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U. S. Nuclear Regulatory Commission
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Subject: River Bend Station - Unit 1
Docket No. 50-458
License No. NPF-47
License Amendment Request (LAR) 96-39, Change to Technical Specifications 3.6.1.3, "Primary Containment Isolation Valves (PCIVs)," 3.6.1.8, "Penetration Valve Leakage Control System (PVLCS)" and 3.6.1.9, "Main Steam-Positive Leakage Control System (MS-PLCS)"

File Nos.: G9.5, G9.42

RBEXEC-96-171
RBF1-96-0417
RBG-43358

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). The proposed changes will permit an increase in the allowable leak rate for the Main Steam Isolation Valves (MSIVs) and delete the Penetration Valve Leakage Control System (PVLCS) and Main Steam-Positive Leakage Control System (MS-PLCS) requirements.

Increasing the allowable leak rate for the MSIVs and deletion of the MS-PLCS is supported by the BWR Owners' Group (BWROG) work. The BWROG formed a MSIV Leakage Committee in 1982 in response to Generic Issue C-8, "MSIV Leakage and LCS Failure." Generic Issue C-8 addressed the safety concerns that reported MSIV leakages are too high and that the Leakage Control System (LCS) will not function at high MSIV leakages. Based on the extensive, ongoing work performed by the BWROG to support resolution of the Generic Issue, the BWROG has developed the technical justification for the proposed Technical Specifications changes and associated exemption requests. The General Electric (GE) report, NEDC-31858P, Rev. 2, "BWROG Report for Increasing MSIV Leakage Rate Limits and Elimination of Leakage Control Systems," describes the safety benefits and provides justification for the proposed changes. Increasing the leakage rate limits for the MSIVs will eliminate unnecessary maintenance.

Deletion of the PVLCS, unique to RBS, is supported by rationale similar to that provided in the GE LCS Report. In addition, the revised LOCA dose calculations using current design data and regulatory guidance, confirm that the on-site and off-site doses resulting from a design basis LOCA, based on airborne and liquid radionuclide release pathways, support the deletion of both the PVLCS and MS-PLCS and the associated increase in MSIV leakage. Deletion of the PVLCS compressors is supported by RBS probabilistic safety assessments (PSAs) regarding long term ADS/SRV backup air supply.

An affidavit supporting the facts set forth in this transmittal and its attachments is provided in Attachment 1. Attachment 2 provides a description of the proposed changes and the associated justification for same. Attachment 3 provides the revised LOCA dose consequences and associated release paths. Attachment 4 provides the cost-savings summary associated with deletion of the PVLCS and MS-PLCS. Attachment 5 provides the preliminary conceptual design for the proposed long term air supply to the PVLCS accumulator tanks (LSV*TK6A and LSV*TK6B). Lastly, Attachment 6 provides a marked-up copy of the affected pages from the RBS Technical Specifications.

Entergy 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 hazard consideration or significantly increase individual or cumulative occupational radiation exposures. Based on the foregoing, Entergy 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.

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 licensee requests for changes requiring NRC staff review that involve high cost without a commensurate safety benefit. The proposed change results in both safety benefits (e.g., occupational dose reduction due to reduced maintenance and testing), as well as, significant economic benefits. Entergy estimates cost savings in excess of $20 million for the remaining licensed life of RBS due to a reduction in unnecessary repair costs to both the PVLCS and MS-PLCS valves (including the MSIVs), reduction in dose exposure to maintenance and test personnel, reduction in outage duration and the extension of the effective service life of these containment isolation valves. These costs exceed the threshold of $100,000 established under the CBLA Program.

Based upon the refueling outage safety improvement and significant resource savings that can be realized by implementation of this proposed change, Entergy is respectfully requesting that this application be reviewed on a schedule sufficient to support the seventh refueling outage (RF-7) currently scheduled to commence September 12, 1997.

This request has been discussed with the NRR project manager for RBS. It has also been reviewed and approved by the RBS Facility Review Committee and the Nuclear Review Board. If you have any questions regarding this request or require additional information, please contact Mr. T. W. Gates at (504) 381-4866.

Sincerely,

J. R. McGaha

JRM/WDR/MGC
attachments

cc: Mr. David L. Wigginton
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Attn: Administrator

BEFORE THE

UNITED STATES NUCLEAR REGULATORY COMMISSION

LICENSE NO. NPF-47

IN THE MATTER OF

ENTERGY GULF STATES, INC.

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 River Bend Station License Amendment Request (LAR) 96-39, Change to Technical Specifications 3.6.1.3, "Primary Containment Isolation Valves (PCIVs)," 3.6.1.8, "Penetration Valve Leakage Control System (PVLCS)" and 3.6.1.9, "Main Steam-Positive Leakage Control System (MS-PLCS);" that I signed this letter 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
PARISH OF WEST FELICIANA

SUBSCRIBED AND SWORN TO before me, a Notary Public, commissioned in the Parish above named, this 6th day of November, 1996.

(SEAL)

Claudia F. Hurst

Claudia F. Hurst
Notary Public

My Commission expires with life

ENTERGY OPERATIONS, INC.
RIVER BEND STATION
DOCKET 50-458/LICENSE NO. NPF-47
LICENSE AMENDMENT REQUEST 96-39

1.0 Licensing Document Involved

This proposed change affects the following Technical Specifications:

3.6.1.3 Primary Containment Isolation Valves (PCIVs)
3.6.1.8 Penetration Valve Leakage Control System (PVLCS)
3.6.1.9 Main Steam-Positive Leakage Control System (MS-PLCS)

2.0 Background

2.1 Penetration Valve Leakage Control System (PVLCS) including ADS/SRV Support

As described in USAR Section 9.3.6, the PVLCS supplements the isolation function of primary containment isolation valves (PCIVs) in process lines that also penetrate the secondary containment. These penetrations are sealed by air from the PVLCS to prevent fission products leaking past the isolation valves and bypassing the secondary containment after a Design Basis Accident (DBA) loss of coolant accident (LOCA).

The PVLCS consists of two independent, manually initiated subsystems, either of which is capable of preventing fission product leakage from the containment post LOCA. Each subsystem is comprised of an air compressor, an accumulator, an injection valve, and three injection headers with separate isolation valves. This system has additional headers, which serve the Main Steam-Positive Leakage Control System (MS-PLCS) and Safety Relief Valve (SRV) actuator air accumulators.

The analyses described in USAR Section 15.6.5 provide the evaluation of off-site dose consequences during accident conditions. During the first 25 minutes following an accident, the isolation valves on lines that penetrate primary containment and also penetrate secondary containment are assumed to leak fission products directly to the environment, without being processed by the Standby Gas Treatment System. The PVLCS is manually initiated 20 minutes following an accident. The analyses take credit for manually initiating PVLCS after 25 minutes and do not assume any further secondary containment bypass leakage. Each process line has two PCIVs and an additional isolation valve outside of the outboard PCIV. The two outboard valves are the designated PVLCS valves. Each PVLCS valve is provided sealing air from its electrically associated division of PVLCS.

The minimum air supply necessary for PVLCS OPERABILITY varies with the system being supplied with compressed air from the PVLCS accumulators. Due to the support system function of PVLCS for SRV actuator air, however, SR 3.6.1.8.1 specifies a minimum PVLCS accumulator pressure of 101 psig, which provides sufficient air for intermediate and long term post LOCA SRV actuation. This minimum air pressure alone is sufficient for PVLCS to support the OPERABILITY of the SRVs. PVLCS provides a long term safety-related backup air supply to the Safety Relief Valve (SRV) and Automatic Depressurization System (ADS) air accumulators. USAR Section 5.2.2.4.1 states that the primary source of air for the SRV and ADS accumulators is from the non-nuclear safety main steam (SVV) air compressors which are backed up by the safety-related PVLCS compressors. The ADS accumulators alone have sufficient capacity to support the ADS valves to depressurize the reactor in the event of a design basis small break LOCA and loss of high pressure injection as described in USAR Section 5.2.2.4. This analysis does not depend on the PVLCS compressors to initially depressurize the reactor. The leakage control system is manually placed in service approximately 20 minutes following a LOCA event. The compressors themselves are load shed from the electrical distribution system for 10 minutes, after an interruption of electrical power. They function to provide a backup air supply to support long-term operability of the ADS/SRV valves.

In the RBS analysis for Station Blackout (SBO), an evaluation of the actual capacity of the SRVs was conducted. This analysis determined that for the 4 hour coping criteria, a maximum 31 SRV actuations will be required. The majority of the actuations will be needed during the earlier portion of the event. The seven ADS accumulators provide sufficient air for a total of 28 to 35 actuations available to control RPV pressure. When the nine remaining non-ADS SRVaccumulators are considered, a minimum of 37 cycles are available. In addition, operator guidance for addressing the SBO event or loss of SVV air supply is available to control RPV pressure by continuous SRV opening, conserving accumulator supply, and providing backup air, if necessary. As a result, the actual relieving capacity is in excess of the expected need without the backup PVLCS air compressor. The air supply to the SRVs is not credited for compliance to the Station Blackout (SBO) Rule. The four-hour SBO coping duration analysis for RBS demonstrates that ADS/SRV accumulators have adequate capacity to mitigate this event. Evaluations performed at RBS have determined the probability of recovering off-site power within various time frames. The results indicated the probabilities are greater than 98% within four hours.

The analysis conducted for the design basis of the PVLCS contains the following conservatisms:

The leakage from the valves is assumed to be from open systems inside containment and outside secondary containment. Many of the systems are closed both inside and outside containment.

RG 1.3 requires 100% noble gases and 25% halogens be assumed to be released with the majority of the off-site dose being contributed by iodine. At TMI-2, the iodine release was only 0.004%, a factor of 6250 reduction.

As a result, the actual off-site effects of the DBA event are expected to be greatly reduced from those evaluated.

2.2 Main Steam-Penetration Leakage Control System (MS-PLCS) including MSIVs

As described in USAR Section 6.7, the MS-PLCS supplements the isolation function of the MSIVs by providing a positive pressure air seal for the fission products that could leak through the closed MSIVs after a DBA LOCA.

The MS-PLCS consists of two independent subsystems. An inboard subsystem, which is connected between the inboard and outboard MSIVs and an outboard subsystem, which is connected to the double disk of main steam shutoff valves and the valve stem packing glands of the outboard MSIVs. These are maintained at a positive air pressure with respect to reactor vessel pressure following system actuation. The MS-PLCS is supplied with compressed air by two separate and redundant compressed air supply subsystems that are integral components of the PVLCS. Each subsystem receives power from a separate division of the emergency power supply. The MS-PLCS is manually initiated approximately 20 minutes following a DBA LOCA, and is designed to control and minimize leakage through the MSIVs for up to 30 days.

The MS-PLCS mitigates the consequences of a DBA LOCA by preventing the release of untreated fission products from the closed MSIVs for this type of an event. The analyses in USAR Section 15.6.5 provide the evaluation of off-site dose consequences.

Each main steam line has two quick-closing, air-operated MSIVs: one inboard and one outboard of the containment penetration. The safety function of the MSIVs is to isolate the reactor system in the event of a LOCA or other events requiring containment or reactor system isolation. For a steam line break, the isolation valves would terminate the blowdown of reactor coolant in sufficient time to prevent an uncontrolled release of radioactivity from the reactor vessel to the environment. For the LOCA, the valves would isolate the reactor from the environment and prevent the direct release of fission products from the containment. Although the MSIVs are designed to provide a leak-tight barrier, industry operating experience has indicated that degradation has occurred in the leak-tightness of these valves. Only that portion of the main steam system from the RPV up to and including the outboard MSIV is part of the reactor coolant pressure boundary. The MSIVs are designed to close against maximum steam flow and differential pressure. Over-pressure protection is provided by the main steam safety relief valves (SRVs) which discharge the steam to the suppression pool. The main steam piping from the vessel nozzle to the outboard MSIVs, including the MSIVs, are designed to the classifications of Seismic Category I, ASME Section III, and Quality Group A.

The MSIVs are globe valves designed to close by diversified sources of spring pressure and pneumatic pressure. In addition, each valve has an air accumulator to assure adequate supply of pneumatic pressure. The MSIVs are designed to close on loss of pneumatic pressure or loss of power to the pilot valves. Each valve has an independent position switch initiating a signal into the RPS scram trip circuit when the valve closes. The MSIVs are signaled to close on the following conditions: (1) low water level in the reactor vessel, (2) high flow rate in the main steam line, (3) low pressure at the inlet to the turbine, (4) high ambient steam line tunnel temperature (outside containment), (5) low condenser vacuum (unless procedurally bypassed), and (6) high turbine building temperature. Once any of the closure signals is initiated, the valves will continue to close and cannot be opened except by manual means. Each valve may be operated by independent remote-manual switches located in the control room. Lights in the control room indicate valve position.

The analysis of a complete, sudden steam line break outside the containment is described in USAR Chapter 15. The analysis shows that the fuel barrier is protected against loss of cooling if MSIV closure is within specified limits, including instrumentation delay, to initiate valve closure after the break. The calculated radiological effects of the radioactive material assumed to be released with the steam are shown to be well within the guideline values for such an accident.

The shortest closing time (approximately 3 seconds) of the MSIVs is also shown in USAR Chapter 15 to be satisfactory. The switches on the valves initiate a reactor scram when specific conditions (extent of valve closure, number of main steam lines affected, and reactor power level) are exceeded (USAR Section 7.2.1). The pressure rise in the system from stored and decay heat may cause the SRVs to open briefly, but the rise in fuel cladding temperature is insignificant and no fuel damage results.

2.3 Existing Dose Calculations

2.3.1 General Regulatory Requirements

The regulatory requirements applicable to the determination of off-site dose consequences of design basis accidents are provided in 10CFR Part 50 and Part 100. The regulations provided in 10CFR Part 50 provide the basis and procedures applicable to the licensing of production and utilization facilities. Section 50.34 of 10CFR Part 50 requires an analysis and evaluation of the design and performance of structures, systems, and components with the objective of assessing the risk to public health and safety resulting from the operation of RBS. The site evaluation is to assume a fission product release from the core based upon a major accident, hypothesized for purposes of site analysis or postulated from considerations of possible accident events, that would result in potential hazards not exceeded by any accident considered credible. These core release source terms are based on TID-14844, "Calculation of Distance Factors for Power and Test Reactor Sites." The evaluation also assumes containment leakage at the expected demonstrable leakage rate and site specific meteorological conditions. With these

conservative assumptions, both the 2 hour Exclusion Area Boundary dose and the duration (30 day) Low Population Zone dose are shown to be less than the 10CFR100 limits of 25 Rem whole body and 300 Rem thyroid.

Post accident Control Room exposures are also governed by the requirements of 10CFR50. General Design Criterion 19 of Appendix A to Part 50 requires that adequate protection be provided to permit access to and occupancy of the Control Room under accident conditions and for the duration of the accident without personnel radiation exposures in excess of 5 Rem to the whole body, or its equivalent to any part of the body (i.e., 30 Rem, thyroid).

2.3.2 Accident Analysis Requirements

Section 50.34 of 10CFR50 requires an analysis and evaluation of the design and performance of structures, systems, and components with the objective of assessing the risk to public health and safety resulting from the operation of RBS. The DBA LOCA is the postulated accident used to establish the adequacy of structures, systems, and components with respect to public health and safety since this accident establishes the limiting radiation doses for all postulated events. The NRC Staff, in Regulatory Guide 1.3, Rev. 2, "Assumptions Used for Evaluating the Potential Radiological Consequences of a Loss of Coolant Accident for Boiling Water Reactors," has provided assumptions and methods which are considered an acceptable means of demonstrating compliance with the regulatory requirements of 10CFR Parts 50 and 100.

Branch Technical Position CSB 6-3 establishes additional requirements relative to the determination of leakage which bypasses the secondary containment. Additionally, Standard Review Plan (SRP) 6.5.5 allows credit for the pressure suppression pool as a fission product cleanup system. The guidance in this SRP replaces the guidance provided in Regulatory Guide 1.3 Position C.1.f.

There is no specific regulatory guidance which provides acceptable Control Room designs relative to the control of post accident radiation doses to Control Room personnel. However, Regulatory Guide 1.78, "Assumptions for Evaluating the Habitability of a Nuclear Power Plant Control Room During a Postulated Hazardous Chemical Release" provides guidance on Control Room design relative to the release of hazardous chemicals. Since the Control Room and associated support systems are provided as engineered safety features, the guidance provided in Regulatory Guide 1.52 is applicable.

2.3.3 Existing Dose Analysis

The design basis accident analysis considers the loss of coolant accident (DBA-LOCA) as the limiting credible accident event. The design basis loss of coolant accident dose analysis was performed in accordance with all applicable regulatory requirements (e.g., NRC Standard Review Plan 15.6.5 and Reg.

Guide 1.3) using conservative computer programs, models, and input parameters. Analysis guidelines set forth in these regulations use the assumptions and procedures originally delineated in TID-14844.

In the existing RBS LOCA dose analysis calculation, the release of radionuclides from the severely damaged core are assumed to consist of 100 percent of the noble gases and 50 percent of the iodine present in an equilibrium core operating at a power level of 3039 MWt for 1000 days prior to the accident. Of this release, 100 percent of the noble gases and 50 percent of the iodine remain airborne. The other 50 percent of the iodine is removed by plate-out or condensation on the drywell and containment surfaces and internals in accordance with TID-14844. The dose consequences of contaminated liquid leakage from the containment are included in this calculation and are combined with the doses due to airborne containment sources. The existing analysis used dose conversion factors from ICRP 2 and used the SWEC (Stone & Webster Engineering Corporation) DRAGON4 computer code. Leakage from valves served by the PVLCS was assumed to be directly to the environment prior to the PVLCS operation at 25 minutes. Leakage through the MSIVs was not considered beyond 25 minutes due to the operation of the MS-PLCS. The existing dose analysis also considered the consequences of leakage from ESF components outside the containment in accordance with NUREG-0800, Standard Review Plan 15.6.5, Rev. 2, Appendix B, "Radiological Consequences of a Design Basis Loss-Of-Coolant Accident: Leakage from Engineered Safety Feature Components Outside Containment."

3.0 <u>Description of Proposed Change</u>

3.1 Penetration Valve Leakage Control System (PVLCS) including ADS/SRV Support

Entergy proposes to delete the PVLCS and either remove the components or abandon in-place. This includes the PVLCS supply to those process lines treated by the PVLCS which include:

1) RWCU to Main Condenser (Penetration 1KJB*Z4)
2) RWCU Backwash to Radwaste (Penetration 1KJB*Z5)
3) Condensate Makeup (Penetration 1KJB*Z134)
4) Fire Protection (Penetration 1KJB*Z41)
5) Ventilation Chilled Water Return (Penetration 1KJB*Z131)
6) Ventilation Chilled Water Supply (Penetration 1KJB*Z132)
7) Service Air (Penetration 1KJB*Z44)
8) Instrument Air (Penetration 1KJB*Z46)
9) Feedwater to RPV-Loop A (Penetration 1KJB*Z3A)
10) Feedwater to RPV-Loop B (Penetration 1KJB*Z3B)

Additionally, this includes elimination of the PVLCS compressor skids including the compressors but retaining the PVLCS accumulator tanks LSV*TK6A and LSV*TK6B as a backup air supply to the ADS/SRV accumulators. A new Seismic Category I line will be installed with a connection external to the Auxiliary Building as an alternate long term air supply in the unlikely event that the plant Instrument Air System (IAS) diesel-driven air compressor backup supply, described in Section 4.1, is not available. A basic drawing of this conceptual design is provided in Attachment 5.

Additionally, Technical Specification 3.6.1.8, "Penetration Valve Leakage Control System (PVLCS)," will be deleted and the reference to the PVLCS in SR 3.6.1.3.9 will be deleted. SR 3.6.1.8.1 air pressure and applicable portions of SR 3.6.1.8.2 system functional testing will be relocated to the Technical Requirements Manual.

3.2 Main Steam-Positive Leakage Control System (MS-PLCS) including MSIVs

Entergy proposes to delete the MS-PLCS and either remove the components or abandon in-place. This includes the supply to those process lines treated by MS-PLCS which include:

1)	Main Steam Line 1A	(Penetration 1KJB*Z1A)
2)	Main Steam Line 1B	(Penetration 1KJB*Z1B)
3)	Main Steam Line 1C	(Penetration 1KJB*Z1C)
4)	Main Steam Line 1D	(Penetration 1KJB*Z1D)
5)	Turbine Plant Miscellaneous Drains	(Penetration 1KJB*Z2)

Additionally, Technical Specification 3.6.1.9, "Main Steam-Positive Leakage Control System (MS-PLCS)," will be deleted. The proposed change will result in the following changes to Technical Specification 3.6.1.3, "Primary Containment Isolation Valves (PCIVs):"

a. Delete SR 3.6.1.3.8
b. Change SR 3.6.1.3.10 to read as follows:

Verify leakage rate for each MSIV is $\leq$100 scfh and the combined maximum pathway leakage rate for all four main steam lines is $\leq$200 scfh when tested at $\geq P_a$. However, if any MSIV exceeds 100 scfh, the acceptance criterion for its retest shall be in accordance with the Primary Containment Leakage Rate Testing Program.

3.3 Dose

3.3.1 Revised Dose Analysis

The proposed change requires a revision to the existing LOCA dose analysis due to potential leakage from the MSIVs and the valves served by the PVLCS. Additional changes were also included in the revised dose analysis to account for changes in regulatory guidance and dose methodology. These changes are

described in the following sections. Attachment 3 provides the LOCA dose consequences and associated release paths.

3.3.2 Airborne Dose Analytical Model Description

The containment, drywell, and annulus are modeled as three separate regions. The following discussion is applicable to the calculation of airborne doses only.

The design basis leak rate of the primary containment (excluding main steam lines and lines sealed by the PVLCS) is 0.26 weight percent of the containment volume per 24 hours at P_a for the duration of the accident and is assumed to be released entirely to the environment initially or the secondary containment later into the accident. In addition, the leakage rate of 170,000 cc/hr (4298 sccm) at P_a through the containment isolation valves served by the PVLCS is considered as bypass leakage circumventing the secondary containment.

Releases of fission products from the PVLCS and MS-PLCS valve leakage are not through the plant stack which requires a different set of χ/Q values. The χ/Q values used for the PVLCS and MS-PLCS valve leakage are the most limiting values for turbine or radwaste building releases.

The Control Room is modeled as a separate region outside of containment. The Control Room doses result from infiltration of radioactivity into the Control Room from the environment. The primary path into the Control Room envelope is through the Control Room Fresh Air (CRFA) filtration supply to the Main Control Room HVAC systems that are available during the DBA. Activity also penetrates the Control Room envelope through inleakage due to door openings.

3.3.3 General Scenario

The activity released from the severely damaged core is assumed to enter the drywell at time zero. Transfer from the drywell to the containment is either through the suppression pool, where a decontamination factor of 10 is taken per SRP 6.5.5, or through drywell leakage per RBS Technical Specifications which bypasses the suppression pool. The bypass flow which enters the containment atmosphere and the activity leaving the suppression pool are assumed to be uniformly mixed in the containment atmosphere. The flow through the horizontal vents into the suppression pool is determined in the calculation along with the drywell bypass flow. Suppression pool scrubbing is assumed to remain effective as long as there is flow from the drywell into the suppression pool.

During the first 24 seconds post-LOCA, the annulus is below -0.25 in WG and the annulus exhaust is isolated (Technical Specification 3.6.4.1). Leakage of radionuclides from the containment to the annulus during this time interval is retained in the annulus until the pressure exceeds -0.25 in WG. The leak rate

from the containment to the annulus during this time is 0.259%/day (i.e., 0.26%/day - 13,500 cc/hr). The 13,500 cc/hr (341sccm) figure is that portion of containment leakage which is assumed to bypass the annulus and enter the Fuel and Auxiliary Buildings. Since the Auxiliary and Fuel Building pressures are not below -0.25 in WG during this initial time interval, the annulus bypass leakage is directly to the environment. Leakage from the drywell to the atmosphere through one failed MSIV valve is assumed to begin at time zero. Leakage from the containment to the atmosphere through the PVLCS (secondary containment bypass) valves is assumed to begin at time zero.

The model conservatively assumes that one inboard MSIV fails open at time zero. An analysis of the section of the main steam line between the inboard and outboard MSIVs following a DBA LOCA shows that for the first two hours the steam line pressure is well above the pressure at which leakage from the containment would be expected to occur. Therefore, the MSIVs associated with the remaining three steam lines are assumed to begin leaking at 2 hours with a total leak rate of 200 scfh for all four main steam lines. Activity from the MSIVs is assumed to leak directly to the environment.

The on-site and off-site doses were determined using the TRANSACT computer code which included the ICRP 30 dose conversion factors. RBS has previously implemented the TRANSACT code currently used as the standard Entergy code for offsite and Control Room dose calculations due to postulated events. This code was utilized with respect to the SER previously issued to RBS by the NRC Staff in response to LAR 95-04 from D. L. Wigginton to J. R. McGaha, Jr. dated January 11, 1996.

The TRANSACT code is derived from the TACT5 code documented in NUREG/CR-5106. RBS has previously 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.

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 Entergy. 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.

3.3.4 Liquid Release Dose Analytical Model Description

Potential leakage of contaminated liquids following a design basis accident can result in the release of radioisotopes outside the containment. Although the RBS design provides barriers to such releases in accordance with regulatory

requirements, these barriers must be assumed to pass some limited amount of leakage in order to allow for realistic equipment performance characteristics and testing methods. Potential leakage sources include penetration leakage, system boundary valve leakage, or leakage from ESF components. Following a LOCA, there is the possibility of leakage of contaminated water into the Auxiliary Building. The potential leakage pathways result in a total leakage rate to the Auxiliary Building of 60 gph as stated in the existing dose analysis and as specified in NUREG-0989, SER Section 15.6.5, dated May 1984.

3.3.5 Regulatory Guidance

NUREG-0800, Standard Review Plan 15.6.5, Rev. 2, Appendix B, addresses one source of leakage of water outside the containment as a potential source of fission product leakage following a design basis LOCA. This source, leakage of water from ESF equipment, is postulated to occur during the recirculation phase for long-term core and primary containment cooling.

3.4 General

The proposed change removes the requirements for 33 Leakage Control Systems (LCS) MOVs from the PVLCS and MS-PLCS (in addition to other valves and components). These MOVs will not serve as PCIVs and as such, have no safety significance. Although some of these MOVs will be retained as normal system isolation or maintenance valves, they will have no safety or leakage control function. Entergy intends to disconnect, remove, and/or cap and abandon the leakage control portions of the PVLCS and MS-PLCS. LCS lines connecting to the secondary containment bypass and main steam line PCIVs will be disconnected and welded/capped closed such that containment integrity is maintained.

4.0 Justification for Proposed Change

4.1 Penetration Valve Leakage Control System (PVLCS) including ADS/SRV Support

The currently installed plant instrument air (IAS) diesel-driven air compressor is capable of supplying the IAS and service air (SAS) systems at a pressure of approximately 135 psig. The IAS supply is hard-piped to both the SVV air dryers and the divisional PVLCS piping. The piping downstream of the PVLCS accumulator tanks provides backup to the SVV divisional air supply headers. The connections from IAS to PVLCS contain isolations which prevent a failure in the IAS piping system from affecting the safety-related PVLCS system. Existing design provides for isolating SAS loads in order to maintain IAS loads with any one of the normal IAS or SAS compressors (three IAS and three SAS electric air compressors) or the IAS diesel-driven air compressor supplying the loads. The IAS diesel-driven compressor is currently started on a weekly basis. Periodic functional testing will be provided to assure its reliable operation. The design of the SVV air supply system, discussed in USAR Section 5.2, includes an air accumulator for each SRV. The seven Automatic

SRV SVV accumulators are capable of providing sufficient air pressure for a minimum of three lifts of each valve. The nine non-ADS SRV SVV accumulators are capable of providing sufficient air pressure for a minimum of one lift of each valve under design conditions. The ADS design basis (reactor isolation and failure of high pressure injection) requires a single lift of six of the seven ADS SRVs. SRVs will be available for long-term post accident/event (beyond DBA) reactor vessel pressure control. As discussed above, the PVLCS provides long-term backup air supply to the SRVs through the PVLCS air accumulators (LSV*TK6A and 6B) which are maintained at pressures at or above 101 psig.

Current guidance directs the operator to verify the availability of an alternate air supply upon the loss of both divisions of PVLCS. With this verification, the operator is directed to maintain the availability of the alternate supply, thereby assuring long-term SRV function. Sufficient time is provided for the operator to confirm availability of the alternate backup air supply to the SRVs especially when considering the unlikelihood of multiple failures resulting in loss of the backup supply (PVLCS) and an event requiring this feature.

This is further substantiated with regard to the SER previously issued to RBS by the NRC staff in response to LAR 94-04 from D. L. Wigginton to J. R. McGaha, Jr. dated June 19, 1995. This SER states that the IAS diesel-driven air compressor, not dependent on off-site power, is an acceptable supply to the PVLCS accumulator tanks which provide long-term backup air supply to the ADS/SRV accumulators.

4.2 Main Steam-Penetration Leakage Control System (MS-PLCS) including MSIVs

The proposed changes to the MS-PLCS are a result of extensive work performed by the BWR Owner's Group (BWROG) in support of the resolution of Generic Issue C-8, "MSIV Leakage and LCS Failure." GE Report NEDC-31858P, Rev. 2, "BWROG Report for Increasing MSIV Leak Rate Limits and Elimination of Leakage Control Systems" dated September 1993, provides technical justification on a generic basis to support these proposed changes and is presently in the final review stages with the NRC staff. As a result of increasing MSIV leakages and the inability of the LCS to function at high MSIV leakages, the NRC prioritized Generic Issue C-8, "MSIV Leakage and LCS Failure," as a high priority item in January 1983. This issue was closed in 1990.

The BWROG report, NEDC-31858P, provides the justification for increasing MSIV leakage limits and for eliminating the requirements for the LCS. In addition, based on an extensive evaluation of valve leakage data, the BWROG concludes that MSIV leakage rates exceeding 500 scfh usually indicate that significant maintenance or valve modification is required; whereas, leakage rates less than 500 scfh can generally be eliminated by proper routine maintenance and testing procedures. This supports the BWROG position that MSIV leakage rates up to 500 scfh are not indicative of substantial defects that would challenge the isolation capability of the MSIVs. This conclusion is verified by LERs in which plant maintenance personnel routinely do not find specific defects when the MSIVs have been disassembled for repairs. This finding

is also consistent with the observation of MSIV leakage performance as documented in NUREG-1169. With concurrence from the valve manufacturers, this report concludes that MSIV leakage rates up to 200 scfh are not an indication of substantial mechanical defects in the valve which would challenge the isolation capability of the valve to fulfill its safety function. Therefore, the proposed increase in the allowable leakage rate to 100 scfh for the MSIVs (not to exceed 200 scfh for all four main steam lines) will not inhibit the isolation capability of the valve.

The BWROG has also found that disassembling and refurbishing MSIVs to meet the unnecessarily low leakage limits frequently contributes to repeated failures. Examples of these maintenance-induced defects include machining-induced seat cracking, machining of guide ribs, excessive pilot valve seat machining, and mechanical defects induced by assembly and disassembly. By not having to disassemble the valves and refurbish them for minor leakage, RBS may avoid introducing one of the root causes of recurring leakage. Industrial experience suggests that by attempting to correct non-existing or minimal defects in the valves, it is likely that some actual defects may be introduced that lead to later leak rate test failures.

The BWROG formed a MSIV Leakage Committee in 1982 to address the increasing MSIV leakage rates and a follow-on MSIV Leakage Closure Committee in 1986 to address alternate actions to resolve on-going but less severe MSIV problems. The MSIV Leakage Committee identified contributors which cause MSIVs to fail the leak rate tests by large margins, developed recommendations to minimize leakages, evaluated alternates for MSIV leakage treatments, and compiled recent history of MSIV leakages and LCS operating experience.

MSIV failure to meet the current Technical Specification limit have been documented in response to surveys conducted by the NRC during the early 1980s and by the BWROG during the middle and late 1980s. As many as 50% of the total "as found" MSIV local leak rate tests were reported in the early NRC survey to exceed the leakage rate limit.

This proposed increase provides a more realistic, but still conservative, limit for the MSIVs. Based on the BWROG study, the proposed increase in the allowable leak rate will increase the chance for a successful local leak rate test to greater than 90%, up from the previous 77% success rate. This increase in successful local leak rate testing will significantly reduce MSIV maintenance costs, reduce dose exposure to maintenance and test personnel, reduce outage durations, extend effective service life of the MSIVs, and minimize the potential for outage extensions at RBS.

A safety-related LCS was required by Regulatory Guide 1.96 in order to reduce the radiological consequences of MSIV leakage. As discussed earlier, Generic Issue C-8 identified the safety concern that MSIV leakage rates, as determined by conservative local leak rate tests, were too high and that the LCS would not function at high MSIV leakage rates. The 1981 NRC survey indicated that 33 percent of the total tests exceeded leakage rates of 100 scfh. Since the process capability of the LCS at RBS is designed for MSIV leakage rates of no more than 100 scfh, the potential existed for

the LCS not to function as analyzed for a design-basis LOCA as described in USAR Section 15.6.5.

The safety significance of the LCS in terms of public risk was addressed in NUREG CR-4330. This NUREG evaluated the possibility of regulatory modification concerning elimination of the LCS requirements and disabling the systems currently installed at BWRs. The NUREG concluded that streamlining the regulatory requirements would have little impact on public risk and that substantial savings in operating costs may be realized. The NUREG also concluded that the increased overall public risk is less than one percent, but the overall savings are estimated over several millions of dollars for the remaining life of the plants.

From a safety perspective, calculations using standard conservative assumptions for considering the off-site consequences of a postulated design basis LOCA confirm that off-site and Control Room doses will be within the regulatory guidelines for the allowable MSIV leakage rate. This calculation is described in Section 15.6.5 of the USAR. However, if MSIV leakages are only moderately higher than the allowable limit, the calculated doses will exceed the regulatory guidelines. Furthermore, as documented in Generic Issue C-8, the LCS will not function if MSIVs exceed the leakage limit by a moderate amount.

It should be noted that the radiological analysis assumes no credit for maintaining the integrity of the secondary plant system. If the secondary system maintains its integrity, the main steam piping and main condenser still provide the capability to process MSIV leakage following a LOCA. The main steam and main condenser systems provide fission product attenuation by providing hold-up and plate-out of fission products that may leak through the MSIVs. Therefore, in addition to the design basis LOCA, a postulated failure of the secondary system to maintain its integrity would also have to be assumed in order to have a direct leakage path through the MSIVs. The combination of both these events (LOCA and secondary system integrity loss) concurring simultaneously is deemed acceptably low.

Based on the BWROG evaluation of MSIV leakage performance, the current Technical Specification allowable MSIV leak rate is extremely limiting and routinely requires the repair and retesting of MSIVs. This unnecessary repair significantly impacts the maintenance work load, often contributes to outage extensions, and has in the past adversely affected the operability of the MSIVs. BWR outage planners routinely schedule several days of contingency to repair and retest the MSIVs. In addition, the needless dose exposure to maintenance and test personnel is inconsistent with As Low As Reasonably Achievable (ALARA) requirements. There have also been many Licensee Event Reports written for MSIV leakages exceeding the Technical Specification limit.

4.3 Dose

The design basis LOCA has been reanalyzed for radiological impacts based on deletion of both the PVLCS and MS-PLCS and increasing the MSIV leakage rate to $\leq$100 scfh

for any one valve and $\leq$200 scfh for a combined maximum pathway leakage for all four main steam lines when tested at $\geq P_a$. The total off-site and on-site LOCA doses for both the airborne and liquid release pathways are well within the applicable regulatory limits. Plant-specific supporting information and results of the revised radiological analysis that justify the proposed changes are included in Attachment 3. As concluded in the supporting information, the increased MSIV allowable leak rate of $\leq$100 scfh for any one valve and $\leq$200 scfh for a combined maximum pathway leakage for all four main steam lines and the deletion of MS-PLCS and PVLCS will not adversely affect the performance of the primary containment isolation function. The radiological analysis demonstrates that the proposed changes do not result in an increase to the dose exposures previously calculated for a design basis LOCA. The revised LOCA dose exposures remain bounded by the guidelines of 10CFR100 for the off-site doses and 10CFR50, Appendix A (General Design Criteria 19) for the Control Room doses.

The whole body (DDE) dose at the Low Population Zone (LPZ) is 2.82 Rem and the Control Room is 0.43 Rem. These values are acceptable since the revised doses are bounded by the Regulatory Guidelines (2.82 versus 25 Rem at the LPZ and 0.43 versus 5 Rem at the Control Room). The associated whole body (DDE) dose at the exclusion area boundary (EAB) is 4.69 Rem which remains bounded by the Regulatory Guideline of 25 Rem.

The thyroid CEDE dose at the LPZ is 62.58 Rem. This is acceptable since the revised dose of 62.58 Rem is significantly less than the Regulatory Guideline (300 Rem). The EAB thyroid CEDE dose is 37.53 Rem, whereas the Control Room thyroid CEDE dose is 11.18 Rem. These values are also acceptable since the revised doses are bounded by the Regulatory Guidelines (37.53 versus 300 Rem at the EAB and 11.18 versus 30 Rem at the Control Room). The Control Room beta (SDE) dose is 9.15 Rem which remains bounded by the Regulatory Guideline of 30 Rem.

4.4 General

The proposed changes will reduce unnecessary MSIV and PVLCS (secondary containment bypass) valve repair costs, avoid unnecessary dose exposure to maintenance and test personnel, reduce outage durations, and extend the effective service life of these containment isolation valves. In addition, the proposed increase in the MSIV leakage limit has potential to significantly reduce recurring valve leakages and minimize the possibility of needless repair which can compromise plant safety.

Both the MS-PLCS and PVLCS leakage control systems are extremely difficult to maintain. The systems contain extensive logic and instrumentation requiring frequent calibration to meet the Technical Specification requirements. The systems have been declared inoperable and required entry into the Limiting Condition of Operation (LCO) on numerous occasions. Additionally, replacement of safety-related environmentally qualified parts has become increasingly difficult with excessive lead times.

The existing leakage control systems have limitations for mitigating leakage. Operation of the systems induces higher leakage by increasing the differential pressure across the valves and is inconsistent with the philosophy of multiple barriers for limiting fission product releases. In actuality, operation of the systems increases containment pressure and thereby increases the containment leakage. Deletion of the PVLCS and MS-PLCS eliminates the concern for containment in-leakage as the source of this leakage is removed. The MS-PLCS and PVLCS systems require multiple logic controls, interlocks, timers, isolation valves, and other equipment to supplement containment integrity. Based on plant operating experience, the MS-PLCS and PVLCS systems do not provide a high degree of reliability as stated above. Additionally, the leakage control systems have limited capacity and MS-PLCS and PVLCS do not function at moderate leakage rates above 100 scfh.

Entergy will institute provisions in the Primary Containment Leakage Rate Testing Program to ensure that any MSIV exceeding the proposed 100 scfh limit will be repaired and retested to meet a leakage rate of less than or equal to 37.5 scfh. This will assure continuation of high quality repair and refurbishment efforts to improve the overall performance and reliability of the MSIVs.

The proposed change removes the requirements for 33 LCS MOVs from the PVLCS and MS-PLCS (in addition to other valves and components). These MOVs will not serve as PCIVs and as such, have no safety significance. Although some of these MOVS will be retained as normal system isolation or maintenance valves, they will have no safety or leakage control function. Consequently, their thermal overloads require no bypassing. Entergy intends to disconnect, remove and/or cap and abandon the PVLCS and MS-PLCS systems. The PVLCS skids will be physically removed. LCS lines connecting to the secondary containment bypass and main steam line PCIVs will be disconnected and welded/capped closed such that containment integrity is maintained. The welding and post-weld examination procedures will be in accordance with the American Society of Mechanical Engineers (ASME) Code, Section XI requirements. Additionally, these welds and/or caps will be periodically tested as part of the primary Containment Integrated Leak Rate Test (CILRT) program in accordance with the requirements of 10CFR50, Appendix J.

An analysis of the impact on core damage frequency (CDF) was performed for removal of the PVLCS compressors for various scenarios. Revision 2 of the RBS PSA was utilized for which the calculated average CDF is 3.55 E-06 per year. The analysis led to the following conclusions:

1) Removal of the PVLCS compressors with replacement by a new connection to the diesel-driven IAS compressor will result in an increase in CDF from 3.55E-06 per year to 3.68E-06 per year or an increase of 4%.

2) Removal of the PVLCS compressors with replacement by a connection to large accumulators such as LSV*TK6A and LSV*TK6B will result in a decrease in CDF from 3.55E-06 per year to 3.40E-06 per year or a decrease of -4.5%.

The MS-PLCS and PVLCS are judged to be of low safety significance since both function only to reduce leakage through an isolated containment following an accident. The PVLCS is also judged to be of low safety significance since the valves which it serves are required to meet specific leakage criteria and the system mitigates only a fraction of the complete containment leakage following an accident. Several studies have documented the minimal impact of increased unfiltered containment leakage, among these are NUREG-1273, "Technical Findings and Regulatory Analysis for Generic Safety Issue II.E.4," "Containment Integrity Check," NUREG/CR-3539, "Impact of Containment Building Leakage on LWR Accident Risk," and NUREG-1493, "Performance-Based Containment Leakage Test Program." These documents indicate that leakage rates significantly in excess of the allowed leakage rates would not result in significant increase in risk to the public. Entergy wishes to prevent unnecessary plant shutdowns for conditions that do not constitute significant reductions in the overall protection of the public health and safety. When considering the remaining 30 years of licensed plant life, the elimination of the PVLCS and MS-PLCS would equate to a cost savings in excess of $20 million (see Attachment 4 for Cost-Savings Summary). Consequently, this request has major cost-savings significance.

5.0 <u>No Significant Hazard Consideration</u>

In accordance with 10CFR50.92, a proposed change to the operating license (Technical Specifications) involves no significant hazard consideration if operation of the facility in accordance with the proposed change would not (1) involve a significant increase in the probability or consequences of any accident previously evaluated, (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. This request is evaluated against each of these criteria as follows:

(1) <u>The operation of River Bend Station, in accordance with the proposed amendment, will not involve a significant increase in the probability or consequences of any accident previously evaluated</u>.

The proposed amendment to delete Technical Specification 3.6.1.8 and 3.6.1.9 involves eliminating the PVLCS and MS-PLCS leakage control requirements from the Technical Specifications. As described in Sections 9.3 and 6.7 respectively, of the USAR, the PVLCS and MS-PLCS are manually initiated about 20 minutes following a design basis LOCA. Since the PVLCS and MS-PLCS are operated only after an accident has occurred, this proposed amendment has no effect on the probability of an accident.

Since MSIV leakage and operation of the PVLCS and MS-PLCS are included in the radiological analysis for the design basis LOCA as described in Section 15.6.5 of the USAR, the proposed amendments will not affect the precursors of other analyzed accidents. The PVLCS and MS-PLCS are not initiators of any previously analyzed accident. The proposed amendments result in acceptable radiological consequences of the design basis LOCA previously evaluated in Section 15.6.5 of the USAR.

The proposed amendment to Technical Specification 3.6.1.3 does not involve a change to structures, components, or systems that would affect the probability of an accident previously evaluated. A plant-specific radiological analysis has been performed to assess the affects of the proposed increase to the allowable MSIV leak rate and deletion of the PVLCS and MS-PLCS in terms of Control Room and off-site doses following a postulated design basis LOCA. This change required a revision to the existing LOCA dose analysis due to the potential leakage from the MSIVs and those valves served by the PVLCS. Additional changes were also included in the revised dose analysis to account for changes in regulatory guidance and dose methodology. Leakage from the drywell to the atmosphere through one failed MSIV and leakage from the containment to the atmosphere through the PVLCS (secondary containment bypass valves) are both assumed to begin at time zero. The model conservatively assumes that one inboard MSIV fails open at time zero and the MSIVs associated with the remaining three main steam lines are assumed to begin leakage at 2 hours with a total leak rate of 200 scfh for all four main steam lines. The design basis leak rate of the primary containment (excluding main steam lines and lines sealed by the PVLCS) is 0.26% of the containment volume by weight per 24 hours for the duration of the accident and is assumed to be released entirely to the environment initially or the secondary containment later into the accident. The leakage of 170,000 cc/hr (4298 sccm) at P_a through the containment isolation valves served by the PVLCS is considered as bypass leakage circumventing the secondary containment. The on-site and off-site doses were determined using the TRANSACT computer code which included the ICRP 30 dose conversion factors. The total off-site and on-site LOCA doses for both the airborne and liquid release pathways resulting from the proposed change are bounded by the applicable regulatory limits.

The analysis demonstrates that dose contributions from the proposed combined MSIV leakage rate limit of 200 scfh and from the proposed deletion of the PVLCS and MS-PLCS result in values bounded by the applicable regulatory limits as compared to the LOCA doses previously evaluated for the off-site and Control Room doses as contained in 10CFR100 and 10CFR50, Appendix A (General Design Criteria 19), respectively. The LOCA doses previously evaluated are discussed in Section 15.6.5 of the USAR.

The whole body (DDE) dose at the Low Population Zone (LPZ) is 2.82 Rem and the Control Room is 0.43 Rem. These values are acceptable since the revised doses are bounded by the Regulatory Guidelines (2.82 versus 25 Rem at the LPZ and 0.43 versus 5 Rem at the Control Room). The associated whole body (DDE) dose at the exclusion area boundary (EAB) is 4.69 Rem which also remains bounded by the Regulatory Guideline of 25 Rem.

The thyroid CEDE dose at the LPZ is 62.58 Rem. This is acceptable since the revised dose of 62.58 Rem is significantly less than the Regulatory Guideline (300 Rem). The EAB thyroid CEDE dose is 37.53 Rem, whereas the Control Room thyroid CEDE dose is 11.18 Rem. These values are also acceptable since the revised doses are well within the Regulatory Guidelines (37.53 versus 300 Rem at the EAB and 11.18 versus 30 Rem at the Control Room). The Control Room beta (SDE) dose is 9.15 Rem which also remains bounded by the Regulatory Guideline of 30 Rem.

In summary, the proposed changes do not result in an increase to the radiological consequences of a LOCA previously evaluated in the USAR. The revised LOCA doses are bounded by the Regulatory Guidelines. The effectiveness of the proposed request even for leakage rates greater than the proposed MSIV allowable leak rate ensures that off-site and Control Room dose limits are not exceeded.

There is no physical change to the ADS/SRVs. The PVLCS accumulator tanks remain the backup air supply to the ADS/SRV accumulators. A qualified long-term backup air supply remains but is supplied from a different source. Therefore, the proposed change does not involve a significant increase in the probability or consequences of an accident previously evaluated.

The proposed change deletes the requirements for the LCS isolation valves which are non-PCIVs. These valves are eliminated and will not be performing a safety function. The LCS lines that are connected to the PCIVs and process piping will be welded and/or capped closed to assure primary containment integrity is maintained. The welding and post-weld examination procedures will be in accordance with the American Society of Mechanical Engineers (ASME) Code, Section XI requirements. These welds and/or caps will be periodically tested as part of the primary Containment Integrated Leak Rate Test (CILRT) program in accordance with the requirements of 10CFR50, Appendix J. The proposed change does not involve an increase in the probability of equipment malfunction previously evaluated in the USAR. In fact, the proposed change reduces the probability of equipment malfunction since, upon implementation, RBS will be operated with fewer process line isolation valves and associated support equipment subjected to postulated failure. The affected LCS MOVs will be eliminated or retained as normal system isolation or maintenance valves having no safety or leakage control function thus requiring no bypassing of their thermal overloads. This proposed change has no effect on the consequences of an accident previously evaluated since the LCS lines will be welded and/or capped closed, thus assuring that primary containment integrity, isolation and leak test capability are not compromised.

Therefore, as discussed above, the proposed changes do not involve a significant increase in the probability or consequences of any accident previously evaluated.

(2) <u>The operation of River Bend Station, in accordance with the proposed amendment, will not create the possibility of a new or different kind of accident from any accident previously evaluated</u>.

The proposed amendment to Technical Specification 3.6.1.3 does not create the possibility for a new or different kind of accident from any accident previously evaluated. The BWROG evaluated MSIV leakage performance and concluded that MSIV leakage rates up to 200 scfh will not inhibit the capability and isolation performance of the valve to isolate the primary containment. There is no new modification which could impact the MSIV operability. The LOCA has been reanalyzed at the proposed maximum combined leakage rate of 200 scfh. Therefore, the proposed change does not create any new or different kind of accident from any accident previously evaluated in the USAR.

The proposed amendment to delete Technical Specification 3.6.1.8 and 3.6.1.9 does not create the possibility of a new or different kind of accident from any accident previously evaluated because the removal of the PVLCS and MS-PLCS does not affect any of the remaining systems at RBS and the LOCA has been reanalyzed with LOCA doses resulting from the proposed change remaining bounded by the applicable regulatory limits.

The PVLCS and MS-PLCS are of low safety significance as discussed in NUREG-1273, Technical Findings and Regulatory Analysis for Generic Safety Issue II.E.4.3, "Containment Integrity Check," and NUREG/CR-3539, "Impact of Containment Building Leakage on LWR Accident Risk."

The proposed change to eliminate the LCS does not create the possibility of a new or different kind of accident from any accident previously evaluated because the removal of the LCS does not adversely affect any of the remaining RBS systems or change system inter-relationships. The associated proposed changes to delete the LCS isolation valves does not create the possibility of a new or different kind of accident. The affected LCS MOVs will be eliminated or retained as normal system isolation or maintenance valves having no safety or leakage control function thus requiring no bypassing of their thermal overloads. The PVLCS and MS-PLCS connections to the process piping will be welded and/or capped closed to assure that primary containment integrity, isolation and leak testing capability are not compromised, therefore eliminating the possibility for any new or different kind of accident.

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

(3) **The operation of River Bend Station , in accordance with the proposed amendment, will not involve a significant reduction in a margin of safety**.

The proposed amendment to Technical Specification 3.6.1.3 does not involve a significant reduction in a margin of safety. The allowable leak rate limit specified for the MSIVs is used to quantify a maximum amount of bypass leakage assumed in the LOCA radiological analysis. Results of the analysis demonstrate calculated doses, assuming the two single active failures of one MSIV to close and one diesel generator to respond are bounded by the requirements of 10CFR100 for the off-site doses and 10CFR50, Appendix A (General Design Criteria 19) for the Control Room doses. The calculated whole body doses are significantly reduced at the LPZ, the Control Room, and the EAB. The calculated thyroid dose is significantly reduced at the LPZ, the Control Room, and the EAB.

The proposed amendment to delete Technical Specification 3.6.1.8 and 3.6.1.9 for the PVLCS and MS-PLCS, does not reduce the margin of safety. In fact, the overall margin of safety is increased. The method is effective to reduce dose consequences of MSIV and the PVLCS leakage over an expanded operating range and will, thereby, resolve the safety concern that the PVLCS and MS-PLCS will not function at leakage rates higher than their design capacity. The method is consistent with the philosophy of protection by multiple leak-tight barriers used in containment design for limiting fission product release to the

environment. Therefore, the proposed method is highly reliable and effective for MSIV leakage and deletion of the PVLCS and MS-PLCS.

The calculation shows that MSIV leakage rates up to 100 scfh per steam line would not exceed the regulatory limits. Therefore, the proposed method provides a substantial safety margin for mitigating the radiological consequences of MSIV leakage beyond the proposed Technical Specification leak rate limit of 200 scfh for all four main steam lines (combined maximum pathway).

Minor increases in containment leakage such as the leakage through the MSIVs, as identified in NUREG-1273, NUREG/CR-3539, and NUREG-1493 have been found to have no significant impact on the risk to the public. Therefore, the proposed change does not result in a significant reduction in a margin of safety.

The backup air supply to the ADS/SRV accumulators remains the PVLCS accumulator tanks. The long-term backup air supply to the ADS accumulators is still from a qualified source and has been evaluated not to increase risk to the general public; therefore, the change does not involve a reduction in the margin of safety.

The proposed change to delete the LCS isolation valves does not reduce the margin of safety. Welded and/or capped closure of the LCS lines assure that primary containment integrity and leak testing capability are not compromised. The affected LCS MOVs will be eliminated or retained as normal system isolation or maintenance valves having no safety or leakage control function thus requiring no bypassing of their thermal overloads. The PVLCS and MS-PLCS connections to the process piping will be welded and/or capped closed to assure that primary containment integrity, isolation and leak testing capability are not compromised, therefore eliminating the possibility for a significant reduction in the margin of safety.

Therefore, as discussed above, the proposed changes do not involve a significant reduction in a margin of safety.

Therefore, as discussed above, the proposed amendment to the Technical Specifications does not involve a significant hazard consideration.

6.0 Environmental Impact Consideration

Entergy has reviewed this request against the criteria of 10CFR51.22 for environmental considerations. This regulation allows for a categorical exclusion provided that (i) the amendment involves no significant hazard consideration, (ii) there is no significant change in the amounts of any effluents that may be released off-site, and (iii) there is no significant increase in individual or cumulative occupational radiation exposure.

This request has been determined by Entergy not to involve a significant hazard consideration. The change will continue to allow for timely and accurate determination of the radiological plant effluents and will not affect the amounts or types of effluents. The

requested change would not increase the individual or cumulative occupational radiation exposure.

Entergy 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.

7.0 Schedule for Attaining Compliance

Based on the refueling outage safety improvement and significant resource savings that can be realized by implementing this proposed change, Entergy is requesting that this application be reviewed on a schedule sufficient to support the seventh refueling outage (RF-7) currently scheduled to commence September 12, 1997.

8.0 Notification of State Personnel

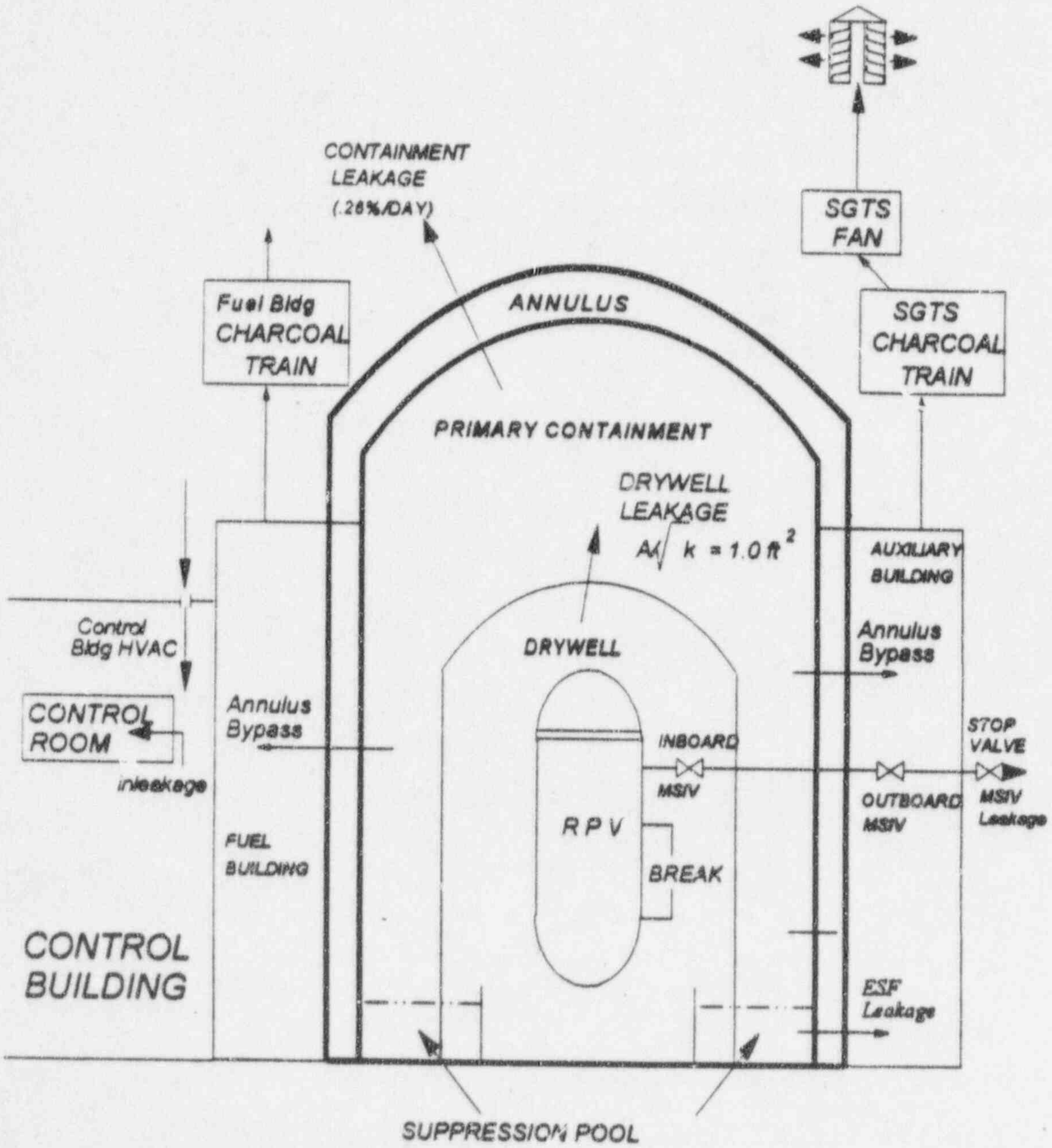
A copy of this amendment request has been provided to the State of Louisiana, Department of Environmental Quality - Radiation Protection Division.

RESULTS
LOCA DOSE CONSEQUENCES

Receptor Location	Dose Consequences		
	Whole Body (Rem)	**Skin (Rem)**	**Thyroid (Rem)**
EAB Dose Limit	25	---	300
Design Basis Dose	4.69	N/A*	37.53
LPZ Dose Limit	25	---	300
Design Basis Dose	2.82	N/A*	62.58
CR Dose Limit	5	30	30
Design Basis Dose	0.43	9.15	11.18

*Beta skin dose not applicable since no regulatory limit has been established.

LOCA FLOW PATHS



COST-SAVINGS SUMMARY
MS-PLCS/PVLCS DELETION

ORGANIZATION	TOTAL SAVINGS ($) - a
Maintenance (3)	
TOTAL	*3,220,000*
Operations (2)	
TOTAL	*420,000*
Design Engineering - a	
TOTAL	*166,500*
Outage Management (3)	
TOTAL	*9,150,000*
System Engineering (2)	
TOTAL	*2,295,000*
System Engineering (3)	
TOTAL	*1,300,000*
MOV 89-10 Program - 33 MOVs (3)	
TOTAL	*808,500*
MOV Signature Test - 33 MOVs (5)	
TOTAL	*359,700*
MOV Refurbishment - 33 MOVs (6)	
TOTAL	*631,125*
ISI (4)	
TOTAL	*210,000*
ISI (6)(b)	
TOTAL	*450,000*
IST (1)	
TOTAL	*480,000*
IST (3)	
TOTAL	*692,000*
GRAND TOTAL	*$20,182,825*

Performance Frequency

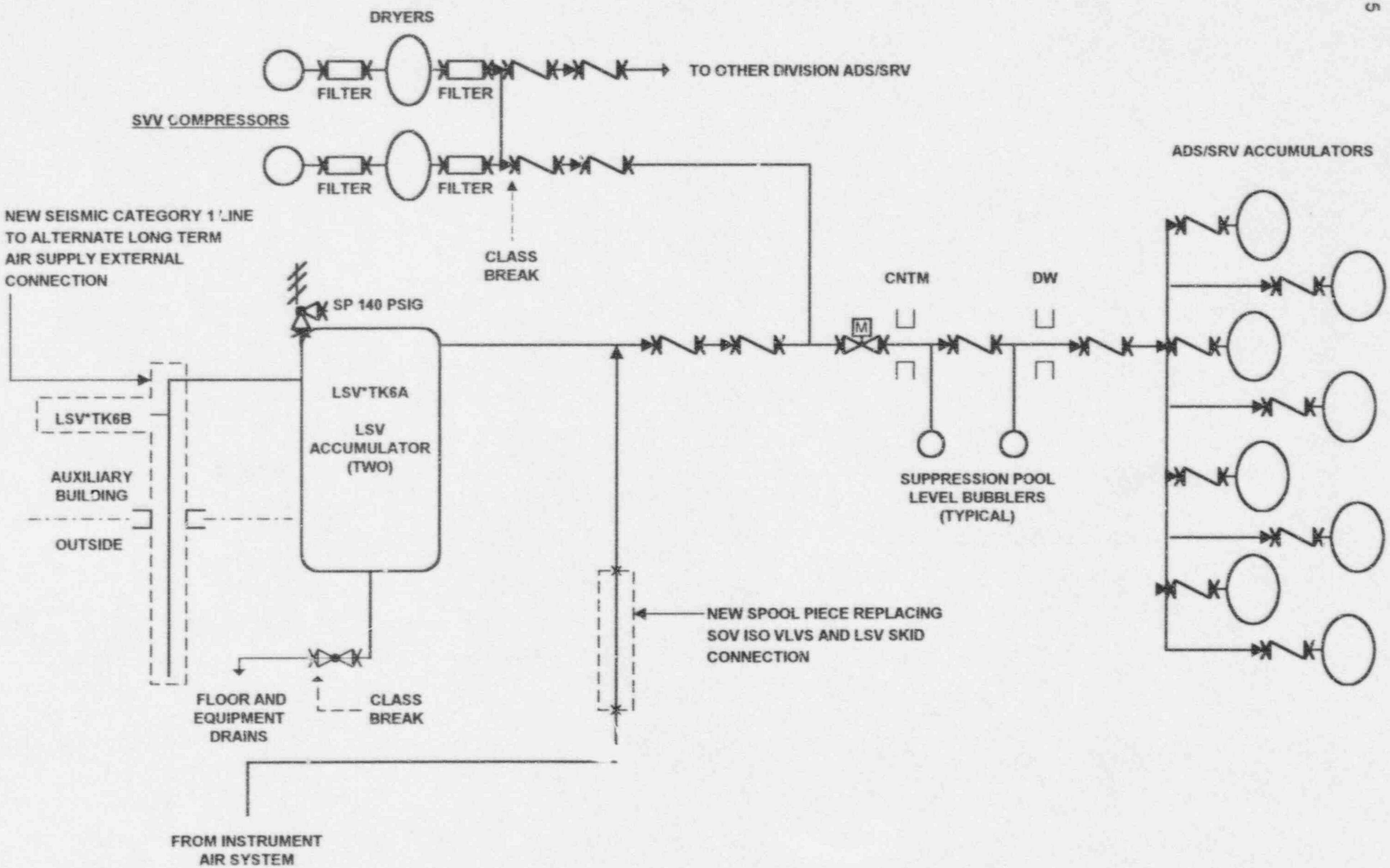
(1) Quarterly
(2) Annually
(3) 18 Months (Refueling)
(4) 3 Years
(5) 7 Years
(6) 10 Years

Notes

a - Remaining licensed life of plant (30 years)
b - Based on Industry Standard values

ESTIMATED COST BASIS :
Manhours based on $50/mhr
Person Rem based on $15K/Rem
Critical-path based on $25K/hr

PROPOSED ADS/SRV ALTERNATE LONG TERM AIR SUPPLY CONCEPTUAL DESIGN



PROPOSED TECHNICAL SPECIFICATION CHANGES

PROPOSED TECHNICAL SPECIFICATION CHANGES

The following are the specific changes to the Technical Specifications which are requested:

1- **SR 3.6.1.3.8**
DELETE

2- **SR 3.6.1.3.9**
DELETE *reference to "equipped with PVLCS"*

3- **SR 3.6.1.3.10**
PROPOSED *replace existing text with:*

> *Verify leakage rate for each MSIV is $\leq$100 scfh and the combined maximum pathway leakage rate for all four main steam lines is $\leq$200 scfh when tested at $\geq P_a$. However, if any MSIV exceeds 100 scfh, the acceptance criterion for its retest shall be in accordance with the Primary Containment Leakage Rate Testing Program.*

4- **3.6.1.8**
DELETE

NOTE: SR 3.6.1.8.1 (applicable air pressure) and SR 3.6.1.8.2 (applicable portions) will be MOVED to Technical Requirements Manual (TRM).

5- **3.6.1.9**
DELETE