BEFORE THE

UNITED STATES NUCLEAR REGULATORY COMMISSION

LICENSE NO. NPF-29

DOCKET NO. 50-416

IN THE MATTER OF

MISSISSIPPI POWER & LIGHT COMPANY
and
SYSTEM ENERGY RESOURCES, INC.
and
SOUTH MISSISSIPPI ELECTRIC POWER ASSOCIATION
and
ENTERGY OPERATIONS, INC.

AFFIRMATION

I, L. F. Daughtery, being duly sworn, state that I am the Acting Director, Nuclear Safety and Regulatory Affairs, GGNS, of Entergy Operations, Inc.; that on behalf of Entergy Operations, Inc., System Energy Resources, Inc., and South Mississippi Electric Power Association I am authorized by Entergy Operations, Inc. to sign and file with the Nuclear Regulatory Commission, this application for amendment of the Operating License of the Grand Gulf Nuclear Station; that I signed this application as the Acting Director, Nuclear Safety and Regulatory Affairs, GGNS, of Entergy Operations, Inc.; and that the statements made and the matters set forth therein are true and correct to the best of my knowledge, information and belief.

L. F. Daughtery

STATE OF MISSISSIPPI
COUNTY OF CLAIBORNE

SUBSCRIBED AND SWORN TO before me, a Notary Public, in and for the County and State above named, this 4th day of August, 1995.

(SEAL)

Richard R. Moore III
Notary Public

MISSISSIPPI STATEWIDE NOTARY PUBLIC
MY COMMISSION EXPIRES JUNE 5, 1998
BONDED THRU STEGALL NOTARY SERVICE
My commission expires:

PROPOSED CHANGE TO THE OPERATING LICENSE

FUEL HANDLING ACCIDENT OPERATIONAL CONDITIONS

GGNS PCOL 93/08 Revision 1

A. AFFECTED TECHNICAL SPECIFICATIONS

The following are the Technical Specifications affected by the proposed change.

Limiting Condition for Operation (LCO)		Affected pages
3.3.6.1	Primary Containment and Drywell Isolation Instrumentation	3.3-51, 3.3.-52, and 3.3-55
3.3.6.2	Secondary Containment Isolation Instrumentation	3.3-62
3.3.7.1	Control Room Fresh Air (CRFA) System Instrumentation	3.3-76
3.6.1.3	Primary Containment Isolation Valves (PCIVs)	3.6-13
3.6.4.1	Secondary Containment	3.6-42 and 3.6-43
3.6.4.2	Secondary Containment Isolation Valves (SCIVs)	3.6-45 and 3.6-47
3.6.4.3	Standby Gas Treatment (SGT) System	3.6-49 and 3.6-50
3.7.3	CRFA System	3.7-6, 3.7-7, and 3.7-8
3.7.4	Control Room AC System	3.7-9, 3.7-10, and 3.7-11
3.8.2	AC Sources - Shutdown	3.8-18, 3.8-19, and 3.8-20
3.8.5	DC Sources - Shutdown	3.8-31, 3.8-32, and 3.8-33
3.8.8	Distribution Systems - Shutdown	3.8-40

Associated Technical Specification Bases changes to be implemented following NRC approval of the proposed Technical Specification changes are detailed in Attachment 3.

B. DISCUSSION:

Following reactor shutdown, decay of the short-lived fission products greatly reduces the fission product inventory present in irradiated fuel. Prior to implementation of the Improved Technical Specifications (Amendment 120 to the Operating License) the Technical Specifications in LCO 3.9.4 required a 24 hour period of reactor subcriticality prior to fuel movement. The implementation of the Improved Technical Specifications relocated this restriction to licensee control under the change controls of 10CFR50.59. This delay is an assumption in the fuel handling accident analysis. The proposed changes are based on a longer decay period (12 days) and takes advantage of the reduced radionuclide inventory available for release in the event of a fuel handling accident. The proposed changes redefine the operability requirements for selected engineered safety feature (ESF) systems such that these systems are only required to be operable during this 12 day decay period.

Implementation of the proposed changes will have a significant impact on outage activities at Grand Gulf Nuclear Station (GGNS) resulting in reduced outage costs and increased flexibility with no impact on safety margin. Currently, moving large equipment into secondary containment such as chemical-deacontamination equipment or safety-relief valves must either be delayed or moved through an alternate entrance.

Additional, the high level of modification work, maintenance, and repair activities during outages, results in wear and tear on access doors to secondary containment and causes the doors to frequently break down which creates a bottle-neck situation for processing personnel and equipment in and out of the radiological control area. When this occurs, contract personnel must be rerouted through other, less convenient access paths. Also, additional door guards are typically employed during outages to ensure that incidents of doors being left open are minimized. These factors coupled with the increased flexibility for scheduling testing and maintenance activities on secondary containment valves, dampers, and instrumentation can result in accrued cost reductions in excess of $500,000 over the remaining operating life of the plant and allow outage resources to be directed elsewhere.

Original License Basis

The fuel handling accident in the auxiliary building is evaluated in the GGNS Updated Final Safety Analysis Report (UFSAR) Section 15.7.4. The design basis analysis is based on the Standard Review Plan (SRP) 15.7.4 and Regulatory Guide (RG) 1.25. The limiting event is the drop of a channeled irradiated fuel assembly onto stored spent fuel bundles. The cause of this event is a failure of the fuel assembly lifting mechanism. The radioactive release causes high radiation signals to isolate the normal ventilation system and initiate the standby gas treatment system. The Technical Specifications define operability, closure times, and surveillance intervals for the fuel handling area ventilation exhaust and pool sweep radiation monitors, the SGT System, and the secondary containment automatic isolation damper/valves and the associated electrical power systems. These systems limit the transport of fission products to the environment such that the radiological effects at the Site Boundary are approximately 1.7 rem whole body and 2.3 rem thyroid.

The fuel handling accident in the containment is evaluated in the UFSAR Section 15.7.6. The design basis analysis is also based on the SRP 15.7.4 and RG 1.25. The limiting event is the drop of an irradiated fuel assembly onto the reactor core with the containment equipment hatch open to the secondary containment. The cause of this event is a failure of the fuel assembly lifting mechanism. The UFSAR evaluates two cases for transporting the radioactivity released from containment to the environment: 1) The activity released from the fuel is conservatively assumed to be completely pulled through the open equipment hatch into the auxiliary building; and 2) The activity released from the fuel is partially pulled through the equipment hatch into the auxiliary building and partially pulled through the containment ventilation system to the environment. For case 2, the radiological effects at the Site Boundary are approximately 1.7 rem whole body and 2.5 rem thyroid. Case 1 is slightly less.

For the fuel handling accidents described above, secondary containment integrity, isolation of the containment and fuel handling area ventilation systems, working in conjunction with the SGT System limit the transport of fission products to the environment and the associated radiological consequences (per SRP 15.7.4 guidelines) to well within the 10CFR100.11 limits. The SRP further defines the fuel handling accident limits as 75 rem thyroid and 6 rem whole body. Because these systems are directly related to mitigating the release of radioactive material and are part of the

primary success path for the design basis fuel handling accident, appropriate operating restrictions are imposed by the Technical Specifications.

Loads in excess of 1140 pounds (heavy loads) are prohibited from traveling over spent fuel assemb.ies in the spent fuel or upper containment fuel storage pool racks. Without appropriate controls, loads weighing less than 1140 pounds (light loads) of sufficient impact energy could result in exceeding the SRP 15.7.4 dose limitations if dropped on irradiated fuel assemblies. This issue was identified via LER-88/016-1 (AECM-89/0025) dated February 1, 1989 (Final Report). The resolution to this LER established administrative controls that involve height/weight limits that control the impact energy of light loads to assure that, in the unlikely event of a drop over irradiated fuel, offsite radiological consequences would be limited to the SRP 15.7.4 limits. This proposed amendment applies the same criteria to establish controls on the handling of recently irradiated fuel, thereby allowing irradiated fuel and "light loads" to be controlled on the same basis.

Reanalysis of Fuel Handling Accident

Entergy Operations, Inc. recently reanalyzed the Fuel Handling Accidents for GGNS. The reanalysis was performed to incorporate ICRP 30 dose conversion factors, consideration of the drop of the fuel handling tool, updated atmospheric dispersion factors (χ/Q factors), and the impact of not crediting various engineered safety feature (ESF) systems that are currently used to reduce the consequences of the analyzed events.

Implementation of the ICRP 30 dose conversion factors for the GGNS Loss of Coolant Accident (LOCA) analysis, revised χ/Q factors, and other significant dose calculation methodology and assumption changes were recently incorporated into the GGNS license basis under 10CFR50.59. Entergy presented the new license basis during a meeting with the NRC Staff on April 6, 1993.

The ICRP 30 conversion factors are based on additional empirical information and improved understanding of radiation effects. Atmospheric dispersion values (χ/Q factors) are based on meteorological conditions in the area surrounding the site. GGNS has been gathering this data since before the plant began operation and has updated the various factors which rely on that data. The major effect of revised meteorological data is on the χ/Q factors which become slightly less favorable.

The reanalysis evaluated the Fuel Handling Accident in the auxiliary building and in containment. Precursors for these events are unchanged from that described in the UFSAR; however, the analysis was expanded to evaluate the effects of various decay time periods beyond 24 hours in conjunction with the assumption that those systems used to mitigate the accident as described in the UFSAR are not available.

The analysis approach was to calculate the whole body and thyroid dose due to a single fuel rod failure. Using the dose due to a single fuel rod failure and the regulatory dose limits (75 rem thyroid, 6 rem whole body), the maximum number of rod failures which could occur without exceeding the dose limits was calculated. Based on the required

impact energy to the cladding which could result in a fuel rod failure, height/weight limitations were developed for various decay periods.

The analysis demonstrated that for the worst case drop the regulatory dose limitations of SRP 15.7.4 are satisfied for decay periods of 12 days or more without credit for the ESF systems discussed above. On or before the 12th day following shutdown, the thyroid dose proved to be limiting with a postulated thyroid dose exceeding the 75 rem limit.

Key assumptions used in the analysis are as follows:

- Regulatory Guide 1.25 [Ref. 6] assumptions are followed with the exception that the ICRP 30 dose conversion factors are used for thyroid dose and whole body dose calculations.

- In accordance with NUREG/CR-5009, a release fraction of 12% was applied to 1-131 for extended burnup fuel.

- Credit for 12 months decay time for GE fuel previously discharged.

- It is conservatively assumed that the 3" and 4" sections of the NF500 telescoping mast and the handling tool fall from the top of the water level.

- Impact energy associated with a struck fuel assembly is absorbed in the entire volume of the fuel rod cladding. No credit is taken for non-cladding items such as tie-plates, water rods, or fuel pellets.

- Credit is taken for the buoyancy force on the dropped object. No credit is taken for drag force on the dropped object.

- All fuel rods of a dropped bundle are assumed to fail due to bending (i.e., no credit is taken for lateral support provided by a fuel channel).

- Atmospheric dispersion factors (X/Q), developed using approved methodologies from Standard Review Plan Section 2.3.4, Rev. 1, are used in the radiological assessment.

- Per RG 1.25, all of the gap fission product inventory is released after a cladding failure. This gap inventory, based on the fraction of the total fission products, is as follows:

    - 10% of the noble gases (excluding Kr-85)
    - 30% of the Kr-85 inventory
    - 10% of the Iodine inventory

- Per RG 1.25, the activity released to the containment/auxiliary building is based on an overall decontamination factor of 100 for Iodine for the first 23 feet of water coverage (additional decontamination for water depths of 46 feet or greater is credited) and a decontamination factor of one for the noble gases (no noble gases are retained in the pool).

- Per RG 1.25, the fission products released to the containment/auxiliary building escape to the environment within two hours.

The limiting drop without accident mitigating functions is within the dose limitations for decay periods of 12 days or more. Therefore, the bounding decay period of 12 days was chosen as the basis for the proposed Technical Specifications changes. Based on these results operability requirements and action requirements were established for the LCOs identified in Section A.

Proposed Changes

This proposed amendment to the GGNS Technical Specifications revises those specifications associated with handling irradiated fuel in the primary or secondary containment and revises certain specifications related to performing CORE ALTERATIONS with irradiated fuel in the reactor pressure vessel. The purpose is to establish a point where operability of those systems typically used to mitigate the consequences of a fuel handling accident is no longer required to meet the current license basis offsite dose limitations (75 rem thyroid, 6 rem whole body).

Specifically, the proposal uses a new term for irradiated fuel that contains sufficient fission products to require operability of accident mitigation systems to meet the accident analysis assumptions and revises the operability requirements for the associated ESF systems. The proposed changes are summarized below.

1) The Applicabilities for the following LCOs are modified from "when handling irradiated fuel assemblies" to "when handling recently irradiated fuel assemblies". Also, the LCO Conditions and Required Actions are modified to reflect the change in the LCO Applicabilities.

   3.3.6.1 Primary Containment and Drywell Isolation Instrumentation
   3.3.6.2 Secondary Containment Isolation Instrumentation
   3.3.7.1 CRFA System Instrumentation
   3.6.1.3 PCIVs (due to changes in the Applicability of LCO 3.3.6.1)
   3.6.4.1 Secondary Containment
   3.6.4.2 SCIVs
   3.6.4.3 SGT System
   3.7.3 CRFA System
   3.7.4 Control Room AC System
   3.8.2 AC Sources - Shutdown
   3.8.5 DC Sources - Shutdown
   3.8.8 Distribution Systems - Shutdown

   Additionally, the Bases for these LCOs will be updated to reflect the requirements (see Attachment 3).

2) The Applicabilities for the following LCOs are modified to no longer require the LCO to be met during CORE ALTERATIONS. Also, the LCO Conditions and Required Actions are modified to reflect the change in the LCO Applicabilities.

3.3.6.1	Primary Containment and Drywell isolation Instrumentation
3.3.6.2	Secondary Containment Isolation Instrumentation
3.3.7.1	CRFA System Instrumentation
3.6.1.3	PCIVs (due to changes in the Applicability of LCO 3.3.6.1)
3.6.4.1	Secondary Containment
3.6.4.2	SCIVs
3.6.4.3	SGT System
3.7.3	CRFA System
3.7.4	Control Room AC System

Additionally the Bases for these LCOs will be updated to reflect the requirements (see Attachment 3).

C. PLANT SPECIFIC JUSTIFICATION

The use of the term "recently irradiated fuel" provides a mechanism for applying a cutoff in fission product decay to various specifications where the concept applies. The 12 day period to be discussed in the Technical Specification Bases has been shown by analysis to provide sufficient decay such that, assuming the design basis fuel handling accident, radiological consequences are within the acceptance criteria of SRP 15.7.4 [Ref. 5] and General Design Criteria 19 [Ref. 3].

The revised Technical Specification requirements incorporate the term "recently irradiated" to establish operational conditions where specific activities represent situations where significant radioactive releases can be postulated. During MODE 4 or 5, these are:

1. During handling of recently irradiated fuel in the primary or secondary containment.

2. During operations with a potential for draining the reactor vessel.

The revised requirements redefine the LCOs' applicability for instrumentation and devices that isolate containment and provide for filtration systems that mitigate the radiological impact of fuel handling accidents. The proposed applicability is consistent with the fuel handling accident assumptions.

As described in the UFSAR [Ref. 2], the accidents postulated to occur during core alterations in addition to fuel handling accidents are: inadvertent criticality due to a control rod removal error or continuous control rod withdrawal error during refueling and the inadvertent loading and operation of a fuel assembly in an improper location. These events are not postulated to result in fuel cladding integrity damage. Since the only accident postulated to occur during CORE ALTERATIONS that results in a significant radioactive release is the fuel handling accident, the proposed Technical Specification requirements omitting CORE ALTERATIONS is justified. The LCO applicabilities related

to operations with a potential for draining the reactor vessel is unaffected by the proposed changes.

Supplemental Plant Specific Analysis Justification

The proposed changes to the Technical Specifications are based, for the most part, on reanalysis of the GGNS fuel handling accident. Each of the assumptions used in the analysis as well as the methodology are notably conservative relative to the conditions typically present when fuel handling occurs. In addition to the conservative assumptions of RG 1.25, additional conservatism is included in the current GGNS reanalysis of this event.

- For fuel assemblies struck by a dropped object, it is assumed that the impact energy is absorbed by the fuel rod cladding. No credit is taken for non-clad items such as tie plates, water rods, or fuel channels. In addition, all of the impact energy is dissipated by failing fuel rods. In other words, each failed fuel rod only absorbs the minimum amount of energy to cause it to fail, thereby, maximizing the number of failures.

- The analysis assumes that the handling tool falls from the top of the water level in containment. This represents a significant fraction (approximately 40%) of the impact energy that is absorbed by the impacted fuel.

- Although the containment ventilation system has charcoal filters, no credit is taken for iodine removal.

- No credit is taken for irradiation strengthening of the 9x9-5 fuel cladding. The yield strength of the cladding increases by approximately 10% after only 10 days of irradiation. Although no specific analysis was performed, it is anticipated that, for periods shorter than 10 days, the low fission product buildup would be more than enough to compensate for the reduced clad strength.

Each of the above conservatisms has a significant impact on the radiological consequences of a fuel handling accident even considering the worst case assumptions imposed by RG 1.25.

Supplemental Plant Specific Shutdown Risk Justification

The containment and associated engineered safety feature systems are only required by the Technical Specifications during the specific activities which are postulated to result in a significant release of radioactivity (e.g., fuel handling accident, draindown). As a result, the Technical Specifications requirements are based on the plant being in specified conditions and are not based on providing requirements associated with shutdown risk considerations. Shutdown risk issues are addressed by utility outage management which follows the guidance of NUMARC 91-06, Guidelines For Industry Actions To Assess Shutdown Management [Ref. 7]. NUMARC 91-06 Section 4.5 discusses the need to assure that secondary containment, for GGNS, closure can be achieved to prevent fission product release during severe accidents. NUMARC 91-06 also identifies that the time to effect

closure should be consistent with plant conditions (e.g., reactor coolant system inventory and decay heat load). Consistent with the industry's commitment in the letter from NUMARC's President, Mr. Byron Lee, Jr., to Mr. James M. Taylor of the NRC [Ref. 8], GGNS has administrative controls in place to meet the recommendations of NUMARC 91-06 Section 4.5 for extended loss of decay heat removal events.

Also, in accordance with Technical Specification 3.9.6, RPV Water Level - Irradiated Fuel, handling irradiated fuel in the reactor vessel can only occur when the water level in the reactor cavity is at the high water level the proposed changes only affect containment requirements during relatively low risk times during refueling outages. Therefore, the proposed changes do not significantly increase the shutdown risk.

Additionally, the proposed Technical Specification changes do not affect the requirements to have the containment systems available any time the unit is in MODE 1, 2, or 3 regardless of whether fuel handling is occurring in the spent fuel pool.

D. JUSTIFICATION FOR GENERIC CHANGES

Following reactor shutdown, decay of the short-lived fission products greatly reduces the fission product inventory present in irradiated fuel. Previously, the Technical Specifications contained requirements which were developed to support fuel handling accident analyses which assumed a 24-hour period of reactor subcriticality prior to fuel movement. The proposed changes are based on performing additional analysis assuming a longer decay period to more realistically model the reduced radionuclide inventory available for release in the event of a fuel handling accident.

The proposed changes redefine the operability requirements for selected engineered safety feature (ESF) systems such that these systems are only required to be operable during the time frame that the limiting fuel handling accident analysis takes credit for them to function to meet the regulatory dose limits. The analytical approach to justify these Technical Specification changes is to calculate the whole body and thyroid doses due to a fuel handling accident at various time intervals. The purpose is to establish a point where operability of those systems typically used to mitigate the consequences of a fuel handling accident is no longer required to meet the regulatory dose limits (75 rem thyroid, 6 rem whole body).

The analyses performed to justify these changes must demonstrate that the regulatory dose limitations of SRP 15.7.4 are satisfied without credit for the ESF systems whose operability requirements are being relaxed.

The proposed changes also deletes operability requirements for dose mitigation systems to be operable during CORE ALTERATIONS. This change is justified since accidents postulated to occur during core alterations in addition to the fuel handling accident are: inadvertent criticality due to a control rod removal error or continuous control rod withdrawal error during refueling and the inadvertent loading and operation of a fuel assembly in an improper location. These events are not postulated to result in fuel cladding integrity damage. Since the only accident postulated to occur during CORE ALTERATIONS that results in a significant radioactive release is the fuel handling

accident, the proposed Technical Specification requirements omitting CORE ALTERATIONS is justified. The applicability related to operations with a potential for draining the reactor vessel is unaffected by the proposed changes.

To implement this Technical Specification change the requesting utility must identify that shutdown risk issues are addressed by utility outage management which follows the guidance of NUMARC 91-06, Guidelines For Industry Actions To Assess Shutdown Management [Ref. 7]. Including NUMARC 91-06 Section 4.5 which discusses the need to assure that primary/secondary containment closure can be achieved to prevent fission product release during severe accidents and identifies that the time to effect closure should be consistent with plant conditions (e.g., reactor coolant system inventory and decay heat load). The utility must also identify that consistent with the industry's commitment in the letter from NUMARC's President, Mr. Byron Lee, Jr., to Mr. James M. Taylor of the NRC [Ref. 8], the utility has administrative controls in place to meet the recommendations of NUMARC 91-06 Section 4.5 for extended loss of decay heat removal events.

E. NO SIGNIFICANT HAZARDS CONSIDERATIONS

This proposed amendment to the Grand Gulf Nuclear Station (GGNS) Technical Specifications (TS) revises those specifications associated with handling irradiated fuel in the primary or secondary containment and CORE ALTERATIONS. Specifically, the proposal uses a new term to describe irradiated fuel that contains sufficient fission products to require operability of accident mitigation systems to meet the accident analysis assumptions. This proposed change revises the operability requirements and required actions for the following Technical Specification Limiting Conditions for Operation (LCOs): Primary Containment and Drywell Isolation Instrumentation, Secondary Containment Isolation Instrumentation, CRFA System Instrumentation, PCIVs, Secondary Containment, SCIVs, SGT System, CRFA System, Control Room AC System, AC Sources - Shutdown, DC Sources - Shutdown, and Distribution Systems - Shutdown.

The Commission has provided standards for determining whether a no significant hazards consideration exists as stated in 10CFR50.92(c). A proposed amendment to an operating license involves a 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.

Entergy Operations Inc. has evaluated the no significant hazards considerations in its request for a license amendment. In accordance with 10CFR50.91(a), Entergy Operations Inc. is providing the analysis of the proposed amendment against the three standards in 10CFR50.92(c). A description of the no significant hazards considerations determination follows:

1. The proposed changes do not significantly increase the probability or consequences of an accident previously evaluated.

   A new term to describe irradiated fuel is used to establish operational conditions where specific activities represent situations where significant radioactive releases can be postulated. These operational conditions are consistent with the design basis analysis. Because the equipment affected by the revised operational conditions is not considered an initiator to any previously analyzed accident, inoperability of the equipment cannot increase the probability of any previously evaluated accident. The proposed requirements in conjunction with existing administrative controls on light loads, bounds the conditions of the current design basis fuel handling accident analysis which concludes that the radiological consequences are within the acceptance criteria of NUREG 0800, Section 15.7.4 and General Design Criteria 19. Therefore, the proposed changes do not significantly increase the probability or consequences of any previously evaluated accident.

   Based on the above, the proposed changes do not significantly increase the probability or consequences of any accident previously evaluated.

2. The proposed changes would not create the possibility of a new or different kind of accident from any previous analyzed.

   The new term to describe irradiated fuel is used to establish operational conditions where specific activities represent situations where significant radioactive releases can be postulated. These operational conditions are consistent with the design basis analysis. The proposed changes do not introduce any new modes of plant operation and do not involve physical modifications to the plant. Therefore, the proposed changes do not create the possibility of a new or different kind of accident from any previous analyzed.

   Therefore, the proposed changes do not create the possibility of a new or different kind of accident from any accident previously analyzed.

3. The proposed changes do not involve a significant reduction in a margin of safety.

   The new term to describe irradiated fuel is used to establish operational conditions where specific activities represent situations where significant radioactive releases can be postulated. These operational conditions are consistent with the design basis analysis and are established such that the radiological consequences are at or below the current GGNS licensing limit. Safety margins and analytical conservatisms have been evaluated and are well understood. Substantial margins are retained to ensure that the analysis adequately bounds all postulated event scenarios. The proposed change only eliminates the excess margin from the analysis. The current margin of safety is retained.

Specifically, the margin of safety for the fuel handling accident is the difference between the 10CFR100 limits and the licensing limit defined by NUREG 0800, Section 15.7.4. With respect to the control room personnel doses, the margin of safety is the difference between the 10CFR100 limits and the licensing limit defined by 10CFR50, Appendix A, Criterion 19 (GDC 19). Excess margin is the difference between the postulated doses and the corresponding licensing limit.

The proposed applicability continues to ensure that the whole-body and thyroid doses at the exclusion area and low population zone boundaries as well as control room, doses are at or below the corresponding licensing limit. The margin of safety is unchanged; therefore, the proposed changes do not involve a significant reduction in a margin of safety.

Therefore, the proposed changes do not result in a significant reduction in a margin of safety.

Based on the above evaluation, operation in accordance with the proposed amendment involves no significant hazards considerations.

E. REFERENCES

1. Grand Gulf Nuclear Station Unit 1 Technical Specifications and Bases, Updated through Amendment 120.

2. Grand Gulf Nuclear Station Final Safety Analysis Report, Updated through Revision 7, Chapter 15.

3. 10CFR50, Appendix A, General Design Criteria 19.

4. NUREG 1434, Standard Technical Specifications, General Electric BWR/6 Plants, Revision 0, September 29, 1992.

5. NUREG 0800, (Standard Review Plan), Section 15.7.4, "Radiological Consequences of Fuel Handling Accidents," Revision 1, July 1981.

6. Regulatory Guide 1.25, "Assumptions Used for Evaluating the Potential Radiological Consequences of a Fuel Handling Accident in the Fuel Handling and Storage Facility for Boiling and Pressurized Water Reactors", 3/23/72.

7. NUMARC 91-06, Guidelines For Industry Actions To Assess Shutdown Management, December 1991.

8. Letter from Mr. Byron Lee, Jr., President and Chief Executive Officer NUMARC, to Mr. James M. Taylor, Executive Director for Operations U.S. NRC, dated December 6, 1991.

Mark-up of Affected Technical Specifications and Bases

GGNS PCOL 93/08 Revision 1