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.

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 ≤ 1125 psig to ≤ 1095 psig and the allowable value change from ≤ 1140 psig to ≤ 1102 psig, the staff finds that these changes are in the conservative direction. Since normal scram setpoint for reactor vessel high pressure is ≤ 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 Standby liquid control system (SLCS).

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.

Standard 2. 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.

Principal Contributors: Hulbert C. Li, L. L. Kintner, M. McCoy
Jin-Sien Guo

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

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