.6 CONTROL ROD WITHDRAWAL BLOCK INSTRUMENTATION

LIMITING CONDITION FOR OPERATION

3.3.6. The control rod withdrawal block instrumentation channels shown in Table 3.3.6-1 shall be OPERABLE with their trip setpoints set consistent with the values shown in the Trip Setpoint column of Table 3.3.6-2.

APPLICABILITY: As shown in Table 3.3.6-1.

ACTION:

a. With a control rod withdrawal block instrumentation channel trip setpoint less conservative than the value shown in the Allowable Values column of Table 3.3.6-2, declare the channel inoperable until the channel is restored to OPERABLE status with its trip setpoint adjusted consistent with the Trip Setpoint value.

b. With the number of OPERABLE channels less than required by the Minimum OPERABLE Channels per Trip Function requirement, take the ACTION required by Table 3.3.6-1.

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

SURVEILLANCE REQUIREMENTS

4.3.6 Each of the above required control rod withdrawal block trip systems and instrumentation channels shall be demonstrated OPERABLE by the performance of the CHANNEL CHECK, CHANNEL FUNCTIONAL TEST and CHANNEL CALIBRATION operations for the OPERATIONAL CONDITIONS and at the frequencies shown in Table 4.3.6-1.

TABLE 3.3.6-1

CONTROL ROD WITHDRAWAL BLOCK INSTRUMENTATION

TRIP FUNCTION	MINIMUM OPERABLE CHANNELS PER TRIP FUNCTION	APPLICABLE OPERATIONAL CONDITIONS	ACTION
1. ROD BLOCK MONITOR(a)			
a. Upscale	2	1*	60
b. Inoperative	2	1*	60
c. Downscale	2	1*	60
2. APRM			
a. Flow Biased Simulated Thermal Power - Upscale	4	1	61
b. Inoperative	4	1, 2, 5	61
c. Downscale	4	1	61
d. Neutron Flux - Upscale, Startup	4	2, 5	61
3. SOURCE RANGE MONITORS			
a. Detector not full in(b)	3	2	61
	2	5	61
b. Upscale(c)	3	2	61
	2	5	61
c. Inoperative(c)	3	2	61
	2	5	61
d. Downscale(d)	3	2	61
	2	5	61
4. INTERMEDIATE RANGE MONITORS			
a. Detector not full in (e)	6	2, 5	61
b. Upscale	6	2, 5	61
c. Inoperative(e)	6	2, 5	61
d. Downscale	6	2, 5	61
5. SCRAM DISCHARGE VOLUME			
a. Water Level-High	2	1, 2, 5**	62
b. Scram Trip Bypassed	1	(1, 2, 5**)	62
6. REACTOR COOLANT SYSTEM RECIRCULATION FLOW			
a. Upscale	2	1	62
b. Inoperative	2	1	62
c. (Comparator) (Downscale)	2	1	62

TABLE 3.3.6-1 (Continued)

CONTROL ROD WITHDRAWAL BLOCK INSTRUMENTATION

ACTION

ACTION 60 - Take the ACTION required by Specification 3.1.4.3.

ACTION 61 - With the number of OPERABLE Channels:

a. One less than required by the Minimum OPERABLE Channels per Trip Function requirement, restore the inoperable channel to OPERABLE status within 7 days or place the inoperable channel in the tripped condition within the next hour.

b. Two or more less than required by the Minimum OPERABLE Channels per Trip Function requirement, place at least one inoperable channel in the tripped condition within one hour.

ACTION 62 - With the number of OPERABLE channels less than required by the Minimum OPERABLE Channels per Trip Function requirement, place the inoperable channel in the tripped condition within one hour.

NOTES

* With THERMAL POWER ≥ (20)% of RATED THERMAL POWER.

** With more than one control rod withdrawn. Not applicable to control rods removed per Specification 3.9.10.1 or 3.9.10.2.

a. The RBM shall be automatically bypassed when a peripheral control rod is selected.

b. This function shall be automatically bypassed if detector count rate is > 100 cps or the IRM channels are on range (2) or higher.

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

d. This function shall be automatically bypassed when the IRM channels are on range 3 or higher.

e. This function shall be automatically bypassed when the IRM channels are on range 1.

TABLE 3.3.6-2

CONTROL ROD WITHDRAWAL BLOCK INSTRUMENTATION SETPOINTS

TRIP FUNCTION	TRIP SETPOINT	ALLOWABLE VALUE
1. ROD BLOCK MONITOR*		
a. Upscale	≤ 0.66 W + (40)%	≤ 0.66 W + (43)%
b. Inoperative	NA	NA
c. Downscale	≥ (5)% of RATED THERMAL POWER	≥ (3)% of RATED THERMAL POWER
2. APRM		
a. Flow Biased Simulated Thermal Power - Upscale	≤ 0.66 W + (42)%*	≤ 0.66 W + (45)%*
b. Inoperative	NA	NA
c. Downscale	≥ (5)% of RATED THERMAL POWER	≥ (3)% of RATED THERMAL POWER
d. Neutron Flux - Upscale Startup	≤ (12)% of RATED THERMAL POWER	≤ (14)% of RATED THERMAL POWER
3. SOURCE RANGE MONITORS		
a. Detector not full in	NA	NA
b. Upscale	≤ (2×10^5) cps	≤ (5×10^5) cps
c. Inoperative	NA	NA
d. Downscale	≥ (3) cps	≥ (2) cps
4. INTERMEDIATE RANGE MONITORS		
a. Detector not full in	NA	NA
b. Upscale	≤ (100/125) of full scale	≤ (110/125) of full scale
c. Inoperative	NA	NA
d. Downscale	≥ (5/125) of full scale	≥ (3/125) of full scale
5. SCRAM DISCHARGE VOLUME		
a. Water Level High	To be specified	NA
b. Scram Trip Bypassed	NA	NA
6. REACTOR COOLANT SYSTEM RECIRCULATION FLOW		
a. Upscale	≤ (__/__) of full scale	≤ (__/__) of full scale
b. Inoperative	NA	NA
c. (Comparator) (Downscale)	≤ (10)% flow deviation	≤ (__)% flow deviation

*The Average Power Range Monitor rod block function is varied as a function of recirculation loop flow (W). The trip setting of this function must be maintained in accordance with Specification 3.2.2.

TABLE 4.3.6-1

CONTROL ROD WITHDRAWAL BLOCK INSTRUMENTATION SURVEILLANCE REQUIREMENTS

TRIP FUNCTION	CHANNEL CHECK	CHANNEL FUNCTIONAL TEST	CHANNEL CALIBRATION(a)	OPERATIONAL CONDITIONS IN WHICH SURVEILLANCE REQUIRED
1. ROD BLOCK MONITOR				
a. Upscale	NA	S/U(b),M	Q	1*
b. Inoperative	NA	S/U(b),M	NA	1*
c. Downscale	NA	S/U(b),M	Q	1*
2. APRM				
a. Flow Biased Simulated Thermal Power - Upscale	NA	S/U(b),M	Q	1
b. Inoperative	NA	S/U(b),M	NA	1, 2, 5
c. Downscale	NA	S/U(b),M	Q	1
d. Neutron Flux - Upscale, Startup	NA	S/U(b),M	Q	2, 5
3. SOURCE RANGE MONITORS				
a. Detector not full in	NA	S/U(b),W(c)	NA	2, 5
b. Upscale	NA	S/U(b),W(c)	Q	2, 5
c. Inoperative	NA	S/U(b),W(c)	NA	2, 5
d. Downscale	NA	S/U(b),W(c)	Q	2, 5
4. INTERMEDIATE RANGE MONITORS				
a. Detector not full in	NA	S/U(b),W(c)	NA	2, 5
b. Upscale	NA	S/U(b),W(c)	Q	2, 5
c. Inoperative	NA	S/U(b),W(c)	NA	2, 5
d. Downscale	NA	S/U(b),W(c)	Q	2, 5
5. SCRAM DISCHARGE VOLUME				
a. Water Level-High	NA	Q	R	1, 2, 5**
b. Scram Trip Bypassed	NA	M	NA	(1, 2, 5**)
6. REACTOR COOLANT SYSTEM RECIRCULATION FLOW				
a. Upscale	NA	S/U(b),M	Q	1
b. Inoperative	NA	S/U(b),M	NA	1
c. (Comparator) (Downscale)	NA	S/U(b),M	Q	1

3/4 3-54

TABLE 4.3.6-1 (Continued)

CONTROL ROD WITHDRAWAL BLOCK INSTRUMENTATION SURVEILLANCE REQUIREMENTS

NOTES:

a. Neutron detectors may be excluded from CHANNEL CALIBRATION.

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

c. When making an unscheduled change from OPERATIONAL CONDITION 1 to OPERATIONAL CONDITION 2, perform the required surveillance within 12 hours after entering OPERATIONAL CONDITION 2.

* With THERMAL POWER ≥ (20)% of RATED THERMAL POWER.

** With any control rod withdrawn. Not applicable to control rods removed per Specification 3.9.10.1 or 3.9.10.2.

APPENDIX B

GEORGIA POWER COMPANY LETTER OF FEBRUARY 26, 1981

AND

SUBMITTAL WITH PROPOSED TECHNICAL SPECIFICATIONS CHANGES

FOR

EDWIN I. HATCH NUCLEAR PLANT UNITS 1 AND 2

Georgia Power

W. A. Widner
Vice President and General Manager
Nuclear Generation

February 26, 1981

U. S. Nuclear Regulatory Commission
Office of Nuclear Reactor Regulation
Washington, D. C. 20555

NRC DOCKETS 50-321, 50-366
OPERATING LICENSES DPR-57, NPF-5
EDWIN I. HATCH NUCLEAR PLANT UNITS 1, 2
BWR SCRAM SYSTEM DESIGN

Gentlemen:

In accordance with the provisions of 10 CFR 50.90, as required by 10 CFR 50.59(c)(1), Georgia Power Company hereby proposes amendments to Operating Licenses NPF-5 and DPR-57. The proposed amendment would be to incorporate revised Technical Specifications in response to your July 7, 1980, letter. The proposed Technical Specifications will strengthen the provisions for assuring continued operability of the control rod drive system during reactor operation by providing surveillance requirements on the scram discharge volume vent and drain valves.

In addition, the pressure relief valves are added to the existing tables of containment isolation valves and included in the normal surveillance requirements.

The proposed changes in no way alter system design or operation, and thus do not create the possibility of a new accident or malfunction of equipment, nor does it increase the probability of previously analyzed accidents or malfunctions. Margins of safety are increased by the addition of periodic surveillance on these valves which become containment isolation valves during the time period following a scram and before the scram is reset.

The Plant Review Board and the Safety Review Board have reviewed the proposed changes to the Technical Specifications and the basis, stated above, for the proposed changes, and have concluded that they do not involve an unreviewed safety question.

Accordingly, we therefore request your review and approval of the proposed changes to the Technical Specifications as shown in the attachments.

Very truly yours,

W. A. Widner

RDB/mb
Attachments

Sworn to and subscribed before me this 26th day of February, 1981.

Notary Public

xc: R. F. Rogers, III

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Franklin Research Center
A Division of The Franklin Institute

ATTACHMENT 1

NRC DOCKETS 50-321, 50-366
OPERATING LICENSES DPR-57, NPF-5
EDWIN I. HATCH NUCLEAR PLANT UNITS 1, 2
PROPOSED CHANGES TO TECHNICAL SPECIFICATIONS

Pursuant to 10 CFR 170.12 (c), Georgia Power Company has evaluated the attached proposed amendment to Operating Licenses DPR-57 and NPF-5 and has determined that:

a) The proposed amendment does not require the evaluation of a new Safety Analysis Report or rewrite of the facility license;

b) The proposed amendment does not contain several complex issues, does not involve ACRS review, and does not require an environmental impact statement;

c) The proposed amendment does not involve a complex issue, an environmental issue or more than one safety issue;

d) The proposed amendment does involve a single safety issue, namely, the addition of scram discharge volume vent valves, drain valves, and pressure relief valves to the existing tables of containment isolation valves.

e) The proposed change is therefore a Class III amendment for one unit and a Class I amendment for the other unit.

ATTACHMENT 2
NRC DOCKET 50-366
OPERATING LICENSE NPF-5
EDWIN I. HATCH NUCLEAR PLANT UNIT 2
<u>PROPOSED CHANGE TO TECHNICAL SPECIFICATIONS</u>

The proposed change to Technical Specifications (Appendix A to Operating License NPF-5) would be incorporated as follows:

Remove Page	Insert Page
3/4 6-23	3/4 6-23
3/4 6-32	3/4 6-32

TABLE 3.6.3-1 (Continued)

PRIMARY CONTAINMENT ISOLATION VALVES

VALVE FUNCTION AND NUMBER	VALVE GROUP[(a)]	ISOLATION TIME (Seconds)
A. Automatic Isolation Valves (Continued)		
25. Traversing Incore Probe Isolation Valve Ball Valves	(b)	NA
26. Vacuum Relief Isolation Valves		
2T48-F309	6	5
2T48-F324	6	5
27. Scram Discharge Volume vent valves		
2C11-F010A	(c)	60
2C11-F010B	(c)	60
28. Scram Discharge Volume drain valve		
2C11-F011	(c)	60

(a) See Specification 3.3.2, Table 3.3.2-1, for isolation signals that operate each valve group.

(b) Closes upon withdrawal of TIP. TIP automatic withdrawal is actuated by either low reactor vessel water level or high drywell pressure.

(c) Isolates on receipt of any scram signal.

TABLE 3.6.3-1 (Continued)

PRIMARY CONTAINMENT ISOLATION VALVES

VALVE FUNCTION AND NUMBER

C. OTHER ISOLATION VALVES (Continued)

25. HPCI exhaust drain isolation valves
    2E41-F022
    2E41-F040

26. RHR relief valve discharge isolation valves
    2E11-F055 A, B[i]
    RV[j]
    RV[j]
    2T49-F009 A, B

27. Core spray test line isolation valves
    2E21-F036 A, B
    2E21-F044 A, B

28. Control air to vacuum breakers isolation valve
    Solenoid valve

29. Scram discharge volume relief valve
    2C11-F012[i]

---

[i] Pressure relief valve.

[j] Thermal relief valve.

ATTACHMENT 3
NRC DOCKET 50-321
OPERATING LICENSE DPR-57
EDWIN I. HATCH NUCLEAR PLANT UNIT 1
PROPOSED CHANGE TO TECHNICAL SPECIFICATIONS

The proposed change to Technical Specifications (Appendix A to Operating License DPR-57 would be incorporated as follows:

Remove Page	Insert Page
3.7 18a	3.7 18a
3.7 20	3.7 20

TABLE 3.7-1

PRIMARY CONTAINMENT ISOLATION VALVES

Isolation Group (g)	Valve Identification	Number of Power Operated Valves Inside	Number of Power Operated Valves Outside	Maximum Operating Time (sec)	Normal Position (a)	Action on Initiating Signal (a)
(f)	Scram Discharge Volume vent valves (C11-F010A, C11-F010B)		2	60	O	GC
(f)	Scram Discharge Volume drain valve (C11-F011)		1	60	O	GC

(g)	Valve Identification	Number of Valves Inside	Number of Valves Outside	(g)	Normal Position	(g)
	Scram Discharge Volume relief valve (C11-F012)		1		C	

Notes to Table 3.7-1 (Concluded)

(f) Valves receive isolation signal on any scram.

(g) Not applicable

APPENDIX C

GEORGIA POWER COMPANY LETTER OF OCTOBER 1, 1981

WITH

ANSWER TO RFI

FOR

EDWIN I. HATCH NUCLEAR PLANT UNITS 1 AND 2

Georgia Power Company
333 Piedmont Avenue
Atlanta, Georgia 30308
Telephone 404 526-7020

Mailing Address:
Post Office Box 4545
Atlanta, Georgia 30302

October 1, 1981

Georgia Power
the southern electric system

J. T. Beckham, Jr.
Vice President and General Manager
Nuclear Generation

RECEIVED
OCT 6 1981
U.S. NUCLEAR REGULATORY COMMISSION

Director of Nuclear Reactor Regulation
U. S. Nuclear Regulatory Commission
Washington, D. C. 20555

NRC DOCKETS 50-321, 50-366
OPERATING LICENSES DPR-57, NPF-5
EDWIN I. HATCH NUCLEAR PLANT UNITS 1, 2
RESPONSE TO FRANKLIN RESEARCH CENTER REQUEST
CONCERNING SCRAM DISCHARGE SYSTEM TECHNICAL SPECIFICATIONS

GENTLEMEN:

Your letter dated September 1, 1981, conveyed to Georgia Power Company a request for additional information from Franklin Research Center (FRC) concerning our February 26, 1981, submittal of proposed modifications to the Technical Specifications regarding the scram discharge volume and associated instruments. The following information is supplied in response to the FRC request:

Item 1

The model Technical Specifications contained in your July 7, 1980, letter placed the scram discharge volume vent and drain valves in section 3/4.1.3.1 of the model Technical Specifications; "Control Rod Operability". Item 1 of the FRC request asked for a reference to the section of the Technical Specifications where the requested change is incorporated.

Our February 26, 1981, letter proposed that these valves be placed in the tables of power operated containment isolation valves instead of the "Control Rod Operability" section. These valves do not affect control rod operability at Plant Hatch. The plant unique geometry of this system at Plant Hatch allows free communication between the scram level switches and the scram discharge volume (SDV). Thus, the level switches, not the vent and drain valves, protect the scram function, and in a sense control rod operability, by providing assurance that the SDV is empty. The vent and drain valves are important, however, insofar as they provide a containment pressure boundary during the time that a scram is sealed-in. For this reason we have chosen to place the valves in the tables of containment isolation valves. The surveillance requirements are therefore different than those proposed by the model Technical Specifications in order to be consistent with the requirements for other comparable containment isolation valves.

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Georgia Power

Director of Nuclear Reactor Regulation
U. S. Nuclear Regulatory Commission
October 1, 1981
Page Two

Items 2 and 3

As indicated in our October 10, 1980, letter the scram level switches are currently covered by Technical Specifications on each unit. For Unit 1, please refer to Specifications 3.1 and 4.1, Tables 3.1-1 and 4.1-1, item 7. For Unit 2, the appropriate reference is Specification 3/4.3.1, tables 3.3.1-1 and 4.3.1-1, item 8. The instrument functional test frequency for Unit 1 is once every three months as initially approved by the Commission on issuance of the Unit 1 Operating License. We have not proposed to modify this specification.

Items 4, 5 and 6

The SDV rod block setpoint and surveillance requirements are specified in Unit 2 Technical Specification Section 3.3.5 and Tables 3.3.5-1, 3.3.5-2 and 4.3.5-1. In reviewing the Technical Specifications for our February 26, 1981 submittal, the absence of a comparable specification in the Unit 1 Technical Specifications was not noted. We agree that it is appropriate to specify the limits and surveillance requirements for the SDV rod block alarm switch and will propose an amendment to the Unit 1 license to incorporate requirements similar to those contained in our Unit 2 Specifications referenced above.

Very truly yours,

W. A. Widner

J. T. Beckham, Jr.

RDB/mb

xc: M. Manry
R. F. Rogers, III