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January 21,2003

U.S. Nuclear Regulatory Commission
Attention: Document Control Desk
Washington, DC 20555

Subject: Report of 10CFR50.59 Safety Evaluations and Commitment Changes – September 01, 2001 through October 31, 2002
Grand Gulf Nuclear Station
Docket No. 50-416
License No. NPF-29

GNRO-2003/00001

Ladies and Gentlemen:

Pursuant to 10CFR50.59(d)(2), Entergy Operations, Inc. hereby submits the summary of 10CFR50.59 evaluations for the September 01, 2001 through October 31, 2002 period. Also attached is the summary of commitment changes for the same period made in accordance with NEI 95-07 Guidelines.

We are now submitting the 10CFR50.59 evaluations summary on a more frequent basis than that required by 10CFR50.59(d)(2). This change has been made to improve the timeliness of information provided to the NRC and to take advantage of recent changes made by the NRC in the area of electronic transmittal of information. If further information is required, please contact this office.

This letter contains no commitments.

Yours truly,

CAB/GWI;gwi

attachments:
1. Table of Contents
2. 10CFR50.59 Evaluations and Commitment Change Evaluation Summary

cc: (See Next Page)

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MEANING OF ACRONYMS			
ARI	Alarm Response Instruction	LOP	Loss of Power
ASTM	American Society for Testing and Materials	MAPL HGR	Maximum Average Planar Linear Heat Generation Rate
CCE	Commitment Change Evaluation	MCPR	Minimum Critical Power Ratio
CMWT	Core Megawatts Thermal	MNCR	Material Nonconformance Report
CR	Condition Report	MOV	Motor Operated Valve
DCP	Design Change Package	MS	Mechanical Standard
EP	Emergency Procedure	MSIV-LCS	Main Steam Isolation Valve Leakage Control System
EPI	Equipment Performance Instruction	NPE	Nuclear Plant Engineering
EPRI	Electric Power Research Institute	NSSS	Nuclear Steam Supply System
ER	Engineering Request	PDMS	Plant Data Management System
ES	Electrical Standard	PPM	Parts Per Million
ESF	Engineered Safety Feature	PRA	Probabilistic Risk Assessment
GE	General Electric	PSW	Plant Service Water
GG	Grand Gulf	RCIC	Reactor Core Isolation Cooling
GGN	Grand Gulf Nuclear	RFO	Refueling Outage
GPM	Gallons Per Minute	RHR	Residual Heat Removal
IOI	Integrated Operating Instruction	RPV	Reactor Pressure Vessel
ISI	In Service Inspection	SCN	Standard Change Notice
IST	In Service Testing	SERI	System Energy Resources, Inc.
LBDC	License Basis Document Change	SGTS	Standby Gas Treatment System
LDC	License Document Change	SOER	Significant Operating Experience Report
LHGR	Linear Heat Generation Rate	SSW	Standby Service Water
LLRT	Local Leak Rate Test	TRM	Technical Requirements Manual
LOCA	Loss of Coolant Accident	UHS	Ultimate Heat Sink

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2002-0001-R01	ER-GG-2000-0083-000, Revision 2	10
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2002-0002	AECM-1986/00319 & AECM-1987/00169	31
2002-0006	AECM-1980/00026	35

Notes: 1. May not have been previously transmitted.

BRIEF DESCRIPTION OF CHANGE, TEST OR EXPERIMENT:

SERI-MS-38, Revision 0, provides GGNS specific quality requirements for the Instrument Air System (P53) and for Diesel Generator Starting Air (P75) and (P81). SERI-MS-38 incorporates the requirements established by the Instrument Society of America (ISA) as described in ISA-S7.3-1975, 1981 Revision, and "Quality Standard for Instrument Air". ISA S7.3 defines industry acceptable levels for oil content, dew point temperature, and particulates. Members of the ISA Standards and Practices Board as of May 1981 included Bechtel Power Corporation and General Electric. ISA S7.3 is also recommended for adoption by the nuclear power industry per SOER 88-01.

REASON FOR CHANGE, TEST OR EXPERIMENT:

Letter Number AECM 89/0032 was written in response to Generic Letter 88-14. The AECM committed to develop a GGNS instrument air system quality standard based on overall system and component needs. SERI-MS-38, Revision 0, "Quality Standard for Instrument Air System and For Diesel Generator Starting Air" was developed to meet the AECM commitment for an instrument air system quality standard.

50.59 EVALUATION SUMMARY AND CONCLUSIONS:

SERI-MS-38 establishes limits in instrument and diesel starting air for oil content concentration, dew point temperature, and maximum entrained particle size for safety-related and non-safety-related components. The standard exempts GGNS from the requirement to monitor for corrosive contaminants and hazardous gases based on discussions in NUREG 1275, Volume 2, and SOER 88-01. It also establishes requirements for compressed air monitoring frequency and preventive maintenance requirements.

The adoption of a maximum 1 ppm oil content concentration is not considered to constitute a change to the previously governing GE Specification 22A2738 since the specification did not define "oil free" air. The adoption of ISA S7.3 dew point temperature requirements is a relaxation from the -40 degrees Fahrenheit in the GE specification. This relaxation does not adversely affect qualified equipment life and component operability or require design specification revision since adherence to SERI-MS-38 will prevent condensate accumulation and arrest oxidation of iron or steel components. The SERI-MS-38 maximum entrained particle size of 3.0 microns for safety-related components and 50 microns for non-safety-related components is a change from the previously allowed 50 microns for both safety-related and non-safety-related components per the GE specification. The new 3.0 micron limit meets ISA S7.3 requirements but is a negligible increase to the 0.9 micron particle size referred to in the FSAR. The FSAR will be revised to reflect the new absolute rating of 3.0 microns.

SERI-MS-38 does not require a change to the technical specifications because the quality of instrument and diesel generator starting air is not addressed in the GGNS Unit One technical specifications.

Adherence to SERI-MS-38 will ensure air quality to support safe shutdown of the plant and will maintain the designed reliability of components needed to mitigate the consequences of an accident. Therefore, adherence to SERI-MS-38 will not increase the probability of occurrence or the consequences of an accident previously evaluated in the FSAR.

Adherence to SERI-MS-38 will not affect diesel generator starting systems redundant equipment design or configuration and will not affect instrument air system piping forming part of the containment boundary. Compliance with the standard will ensure the quality of compressed air for starting the diesel generators and for operation of safety-related components and equipment important to safety served by instrument air. Therefore, adherence to SERI-MS-38 will not increase the probability or the consequences of a malfunction of equipment important to safety previously evaluated in the FSAR. It will not create the possibility of an accident of a different type than any already evaluated in the FSAR.

SERI-MS-38 does not require physical modification to any compressed air systems and does not affect any compressed air system operation with the exception of periodic air quality sampling and air system component inspections. Therefore, adherence to SERI-MS-38 will not create the possibility of a malfunction of equipment important to safety different than previously evaluated in the FSAR.

Since the technical specifications do not address instrument and diesel generator starting air quality, they do not establish any margin of safety based on the quality of instrument or diesel generator starting air. Therefore, adherence to SERI-MS-38 will not reduce any inherent margin of safety used as the basis for any technical specification.

Therefore, no technical specification change is required and no unreviewed safety questions are created as a result of SERI-MS-38 adherence.

BRIEF DESCRIPTION OF CHANGE, TEST OR EXPERIMENT:

ER-GG-2000-0892-000, Revision 0, is providing on-line leak repair instructions for Reactor Water Clean Up System (RWCU) Outboard Isolation Valve Q1G33F039 and Inboard Isolation Valve Q1G33F040.

REASON FOR CHANGE, TEST OR EXPERIMENT:

Valves Q1G33F039 and Q1G33F040 have pressure seal leakage and are emitting steam from some of the eight pressure seal ring knockout holes in the valve body. The valves have been previously drilled and tapped at the bottom of the pressure seal ring and Furmanite sealant compound has been injected to try and fill any voids to control leakage. These efforts have not been completely successful. The Q1G33F039 leakage has been controlled for only a short period of time; then leakage re-occurred. Based on information obtained from the valve vendor, the pressure seal ring could be located as much as ¼" lower than the locations the vendor had initially recommended for drilling and injecting. The attempt to control leakage on the Q1G33F040 was also not successful due to the unavailability of a clear pathway for the sealant compound to travel to the voids. This repair will consist of drilling and tapping the segment ring knockout holes on the Q1G33F039 Valve and injecting Furmanite sealant compound downstream of the leakage. This repair method will allow the Furmanite compound to dam on the downstream side of the pressure seal ring and fill the void areas near the valve bonnet and should provide a means of controlling the leakage.

The QIG33F040 Valve will have the same repair methods implemented except this valve body has already had the segment ring holes drilled and tapped to facilitate the installation of on-line sealant shutoff adaptors. It should be noted that a similar on-line leak seal was successfully performed in the past on this valve utilizing the segment ring knockout holes.

These repair instructions governed by this 50.59 will be valid until the first forced outage of sufficient duration to allow the final repair of the valve or until RFO 11.

The repair described in this ER is different than that recommended by EPRI and is being utilized because the initial attempts to inject the seal ring from below (as recommended by EPRI) have been unsuccessful. Backside injection requires reevaluation of the bonnet bolt stresses for additional loads.

SAFETY EVALUATION SUMMARY AND CONCLUSIONS:

Valves Q1G33F039 and Q1G33F040 are the RWCU Return to RHR and Feedwater Outboard and Inboard Containment Isolation Valves and are located in the Auxiliary and Containment Steam Tunnels. These valves currently have pressure seal ring leakage and are emitting steam from some of eight valve body segment ring knockout holes located downstream of the pressure seal ring. The valve will be injected with on-line sealant compound by utilizing the valve segment ring knockout holes, which will allow Furmanite compound to be injected on the downstream side of the pressure seal ring. This location for the shutoff adaptors should allow the on-line sealant compound to fill the valve body voids between the bonnet and downstream side of the pressure seal ring and control the current leakage. This repair will be valid until the

first forced outage of sufficient duration to allow the final repair/rework of the valves or until RFO 11 when the final repair will be implemented.

A calculation was performed utilizing a maximum system pressure of 1445 psig to evaluate the bonnet and yokearm bolting stresses. This evaluation has shown that based on the maximum operating pressure of 1445 psig and utilizing the highest conservative as left valve stem thrust force of 29051, where applicable, for Valves Q1G33F039 and Q1G33F040 this repair will maintain the valves safety-related operation and containment isolation function following the injection of on-line sealant through the segment ring holes. A limited amount of the sealant will be injected into the valves to seal around the pressure seal ring. Injection of the sealant compound on the downstream side of the pressure seal ring is an alternate location, which is not usually used for performing on-line leak repairs, but has been evaluated and found to be acceptable for use on Valves Q1G33F039 and Q1G33F040. This allows damming the sealant by utilizing the voids between the valve body and valve bonnet. The installation of the shutoff adaptors and injection of the Furmanite sealant compound will not adversely affect the structural integrity of the valves. Furmanite evaluations has also evaluated the valve body stresses for adding the shutoff adaptors and for injection pressures in Furmanite Procedures No. N-2000263 and N-2000264. The Furmanite and GGNS calculations have shown all stresses in the valves will remain within ASME Section III code allowables for Class 2 valves. Also the actual injection pressure of the Furmanite compound inside each valve body will be held to 1250 psig, which is less than the design pressure of 1500 psig for the valves as specified on Vendor Drawing M-242.0-Q1-1.2-117, Rev. 5. After the injection of the sealant, the valves will be partially stroked to verify the stem movement in the close safety direction. The Q1 G33F039 and Q1G33F040 Valves will therefore be capable of performing their safety-related function as primary containment isolation valves and will not increase the possible offsite radiation dose, and therefore not affect the health and safety of the public.

While the valve repair may change the operational load path and affect the non-code portions of the valve, it does not alter the original valve design because the load path that prevents disassembly of the valve's pressure boundary during operation or accident conditions remains the same. As in the original design, the segmented thrust ring still provides the positive locking mechanism that retains the bonnet inside the valve body. The pressure seal ring (a gasket) is being partially or completely replaced with an injected sealant that depends on non-code portions of the valve to retain the sealant in position similar to a packing gland assembly. Based on evaluations it is the position of Central Engineering Programs and GGNS Design Engineering that the code boundary is unaffected by the described repair. However, the repair does alter the stresses on the non-code portions of the valve and a thorough evaluation of that effect on the integrity of the valve's actuator assembly has been performed per calculation NPE-G33F039/F040, Revision 11.

Calculation NPE-G33F0391F040, Revision 11, assumed that the on-line leak sealant would apply even stresses to the body-to-yokearm bolting. This assumption is acceptable since non-uniform filling will result in minor stress variation that will not challenge margins provided in code allowable stress.

The final repair/rework of the valves will consist of replacing the valves with like-for-like components or if not possible removing the shutoff adaptors from the segment ring holes and removing the Furmanite compound from the valve body and other components. Also, final repair/rework will be made to the valve bodies to close valve body openings made by prior injection attempts per ER-GG-2000-0880-000, Revision 0, response instructions. This final repair/rework will consist of inserting a safety related threaded gland plug into the hole and installing a seal weld for leak tightness.

This on-line repair will not affect the pipe break accidents identified in UFSAR Appendix 3C, Section 3C.2.2, since the valves will maintain their original design function. Also, this repair will not affect the missile evaluations identified in UFSAR Section 3.5. This repair is not creating any new missiles since the shutoff adaptors are similar to the nut-bolt combinations discussed in UFSAR 3.5.1.2.2.1 that only have a small amount of stored energy and thus are of no concern as potential missiles.

One risk involved in performing a leak repair is injecting too much sealant into a valve to seal a leak. ER-GG-2000-0892-000, Revision 0, will administratively control the amount of sealant as well as the pressure being injected into the valve. Controlling the amount of sealant and pressure ensures valve component stresses will not be increased to values higher than code allowable stresses and that the sealant will not be introduced into the piping, in a manner that could cause the piping to be plugged or excessive sealant to be injected into the reactor vessel.

BRIEF DESCRIPTION OF CHANGE, TEST OR EXPERIMENT:

In 1996, DCP 88/0249 modified the Inboard MSIV-LCS. This modification de-energized the Inboard Blower (E32-C00I), Inboard Heaters (E32-B00IA, E, J, N), the dilution air flow trip and alarm, the short term depressurization valves, and the inboard flow element trips and alarm. Basically, this modification made the Inboard MSIV-LCS much more passive. During the Design Basis Documentation (DBD) process, Chapter 7 of the UFSAR was found to contain information that should have been changed when DCP 88/0249 was done. Additionally, the review found that the documentation related to the requirements for the E32F003A, E, J, and N valves (leaving them open vice the original closed position) were not specifically addressed in the original 50.59. To ensure adequate documentation and because the 50.59 rule has changed, a new 50.59 addressing the modification is done. This will consolidate the information into one document and address any additional changes needed in the UFSAR; i. e., revising UFSAR Table 6.7-1 (Single Failure analysis Table) and specifically addressing missing changes to UFSAR Chapters 7 & 8. These items were not addressed by Safety Evaluation 94-086-R00.

REASON FOR CHANGE, TEST OR EXPERIMENT:

This 50.59 deals with items not addressed by Safety Evaluation 94-086-R00 (50.59 for DCP 88/0249).

SAFETY EVALUATION SUMMARY AND CONCLUSIONS:

DCP 88/0249's modification meets the requirements of Reg. Guide 1.96. The original 50.59 (94-086-R00) dealt with establishing the Outboard MSIV-LCS as the primary leakage control system which is discussed in UFSAR Chapter 7. Establishing the Outboard MSIV-LCS as the Primary MSIV-LCS provides for two MSIVs per steam lines as radiological barriers before reaching the leakage control system and assists in meeting the single failure criteria for MSIV-LCS. If Division II (Outboard MSIV-LCS) is unavailable, then the single failure has occurred (DIV II failure) and then Division I (Inboard MSIV-LCS) can be initiated and the MSIVs will be leaking at their technical specification rates for offsite dose assumptions. With a primary and secondary MSIV-LCS initiation established, the need for the inboard high MSIV leakage flow trip is redundant and can be deleted and not change any failures associated with Reg. Guide 1.96 Guidelines. The single failure of a check valve on the Inboard MSIV-LCS and outboard blower suction is being added to UFSAR Table 6.7-1. The check valve failure, either open or closed, has no impact on the operation of either MSIV-LCS. There are no new single failures created as a result of changing UFSAR Chapters 7 and 8. There are no changes to the inboard MSIVs divisional power (Div I). The passive Inboard MSIV-LCS directs MSIV leakage to the Auxiliary Building through the bleed-off valves (E32F003A, E, J, N) in order to be processed by SGTS. The functionality of SGTS is unchanged by the modified Inboard MSIV-LCS. The Inboard MSIV-LCS is still designed to be manually actuated within 20 minutes post-LOCA if the Outboard MSIV-LCS is unavailable for processing MSIV leakage. The vessel pressure and steam line pressure interlocks are provided to prevent inadvertent operation and are unchanged by the modification.

BRIEF DESCRIPTION OF CHANGE, TEST OR EXPERIMENT:

This modification removes the RHR injection point through the reactor vessel head spray nozzle. This change deletes the head spray mode of operation of the RHR System. The head spray mode of operation is not credited for accident mitigation in the safety analysis. The purpose of the head spray mode was reactor level control and was not used at GGNS. Portions of the piping that connected to the reactor vessel head tee are being removed. A blind fitting (i. e., flange) will be installed at the vessel tee. The RHR piping and leak-off line for the E51F066 Check Valve is being removed (SCN 01-0012A to MS-02). The RHR/RCIC Inboard Containment Isolation Check Valve, E51F066, is being removed (SCN 01-0001A to 9645-M-242.*2).* Hardware associated with the RHR piping in the insulation head is being removed. The containment penetration for this line located in the vessel cavity will be isolated. Portions of the RHR piping from the bulkhead to the containment penetration (No. 18) are being removed. The change electronically removes power and controls from the E12F394 Valve and the valve is being left in place as a manual locked closed valve (see SCN 01-0003A to MS-25 and SCN 01/0001A to ES-18). Also, the electrical power and controls will be removed from the valve stem leak detection circuits for both the E12F394 and E51F066 Valves. All power and control cables will be lifted at both ends and will be relabeled as SPARE cables. The relay that closes the E12F394 Valve upon an isolation signal from the NSSS system will be removed and the electrical circuit, from which it is being removed, will be functionally tested as required by the ER. With the head spray line removed this valve no longer needs to be motor operated. Changing the valve to a manual valve removes MOV testing requirements (SCN 01-0003A to MS-25 and SCN 01-0001A to ES-18). An ASME Class 2, welded cap (i. e., blind fitting) will be installed in-line on the Auxiliary Building side of Penetration 18 to isolate the containment penetration. AC Electrical Power System Calculations EC-Q1111-90016, Rev. 13, Supplement 1; E0045Q, Rev. 1, Supplement 4; and EC-Q1111-90028, Rev. 5, Supplement 2, reflect the removal of equipment, cable and voltage requirements associated with E12F394 and are evaluated as part of E12F394 removal. E12F395 will change to a locked closed manual valve. The function of the E12F023 Valve after the modification will be to allow fire water injection into the reactor vessel via the test connection containing Valves E12F061 and E12F062 (see Attachment 26 of Procedure 05-S-01-EP-2).

REASON FOR CHANGE, TEST OR EXPERIMENT:

The change saves outage time and dose spent in pipe removal and assembly of the head spray line. It also eliminates a mode of operation that is no longer used. Also, a benefit is the elimination of local leak rate testing of Containment Isolation Valves E12F394 and E12F023. An additional benefit of the modification is the removal of a GL 96-06 penetration over pressurization concern for Penetrations 18 and 311.

SAFETY EVALUATION SUMMARY AND CONCLUSIONS:

This change has no direct input to any UFSAR Chapter 15, Accident *Analysis* and supports the current operating modes of the RHR systems. As a result, this change does not increase the frequency of occurrence of an accident evaluated previously in the UFSAR. The optional RHR head spray flow path is being isolated by a locked closed Valve E12F394 and blind fitting at

Penetration 18 and no longer provides RPV head spray. This optional flow path is not required to meet current UFSAR nominal shutdown requirements. The penetration is currently used as an optional flow path requirement given in Emergency Procedure 05-S-01-EP-2, Attachment 25, Injection into RPV with Condensate Transfer. This flow path will be deleted from the attachment. This flow path through the head spray is not credited for accident mitigation in the safety analysis. In the EPs, currently RHR B has five flow paths for condensate transfer injection. With the removal of head spray, RHR B will have four flow paths for condensate transfer injection which is how many RHR A has available in the same EP attachment.

This change is designed to meet the same safety- and non-safety-related pressure boundary, seismic, and tornado protection requirements established in the design and licensing basis for the RHR A system. All essential plant systems and equipment will function as assumed in the *Accident Analysis.* Therefore, this change will have no effect on any consequences of the accidents evaluated previously in the UFSAR, will not change offsite dose to the public, will not affect any fission product barriers, and does not alter any assumptions previously made in evaluating the radiological consequences of an accident described in the UFSAR. As a result, this change will not increase the consequences of an accident or create an accident of a different type than any evaluated previously in the UFSAR.

All essential and nonessential plant systems and equipment are designed to meet the current licensing and design requirements. This modification does not change the RHR system actuation, flow parameters, or the pressure boundary requirements and they will function as assumed in the *Accident Analysis.* Therefore, this change will not increase the consequences of a malfunction of equipment important to safety or create the possibility of a malfunction of equipment important to safety of a different type than any evaluated previously in the UFSAR. Also, this change will not reduce the margin of safety as defined in the basis for any technical specification.

Based on the results of this safety evaluation, the effects associated with this modification are inconsequential and, therefore, do not constitute an unreviewed safety question (USQ).

This design change is internal to the plant, does not affect plant power levels, and does not affect plant influents or effluents. Therefore, this design change does not represent a change to the Environmental Protection Plan or a change that will affect the environment. There is no potential for an unreviewed environmental question; therefore, there is no need to perform an environmental evaluation.

The piping containing Penetration 18/311 will be drained to address a GL 96-06 concern about over pressurization of containment penetrations. Draining this penetration piping will remove this penetration as being subject to penetration over pressurization.

The IST and ISI Program Section will be affected by removing select safety-related components from the IST Pump and Valve Program. This criteria contains the bases used to determine if components are within the scope of ASME Operation and Maintenance (OM) Standard, 1987 Edition with OMa-1988 Addenda, Parts 1, 6, and 10 as required by 10 CFR 50.55a. For valves (ASME OM Part 10), the scope is limited to certain active and passive valves and relief devices (and their actuating and position indicating systems) that are required to perform a specific

function in shutting down the reactor to the cold shutdown condition or in mitigating the consequences of an accident. The valves, E5IF066, E12F395, E12F394, E12F061, E12F062, E12F019, and E12F023 are being affected by ER 2000-0083-000 and will be removed from the program.

BRIEF DESCRIPTION OF CHANGE, TEST OR EXPERIMENT:

This modification removes the RHR injection point through the reactor vessel head spray nozzle. This change deletes the head spray mode of operation of the RHR System. The head spray mode of operation is not credited for accident mitigation in the safety analysis. The purpose of the head spray mode was reactor level control and was not used at GGNS. Portions of the piping that connected to the reactor vessel head tee are being removed. A blind fitting (i. e., flange) will be installed at the vessel tee. The RHR piping and leak off line for the E51F066 Check Valve is being removed (SCN 01-0012A to MS-02). The RHR/RCIC Inboard Containment Isolation Check Valve, E51F066, is being removed (SCN 01-0001A to 9645-M-242.2). Hardware associated with the RHR piping in the insulation head is being removed. The containment penetration for this line located in the vessel cavity will be isolated. Portions of the RHR piping from the bulkhead to the containment penetration (No. 18) are being removed. The change electrically removes power and controls from the E12F394 Valve and the valve is being left in place as a manual locked closed valve (see SCN 01-0003A to MS-25 and SCN 01/0001A to ES-18). Also the electrical power and controls will be removed from the valve stem leak detection circuits for both the E12F394 and E51F066 Valves. All power and control cables will be lifted at both ends and will be relabeled as SPARE cables. The relay that closes the E12F394 Valve upon an isolation signal from the NSSS system will be removed and the electrical circuit, from which it is being removed, will be functionally tested as required by the ER. With the head spray line removed this valve no longer needs to be motor operated. Changing the valve to a manual valve removes MOV testing requirements (SCN 01-0003A to MS-25 and SCN 01-0001A to ES-18). An ASME Class 2, welded cap (i. e., blind fitting) will be installed in line on the Auxiliary Building side of Penetration 18 to isolate the containment penetration. AC Electrical Power System Calculations EC-Q1111-90016, Rev. 13, Supplement 1; E0045Q, Rev. 1, Supplement 4; and EC-Q1111-90028, Rev. 5, Supplement 2, reflect the removal of equipment, cable, and voltage requirements associated with El2F394 and are evaluated as part of E12F394 removal. E12F395 will change to a locked closed manual valve. The function of the E12F023 Valve after the modification will be to allow fire water injection into the reactor vessel via the test connection containing E12F061 and E12F062 Valves (see Attachment 26 of Procedure 05-S-01-EP-2).

REASON FOR CHANGE, TEST OR EXPERIMENT:

The change saves outage time and dose spent in pipe removal and assembly of the head spray line. It also eliminates a mode of operation that is no longer used. Also, a benefit is the elimination of local leak rate testing of Containment Isolation Valves E12F394 and E12F023. An additional benefit of the modification is the removal of a GL 96-06 penetration over pressurization concern for Penetrations 18 and 311.

SAFETY EVALUATION SUMMARY AND CONCLUSIONS:

This change has no direct input to any UFSAR Chapter 15, *Accident Analysis* and supports the current operating modes of the RHR systems. As a result, this change does not increase the frequency of occurrence of an accident evaluated previously in the UFSAR. The optional RHR head spray flow path is being isolated by a locked closed Valve E12F394 and blind fitting at Penetration 18 and no longer provides RPV head spray. This optional flow path is not required to meet current UFSAR nominal shutdown requirements. The penetration is currently used as

an optional flow path requirement given in Emergency Procedure 05-S-01-EP-2, Attachment 25, Injection into RPV with Condensate Transfer. This flow path will be deleted from the attachment. This flow path through the head spray is not credited for accident mitigation in the safety analysis. In the EPs, currently RHR B has five flow paths for condensate transfer injection. With the removal of head spray, RHR B will have four flow paths for condensate transfer injection which is how many RHR A has available in the same EP attachment.

This change is designed to meet the same safety- and non-safety-related pressure boundary, seismic, and tornado protection requirements established in the design and licensing basis for the RHR A system. All essential plant systems and equipment will function as assumed in the *Accident* Analysis. Therefore, this change will have no effect on any consequences of the accidents evaluated previously in the USAR, will not change offsite dose to the public, will not affect any fission product barriers, and does not alter any assumptions previously made in evaluating the radiological consequences of an accident described in the UFSAR. As a result, this change will not increase the consequences of an accident or create an accident of a different type than any evaluated previously in the UFSAR.

All essential and nonessential plant systems and equipment are designed to meet the current licensing and design requirements. This modification does not change the RHR system actuation, flow parameters, or the pressure boundary requirements and they will function as assumed in the *Accident Analysis.* Therefore, this change will not increase the consequences of a malfunction of equipment important to safety or create the possibility of a malfunction of equipment important to safety of a different type than any evaluated previously in the UFSAR. Also, this change will not reduce the margin of safety as defined in the basis for any technical specification.

Based on the results of this safety evaluation, the affects associated with this modification are inconsequential and, therefore, do not constitute an unreviewed safety question (USQ).

This design change is internal to the plant, does not affect plant power levels, and does not affect plant influents or effluents. Therefore, this design change does not represent a change to the Environmental Protection Plan or a change that will affect the environment. There is no potential for an unreviewed environmental question; therefore, there is no need to perform an environmental evaluation.

The piping containing Penetrations 18 and 311 will be drained to address a GL 96-06 concern about over pressurization of containment penetrations. Draining this penetration piping will remove this penetration as being subject to penetration over pressurization.

The IST and ISI Program Section will be affected by removing select safety-related components from the IST Pump and Valve Program. This criteria contains the bases used to determine if components are within the scope of ASME Operation and Maintenance (OM) Standard, 1987 Edition with OMa-1988 Addenda, Parts 1, 6, and 10 as required by 10 CFR 50.55a. For valves (ASME OM Part 10), the scope is limited to certain active and passive valves and relief devices (and their actuating and position indicating systems) that are required to perform a specific function in shutting down the reactor to the cold shutdown condition or in mitigating the

consequences of an accident. The Valves, E51F066, E12F395, E12F394, E12F061, E12F062, E12F019, and E12F023 are being affected by ER 2000-0083-000 and will be removed from the program.

BRIEF DESCRIPTION OF CHANGE, TEST OR EXPERIMENT:

This modification removes the RHR injection point through the reactor vessel head spray nozzle. This change deletes the head spray mode of operation of the RHR System. The head spray mode of operation is not credited for accident mitigation in the safety analysis. The purpose of the head spray mode was reactor level control and was not used at GGNS. Portions of the piping that connected to the reactor vessel head tee are being removed. A blind fitting (i.e., flange) will be installed at the vessel tee. The RHR piping and leak off line for the E51F066 Check Valve is being removed (SCN 01-0012A to MS-02). The RHR/RCIC Inboard Containment Isolation Check Valve, E51F066, is being removed (SCN 01-0001A to 9645-M-242.2). Hardware associated with the RHR piping in the insulation head is being removed. The containment penetration for this line located in the vessel cavity will be isolated. Portions of the RHR piping from the bulkhead to the containment penetration (No. 18) are being removed. The change electrically removes power and controls from the E12F394 Valve and the valve being left in place as a manual locked closed valve (see SCN 01-0003A to MS-25 and SCN 01/0001A to ES-18). Also, the electrical power and controls will be electrically removed from the valve stem leak detection circuits for both the E12F394 and E51F066 Valves. All power and control cables in the Auxiliary and Control Buildings will be lifted and will be relabeled as SPARE cables. The cables in the Containment and Drywell will remain terminated but will be noted in PDMS that these cables are considered as spares. These cables will be de-energized. The relay that closes the E12F394 Valve upon an isolation signal from the NSSS system will be removed and the electrical circuit, from which it is being removed, will be functionally tested as required by the ER. With the head spray line removed this valve no longer needs to be motor operated. Changing the valve to a manual valve removes MOV testing requirements (SCN 01-0003A to MS-25 and SCN 01-0001A to ES-18). An ASME Class 2, welded cap (i.e., blind fitting) will be installed in line on the Auxiliary Building side of Penetration 18 to isolate the containment penetration. AC Electrical Power System Calculations EC-Q1111-90016, Rev. 13, Supplement 1; E0045Q, Rev. 1, Supplement 4; and EC-Q1111-90028, Rev. 5, Supplement 2, reflect the removal of equipment, cable and voltage requirements associated with E12F394 and are evaluated as part of E12F394 removal. E12F395 will change to a locked closed manual valve. The function of the E12F023 Valve after the modification will be to allow fire water injection into the reactor vessel via test connection containing E12F061 and E12F062 Valves.

REASON FOR CHANGE, TEST OR EXPERIMENT:

The change saves outage time and dose spent in pipe removal and assembly of the head spray line. It also eliminates a mode of operation that is no longer used. Also a benefit is the elimination of local leak rate testing of Containment Isolation Valves E12F394 and E12F023. An additional benefit of the modification is the removal of a GL 96-06 penetration over pressurization concern for Penetrations 18 and 311.

SAFETY EVALUATION SUMMARY AND CONCLUSIONS:

This change has no direct input to any UFSAR Chapter 15, *Accident Analysis* and supports the current operating modes of the RHR systems. As a result, this change does not increase the frequency of occurrence of an accident evaluated previously in the UFSAR. The optional RHR head spray flow path is being isolated by a locked closed Valve E12F394 and a blind fitting at

Penetration 18 and no longer provides RPV head spray. This optional flow path is not required to meet current UFSAR nominal shutdown requirements. The penetration is currently used as an optional flow path requirement given in Emergency Procedure 05-S-O1-EP-2, Attachment 25, Injection into RPV with Condensate Transfer. This flow path will be deleted from the attachment. This flow path through the head spray is not credited for accident mitigation in the safety analysis. In the EPs, currently RHR B has five flow paths for condensate transfer injection. With the removal of head spray, RHR B will have four flow paths for condensate transfer injection which is how many RHR A has available in the same EP attachment.

This change is designed to meet the same safety- and non-safety-related pressure boundary, seismic, and tornado protection requirements established in the design and licensing basis for the RHR B system. All essential plant systems and equipment will function as assumed in the *Accident Analysis.* Therefore, this change will have no effect on any consequences of the accidents evaluated previously in the UFSAR, will not change offsite dose to the public, will not affect any fission product barriers, and does not alter any assumptions previously made in evaluating the radiological consequences of an accident described in the UFSAR. As a result, this change will not increase the consequences of an accident or create an accident of a different type than any evaluated previously in the UFSAR.

All essential and nonessential plant systems and equipment are designed to meet the current licensing and design requirements. This modification does not change the RHR system actuation, flow parameters, or the pressure boundary requirements and they will function as assumed in the *Accident Analysis.* Therefore, this change will not increase the consequences of a malfunction of equipment important to safety or create the possibility of a malfunction of equipment important to safety of a different type than any evaluated previously in the UFSAR. Also, this change will not reduce the margin of safety as defined in the basis for any technical specification.

Based on the results of this safety evaluation, the affects associated with this modification are inconsequential and, therefore, do not constitute an unreviewed safety question (USQ).

This design change is internal to the plant, does not affect plant power levels, and does not affect plant influents or effluents. Therefore, this design change does not represent a change to the Environmental Protection Plan or a change that will affect the environment. There is no potential for an unreviewed environmental question; therefore, there is no need to perform an environmental evaluation.

The piping containing Penetrations 18/311 will be drained to address a GL 96-06 issue concerning over pressurization of containment penetrations. Draining this penetration piping will remove this penetration as being subject to penetration over pressurization.

The IST and ISI Program Section will be affected by removing select safety-related components from the IST Pump and Valve Program. This criteria contains the bases used to determine if components are within the scope of ASME Operation and Maintenance (OM) Standard, 1987 Edition with OMa-1988 Addenda, Parts 1, 6, and 10 as required by 10 CFR 50.55a. For valves (ASME OM Part 10), the scope is limited to certain active and passive valves and relief devices

(and their actuating and position indicating systems) that are required to perform a specific function in shutting down the reactor to the cold shutdown condition or in mitigating the consequences of an accident. The Valves, E51F066, E12F395, E12F394, E12F061, E12F062, E12F019 and E12F023 are being affected by ER 2000-0083-000 and will be removed from the program.

BRIEF DESCRIPTION OF CHANGE, TEST OR EXPERIMENT:

The evaluated ER will make the following changes to the non-safety portion of the Division I and II Diesel Generator Starting Air Systems (DGSAS). These changes will be made on all four trains. Replace the existing system traps (N1P75D015, D025, and D028) with timed solenoid valves. Install a new filter/separator on the discharge of the aftercooler (1P75B013). Raise the compressor suction one foot on both motor driven compressors. Revise MS-38 to allow for a less restrictive dew point requirement for the Div. I and II diesel generator starting air systems of +22 degrees F. The new dew point will meet the requirements of ISA S7.3, Quality Standard for Instrument Air. It will also authorize the like-for-like replacement of the non-safety pipe from the Compressors (N1P75C012A, B and N1P75CO13A, B) to the Dryer Towers (N1P75B012).

REASON FOR CHANGE, TEST OR EXPERIMENT:

The current traps are not effective in removing liquid water which condenses in the aftercooler and pipe leading to the dryers. This liquid saturates the desiccant in the dryers and prevents it from working. Saturated desiccant can not be regenerated. The new filter/separator and solenoid valves will prevent moisture carry over to the tower and allow it to work properly.

SAFETY EVALUATION SUMMARY AND CONCLUSIONS:

The modifications do not adversely impact the reliability of the diesel generator starting air system. The new dew point requirement is low enough to minimize the formation of corrosion in the starting air system leading to the starting air headers. The strainers prior to the headers will be regularly inspected for corrosion products. All required portions of the diesel generator starting air and control air systems are protected by strainers or filters. Additionally, the formation of rust beyond these strainers/filters is very unlikely since the materials down stream are stainless steel or copper. Additionally, the dew point in the control air system will decrease since it operates at a lower pressure than the supplied starting air.

Performing the work online was evaluated and determined not to impact the operability of the starting air system or the diesel.

BRIEF DESCRIPTION OF CHANGE, TEST OR EXPERIMENT:

In order to resolve the over pressurization issue of Generic Letter 96-06, Penetration *87/325* was initially proposed with a bypass line to relieve pressure. Due to unforeseen circumstances, the bypass line may not be a viable option. As a contingency, a Class 1 pressure relief valve (G33F267) will be installed in the Auxiliary Building Steam Tunnel. The penetration was identified in Engineering Report GGNS-97-0002, Rev. 1, and CR-GGN-1999-1147. As part of the modification, the installed bypass line will be removed.

REASON FOR CHANGE, TEST OR EXPERIMENT:

USNRC Generic Letter 96-06 raises the concern that during a postulated accident condition, some piping inside the Containment/Drywell may be heated beyond its maximum operating temperature. The concern is that water trapped in isolated piping sections (isolated by closed valves) penetrating the Containment/Drywell would thermally expand and produce extremely high pressures that could potentially challenge the piping and penetration integrity, which could affect the health & safety of the public.

Containment/drywell Penetrations 87 and *325* (RWCU suction) may need to have a relief valve installed versus relying on the pressure relieving bypass line on G33F001 installed by ER GG-1997-0022-003. This is because the thrust of G33F001 may not be able to be lowered due to LLRT concerns. As part of the modification, the installed bypass line will be removed.

The location of the relief valve is the Auxiliary Building Steam Tunnel. The reason for the Auxiliary Building Steam Tunnel versus locating it in the Drywell is as follows. There is a concern for dose due to installation in the Drywell. ALARA practices called for trying to locate the relief valve if possible in the Auxiliary Building Steam Tunnel. Another reason is leakage in the Drywell. If the relief valve should begin to leak into the Drywell this would be a high energy leakage path that could not be isolated and repaired on line. Locating the relief valve in the Auxiliary Building Steam Tunnel, there is a potential success path for isolating the leak and attempting to get the relief valve to reseat. Another reason for locating the relief valve in the auxiliary building steam tunnel is the opening of the relief valve is only 1/4"±l/8" so the opening could be as large as 3/8". If the relief valve were to fail full open, the orifice opening is 0.328" or 0.0845 inches squared. This is the limiting opening size. The leakage would be bounded by the High Energy Line Break Analysis for the Auxiliary Building Steam Tunnel and the leak could be isolated by closing G33F001, G33F004, and G33F252. The leakage should be minimal and capable of being handled by the leak detection system for the Auxiliary Building Steam Tunnel.

SAFETY EVALUATION SUMMARY AND CONCLUSIONS:

The following change meets all design basis requirements, and will provide a pressure relief mechanism and/or assure structural integrity to resolve the over pressurization issue described in GL 96-06 for Penetration *87/325.* The modification is the installation of a relief valve for line 6"-DBA-9 to provide pressure relief for Containment and Drywell Piping Penetrations 87/325. The installation of the relief valves will consist of installing a 3/4" branch connection off the main pipe with a set of flanges and a safety-related relief valve attached at its end. Since the relief valve is intended to protect the safety related piping between the isolation valves, the installed relief valve is procured to ASME Section III requirements. The relief valve is installed to protect

the safety-related portion of piping from catastrophic failure under post LOCA conditions only. Contrary to the normal function of a relief valve, the new relief valve is not intended to continuously preserve the safety-related piping pressure boundary during a normal or upset condition. For this reason, the relief valve can be considered to perform a passive function during normal system operation and is typically set to a pressure well above the operating system pressure. Normally, relief valves that experience seat leakage are inherently the valves that are challenged due to a close margin between the operating pressure of the system and the lift point set pressure of the relief valve. Additionally, this relief valve will be tested at least once every five years. During this testing the valve will be "as left" leakage tested at 90% of the set pressure prior to being installed back in the plant. Since the relief valve set pressure will not be challenged during normal or upset operations, it is therefore expected to remain leak tight. The relief valve may be required in a post LOCA condition to release a minute amount of fluid only to immediately decrease the pipe internal pressure. Therefore, since the pressure will quickly subside as soon as the valve disk begins unseating and a minute amount of fluid is leaked/discharged, a minimum valve relieving capacity is actually needed. It is expected that complete opening of the relief valve disk will not occur. Therefore the size and relieving capacity of the relief valve are not critical design parameters and the ¾" valve installed is obviously adequate for this purpose. The relief valve will be added in sections of pipe such that no existing stop valve or other device could reduce the penetration overpressure protection. There are no ASME Code requirements dictating the installation of tail pipes. Also, a cursory review of 29CFR1910, "Occupational Safety and Health Standards," has been conducted. The review revealed no Occupational Safety and Health Administration (OSHA) requirements related to the use of relief valve tail pipes. As a precaution however, the relief valve discharge nozzle has been oriented in a way not to directly affect any adjacent equipment. The newly installed relief valve for Reactor Water System Penetrations 87 and 325 is not located in a normally accessible area and does not represent a personnel hazard should unlikely relief valve discharge occur. Since the relief valve will be required to release just a minute amount of fluid during an accident condition only, the existing bounding accident environmental parameters will not be impacted. Therefore, the original environmental qualification of the equipment inside Containment or the Auxiliary Building Steam Tunnel is not impacted. Relief valves instead of rupture discs were selected for installation to ensure the availability of the affected systems after a small break LOCA event as Grand Gulf emergency procedures restore some of these systems to help mitigate accident consequences. The additions of the small bore branch (including relief valve and flanges) has been evaluated along with the existing piping in NPE Stress Calculations PDS-2193, Supplement 1, Rev. 0; PDS-2741, Rev. 0; and PDS-106, Rev. 7, for all plant conditions (including the elevated relief pressure) to meet the design requirements of ASME Section III, Subsections NB-3600, Code Cases *1555* and 1574; ANSI B31.1; Bechtel Document Number M-18; and Drawing 9645-M-1398. Also, new revised Calculation NPE-E12F394/G33F001/F004/F250/F251/F252/F253, Rev. 12, was necessary to qualify the affected isolation valves. There are no pipe break jet impingement cones postulated in the area of the newly added relief valve and the penetrations boundaries. Therefore, the 3/4" line cannot fail due to jet impingement caused by an adjacent main line break. Also, failure due to suppression pool swell is not expected since the line is installed in the Auxiliary Building Steam Tunnel. The valve is designed for relief pressure well above maximum operating pressures for the system to prevent inadvertent discharges. The new relief valve will be tested in accordance with Procedure 07-S-14-395, "General Maintenance Instruction-Safety and Relief Valve Program-Safety Related". Divisional failure possibilities were reviewed for the penetration valve. Various failure scenarios were considered and no new unevaluated effect due to this modification was

identified. All the above modifications and changes will assure piping systems and containment integrity under over pressurization conditions post-LOCA. As part of the modification, the installed bypass line will be removed. The plugs put in place of the bypass line will meet the requirements of a Class 1 boundary.

BRIEF DESCRIPTION OF CHANGE, TEST OR EXPERIMENT:

ER-GG-2000-0113-000, Rev. 0, makes two changes. One change establishes a 3-hour fire resistance rating for a non-standard fire barrier penetration configuration based on evaluation rather than actual fire endurance testing. This non-standard fire barrier penetration configuration is utilized in the double wall configuration separating the Auxiliary Building and the Control Building. The second change accepts-as-is 15 fire rated penetration seals as acceptable for the hazards in the area, although they may not provide a 3-hour fire resistance rating. The 15 penetrations are as follows: CE156GA, CE-352G, CE-365G, CE-200DA, CE-201 DA, CE-202DA, CE-205DA, CE208DA, CE-230DA, CE-259DA, CE-270DA, CE-277DA, CE-129BA, CE-192BA and CE-193BA.

REASON FOR CHANGE, TEST OR EXPERIMENT:

The penetrations described above are located in 3-hour rated fire barriers and are described in the UFSAR and an NRC SER as providing a 3-hour fire resistance rating based on fire testing in accordance with ASTM E-119. Contrary to this requirement, these penetrations are not bounded by actual fire test in accordance With ASTM E-119. Therefore, this ER provides the engineering evaluations necessary to establish the adequacy of these fire rated penetrations.

SAFETY EVALUATION SUMMARY AND CONCLUSIONS:

License Condition 2.C.41 allows GGNS to make changes to the approved Fire Protection Program through the 50.59 process if those changes do not adversely affect the ability to achieve and maintain safe shutdown in the event of a fire. Therefore, from the fire protection standpoint, the basis for evaluation is "no adverse effect on the ability to achieve and maintain safe shutdown in the event of a fire". Generic Letter 86-10, Enclosure No.1, Interpretation No. 4, states "Where fire area boundaries are not wall-to-wall, floor-to-ceiling boundaries with all penetrations sealed to the fire rating required of the boundaries, licensees must perform an evaluation to assess the adequacy of fire boundaries in their plants to determine if the boundaries will withstand the hazards associated with the area". As documented in Fire Protection Evaluation 2000-00075, 15 non-standard penetration configurations have been evaluated and found capable of withstanding the hazards associated with areas on either side of the affected penetrations. Additionally, Fire Protection Evaluation (FPE) 2000-0074 establishes a 3-hour fire resistance rating for a non-standard fire barrier penetration configuration between the Control and Auxiliary Building, based on evaluation rather than actual fire endurance testing. FPEs 2000-0074 & 75 also determine that safe shutdown capability will not be affected. Thus, the ability to achieve and maintain safe shutdown conditions in the event of a fire, as presently analyzed in the UFSAR, has not been adversely affected.

BRIEF DESCRIPTION OF CHANGE, TEST OR EXPERIMENT:

Operator actions are necessary to ensure isolation of a potential leak path to the outside through SSW piping under certain conditions of operation post-LOP/LOCA. The concern is that post-LOP/LOCA a delayed failure of the divisional power would prevent the capability to isolate Penetration 90 or 91 depending on which division is lost. Credit would be taken for a secondary containment liquid loop seal created by the SSW piping and water inventory between the Drywell Purge Compressor and the UHS Cooling Tower. Operator actions would be credited for ensuring long term isolation of the containment penetrations (90 and 91) via Step 6.7.27 from the Conduct of Operations Instructions being incorporated into the emergency procedures. Another scenario involving Auxiliary Building flooding due to the passive failure of the radiation monitor pump seal will require a change to the ARIs to address the SSW radiation monitor as a potential source of leakage in the Auxiliary Building.

REASON FOR CHANGE, TEST OR EXPERIMENT:

Containment Penetrations 90 and 91 (SSW return from the drywell purge compressors) may not comply with all aspects of regulatory guidance. Specifically, since the inboard and outboard isolation valves for these penetrations are powered from the same divisional power source (e.g., both isolation valves for Penetration 90 are powered from Division I) and since the SSW system's design basis post accident short term response includes postulation of a single active failure of a divisional power source, the ability to remote manually isolate Penetration 90 or 91 following a design basis accident may not exist.

SAFETY EVALUATION SUMMARY AND CONCLUSIONS:

There are two scenarios considered for SSW. One is a containment isolation valve failure which involves the failure of a division of SSW with the SSW containment penetration unisolated. The other SSW failure is a flooding concern which involves both divisions of SSW running and failure of one SSW division's radiation monitor seal.

For the containment isolation concern, the failure of a division of SSW with the SSW containment penetration isolation valve not closed, credit would be taken for a secondary containment liquid loop seal created by the SSW piping and water inventory between the drywell purge compressor and the UHS cooling tower. Operator actions would be credited for ensuring long term isolation of Containment Penetration 90 or 91 via Step 6.7.27 from the Conduct of Operations Instructions being incorporated into the emergency procedures to maintain the loop seal.

For the Auxiliary Building flooding concern, with both divisions of SSW running and one SSW division's radiation monitor pump seal failure (a passive failure of the seal after 24 hours into an accident), the DI7 sample pump seal is assumed to fail and leakage would begin at this point. The Auxiliary Building Sump ARIs will be changed to include the SSW radiation monitor pump as a source of leakage in the Auxiliary Building.

From ANS 58.9-1981, these requirements about single failure are presented.

3.4 During the short term (accident), the single failure may be limited to an active failure.

3.5 During the long term (accident), assuming no prior failure during the short term (accident), the limiting single failure considered can be either active or passive.

3 6 The design flow for a passive failure shall be defined by analysis of realistic passive failure mechanisms in the system, considering conditions of operation and possible failure or leakage modes as appropriate.

3.7 The designer shall consider an operator error as a potential single active failure. (In the case of SSW radiation monitor, an operator error would be not closing one valve on the penetration.)

3.8 If suitable time and means for detection, diagnosis, and correction of single failures are provided, operator actions for mitigation of consequences of single failure shall be allowed.

4.4 Limited leakage passive failures need not be considered in the analysis required, if the unit is designed such that a failure does not result in loss of the required safety functions.

For the Auxiliary Building flooding issue, the 24 hour requirement is taken from SECY 77-439 concerning a passive failure in a system. It reads, "In the study of passive failures it is current practice to assume fluid leakage owing to gross failure of a pump or valve seal during the long term cooling mode following a LOCA (24 hours or greater after the event) but not pipe breaks".

In conjunction with the SSW loop seal, procedurally operators isolate the SSW piping at the SSW pump house as an enhancement. This procedure change is designed to meet the same safety- and non-safety-related pressure boundary, seismic, and tornado protection requirements established in the design and licensing basis for the SSW system. All essential plant systems and equipment will function as assumed in the *Accident Analysis.* Therefore, this change will have no effect on any consequences of the accidents evaluated previously in the UFSAR, will not change offsite dose to the public, will not affect any fission product barriers, and does not alter any assumptions previously made in evaluating the radiological consequences of an accident described in the UFSAR. As a result, this change will not increase the consequences of an accident or create an accident of a different type than any evaluated previously in the UFSAR.

This SSW radiation monitor fluid leakage is approximately 15 gpm. It is bounded by the overall allowable SSW leakage for 30 days of SSW leakage. The leakage is not assumed to start for 24 hours based on it being a passive failure of safety-related equipment. The pump seal leakage is based on maintaining the SSW piping integrity for 24 hours following a severe accident at which time operator actions are credited for manually securing the failed SSW

radiation monitor pump seal. This 24 hour basis is consistent with NRC-accepted standard industry practice for PRA evaluations of severe accidents and is considered conservative for parameters controlled by the emergency procedures.
The selection of isolating the SSW line using the valves described in Conduct of Operations, 01-S-06-2, 6.7.27, was based on isolating the SSW containment penetration in the SSW Pump House instead of at the containment penetration. This was to keep operator exposure to post-accident doses to a minimum. From Bechtel Mechanical Calculation 5.6.8-N, Revision 0, the post-accident dose rates at Penetrations 90 and 91 are 317 rad/hr in the RPV vibration instrument test room (1A319). The SSW valves isolated at the SSW Pump House are butterfly valves and are not redundant air operated valves like the typical secondary containment isolation valves. However, in order to establish secondary containment, the SSW Pump House supply and return valves are closed. For post-accident/emergency conditions where the containment penetration is open, it is desired to close the SSW pump supply and return valves for a failed division of SSW. This is done to establish a barrier until the failed SSW division diesel can be restored or other actions can be determined for isolating the SSW containment penetration.

For SSW containment penetration there is only one scenario for consideration. Assuming a LOP/LOCA with the failure of a division of SSW is a single failure. The failure of the SSW pump and diesel is considered the single failure. The penetration would be isolated by the loop seal that would be in the SSW piping between the Auxiliary Building and SSW Pump House. As an additional precautionary measure, the emergency procedures would be revised to require isolating the SSW piping at the pump house.

Another failure identified outside of the penetration isolation which is an Auxiliary Building flooding concern is the failure of the radiation monitor pump seal. Failure of a SSW radiation monitor pump seal at operating SSW pressure is a legitimate single failure in that there are both divisions of SSW running, and after 24 hours a single passive failure occurs. That is one of the SSW radiation monitor pump seals. The leakage from the SSW radiation monitor pump seal is approximately 15 gpm (Reference Calculation MC-Q1P41-02005, Rev. 0, SSW Radiation Monitor Pump Seal Leakage) and is bounded by the overall allowable SSW leakage for 30 days of SSW inventory leakage (Reference Calculation MC-Q1P41-86007, Rev. 0, Standby Service Water Ultimate Heat Sink Performance). It would be prudent, however, to have operations isolate the SSW radiation monitors pumps after 24 hours after the LOCA in order to prevent flooding of the Auxiliary Building hallway. If the seals were to fail, the leak would be detected by requirement of ARIs which cover Auxiliary Building flooding. Auxiliary Building flooding is identified to the operator through Auxiliary Building Sump Level high annunciators. The annunciators have ARIs that provide sources of flooding/sump level high in the Auxiliary Building. For ER-GG-2002-0102, there are Auxiliary Building ARIs that will be updated to reflect the SSW radiation monitor sump pumps as a potential source of leakage. The SSW radiation monitor pump seals are located in the corridors on Elevation 93' of the Auxiliary Building. Operator dose exposure while securing Auxiliary Building corridor leakage is minimal because the exposure is away from containment leakage paths.

Nuclear Management Manual Procedure DC-306, "Commercial Grade Item Evaluation", reads "This procedure establishes methodology to provide reasonable assurance that commercial

grade items used in safety-related applications at Entergy Nuclear South (EN-S) will perform their intended safety functions." Commercial Grade Dedication is used for safety related applications at Grand Gulf and taking credit for the SSW radiation monitor pump seals as a pressure boundary is consistent with the Commercial Grade Dedication program.

ER-GG-1996-0166-001, Revision 0, classified the pump as safety-related including the safety function of maintaining pressure integrity. The new 8-1 pump seals are rated for 1250 psig hydrostatic and 450 psig at 3450 rpm. The pressure retaining components and supports of 1D17J005 and 1D17J006 are upgraded to safety related, seismic category I. The seals are pressure retaining components for the radiation monitor pumps.

The pump seals are evaluated by the document EPRI CGI Joint Utility Task Group Commercial Grade Item Evaluation for Circular Cross Section O-Rings Manufactured by Parker Seal Group, O-Ring Division (TE Number CGIOROI).

All essential and nonessential plant systems and equipment are designed to meet the current licensing and design requirements. This modification does not change the SSW system actuation, flow parameters, or the pressure boundary requirements and they will function as assumed in the *Accident Analysis.* Therefore, this change will not increase the consequences of a malfunction of equipment important to safety or create the possibility of a malfunction of equipment important to safety of a different type than any evaluated previously in the UFSAR. Also, this change will not reduce the margin of safety as defined in the basis for any technical specification.

Based on the results of this safety evaluation, the effects associated with this modification are inconsequential and, therefore, do not constitute an unreviewed safety question (USQ).

This change is procedural to the plant, does not affect plant power levels, and does not affect plant influents or effluents. Therefore, this design change does not represent a change to the Environmental Protection Plan or a change that will affect the environment. There is no potential for an unreviewed environmental question; therefore, there is no need to perform an environmental evaluation.

BRIEF DESCRIPTION OF CHANGE, TEST OR EXPERIMENT:

This review evaluates the changes necessary to implement the Appendix K power uprate requested by GNRO 2002-0008. These changes consist of the following three items:

(I) Overpower Reportability
Consistent with GE and industry positions and the attached technical discussion, GGNS will Maintain the existing 2% overpower threshold for reportability.

(II) Steam Tables
When available, the ASME 1997 steam tables can be applied in the plant calorimetric heat balance in lieu of the current ASME 1967 tables. These revised steam tables are not necessary for power uprate implementation and may be implemented at a later date.

(III) Uprate Testing
The testing plan provides the verification that GGNS feed water and turbine control systems will respond as expected and core operating limits are maintained within technical specifications when reactor power is increased to 3898 MW. Additionally, chemistry and radiological conditions will be monitored for compliance with technical specifications and other administrative limits.

REASON FOR CHANGE, TEST OR EXPERIMENT:

These changes are necessary to implement the Appendix K power up rate.

SAFETY EVALUATION SUMMARY AND CONCLUSIONS:

No new accident scenarios, failure mechanisms, or limiting single failures are introduced as a result of the proposed changes. All systems, structures, and components previously required for the mitigation of a transient remain capable of fulfilling their intended design functions. The proposed changes have no adverse effects on any safety-related system or component and do not challenge the performance or integrity of any safety related system.

The testing portion of this ER will utilize the methodology of Startup Tests 22 and 23B, but to a lesser magnitude. These tests will verify the automatic control function of the Turbine Control and Feed Water Control System at 3641 CMWT, 3833 CMWT, and 3898 CMWT. Reactor Engineering will monitor core operating parameters to ensure that power distribution limits are not exceeded. Chemical and radiological conditions will be monitored during the performance of these tests to assure that administrative and regulatory limits are not exceeded. Peer checking and independent verification will be used prior to critical evolutions, and management oversight will be maintained throughout the performance of these tests.

BRIEF DESCRIPTION OF CHANGE, TEST OR EXPERIMENT:

This safety evaluation assesses the reload-related changes associated with Cycle 13 operation as presented in the Core Operating Limits Report (COLR) located in the Operating License Manual (OLM). Cycle 13 has been designed for 492 effective full power days with a core consisting of 240 fresh ATRIUM-10 assemblies, 204 once-burnt ATRIUM-10 assemblies, 228 twice-burnt GE11 assemblies, and 128 thrice-burnt GE11 assemblies. There are no TS, TS Bases, or TRM changes required to operate with this new core; however, the UFSAR does require updates. The Cycle 13 core has been designed and analyzed for a 1.7% power up rate at a rated thermal power of 3898 MWt in support of the Cycle 13 implementation of the Appendix K up rate. As such, the reload analyses are applicable to both the current power level of 3833 MWt as well as the up rated power level of 3898 MWt. The Appendix K up rate is being reviewed by the NRC and is not addressed in this evaluation. Individual design changes on GGNS systems are assessed in the safety evaluation associated with the specific change package and are not addressed in this evaluation. Attachment 1 provides a detailed description of the Cycle 13 reload analysis and the issues considered in this evaluation.

REASON FOR CHANGE, TEST OR EXPERIMENT:

Cycle 13 operation will require new core operating limits and the Core Operating Limits Report has been revised to include these new limits. These limits include flow-, power-, and exposure-dependent LHGR, MAPLHGR, and MCPR limits.

SAFETY EVALUATION SUMMARY AND CONCLUSIONS:

The Cycle 13 core configuration and operation has been evaluated with respect to mechanical, neutronic, thermal-hydraulic, dose, thermal performance, and methods considerations for GGNS. This evaluation concludes that the reload-related changes associated with Cycle 13 operation will not constitute an unreviewed safety question.

BRIEF DESCRIPTION OF CHANGE, TEST OR EXPERIMENT:

This safety evaluation assesses the reload-related changes associated with Cycle 13 operation as presented in the Core Operating Limits Report (COLRI located in the Operating License Manual (OLM). Cycle 13 has been designed for 492 effective full power days with a core consisting of 240 fresh ATRIUM-10 assemblies, 204 once-burnt ATRIUM-10 assemblies, 228 twice-burnt GE11 assemblies, and 128 thrice-burnt GE11 assemblies. There are no TS, TS Bases, or TRM changes required to operate with this new core; however, the UFSAR does require updates. The Cycle 13 core has been designed and analyzed for a 1.7% power up rate at a rated thermal power of 3898 MWt in support of the Cycle 13 implementation of the Appendix K up rate. As such, the reload analyses are applicable to both the current power level of 3833 MWt as well as the up rated power level of 3898 MWt. The Appendix K up rate is being reviewed by the NRC and is not addressed in this evaluation. Individual design changes on GGNS systems are assessed in the safety evaluation associated with the specific change package and are not addressed in this evaluation. Attachment 1 provides a detailed description of the Cycle 13 reload analysis and the issues considered in this evaluation. This revision provides additional evaluations associated with the explicit incorporation of the increased GE11 channel bow described in CR-GGN-2002-01810 into the Cycle 13 core design and operation.

REASON FOR CHANGE, TEST OR EXPERIMENT:

Cycle 13 operation will require new core operating limits and the Core Operating Limits Report has been revised to include these new limits. These limits include flow-, power-, and exposure-dependent LHGR, MAPLHGR, and MCPR limits.

SAFETY EVALUATION SUMMARY AND CONCLUSIONS:

The Cycle 13 core configuration and operation has been evaluated with respect to mechanical, neutronic, thermal-hydraulic, dose, thermal performance, and methods considerations for GGNS. This evaluation concludes that the reload-related changes associated with Cycle 13 operation will not constitute an unreviewed safety question.

BRIEF DESCRIPTION OF CHANGE, TEST OR EXPERIMENT:

Currently, TRM 7.6.3.3.g.1 requires MSIV fast closure testing to be performed during startup at 600 psig reactor pressure. This testing is done for Technical Specification 3.6.1.3 under Surveillance Requirement 3.6.1.3.6, which is MSIV fast closure testing. This evaluation deletes the TRM requirement. Procedure 06-OP-1B21-V-0001 will need to be revised to allow MSIV fast closure testing during controlled reactor shutdown with low steam flow or at any reactor pressure and temperature. There is a minor procedure change for IOI 03-1-01-3. The tables will be moved from IOI 03-1-01-3 to 06-OP-1B21-V-0001 because it currently fast closes the MSIV and records the required information. This will eliminate the need for MSIV fast closure testing during startup after an outage. The IST procedure for MSIV fast closure will still be updated to reflect MSIV fast closure testing at any reactor pressure and temperature.

REASON FOR CHANGE, TEST OR EXPERIMENT:

During controlled reactor shutdown MSIV fast closure testing is performed by IOI 03-1-01-03. IST MSIV fast closure testing is currently performed and credited during startup. This may not be the optimum time to test the MSIVs. During controlled shutdown it would be desirable to fast closure test the MSIVs during shutdown around 60 psig reactor pressure. Fast closure testing during shut down reduces the possibility of preconditioning the MSIVs prior to fast closure testing. Currently, the valves may have maintenance performed on them during an outage, thus giving the impression of preconditioning of the MSIVs. Fast closure testing of the MSIVs during controlled shutdown of the reactor would eliminate this situation and would be an IST enhancement because no perceived MSIV preconditioning occurs.

SAFETY EVALUATION SUMMARY AND CONCLUSIONS:

For the MSIV fast closure testing, the requirements are going to be deleted in TRM Section 7.6. 3.3.g.1. This testing currently performed by Integrated Operating Instruction 03-1-01-3 but not utilized for IST testing will be moved to 06-OP-1B21-V-0001. This will allow MSIV fast closure testing at any reactor pressure and temperature. The procedure for this testing is 06-OP-1B21-V-0001. Procedure 06-OP-1B21-V-0001 needs to be revised to include MSIV fast closure testing during controlled shutdown conditions with reactor at low steam flow or at any reactor pressure and temperature.

Stroking of the MSIVs will be performed per the requirements of GIN 2001/00986. If the valve is "wet" (i. e, the steam line is filled with steam) the valve can be repeatedly stroked with no minimum wait time between valve cycles and the valve will not experience valve galling or damage due to stroking. Although experience has shown that "cold" stroke of the MSIVs is consistent with the "hot" stroke of the MSIVs, 06-OP-1B21-V-000I will be revised with enhanced stroke time acceptance criteria.

If fast closure testing of the MSIVs is performed at 60 psig reactor pressure, the saturation temperature is greater than 300 degrees F. The MSIVs are still "hot" even with the test pressure reduced from 600 psig to 60 psig. The MSIV fast closure time will be unaffected by reducing the test pressure to 60 psig. The MSIVs fast closure stroke time can be performed at any pressure

and temperature. Limitations about stroking the MSIVs "cold" and "dry" has been communicated to operations via GIN 2001/00986.
The requirement to MSIV fast closure test at 600 psig was not found in any of the MSIV technical specifications, vendor manual or design documents for the MSIVs. It appears that there was no basis for this reactor pressure other than the desire to test MSIVs "wet" and "hot". Therefore, 600 psig was chosen as the test pressure apparently arbitrarily based on steam conditions.

Changing the TRM to allow MSIV fast closure testing is an IST enhancement. It eliminates preconditioning of the MSIVs. This TRM change does not affect MSIV design functions and continues to provide safety pressure boundary, seismic, and tornado protection requirements established in the design and licensing basis for the MSIVs. All essential plant systems and equipment will function as assumed in the *Accident Analysis.* Therefore, this change will have no effect on any consequences of the accidents evaluated previously in the UFSAR, will not change offsite dose to the public, will not affect any fission product barriers, and does not alter any assumptions previously made in evaluating the radiological consequences of an accident described in the UFSAR. As a result, this change will not increase the consequences of an accident or create an accident of a different type than any evaluated previously in the UFSAR.

The MSIVs are designed to meet the current licensing and design requirements. This evaluation does not change the MSIV system actuation, flow parameters, or the pressure boundary requirements and they will function as assumed in the *Accident Analysis.* Therefore, this change will not increase the consequences of a malfunction of equipment important to safety or create the possibility of a malfunction of equipment important to safety of a different type than any evaluated previously in the UFSAR. Also, this change will not reduce the margin of safety as defined in the basis for any technical specification.

Based on the results of this safety evaluation, the effects associated with this evaluation are inconsequential and, therefore, do not constitute an unreviewed Safety Question (USQ).

This change is an enhancement to MSIV fast closure testing allowing testing to occur either during a controlled reactor shutdown at low steam flow or any reactor pressure. It does not affect plant influents or effluents. Therefore, this evaluation does not represent a change to the Environmental Protection Plan or a change that will affect the environment. There is no potential for an unreviewed environmental question, therefore, there is no need to perform an environmental evaluation.

Commitment Number: 16164

Source Document Number: AECM-1990/00156

COMMITMENT CHANGE TITLE:

Key vendor contact process

COMMITMENT DESCRIPTION:

Original Commitment Description:

Grand Gulf will implement a procedure by 1/1/91 which will require documented contact with key non-NSSS vendors on an annual basis. This procedure will also control the list of non-NSSS vendors to be contacted annually.

Revised Commitment Description:

Grand Gulf/Entergy Operations, Inc. will implement a revised process by 12/31/01 which will require documented contact with key non-NSSS vendors once every other calendar year. The next vendor contact will be completed in the calendar year 2002. This process will also control the list of non-NSSS vendors to be contacted.

JUSTIFICATION FOR CHANGE OR DELETION:

Generic Letter 90-03 requires licensees to maintain a vendor interface program which is a good faith documented effort to periodically contact the vendors of key non-NSSS safety-related components (such as auxiliary feed water pumps, batteries, inverters, battery chargers, cooling water pumps, and valve operators) to obtain any technical information applicable to this equipment. As documented by letters CEO 98/00079, CEO 99/00086, and CEO 2000-00089, Entergy Operations has contacted approximately 44 vendors per year for the last 3 years to request updated technical information related to approximately 510 technical publications. In response to these requests, approximately 43 documents were submitted to Entergy as updated information. Only a small percentage of the documents received were found to be applicable to plant equipment. None of the information received resulted in any corrective actions or plant modifications. Therefore, changing the frequency of Entergy's periodic contact with key non-NSSS vendors to every other calendar meets the intent of the Generic Letter and should have no adverse effect on plant equipment.

Commitment Number: 24251 and 24254

Source Document Number: AECM-1986/00319 and AECM-1987/00169

COMMITMENT CHANGE TITLE:

Control Room air conditioner and switchgear room cooler flow tests

COMMITMENT DESCRIPTION:

Original Commitment Description:

(i) Commitment Description based on AECM-86/0319, Page 3, Paragraph C.3:

Flow to the control room air conditioning (A/C) unit will be monitored periodically and evaluated for degradation. This monitoring will include weekly observation of the unit's compressor discharge pressure. This observation will be utilized to detect potential problems in overall system performance that may be due to degraded SSW System flow. In addition, SSW flow rate to the condenser will be measured on a monthly basis. These flow monitoring activities will be performed only if the system is required to be operable and as long as the subject flow rate is less than the established design value. *(Please note that this commitment has been completely superceded by subsequent commitments made in conjunction with AECM-87/0169, as discussed below.)*

(ii) Commitment Description based on AECM-87/0169, Page 10 of Attachment, Paragraph f:

As a long term corrective measure, a flow monitoring program has been established to provide flow performance and trending information. The primary objective of the monitoring program is to identify the need for flushing or cleaning those SSW components which are serviced by PSW during normal operating conditions. On a monthly basis, flow data will be measured and recorded for the ESF switchgear room coolers and the "A" control room A/C unit. The "B" control room A/C unit is excluded from the periodic monitoring due to its relatively high measured flow rate (in excess of 180 gpm). If, however, significant deterioration in flow to the "A" control room A/C unit is observed in the monitoring program, the "B" side unit flow will be confirmed to be acceptable.

Flow thresholds have been established to assure flow rates are maintained above the minimum design flow values. If measured flow is confirmed to be below this threshold, an evaluation will be performed to determine actions necessary to restore the flow or to increase the monitoring to assure flow is maintained above minimum design flow. This evaluation will include consideration of any trend noted and the potential for sudden further changes in cooler flow. Should measured flow fall below the minimum design flow, the affected component will be considered inoperable. At the second refueling outage, one SSW division will be selected and each components flow will be measured. Furthermore, a review of the program's accumulated data will be performed. Based on this data and subsequent review, the need for additional flushing or cleaning will be determined. Threshold and monitoring periodicity will also be revised, if necessary, based on that evaluation.

This program of cooler performance monitoring represents SERI's long term program and supercedes previous interim commitments to monitor the ESF switchgear room coolers and the control room A/C units (AECM86/0283, AIECM-86/0309, and AECM- 86/0319).
The flow monitoring program has been implemented via plant procedures.

Revised Commitment Description:

(i) Commitment Description based on AECM-86/0319, Page 3, Paragraph C.3:

Flow to the control room air conditioning unit will be monitored periodically and evaluated for degradation. This monitoring will include weekly observation of the unit's compressor discharge pressure. This observation will be utilized to detect potential problems in overall system performance that may be due to degraded SSW System flow. In addition, SSW flow rate to the condenser will be **periodically** measured **on at least a semi-annual basis.** These flow monitoring activities will be performed only if the system is required to be operable.

(ii) Commitment Description based on AECM-87/0169, Page 10 of Attachment, paragraph f:

A flow monitoring program has been established to provide flow performance and trending information. The primary objective of the monitoring program is to identify the need for flushing or cleaning those SSW components which are serviced by PSW during normal operating conditions. On a **quarterly** basis, flow data will be measured and recorded for the ESF switchgear room coolers. On a **semi-annual** basis, flow data will be measured and recorded for the "A" and "B" main control room A/C units. **If adverse trends or significant deterioration in flow to any of these coolers is observed in the monitoring program, corrective actions will be implemented to promptly restore SSW flows to acceptable levels.**

Flow thresholds have been established to assure flow rates are maintained above the minimum design flow values. **If measured flow is confirmed to be below this threshold, corrective actions will be performed to restore the flow or to increase the monitoring to assure flow is maintained above minimum-design flow.** Should measured flow fall below the minimum design flow, the affected component will be considered inoperable. **At least every 3 years, each SSW division will be flow balanced and each component's flow will be measured and verified acceptable.** Furthermore, a review of the program's accumulated data will be performed. Based on this data and subsequent review, the need for additional flushing or cleaning will be determined. Threshold and monitoring periodicity will also be revised, if necessary, based on that evaluation.

This program of cooler performance monitoring represents **ENTERGY's** long term program and supercedes previous interim commitments to monitor the ESF switchgear room coolers and the control room A/C units (AECM-86/0283, AECM-86/O309, and AECM-86/0319).

The flow monitoring program has been implemented via plant procedures. **The applicable SSW flow rate acceptance criteria for each component has been included in these procedures.**

JUSTIFICATION FOR CHANGE OR DELETION:

(i) Commitment based on AECM-86/0319, Page 3, Paragraph C.3:

The proposed revision would read "**will be periodically** measured **on at least a semi-annual** basis" in place of "will be measured on a monthly basis". This revision is proposed to ensure consistency is maintained between multiple source documents. As stated above, the commitment based on AECM-86/0319, Page 3, Paragraph C 3, has been completely superceded by subsequent commitments made in conjunction with AECM-87/0169 which is discussed below.

(ii) Commitment based on AECM-87/0169, Page 10 of Attachment, Paragraph f:

In the first paragraph, the proposed revision would read "a **quarterly** basis for the ESF switchgear room coolers" and "a **semi-annual** basis for the "A" and "B" main control room A/C units". Additionally, the last sentence was revised to read, "**If adverse trends or significant deterioration in flow to any of these coolers is observed in the monitoring program, corrective actions will be implemented to promptly restore SSW flows to acceptable levels".** Justification for the proposed flow test frequency change is adequately provided by the data collected from these testing activities since the 1999 time frame. In 1999, significant improvements were implemented with respect to ESF switchgear room cooler flow testing activities and, since that time, very few flow related problems have been experienced. Since 1999, trends of the associated SSW flow rate data have indicated relatively stable, acceptable flow rates to each of the ESF switchgear room coolers. Additional justification for the proposed change is provided by the fact that other programs are in place to periodically monitor the SSW flow through these coolers. Specifically, periodic SSW system flow verifications and SSW system flow balances are conducted to ensure these coolers, as well as other SSW components, receive adequate flow. SSW flow rate acceptance criteria are specified in these flow verification and balancing procedures (17-S-06-22, 23, and 24) as well as requirements for implementing corrective actions if discrepancies are noted. Also, each of the ESF switchgear room coolers is monitored and trended as part of the GGNS Thermal Performance Program (reference AECM-90/0007) where SSW flow rates to the individual coolers are recorded and verified adequate. Thermal performance tests for the Main Control Room A/C units are periodically conducted based on technical specification requirements. Based on the overall monitoring activities completed through these various items, numerous and adequate actions are being performed to provide reasonable assurance that the affected coolers are maintained in an operable condition. Thus, the proposed frequency change for EPI Flow Rate testing will not diminish the ability of these components to perform their design related functions.

In the second paragraph, the proposed revision would read **"If measured flow is confirmed to be below this threshold, corrective actions will be performed to restore the flow or to increase the monitoring to assure flow is maintained above minimum design flow."** and **"At least every 3 years, each SSW division will be flow balanced and each component's flow will be measured and verified acceptable."** Justification for the proposed changes in the second paragraph is that these changes are primarily editorial in nature. The proposed changes are considered necessary to indicate that corrective actions will be completed to restore flow

rates to desired levels, with or without a specific evaluation being completed. Additionally, the frequency for SSW System flow balance testing is every 18 months and these tests can now be performed outside-of refueling outages. Thus, the proposed change to indicate a test frequency of "at least every 3 years" would essentially meet the original commitment while allowing the current flow balance test frequency to be maintained or extended, based on plant needs and conditions.

In the third paragraph, the proposed revision would read "... **ENTERGY's** ..." in place of "... **SERI's** ..." due to changes in the Company's name. Justification for the proposed change in the third paragraph is not considered necessary due to the editorial nature of this change.

Commitment Number: 22996 and 34029

Source Document Number: AECM-1980/00026 and LBDC 2002-100

COMMITMENT CHANGE TITLE:

Valve lineup checks

COMMITMENT DESCRIPTION:

Original Commitment Description:

The position of each manually operated valve is identified in a valve lineup sheet. Valve lineup checks are conducted as required by technical specifications to verify system flow paths. (In addition, valve lineup checks on ESF systems are conducted after *each* refueling outage and following any major work on the system. Valve lineup checks will be conducted on the other accessible safety related systems during the cycle.) For safety-related systems/components, this valve lineup has independent verifications. Where appropriate, valves are locked in their designated position to prevent inadvertent repositioning.

Revised Commitment Description:

The position of each manually operated valve is identified in a valve lineup sheet. Valve lineup checks are conducted as required by technical specifications to verify system flow paths. For safety-related systems/components, this valve lineup has independent verifications. Where appropriate, valves are locked in their designated position to prevent inadvertent repositioning.

JUSTIFICATION FOR CHANGE OR DELETION:

GGNS has programs in place which require that valves in safety-related systems which are repositioned for maintenance, modification, or surveillance testing purposes be returned to their correct positions. In addition, system lineup changes other than those covered by normal operating procedures are logged and abnormal lineups are covered during the shift turnover. Limiting the checks to be performed during operation has been proposed primarily based on ALARA and personnel safety considerations. Based on the programmatic controls mentioned above, the incremental benefit of this additional check is not deemed an effective use of resources.