December 30, 1987

Docket No.: 50-416

Mr. Oliver D. Kingsley, Jr.
Vice President, Nuclear Operations
System Energy Resources, Inc.
Post Office Box 23054
Jackson, Mississippi 39205

Dear Mr. Kingsley:

SUBJECT: ISSUANCE OF AMENDMENT NO. 41 TO FACILITY OPERATING LICENSE NO. NPF-29 - GRAND GULF NUCLEAR STATION, UNIT 1, REGARDING CHANGES TO TECHNICAL SPECIFICATIONS FOR ANTICIPATED TRANSIENT WITHOUT SCRAM (ATWS) MODIFICATIONS (TAC NO. 65969)

The Nuclear Regulatory Commission has issued the enclosed Amendment No. 41 to Facility Operating License No. NPF-29 for the Grand Gulf Nuclear Station, Unit 1. This amendment consists of changes to the Technical Specifications (TS) in response to your application dated August 13, 1987, as revised October 23, November 25, December 22, and December 27, 1987.

The amendment provides interim changes to the TS for the standby liquid control system and the ATWS recirculation pump trip system to reflect modifications made to conform to 10 CFR 50.62.

A copy of the Safety Evaluation is also enclosed. The Notice of Issuance will be included in the Commission's biweekly Federal Register notice.

Sincerely,

L L Kintner

Lester L. Kintner, Project Manager
Project Directorate II-1
Division of Reactor Projects-I/II
Office of Nuclear Reactor Regulation

Enclosures:
1. Amendment No. 41 to NPF-29
2. Safety Evaluation

cc w/enclosures:
See next page

Distribution:
See attached list

(GRAND GULF 1)

	LLK	EGA
LA:PDII-1:DRPR:NRR	PM:PDII-1:DRPR:NRR	PDII-1:DRPR:NRR
PDAnderson:dlm	LLKintner	EGAdensam
12/30/87	12/28/87	12/30/87

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PDR ADOCK 05000416
P PDR

Mr. Oliver D. Kingsley, Jr.
System Energy Resources, Inc.

Grand Gulf Nuclear Station (GGNS)

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MISSISSIPPI POWER & LIGHT COMPANY

SYSTEM ENERGY RESOURCES, INC.

SOUTH MISSISSIPPI ELECTRIC POWER ASSOCIATION

DOCKET NO. 50-416

GRAND GULF NUCLEAR STATION, UNIT 1

AMENDMENT TO FACILITY OPERATING LICENSE

Amendment No. 41
License No. NPF-29

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

   A. The application for amendment by System Energy Resources, Inc., (the licensee), dated August 13, 1987, as revised October 23, 1987, November 25, 1987, December 22, and December 27, 1987, 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-29 is hereby amended to read as follows:

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(2) Technical Specifications

The Technical Specifications contained in Appendix A and the Environmental Protection Plan contained in Appendix B, as revised through Amendment No. 41 , are hereby incorporated into this license. System Energy Resources, Inc. shall operate the facility in accordance with the Technical Specifications and the Environmental Protection Plan.

3. This license amendment is effective as of its date of issuance.

FOR THE NUCLEAR REGULATORY COMMISSION

/S/

Gus C. Lainas, Assistant Director
for Region II Reactors
Division of Reactor Projects-I/II
Office of Nuclear Reactor Regulation

Attachment:
Changes to the Technical
Specifications

Date of Issuance: December 30, 1987

* SEE PREVIOUS CONCURRENCES

(GRAND GULF 1)

LA:PD21:DRPR	PM:PD21:DRPR	EMEP	L. Reyes by w/ edits conc Egan REGII:DD	OGC
PAnderson:dlm	LLKintner	*LMarsh		*JScinto
12/30/87	12/30/87	12/29/87	12/30/87	12/29/87
D:PD21:DRPR	DRPA:DRPR			
EAdensam	GLainas			
12/30/87	12/30/87			

ATTACHMENT TO LICENSE AMENDMENT NO. 41

FACILITY OPERATING LICENSE NO. NPF-29

DOCKET NO. 50-416

Replace the following pages of the Appendix A Technical Specifications with the attached pages. The revised pages are identified by Amendment number and contain vertical lines indicating the area of change. The corresponding overleaf page(s) have been provided to maintain document completeness.

Remove	Insert
3/4 1-17	3/4 1-17 (overleaf)
3/4 1-18	3/4 1-18
3/4 1-19	3/4 1-19
3/4 1-20	3/4 1-20
3/4 3-37	3/4 3-37
--	3/4 3-37a
3/4 3-38	3/4 3-38 (overleaf)
3/4 3-39	3/4 3-39
3/4 3-40	3/4 3-40 (overleaf)
3/4 6-41	3/4 6-41 (overleaf)
3/4 6-42	3/4 6-42
B 3/4 1-3	B 3/4 1-3 (overleaf)
B 3/4 1-4	B 3/4 1-4
B 3/4 1-4a	B 3/4 1-4a
B 3/4 3-1	B 3/4 3-1 (overleaf)
B 3/4 3-2	B 3/4 3-2
B 3/4 3-3	B 3/4 3-3
B 3/4 3-3a	B 3/4 3-3a

SURVEILLANCE REQUIREMENTS

4.1.4.2 The RPCS shall be demonstrated OPERABLE by:

a. Verifying the OPERABILITY of the rod pattern controller function when THERMAL POWER is less than the low power setpoint by selecting and attempting to move an inhibited control rod:

   1. After withdrawal of the first insequence control rod or gang for each reactor startup.

   2. As soon as the rod inhibit mode is automatically initiated at the RPCS low power setpoint, 20 +15, -0% of RATED THERMAL POWER, during power reduction.

   3. The first time only that a banked position, N1, N2, or N3, is reached during startup or during power reduction below the RPCS low power setpoint.

b. Verifying the OPERABILITY of the rod withdrawal limiter function when THERMAL POWER is greater than or equal to the low power setpoint by selecting and attempting to move a restricted control rod in excess of the allowable distance:

   1. As each power range above the RPCS low power setpoint is entered during a power increase or decrease.

   2. At least once per 31 days while operation continues within a given power range above the RPCS low power setpoint.

c. Verifying each RPCS bypass switch is in the unbypassed position or is in compliance with ACTION b.3 of this specification:

   1. At least once per 24 hours.

   2. Prior to a control rod movement, except by scram, following a power reduction to less than or equal to the low power setpoint.

3/4.1.5 STANDBY LIQUID CONTROL SYSTEM

LIMITING CONDITION FOR OPERATION

3.1.5 Two standby liquid control system subsystems shall be OPERABLE.

APPLICABILITY: OPERATIONAL CONDITIONS 1, 2 and 5*.

ACTION:

a. In OPERATIONAL CONDITION 1 or 2:

1. With one system subsystem inoperable, restore the inoperable subsystem to OPERABLE status within 7 days or be in at least HOT SHUTDOWN within the next 12 hours.

2. With both standby liquid control system subsystems inoperable, restore at least one subsystem to OPERABLE status within 8 hours or be in at least HOT SHUTDOWN within the next 12 hours.

b. In OPERATIONAL CONDITION 5*:

1. With one system subsystem inoperable, restore the inoperable subsystem to OPERABLE status within 30 days or insert all insertable control rods within the next hour.

2. With both standby liquid control system subsystems inoperable, insert all insertable control rods within one hour.

SURVEILLANCE REQUIREMENTS

4.1.5 Each standby liquid control system subsystem shall be demonstrated OPERABLE:

a. At least once per 24 hours by verifying that;

1. The temperature of the sodium pentaborate solution is within the limits of Figure 3.1.5-1.

2. The available volume of sodium pentaborate solution is greater than or equal to 4530 gallons.

3. The heat tracing circuit is OPERABLE by determining the temperature of the pump suction piping is within the limits of Figure 3.1.5-1.

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

SURVEILLANCE REQUIREMENTS (Continued)

b. At least once per 31 days by;

1. Starting both pumps and recirculating demineralized water to the test tank.

2. Verifying the continuity of the explosive charge.

3. Determining that the available weight of sodium pentaborate is greater than or equal to 5800 lbs and the concentration of boron in solution is within the limits of Figure 3.1.5-1 by chemical analysis.*

4. Verifying that each valve, manual, power operated or automatic, in the flow path that is not locked, sealed, or otherwise secured in position, is in its correct position.

c. Demonstrating that, when tested pursuant to Specification 4.0.5, the minimum flow requirement of 41.2 gpm at a pressure of greater than or equal to 1300 psig is met, without actuation of the pump relief valve.

d. At least once per 18 months during shutdown by;

1. Initiating one of the standby liquid control system subsystems, including an explosive valve, and verifying that a flow path from the pumps to the reactor pressure vessel is available by pumping demineralized water into the reactor vessel. The replacement charge for the explosive valve shall be from the same manufactured batch as the one fired or from another batch which has been certified by having one of that batch successfully fired. Both system subsystems shall be tested in 36 months.

2. Demonstrating that the pump relief valve opens within 3% of the system design pressure and verifying that the relief valve does not actuate during recirculation to the test tank.

3. **Demonstrating that all heat traced piping between the storage tank and the reactor vessel is unblocked by pumping from the storage tank to the test tank and then draining and flushing the piping with demineralized water.

4. Demonstrating that the storage tank heater is OPERABLE by verifying the expected temperature rise for the sodium pentaborate solution in the storage tank after the heater is energized.

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*This test shall also be performed anytime water or boron is added to the solution or when the solution temperature drops below the limit of Figure 3.1.5-1.

**This test shall also be performed whenever both heat tracing circuits have been found to be inoperable and may be performed by any series of sequential, overlapping or total flow path steps such that the entire flow path is included.

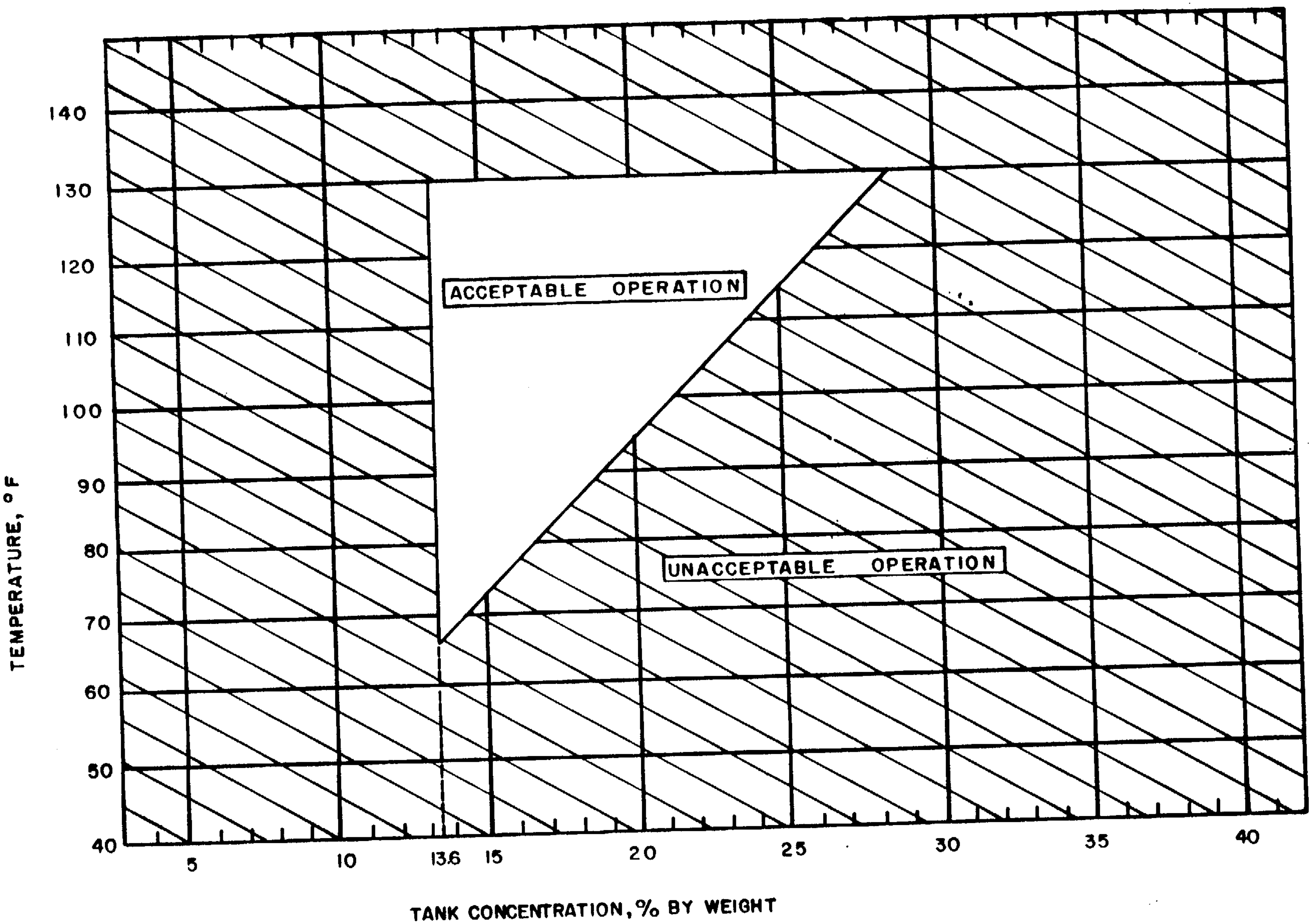

FIGURE 3.1.5-1 SODIUM PENTABORATE SOLUTION TEMPERATURE/CONCENTRATION REQUIREMENTS

3/4.3.4 RECIRCULATION PUMP TRIP ACTUATION INSTRUMENTATION

ATWS RECIRCULATION PUMP TRIP SYSTEM INSTRUMENTATION

LIMITING CONDITION FOR OPERATION

3.3.4.1 The anticipated transient without scram recirculation pump trip (ATWS-RPT) system instrumentation channels shown in Table 3.3.4.1-1 shall be OPERABLE with their trip setpoints set consistent with values shown in the Trip Setpoint column of Table 3.3.4.1-2.

APPLICABILITY: OPERATIONAL CONDITION 1.

ACTION:

a. With an ATWS-RPT system instrumentation channel trip setpoint less conservative than the value shown in the Allowable Values column of Table 3.3.4.1-2, declare the channel inoperable until the channel is restored to OPERABLE status with the channel setpoint adjusted consistent with the Trip Setpoint value.

b. With the number of OPERABLE channels one less than required by the Minimum OPERABLE Channels per Trip System requirement for one or both Trip System(s), restore the inoperable channel(s) to OPERABLE status within 14 days or be in at least STARTUP within the next 6 hours.

c. With the number of OPERABLE channels two or more less than required by the Minimum OPERABLE channels per Trip System requirement for one trip system and:

   1. If the inoperable channels consist of one reactor vessel water level channel and one reactor vessel pressure channel, place both inoperable channels in the tripped condition* within one hour or declare the trip system inoperable.

   2. If the inoperable channels include two reactor vessel water level channels or two reactor vessel pressure channels, declare the trip system inoperable.

d. With one trip system inoperable, restore the inoperable trip system to OPERABLE status within 72 hours or be in at least STARTUP within the next 6 hours.

e. With both trip systems inoperable, restore at least one trip system to OPERABLE status within one hour or be in at least STARTUP within the next 6 hours.

*The inoperable channels need not be placed in the tripped condition where this would cause the Trip Function to occur.

INSTRUMENTATION

SURVEILLANCE REQUIREMENTS

4.3.4.1.1 Each ATWS recirculation pump trip system instrumentation channel shall be demonstrated OPERABLE by the performance of the CHANNEL CHECK, CHANNEL FUNCTIONAL TEST, and CHANNEL CALIBRATION operations at the frequencies shown in Table 4.3.4.1-1.

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

TABLE 3.3.4.1-1

ATWS RECIRCULATION PUMP TRIP SYSTEM INSTRUMENTATION

TRIP FUNCTION	MINIMUM OPERABLE CHANNELS PER TRIP SYSTEM[(a)]
1. Reactor Vessel Water Level - Low Low, Level 2	2
2. Reactor Vessel Pressure - High	2

(a) One channel may be placed in an inoperable status for up to 2 hours for required surveillance provided the other channel is OPERABLE.

TABLE 3.3.4.1-2

ATWS RECIRCULATION PUMP TRIP SYSTEM INSTRUMENTATION SETPOINTS

TRIP FUNCTION	TRIP SETPOINT	ALLOWABLE VALUE
1. Reactor Vessel Water Level - Low Low, Level 2	≥ -41.6 inches*	≥ -43.8 inches
2. Reactor Vessel Pressure - High	≤ 1095 psig	≤ 1102 psig

*See Bases Figure B3/4 3-1.

TABLE 4.3.4.1-1

ATWS RECIRCULATION PUMP TRIP ACTUATION INSTRUMENTATION
SURVEILLANCE REQUIREMENTS

TRIP FUNCTION	CHANNEL CHECK	CHANNEL FUNCTIONAL TEST	CHANNEL CALIBRATION
1. Reactor Vessel Water Level - Low Low, Level 2	S	M	R*
2. Reactor Vessel Pressure - High	S	M	R*

*Calibrate trip unit at least once per 31 days.

TABLE 3.6.4-1 (Continued)

CONTAINMENT AND DRYWELL ISOLATION VALVES

SYSTEM AND VALVE NUMBER		PENETRATION NUMBER
Containment (Continued)		
RHR Pump "B" Test Line	E12-F335	67(O)(c)
RHR "B" Test Line To Suppr. Pool	E12-F290B-B	67(O)(d)
Inst. Air to ADS	P53-F006	70(I)
LPCS Relief Valve Vent Header	E21-F018	71A(O)(d)
RHR Pump "C" Relief Valve Vent Header	E12-F025C	71B(O)(d)
RHR "C" Relief Valve Vent Hdr. to Suppr. Pool & Post-Acc. Sample Return	E12-F406	71B(I)(c)
RHR Shutdown Vent Header	E12-F036	73(O)
RHR Shutdown Suction Relief Valve Disch.	E12-F005	76B(O)
RHR Heat Ex. "A" Relief Vent Header	E12-F055A	77(O)(d)
RHR Heat Ex. "A" Relief Vent Header	E12-F103A	77(O)(d)
RHR Heat Ex. "A" Relief Vent Header	E12-F104A	77(O)(d)
SSW "A" Supply	P41-F169A	89(I)(c)
SSW "B" Supply	P41-F169B	92(I)(c)
Ctmt. Leak Rate Test Inst.	M61-F015	110A(I)
Ctmt. Leak Rate Test Inst.	M61-F014	110A(O)
Ctmt. Leak Rate Test Inst.	M61-F019	110C(I)
Ctmt. Leak Rate Test Inst.	M61-F018	110C(O)
Ctmt. Leak Rate Test Inst.	M61-F017	110F(I)
Ctmt. Leak Rate Test Inst.	M61-F016	110F(O)
b. Drywell		
LPCI "A"	E12-F041A	313(I)
LPCI "B"	E12-F041B	314(I)
LPCI "B"	E12-F236	314(O)
CRD to Recirc. Pump A Seals	B33-F013A	326(I)

TABLE 3.6.4-1 (Continued)

CONTAINMENT AND DRYWELL ISOLATION VALVES

SYSTEM AND VALVE NUMBER		PENETRATION NUMBER
Drywell (Continued)		
CRD to Recirc. Pump A Seals	B33-F017A	326(0)
Instrument Air	P53-F008	335(I)
Standby Liquid Control	C41-F007	328(I)
Standby Liquid Control	C41-F006	328(0)
Standby Liquid Control-Drain	C41-F218	328(I)
Cont. Cooling Water Supply	P42-F115	329(I)
Drywell Chilled Water Supply	P72-F147	332(I)
Condensate Flush Conn.	B33-F204	333(I)
Condensate Flush Conn.	B33-F205	333(0)
Combustible Gas Control	E61-F002A	339(0)
Combustible Gas Control	E61-F002B	338(0)
Combustible Gas Control	E61-F004A	340(0)
Combustible Gas Control	E61-F004B	340(0)
Upper Containment Pool Drain	G41-F265	342(0)
CRD to Recirc. Pump B Seals	B33-F013B	346(I)
CRD to Recirc. Pump B Seals	B33-F017B	346(0)
Service Air	P52-F196	363(I)
Cont. Leak Rate Test Inst.	M61-F021	438A(I)
Cont. Leak Rate Sys.	M61-F020	438A(0)
BLIND FLANGES		
Cont. Leak Rate Sys.	NA	40(I)(0)
Cont. Leak Rate Sys.	NA	82(I)(0)
Containment Leak Rate System	NA	343(I)(0)

CONTROL RODS (Continued)

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

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

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

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

3/4.1.4 CONTROL ROD PROGRAM CONTROLS

The rod withdrawal limiter system input power signal orginates from the first stage turbine pressure. When operating with the steam bypass valves open, this signal indicates a core power level which is less than the true core power. Consequently, near the low power setpoint and high power setpoint of the rod pattern control system, the potential exists for nonconservative control rod withdrawals. Therefore, when operating at a sufficiently high power level, there is a small probability of violating fuel Safety Limits during a licensing basis rod withdrawal error transient. To ensure that fuel Safety Limits are not violated, this specification prohibits control rod withdrawal when a biased power signal exists and core power exceeds the specified level.

Control rod withdrawal and insertion sequences are established to assure that the maximum insequence individual control rod or control rod segments which are withdrawn at any time during the fuel cycle could not be worth enough to result in a peak fuel enthalpy greater than 280 cal/gm in the event of a control rod drop accident. The specified sequences are characterized by homogeneous, scattered patterns of control rod withdrawal. When THERMAL POWER is greater than 20% of RATED THERMAL POWER, there is no possible rod worth which, if dropped at the design rate of the velocity limiter, could result in a peak enthalpy of 280 cal/gm. Thus requiring the rod pattern controller function to be OPERABLE when THERMAL POWER is less than or equal to 20% of RATED THERMAL POWER provides adequate control.

CONTROL ROD PROGRAM CONTROLS (Continued)

The RPCS provides automatic supervision to assure that out-of-sequence rods will not be withdrawn or inserted. A rod is out of sequence if it does not meet the criteria of the Banked Position Withdrawal Sequence as described in the FSAR. The RPCS function is allowed to be bypassed in the Rod Action Control System (RACS) if necessary, for example, to insert an inoperable control rod, return an out-of-sequence control rod to the proper in-sequence position or move an in-sequence control rod to another in-sequence position. The requirement that a second qualified individual verify such bypassing and positioning of control rods ensures that the bases for RPCS limitations are not exceeded. In addition, if THERMAL POWER is below the low power setpoint, additional restrictions are provided when bypassing control rods to ensure operation at all times within the basis of the control rod drop accident analysis.

The analysis of the rod drop accident is presented in Section 15.4 of the FSAR and the techniques of the analysis are presented in a topical report, Reference 1, and two supplements, References 2 and 3.

The RPCS is also designed to automatically prevent fuel damage in the event of erroneous rod withdrawal from locations of high power density during higher power operation.

A dual channel system is provided that, above the low power setpoint, restricts the withdrawal distances of all non-peripheral control rods. This restriction is greatest at highest power levels.

3/4.1.5 STANDBY LIQUID CONTROL SYSTEM

The standby liquid control system provides a backup capability for bringing the reactor from full power to a cold, xenon-free shutdown, assuming that the withdrawn control rods remain fixed in the rated power pattern. To meet this objective it is necessary to inject a quantity of boron which produces a concentration of 660 ppm in the reactor core in approximately 90 to 120 minutes. A minimum available quantity of 4530 gallons of sodium pentaborate solution containing a minimum of 5800 lbs. of sodium pentaborate is required to meet a shutdown requirement of 3%. There is an additional allowance of 165 ppm in the reactor core to account for imperfect mixing and leakage. The time requirement was selected to override the reactivity insertion rate due to cooldown following the xenon poison peak and the required pumping rate is 41.2 gpm. The minimum storage volume of the solution is established to allow for the portion below the pump suction that cannot be inserted. The temperature requirement is necessary to ensure that the sodium pentaborate remains in solution.

1. C. J. Paone, R. C. Stirn and J. A. Woolley, "Rod Drop Accident Analysis for Large BWR's," G. E. Topical Report NEDO-10527, March 1972

2. C. J. Paone, R. C. Stirn and R. M. Young, Supplement 1 to NEDO-10527, July 1972

3. J. M. Haun, C. J. Paone and R. C. Stirn, Addendum 2, "Exposed Cores," Supplement 2 to NEDO-10527, January 1973

STANDBY LIQUID CONTROL SYSTEM (Continued)

With redundant pumps and explosive injection valves and with a highly reliable control rod scram system, operation of the reactor is permitted to continue for short periods of time with the system inoperable or for longer periods of time with one of the redundant components inoperable.

Relief valves are provided on the SLCS pump discharge piping to protect the SLCS pump and piping from overpressure conditions. Testing of the relief valve setpoint and verifying that the relief valve does not open during steady state operation of the SLCS pumps demonstrates OPERABILITY of the relief valve. The relief valves are ASME Class 2 valves and, as such, have a ± 3% tolerance in the opening pressure from the set pressure, per the ASME Code (Section III - Division 1 Subsection NC-7614.2(b), 1974 Edition).

Surveillance requirements are established on a frequency that assures a high reliability of the system. Once the solution is established, boron concentration will not vary unless more boron or water is added, thus a check on the temperature and volume once each 24 hours assures that the solution is available for use.

Replacement of the explosive charges in the valves at regular intervals will assure that these valves will not fail because of deterioration of the charges.

Compliance with the NRC ATWS Rule 10CFR50.62 has been demonstrated by means of the equivalent control capacity concept using the plant specific minimum parameters. This concept requires that each boiling water reactor must have a standby liquid control system with a minimum flow capacity and boron content equivalent in control capacity to 86 gpm for 13% weight sodium pentaborate solution (natural boron enrichment) used for the 251-inch diameter reactor vessel studied in NEDE-24222, Reference 4. The described minimum system parameters (82.4 gpm, 13.6% weight with natural boron enrichment) provides an equivalent control capacity to the 10CFR 50.62 requirement. The techniques of the analysis are presented in a licensing topical report NEDE-31096-P, Reference 5.

Only one subsystem is needed to fulfill the system design basis, and two subsystems are needed to fulfill ATWS rule requirements. An SLCS subsystem consists of the storage tank, one divisional pump, explosive type valve, and associated controls, and other valves, piping, instrumentation, and controls necessary to prepare and inject neutron absorbing solution into the reactor.

4. "Assessment of BWR Mitigation of ATWS, Volume II," NEDE-24222, December 1979.

5. L. B. Claasen et al., "Anticipated Transients Without Scram, Response to NRC ATWS Rule 10CFR50.62," G. E. Licensing Topical Report prepared for the BWR Owners' Group, NEDE-31096-P, December 1985.

BASES

3/4.3.1 REACTOR PROTECTION SYSTEM INSTRUMENTATION

The reactor protection system automatically initiates a reactor scram to:

a. Preserve the integrity of the fuel cladding.

b. Preserve the integrity of the reactor coolant system.

c. Minimize the energy which must be absorbed following a loss-of-coolant accident, and

d. Prevent inadvertent criticality.

This specification provides the limiting conditions for operation necessary to preserve the ability of the system to perform its intended function even during periods when instrument channels may be out of service because of maintenance. When necessary, one channel may be made inoperable for brief intervals to conduct required surveillance.

The reactor protection system is made up of two independent trip systems. There are usually four channels to monitor each parameter with two channels in each trip system. The outputs of the channels in a trip system are combined in a logic so that either channel will trip that trip system. The tripping of both trip systems will produce a reactor scram. The bases for the trip settings of the RPS are discussed in the bases for Specification 2.2.1.

The measurement of response time at the specified frequencies provides assurance that the protective functions associated with each channel are completed within the time limit assumed in the accident analysis. No credit was taken for those channels with response times indicated as not applicable. Response time may be demonstrated by any series of sequential, overlapping or total channel test measurement, provided such tests demonstrate the total channel response time as defined. Sensor response time verification may be demonstrated by either (1) inplace, onsite or offsite test measurements, or (2) utilizing replacement sensors with certified response times.

3/4.3.2 ISOLATION ACTUATION INSTRUMENTATION

This specification ensures the effectiveness of the instrumentation used to mitigate the consequences of accidents by prescribing the OPERABILITY trip setpoints and response times for isolation of the reactor systems. When necessary, one channel may be inoperable for brief intervals to conduct required surveillance. Some of the trip settings may have tolerances explicitly stated where both the high and low values are critical and may have a substantial effect on safety. Negative barometric pressure fluctuations are accounted for in the trip setpoints and allowable values specified for drywell pressure-high. The setpoints of other instrumentation, where only the high or low end of the setting have a direct bearing on safety, are established at a level away from the normal operating range to prevent inadvertent actuation of the systems involved.

Except for the MSIVs, the safety analysis does not address individual sensor response times or the response times of the logic systems to which the sensors are connected. For D.C. operated valves, a 3 second delay is assumed before the valve starts to move. For A.C. operated valves, it is assumed that

ISOLATION ACTUATION INSTRUMENTATION (continued)

the A.C. power supply is lost and is restored by startup of the emergency diesel generators. In this event, a time of 10 seconds is assumed before the valve starts to move. In addition to the pipe break, the failure of the D.C. operated valve is assumed; thus the signal delay (sensor response) is concurrent with the 10 second diesel startup. The safety analysis considers an allowable inventory loss in each case which in turn determines the valve speed in conjunction with the 10 second delay. It follows that checking the valve speeds and the 10 second time for emergency power establishment will establish the response time for the isolation functions. However, to enhance overall system reliability and to monitor instrument channel response time trends, the isolation actuation instrumentation response time shall be measured and recorded as a part of the ISOLATION SYSTEM RESPONSE TIME.

Operation with a trip set less conservative than its Trip Setpoint but within its specified Allowable Value is acceptable on the basis that the difference between each Trip Setpoint and the Allowable Value is equal to or greater than the drift allowance assumed for each trip in the safety analyses.

3/4.3.3 EMERGENCY CORE COOLING SYSTEM ACTUATION INSTRUMENTATION

The emergency core cooling system actuation instrumentation is provided to initiate actions to mitigate the consequences of accidents that are beyond the ability of the operator to control. This specification provides the OPERABILITY requirements, trip setpoints and response times that will ensure effectiveness of the systems to provide the design protection. Negative barometric pressure fluctuations are accounted for in the trip setpoints and allowable values specified for drywell pressure-high. Although the instruments are listed by system, in some cases the same instrument may be used to send the actuation signal to more than one system at the same time.

Operation with a trip set less conservative than its Trip Setpoint but within its specified Allowable Value is acceptable on the basis that the difference between each Trip Setpoint and the Allowable Value is equal to or greater than the drift allowance assumed for each trip in the safety analyses.

3/4.3.4 RECIRCULATION PUMP TRIP ACTUATION INSTRUMENTATION

The anticipated transient without scram recirculation pump trip (ATWS-RPT) system provides a means of limiting the consequences of the unlikely occurrence of a failure to scram during an anticipated transient. The response of the plant to this postulated event has been evaluated in General Electric Company report NEDC-32408 dated March 1987. The results of the analysis show that the Grand Gulf ATWS-RPT design provides adequate protection for these events in which the normal scram paths fail.

The ATWS-RPT provides fully redundant trip of the recirculation pump motors so that the pumps coast down to zero speed. This trip function reduces core flow creating steam voids in the core, thereby decreasing power generation and limiting any power or pressure excursions. The Grand Gulf ATWS-RPT design provides compliance with the requirements of the NRC ATWS Rule 10CFR50.62.

RECIRCULATION PUMP TRIP ACTUATION INSTRUMENTATION (Continued)

The ATWS-RPT and Alternate Rod Insertion (ARI) system use common setpoints and trip channels (transmitters and trip systems). Therefore, the ARI trip function and the RPT trip function will be initiated simultaneously. The instrumentation setpoints for the RPV pressure and water level trip channels are established such that the normal scram paths for these variables would already be initiated.

The end-of-cycle recirculation pump trip (EOC-RPT) system is a part of the Reactor Protection System and is an essential safety supplement to the reactor trip. The purpose of the EOC-RPT is to recover the loss of thermal margin which occurs at the end-of-cycle. The physical phenomenon involved is that the void reactivity feedback due to a pressurization transient can add positive reactivity to the reactor system at a faster rate than the control rods add negative scram reactivity. Each EOC-RPT system trips both recirculation pumps, reducing coolant flow in order to reduce the void collapse in the core during two of the most limiting pressurization events. The two events for which the EOC-RPT protective feature will function are closure of the turbine stop valves and fast closure of the turbine control valves.

A fast closure sensor from each of two turbine control valves provides input to the EOC-RPT system; a fast closure sensor from each of the other two turbine control valves provides input to the second EOC-RPT system. Similarly, a closure sensor for each of two turbine stop valves provides input to one EOC-RPT system; a closure sensor from each of the other two stop valves provides input to the other EOC-RPT system. For each EOC-RPT system, the sensor relay contacts are arranged to form a 2-out-of-2 logic for the fast closure of turbine control valves and a 2-out-of-2 logic for the turbine stop valves. The operation of either logic will actuate the EOC-RPT system and trip both recirculation pumps.

Each EOC-RPT system may be manually bypassed by use of a keyswitch which is administratively controlled. The manual bypasses and the automatic Operating Bypass at less than 40% of RATED THERMAL POWER are annunciated in the control room. The automatic bypass setpoint is feedwater temperature dependent due to the subcooling changes that affect the turbine first-stage pressure-reactor power relationship. For RATED THERMAL POWER operation with feedwater temperature greater than or equal to 420°F, an allowable setpoint of $\leq$ 26.9% of control valve wide open turbine first-stage pressure is provided for the bypass function. This setpoint is also applicable to operation at less than RATED THERMAL POWER with the correspondingly lower feedwater temperature. The allowable setpoint is reduced to $\leq$ 22.5% of control valve wide open turbine first-stage pressure for RATED THERMAL POWER operation with a feedwater temperature between 370°F and 420°F. Similarly, the reduced setpoint is applicable to operation at less than RATED THERMAL POWER with the corresponding lower feedwater temperature.

The EOC-RPT system response time is the time assumed in the analysis between initiation of valve motion and complete suppression of the electric arc, i.e., 190 ms. Included in this time are: the response time of the sensor, the response time of the system logic and the breaker interruption time. Breaker interruption time includes both breaker response time and the manufacturer's design arc suppression time of 12 ms.

RECIRCULATION PUMP TRIP ACTUATION INSTRUMENTATION (Continued)

Operation with a trip set less conservative than its Trip Setpoint but within its specified Allowable Value is acceptable on the basis that the difference between each Trip Setpoint and the Allowable Value is equal to or greater then the drift allowance assumed for each trip in the safety analyses.

3/4.3.5 REACTOR CORE ISOLATION COOLING SYSTEM ACTUATION INSTRUMENTATION

The reactor core isolation cooling system actuation instrumentation is provided to initiate actions to assure adequate core cooling in the event of reactor isolation from its primary heat sink and the loss of feedwater flow to the reactor vessel without providing actuation of any of the emergency core cooling equipment.

Operation with a trip set less conservative than its Trip Setpoint but within its specified Allowable Value is acceptable on the basis that the difference between each Trip Setpoint and the Allowable Value is equal to or greater than the drift allowance assumed for each trip in the safety analyses.

3/4.3.6 CONTROL ROD BLOCK INSTRUMENTATION

The control rod block functions are provided consistent with the requirements of the specifications in Section 3/4.1.4, Control Rod Program Controls and Section 3/4.2 Power Distribution Limits. The trip logic is arranged so that a trip in any one of the inputs will results in a control rod block.

The OPERABILITY of the control rod block instrumentation in OPERATIONAL CONDITION 5 is to provide diversity of rod block protection to the one-rod-out interlock.

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

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

SUPPORTING AMENDMENT NO. 41 TO FACILITY OPERATING LICENSE NO. NPF-29

MISSISSIPPI POWER & LIGHT COMPANY

SYSTEM ENERGY RESOURCES, INC.

SOUTH MISSISSIPPI ELECTRIC POWER ASSOCIATION

GRAND GULF NUCLEAR STATION, UNIT 1

DOCKET NO. 50-416

1.0 INTRODUCTION

By letters dated August 13, 1987, October 23, 1987, November 25, 1987, December 22, and December 27, 1987, System Energy Resources, Inc. (SERI or the licensee), requested an amendment to Facility Operating License No. NPF-29 for the Grand Gulf Nuclear Station, Unit 1. The proposed amendment would provide interim changes to the Technical Specifications (TS) for the standby liquid control system (SLCS) and the Anticipated Transient Without Scram recirculation pump trip (ATWS-RPT) system to reflect modifications to these systems. The modifications to these systems will be made during the second refueling outage to conform to 10 CFR 50.62. A third system required by 10 CFR 50.62, the alternate rod insertion (ARI) system, which will be installed during the second refueling outage, will not require changes to the TS at this time. The staff will provide guidance on a generic basis regarding TS requirements for the ATWS-RPT and ARI systems at a later date. Evaluation of changes to the TS for the ATWS - RPT system and the SLCS are provided below.

The letters dated December 22, and December 27, 1987, requested emergency consideration for the issuance of the license amendment pursuant to 10 CFR 50.91(a)(5). Our evaluation of the licensee's explanation of emergency circumstances is provided in Section 3.0 of this safety evaluation. The December 22, and December 27, 1987, letters did not change the substance of previous submittals which were noticed in the Federal Register on December 4, 1987. Therefore, it is not necessary to renotice the proposed amendment.

2.0 EVALUATION

2.1 ATWS recirculation pump trip system.

The TS for the ATWS - RPT system would be changed by adding Actions c, d and e in TS Section 3.3.4.1, decreasing the trip setpoint and allowable value for the reactor vessel high pressure actuation instrumentation in Table 3.3.4.1-2, and revising the Bases Section 3/4.3.4 to reflect the modifications.

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By letter dated April 3, 1987, the licensee committed to upgrade the ATWS - RPT system to the designs described in BWROG licensing topical report NEDE-31096-P and the applicable conditions of the NRC safety evaluation of this topical report. The licensee will modify the existing ATWS - RPT system during the second refueling outage. It will utilize redundant breakers for each power feed. The trip function will use energized-to-trip logic and either one of the redundant logics will trip both pumps. The proposed Action Statements c, d and e to be added in TS Section 3.3.4.1 will cover all possible situations for the modified ATWS-RPT system operating conditions. The staff finds that these action statements will allow for greater operational flexibility. They are comparable to the existing end-of-cycle (EOC) RPT action statements, and are similar to action statements in the Perry (BWR/6) TS, which have been approved by the staff. Therefore, the staff finds these proposed action statements to be acceptable.

With respect to the reactor vessel high pressure trip setpoint change from $\leq$ 1125 psig to $\leq$ 1095 psig and the allowable value change from $\leq$ 1140 psig to $\leq$ 1102 psig, the staff finds that these changes are in the conservative direction. Since normal scram setpoint for reactor vessel high pressure is $\leq$ 1064.7 psig, the ATWS-RPT setpoint will not significantly challenge scram actuation. The staff finds these changes acceptable.

The changes in the TS Bases Section 3/4.3.4, are made to reference the analysis performed by General Electric in a topical report, NEDC-32408, dated March 1987, and describe the modifications made to the ATWS-RPT and ARI systems. The staff finds the changes to the TS Bases to be acceptable.

The staff finds that proposed changes in the TS for the ATWS - RPT system, as described in the licensee's submittals are acceptable on an interim basis. Technical Specification requirements may be changed when generic TS for the ARI and ATWS - RPT systems are finalized by the staff.

2.2 <u>Standby liquid control system (SLCS).</u>

The TS for the SLCS would be changed by (1) adding maximum temperature and minimum concentration limits in TS Figure 3.1.5-1, "Sodium Pentaborate Solution Temperature/Concentration Requirements," (2) changing surveillance requirements in TS Section 4.1.5 to meet requirements for ATWS, as well as other design basis accidents, and (3) revising the TS Bases Section 3/4.1.5 to reflect system modifications for ATWS.

With respect to TS Figure 3.1.5-1, the curve of temperature versus concentration required to keep sodium pentaborate in solution remains the same, but minimum sodium pentaborate concentration is limited to that required for an ATWS (13.6% by weight) and maximum temperature is limited to that required to assure adequate suction head for the SLCS pumps (130°F). These changes are consistent with ATWS requirements and are acceptable.

Surveillance requirements in TS Section 4.1.5 are modified as follows.

1. The limiting temperature of the SLCS pump suction piping is changed from "greater than or equal to 70°F" to the requirements of the new TS Figure 3.1.5-1. This change provides consistency with the temperature limits in the new figure and is, therefore, acceptable.

2. The minimum available volume of sodium pentaborate solution and weight of sodium pentaborate are changed to ensure that the present requirements for design basis accidents are maintained together with the new ATWS requirements. Therefore, these changes are acceptable.

3. For the surveillance test to determine adequate flow rate in the SLCS, the minimum test pressure is increased from 1220 psig to 1300 psig to account for additional injection piping pressure losses resulting from the two-pump flow rate of 82.4 gpm for ATWS accidents. The 1300 psig is based on a conservative analytical calculation that is acceptable to the staff. When the actual losses are measured in the testing following completion of the physical modification, the TS may need to be amended. In the interim, a minimum test pressure of 1300 psig is acceptable.

4. The pump relief valve setpoint would be increased to provide assurance that the relief valve would not open during system operation. The revised setpoint is identified as "system design pressure" in proposed TS Section 4.1.5.d.2 based on the licensee's commitment to increase the system design pressure to "approximately 1500 psig" by reanalyzing the piping to demonstrate it will satisfy ASME Code, Section III allowable stress values at pressures up to 1500 psig. The use of "system design pressure" in the specification is acceptable, provided the analyses show that the system design pressure is found to be equal to or greater than 1500 psig.

The licensee's submittals included a description of planned testing of two pump operation to assure that a total flow rate of 82.4 gpm can be achieved. The submittals also included revised TS Bases 3/4.1.5, which states that two SLCS subsystems are needed to fulfill 10 CFR 50.62 requirements. Each subsystem includes one pump with a flow rate of at least 41.2 gpm. The post-implementation acceptance test will consist of simultaneous operation of both pumps through the test loop while applying a steady back pressure of 1300 psig and verifying that the system flow rate is at least 82.4 gpm. A single pump surveillance test performed at least once per 18 months to verify a flow rate of at least 41.2 gpm at the same backpressure as the two pump test is included in proposed TS Section 4.1.5.c. Based on this description of the two-pump test and the revised pump relief valve setpoint of 1500 psig which ensures the valve will not lift during the test, we find the change in TS Bases 3/4.1.5 to be acceptable.

The licensee has proposed the addition of a drywell isolation valve in TS Table 3.6.4-1 because of modifications to the SLCS discharge piping. This isolation valve is in the drain line from the SLCS discharge piping. The drain line has been moved from outside the drywell to inside the drywell, between the two drywell isolation valves for the SLCS discharge piping.

This drain line will be isolated from the SLCS discharge line by two normally closed valves. One of the two valves (F218) will serve as an inboard drywell isolation valve and is added to TS Table 3.6.4.1. The two drywell isolation valves (F006 and F007) for the SLCS discharge line and their test connection isolation valve (F026) will not be changed.

The Grand Gulf plant has a Mark III containment. Drywell isolation is not designed to prevent the uncontrolled release of radioactivity from the containment to the environment. However, uncontrolled bypass leakage paths between the drywell and the containment could produce pressurization of the containment and increase containment pressure during the blowdown from a loss of coolant accident (LOCA). The licensee has previously reviewed the allowable drywell bypass leakage and determined the amount of steam that could bypass the suppression pool without exceeding the containment design pressure. Since the drywell isolation valve (F006) remains in its current position and relocation of the drain line from outside to inside of the drywell improves drywell integrity, the changes will not increase the potential for drywell bypass. Therefore, the staff concludes that the proposed TS changes are acceptable.

The licensee has proposed ASME Code classification changes to the SLCS discharge piping. The ASME Code, Section III, Class 1 to Class 2 interface has been moved from the explosive valves (F004A and F004B) to the outboard drywell isolation valve in the discharge piping (F006). This change meets the requirements for reactor coolant pressure boundary in 10 CFR 50.2 and 10 CFR 50.55a(c)(1), in that the ASME Code, Section III, Class 1, piping extends to the outboard isolation valve. Therefore, this change is acceptable to the staff.

The staff concludes that the proposed TS changes for the SLCS system are in accord with the NRC staff approved modifications to be made to comply with 10 CFR 50.62 regarding ATWS requirements and will maintain the specifications for other design basis accidents. Accordingly, the staff concludes the proposed TS changes are acceptable.

3.0 EXIGENT OR EMERGENCY CIRCUMSTANCES

The initial application requesting changes to the TS to reflect ATWS-related equipment changes was filed on August 13, 1987. The ATWS-related equipment was scheduled to be installed in the second refueling outage. Based on the original (November 5, 1987) refueling outage (RF02) schedule, SERI planned to begin the outage by opening the generator output breakers on November 7, 1987. The schedule at that time called for resynchronizing the generator to the grid on January 5, 1988, thus ending RF02. That schedule showed a reactor restart (Mode 2) on January 3, 1988.

SERI management attention to schedule and timely reaction to potential delays have resulted in a positive impact on the schedule. Specific management decisions which resulted in net gains in the schedule were:

- Rework of a main steam isolation valve which had the potential for impacting the critical path activities. By rearranging other scheduled work activities this potential delay was avoided.

- Another potential delay pertaining to the Emergency Standby Diesel Generator turbo-chargers was absorbed by resequencing the ECCS testing and completing the test in less than scheduled time.

- SERI stopped non-essential vibration monitoring instrumentation removal activities early so that it would not become a critical path activity. As part of the pre-outage planning and development, General Electric had evaluated and concurred with not completing the removal of this instrumentation during this refueling outage.

- SERI's original planning called for Christmas Eve to be a holiday for workers except those working critical path jobs. Management later decided this was not feasible and therefore made this a normal work day.

- The operational hydrostatic test of reactor pressure vessel which is the current critical path activity is scheduled to be completed December 28, 1987, and is currently ahead of schedule.

- The original schedule included a 5-day window for system restoration and paperwork closeout. This has been reduced by management attention throughout the outage. This attention has caused the systems to be restored and the paperwork to be closed out as the work was completed, thus resulting in an anticipated savings of one to two days in the schedule.

- I&C surveillances are outage critical path activities. Special steps have been taken to insure that these surveillances are managed and executed effectively. Dedicated I&C teams along with dedicated management representatives from both Operations and I&C have resulted in overall schedule savings for I&C surveillances.

Throughout the outage SERI has maintained senior members of management on site 24 hours a day. This has resulted in problems being expeditiously resolved thus preventing impacts on scheduled activities and has increased attention to problems which could have impacted the overall schedule. This posture toward outage management has resulted in continuous management attention to schedule and has resulted in critical path activities being as much as two days ahead of schedule at some points during the outage. At this point in the outage, SERI anticipates going to Mode 2 on January 1, 1988.

The review of the application, including a request for additional information on August 21, 1987, several conference calls, and a meeting on November 20, 1987, resulted in the Federal Register notice being published on December 4, 1987. SERI delayed the request for emergency processing of the amendment until mid-December to allow more refinement in the outage schedule and more certainty in the need for the amendment prior to Janaury 4, 1988, when the 30 day comment period expires. By letter dated December 22, 1987, as supplemented December 27, 1987, the licensee requested emergency consideration for the license amendment regarding ATWS modifications pursuant to 10 CFR 50.91(a)(5) so that the amendment would be issued by December 30, 1987, in order to avoid a delay in resumption of power operation.

The staff has reviewed the licensee's explanation of the circumstances justifying consideration of this amendment on an emergency basis. The need for the amendment was due to the shortening of the outage schedule during December 1987. During the review period, several issues in the initial application were discussed and resolved, principally: (1) issuance of interim TS on the ATWS-RPT system and no TS for the ARI system until the staff develops generic ATWS TS; (2) an increase in SLCS design pressure from 1400 psig to 1500 psig to assure functioning of the two-pump ATWS mode, without bypass through the SLCS pump relief valve; (3) retention of ASME Code, Class 1, for the SLCS discharge piping from the drywell outboard isolation valve to the high pressure core spray system; and (4) a change from Class 1 to Class 2 piping from the explosive valves to the outboard isolation valve. The licensee's commitments in its November 25, 1987 letter demonstrate good faith efforts to resolve the issues. Based on this review, the staff finds that the licensee used its best efforts to apply for the subject amendment in a timely manner and that it had not acted in a manner to create the emergency to take advantage of these procedures.

4.0 FINAL NO SIGNIFICANT HAZARDS CONSIDERATION DETERMINATION

The Commission has provided standards in 10 CFR 50.92(c) for determining whether a significant hazards consideration exists. A proposed amendment to an operating license for a facility involves no significant hazards consideration if operation of the facility in accordance with the proposed amendment would not: (1) involve a significant increase in the probability or consequences of an accident previously evaluated; or (2) create the possibility of a new or different kind of accident from any accident previously evaluated; or (3) involve a significant reduction in a margin of safety.

The licensee has provided an analysis of no significant hazards consideration in its November 25, 1987 submittal. The licensee concluded that its proposed ATWS modifications and associated TS changes meet the three standards in 10 CFR 50.92(c).

The NRC staff has reviewed the licensee's submittal including its analysis about the issue of no significant hazards considerations. The staff concludes that the three standards in 10 CFR 50.92 are met for the following reasons.

Standard 1. The proposed amendment does not involve a significant increase in the probability or consequences of an accident previously evaluated.

The changes in the ATWS-RPT system would not increase the probability of an ATWS since the ATWS-RPT system would perform a mitigation function (tripping recirculation pumps thus increasing core steam voids and reducing core power) and the changes, therefore, would not affect ATWS precursors. These changes would not increase the consequences of an ATWS because the modified ATWS-RPT system would provide a redundant, and more reliable, trip of the recirculation pump motors. In addition, decreasing the trip set point would result in an earlier trip during an ATWS, thereby reducing consequences. The changes to the ATWS-RPT system would not affect the probability or consequences of previously evaluated accidents other than the ATWS because this system is designed specifically for the ATWS.

The addition of the ARI system would not increase the probability of an ATWS because the ARI system would perform a mitigation function (insert control rods) and the ARI system would not affect ATWS precursors. The ARI system would not increase the consequences of an ATWS because the ARI system provides another means of shutting down the reactor in the event of an ATWS. The ARI system is designed specifically for an ATWS and, therefore, addition of the ARI system would not affect the probability or consequences of other previously evaluated accidents.

The SLCS is a mitigation feature for the ATWS and other accidents previously analyzed and, therefore, the proposed changes would not affect the probability of an accident. The changes to the SLCS would provide for operation of both pumps simultaneously with slight changes in sodium pentaborate concentration and storage tank volume to meet 10 CFR 50.62 requirements. This change would decrease the time required to inject sodium pentaborate into the reactor and, therefore, would decrease the consequences of an ATWS. The changes would not significantly affect the consequences of accidents other than an ATWS, because the capability for single pump operation is retained. The SLCS piping would be modified to inject sodium pentaborate into the high pressure core spray header instead of the bottom of the reactor vessel to provide more effective mixing with water in the reactor vessel, thus reducing consequences of an accident. Movement of the ASME Code, Class 1, boundary in the SLCS discharge piping will not significantly increase the probability or consequences of an accident because the requirements of 10 CFR 50.55a(c)(1) for the reactor coolant pressure boundary are met for the modified piping. The proposed increase in SLCS pump relief valve setpoint would not increase the probability of piping failure because the piping will be reanalyzed and modified as necessary to demonstrate the ASME Code Section III criteria for design pressure are met.

<u>Standard 2.</u> The proposed amendment does not create the possibility of a new or different kind of accident from any previously analyzed.

The actuation instrumentation for the ATWS-RPT system would be modified to provide a redundant, more reliable trip of the recirculation pumps and to actuate the ARI valves. Since the ARI system would use the transmitters and trip units of the ATWS-RPT system, the only additional components added for the ARI system would be the ARI valves in three parallel vent pipes to vent the control rod scram pilot air header, thus causing control rod insertion. The ATWS-RPT system is electrically independent and separated from the reactor protection system (RPS) and all other Class 1E circuits. Therefore, failure of the ATWS-RPT system or any component of the ATWS-RPT will not create the possibility of a new or different kind of accident from any previously analyzed.

The ARI system will provide three additional vent paths from the existing scram pilot headers consisting of two ARI valves per vent path. The ARI system will utilize the same trip system as the RPT with the same trip logic. The ARI system will utilize no components associated with the RPS and is electrically independent from RPS and all other Class 1E circuits. Failure of any ARI valve in any mode (open or closed) would not inhibit the RPS scram function. The RPS scram function is to de-energize the scram solenoid valves on each hydraulic control unit. Because the scram solenoid valves are downstream of the ARI vent paths, the scram solenoid valves will be capable of performing their safety function (venting the RPS scram valves) regardless of the position of the ARI valves. The modifications associated with the installation of the ARI system will create no new system or component failure modes and therefore will not create the possibility of a new or different kind of accident from any previously analyzed.

The SLCS as modified will continue to meet the original design bases described in the GGNS FSAR (Section 9.3.6.1) and will continue to meet its design function (Section 9.3.6.2) with the added capability to run both pumps simultaneously to achieve the control capability required by 10 CFR 50.62. All modifications have been designed such that the original design bases of the SLCS are still valid. Modifications affecting Class 1E circuits have been designed to meet the applicable physical separation and independence criteria. Modifications to piping meet the applicable design requirements which will assure that the piping will function in its intended manner. The failure modes of the SLCS and components have been previously evaluated. The modifications to the existing SLCS meet all applicable design requirements, and no new failure modes are introduced. Therefore the modifications to the SLCS do not create the possibility of a new or different kind of accident from any previously analyzed.

Standard 3. The proposed amendment does not involve a significant reduction in a margin of safety.

The increase in the SLCS pump relief valve setpoint provides additional margin (100 psi versus 80 psi) between the relief valve setpoint and the SLCS operating pressure, thus increasing the reliability of operation. The decrease in the reactor pressure-high trip setpoint for the ATWS-RPT and connected ARI system would initiate these systems sooner, thus increasing the margin to core damage. The reduction of the trip setpoint increases slightly the likelihood of scramming the reactor with the ARI systems before a normal scram is initiated during a pressure transient, assuming extremes of instrument accuracy and drift. However, the consequences of a trip of the ARI system before a normal reactor trip would be to reduce the peak pressure in the transient, thereby increasing the margin to core damage. The change in the interface between ASME Code Class 1 piping and ASME Code Class 2 piping does not significantly affect the margin of safety for RCPB integrity because the modified piping meets the requirements of 10 CFR 50.55a(c)(1). Therefore, the proposed changes do not involve a significant reduction in the margin of safety.

Based on the review of the licensee's submittal and the evaluation above, the staff has made a final determination that the licensee's amendment request does not involve a significant hazards consideration.

5.0 ENVIRONMENTAL CONSIDERATION

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

6.0 CONCLUSION

The Commission made a proposed determination that this amendment involves no significant hazards consideration, which was published in the Federal Register (52 FR 46136) on December 4, 1987, and consulted with the State of Mississippi. No public comments or requests for hearing were received, and the State of Mississippi did not have any comments.

The staff has 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 the security nor to the health and safety of the public.

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Dated: December 30, 1987

AMENDMENT NO. 41 TO FACILITY OPERATING LICENSE NO. NPF-29 - GRAND GULF, UNIT 1

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