ENTERGY

Entergy Operations, Inc.
River Bend Station
5485 U.S. Highway 61
P.O. Box 220
St. Francisville, LA 70775
(504) 381-4374
FAX (504) 381-4872

JOHN R. McGAHA, JR.
Vice President
Operations

August 17, 1995

U. S. Nuclear Regulatory Commission
Document Control Desk
Mail Station P1-37
Washington, DC 20555

Subject: River Bend Station - Unit 1
Docket No. 50-458
License No. NPF-47
License Amendment Request (LAR) 95-04, Change to Technical Specifications Concerning Fuel Handling Accident Conditions

File Nos.: G9.5, G9.42

RBEXEC-95-122
RBF1-95-0174
RBG-41728

Gentlemen:

In accordance with 10CFR50.90, Entergy Operations, Inc. (EOI) hereby applies for amendment of Facility Operating License No. NPF-47, Appendix A - Technical Specifications, for River Bend Station (RBS). This request consists of a proposed change to Technical Specifications concerning engineered safety systems which respond to Fuel Handling Accident conditions.

The subject request is being submitted as part of the cost beneficial licensing action (CBLA) program established within NRR where increased priority is granted to qualifying licensee requests. Implementation of the proposed changes will have a significant impact on outage activities at RBS resulting in reduced outage costs and increased flexibility while maintaining an acceptable safety margin. The increased flexibility for scheduling testing and maintenance activities on plant equipment can result in accrued cost reductions in excess of $634,000 per outage; of which, $25,000 is direct cost, $252,000 is critical path time, and $357,000 is replacement power. In accordance with NRC Administrative Letter 95-02, replacement power costs are not to be considered in the cost benefit of a requested licensing action; therefore, the

A001

cost benefit for this request is estimated at $277,000 per outage. These costs exceed the threshold of $100,000 over the life of the plant established under the CBLA program. These cost savings are expected to benefit all remaining refueling outages of the plant due to the continued reduction in resources and increased efficiencies this change allows.

Enclosure 2 provides a description of the proposed changes and the associated justification (including a Basis For No Significant Hazards Consideration). Enclosure 3 contains a table of the affected Technical Specifications and summary of the associated changes. A marked-up and revised copy of the affected pages from the Improved Technical Specifications (reference EOI letter RBG-41632, Rev. 2 to LAR 93-14, dated June 30, 1995), as approved in Amendment 81 dated July 20, 1995, is provided in Enclosure 4. Enclosure 1 is an affidavit supporting the facts set forth in this letter and enclosures. This request has been reviewed and approved by the RBS Facility Review Committee and the Nuclear Review Board.

As discussed in a meeting between representatives of the BWR6s and the NRC on July 20, 1995, this request will have minimal impact to shutdown risk. This evaluation is based on the following: 1) the proposed changes will not remove requirements for systems to mitigate potential vessel drain-down events (ECCS); 2) the proposed changes will not remove requirements for systems required for decay heat removal (DHR/RHR); 3) the loss of DHR has been identified as a concern prior to reaching high water level over the vessel; and 4) the proposed changes require this high water level. In addition to the information presented in the July 20, 1995 meeting, River Bend has programs in place to effectively close the containment resulting in additional protection. Therefore, the requested changes can be implemented and not preclude effective actions to address shutdown risk concerns.

EOI has reviewed the proposed change against the criteria of 10CFR51.22 for categorical exclusion from environmental impact considerations. The proposed change does not involve a significant hazards consideration nor does it significantly increase individual or cumulative occupational radiation exposures. Based on the foregoing, EOI concludes that the proposed change meets the criteria given in 10CFR51.22(c)(9) for a categorical exclusion from the requirement for an Environmental Impact Statement.

Based upon the refueling outage improvement and significant resource savings that can be realized by implementing this proposed change, EOI is requesting that this application be reviewed on a schedule sufficient to support the sixth refueling outage (RF-6) currently scheduled to begin January 6, 1996.

If you have any questions regarding this request or require additional information, please contact me or my staff.

Sincerely,

JRM/BMB/jr
enclosures

xc: Mr. David L. Wigginton
U. S. Nuclear Regulatory Commission
11555 Rockville Pike
M/S OWFN 13-H-15
Rockville, MD 20852

NRC Resident Inspector
P. O. Box 1051
St. Francisville, LA 70775

U. S. Nuclear Regulatory Commission
Region IV
611 Ryan Plaza Drive, Suite 400
Arlington, TX 76011

Louisiana Department of Environmental Quality
Radiation Protection Division
P. O. Box 82135
Baton Rouge, LA 70884-2135
Attn: Administrator

BEFORE THE

UNITED STATES NUCLEAR REGULATORY COMMISSION

LICENSE NO. NPF-47

DOCKET NO. 50-458

IN THE MATTER OF

GULF STATES UTILITIES COMPANY

CAJUN ELECTRIC POWER COOPERATIVE AND

ENTERGY OPERATIONS, INC.

AFFIRMATION

I, John R. McGaha, state that I am Vice President-Operations of Entergy Operations, Inc., at River Bend Station; that on behalf of Entergy Operations, Inc., I am authorized by Entergy Operations, Inc. to sign and file with the Nuclear Regulatory Commission, this License Amendment Request, (LAR) 95-04, Change to Technical Specification Concerning Fuel Handling Accident Conditions; that I signed this request as Vice President-Operations at River Bend Station 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.

John R. McGaha

John R. McGaha

STATE OF LOUISIANA
WEST FELICIANA PARISH

SUBSCRIBED AND SWORN TO before me, Notary Public, commissioned and qualified in and for the Parish and State above named, this 17th day of August, 1995.

(SEAL)

Claudia F. Hurst

Claudia F. Hurst
Notary Public

My Commission expires with life.

ENCLOSURE 2

ENTERGY OPERATIONS INCORPORATED
RIVER BEND STATION
DOCKET 50-458/LICENSE NO. NPF-47

FUEL HANDLING ACCIDENT OPERATIONAL CONDITIONS

(LAR 95-04)

DOCUMENT INVOLVED: Technical Specifications

ITEMS: Technical Specifications affected are listed in Enclosure 3. The specifications identify related changes to those previously submitted as the Improved Technical Specifications (LAR 93-14). Enclosure 4 contains the markup of the Improved Technical Specifications as approved in Amendment 81 dated July 20, 1995.

Reason for Request

Following reactor shutdown, decay of the short-lived fission products greatly reduces the fission product inventory present in irradiated fuel. The current restriction of a 24-hour period of reactor subcriticality prior to any fuel movement in the vessel, an assumption in the existing fuel handling accident analysis (FHA), is maintained. The proposed changes are based on fuel cycles through 2000 and a longer decay period which take advantage of the reduced radionuclide inventory available for release in the event of a fuel handling accident. The longer decay period is currently calculated to be 11 days. The previous limits remain in effect prior to reaching this decay time. 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 calculated decay period. Currently the affected systems are: Isolation Actuation, Radiation Monitoring, Primary Containment, Secondary Containment, Fuel Building Ventilation, Standby Service Water System, Main Control Room Air Conditioning System and AC Sources, DC Sources & Distribution during shutdown.

Similar requests have been proposed by other Entergy plants -- Grand Gulf Nuclear Station on November 9, 1994, as revised on August 4, 1995 and Arkansas Nuclear One on May 19, 1995. The use of extended decay time in evaluating the effects of a fuel handling accident has recently been accepted by the NRC in Amendments 194 & 171 for Calvert Cliffs Units 1 & 2.

As discussed in a meeting between representatives of the BWR6s and the NRC on July 20, 1995, this request will have minimal impact to shutdown risk. This is based on: 1) the proposed changes will not remove requirements for systems to mitigate potential vessel drain-down events (ECCS); 2) the proposed changes will not remove requirements for systems

required for decay heat removal (DHR/RHR); 3) the loss of DHR has been identified as a concern prior to reaching high water level over the vessel; and 4) the proposed changes continue to require this high water level during fuel movement. In addition to the information presented in July 20, 1995 meeting, River Bend has programs in place to effectively close the containment resulting in additional protection. Therefore, the requested changes can be implemented and not preclude effective actions to address shutdown risk concerns.

Implementation of the proposed changes will have a significant impact on outage activities at River Bend Station (RBS) resulting in reduced outage costs and increased flexibility while maintaining an acceptable safety margin. Currently, moving large equipment such as chemical-decon equipment or safety-relief valves into containment must be scheduled to minimize lost time and is expected to affect the critical path of the outage at least twice. This results in work being delayed or rescheduled to less efficient times in an outage. Because of the high level of modification, maintenance, and repair activities during outages, wear and tear on the two airlock doors to containment causes the doors to break down resulting in increased repair cost. These repairs also create a bottle-neck situation for processing personnel and equipment in and out of the containment and drywell, including rerouting through the remaining door further delaying work. In addition, the actual establishment of the containment boundary an estimated two times per outage further restricts access and requires additional resources.

Present estimates of outage cost per hour, when on the critical path, are approximately \$12,000/hr, in addition to replacement power cost of approximately \$17,000/hr. Coupled with increased flexibility for scheduling testing and maintenance activities on primary containment valves, instrumentation and secondary containment (Fuel Building) dampers and instrumentation, these factors can result in accrued cost reductions in excess of \$634,000 per outage, of which \$25,000 is direct cost, \$252,000 is critical path time and \$357,000 is replacement power. In accordance with NRC Administrative Letter 95-02, replacement power costs are not to be considered in the cost benefit of a requested licensing action; therefore, the cost benefit for this request is estimated at \$277,000 per outage. This cost savings is expected to benefit all remaining refueling outages of the plant due to the continued reduction in resources and increased efficiencies this change allows.

Original License Basis

The original design basis analysis (DBA) evaluated a number of events and locations. This analysis is based on guidance in the NRC Standard Review Plan (SRP) 15.7.4 and Regulatory Guide (RG) 1.25. The event which results in the largest postulated number of failed fuel rods (125) is the drop of a bundle into the reactor core with the vessel head off. However the RBS configuration in this event is with Primary Containment isolated per Technical Specification 3.6.1.10, "Primary Containment-Shutdown." As discussed in SRP 15.7.4, no radiological consequences need be calculated due to the low leakage of the Primary Containment (0.26 % per day). The results of an FHA in secondary containment (Fuel Building) determined the limiting event as a drop of a channeled irradiated fuel assembly onto stored spent fuel bundles.

This event results in a maximum of 123 fuel pins failed and comparable radiological effects. Because an FHA in the Fuel Building results in comparable fuel damage and a higher radiological release, the event was used for the initial licensing of the plant. This event is evaluated in the RBS Safety Analysis Report (SAR) Section 15.7.4, Radiological Consequences of Fuel Handling Accidents.

As described in SAR 15.7.4, the cause of this event is a failure of the fuel assembly lifting mechanism. The RBS TS 3.3.6.2, 3.6.4.5 and 3.6.4.7, requires the Fuel Building secondary containment and associated isolation be OPERABLE and the Fuel Building filtration to be operating during irradiated fuel movement; therefore any release is filtered in accordance with guidance in RG 1.25. The TSs define operability, closure times, and surveillance intervals for the required systems. These systems limit the transport of fission products to the environment such that the radiological effects at the Site Boundary are approximately 1.3 rem whole body and 2.1 rem thyroid with a main control room effect of 0.07 rem whole body and 3.9 rem thyroid, as documented in SAR 15.7.4 Table 15.7-13. Therefore, the radiological consequences, determined in accordance with SRP 15.7.4 guidelines, are well within the 10CFR100.11 limits (e.g., less than 75 rem thyroid and 6 rem whole body). As noted above, the FHA in the Primary Containment was not specifically evaluated, per allowance in SRP 15.7.4. In a later submittal dated September 28, 1988, supplemented November 30, 1988, and February 6, 1989, RBS requested additional vent and drain lines to be open for 10CFR50 Appendix J valve leak rate testing. This request was approved in Amendment 35 dated March 3, 1989.

Loads in excess of 1200 pounds (heavy loads) are controlled by load restrictions in the Technical Requirements Manual from traveling over spent fuel assemblies in the spent fuel or upper containment fuel storage pool racks. With administrative controls in place, loads weighing less than 1200 pounds (light loads) will not result in exceeding the SRP 15.7.4 dose guidelines if dropped on irradiated fuel assemblies. Presently established administrative controls impose height/weight limits to control the impact energy of light loads. These limits assure that, in the unlikely event of a drop of less than 1200 pounds over irradiated fuel, offsite radiological consequences will not exceed SRP 15.7.4 guidelines. This proposed amendment does not change this restriction or the effects of previously evaluated events.

Methodology of Analysis

The original design basis method used at RBS employed the Stone & Webster DRAGON4 code for the FHA in the Fuel Building. This original analysis was reviewed in NUREG-0989, RBS Safety Evaluation Report (SER) Section 15.7.4. This method was also employed for the September 1988 submittal allowing containment vent and drain lines to be opened as noted above. RBS has recently implemented the TRANSACT code, used as the standard Entergy code for offsite dose and control room dose calculations due to postulated events. The TRANSACT code is derived from the TACT V code documented in NUREG/CR-5109. RBS has benchmarked the TRANSACT code. This benchmarking used the analysis of the

September 1988 submittal under DRAGON4 and a similar analysis under TRANSACT. The guidance and assumptions of SRP 15.7.4, RG 1.25 and initial licensing basis and revisions accepted in Amendment 35 are maintained in the benchmark, including:

- Primary containment leak rate of 0.26% per day.
- Additional leakage of 70.2 cfm from open LLRT lines.
- DRAGON4 dose conversion factors are used (based on ICRP-2 model).
- Retention of noble gases in the fuel pool is negligible.
- The event occurs 80 hours after shutdown.
- Airborne nuclides released from the pool are mixed and diluted in 50% of containment atmosphere.

ICRP-30 dose conversion factors have also been applied in conjunction with the benchmark, vice the earlier ICRP-2 values. The newer document (ICRP-30) is the result of refinements in the dose effects of radiation. The use of ICRP-30 in lieu of ICRP-2 for work of this type has previously been evaluated at EOI's Grand Gulf Nuclear Station and included in their evaluation of this event. A comparison of the two codes and the ICRP-30 change is contained below identifying the effects in rem on whole body and thyroid for the exclusion area boundary (EAB), low population zone (LPZ) and main control room (MCR).

	DRAGON w/ICRP-2	TRANSACT w/ICRP-2	TRANSACT w/ICRP-30
Whole body			
EAB	1.39E-2	1.23 E-2	3.55 E-3
LPZ	2.56E-2	2.49 E-2	7.12 E-3
MCR	8.10E-3	1.64 E-2	4.52 1E-3
Thyroid			
EAB	2.03	1.99	1.45
LPZ	2.99	2.99	2.19
MCR	5.63E-1	5.5 E-1	4.03 E-1

This benchmarking demonstrates excellent agreement between the codes for all values but the MCR gamma dose for which TRANSACT uses a more conservative model. The TRANSACT MCR gamma doses are greater than those calculated by DRAGON4 because TRANSACT uses a simplified finite-cloud dose model. The model used by TRANSACT is a simpler model than that used by DRAGON4 but is conservative and bounding. The TRANSACT model is consistent with the model presented by the NRC in the 13th AEC Air Cleaning Conference by

K. G. Murphy and Dr. K. N. Campe. Furthermore, thyroid doses will prove more limiting than whole body doses. Therefore, TRANSACT is acceptable for use in this application.

Reanalysis of Fuel Handling Accident

This request will revise the original design basis of the plant by evaluating a new limiting event: the drop of an irradiated fuel assembly onto the reactor core with the containment equipment hatch open to the environment. The original 24-hour case will also remain as previously analyzed. The cause of this event remains a failure of the fuel assembly lifting mechanism. The new DBA is also based on SRP 15.7.4 and RG 1.25. This analysis evaluates the transport of the radioactivity released as conservatively assumed to be completely pulled through the open primary containment equipment hatch into the environment within two hours, as originally assumed. In this analysis the Main Control Room filters are conservatively assumed to be unavailable; therefore, the operations staff is directly exposed. For this event the limiting offsite radiological effects at the Site Boundary are approximately 0.0823 rem whole body and 46.6 rem thyroid. The radiological effects in the Main Control Room are approximately 2.23 E-3 rem whole body and 24.5 rem thyroid.

Entergy Operations, Inc., recently reanalyzed the FHA for RBS using the TRANSACT code. The reanalysis was performed incorporating ICRP-30 dose conversion factors, consideration of the drop of the fuel handling tool and the impact of not crediting various ESF systems that are currently used to reduce the consequences of the analyzed events. The ICRP-30 dose conversion factors are based on additional empirical information and improved understanding of radiation effects. Atmospheric dispersion values (X/Q factors) are based on meteorological conditions in the area surrounding the site and are unchanged.

This analysis demonstrated that for the worst case bundle drop, the regulatory dose guidelines of SRP 15.7.4 are satisfied for a decay period of 11 days or more without credit for the ESF systems discussed above. On or before the calculated decay period after shutdown for the analysis, the Main Control Room thyroid dose proved to be limiting. Therefore, the ESF systems necessary to support the analysis (including MCR filters and Primary Containment), and all previous limits and TS requirements will remain in effect prior to the limit when handling fuel. RG 1.25 [Ref. 6] assumptions are followed, except that ICRP-30 dose conversion factors are used for thyroid dose and whole body dose calculations and a higher radial peaking factor of 1.65 is assumed. Key assumptions used in the analysis are as follows:

- It is conservatively assumed that three sections of the telescoping mast, and the handling tool, of the Reactor Building refueling bridge fall in accordance with the analysis in GESTAR II. The water level is controlled at 23 feet above the reactor flange, when handling fuel over the reactor, and at 23 feet over spent fuel, as required by TS 3.9.6 and 3.7.6, respectively.

- None of the energy associated with the dropped fuel assembly is absorbed by the fuel material (uranium dioxide).

- Credit is taken for the buoyancy force on the dropped object (spent fuel bundle). No credit is taken for drag force on the dropped object when falling through the water.

- Atmospheric dispersion rate and X/Q values are determined in accordance with RG 1.145, Atmospheric Dispersion Models for Potential Accident Consequence Assessments at Nuclear Power Plants.

- For this analysis and in accordance with 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, corresponds to 10% of the noble gases (excluding Kr-85), 30% of the Kr-85 inventory and 10% of the Iodine inventory. In the Grand Gulf submittal Iodine inventory was calculated in accordance with NUREG/CR-5009. This results in a release fraction of 12% of I-131 for extended burnup fuel. RBS is not currently using extended burnup fuel. The earliest refueling outage that extended burnup fuel will be in the core is the end of Cycle 9 which will not be earlier than the year 2000. The use of extended burnup fuel could impact a number of present design basis analyses. Prior to the use of extended burnup fuel, the FHA analysis would be reviewed and any necessary license changes made prior to fuel handling of any high burnup fuel at that time.

- For this analysis and in accordance with RG 1.25, the activity released to the Primary Containment or Fuel Building is based on an overall decontamination factor of 100 for Iodine for 23 (or more) feet of water coverage above the fuel and a decontamination factor of 1 for the noble gases (no noble gases are retained in the pool). The current analysis did not use decontamination factors greater than 100 for water depths greater than 23 feet.

- Per RG 1.25, the fission products released to the Primary Containment or Fuel Building escape to the environment within two hours.

The precursors for this reanalysis of the FHA in the Fuel Building and Primary Containment are unchanged from those described in the SAR; 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 SAR are not available. In the development of this submittal previous conservatisms contained in the original analysis were also reevaluated. Among the conservative assumptions previously used is the number of fuel pins determined to fail due to mechanical stress and strain. The GE analysis contained in GESTAR II at the time of original licensing determined 101 fuel pins fail. This number was used as a basis in the analysis for the Grand Gulf and Clinton stations. The RBS analysis, contained in Section 15.7.4 of the USAR, assumed 123 pins failed in the Fuel Building event. During the development of this request RBS has confirmed that the initial

conditions and assumptions contained in GESTAR II, Section 2.2.3.5, can be applied to RBS design and operation. Therefore, for this analysis, the revised FHA results include the effects of 104 failed fuel pins. This value differs slightly from the GESTAR II value to account for the use of GE11 fuel bundles starting with Cycle 7.

The current River Bend analysis also includes an additional margin in the limiting location of the Main Control Room dose as a conservatism to address any changes in plant or fuel design in the future. This margin will be used as necessary to maintain the calculated limits of the event below those specificed in Gernal Design Criterion (GDC) 19 and accepted SRP 15.7.4 guidelines.

The current analysis identifies the limiting drop, without accident-mitigating functions, is within the regulatory dose limitations for decay periods of 11 days or more. Based on these results operability requirements were established in the Technical Specifications for Secondary Containment Isolation Instrumentation, Control Room Fresh Air Instrumentation, Primary Containment Airlocks, Secondary Containment Isolation Dampers, Fuel Building, Fuel [illegible]ing Ventilation System-Fuel Handling, Primary Containment-Shutdown, Control Room Fresh Air, Control Room Air Conditioning System, AC and DC sources- Shutdown, Inverters-Shutdown and Distribution Systems - Shutdown. Additional information was included in the BASES for the above specifications and Secondary Containment-Operating and Fuel Building Ventilation-Operating.

Margins and Conservatisms

The proposed changes to the TSs are based on reanalysis of the RBS fuel handling accident and revision of the applicability of CORE ALTERATION requirements. Each of the assumptions used in the analysis, as well as the methodology, are conservative relative to the conditions typically present when fuel handling occurs. In addition to the conservative assumptions of Regulatory Guide 1.25, further conservatism is included in the RBS reanalysis of this event:

- The analysis assumes that the handling tool falls from the top of the water level in containment. This represents a significant fraction (approximately 35 %) of the impact energy that is absorbed by the impacted fuel.

- No credit is taken for irradiation strengthening of the 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.

In addition to conservatisms in the analysis, EOI has implemented NUMARC 91-06 for shutdown operations at RBS. Shutdown Operations Protection Plan and Primary-Secondary

Containment Integrity procedures presently includes guidance for closing the containment hatch and other significant openings in containment, in addition to the requirements contained in the license and design basis.

In addition to the above actions taken to reduce the effects of an event, the following conservatisms are included in the analysis. None of these effects have been credited in this analysis; and therefore, provide significant margin in the results of this request.

- No allowance was made for deformation and energy absorption to the top and bottom structure of the fuel assembly, the spent fuel rack structure, or deformation or energy absorption in the fuel pellets.

- A spent fuel assembly was always assumed to drop onto one or more other fuel assemblies when possible (in the core or the spent fuel pool). In the canal or transfer tube, since the assembly is removed from others, damage to only one assembly can result.

- No allowance was taken for the hydrodynamic drag and energy dissipation of the fuel assembly falling through water.

A further conservatism results by neglecting plate-out inside the reactor containment building prior to release. TID-14844, "Calculation Of Distance Factors For Power And Test Reactor Sites," also states that cloud depletion as ground deposition (particulate fallout) is not assumed during cloud travel. Such deposition during cloud travel would further reduce the dose at the low population zone distance.

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. None of these margins have been included in the River Bend analysis and therefore the results of the analysis are conservative.

Control of Assumptions and Limits

RBS has identified those assumptions and initial conditions contained in this analysis which are fuel dependent and which could affect the results of the analysis. These items will be checked in future reload analyses to maintain the results within those proposed in this submittal.

Proposed Changes and Justification

This proposed amendment to the RBS TSs revises those specifications associated with handling irradiated fuel in the primary or secondary containment. 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) and GDC 19 operator limits of 30 rem thyroid and 5 rem whole body. Specifically, the proposal adds new information for irradiated fuel to each associated specification BASES that contains sufficient fission products to require operability of accident mitigation systems to meet the accident analysis assumptions. The enhanced information revises the OPERABILITY requirements for Secondary Containment Isolation Instrumentation, Control Room Fresh Air Instrumentation, Primary Containment Airlocks, Secondary Containment Isolation Dampers, Fuel Building, Fuel Building Ventilation System-Fuel Handling, Primary Containment-Shutdown, Control Room Fresh Air, Control Room Air Conditioning System, AC and DC sources- Shutdown, Inverters-Shutdown and Distribution Systems - Shutdown. Additional information was included in the BASES for the above specifications and Secondary Containment-Operating and Fuel Building Ventilation-Operating.

The new information for recently irradiated fuel provides a mechanism for applying a cutoff in fission product decay to various specifications where the concept applies. The decay period 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 NUREG 0800, Section 15.7.4 [Ref. 5] and GDC 19 [Ref. 3].

Technical Sp. fication APPLICABILITY statements are revised with "recently irradiated" replacing "irra. ated." Also, ACTION statements are revised, as appropriate, to reflect the proposed changes. Note that the markup for the proposed changes are consistent with the ITS terminology and NUREG 1434.

The revised APPLICABILITY and ACTIONs incorporate the new term of recently irradiated to establish operational conditions where specific activities represent situations where significant radioactive releases can be postulated. During MODEs 4 or 5, these are: 1) When handling recently irradiated fuel; and 2) During operations with a potential for draining the reactor . sel. The APPLICABILITY and ACTIONs for electrical power systems only address fuel handling because the other conditions are implicitly included by the requirement that these systems be operable in MODEs 4 and 5.

The revised APPLICABILITY redefine the LCOs' applicability for instrumentation/devices that initiate alarms, isolate containment, and provide for filtration systems, including support systems, that mitigate the radiological impact of fuel handling accidents. The proposed APPLICABILITY is consistent with the fuel handling accident assumptions. The applicability to recently irradiated fuel bounds events where this fuel is dropped onto other recently irradiated fuel. As described in the SAR [Ref. 2], the accidents postulated to occur during core alterations are: inadvertent criticality due to a control rod removal error, 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 during shutdown. Since the only accident postulated to occur during CORE ALTERATIONS that results in a significant radioactive release is the fuel handling accident, the relationship to CORE ALTERATIONS is not appropriate. Therefore, the proposed LCO applicability for handling recently irradiated fuel assemblies is justified. The

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

The previous limit of 24 hours for moving fuel is maintained. This 24-hour minimum requirement for reactor subcriticality prior to fuel movement ensures that sufficient time has elapsed to allow the radioactive decay of the short-lived fission products.

During the decay period for recently irradiated fuel, selected ESF systems are required to limit the radiological consequences of a fuel handling accident to within regulatory limits. These decay times are consistent with the assumptions used in the accident analyses. The BASES of each specification is revised to include the basis for the decay period used in the new definition for recently irradiated fuel. This location in the BASES to describe the concept and limits for recently irradiated fuel was chosen to minimize changes in the future.

NO SIGNIFICANT HAZARDS CONSIDERATIONS:

This proposed amendment to the River Bend Station (RBS) Improved Technical Specifications (ITS) revises those specifications associated with handling irradiated fuel in the primary or secondary containment and CORE ALTERATIONS. Specifically, the proposal adds additional information to the BASES 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 Secondary Containment Isolation Instrumentation, Control Room Fresh Air Instrumentation, Primary Containment Airlocks, Secondary Containment Isolation Dampers, Fuel Building, Fuel Building Ventilation System-Fuel Handling, Primary Containment-Shutdown, Control Room Fresh Air, Control Room Air Conditioning System, AC and DC sources- Shutdown, Inverters-Shutdown and Distribution Systems - Shutdown. Additional information was included in the BASES for the above specifications and Secondary Containment-Operating and Fuel Building Ventilation-Operating. Detailed listing of the individual specifications are included in Enclosure 3.

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. EOI has evaluated the no significant hazards considerations in this request for a license amendment. In accordance with 1OCFR50.91(a), EOI 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.**

The proposed limits on recently 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 applicability in conjunction with existing administrative controls on light loads, bounds the conditions of the current design basis fuel handling accident analysis. The analysis also concludes the limiting offsite radiological consequences are well within the acceptance criteria of NUREG 0800, Section 15.7.4 and GDC 19. The analysis is also conducted in a conservative manner containing margins in the calculation of mechanical analysis, iodine inventory and iodine decontamination factor. Each of these conservatisms will further decrease the consequences. Therefore, the proposed changes do not significantly increase the probability or consequences of any previously evaluated accident.

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

The proposed limits are used to establish operational conditions where specific activities represent situations where significant radioactive releases can be postulated. In addition, the changes to operation are consistent with previous limits -- only allowing increased flexibility after the radiological consequences are assured to remain within accepted limits. Therefore, 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.

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

The revised limits are 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 RBS licensing limit. Safety margins and analytical conservatisms have been evaluated and are well understood. Conservative methods of analysis are maintained through the use of accepted methodology and benchmarking the proposed methods to previous analysis. Margins are retained to ensure that the analysis adequately bounds all postulated event scenarios. The proposed change only eliminates some excess conservatism from the analysis.

EOI has implemented NUMARC 91-06 guidelines for shutdown operations at RBS. Shutdown Operations Protection Plan and Primary-Secondary Containment Integrity procedures presently include guidance for closure of the containment hatch and other

significant openings in containment, in addition to the requirements contained in the license and design basis. This additional protection will enhance the ability to limit offsite effects.

Acceptance limits for the fuel handling accident are provided in 10CFR100 with additional guidance provided in NUREG 0800, Section 15.7.4. Excess margin is the difference between the postulated doses and the corresponding licensing limit. In the initial review of River Bend Station for operation (NUREG-0989, Section 15.7.4), the NRC accepted the design and analysis based on meeting the guideline dose limits of 10CFR100 and SRP 15.7.4. 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 below the corresponding licensing limit. These margins are unchanged; therefore, the proposed changes do not involve 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.

Environmental Impact Consideration

RBS has reviewed this request against the criteria of 10CFR51.22 for environmental considerations. The request does not affect any system discharging radwaste to the environment or monitoring any such discharge. Also, the request does not adversely affect any system designed to monitor or isolate gaseous radioactive effluents to the environment. Therefore, the request does not involve a significant hazards consideration, does not significantly increase the types or quantity of effluent that may be released offsite, and does not significantly increase individual or cumulative occupational radiation exposures. Based on the foregoing, RBS concludes that the proposed change meets the criteria given in 10CFR51.22 (c)(9) for a categorical exclusion from the requirement for an Environmental Impact Statement.

E. REFERENCES:

1. River Bend Station Unit 1 Technical Specifications and Bases, Updated through Amendment 82.

2. River Bend Station Updated Safety Analysis Report.

3. 10CFR50, Appendix A, General Design Criterion 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. Grand Gulf Nuclear Station submittal GNRO-94/00131, dated November 9, 1994

8. Arkansas Nuclear One submittal dated May 19, 1995

9. NRC letter dated August 31, 1994 issuing amendments 171/194 to units 1 & 2 of Calvert Cliffs Nuclear Power Plant

10. TID-14844, "Calculation Of Distance Factors For Power And Test Reactor Sites" dated March 1962

Enclosure 3

Affected Technical Specifications
&
Summary of Change

This request will provide additional information concerning the limits of "recently irradiated fuel" which will allow various ESFs, presently being required with irradiated fuel, no longer being required. The table below identifies the changes for the Improved Technical Specification (ITS). Enclosure 4 contains the markup of the specifications. Enclosure 5 contains the associated BASES changes for information.

<u>ITS</u> <u>Title/Summary</u>

3.3.6.2	SECONDARY CONTAINMENT ISOLATION INSTRUMENTATION, note a
3.3.7.1	CONTROL ROOM FRESH AIR SYSTEM INSTRUMENTATION, note b
3.6.1.2	PRIMARY CONTAINMENT AIRLOCKS; Applicability & Action E
3.6.1.10	PRIMARY CONTAINMENT-SHUTDOWN; Applicability & Action A
3.6.4.2	SECONDARY CONTAINMENT ISOLATION DAMPERS; Applicability & Action D
3.6.4.5	FUEL BUILDING; Applicability & Action A
3.6.4.7	FUEL BUILDING VENTILATION SYSTEM-FUEL HANDLING; Applicability & Action B
3.7.2	CONTROL ROOM FRESH AIR SYSTEM; Applicability, Actions C & E
3.7.3	CONTROL ROOM AIR CONDITIONING SYSTEM; Applicablity, Actions D & E
3.8.2	AC SOURCES-SHUTDOWN; Applicablity, ActionsA & B
3.8.5	DC SOURCES-SHUTDOWN; Applicablity & Action A
3.8.8	INVERTERS-SHUTDOWN; Applicablity & Action A
3.8.10	DISTRIBUTION SYSTEMS-SHUTDOWN; Applicablity & Action A

BASES **In addition to the identified changes above the following additional specifications BASES will be changed upon approval of this request.**

B3.6.4.1	SECONDARY CONTAINMENT-OPERATING; APPLICABLE SAFETY Analysis & Applicability
B3.6.4.6	FUEL BUILDING VENTILATION SYSTEM-OPERATING

Enclosure 4

Markup of Improved Technical Specifications