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July 6, 2001

U.S. Nuclear Regulatory Commission
Washington, DC 20555-0001

Attention: Document Control Desk

Subject: Grand Gulf Nuclear Station
Docket No. 50-416
License No. NPF-29
Report of 10CFR50.59 Safety Evaluations and Commitment Changes – September 16, 2000 through April 30, 2001

GNRO-2001/00051

Ladies and Gentlemen:

Pursuant to 10CFR50.59(d)(2), Entergy Operations, Inc. hereby submits the Summary of the 10CFR50.59 Evaluations for the period September 16, 2000 through April 30, 2001. Also attached is the SUMMARY of the commitment change made in accordance with guidelines of NEI 95-07 for the same period.

We are now submitting Summary of 10CFR50.59 Evaluations 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. This letter does not contain any commitments. If further information is required, please contact this office.

Yours truly,

JCR/ACG
enclosure: Commitment Change Evaluation Summary
cc (See Next Page)

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10CFR50.59 SUMMARY FOR THE PERIOD
STARTING SEPTEMBER 16, 2000 AND ENDING APRIL 30, 2001

LEGEND OF ACRONYMS			
LDC	Licensing Document Change	FHAR	Fire Hazard Analysis
ER	Engineering Request	PAP	Plant Administrative Procedure
DCP	Design Change Package	CR	Condition Report
MAI	Maintenance Action Item	TSTI	Technical Special Test Instructions
CCE	Commitment Change Evaluation	CN	Change Notice
NMM	Nuclear Management Manual	SOI	System Operating Instructions
TA	Temporary Alteration	DCS	Design Change Standard

Evaluation Number: 2000-0014-R01 Document Evaluated: LDC-2000-059

BRIEF DESCRIPTION OF CHANGE, TEST OR EXPERIMENT:

Cooling to the Hot Lab is provided by air handler V41B002. This airhandling unit is non-seismic and non-safety related. This ER will add a filter in the supply ductwork to remove any particulate matter that may be in the airstream. In order to support filter installation, SV41-TIC-R00I, SV41-TIS N009 and a supporting instrument airline, conduit and temperature switch sensing element require relocation.

REASON FOR CHANGE, TEST OR EXPERIMENT:

As described in GGCR1998-0918-00, airhandler V41B002 emits a “black dust” that enters the hot lab and adversely affects equipment. Despite repeated attempts to locate the source of the dust and remove it, the dust still persists. This ER will add a filter in the supply ductwork to the hot lab to remove the dust from the airstream.

SAFETY EVALUATION SUMMARY AND CONCLUSIONS:

The modifications made by this change meet the necessary requirements of all applicable codes. The modification is taking place in the radwaste building in an area that does not contain any safety related equipment. The changes will have no affect on any equipment which is considered important to safety and does not cause any equipment to be operated outside design limits. No new failure modes are created and there is no increase in previously identified failure modes for equipment which is important to safety. The changes will not degrade any important to safety system, component, or structure nor will they degrade or prevent actions described in the SAR accident analysis. No credit is taken for the radwaste building HVAC system to mitigate the consequences of an accident. It is concluded that this modification will not adversely affect Grand Gulf Nuclear Station. UFSAR figure 9.4-005B is being revised to reflect the addition of the filter in the radwaste Building HVAC system.

Evaluation Number: 2000-0019-R01 Document Evaluated: FSAR Section 9.5.5.2

BRIEF DESCRIPTION OF CHANGE, TEST OR EXPERIMENT:

UFSAR Subsection 9.5.5 states that the standby diesel engines and the HPCS engines "... can operate for a minimum of 30 days without additional water being required" by their jacket water systems. A review of the design basis documentation showed that there is no basis for this 30 day period. This change removes the 30 day period. The jacket water inventory period is being removed and is consistent with the guidance provided in NUREGs 0800, and 0831. NUREG 0800 Section 9.5.5 and NUREG 0831, Section 9.6.1 do not specify any time requirements for diesel jacket cooling water. NUREG 0800 Section 9.5.5 states that a provision be made to identify leakage and have contingencies for dealing with the leakage. The Grand Gulf Diesel Generator Jacket cooling water design meets these requirements. Each diesel jacket cooling water system has "Diesel Generator Trouble" annunciation in the control room that alarms when jacket cooling water inventory is low, and local "Low jacket water" inventory annunciation at the local diesel generator panel. Jacket water is not a consumable and will not require replenishing given a leak tight system. The only source of loss is evaporative for Div I and II and these are extremely small. The HPCS diesel engine jacket water system is a closed and unvented system. The jacket water expansion tank is closed by a pressure relieving cap (i.e., very similar to the radiator caps used on automobiles). Therefore, there will be no evaporation of water from the HPCS diesel engine jacket water system; consequently, provided there are no jacket water leaks, the HPCS diesel engine jacket water inventory is sufficient to run 7 days continuously without being refilled, This is not a physical change to the plant but to UFSAR Section 9.5.5 to remove the 30 day period.

REASON FOR CHANGE, TEST OR EXPERIMENT:

The existing requirements for the jacket water have been reviewed. The review shows no basis for the 30 day period and is therefore being deleted for the Division I, II and III in UFSAR 9.5.5.

SAFETY EVALUATION SUMMARY AND CONCLUSIONS:

This evaluation removes the leakage period for 30 days in UFSAR Section 9.5.5. The guidance set forth in NUREG 0831, 9.6.1. NUREG 0800, 9.5.5 specifies no requirements concerning diesel generator leakage other than having a capability of detecting and controlling the leakage. The diesels at Grand Gulf have "Diesel Generator Trouble" annunciation in the control room that alarms sending an operator to the local diesel generator panel which ahs annunciation for indicating when diesel generator cooling water standpipe level or expansion tank level (HPCS) low. The jacket water inventory low annunciation has the operator replenish the jacket water. Combined with the guidance established in NUREG 0831, 9.6.1, and the diesel generator cooling water standpipe level annunciation provides the basis for removing the 30 day inventory period.

Evaluation Number: 2000-0036-R00

Document Evaluated: ER 2000-0246-00-00
CN 2000-0023

BRIEF DESCRIPTION OF CHANGE, TEST OR EXPERIMENT:

ER 2000-0246-00-00 and CN 2000-0023 are providing a leak repair for the Reactor Water Clean Up (RWCU) System Outboard isolation valve Q1G33F039.

REASON FOR CHANGE, TEST OR EXPERIMENT:

Valve Q1G33F039 has a pressure seal leak and is leaking steam from at least one of eight pressure seal retaining ring knock-out holes in the valve body. The valve will be drilled and tapped at the bottom of the pressure seal ring and Furmanite sealant compound will be injected to fill the voids allowing leakage control. This repair will be valid until the first forced outage of sufficient duration to allow the final repair of the valve or until RF11.

SAFETY EVALUATION SUMMARY AND CONCLUSIONS:

Valve Q1G33F039 is the RWCU Return to Feedwater Outboard Containment Isolation Valve and is located in the Auxiliary Steam Tunnel. The valve has a pressure seal leak and is leaking steam from at least one of eight pressure seal retaining ring knock-out holes in the valve body. The valve will be drilled and tapped at various locations around the valve body at the bottom of the pressure seal ring and Furmanite sealant compound will be injected to fill the voids. This repair will be valid until the first forced outage of sufficient duration to allow the final repair of the valve or until RFI1 when the final repair will be implemented. This repair will maintain the valve's design basis condition and maintains its required containment isolation function during a postulated accident. The sealant is not a pressure retaining component and only a limited amount of the sealant will be injected into the valve. Limiting the amount of sealant and pressure to be injected will ensure the pressure retaining components of the valve are not moved to non-code components such as the yoke. Only sufficient sealant will be injected to seal around the pressure seal ring. Injection of the sealant at the bottom of the pressure seal ring is the appropriate location. This allows the valve internal pressure to direct the sealant upwards into the voids and controls the leaks with a minimum of the sealant. The installation of the nuclear qualified, pressure boundary shutoff adapters and injection of the Furmanite sealant compound will not adversely affect the structural integrity of the valve. The stress evaluation of the valve body was performed by Furmanite in Procedure No. N-2000205, Rev. 1. This evaluation shows all stresses in the valve will remain within ASME Section III code allowables for a Class 2 valve. Also the actual injection pressure of the Furmanite compound inside the valve body will be held to 1420 psig, which is less than the design pressure of 1500 psig for the valve as specified in Specification M-242.0. The sealant will not be injected at any locations near the valve stem or in quantities that would be able to affect the stem voids or packing, therefore valve stroke time as required in the TRM will not be affected. However, after the injection a partial stroke will be performed to assure the free movement of the valve stem, which is in accordance with the expectations of the nuclear industry as specified in NRC Inspection Manual Part 9900. The valve will therefore be capable of performing its safety-related function as a Primary Containment Isolation Valve and will not increase the possible offsite radiation dose, and therefore not affect the health and safety of the public.

The final repair of the valve will consist of removing the shutoff adapters and inserting a safety related threaded gland plug into the hole. The thread engagement of the plug will be sufficient to withstand the internal design pressure of the valve. The gland plug will then be seal welded to ensure leak tightness.

This repair will not affect the pipe break accidents identified in UFSAR Appendix 3C, Section 3C.2.2, since the valve will maintain its original design basis. 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 adapters are similar to the nut bolt combinations discussed in UFSAR 3.5.1.2.2.i that only have a small amount of stored energy and thus are of no concern as potential missiles. The shutoff adapters will have only 1251bs of propulsive force applied for a few milliseconds.

This repair will install Furmanite shutoff adapters, which involve drilling and tapping into the valve body at the Bottom of the pressure seal ring and requires an 1/8" to 3/16" (max.) diameter hole be drilled through the body to facilitate the installation of the Furmanite shutoff adapters. Installation of the Furmanite shutoff adapter will be at 90^0 perpendicular to the valve or slightly on a downward angle (no more than *5* degrees) to enter the valve body below the pressure seal ring. The valve body is an ASME pressure boundary component. This small hole could be a containment leakage path. The postulated leakage path would be past the inboard disc of the valve then through the hole. LCO 3.6.1.3 requires the penetration be isolated within 4 hours due to an inoperable Containment Isolation Valve. However, the time required for the hole to be drilled through the valve body and open to the atmosphere is very short compared to the LCO time limit of 4 hours.

One risk involved in performing a leak repair is injecting too much sealant into a valve to seal a leak. ER 2000-0246-00-00 and CN 2000-0023 will administratively control the amount of sealant pressure being injected into the valve. Controlling the amount of sealant and pressure ensures the 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.

Note: The above repair is not a code repair

Evaluation Number: 2000-0037-R00 Document Evaluated: LDC 2000-061

BRIEF DESCRIPTION OF CHANGE, TEST OR EXPERIMENT:

This change evaluates compensatory actions taken under ER 2000-0770 until final resolution and disposition of CR 2000-1352 is achieved. CR GGN-2000-1352 identified a condition where the EOC-RPT did not actuate per the design requirements when the reactor scrammed on 9/15/2000. Subsequent investigation concluded that the turbine control valve (TCV) fast closure trip setpoint may not fully trip all channels of the RPS and EOC-RPT functions for all TCV fast closure events initiated by the turbine control system. The EHC load reject device results in porting secondary control fluid to a final value that does not ensure the current trip setpoint is reached on all trip units. ER 2000-0770-00 conservatively increases the TCV fast closure trip setpoint to ensure that all RPS and EOC-RPT trip channels reliably actuate in response to a TCV fast closure. TRM Tables TR3.3.1.1-1 and TR3.3.4.1-1 are being changed by this evaluation.

In addition, the turbine speed demand setpoint is lowered to force the turbine load and associated fluid pressure when responding to a TCV fast closure to a point where the fluid pressure is below the TCV fast closure trip setpoint. This is based on previous valve test data.

REASON FOR CHANGE, TEST OR EXPERIMENT:

A load reject condition without the associated turbine load reject trip failed to actuate. The N41-M791 Load Reject Relay (LRR) was found to be set up incorrectly and this condition has been corrected. When set correctly the TCV control fluid is dumped on initiation of the relay thereby ensuring the TCV fast closure trip setpoint is reached on all trip units. Raising the TCV Fast Closure Trip setpoint will increase the margin for initiating the RPS and EOC-RPT due to TCV Fast Closure. This additional margin will ensure that all required trip channels are actuated regardless of the actual sequence of initiation using any of the designed methods of EHC or MHC load reject trips.

SAFETY EVALUATION SUMMARY AND CONCLUSIONS:

This change increases the TCV Fast Closure, trip setpoint from ≥44.3 psig to ≥46 psig. With the reactor and turbine generator at power, fast closure of the TCVs can result in a significant addition of positive reactivity to the core as nuclear system pressure rapidly increases. The TCV fast closure trip system initiates an anticipatory RPS and EOC-RPT actuations early compared to either the neutron monitoring system or nuclear system high pressure trips for fast pressurization events. This anticipatory trip is credited in the safety analysis to provide a satisfactory margin to the fuel thermal operating limits for moderate frequency pressurization transients such as the generator load rejection event. The RPS and EOC-RPT functions anticipate the addition of positive reactivity that results from the rapid pressure increase, thus effectively mitigating the pressurization transient. The TCV fast closure trip setting is selected to provide timely indication of control valve fast closure. The trip setpoint increase from ≥44.3 to ≥46 will increase the reliability to detect TCV fast closure events for RPS and EOC-RPT initiation. There are no unreviewed safety questions or reductions in the margins of safety associated with increasing the TCV fast closure trip setpoint to $\geq$46 psig.

While not a safety related change, additional margin is obtained by lowering the main turbine speed demand setpoint. In response to a partial Generator Load Reject, the generator control system will change from load demand to speed demand. In speed demand, the EHC system will respond by lowering the EHC fluid pressure below the TCV Fast Closure Trip setpoints, thereby ensuring these functions occur as designed.

Evaluation Number: 2000-0038-R00

Document Evaluated: LDC-2000-061
ER 2000-0770-00 and ER 2000-0770-01

BRIEF DESCRIPTION OF CHANGE, TEST OR EXPERIMENT:

This change evaluates compensatory actions taken under ER 2000-0770-00 and ER 2000-0770-01 until final resolution and disposition of CR 2000-1352 is achieved. CR GGN-2000-1352 identified a condition where the EOC-RPT did not actuate per the design requirements when the reactor scrammed on 9/15/2000. Subsequent investigation concluded that the turbine control valve (TCV) fast closure trip setpoint may not fully trip all channels of the RPS and EOC-RPT functions for all TCV fast closure events initiated by the turbine control system. The EHC load reject device results in porting secondary control fluid to a final value that does not ensure the current trip setpoint is reached on all trip units. ER 2000-0770-000 conservatively increases the TCV fast closure trip setpoint to ensure that all RPS and EOC-RPT trip channels reliably actuate in response to a TCV fast closure. TRM Tables TR3.3.1.1-1 and TR3.3.4.1-1 are being changed by this evaluation.

ER 2000-0770-000 had required the turbine speed demand setpoint be lowered. The intent was to gain additional margin over that obtained by changing the trip set point to ≥ 46 psig. Due to operational consideration and its potential impact on bypass valves coming open, ER 2000-0770-001 deletes the requirement to lower the speed demand. The speed demand will remain unchanged. The margin improvement obtained by raising the trip setpoint is sufficient to ensure that the trip points are reached.

REASON FOR CHANGE, TEST OR EXPERIMENT:

A load reject condition without the associated turbine load reject trip failed to actuate. The N41-M791 Load Reject Relay (LRR) was found to be set up incorrectly and this condition has been corrected. When set correctly the TCV control fluid is dumped on initiation of the relay thereby ensuring the TCV fast closure trip setpoint is reached on all trip units. Raising the TCV Fast Closure Trip setpoint will increase the margin for initiating the RPS and EOC/RPT due to TCV Fast Closure. This additional margin will ensure that all required trip channels are actuated regardless of the actual sequence of initiation using any of the designed methods of EHC or MHC load reject trips.

SAFETY EVALUATION SUMMARY AND CONCLUSIONS:

This change increases the TCV Fast Closure, trip setpoint from ≥44.3 psig to ≥46 psig. With the reactor and turbine generator at power, fast closure of the TCVs can result in a significant addition of positive reactivity to the core as nuclear system pressure rapidly increases. The TCV fast closure trip system initiates an anticipatory RPS and EOC-RPT actuations early compared to either the neutron monitoring system or nuclear system high pressure trips for fast pressurization events. This anticipatory trip is credited in the safety analysis to provide a satisfactory margin to the fuel thermal operating limits for moderate frequency pressurization transients such as the generator load rejection event. The RPS and EOC-RPT functions anticipate the addition of positive reactivity that results from the rapid pressure increase, thus effectively mitigating the pressurization transient. The TCV fast closure trip setting is selected to provide timely indication of control valve fast closure. The trip setpoint increase from $\geq$44.3 to $\geq$46 will increase the reliability to detect TCV fast closure events for RPS and EOC-RPT initiation. There are no unreviewed safety questions or reductions in the margins of safety associated with increasing the TCV fast closure trip setpoint to $\geq$46 psig.

Evaluation Number: 2000-0039-R00

Document Evaluated: LDC 2000-048, ER 2000-0074-00-00, 2000-0074-01-00 and SCN 2000-0003A against GGNS-DCS-01

BRIEF DESCRIPTION OF CHANGE, TEST OR EXPERIMENT:

ERs 00/0074-00-00, 00/0074-01-00, SCN 00/0003A against GGNS-DCS-01 and SCN 00/0004A against GGNS-DCS-01 are processed to authorize the following modifications for Inboard MSIVs 1B21-F022A-D and Outboard MSIVs 1B21 -F028A-D.

- Installation of a back-seated poppet, including live-loaded packing
- Installation of a nose guide poppet
- Incorporation of anti-rotation features
- Installation of a floating pilot poppet
- Installation of a forged, one-piece stem

The associated LDC revises the MSIV description found in UFSAR Section 5.4.5 and as depicted on Figure 5.4-9.

REASON FOR CHANGE, TEST OR EXPERIMENT:

Previous Local Leak Rate Testing failures of the MSIVs have, at times, resulted in GGNS exceeding the allowable Technical Specification leakage limits. A Significant Event Response Team (SERT) investigated the failures for the failures to determine the root causes and contributing causes. To address the SERT findings, the valve manufacturer (Atwood & Morrill Co.) proposed several modifications that have evolved as various plants have experienced MSIV failures. GGNS evaluated the vendor recommendations and concluded that the best balanced choice to address the failures is a combination of modifications (as listed above).

SAFETY EVALUATION SUMMARY AND CONCLUSIONS:

The proposed modifications incorporate lessons learned with regard to MSIV problems that have occurred throughout the industry. The changes have been recommended by the valve vendor and have been thoroughly evaluated with regard to material compatibility, weight changes, mechanical loading, flow characteristics, seismic qualification, etc. and has determined that the modified valves meet all requirements of the original valve design. The changes do not affect design conditions of the MSIVs, including environmental conditions. All required materials are procured as safety-related. The existing level of redundancy, diversity and separation are not affected. The changes will not affect the function of the valves or how the valves interact with other systems and components. The modified assemblies will be rated for the system operating parameters consistent with the originally supplied assemblies. No new missile sources are postulated and line breaks are unaffected. The modified valve internals are expected to provide an increased level of reliability from the originally supplied items mainly due to the anti-rotation features. The changes provide a more robust valve design that can be expected to outlast the original design. Stress limits remain within the ASME Code requirements. All modifications and materials are in accordance with the Codes and Standards cited in the MSIV specification, or have been reconciled to be equal to or better than the original materials. Pipe stress and pipe support analyses have been evaluated and found to be acceptable for the proposed changes. The modifications will assist in valve closure and seat tightness by addressing the root and contributing causes of previous failures; otherwise, valve operating characteristics are not affected. The ability of the MSIVs to function as required by various accident and transient analyses, including valve stroke times, are not affected by these changes. Impacts to UFSAR text and figures that describe the MSIVs have been identified and are included as shown on page 1 of this evaluation.

Since the required design functions and safety analysis criteria continue to be met with the proposed modifications, the changes will not increase the probability of occurrence or consequences of previously evaluated accidents or equipment malfunctions, or create the possibility for an accident or a malfunction of equipment important to safety of a different type than that previously analyzed. In addition, the changes do not affect the Technical Specifications or the basis for any margin of safety. Therefore, these changes do not involve an unreviewed safety question.

Evaluation Number: 2000-0040-R00 Document Evaluated: LDC 2000-035

BRIEF DESCRIPTION OF CHANGE, TEST OR EXPERIMENT:

Change the sentences requiring RP personnel to bring radiation survey equipment to a fire scene; to requiring "RP personnel shall respond to fires involving radiologically posted areas with appropriate instrumentation. RP may not respond to fire outside radiologically posted areas."

REASON FOR CHANGE, TEST OR EXPERIMENT:

SAR Section 9B.7.1 requires RP personnel to bring radiation survey equipment to monitor the fire area for potential radiation exposure hazards. It does not take into account fires outside radiologically posted areas which do not need RP support since there is no threat of radiation exposure.

SAFETY EVALUATION SUMMARY AND CONCLUSIONS:

By changing SAR Section 9B.7.1 the original intent of the SAR will be retained while eliminating the unnecessary requirement for RP personnel to respond to fires where no radiation exposure hazard exists. The change should read as follows:

"RP personnel shall respond to fires involving radiologically posted areas with appropriate instrumentation. RP may not respond to fire outside radiologically posted areas."

Evaluation Number: 2000-0041-R00 Document Evaluated: LDC 2000-033

BRIEF DESCRIPTION OF CHANGE, TEST OR EXPERIMENT:

GGNS has developed a Plant Operations Manual that incorporates a battery capacity test and changes the current functional test frequencies, to enhance the overall reliability of the 8 hour Emergency Lights required for Appendix R considerations.

REASON FOR CHANGE, TEST OR EXPERIMENT:

GGNS has emergency lighting units with at least an 8-hour battery power supply provided in all areas needed for operation of safe shutdown equipment and in access and egress routes thereof, as required per 10CFR50 Appendix R. These Appendix R emergency lighting units are currently tested and inspected on a weekly basis by performing a 60 second functional test and visual observation of indicators. In addition to this weekly testing, a more detailed monthly test and inspection is performed by 07-S-12-108 which performs the same functional test, checks/verifies voltages, electrolytic levels and provides general cleaning instructions.

In an effort to improve the testing of these emergency lighting units to incorporate a battery capacity test, GGNS utilized several industry documents to facilitate uniformity with the industry. These documents referenced are EPRI TR-100249R1, *"Emergency Battery Lighting Unit Maintenance and Application Guide"* and EPRI TR106826, *"Battery Performance Monitoring by Internal Ohmic Measurements".* The result is development of Plant Operations Manual 07-S-12-143 which will incorporate the recommendations of these documents to improve the scope of testing for these Appendix R emergency lighting units by verifying the capacity of the batteries installed is adequate and provides confidence that the lights will be available for the required 8-hours in addition to the current testing requirements of 07-S-12-108.

EPRI TR-100249R1 recommends functional testing these emergency lighting units on a Quarterly basis and battery capacity testing and other detailed inspection on a Semi-annual frequency. These changes will align GGNS with recognized industry standards and improve overall testing of the Appendix R emergency lighting units.

Currently GGNS SAR Section 9.5.3.4 states, "The normal DC lighting system will be inspected and tested weekly to ensure the operability of the automatic switches and other components in the system".

SAFETY EVALUATION SUMMARY AND CONCLUSIONS:

This new test procedure, in conjunction with the periodicity change for functional testing from weekly to quarterly to be consistent with industry standards and will enhance the assurance of performance of the Appendix R 8-hour emergency lighting units. This change will not increase the probability of occurrence or consequences of an accident previously evaluated in the SAR. Nor will it increase the probability of occurrence or the consequences of a malfunction of equipment important to safety previously evaluated in the SAR. It will not create a new equipment malfunction not evaluated in the SAR nor will it create the possibility of an accident of a different type than any previously evaluated in the SAR. No margin of safety as defined in the basis for any Technical Specification will be affected. These emergency lighting units have a passive interface with some equipment important to safety since they are mounted for Seismic II/I considerations, however, since their mounting configuration is not affected by these changes no change in any interface will result. They are listed in Appendix B to the GGNS Q-List. They are used to illuminate equipment and ingress/egress routes during a period when the plant may be required to be remotely shutdown outside the control room. The improved testing via this new procedure will provide a means to test and verify the emergency light units' battery capacity to ensure they will perform their intended function for the required time period as required per 10CFR50, Appendix R. Extending the functional check and visual inspection from weekly to quarterly will not adversely affect the operation nor decrease any assurance of operation by these emergency lighting units. No unreviewed safety question will result from this modification. All illumination levels currently provided by these lighting units will remain unchanged. GGNS UFSAR Section 9.5.3.4 requires updating to change the referenced period for testing from weekly to quarterly.

Evaluation Number: 2000-0042-R00

Document Evaluated: LDC-2000-057
ER 1997-0285-01-00

BRIEF DESCRIPTION OF CHANGE, TEST OR EXPERIMENT:

ER 97/0285-01-00 provides for replacement of valve 1E22F004 with a valve having a higher ANSI pressure rating. The existing Limitorque actuator is being retained except that the 100 ft-lb actuator motor (12.8 hp) is replaced with a 150 ft-lb actuator motor (19.2 hp). Equalization line 3/4"-DBB-164 and associated valve 1E22F305 are being retained to continue to provide protection against pressure-locking; however, the connection point of the line at the valve is being moved from the valve bonnet to the body. Valve internal components (i.e. valve disc, guides and seat) have stellite hard facing to improve performance of the valve. After performance of the modification, the valve will be stroke time tested to establish a new baseline and to ensure that new stroke times continue to meet USFAR requirements.

Associated LDC No. 2000-057 revises UFSAR Tables 8.3-3 and 8.3-4, and Figures 6.2-77, 6.3-1 and 8.3-12b to reflect the change in motor ratings and relocation of the pressure equalization line connection point. The LDC also revises UFSAR Table 3.9-2ac to capture design and stress values for the replacement valve.

SCN 00/0001A removes 1E22F004 from the scope of GE Purchase Specification 21A9457, and SCN 00/0010A adds 1 E22F004 to the scope of GGNS-M-242.0. SCN 00/0009A revises information for E22F004 in GGNS-MS-25.0. SCN 00/0005A changes the applicable drawing for 1 E22F004 referenced in GGNS-DCS-01.

Additionally, calculations PC-Q1E22-00001, Rev. 0, MC-Q1111-93035 Rev. 12, M-242.0-Q1E22F004-8.0-1, Rev. 0, MC-Q1111-91123 Rev. 24, EC-Q1111-90016 Rev. 13, EC-Q1111-90028 Rev. 5, EC-Q1E22-00-001 Rev. 0, NPE-PDS-63 Rev. 9, NPE-PDS-63 Supplement 1, Rev. 0, NPE-PDS-63 Supplement 2, Rev. 0 NPE-PDS-2704 Rev. 1, and SR-002 Rev. 1 were developed to support the changes.

REASON FOR CHANGE, TEST OR EXPERIMENT:

Entergy is implementing a "margin improvement" program to enhance the operation, maintenance, and reliability of selected Motor Operated Valves (MOVs) at the Grand Gulf Nuclear Station. By replacing specific components of selected valves, the margin between required torque/thrust versus maximum available torque/thrust can be increased. The current configuration for actuator 1E22F004 provides sufficient torque/thrust to operate the subject valve under design conditions; however, the operating thrust margin (i.e., the difference between the actuator output thrust required to stroke the valve and the thrust capability of the actuator with associated inaccuracies) is small. The operating torque/thrust margin improvement on MOV 1E22F004 will be accomplished by increasing the output capability of the actuator by installing a larger motor and by providing a valve suitable for the increased forces applied by the actuator.

SAFETY EVALUATION SUMMARY AND CONCLUSIONS:

This Safety Evaluation documents the fact that the proposed changes do not result in an Unreviewed Safety Question for the following reasons. The changes do not cause a greater reliance to be placed on any specific system, structure or component to perform a safety function. The changes do not degrade the performance or reliability of a safety system assumed to function in the accident analysis. The changes do not put the plant operation in an unanalyzed region. The changes herein are bounded by the analysis in the Technical Specifications, the TRM and the SAR. Moreover, the changes do not adversely affect the overall performance or reliability of a safety system in a manner that could lead to an accident occurring.

Because the changes described above will meet or exceed the requirements of the original design (component integrity, capacity, functionality, code requirements, etc.) and existing analyses, the changes will not degrade the performance of any important to safety systems, components or structures, nor will they degrade or prevent actions described in the SAR accident analysis. The changes do not increase the probability of occurrence or increase the consequences of an accident or malfunction of equipment important to safety previously evaluated in the SAR, and do not create a different type of accident or malfunction than previously evaluated in the SAR. The Technical Specifications and Technical Requirements Manual are not affected, and the margin of safety as defined in the basis for any Technical Specification remains unchanged. Therefore, these changes do not constitute an Unreviewed Safety Question.

Evaluation Number: 2000-0043-R00

Document Evaluated-Procedure:
Temporary Alteration 2000/007

BRIEF DESCRIPTION OF CHANGE, TEST OR EXPERIMENT:

Temporary Alteration 00/007 will allow installation of a pressure gauge on one or all three of the P21 System Sulfuric Acid Storage Tanks (SP21A003A, A003B, and or A003C). Once installed, the gauge can be used by Operations Department personnel to monitor internal pressure in the storage tank. The pressure gauge will be connected using temporary tubing and a spare flange connection at the top of each of the acid storage tanks. The connection is above the normal liquid level of acid stored in these tanks. This will eliminate the potential for inadvertent draining of the acid storage tanks. The gauge and associated tubing will be within the confines of the existing concrete containment berm around the acid storage tanks. In the event leakage of acid should occur as a result of installing the pressure gauges or through subsequent use of the gauges, the berm will be capable of capturing and retaining the acid. As such, installation of the temporary gauges will not represent any new or abnormal environmental concerns.

REASON FOR CHANGE, TEST OR EXPERIMENT:

The design rated pressure of the acid storage tanks (SP21A003A, B, and C) is 15 psig based on data available from Data Sheet included in Specification M-100.0. The storage tanks are vented to the atmosphere through a desiccant-filled breather. As documented in CR-GGN-2000-0746, the breather screens have been observed to clog and even rupture or catastrophically fail. While the storage tanks are normally maintained at atmospheric pressure, the internal pressure increases during acid off-loads. If the vent (breather screen) is blocked due to screen clogging or failure, the increase in tank pressure can potentially challenge the design rated pressure for the acid storage tanks. The pressure gauges installed by this Temp Alt will provide Operations Department personnel with a means to monitor internal tank pressure during normal operations as well as during acid off-loads. Based on tank pressures, Operations Department personnel will be in a position to safely off-load sulfuric acid and if necessary, terminate acid off-loads to minimize total tank pressure.

SAFETY EVALUATION SUMMARY AND CONCLUSIONS:

The Sulfuric Acid Storage Tanks, SP21A003A, A003B, and A003C are part of the P21 System. As stated in UFSAR Section 9.2.3.3, the P21 System serves no safety function aside from the Containment Isolation Valve function provided by affected P21 System valves. There are no Containment Isolation Valves or other safety related components in or around the sulfuric acid storage tanks and as such, the proposed change will not impact the plant's safety-related components or structures. The scope of the proposed change is limited to installing a temporary tubing connection and pressure gauge to one, two or all three of the acid storage tanks. The pressure gauges will be mounted locally to existing structures in the immediate area of the acid storage tanks. As such, there is no potential for the proposed change to impact the availability or operation of other plant equipment or systems. The proposed change directly affects the sulfuric acid storage tanks, however, this change will not adversely impact other

P21 System equipment. The impact of the proposed change on the sulfuric acid storage tank vessels will be limited to removal of a blind flange and installing a temporary flange, tubing and a pressure gauge in place of the blind flange. The materials used in the proposed change will be selected such that they will provide adequate structural characteristics to maintain the integrity of the storage tank. As such, based on evaluation of the proposed change, it has been determined that the temporary pressure gauges can be installed and used without adversely impacting the plant equipment or structures. Neither the GGNS Technical Specifications or the Technical Requirements Manual (TRM) address the Sulfuric Acid Storage Tank portion of the P21 System. As such, the proposed change will not conflict with or necessitate a change to these documents. Based on conclusions reached by this evaluation, implementation of the proposed change does not introduce or represent an Unreviewed Safety Question or an Unreviewed Environmental Question.

Evaluation Number: 2000-0044-R00

Document Evaluated: LDC 2000-040, ER 2000/0081-00-00

BRIEF DESCRIPTION OF CHANGE, TEST OR EXPERIMENT:

This ER installs flush taps in portions of the Loop A and B piping of the Residual Heat Removal (RHR) system in the Auxiliary Building and will replace associated 1" drain lines and valves with 2" drain lines and valves to facilitate the removal of crud from inside of the RHR piping.

Associated LDC No. 2000-040 makes the following changes: UFSAR Figures 5.4-16 Sheet 1, 5.4-16-02 and 5.4-17 are being revised to reflect the piping changes described above.

Additionally, calculations MC-Q1E12-00-011 Rev. 0, MC-Q1E12-00-012 Rev. 0, MC-Q1E12-00-013 Rev. 0, MC01E12-00-014 Rev. 0, XC-Q1J11-96007 Rev. 1, XC-Q1111-92010 Rev. 5, NPE-PDS-2000 Rev. 5, NPE-PDS-2006 Rev. 3, NPE-PDS-2320 Rev. 3, NPE-PDS-46 Supplement 1, NPE-PDS-204A Supplement 1, NPE-PDS-69A Supplement 1, NPE-PDS-204 Supplement 1, NPE-PDS-69C Supplement 1, NPE-PDS-65 Supplement 1, NPE-PDS-365 Supplement 1, N1E12G275A01 Supplement 1, Q1E12G168H01 Rev. 1, and N1E51G159C03 Rev. 0 were developed, revised or supplemented to support the changes.

REASON FOR CHANGE, TEST OR EXPERIMENT:

Some of the RHR headers in the Auxiliary Building currently have high dose rates as a result of buildup of radioactive material in the piping due to infrequent use. To reduce the radiation levels in the vicinity of the piping, flush taps are being installed to facilitate the use of hydrolasing to clean the piping. To accommodate the hydrolasing, the 1" drain lines and valves associated with these headers are being replaced with 2" drain lines and valves.

SAFETY EVALUATION SUMMARY AND CONCLUSIONS:

The RHR system is discussed in various sections of the UFSAR and shown schematically in Figures 5.4-16 Sheet 1, 5.4-16-02 and 5.4-17, RHR System Piping and Instrument Diagrams. The addition of the flush taps and the replacement of 1" drain lines and valves with 2" drain lines and valves does not change the description of the RHR system contained in the text of the UFSAR, however, it does change Figures 5.4-16 Sheet 1, 5.4-16-02 and 5.4-17.

An electronic search was performed of the GGNS UFSAR, Fire Hazards Analysis, SER (including supplements), Technical Specifications, TRM, Operating License, Design Engineering Criteria documents (SDCs, ABDs, and TDCs). Keywords (and their variations) used for the search were: flush, tap(s), flange(s), crack(s), moderate-energy 1E12-F053A,B and 1E12-F050A,B. The following relevant UFSAR Sections, Tables and Figures were identified and reviewed for impact: Table 3.2-1 Items IX-4 and IX-9; Sections 3.5.1.1.2; 3.5.1.1.3; Table 3.5-5; Section 3.6A, Figures 3.6A-4, 3.6A-6, 3.6A-9, 3.6A-12 and 3.6A-25; Table 3.9-3C; Appendix 3C, Section 5.4.7.2.7; Figures 5.4-16 Sheet 1,5.4-16-02, and 5.4-17;

Table 5.4-3; Sections 6.3, 7.6.1.3.3 and 7.6.1.3.3.1, 12.1.2.2, 15.6.6; (TRM) Chapter 16 Tables TR3.4.6-1 and Table 6.8.2-1. Additionally, SER Sections 3.5.1.1 and 12.1 were identified and reviewed for impact. The only change to the licensing basis documents identified was to UFSAR Figures 5.4-16 Sheet 1, 5.4-16-02, and 5.4-17 to show the new flush taps, increased size for drain lines and valves, and applicable notes denoting size of flush taps.

The proposed changes are within the existing licensing basis of the Grand Gulf Nuclear Station. This Safety Evaluation documents the fact that the proposed changes do not result in an Unreviewed Safety Question for the following reasons: the changes do not cause a greater reliance to be placed on any specific system, structure or component to perform a safety function; the changes do not degrade the performance of a safety system assumed to function in the accident analysis and do not decrease the reliability of safety systems assumed to function in the accident analysis; the changes do not put the plant operation in an unanalyzed region; the changes herein are bounded by the analysis in the Technical Specifications, the TRM and the SAR; and the changes do not adversely affect the overall performance or reliability of a safety system in a manner that could lead to an accident occurring. Because the changes described above will meet or exceed the requirements of the original design (component integrity, capacity, functionality, code requirements, etc.) and existing analyses, the changes will not degrade the performance of any important to safety systems, components or structures nor will they degrade or prevent actions described in the SAR accident analysis. The changes do not increase the probability of occurrence or increase the consequences of an accident or malfunction of equipment important to safety previously evaluated in the SAR and do not create a different type of accident or malfunction than previously evaluated in the SAR. The Technical Specifications and Technical Requirements Manual are not affected, and the margin of safety as defined in the basis for any Technical Specification remains unchanged. Therefore, these changes do not constitute an Unreviewed Safety Question.

Evaluation Number: 2000-0045-R00

Document Evaluated: LDC-2000-042
(ER 1997-0022-02-00)

BRIEF DESCRIPTION OF CHANGE, TEST OR EXPERIMENT:

In order to resolve the over-pressurization issue of Generic Letter 96-06, the following modifications or actions are necessary for some of the penetrations evaluated for these conditions. The penetrations listed below were identified in Engineering Report GGNS-97-0002 Rev 1, CR GGN-1999-1147 and CR GGN1999-1256.

Pen No.	Line No. to be protected from Over Pressurization	Add Relief Valve on Line No.	New Relief Valve Component No.	New Relief Valve Set Pressure (Max Op. P)	Replace Bolts with High Strength Bolts for Valve No.	Mod. Location	Revise Ops Proc.
18 & 311 (Guard Pipe)	6"-DBA-30 6"-DBA-32	See Note § Below					
19 & 312 (Guard Pipe)	3"-DBA-23	4"-EBB-1 4"-EBZ-5					
43	4" & 6" EBB-1	8"-HBD-394	Q1G33F264 Q1G33F261	1430 (1105) 1430 (1105)		AUX CTMT	
44	10"-HBB-35	8"-HBD-429	Q1P42F263	270 (65)		CTMT	
45	10"-HBB-38	6"-HBD-43	Q1P42F264	270 (65)		CTMT	
54	12"-HBB-4						
56	6"-HBB-34	6"-HBD-43	Q1P11F151	170 (110)	Q1P11F075	CTMT AUX	
86	2" & 4"-HBB-155	2"-HBB-155 2"-JCD-33	Q1P21F390 Q1P21F391	300 (75) 270 (65)		CTMT CTMT	
87 & 325 (Guard Pipe)	6"-DBA-9 & 6"-DBA-90	See Note § Below					
88	4"-DBB-103	4"-DBB-103 4"-DBZ-3	Q1G33F263 Q1G33F262	1600 (1220) 1600 (1220)		CTMT CTMT	
90	2" & 3"-HBB-143						

<u>Note *</u>: The ¾' -DCD-25 line is newly added along with pipe support C02 for this modification.

<u>Note §</u>: Penetrations 18 and 311 currently should be addressed in a separate modification package ER 2000-0083-00-02. Penetrations 87 and *325* should also be addressed in a modification package ER 97-0022-03. ER 2000-0083-00-02 and ER 97-0022-03 must be implemented prior to start-up from RFI1 to avoid over-pressurization of the penetration piping.

Pen No.	Line No. to be protected from Over Pressurization	Add Relief Valve on Line No.	New Relief Valve Component No.	New Relief Valve Set Pressure (Max Op. P)	Replace Bolts with High Strength Bolts for Valve No.	Mod. Location	Revise Ops Proc.
91	2" & 3"-HBB-148						
329	8"-HBB-36	8"-HBD-394	Q1P42F263 Same RV as Pen. 45	270 (65)	Q1P42F114	CTMT	
330	8"-HBB-37	8"-HBD-429	Q1P42F263 Same RV as Pen. 45	270 (65)		CTMT	Same as Pen. 45
366	6"-DBB-140	6"-DBB-140	Q1G33F265	1600 (1220)		CTMT	
465	¾"-DCB-6	¾" – DCB-6 ¾"–DCD-25	Q1B33F259 Q1B33F260	1600 (1100) 1600 (1100)		CTMT CTMT	

This ER will provide instructions for the installation of a relief valve on each of lines 4"-EBB-1, 4"-EBZ-5, 8"-HBD-394, 8"-HBD-429, 6"-HBD-43, 2"-HBB- *155,* 2"-JCD-33, 4"-DBB- 103, 4"-DBZ-3, 6"-DBB- 140 and %"-DCB-6 to provide pressure relief for piping penetrations 43, 44, 45, *56,* 86, 88, 329, *330,* 366 & 465. The ER will also replace body to bonnet flange bolts for gate valves Q1P11F075 (Ctmt Pen. *56)* and QIP42F114 (Dwl Pen. *329).* In addition, appropriate Operations procedures will be revised to add operational limitations and/or ensure that there is no fluid trapped in the piping between the isolation valves for penetrations 19, 45, *54,* 90, 91, *312* and *330.*

SCN 00/0001A for M-912.0 Rev 0 to include the new relief valves Q1G33F261 (pen. 43), Q1G33F264 (pen.43), Q1P42F263 (pen. 44 & 329), Q1P42F264 (pen. 45 & 330), QIPI1F151 (pen. 56), Q1P21F390 (pen. 86),Q1P21F391 (pen. 86), Q1G33F262 (pen. 88), Q1G33F263 (pen. 88), Q1G33F265 (pen. 366), Q1B33E259 (pen. 465) & Q1B33F260 (pen. 465).

SCN 00/0009A for MS-02 Rev 49 to add a reference note for GL 96-06 predicted pressures, and also include the new line number(s).

SCN 00/0009A for M-242.0 Rev 57 to include the GL 96-06 pressures for valves G33F028 (pen. 43),G33F034 (pen. 43), P42F035 (pen. 44), P42F066 (pen. 44), P42F114 (pen. 329), P42E068 (pen. 45), P42F067(pen. 45), P42F117 (pen. 330), P42F116 (pen. 330), P11F075 (pen. 56), P11F004 (pen. 56), G41F047 (pen.56), B33F205 (pen. 333), G33F054 (pen. 88), G33F053 (pen. 88), G33F252 (pen. 366) & G33F253 (pen.366).

SCN 00/0002A for M-25 1.0 Rev 37 to include the GL 96-06 pressures for valves P21F017 (pen. 86), P21F018 (pen. 86), B33F019 (pen. 465), B33F020 (pen. 465), B33F128 (pen. 47), B33F021 (pen. 465) & B33F129 (pen. 465).

Licensing Document Change 2000~042 to revise TRM and UFSAR tables to include the new relief valves Q1G33F264, Q1P21F390, Q1G33E263, Q1G33F265 & Q1B33F259 added between the isolation valves for penetrations 43, 86, 88, 366 & 465 respectively.

Implementation of ER 97-0022-02, ER 97-0022-03 and ER 2000-0083-00 and closure of the licensing commitment A-34648 and A-33119 will complete the resolution of GL 96-06.

Enclosure 1 to this 10CFR50.59 evaluation provides sketches for the various scenarios related to this modification.

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.

Engineering Report GGNS-97-0002 initially identified 18 Grand Gulf penetrations, 12 Containment (36, 39, 43, 47, 49, 50, 51, 54, 58, 81, 84 & 86) and 6 Drywell (330, 331, 333, 348, 349 & 364), susceptible to increased pressures per GL 96-06. All but four penetrations (43, 54, 86 & 330) were addressed in ER's 97-0022-00-01 and 97-0022-01-01. Additional reviews of the GL 96-06 issue resulted in the identification of 16 additional susceptible penetrations via CR GGN-1999-1147 (38, 56, 87. 88, 325, 366 and 465) and CR GGN1999-1256 (18, 19, 44, 45, 90, 91, 311, 312, 329). Penetration 38 has been already addressed in ER's 97-0022-00-01 and 97-0022-01-01 as it was worked in tandem with penetration 39. Penetrations 18 and 311 will be addressed in a separate modification package ER 2000-0083-00-02. ER 97-0022-02 addresses 15 of the 17 remaining penetrations (19, 43, 44, 45, 54, 56, 86, 88, 90, 91, 312, 329, 330, 366 & 465) and ER 97-0022-03 addresses the last two penetrations 87 and 325. ER 97-0022-02 provides twelve new relief valves and replaces bolts for two isolation valves for the affected penetrations.

Note: Throughout this Safety Evaluation and ER, the terms "inboard" and "outboard" refer to location relative to the Reactor Pressure Vessel with "inboard" being closer to the RPV than "outboard". This applies to valves as well as valve discs.

SAFETY EVALUATION SUMMARY AND CONCLUSIONS:

The following changes meet 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 penetrations addressed in the ER:

- **Installation of relief valves for lines 4"-EBB-1, 4"-EBZ-5, 8"-HBD-394, 8"-HBD-429, 6"-HBD-43, 2"-HBB-155, 2"-JCD-33, 4"-DBB-103, 4"-DBZ-3, 6"-DBB-140, 3/4"-DCB-6 and *3/4"-DCD-25* (total 12 relief valves)** to provide pressure relief for 10 Containment and Drywell piping penetrations 43, 44, 45, 56, 86, 88, 329, 330, 366 & 465.

- **Replacement of body to bonnet flange bolts for valves Q1P11F075 (Containment Pen. 56) and Q1P42F114 (Drywell Pen. 329)** to assure the structural integrity of these valves when subjected to increased pressures.

- **Installation of a new ¾" pipe connected to the existing tubing line for penetration 465. Installation of a new pipe support** for the newly added pipe.

- **Revision to appropriate Operations procedures** to ensure there is no fluid trapped in the piping between the isolation valves for penetrations 19, 45, 54, 90, 91, 3 12 & 330.

See Enclosure 1 for detailed sketches and discussion of pressure relief scenarios.

The installation of the relief valves will consist of installing 3/4" or 1/2" branch connections off the main pipes, each with a set of flanges and a safety related relief valve attached at its end. Since the relief valves are intended to protect the safety related piping between the isolation valves, the installed relief valves are procured to ASME Section III requirements.

The relief valves are 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 valves are not intended to continuously preserve the safety-related piping pressure boundary during a normal or upset condition. For this reason the relief valves can be considered to perform a passive function during normal system operation and are typically set to a pressure well above the line design 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. Based on the information provided in the above table, there is sufficient margin between the operating pressure and the relief valve set pressure to assure that the valves are not going to be challenged. Additionally, these relief valves will be tested at least once every ten 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 valves set pressures will not be challenged during normal or upset operations, they are therefore expected to remain leak tight.

The relief valves 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 valves are not critical design parameters and the ½" or ¾" valves installed are obviously adequate for this purpose.

The relief valves will be added in sections of pipe such that no existing stop valve or other device could reduce the penetration overpressure protection. In the case of penetrations 45 and 330, administrative controls are being initiated to assure that only one valve is closed for each of the penetrations so the relief valves would perform their safety function.

There are no ASME Code requirements dictating the installation of tail pipes. Also a cursory review of 29CFRI910 "Occupational Safety and Health Standards" has been conducted. The review revealed no OSHA requirements related to the use of relief valve tail pipes. As a precaution however, the relief valves discharge nozzle has been oriented in a way not to directly affect any adjacent equipment.

The newly installed relief valves for Reactor water systems penetrations 43, 88, 366 & 465 are not located in normally accessible areas and do not represent a personnel hazard should unlikely relief valve discharge occur. Only the new relief valves installed for penetrations 44, 45, 56 and 86 are located in the Containment building in normally accessible areas. However, these valves are located in the overhead areas on moderate energy "clean" systems and cannot directly discharge on plant personnel. The only improbable personnel exposure would be warm (130 ^{0}F maximum) CCW or Demineralized water falling/dripping down.

There are no ASME Code requirements dictating the installation of rupture discs. The only requirement is that a rupture disc, if selected for installation, must be installed in conjunction with a relief valve. Therefore installation of rupture discs may be considered only a variation of the proposed design. It should be noted that each rupture disc would require the installation of an additional pipe support (tieback to the header pipe) as well as a pressure gauge or a free vent between the valve and disc to permit the detection of disc rupture or leakage. It should also be noted that the disc will not burst at its design pressure if back pressure builds up in the space between the disc and the safety relief valve which will occur should leakage develop in the rupture disc due to corrosion or other causes.

The above installations of relief valves will have no impact on current safety analyses (reference safety analysis group response to EAR MC-00-0010).

Since the relief valves 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 are 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 branches (including relief valve and flanges) have been evaluated along with the existing piping in stress calculations PDS-2193 Supplement 1 Rev 0 and PDS-2741 Rev 0 for all plant conditions (including the elevated relief pressure) to meet the design requirements of ASME Section III, Subsections NC-3600. Code Case 1606-1, ANSI B31.I, M-18 and drawing 9645-M-1398. Also new revised calculations NPE-P42F066/F067/F068 Rev. 4, CC-QI111-99001 Rev.2, NPE-EslF059 Rev. 6, NPE-641F029/ F044/ P42F114/ FI16/ FI17 Rev. 8, and NPE-E12F394/G33F001/F004/F250/F251/ F252/F253 Rev. 12 were necessary to qualify the affected isolation valves.

There are no pipe break jet impingement cones postulated in the area of the newly added relief devices and the penetrations boundaries. Therefore, the 3/4" or 1/2" lines 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 lines are installed above elevation 144' and are located at least 20 ft above the normal suppression pool elevation (11 1'-I0"). This will eliminate any significant poolswell loads acting on the and relief valve connections.

Isolation valves QIPI1F075 (Containment Pen. 56) and QIP42FI14 (Drywell Pen. 329) will require replacement of the existing SA-193 Grade B (allowable stress value S = *25* ksi) body to bonnet flange bolts with bolts of SA-540, grade B22 Class 1 material (S = 33 ksi). The higher strength bolt material will allow the valves to withstand elevated predicted pressures. Calculations CC-Q1111-99001, Rev.2 for valve P11F075 and NPE-G41F029/F044/P42F114/FI16/FI17, Rev. 8 for valve P42F1 14 demonstrate the structural integrity of the valves with this change.

The equivalent thrust force to close the affected air operated valves (P11F075 and G41F047 for penetration 56) is based on the current requirement of maintaining an 80 psig actuator pressure. However there is a potential that the post closure air pressure will be considerably less than 80 psi while the line pressure may increase to the new relief valve relieving pressure of 187 psig during a LOCA event. The LOCA pressure will then exceed the current MS-02 line pressure (normal pressure is 50 psig, maximum pressure is 100 psig and design pressure is 150 psig). An evaluation has been performed to ensure that stem ejection forces cannot exceed expected valve unwedging forces during accident conditions. Therefore, undesired opening of AOV isolation valves P11F075 and G41F047 will not occur due to stem ejection forces induced by valve internal pressure (reference response to EAR MC-00-0011)
Pool level increase due to relief valve discharge during a LOCA condition is insignificant when compared to level increase from other sources. No adverse chemistry concerns are created post LOCA by the introduction of small quantities of water from these systems into the suppression pool. The valves are designed for relief pressures well above maximum operating pressures for the systems to prevent inadvertent discharges. These relief valves will be periodically disassembled and inspected in compliance with the plant relief valve program

CEP-IST-002, Rev. 0. The new relief valves will be tested in accordance with procedure 07-S-14-395 "General Maintenance Instruction — Safety and Relief Valve Program — Safety Related".

After the P21 (penetration 86) system relief valves are installed (set pressure 300 psig), the pressure increase described in GL 96-06 could cause the P21 system to experience pressures above 275 psig. However this will occur for less than 2% of the total system operation time. Therefore the classification of the P21 lines as moderate energy remains applicable, and there is no change to the original HELB/MELB evaluation. Since these new relief valves and branch pipes are installed on moderate energy lines, the branch line connections can not eject as a missile due to a failure at the connecting weld.

The new P42 (penetration 44, 45, 329 & 330) and P11 (penetration 56) relief valves and branch pipes are installed on moderate energy lines, therefore the branch line connections cannot eject as a missile due to a failure at the connecting weld. On the other hand, the new relief valves and branch pipes for the G33 (penetrations 43, 88, 366) and B33 (penetration 465) systems are installed on high-energy lines. Walkdowns have been performed to identify potential targets should the branch line connections eject as a missile due to a failure at the connecting weld. No potential targets were identified as a result of the walkdowns.

Divisional failure possibilities were reviewed for all penetration valves. Various failure scenarios were considered and no new unevaluated effect due to this modification was identified.

No hardware modification is required for penetrations 19 & 312. However implementation of Operations procedural controls is necessary to ensure there is no fluid trapped in the piping between the isolation valves. Design Engineering recommends maintaining inboard isolation valve B21F016 closed once control valve B21F033 is fully closed (i.e. at> 50% power). Opening B21F021 for a brief period of time to ensure the line is drained is recommended especially if B21F016 is closed much later than B21F033. This will ensure that future condensed water will be prevented from being trapped between the Containment/Drywell isolation valves. No water hammer is expected in the line upon opening of valve B2IF016. The condensed water upstream of the valve will be below 150 ^{0}F (reference 30) and the valve opening is slow and will take 16-20 ≤seconds (TRM table TR3 .6.1.3-1, Page 3.6-17-I). Therefore, no flashing into steam will occur and no significant loading due to unbalanced fluid transient forces will be present. The possibility exists for leakage past valve B21F016 to cause the refilling of the penetration. However, since there is no vent path it is not considered credible that leakage could fill the penetration solid with water. At least a small air bubble will be present and will be adequate to prevent over-pressure. Also it can be easily postulated that if the valve has exhibited leakage during normal conditions, the leaked fluid into the penetration will tend to escape back in the opposite direction under accident over-pressure conditions.

Operations Alarm Response Instructions for penetrations 45 & 330 will be revised to close only Drywell outboard valve QIP42F117 and Containment Inboard valve QIP42F068 when isolating CCW and allow pressure relief via existing relief valves P42F225 and P42F255. The need to isolate is very unlikely since the CCW piping has been designed to resist all seismic/hydrodynamic events in compliance with II/I criteria and does not receive an automatic isolation signal so that CCW flow may continue to the Recirculation pumps on a LOCA event. No SOI or ONEP instructions require closing the isolation valves. The operators typically close the valves on a CCW in-Containment trouble alarm and surge tank level drop (make-up is slow). In the unlikely event where QIP42FI17 fails to close, Drywell inboard valve QIP42F116 may be isolated. Also in the unlikely event where Q1P42F068 fails to close, the Containment outboard isolation valve QIP42F067 may be closed. Since QIP42F116 and Q1P42F067 could isolate the Containment piping without relief, a relief valve (set pressure 270 psig) will be added downstream of valve P42F1 17 and upstream of P42F068. Therefore pressure in penetrations 45 & 330 will be limited to no more than the pressure required to lift the relief valve (i.e. 270 psig + 10% = 297 psig). This section of piping has been qualified in calculation PDS-2741 Rev 0. The valves have also been qualified for 300 psig in calculations NPE-P42F066/F067/F068, Rev. 4 & NPE-G41F029/F044/P42F114/FI 16/FI17, Rev. 8

No hardware modification is required for penetration 54. However implementation of Operations procedural controls is necessary to maintain this line drained between valves G41F215 and G41F053 to ensure there is no fluid trapped in the piping between the valves. The Operations Department will determine specific line-up, methodology and administrative controls necessary to achieve this action.

The isolation valves for SSW penetrations 90 and 91 are normally closed. The isolation valves for each penetration are both powered from the same electrical division. The valves receive an automatic opening signal on a LOCA. However a potential division failure would keep both isolation valves for a penetration closed.

No hardware modification is required for penetrations 90 & 91. However implementation of Operations procedural controls is necessary to ensure there is no fluid trapped in the piping between the isolation valves upon a divisional failure post LOCA. The ER requires maintaining one of the isolation valves open. This would eliminate the need to add a new relief valve between the isolation valves. Leaving one SSW Containment isolation valve open still allows SSW to meet the criteria of GDC 56 (Primary Containment Isolation). The normal post accident position of the valves is open. Therefore leaving one Containment isolation valve in the open position is anticipatory of the post accident condition of the SSW system and will provide greater safety. The one closed isolation valve provides Containment isolation in the event of single failure of the other isolation valve or power supply. The closed isolation valve provides Containment isolation with the other valve in the open position. Both valves will continue to be tested per the 1ST program.

All the above modifications and changes will assure piping systems and Containment integrity under over-pressurization conditions post LOCA.

Evaluation Number: 2000-0046-R00

Document Evaluated: LDC-2000-015
ER 2000-0118-00-00

BRIEF DESCRIPTION OF CHANGE, TEST OR EXPERIMENT:

This ER will remove the disc position indication switch, position switch IEI2NI13C, the position switch actuator rod and associated wires and conduit, from valve IEI2F04IC. It will also remove the indicating lights for the disc position indication of the valve from main control room panel 1H13P601-17C. The disc position indication switch is used for remote disc position indication during valve testing.

REASON FOR CHANGE, TEST OR EXPERIMENT:

Valve 1EI2F041 C has a history of the position switch actuator rod causing the valve disc to stick resulting in dual indication in the control room. This frequently requires valve disassembly to correct

SAFETY EVALUATION SUMMARY AND CONCLUSIONS:

No changes are being made that affect basic system design functions. The disc position indication switch is used for remote disc position indication during valve testing. The changes do not affect the ability of valve IEI2F04IC to perform its function under accident conditions. These changes do not result in a new pathway for the release of radioactive materials and do not affect offsite dose. No assumptions utilized in evaluating consequences of an accident will be altered. No new failure modes are created and there is no increase in previously identified failure modes for equipment important to safety. No assumptions utilized in evaluating the consequences of a malfunction of equipment important to safety will be altered by this change. This modification does not introduce any new failure modes and does not affect equipment other that the check valve and its associated disc position indication. Secondary and indirect effects (Fire protection, fire loading, pipe break, electrical shorts) have been reviewed and no increased probability of failure of equipment important to safety due to these concerns has been identified. These changes will not degrade any important to safety systems, components or structures nor will they degrade or prevent actions described in the SAR analysis. They do not increase the probability of occurrence or increase the consequences of malfunction of equipment important to safety of a different type than previously evaluated in the SAR. The Technical Specifications are not affected and the margin of safety is unchanged.

Evaluation Number: 2000-0047-R00

Document Evaluated: LDC 2000-044
ER 1998-0427-02-00

BRIEF DESCRIPTION OF CHANGE, TEST OR EXPERIMENT:

ER Package 1998/0427-02-00 physically removes Makeup Water Treatment equipment that is no longer in use. The equipment to be removed originally functioned to support the regeneration activities associated with water purification. This function is presently performed by a vendor provided water treatment trailer. The major components to be removed include the Caustic Dilution Water Heater Tank (SP21A005), the Sulfuric Acid Day Tank (SP21A007), and the two Acid Injection Pumps (SP21C003A/B). In addition to these larger components, associated valves, instrumentation, and piping will also be removed. SCN 00/0001A to 9645-M-111.0 and SCN 00/0010A to GGNS-MS-02 are being generated in conjunction with ER 1998/0427-02-00 to reflect the deletion of equipment and piping respectively. Supplement 1 to Revision 5 of Calculation EC-Q1111-90028 was developed to document the removal of electrical loads associated with the removed equipment.

LDC 2000-044 updates UFSAR text, tables and figures to reflect the removal of the equipment.

REASON FOR CHANGE, TEST OR EXPERIMENT:

The purpose of ER Package 1998/0427-02-00 is to physically remove the Makeup Water Treatment equipment to create an open area for the new Plant Air System air dryer skids which are to be installed per ER Package 1998/0427-00-00 (mechanical scope) and 1998/0427-01-00 (electrical scope).

SAFETY EVALUATION SUMMARY AND CONCLUSIONS:

The changes within the scope of this ER and associated SCNs do not impact any important to safety equipment. The equipment that is being removed has not been utilized since the early stages of the plant's life. UFSAR Section 9.2.3 and UFSAR Figure 9.2-11 note 28 currently reflect that the equipment that is being removed is not normally used in plant operation and that a vendor provided water treatment trailer is utilized instead. This ER permanently removes a portion of the equipment that has previously been functionally abandoned in place. As stated above, the equipment is being removed to allow the future installation of new Plant Air System air dryers that will replace the existing Instrument Air System air dryers. There is no adverse impact to the conclusions of Calculation EC-Q1111-90028, AC Electrical Power Systems Calculations, due to the removal of the electrical loads associated with the equipment removed since the removal of the relatively small loads actually has a positive impact on the electrical system.

These changes do not degrade below the current design basis the performance of a safety system assumed to function in the accident analysis and do not decrease the reliability of safety systems assumed to function in the accident analysis. These changes do not cause a greater reliance to be placed on any specific system, structure or component to perform a safety function. These changes do not adversely affect the overall performance or reliability of a safety system in a manner that could lead to an accident occurring. These changes do not cause a safety system to be operated outside of its design basis limits. Additionally, the margin of safety as defined in the bases for the Technical Specifications has not been reduced.

Evaluation Number: 2000-0048-R00

Document Evaluated: LDC 2000-045
ER 1998-0559-00-00

BRIEF DESCRIPTION OF CHANGE, TEST OR EXPERIMENT:

Modify the control logic for 1st stage SJAE condenser isolation valves 1N62-F003A&B such that a single 2nd stage steam flow transmitter failure will not cause the valve to close.

REASON FOR CHANGE, TEST OR EXPERIMENT:

The intent of ER 98/0559 is to improve the fault tolerance of the SJAE isolation function.

SAFETY EVALUATION SUMMARY AND CONCLUSIONS:

The proposed modification will alter the logic for closing the 1st stage SJAE condenser isolation valves on low 2nd stage steam flow. The current logic configuration would result in condenser isolation if a single downscale failure of a 2nd stage steam flow transmitter occurred. The modified logic will detect failures of the 2nd stage steam flow transmitters (upscale or downscale) and revert to a one-out-of-one scheme for isolation based on the remaining transmitter signal. With the proposed modification, a single transmitter failure downscale will result in annunciation in the control room only, not isolation. The modification will not change the isolation setpoint or the logic and setpoint for the existing low flow annunciator. The modification will add control room annunciation and computer points for 2nd stage SJAE flow transmitter failures and drift. The modification will not impact the Operators ability to control the condenser isolation valves with the existing control room handswitches.

Per UFSAR section 10.4.2.3, 'the condenser air removal system has no safety-related function. Failure of the system will not compromise any safety-related system or component and will not prevent safe reactor shutdown.' No new interfaces with safety related systems, structures or components will be created by the proposed modification.

The condenser air removal system serves no function to mitigate the radiological consequences of an accident or malfunction of equipment. The proposed modification will not increase the probability of an "Offgas System Leak / Failure' as described in UFSAR 15.7.1, and will not impact the radiological consequences of this event.

The change will not impact the assumed initial conditions for Offgas System Leak/Failure event or reduce the margin of safety as defined in the basis for the associated Offgas System Technical Specifications.

The proposed modification will not increase the probability of 'Loss of Condenser Vacuum' as described in UFSAR 15.2.5, and will not impact the radiological consequences of this event. A single downscale failure of a 2nd stage SJAE flow transmitter at present would lead to condenser isolation and possible loss of condenser vacuum. The proposed modification will eliminate this single failure vulnerability.

Evaluation Number: 2000-0049-R00

Document Evaluated: LDC 2000-015
ER 2000-0061-02-00

BRIEF DESCRIPTION OF CHANGE, TEST OR EXPERIMENT:

The selected carbon steel piping, and lateral/branch fitting, 1"-HBD-I325, shall be replaced with stainless steel piping and lateral/branch fitting 1"-HCD-665. The components being replaced are in the GGNS Erosion/Corrosion Program (MS-41) as item 551. This change will provide an improved erosion resistant material to resolve the existing piping segment erosion problem. The existing piping configuration will be maintained, and there is no adverse impact on existing pipe support loads or piping stresses due to this replacement.

REASON FOR CHANGE, TEST OR EXPERIMENT:

The GGNS Flow Accelerated Corrosion (FAC) inspection program is intended to identify, monitor, and replace different degraded components of the plant that are damaged by FAC. This ER is scheduled for the next refueling outage (RF11) based on wall thinning identified by the FAC program during inspections performed during pervious refueling outages. The 1"-HBD-1325 piping downstream of valve N1N33F177 to 18"-HBD-1132 (Stub Tube) is experiencing wall thinning. The 18" Stub Tube is attached to the H. P. Condenser at connection 143.

The purpose of this ER is to replace approximately 2 inches of carbon steel small bore pipe and fittings with stainless steel pipe and fittings to provide enhanced erosion resistance. The change includes the replacement of pipe, and the lateral/branch fitting to 18"-HBD-1132.

SAFETY EVALUATION SUMMARY AND CONCLUSIONS:

Neither the (N33) Seal Steam System, nor (N19) Condensate System have a safety-related function as defined in Section 3.2 of the FSAR. The system analysis has shown that failure of either system will not compromise any safety-related equipment or component, and will not prevent safe shutdown. The modifications made by ER 2000/0061-02-00 will in no way impact any of the accident analyses presented in the FSAR. Replacement of the carbon steel pipe and fittings with stainless steel pipe and fittings will aid in the prevention of significant levels of wear in these systems. The piping being replaced is in the GGNS Erosion/Corrosion Program (MS-41) as item 551. No new failure modes are being created thus no possibility of an accident or malfunction of a different type than previously analyzed is possible. Failure of the systems will not compromise any safety-related system or component, and will not prevent safe reactor shutdown. The margin of safety will not be reduced. The piping and fittings installed by this design change meet ANSI B31.1 code requirements and is supported for the appropriate design loads.

Evaluation Number: 2000-0050-R00

Document Evaluated: LDC 2000-015
ER-2000-0062-02-00

BRIEF DESCRIPTION OF CHANGE, TEST OR EXPERIMENT:

The selected carbon steel piping, and fittings, 4"-EBD-6, shall be replaced with stainless steel piping 4"-ECD-25. The piping being replaced is in the GGNS Erosion/Corrosion Program (MS-41) as item 277. This change will provide an improved erosion resistant material to resolve the existing piping segment erosion problem. The existing piping configuration will be maintained, and there is no adverse impact on pipe support loads or piping stresses due to this replacement.

REASON FOR CHANGE, TEST OR EXPERIMENT:

The GGNS Flow Accelerated Corrosion (FAC) inspection program is intended to identify, monitor, and replace different degraded components of the plant that are damaged by FAC. This ER is scheduled for the next refueling outage (RF11) based on wall thinning identified by the FAC program during inspections performed during pervious refueling outages. The 4"-EBD-6 piping downstream of restricting orifice N1B21-RO-D032 to the 4"-EBD-6 90° elbow marked spool piece 7S-7 is experiencing wall thinning.

The purpose of this ER is to replace approximately five feet of carbon steel pipe and fittings with stainless steel pipe and fittings to provide enhanced erosion resistance.

SAFETY EVALUATION SUMMARY AND CONCLUSIONS:

The portion of the (B21) Nuclear Boiler System affected by this modification has no safety-related function as defined in Section 3.2 of the FSAR. The modifications made by ER 2000/0062-02-00 will in no way impact any of the accident analyses presented in the FSAR. Replacement of the carbon steel pipe and fittings with stainless steel pipe and fittings will aid in the prevention of significant levels of wear in these systems. The stainless steel replacement piping and fittings will maintain the existing pipe routing. This modification is a material change only. The piping being replaced is in the GGNS Erosion/Corrosion Program (MS-41) as item 277. No new failure modes are being created thus no possibility of an accident or malfunction of a different type than previously analyzed is possible. Failure of these components will not prevent safe reactor shutdown. The margin of safety will not be reduced. The piping and fittings installed by this design change meet ANSI B31 .1 code requirements and is supported for the appropriate design loads.

Evaluation Number: 2000-0051-R00

Document Evaluated: LDC-2000-015
ER 2000-0094-00-00

BRIEF DESCRIPTION OF CHANGE, TEST OR EXPERIMENT:

ER 2000-0094-00-00 was initiated to effect the removal of Injection Water Control Fluid Relays 1N32D135A, B and C, Bypass Injection Water Pressure Interlock Relays 1N32D136A, B and C, and Hydraulic Timing Relays 1N32D138A, B and C. These devices are being removed from the Hydraulic Control Equipment Rack for Main Steam Bypass Control to eliminate the requirements for preventative maintenance on nonfunctional equipment. The configuration of the rack, located on the turbine front standard, will be modified only as required to maintain operability of system components that will remain. Most pipe sections will be capped or have blind flanges installed. The section of Bypass Startup Fluid piping through 1N32D1 36A, B and C will be hard piped to maintain the flow path to condenser low vacuum trip logic. No new equipment will be installed by this package.

REASON FOR CHANGE, TEST OR EXPERIMENT:

Original system design configuration was implemented to cause a turbine trip within 10 seconds if water injection was not established to the pressure breakdown assembly of the respective bypass valve. This injection water trip function for the bypass valves was deleted to minimize inadvertent turbine trip and the condensate lines associated with these devices were eliminated by MCP 93/1079 (Safety Evaluation 93-0133-R00). Accordingly, UFSAR Section 10.4.4 was revised to delete the water injection failure references. This rendered the turbine control system devices listed above nonfunctional. However, left installed, these devices increase the potential for control fluid leaks at their respective locations and create a preventative maintenance burden to ensure system integrity is maintained. Therefore, these components and associated others that are no longer required will be removed and portions of the respective piping abandoned in place.

SAFETY EVALUATION SUMMARY AND CONCLUSIONS:

The Turbine Generator Control Fluid System (N32) and the Condensate System (N19) serve no safety function. System analyses have shown that failure of these systems will not compromise any safety-related plant system or component and that there will be no impact on the ability to safely shutdown the reactor. UFSAR Section 3.2, which classifies plant systems' piping, does not address System N32 piping. System N19 piping is classified as 'other', indicating that loss of its function would not impact safe plant shutdown. This piping and the associated valves are classified as non-safety related, non-seismic, quality group D and ANSI B31 .1. Removal of the Injection Water Control Fluid Relays, the Bypass Injection Water Pressure Interlock Relays and the Hydraulic Timing Relays will reduce preventative maintenance requirements for the system and the potential for system control fluid leakage. UFSAR revision is only required for Figure 10.4-010 (P&ID M-1053A) to reflect closure of vent isolation valve 1N19F218 and the abandoned status of its associated piping. P&IDs for System N32 are not included in the UFSAR.

Evaluation Number: 2000-0052-R01 Document Evaluated: ER 2000-0892-00-01

BRIEF DESCRIPTION OF CHANGE, TEST OR EXPERIMENT:

ER GG-2000-0892-00-01 is providing on-line leak repair instructions for the 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 a ¼" lower than the locations where 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 Q1G33F040 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 adapters (Ref. MNCR 0302-92 Disposition). 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 for MNCR 0302-92.

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 RF11.

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 QIG33F040 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 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 adapters 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 RF 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 has shown 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 Q1G3F039 and Q1G33F040. This allows damming the sealant by utilizing the voids between the valve body and valve bonnet. The installation of the shutoff adapters 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 adapters 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-QI-I2-117, 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 Q1G33F039 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, Rev. 11.

Calculation NPE-G33F039/F040, Rev. 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 adapters 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 ER-GG-2000-0880-000-0 response instructions. These 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 adapters are similar to the nut bolt combinations discussed in UFSAR 3.5.1.2.2.i 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-2000-0892-000-1 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.

Evaluation Number: 2000-0053-R00

Document Evaluated: LDC-2000-060
ER-2000-0078-00-R02

BRIEF DESCRIPTION OF CHANGE, TEST OR EXPERIMENT:

ER 2000/0078-00-R02 will completely replace the existing C84 Meteorological Monitoring System with new measuring instruments, two new towers, new electronics, new data acquisition units, and backup power sources. This change will require changes to UFSAR section 2.3.3 and UFSAR Table 2.3-170. These changes will be made under LDC 2000-060 and evaluated with this 50.59 safety evaluation.

REASON FOR CHANGE, TEST OR EXPERIMENT:

The C84 Meteorological Monitoring System at GGNS, as described in the UFSAR section 2.3.3.2, is showing the effects of age. The existing system and its components/parts are becoming obsolete and are requiring additional maintenance to keep it operational. There have been ERs issued to upgrade obsolete components and to "abandon-in-place" the DEC computer which was not Y2K compliant. There are open CRs against the system that will require an abundance of engineering and maintenance cost to resolve. The new equipment is more reliable, more accurate, requires less power to operate, and will require minimal maintenance.

SAFETY EVALUATION SUMMARY AND CONCLUSIONS:

The C84 Meteorological System is a non-safety-related system. It is used to monitor meteorological conditions at the site. This information is submitted to the NRC in an annual report to satisfy the requirements of Regulatory Guide 1.23, Rev.0. The meteorological information is also used to determine the atmospheric stability class for controlled and uncontrolled radioactive releases.

This change will not lessen the reliability of the system to perform its intended function. The new instruments, power supplies, and data collecting/storing equipment is more accurate and more reliable than the existing equipment. The back-up battery power supply is capable of providing power to the C84 instruments and electronics for approximately 30 days therefore making it acceptable to remove the propane generator and battery bank of the existing system. The new permanent 50 meter tower is a non-safety related structure and will meet the requirements of the United Building Code (UBC) 1997 as stated in section 3.7 of the UFSAR.

Since this system is a monitoring system only and does not provide any type of automatic function/actuation of equipment, this change will not increase the probability of occurrence of an accident previously evaluated in the SAR. During accident conditions, this system continues to monitor, display, and record weather conditions near the plant. This data is used during an accident to determine/project what direction the radioactive plume is going and at what speed. This change to the Meteorological Monitoring System will install equipment that will provide the same monitoring capabilities as the existing system and is more reliable and accurate than the existing equipment. This change will not create any new release paths nor will it challenge any of the existing pressure boundaries. Therefore, this change will not increase the consequences of an accident previously evaluated in the SAR.

The system does not interface with equipment important to safety and does not provide any type of automatic actuation of any equipment except an alarm in the control room that notifies the operators if problems occur with the system. Therefore, this change will not increase the probability of occurrence or the consequences of a malfunction of equipment important to safety previously evaluated in the SAR. The new system will record and provide the same type data to the plant via the plant data system (PDS). Failure of this system can not affect the operating components or systems in the plant nor can it cause any type of radiological accident. Therefore, this change will not create the possibility of an accident of a different type than previously evaluated in the SAR.

The Meteorological Monitoring System for GGNS is completely isolated from the plant except for the two communication lines that transmit data to the plant and the equipment that translates the data in the plant. The new equipment that will be installed in the SC91P030 panel in the plant will interface with the PDS HUB only and will not interface with equipment important to safety. The power in the SC91P030 panel is BOP power. Therefore, a failure of the new Meteorological Monitoring System to be installed per this change will not create the possibility of a malfunction of equipment important to safety of a different type than any previously evaluated in the SAR.

This requirements for this system are in TRM section 6.3. The requirements in this section are for the surveillance intervals and the minimum required channels needed for the system to be operable. Failure to maintain the minimum required channels does not require any LCO to be initiated or entered. The new system will contain at least the minimum required channels as described in the TRM section 6.3. Therefore, this change will not reduce the margin of safety as defined in the basis for any Technical Specification.

Evaluation Number: 2000-0053-R01

Document Evaluated: LDC-2000-015, ER-2000-0078-00-R02 and SCN 2000/0012A

BRIEF DESCRIPTION OF CHANGE, TEST OR EXPERIMENT:

This revision 1 to the 50.59 evaluation was done to incorporate the three additional changes made to UFSAR mark-ups of LDC 2000-060. It should be noted that these changes do not affect the original evaluation conclusions. The changes made to the three UFSAR pages are as follows:

1) Page 2.3-36 and 37 was remarked-up to show that the data package sent from the MET shack electronics to PDS is done at an interval of less than or equal to (≤) ten seconds instead of at ten seconds.

2) Page 2.3.-37 was changed to note that the sigma theta values are transmitted at 15 minutes and hourly only. This is acceptable since sigma theta is an averaged value.

3) Page 2.3-39 was remarked-up to change the location of the rain gauge. The original markup had the location as being at the backup tower. The rain gauge is actually mounted near the Primary Tower.

ERCN#002 is also being added to this revision. The ERCN makes minor installation changes that do not affect the original 50.59 evaluation.

ER 2000/0078-00-R02 will completely replace the existing C84 Meteorological Monitoring System with new measuring instruments, two new towers, new electronics, new data acquisition units, and backup power sources. This change will require changes to UFSAR section 2.3.3 and UFSAR Table 2.3-170. These changes will be made under LDC 2000-060 and evaluated with this 50.59 safety evaluation.

REASON FOR CHANGE, TEST OR EXPERIMENT:

The C84 Meteorological Monitoring System at GGNS, as described in the UFSAR section 2.3.3.2, is showing the effects of age. The existing system and its components/parts are becoming obsolete and are requiring additional maintenance to keep it operational. There have been ERs issued to upgrade obsolete components and to "abandon-in-place" the DEC computer which was not Y2K compliant. There are open CRs against the system that will require an abundance of engineering and maintenance cost to resolve. The new equipment is more reliable, more accurate, requires less power to operate, and will require minimal maintenance.

SAFETY EVALUATION SUMMARY AND CONCLUSIONS:

The C84 Meteorological System is a non-safety-related system. It is used to monitor meteorological conditions at the site. This information is submitted to the NRC in an annual report to satisfy the requirements of Regulatory Guide 1.23, Rev.0. The meteorological information is also used to determine the atmospheric stability class for controlled and uncontrolled radioactive releases.

This change will not lessen the reliability of the system to perform its intended function. The new instruments, power supplies, and data collecting/storing equipment is more accurate and more reliable than the existing equipment. The back-up battery power supply is capable of providing power to the C84 instruments and electronics for approximately 30 days therefore making it acceptable to remove the propane generator and battery bank of the existing system. The new permanent 50 meter tower is a non-safety related structure and will meet the requirements of the United Building Code (UBO) 1997 as stated in section 3.7 of the UFSAR.

Since this system is a monitoring system only and does not provide any type of automatic function/actuation of equipment, this change will not increase the probability of occurrence of an accident previously evaluated in the SAR. During accident conditions, this system continues to monitor, display, and record weather conditions near the plant. This data is used during an accident to determine/project what direction the radioactive plume is going and at what speed. This change to the Meteorological Monitoring System will install equipment that will provide the same monitoring capabilities as the existing system and is more reliable and accurate than the existing equipment. This change will not create any new release paths nor will it challenge any of the existing pressure boundaries. Therefore, this change will not increase the consequences of an accident previously evaluated in the SAR.

The system does not interface with equipment important to safety and does not provide any type of automatic actuation of any equipment except an alarm in the control room that notifies the operators if problems occur with the system. Therefore, this change will not increase the probability of occurrence or the consequences of a malfunction of equipment important to safety previously evaluated in the SAR. The new system will record and provide the same type data to the plant via the plant data system (PDS). Failure of this system can not affect the operating components or systems in the plant nor can it cause any type of radiological accident. Therefore, this change will not create the possibility of an accident of a different type than previously evaluated in the SAR.

The Meteorological Monitoring System for GGNS is completely isolated from the plant except for the two communication lines that transmit data to the plant and the equipment that translates the data in the plant. The new equipment that will be installed in the SC91P030 panel in the plant will interface with the PDS HUB only and will not interface with equipment important to safety. The power in the SC91P030 panel is BOP power. Therefore, a failure of the new Meteorological Monitoring System to be installed per this change will not create the possibility of a malfunction of equipment important to safety of a different type than any previously evaluated in the SAR.

This requirements for this system are in TRM section 6.3. The requirements in this section are for the surveillance intervals and the minimum required channels needed for the system to be operable. The new system will contain at least the minimum required channels as described in the TRM section 6.3. Therefore, this change will not reduce the margin of safety as defined in the basis for any Technical Specification.

Evaluation Number: 2000-0054-R00

Document Evaluated: LDC-2000-037
ER 1997-0287-00-00

BRIEF DESCRIPTION OF CHANGE, TEST OR EXPERIMENT:

The following hardware modifications are proposed for the Reactor Core Isolation Cooling (RCIC) System Test Return to Condensate Storage Tank (CST) Isolation Motor Operated Valve (MOV) IE51F059:

a. Increase valve yoke strength by welding stiffener plates to the yoke.

b. Replace valve stem with one fabricated from a higher strength material.

c. Replace the welded packing gland leakoff nipple with a welded plug to facilitate future disassembly of the valve.

d. Increase output capability, under design basis degraded voltage conditions, of the Limitorque actuator by changing the overall actuator gear ratio to 67.5:1; this will also increase the operating time (stroke time) of the MOV up to 50 seconds. The allowable thrust on the actuator will be increased to 162% of the Actuator Thrust Rating by relying on actuator testing performed by Kalsi Engineering as permitted by Standard GGNS-MS-25.0 paragraph 5.11. This, along with the increased allowable thrust resulting from the strengthening of the valve yoke, will increase the maximum allowable stem thrust for the valve.

The following documents are associated with ER 97/0287-00-00 and are covered under this evaluation:

SCN 00/0008A against GGNS-MS-25.0, Rev. 0: provides documentation of new Limiting Components, Limiting Component Stress Allowable Thrust, Maximum Allowable Thrust, KALSI actuator thrust rating increase, new stem material and calculation references.

SCN 00/0007A against 9645-M-242.0, Rev. 57 changes the valve specification data sheet to indicate the new stem speed.

SCN 00/000lA against G.E. Specification 22A2134, Rev. 5: permits increasing the RCIC Test Return To OST MOV stroke time to greater than 30 seconds by increasing the RCIC injection time to 50 seconds when ROIC is in the functional flow test mode.

DECCN P106/SDC-E51 against System Design Criteria SDC-E51, Rev. 1: permits increasing the RCIC Test Return To CST MOV stroke time to greater than 30 seconds by increasing the RCIC injection time to 50 seconds when RCIC is in the functional flow test mode.

The following engineering calculations have been prepared or revised to provide the design bases for the proposed modification;

NPE-E51 F059, Revision 5, Supplement to the Powell Seismic Calculation D-69260

MC-Q1111-91123, Revision 23, Gate and Globe Motor Operated Valve Maximum Allowable Thrust

MC-Q1111-94015, Rev. 3, Degraded Voltage Torque Calculations for DC Motor Operated Valves

MC-Q1111-96002, Rev. 5, Calculation of Overall Actuator Ratio (OAR) for Generic Letter 89-10 Motor Operated Valves

MC-Q1E5100015, Rev. 0, Revised Stroke Times for Motor Operated Valves

REASON FOR CHANGE, TEST OR EXPERIMENT:

The RCIC Test Return to CST MOV is included in the scope of the GGNS MOV Program that was established in response to Generic Letter 89-10. GGNS has committed to maintaining an adequate operating margin for these valves. The operating margin is defined as the margin between required torque/thrust and the maximum available torque/thrust that can be generated by the actuator. Sufficient operating margin is necessary to account for possible future degradation in actuator/valve performance from such causes as wear, aging and lubricant degradation. The RCIC Test Return To OST MOV currently has an operating margin that is less than the 10% that is recommended by the NRC. The proposed Nuclear Change will increase the operating margin to greater than 10% by increasing the actuator output capability and strengthening the valve yoke and valve stem to withstand greater thrust forces. Currently, the packing gland leakoff nipple must be cut off and reattached for any work that requires valve disassembly. Since the leakoff connection is not used to divert/detect packing leakage (i.e., the leakoff nipple is capped), it will be modified by removing the capped nipple and welding a plug in the orifice. This will facilitate future disassembly of the valve.

SAFETY EVALUATION SUMMARY AND CONCLUSIONS:

The evaluations/calculations performed to support the modification of RCIC test return motor operated valve (MOV) 1E51F059 conclude that the function and operation of the valve are not changed and the reliability and operability of the valve are not adversely affected by the proposed change. Materials, parts and special processes are in accordance with the applicable codes and standards for the valve. Calculation NPE-E51F059 concludes that the yoke modification will increase the allowable stem thrust that can be applied to open and close the valve and that seismic qualification of the MOV will be maintained under the higher thrust load. Calculation MC-Q1111-91123 concludes that the allowable thrust for the actuator/valve assembly will be increased by the yoke modification and by invoking the KALSI allowable actuator thrust rating. Calculation MCQ1111-94015 concludes that the actuator output torque capability under design basis degraded voltage conditions will be increased by the change to the actuator gear ratio. The response to EAR PC-00-019 indicates that the stroke time increase resulting from the gear ratio change does not adversely impact the voltage performance of the RCIC test return MOV or the other Division I DC system components and does not affect the thermal overload size requirement for the valve. The response to EAR PC-00-018 indicates that there is no impact on piping stress or pipe supports resulting from the added weight of the yoke stiffener plates. The wall thickness of the valve bonnet area in the area of the gland leakoff connection will be maintained by the welded plug to be installed. Standard GGNS-MS-25.0, referenced in UFSAR 5.4.6.2.2.2, will be changed to provide the upgraded parameters for MOV performance that will result in an increase in the operating margin of the ROIC test return MOV.

Hardware modifications to the RCIC Test Return to CST MOV 1E51F059, specifically a change in the actuator gear ratio, will increase the stroke time of this valve under dynamic conditions. The design stroke time during system operations is estimated to be 42 seconds utilizing methodology presented in the final report of the "BWR Owners' Group DC Motor Performance Methodology —Predicting Capability and Stroke Time in DC Motor-Operated Valves" (NEDC-32958). The increased stroke time of the "operational" stroke time over the "design" stroke time is due to a combination of degraded voltage supplied to the actuator due to increased thermal resistance, and due to valve operation against fluid under design pressure and flow. For conservatism, an operational valve stroke time of 50 seconds will be utilized in order to bound the estimated 42 second operational stroke time derived utilizing the current BWR Owners' Group methodology as presented in NEDC-32958.

As discussed in UFSAR Sections 5.4.6.1.1, 5.4.6.3, 7.4.1.1.3.2, and 7.4.1.1.3.6, current RCIC startup time from receipt of an actuation signal to delivery of design flow is within 30 seconds. The increase in stroke time of the Test Return to CST MOV would result in an increase in RCIC response time to 50 seconds only in situations in which RCIC is initiated during full-flow functional testing. For normal conditions in which RCIC is in standby, the response time of 30 seconds is unaffected by this change.

The RCIC System is designed to assure that sufficient reactor water inventory is maintained in the reactor vessel to permit adequate core cooling should the reactor vessel be isolated and feedwater supply be unavailable. Under these conditions, the RCIC System and High Pressure Core Spray (HPCS) System perform similar functions. Several transient events (Anticipated Operational Occurrences and Infrequent Events) credit HPCS/RCIC for long term water level control. For these transients, RCIC is the preferred source of makeup coolant because of its relatively small capacity, which allows easier control of RPV water level. However, RCIC is not an Engineered Safety Feature (ESF) system and no credit is explicitly taken in safety analyses for RCIC system operation except as follows. For the Control Rod Drop Accident (CRDA) analysis, RCIC is used in conjunction with HPCS to maintain reactor water level. For this Design Basis Accident, RCIC is required to meet single failure criteria. ROIC is also explicitly credited in the Station Blackout (SBO) Analysis (UFSAR Appendix 8A) to maintain vessel level and core cooling over the four hour SBO coping duration.

RCIC Test Return to CST MOV 1E51F059 is used, in conjunction with Flow Control Valve (FCV) 1E51F022, to provide a full flow test return path from the RCIC pump to the CST to permit full flow functional testing of ROIC during normal plant operation. As discussed above, in the unlikely event of a RCIC actuation coincident with full flow functional testing and 1ES1F059 full open, the requested increase in F059 stroke time could result in an increase in RCIC startup time (from actuation to delivery of design flow to the reactor vessel) to as high as 50 seconds. This is some 20 seconds beyond the current 30 second time quoted in the UFSAR. Upon receipt of an actuation signal, interlocks will automatically close F059 and the valve will be partially closed after 30 seconds. Therefore, RCIC flow diverted from the reactor vessel to the test return line during the remaining 20 seconds required for F059 to stroke fully closed is considered negligible. The additional 20 second delay will not prevent the RCIC system from performing its required design function (long term core cooling and vessel inventory control) during a CRDA or SBO.

Since the required design functions and safety analysis criteria would continue to be met with the RCIC Test Return MOV hardware changes and subsequent increased response time of the RCIC System under test conditions, the hardware changes and increased stroke time of the RCIC test return MOV will not increase the probability of occurrence or consequences of previously evaluated accidents or equipment malfunctions or create the possibility for an accident or a malfunction of equipment important to safety of a different type than that previously analyzed. Therefore, this change does not involve an unreviewed safety question.

Evaluation Number: 2000-0055-R00 Document Evaluated: LDC 2000-075

BRIEF DESCRIPTION OF CHANGE, TEST OR EXPERIMENT:

UFSAR Appendix 6A Section 3B.8.1.2 (page 6A-9), "Drywell Bubble Pressure and Drag Loads Due to Pool Swell" and Section 3BL.1.c of Attachment L to Appendix 3B (page 6A-29), "Submerged Structure Loads Due to LOCA and SRV Actuations - LOCA Air Bubble Loads," are being revised to reflect that the design of the floor mounted portion of the ECCS/RCIC suction strainer Q1M24D001) has been analyzed utilizing the GESSAR II load definition (method of images) for LOCA air bubble loads.

REASON FOR CHANGE, TEST OR EXPERIMENT:

During a meeting with USNRC at Perry Nuclear Power Plant following installation of the ECCS/RCIC suction strainer, an issue regarding the basis/applicability of the coefficients used to calculate the acceleration drag volumes (hydrodynamic mass) for the perforated strainer sections was raised by the NRC. Although, it was generally accepted that the perforations would greatly reduce the drag volumes, it was felt that available empirical data was insufficient to provide accurate prediction of the reduced acceleration drag volumes. Since the strainer design for Perry and GGNS are similar, the issues raised by the NRC also applies to the GGNS ECCS/RCIC suction strainer.

Consequently, a test program was developed to perform strainer-model testing with the purpose of determining the strainer drag coefficients with specific applicability to the hydrodynamic load associated with Main Steam Safety Relief Valve (SRV) actuations and LOCA blowdown loadings. The test program utilized a 1/3 scale model of the ECCS/RCIC suction strainer and included various specimens of solid and perforated cylinders for validation of the test program.

Strainer drag coefficients determined at the conclusion of the tests indicated that the reductions in acceleration drag provided by the perforations are consistent with the original design. However, an unanticipated outcome of the test was the discovery of the boundary proximity effect on the strainer base plate as a result of its close proximity to the suppression pool floor.

In order to evaluate the effects of the LOCA air bubble loads on the ECCS/RCIC suction strainer, the GESSAR II load definition (method of images) was utilized. This method allows the application of empirical data extracted from testing in the dad analysis of the strainer. Subsequent strainer load calculations indicate that the strainer design is not adversely affected by the test results and the strainer will maintain its capability to perform its intended safety related design functions.

SAFETY EVALUATION SUMMARY AND CONCLUSIONS:

The change in methodology for determining the impact of LOCA air bubble loads on the ECCS/RCIC suction strainer does not require any physical change to the facility. The use of the GESSAR II load definition (method of images) for these loads allows the application of the test data in the load analysis to ensure that the ECCS/RCIC suction strainer is adequately designed. As stated above, the test results and analysis results indicate that the ECCS/RCIC suction strainer design, construction and installation meet all applicable regulatory requirements. Additionally, the new load determination does not adversely impact the previous conclusions related to the Humphrey Concerns evaluated in MAEC-87-0077 (NRC SER for resolution of Humphrey Concerns).

Therefore, the changes do not increase the probability of occurrence or consequences of an accident or malfunction of equipment important to safety previously evaluated in the SAR, and do not create a different type of accident or malfunction than previously evaluated in the SAR. The Technical Specifications and Technical Requirements Manual are not affected, and the margin of safety as defined in the basis for any Technical Specification remains unchanged. Therefore, these changes do not constitute an Unreviewed Safety Question.

Evaluation Number: 2000-0056-R00 Document Evaluated: LDC 2000-038

BRIEF DESCRIPTION OF CHANGE, TEST OR EXPERIMENT:

LDC No. 2000-038 updates UFSAR Sections 3.3.2.1; 3.5.1.4; 3.8.1.4.2; 3.8.1.5.2; 3.9.1.1.1.1; 3.9.1.1.1.2; 3.9.1.1.1.3; 3.9.1.1.1.9; 3.9.3.1.1.1.15; 3.9.3.1.1.1.16; 3.9.2.3.2; and Tables 3.8-6; 3.8-7; 3.8-14; 3.8-15; 3.8-21; 3.8-22; 3.8-23; 3.9-2c; 3.9-2e; 3.9-2j; 3.9-2k; 3.9-21; 3.9-2n; 3.9-20; 3.9-2q; 3.9-2r; 3.9-2x; 3.9-2z; 3.9-25b; 3.9-29c; 3.9-32a; 3.9-32b; 3.9-32c; 3.9-32d; 3.9-32e; 3.9-32f; 3.11-2; 3.11-3

See Sections "B" and "C" below for details of the changes and basis.

REASON FOR CHANGE, TEST OR EXPERIMENT:

LDC 2000-038 updates the UFSAR to correct miscellaneous typographical errors and other discrepant items discovered during the UFSAR Chapter 3 verification review which are documented in the various discrepancies included in Engineering Report GGNS-00-0005 and referenced in the LDC.

SAFETY EVALUATION SUMMARY AND CONCLUSIONS:

The changes to the UFSAR do not represent any physical changes to the plant, they correct typographical errors, add clarifications, correct omissions and correct conflicts between the UFSAR and current design documentation (calculations, stress reports, GE design reports and other documents, standards, etc.).

These changes do not degrade below the current design basis the performance of, and do not decrease the reliability of, any safety systems assumed to function in the accident analysis. These changes do not cause a greater reliance to be placed on any specific system, structure or component to perform a safety function. These changes do not adversely affect the overall performance or reliability of a safety system in a manner that could lead to an accident occurring. These changes do not cause a safety system to be operated outside of its design basis limits. Additionally, the margin of safety as defined in the bases for the Technical Specifications has not been reduced.

A number of the changes involve calculated values of stresses or loads on certain components listed in Tables in Section 3.8 and 3.9 of the UFSAR. In some cases the Code Allowables are revised to correct the information in the UFSAR to agree with the values specified in the calculations, seismic qualification packages and stress reports. Calculations, qualification packages and stress reports (both AE/Entergy and GE) which support the numbers presented in the tables have been revised for various reasons, including design changes to the system or component, new loads issues, general update to the calculations, snubber reduction, etc. All of the revised calculated stress numbers are within the Code allowables. No changes in calculation methodology are identified in the calculation revisions that support the changes.

Evaluation Number: 2000-0057-R00 Document Evaluated: ER 2000-0763-00-00

BRIEF DESCRIPTION OF CHANGE, TEST OR EXPERIMENT:

ER-GG-2000-0763 is installing a new oil cooler for the Division 3 (HPCS) Diesel Generator. Woodward provides an oil cooler for the governor as a standard option. The new cooler is to be cooled by engine jacket water. This design modification provides the configuration change details and evaluations necessary for this change.

REASON FOR CHANGE, TEST OR EXPERIMENT:

CR GGN-2000-1284 identified the concern that the Division 3 diesel engine governor oil temperature during operation was significantly greater than the maximum design temperature (200 degrees F) provided by the governor manufacturer (Woodward Governors). Woodward provides an oil cooler for the governor as a standard option. This design modification provides the configuration change details and evaluations necessary for this change.

Also, CR GGN-2000- 1431 noted minor discrepancies between the HPCS DG P&IDs and actual 'plant configuration. These discrepancies have been resolved by changes to the P&ID DCDs included as part of this package.

SAFETY EVALUATION SUMMARY AND CONCLUSIONS:

Installation of the new governor oil cooler is to be done in accordance with the existing requirements for the HPCS DG (Seismic, piping/tubing, etc.) and therefore this change is acceptable. Resolution of the P&ID discrepancies (CR GGN-2 000-1431) requires only drawing changes to reflect the actual plant configuration, there is no operational impact associated with these drawing changes.

Evaluation Number: 2000-0058-R00 Document Evaluated: T. A. 2000-009

BRIEF DESCRIPTION OF CHANGE, TEST OR EXPERIMENT:

This temporary alteration will remove the lower portion of the "A" offgas regenerative skid dryer/chiller loop seal piping between the 1N64F207A connection to the loopseal and the reducer to the four (4") piping and replace it with a clear tygon tubing secured by clamps at each end.

REASON FOR CHANGE, TEST OR EXPERIMENT:

To have capability to monitor and/or remove loop seal sediment and to establish a performance trend.

SAFETY EVALUATION SUMMARY AND CONCLUSIONS:

This temporary alteration will not affect operability, functionality or radiation monitoring of the offgas system but it will provide capability to monitor or remove restricting sediment and/or restriction of the "A" drain loop seal. Function and operability of this system nor any other (safety or non-safety related) system will be affected. The standard operating procedure of this system will not be affected by this temporary alteration. Failure of the system will not impact normal plant operation. The portion of the system used in this temporary alteration is not part of normal offgas flow and is lined up to be part of the regenerative cycle for an out of service bed (SAR Sec 15.7.1).

Evaluation Number: 2000-0059-R00

Document Evaluated: LDC 2000-068 (Standard MS-38)

BRIEF DESCRIPTION OF CHANGE, TEST OR EXPERIMENT:

Standard MS-38, "Quality Standard for Instrument Air System and Diesel Generator Starting Air," has been revised to reflect correct limits for dewpoint based upon the results of Calculation MC-Q1111-00026, Rev. 0, "Instrument Air, ADS and Diesel Starting Air Dew Point Requirements" which was developed to provide a basis for the requirements contained in MS-38. The revision to MS-38 results in a change to UFSAR section 9.3.1.2 to reflect the maximum allowable dewpoints. The ADS air supply downstream of the ADS booster compressors has been revised from +20^{0}F to +22^{0}F (less restrictive) and for the air supply to the balance of the components (non-ADS) served by the Instrument Air system has been revised from -19^{0}F to -23^{0}F (more restrictive) based on the results of Calculation MC-Q1111-00026. The acceptance criteria in MS-38 for maximum allowable dewpoint for the Diesel Generator starting air systems was also revised from +20^{0}F to -23^{0}F (more restrictive), also based on the results of Calculation MC-Q1111-00026. This change in acceptance criteria for the Diesel Generator starting air system did not result in a change to the UFSAR.

REASON FOR CHANGE, TEST OR EXPERIMENT:

CR-GGN-1 999-0905 was written to document that during DBD review of the Standby Diesel Generators and during the assessment of past CRs written identifying high dewpoints of Standby Diesel starting air, an error was identified in Mechanical Standard MS-38. This standard prescribes air quality standards for the Instrument Air system and Diesel Generator (DG) starting air system. Standard MS-38 requires that the relative humidity of air, in carbon steel compressed air piping, be maintained at a relative humidity of ≤2% in order to arrest oxidation. Additionally the standard considers the dewpoint necessary to prevent the formation of condensation inside the piping. The most restrictive dewpoint is utilized as the acceptance criteria. UFSAR Section 9.3.1.2 prescribes specific dewpoint limitations for Instrument Air and UFSAR Sections 9.5.6.2.1 and 9.5.6.2.2 (for Standby and HPCS DG starting air, respectively) discuss relative humidity of the compressed starting air being maintained low enough to prevent formation of corrosion in carbon steel piping. Standard MS-38 was found to contain non-conservative limits for Instrument Air and DG starting air dewpoint, whereas the limit for ADS air was slightly over-conservative. The impact of this discrepancy was that, even if compressed air was maintained within prescribed limits of Standard MS-38, existing Instrument Air and DG starting air piping and tanks could have been subjected to a relative humidity above 2% and, therefore, at risk of allowing formation of corrosion/rust in carbon steel components. As a result, calculation MC-Q1111-00026 was generated to determine the correct dewpoint criteria, and MS-38 is being revised to incorporate the results.

SAFETY EVALUATION SUMMARY AND CONCLUSIONS:

The Instrument Air system has no safety-related functions. However, many components that utilize instrument air are safety-related. Additionally, some non-safety related components that are supplied with instrument air are potential initiators of analyzed transients.

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The Diesel Generator (DG) starting air systems are safety related. They provide the motive power for starting the Division I, II and III DG engines. For Divisions I and II DGs, the starting air system also supplies the pneumatic control system that controls their starting and stopping.

The revision to standard MS-38 does not make any physical change to the plant. The changes are restricted to MS-38 acceptance criteria (as duplicated in the UFSAR) utilized in determining the satisfactory operation of the Instrument Air system and DG starting air system air dryers, i.e., acceptable dew point that will result in no condensation and no long term corrosion in carbon steel piping. The revised dewpoint acceptance criteria are consistent with the results of calculation MC-Q1111-00026, which determined the specific acceptance criteria necessary to fulfill the overall requirement for a relative humidity of ≤2% in the air systems. With the exception of ADS air, the changes are actually more restrictive. The change in ADS air criterion is minimal and, according to calculation MC-Q1111-00026, provides assurance that the overall requirement of ≤2% relative humidity is still satisfied. The existing Instrument Air Dryers, the new Plant Air Dryers being installed by ERs 98/0427-00 and 01, and the DG air dryers are capable of providing air that meets the revised dewpoint requirements.

The changes to Standard MS-38 and the UFSAR do not cause a greater reliance to be placed on any specific system, structure or component to perform a safety function. The changes do not degrade the performance or reliability of a safety system assumed to function in the accident analysis. The changes do not put the plant operation in an unanalyzed region. The changes do not adversely affect overall performance or reliability of any system in a manner that could lead to an accident occurring. Therefore, the revision to MS-38 and the change to the UFSAR do not increase the probability of occurrence or consequences of an accident or malfunction of equipment important to safety previously evaluated in the SAR, and do not create a different type of accident or malfunction than previously evaluated in the SAR. The Technical Specifications and Technical Requirements Manual are not affected, and the margin of safety as defined in the basis for any Technical Specification remains unchanged. Therefore, the changes do not constitute an Unreviewed Safety Question.

Evaluation Number: 2000-0060-R00

Document Evaluated: ER 1998-0427-00-00 and ER 1998-0427-01-00

BRIEF DESCRIPTION OF CHANGE, TEST OR EXPERIMENT:

ERs 1998/0427-00-00 (mechanical scope) and 1998/0427-01-00 (electrical scope) install two new heated desiccant type plant air dryers to replace the existing heat-less desiccant type instrument air dryers. A Bailey Infi-90 DCS Control Panel will be installed to control the new dryers. Output from the new Infi-90 OCS will be connected to the INFI-NET Communication Loop during RFI I to provide control room monitoring and facilitate system maintenance, alarm management and trouble shooting. The existing control room annunciator window for air dryer trouble' will be retained and its input will be from the new Plant Air Dryers. The associated computer point (which is redundant to the annunciator) is being eliminated. A new system, the Plant Air System (P51) is being created by these ERs and the new air dryers, Infi-90 control panel and associated components are designated as system P51 components. Associated LDC No. 2000-067 revises the following UFSAR sections, tables and figures to reflect the modifications: UFSAR Tables 1.7-1, 1.10-1; Sections 1.2.2, 9.3.1; Figures 8.3-7A, 9.3-1, 9.3-2F, 9.3-3-2 and 9.3-31 (new).

Associated SCNs included in this evaluation are: SCN 00/001 IA to JS-11, Rev. 1, SCN 00/0001A to GES-0I Rev. 4, SCN 00/0001A to M-203.0 Rev. 34, SCN 00/0002A to M-204.0 Rev. 0, SCN 00/0012A to MS-02 Rev. 49, SCN 00/0001A to MS-03 Rev. I and SCN 00/0001A to MS-38 Rev. 1.

Calculations issued in support of these changes are: EC-N1L11-95002 Rev. 2 Supplement 1, EC-Q111190028 Rev. 5 Supplement 1, 7.2.2 Rev. C, 7.2.10 Rev. B, NPE-PDS-2742 Rev. 0, NPE-PDS-2743 Rev. 0, C-S-353.0 Rev. 0 Supplement 12, 425A.4517 Rev 0, 425A.4518 Rev 0, 425A.4519 Rev 0, CC-N 1111-00003 Rev. 0, CC-N1111-00004 Rev. 0.

REASON FOR CHANGE, TEST OR EXPERIMENT:

The existing Instrument Air and Service Air system compressors and air dryers are unreliable due to obsolescence and parts problems. The Instrument Air System dryers are failure prone and require a significant amount of maintenance due to their age and design. The new dryers are more reliable, effective, and cost efficient than the existing dryers which are near the end of their useful life. The proposed changes will result in improved performance and increased reliability of the air dryers that serve the instrument air system.

SAFETY EVALUATION SUMMARY AND CONCLUSIONS:

The instrument air system has no safety-related functions. The system is equipped with primary containment, secondary containment and Drywell isolation valves that are not affected by these changes. Failure of the instrument air and plant air systems will not compromise any safety-related system or component and will not prevent safe reactor shutdown. As described in UFSAR Section 15.2.10, a loss of the instrument air system will result in reactor shutdown due to the opening of the control rod scram valves and/or the closing of the main steam line isolation valves. The loss of instrument air will not interfere with safe shutdown of the reactor since all equipment using instrument air is designed to fail to a position that is consistent with the safe shutdown of the plant. Air-operated equipment that must be available for use in the event of a failure of the instrument air system is equipped with a backup source (e.g., accumulators) to provide the required air supply. The consequences of a loss of instrument air do not result in any temperature or pressure transient in excess of the criteria for which the fuel, pressure vessel, or containment is designed. Existing failure analyses of the instrument air system (complete loss of instrument air) bound potential failures of the new components. The changes made by these ERs have been evaluated under the design change process to insure they comply with all design requirements. The changes do not cause a greater reliance to be placed on any specific system, structure or component to perform a safety function. The changes do not degrade the performance or reliability of a safety system assumed to function in the accident analysis. The changes do not put the plant operation in an unanalyzed region. Existing Technical Specification and TRM requirements bound the proposed changes. And, the changes do not adversely affect the overall performance or reliability of a safety system in a manner that could lead to an accident occurring.

Therefore, the changes do not increase the probability of occurrence or consequences of an accident or malfunction of equipment important to safety previously evaluated in the SAR, and do not create a different type of accident or malfunction than previously evaluated in the SAR. The Technical Specifications and Technical Requirements Manual are not affected, and the margin of safety as defined in the basis for any Technical Specification remains unchanged. Therefore, these changes do not constitute an Unreviewed Safety Question.

Evaluation Number: 2001-0001-R00

Document Evaluated: LDC-2000-015
ER 2000-0151-00-00

BRIEF DESCRIPTION OF CHANGE, TEST OR EXPERIMENT:

P47 Chemical Feed Booster Pump SP470004B has a bad motor. The manufacturer's replacement pump has 1/4" larger inlet/outlet connections that are easily adaptable to the existing piping. The function of these pumps is not to pump chemicals but to provide sufficient pressure for treated PSW water to be reintroduced into the PSW supply piping. This ER facilitates the use of the manufacturer's recommended replacement pump for either SP47C004A or B to sustain the chemical injection function for the PSW system.

REASON FOR CHANGE, TEST OR EXPERIMENT:

The replacement pump for SP47C004A or B is an equivalent pump except for the size of the inlet/outlet connections. The size of these NPT connections is documented in theC0DB and on the P&ID.

SAFETY EVALUATION SUMMARY AND CONCLUSIONS:

The new pump is an acceptable replacement for the older model so there is no change in the function of the PSW chemical feed subsystem and no adverse impact on any system or component important to safety. There is no unresolved safety question.

Evaluation Number: 2001-0002-R00 Document Evaluated: LDC 2000-078

BRIEF DESCRIPTION OF CHANGE, TEST OR EXPERIMENT:

Neutron transmission testing ("Blackness Testing") has been conducted at GGNS since August of 1988 in response to NRC Information Notice IN 87-43 and Generic Letter (96-04) regarding degradation of Boraflex material in high density fuel storage racks (Reference 1). Before the last Blackness Test Campaign, all spent fuel rack cells in which fuel could be stored were qualified to store fuel. The Blackness Test Campaign results indicated the Blackness Test area has increased Boraflex degradation and should not be fully loaded with fuel. Therefore, spent fuel storage racks will be divided into two regions, to be designated Region I or Region II. Region I racks are those areas which are below the Boraflex panel dose threshold for accelerated gapping and are bounded by the EPRI model for shrinkage. These locations are qualified to be fully loaded with all fuel types discussed in the criticality analysis (including the new SPC fuel design — Atrium 10, Reference 3). Region I racks in which the accumulated dose is predicted to exceed the panel dose threshold will not be allowed to store fuel until these locations can be reallocated to Region II racks. Region II (currently only the Blackness Test area but could be additional cells in the future) locations are those which are at or above the Boraflex dose threshold for accelerated gapping; no credit is taken for Boraflex panels in the criticality analysis in these locations. Region II locations shall be grouped in 4x4 arrays in a "6 of 16 blocked" configuration (see Attachment I). Certain cells in this array will be administratively prohibited as well as physically blocked from storing any fuel consistent with the requirement of NUREG-0800 Standard Review Plan section 9.1.2.

In addition, Blackness Testing will no longer be performed at GGNS. The existing Blackness Test data is sufficient to fully characterize Region I locations which are below the Boraflex panel dose threshold for accelerated gaping. Region II locations will no longer receive credit for Boraflex panels. Thus, continued Blackness testing provides no new useful information.

REASON FOR CHANGE, TEST OR EXPERIMENT:

The spent fuel pool monitoring program committed GGNS to perform Blackness Testing (Reference 1) in rack locations which bounded all other locations in terms of accumulated panel dose. As required by the Blackness Testing program, newly discharged fuel was temporarily placed in the test area so that other areas of the racks would be bounded (in panel dose) by the test area. If the test area could meet the assumptions in the criticality analysis, the remainder of the racks would also be within the bounds of the analysis. Blackness Testing results were used to determine the amount of Boraflex gap formation and shrinkage that had occurred, to determine or confirm that the data is consistent with the EPRI Boraflex shrinkage/gap models, and to confirm that the criticality analysis assumptions concerning Boraflex gapping remain bounding for the GGNS in-service racks. From the last Blackness Test Campaign, it was discovered that the Boraflex degradation exceeded the GGNS storage rack criticality analysis assumptions in the Blackness Test Area. Based upon the EPRI Boraflex models, gapping was expected to reach an equilibrium, or stable level where gap size, frequency, and distribution in the Boraflex panel did not change with additional panel dose. Reference 1 contains commitments to the NRC to continue Blackness Testing until equilibrium was reached. But, as a result of the most recent Blackness Test results, it was concluded (Reference 2) that the EPRI model for panel shrinkage/gapping is valid below a threshold dose of 2E10 Rads but that degradation can accelerate beyond this threshold. In addition, as determined in Reference 2, an equilibrium condition may not be obtainable; or, if an equilibrium condition is obtainable, it is at a high level of Boraflex panel gapping. Thus it is no longer possible to reasonably meet the prior commitment. Therefore, it is necessary to divide the spent fuel racks into two regions for criticality control. The previous Blackness test area in which the panel dose has increased past the threshold for accelerated gapping will now be considered Region II with fuel loading restrictions. Region I locations are below the panel dose threshold for accelerated Boraflex gapping and can be fully loaded with fuel qualified in the criticality analysis. Currently, all racks in the upper containment pool (UCP) and all in the spent fuel pool (SFP) (except the Blackness Test area) are Region I, but could be reallocated to Region II if panel dose predictions indicate it is necessary. As panel dose increases in Region I to the threshold for accelerated Boraflex gapping, these locations will be reallocated to Region II with fuel loading restrictions, therefore Region II could potentially include cells other than the Blackness Test Area in the future. The above measures, along with continued rack monitoring described below meet both criticality control requirements and the intent of the original Blackness Testing commitment.

SAFETY EVALUATION SUMMARY AND CONCLUSIONS:

The spent fuel pool monitoring program committed GGNS to perform Blackness Testing in a specific set of rack locations in which newly discharged fuel was placed after each refueling outage so that the test area had the highest Boraflex dose levels. Blackness Testing results were used to determine the amount of Boraflex degradation that had occurred, to determine or confirm that the data is consistent with the EPRI Boraflex shrinkage/gap models, and to confirm that the criticality analysis assumptions concerning Boraflex gapping remain bounding for the GGNS in-service racks. Seven test campaigns were conducted. From the last Blackness Test campaign, it was discovered that the Boraflex degradation exceeded the GGNS storage rack criticality analysis assumptions in the Blackness Test Area. Loading of fuel into this area was administratively prohibited under GL 91-18 provisions. To allow fuel loading in the current Blackness Test area (containing increased Boraflex degradation) and to satisfy the requirements of the current criticality analysis, the spent fuel storage racks shall be divided into two regions. Region I racks are areas which are below the Boraflex panel dose threshold for accelerated gapping and are bounded by the EPRI model for shrinkage. Region II (currently only the Blackness Test area, but could include other cells in the future) rack locations are those which are at or above the Boraflex dose threshold for accelerated gapping and no credit is taken for Boraflex panels in the criticality analysis in these locations. In addition, Blackness Testing will no longer be performed at GGNS. The existing Blackness Test data is sufficient to fully characterize Region I locations which are below the Boraflex panel dose threshold for accelerated gapping. As Region I locations increase in dose (calculated by RACKLIFE or an equivalent methodology) to the threshold dose where accelerated degradation is known to occur, these locations will be reallocated to Region II and applicable loading restrictions imposed. Appropriate procedural controls will be developed to implement this approach.

The proposed UFSAR change to divide the spent fuel racks into two regions and no longer conduct Blackness Testing at GGNS does not create any unresolved safety questions. Blackness Testing results indicate there is increased degradation in the test area and therefore no credit is taken for Boraflex in these racks in the criticality analysis. The Blackness Test results to date are sufficient to characterize those racks whose panel dose remains below the threshold for accelerated gapping and future rack monitoring will continue using spent fuel pool silica and fuel movement data to determine the B_4C loss and panel dose to the racks. Using panel dose and B_4C loss information, and projections from the RACKLIFE program or an equivalent methodology, the time to reallocate a Region I location to a Region II location can be predicted. Therefore Blackness Testing will no longer be required for the GGNS spent fuel racks. Since rack monitoring for Boraflex degradation will continue and criticality analysis assumptions verified, the elimination of Blackness Testing does not create the possibility for any accidents or malfunctions that have not been evaluated in the UFSAR and also does not increase the probability of an accident or malfunction currently analyzed in the SAR. The spent fuel racks will continue to be able to meet the design requirement to maintain keff less than 0.95 per TS 4.3.1. The restrictions placed on loading fuel into Region II (Reference 3) do not create the possibility of a fuel misload nor inadvertent criticality because the locations which are restricted will be physically blocked from storing fuel. An event analysis was conducted (Reference 4, GIN 2001-00108) to determine the probability that a error could result in any of the restricted locations being inadvertently loaded with a spent fuel bundle which would result in a condition outside the bounds of the existing criticality analysis. Note that this is the probability that a bundle is placed in a restricted location (a condition outside the existing criticality analysis) not the probability of occurrence of a criticality event itself. The analysis conclusions included that the minimum number of errors that would have to occur in order to inadvertently load a fuel bundle into a restricted location would be at least four and that the probability of this event is approximately 2.98E-06. Based on this evaluation which considers GGNS past experience with fuel movement errors, the use of blade guides or other blocking devices in spent fuel pool (and possibly upper containment pool in the future) restricted locations is acceptable and is considered to be non risk significant.

Evaluation Number: 2001-0003-R00

Document Evaluated: LDC-2000-015
ER 2000-0118-01-00

BRIEF DESCRIPTION OF CHANGE, TEST OR EXPERIMENT:

This ER will remove the disc position indication switch, position switch 1E12N113A, the position switch actuator rod and associated wires and conduit, from valve 1El2F04lA. It will also remove the indicating lights for the disc position indication of the valve from main control room panel 1H13P601-20C. The disc position indication switch is used for remote disc position indication during valve testing.

REASON FOR CHANGE, TEST OR EXPERIMENT:

Valve 1E12F041A has a history of the position switch actuator rod causing the valve disc to stick resulting in dual indication in the control room. This frequently requires valve disassembly to correct

SAFETY EVALUATION SUMMARY AND CONCLUSIONS:

No changes are being made that affect basic system design functions. The disc position indication switch is used for remote disc position indication during valve testing. The changes do not affect the ability of valve 1El2FO41A to perform its function under accident conditions. These changes do not result in a new pathway for the release of radioactive materials and do not affect offsite dose. No assumptions utilized in evaluating consequences of an accident will be altered. No new failure modes are created and there is no increase in previously identified failure modes for equipment important to safety. No assumptions utilized in evaluating the consequences of a malfunction of equipment important to safety will be altered by this change. This modification does not introduce any new failure modes and does not affect equipment other that the check valve and its associated disc position indication. Secondary and indirect effects (Fire protection, fire loading, pipe break, electrical shorts) have been reviewed and no increased probability of failure of equipment important to safety due to these concerns has been identified. These changes will not degrade any important to safety systems, components or structures nor will they degrade or prevent actions described in the SAR analysis. They do not increase the probability of occurrence or increase the consequences of malfunction of equipment important to safety of a different type than previously evaluated in the SAR. The Technical Specifications are not affected and the margin of safety is unchanged.

Evaluation Number: 2001-0004-R00

Document Evaluated: LDC-2000-015
ER 2000-0118-02-00

BRIEF DESCRIPTION OF CHANGE, TEST OR EXPERIMENT:

This ER will remove the disc position indication switch, position switch 1E12NI131B, the position switch actuator rod and associated wires and conduit, from valve 1EI2F04IB. It will also remove the indicating lights for the disc position indication of the valve from main control room panel 1H13P601-17C.

REASON FOR CHANGE, TEST OR EXPERIMENT:

Valve IE12F04IB has a history of the position switch actuator rod causing the valve disc to stick resulting in dual indication in the control room. This frequently requires valve disassembly to correct

SAFETY EVALUATION SUMMARY AND CONCLUSIONS:

No changes are being made that affect basic system design functions. The changes do not affect the ability of valve IE12F04IB to perform its function under accident conditions. These changes do not result in a new pathway for the release of radioactive materials and do not affect offsite dose. No assumptions utilized in evaluating consequences of an accident will be altered. No new failure modes are created and there is no increase in previously identified failure modes for equipment important to safety. No assumptions utilized in evaluating the consequences of a malfunction of equipment important to safety will be altered by this change. This modification does not introduce any new failure modes and does not affect equipment other that the check valve and its associated disc position indication. Secondary and indirect effects (Fire protection, fire loading, pipe break, electrical shorts) have been reviewed and no increased probability of failure of equipment important to safety due to these concerns has been identified. These changes will not degrade any important to safety systems, components or structures nor will they degrade or prevent actions described in the SAR analysis. They do not increase the probability of occurrence or increase the consequences of malfunction of equipment important to safety of a different type than previously evaluated in the SAR. The Technical Specifications are not affected and the margin of safety is unchanged.

Evaluation Number: 2001-0005-R00

Document Evaluated: LDC-2000-053
ER 2000-0052-00-00

BRIEF DESCRIPTION OF CHANGE, TEST OR EXPERIMENT:

This ER installs a new Division II (Loop B) Standby Service Water (SSW) pump. The new pump will be manufactured utilizing stainless steel components.

The associated LDC makes the following change:

UFSAR Table 3.9-25d is being revised to reflect the results of the seismic analysis for the new pump. Calculation C-C811, Rev. 1 and vendor document M-931 .0-Q1P41C00IB-7.0-1 -0 support this UFSAR change.

REASON FOR CHANGE, TEST OR EXPERIMENT:

The utilization of stainless steel components in the new pump will enhance reliability of the pump and minimize required maintenance compared to the existing Div II SSW pump which utilizes predominantly carbon steel components. The new pump will be manufactured such that it will have the same fit and function as the existing pump and will have the same performance characteristics as the existing pump.

SAFETY EVALUATION SUMMARY AND CONCLUSIONS:

The Div II SSW pump 1P41C00IB-B is discussed in various sections of the UFSAR including Table 3.9-25d (Summary of Results, Equipment: Standby Service Water Pumps) which contains the stress analysis and limits for the pump and UFSAR Table 3.2-1 which indicates that the SSW pumps are Safety Class 3, Quality Group C, constructed to ASME Section 111-3 and Seismic Category I. Installation of the new pump requires a change to the information contained in UFSAR Table 3.9-25d to reflect the seismic analysis of the new pump. Since the new pump is essentially a like-for-like replacement, except for materials of construction, there is no change to the operation of the SSW system and therefore, there is no other licensing basis impact. The Technical Specifications, Operating License, TRM and Fire Hazards Analysis do not require a change as a result of issuance of this ER for the installation of the new pump. The new pump was procured such that there will be no adverse impact on diesel generator loading, no additional heat rejected to the SSW system via pump motor cooler or pump work (i.e., will have the same performance characteristics and same power requirements as the existing pump). The existing motor will be used for the new pump.

During the evaluation performed for the replacement stainless steel pump, a nonconformance was discovered in Calculation 2.2.67-Q Rev. 0. The calculation was found to contain non-conservative values for the amount of interference that may occur between Standby Service Water Pump 1P41C00I B-B and the basin wall during a safe shutdown earthquake; this was documented in CR-GGN-2000-1858. Supplement 1 to Calculation 2.2.67 Rev. 0, has been developed to incorporate the specific structural information related to the new stainless steel pump and to use the correct displacement in determining the required clearances between the pump and the wall needed to prevent interference between the pump and the basin wall. As a result of the new displacements calculated for SSW Pump 1P41C00IB-B, the recesses constructed in the basin walls to accommodate the movement of the original "B" SSW Pump during a safe shutdown earthquake will be enlarged. The enlargement of the recesses were evaluated by the subject ER and found to have no adverse impact on the structural integrity of the "B" SSW basin.

An electronic search was performed of the GGNS UFSAR, Fire Hazards Analysis, SER (including supplements), Licensing Commitments, Technical Specifications, TRM and Operating License. Keywords (and their variations) used for the search were: SSW pump, standby service water pump, basin, and pump AND material. The following relevant UFSAR Sections were identified and reviewed for impact: UFSAR Section 1.2.2.8.1, Table 1.3-3, Table 3.2-1, Section 3.9.3.2.2.1.1, Table 3.9-3b, Table 3.9-25d, Table 8.3-2, Section 9.2.1, Table 9.2-3, Table 9.2-4, Table 9.2-16, Table 9.2-17, Figures 9.2-1, 9.2-2, 9.2-3, and 9.2-4. Additionally, the following licensing commitments and documents were identified and reviewed for impact: commitments A-6306, A-6309, A-7083, A-7573, A-7585, A-7846, A-7961, A-9236, Operating License Conditions C.2.40; and Technical Specifications and Bases 3.7.1

The proposed change is within the existing licensing basis of the Grand Gulf Nuclear Station. This Safety Evaluation documents the fact that the proposed change does not result in an Unreviewed Safety Question for the following reasons: the change does not cause a greater reliance to be placed on any specific system, structure or component to perform a safety function; the change does not degrade the performance of a safety system assumed to function in the accident analysis and does not decrease the reliability of safety systems assumed to function in the accident analysis; the change does not put plant operation in an unanalyzed region; the change herein is bounded by the analysis in the Technical Specifications, the TRM and the SAR; and the change does not adversely affect the overall performance or reliability of a safety system in a manner that could lead to an accident occurring.

Because the change described above will meet or exceed the requirements of the original design (component integrity, capacity, functionality, etc.) and existing analyses, the change will not degrade the performance of any safety systems, components, or structures nor will it degrade or prevent actions described in the SAR accident analysis. The change does not increase the probability of occurrence or increase the consequences of an accident or malfunction of equipment important to safety previously evaluated in the SAR and does not create a different type of accident or malfunction than previously evaluated in the SAR. The Technical Specifications and the Technical Requirements Manual are not affected, and the margin of safety as defined in the basis for any Technical Specification remains unchanged. Therefore, this change does not constitute an unreviewed safety question.

Evaluation Number: 2001-0006-R00

Document Evaluated: Design Change Standard DCS-11

BRIEF DESCRIPTION OF CHANGE, TEST OR EXPERIMENT:

The proposed change will allow de-energizing liquid level detectors' (GESTRA Probes) alarm function from the control room when the probes fail. Also all GESTRA Probes alarm function can be restored once the probes are repaired. The change will require to open the drain valves once every twelve hour period whenever alarm function of the failed probe is deleted. The extent of the design change standard is to allow de-energize or re-energize the annunciators that are associated with the Main and Reheat Steam lines and Seal Steam Lines GESTRA probes. No other systems or functions are affected by these electrical modifications.

REASON FOR CHANGE, TEST OR EXPERIMENT:

The GESTRA probes located in the Main and Reheat Steam lines and Seal Steam Lines have failed during the operating cycle. This causes operations personnel to react and correct spurious alarms and it also requires Temporary Alterations to be processed to defeat the sealed in alarms.

SAFETY EVALUATION SUMMARY AND CONCLUSIONS:

Design Change Standard, DCS-11 will be used in conjunction with a Condition Report written against the failed level probe annunciator. DCS-11 provides a method for removing a probe's annunciator from service until the probe can be repaired or replaced. The GESTRA probe repair is prioritized according to system pressure. The level probes associated with the high pressure steam lines should be repaired first. The level probes associated with low steam supply pressure are lower priority than the high pressure level probes. The function of the GESTRA probe is to provide control room indication whenever drain line level is high and provide information to the operator to open the associated drain line valves. From the review of the plant data over a one year period it was concluded that a twelve hour period for opening the valves was adequate to ensure proper draining of the condensate. De-energizing the GESTRA level probe alarm function will not place the plant in an unsafe condition. The change requires that the drain valve associated with the failed level probe be opened once every twelve hour period. Additionally, the drain valve associated with the failed level probe is to be opened prior to planned power changes. At which time the drain valve may be left open or cycled until a steady power condition is reached. The GESTRA level probes are non-safety related equipment. The Main Steam in the Turbine Building and Seal Steam systems are non-safety related. These systems are not part of the reactor coolant pressure boundary nor are they required for safe shutdown of the plant. The change will not alter the design, function or operation of any equipment important to safety as evaluated in the UFSAR. The extent of electrical modifications is to remove from service the annunciators that are associated with the N11 and N33 GESTRA probes. Upon repairing the probes, the annunciators can be restored to service. No other Systems or functions are affected by these electrical modifications. The appropriate drawings have been identified that will be revised to indicate the annunciators that have been removed from service, as appropriate. Upon repair of the probes the annunciators can be returned to service.

Evaluation Number: 2001-0007-R00

Document Evaluated: LDC 2001-007 (Report 1997-0023)

BRIEF DESCRIPTION OF CHANGE, TEST OR EXPERIMENT:

The UFSAR consistency review for the Control Rod Drive (CRD) hydraulic system identified several discrepancies with design documents. The proposed change will revise UFSAR applicable sections to resolve these discrepancies. The changes include correcting the number of transient cycles for reactor startup/shutdown, changing housing material grade, revising type of alarm for charging header pressure, deleting references to subsection 7.6.1.6, correcting figure numbers for layout of CRD hydraulic system, correcting number of stabilizer valves and correcting rod block function of scram discharge volume high water level scram trip bypass mode switch position.

REASON FOR CHANGE, TEST OR EXPERIMENT:

These discrepancies were due to previous UFSAR changes, incorrect figure reference number and incorrect design information. All changes have been reviewed with design documents (GE Specifications, SDC and drawings) and are software related only. These changes will not affect equipment function or performance of the CRD hydraulic system.

SAFETY EVALUATION SUMMARY AND CONCLUSIONS:

The UFSAR consistency review for the CRD hydraulic system identified several discrepancies with design documents (GE Specifications, SDC and drawings). The proposed change will revise UFSAR applicable sections to resolve these editorial type (correcting inconsistencies within UFSAR sections) discrepancies and correct information without changing intent or scope of the CRD hydraulic system. These discrepancies were due to previous UFSAR changes, incorrect figure reference number and incorrect information. All changes have been reviewed with design documents (GE Specifications, SDC and drawings) and are software related only. These changes will not affect equipment function or performance of the CRD hydraulic system. The change will not alter the design, function or operation of any equipment important to safety as evaluated in the UFSAR.

Evaluation Number: 2001-0008-R00 Document Evaluated: LDC 2001-004

BRIEF DESCRIPTION OF CHANGE, TEST OR EXPERIMENT:

Change the title of the Manager, Training and Emergency Preparedness to Manager, Training and Development. Also shift the reporting responsibilities of both the Manager, Training and Development and the Manager, Emergency Preparedness. The Manager, Training and Development now directly reports to the Director, Training and Development and is matrixed to the Site Vice President. The Manager Emergency Preparedness now reports to the Director, Nuclear Safety Assurance.

REASON FOR CHANGE, TEST OR EXPERIMENT:

Job title changes and reporting responsibility changes.

SAFETY EVALUATION SUMMARY AND CONCLUSIONS:

There are no unanswered safety questions. This is only a change to the UFSAR, TRM, and Emergency Plan to change the titles and the reporting responsibilities. The E Plan change is being controlled by LDC # 1999-0058. The TRM change is being controlled by LDC 2001-010.

Evaluation Number: 2001-0009-R00

Document Evaluated: LDC-2000-073, ER 97-0022-03-00

BRIEF DESCRIPTION OF CHANGE, TEST OR EXPERIMENT:

In order to resolve the over-pressurization issue of Generic Letter 96-06, the following modifications or actions are necessary for some of the penetrations evaluated for these conditions. The penetrations listed below were identified in Engineering Report GGNS-97-0002 Rev 1, CR GGN-1999-1147 and CR GGN-1999-1256.

Pen. No.	Line No. to be Protected from Over-Pressurization	Add Bypass Line for Valve No.	Reduce Maximum Closing Thrust for Valve No.	Add Relief Valve On Line No.	Mod. Location	Revise Operations Procedure
18 & 311♦ (GuardPipe)	6"-DBA-30 6"-DBA-32					✓
87 & 325* (Guard Pipe)	6"-DBA-9 & 6"-DBA-90	Q1G33F001-B, Powell, 600#, MOV Gate	Q1G33F001		DWL	✓

Note ♦: Penetrations 18 and 311 were initially planned to be worked in tandem with another modification (ER 2000-0083-00-02) consisting of removing valve 1E51F066 and the downstream piping, and then draining the penetration piping. Since the modification was cancelled for RF11, operations procedures will be modified to ensure the penetration piping is drained and no fluid is trapped between the isolation valves.

Note *: It should be noted that this modification will also help relieve the piping downstream of the inboard isolation valve G33F252 for Drywell penetration 366, thereby protecting the isolation valve from damage due to over-pressurization. Penetration 366 was previously addressed in ER 97-0022-02 with this anticipated change.

This ER provides instructions for the following:

Installation of a bypass line and reduction of the closing thrust for valve Q1G33F001. The bypass line, in conjunction with the existing relief path to the Reactor Vessel, will provide relief for Containment and Drywell piping penetrations 87 & 325. The reduction in the valve closing thrust will allow gate valve disc flexing in order to initiate bypass flow. This change will also require a revision to Operations procedures to add operational limitations.

Revision to appropriate Operations procedures to ensure there is no fluid trapped in the piping between the isolation valves for penetrations 18 & 311.

Enclosure 1 to this 10CFR50.59 evaluation provides detailed sketches for the various configurations related to this modification.

SCN 00/0014A for MS-02, Rev 49, was generated to add a reference note for GL 96-06 predicted pressures, and also to include the new line number for the bypass line.

SCN 00/0011A for M-242.0, Rev 57, was generated to include the GL 96-06 pressures for valves G33F001 (pen. 87 & 325) & G33F004 (pen. 87 & 325). The GL 96-06 pressure for connecting valve G33F252 (pen. 366) has been included in M-242.0 via package ER 97-0022-02-00.

SCN 00/0011A for MS-25, Rev 12, was generated to revise closing thrust limits for valve G33F001 and also to include limits for the minimum opening speed of valves E12F023 and B21F016. It should be noted that B21F016 was previously addressed in ER 97-0022-02.

Licensing Document Change 2000-073 was generated to revise TS bases section B3.6.1.3, UFSAR section 5.4.8.2 and TRM & UFSAR sections/tables TR3.6.1.3-1 to add a note for valve G33F001 indicating that valve G33F001 must not be open when Reactor pressure exceeds 500 psig. Also the LDC will add a minimum required opening time for valve E12F023 to ensure no water hammer occurs in the piping should the valve be opened inadvertently.

Implementation of ER 97-0022-03 and ER 97-0022-02, and closure of the licensing commitment A-34648 and A-33119 will complete the resolution of GL 96-06.

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.

Engineering Report GGNS-97-0002 initially identified 18 Grand Gulf penetrations, 12 Containment (36, 39, 43, 47, 49, 50, 51, 54, 58, 81, 84 & 86) and 6 Drywell (330, 331, 333, 348, 349 & 364), susceptible to increased pressures per GL 96-06. All but four penetrations (43, 54, 86 & 330) were addressed in ER's 97-0022-00-01 and 97-0022-01-01. Additional reviews of the GL 96-06 issue resulted in the identification of 16 additional susceptible penetrations via CR GGN-1999-1147 (38, 56, 87, 88, 325, 366 and 465) and CR GGN-1999-1256 (18, 19, 44, 45, 90, 91, 311, 312, 329). Penetration 38 has been already addressed in ER's 97-0022-00-01 and 97-0022-01-01 as it was worked in tandem with penetration 39. ER 97-0022-02 addressed 15 of the 19 remaining penetrations (19, 43, 44, 45, 54, 56, 86, 88, 90, 91, 312, 329, 330, 366 & 465). Penetrations 18 and 311 were initially planned to be addressed in a separate modification package ER 2000-0083-00-02. However ER 2000-0083-00-02 was removed from RF11 scope. Therefore, ER 97-0022-03 will address the last four penetrations 18, 87, 311 and 325. ER 97-0022-03 adds a bypass line around an isolation valve and requires the implementation of procedural changes for the affected penetrations.

Note: Throughout this Safety Evaluation and ER, the terms "inboard" and "outboard" refer to location relative to the Reactor Pressure Vessel with "inboard" being closer to the RPV than "outboard". This applies to valves as well as valve discs.

SAFETY EVALUATION SUMMARY AND CONCLUSIONS:

The above changes meet 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 penetrations addressed in ER 97-0022-003 for the following reasons:

The ½" safety related ASME class 2 bypass line will be installed from a hole at a location between the two gate valve discs for Drywell isolation valve Q1G33F001 and will discharge on the upstream side of the valve outside the penetration boundary. It is demonstrated via testing as well as engineering analysis (per Engineering Report GGNS-00-0009 Rev 0) that the disc will experience significant bypass leakage at a differential fluid pressure across the disc above 800 psid, provided the disc thrust does not exceed 23,335 lbf (total stem thrust limit will be higher). The modified valve will meet the requirements to allow bypass leakage once the stem closing thrust is lowered. The bypass leakage will occur from the pressurized side past the first disc into the bonnet cavity and then through the drilled hole in the valve body and the bypass line. The pressure build-up between the penetration isolation valves will then be relieved into the portions of the piping that are outside the penetration boundary. Positive relief will be assured since the upstream side of the valve is open to the Reactor. This configuration will provide pressure relief for Containment and Drywell penetrations 87 & 325 piping located between the isolation valves without affecting the Containment or Drywell leakage limits or requirements. These leakage limits are not affected since only one disc is bypassed in Drywell isolation valve G33F001. The bypassed disc on the isolation valve (inboard disc on Drywell inboard isolation valve) is not considered part of the Drywell isolation boundary.

Only the outboard disc of the inboard isolation valve is considered as a pressure-retaining disc and is effective as a Drywell isolation boundary.

It should be noted that the MEDP for valve G33F001 would not exceed 200 psig when closing the valve since SOI 04-1-01-G33-1 (reference 54) requires closing the valve to switch to the post-pump mode of operation. Therefore the maximum closing thrust for valve G33F001 can be reduced to a value below 23,335 lbf to ensure that disc flexing and bypass leakage occur. The opening thrust must however be maintained at its current value. It should be cautioned that the reduction in the closing thrust may render the valve, under certain improbable HELB conditions, incapable of closing if system pressure were above 500 psig (reference 60). Therefore, SOI 04-1-01-G33-1, step 4.1.2 c(2) will be revised to emphasize that valve G33F001 must not be opened when the system pressure exceeds 500 psig. IOI 03-1-01-1 (reference 64), step 6.2.10 will also emphasize that the valve G33F001 shall be closed prior to Reactor pressure reaching 500 psig. A similar note will be added in the IOI 03-1-01-06 (reference 65) for performing the reactor vessel hydro in case the valve is needed for isolation function during a reactor vessel hydro. As an additional precautionary measure, the IOI will be revised to require peer verification for the closure of valve G33F001 upon aligning the RWCU system to the post-pump mode. Finally Surveillance Procedure 06-OP-1G33-C-0002 (reference 66) will include a caution not to open valve G33F001 at system pressure above 500 psig. A note has been added to the TRM and UFSAR tables for valve G33F001 to indicate that the valve is only operable with the valve closed when reactor vessel pressure is greater than 500 psig. Finally, reducing the closing thrust is not expected to adversely impact the valve Local Leak Rate Test as this valve has historically had good and successful LLRTs.

It should be noted that this does not represent a substantial operational limitation as valve G33F001 will be capable of stroking close at all Reactor pressures under normal operating or system recovery due to pump trip conditions. It is only in the unlikely event of a Reactor blowdown due to a HELB that the valve, if opened, could be incapable to fully close under the resultant differential pressure. The valve will be considered INOP if open at pressures above 500 psig and should be re-closed to restore operability. In the unlikely case where valve G33F001 is mispositioned (i.e. open position) at system pressure above 500 psig, Operations will detect the problem in a timely manner within minutes as RWCU Demineralizers temperature high alarms 1G33-TAH-L609 and 1G33-TAH-L611 will activate. ARI 04-1-02-1H13-P680-11A-C6 "RWCU Filter Demin. Inlet Temperature Hi 140 F" (reference 67) and ARI 04-1-02-1H13-P680-11A-D6 "RWCU Filter Demin. Inlet Temperature Hi 130 F" (reference 68) will include an additional possible cause for the alarms as G33F001 being open. The valve "out-of –position" condition will be corrected within a short period of time during which the probability of a HELB in conjunction with a failure of valve G33F004 to close is practically nil.

This modification also helps relieve the piping downstream of the inboard isolation valve G33F252 for Drywell penetration 366, thereby protecting the isolation valve from damage due to over-pressurization. The piping located between the isolation valves for penetration 366 as well the piping upstream of the outboard isolation valve G33F253 were previously addressed in ER 97-0022-02 with this anticipated change. The modification to be implemented for valve G33F001 will therefore complete the resolution of GL 96-06 over-pressurization concerns for penetration 366.

No hardware modification is required for penetrations 18 and 311. However implementation of Operations procedural controls is necessary to ensure there is no fluid trapped in the piping between the isolation valves. Draining the line will be achieved via manual valves E12F344, E12F345 & E12F397 to the Clean Radwaste (CRW) Drain. The Operations Department has agreed upon the proposed solution.

The Firewater connection is currently included in the EPs as an alternate injection path for reactor vessel makeup via the outboard isolation valve E12F023 to the Feedwater system. This path could not be removed from the EPs as GGNS calculations require the availability of this Fire Water supply. Therefore it is speculated that water could travel back into the penetration piping during a post accident condition when Fire Water may be injected in the reactor. A similar condition could occur if valve E12F023 (LLRT air tested) leaks by during normal conditions. Consequently, water may be trapped between the inboard isolation valve E12F394 and the outboard check valve E12F019. However, since there is no vent path it is not considered credible that the penetration could be filled solid with water. At least a small air bubble will be present and will be adequate to prevent over-pressure. The air bubble cannot travel back upstream of check valve E12F019 as valve E12F394 is located at a higher elevation (133') than the check valve (elevation 129'-6"). Also the Emergency Procedures and the alternate reactor injection mode using Fire Water are considered outside the scope of the plant Design Bases, and need not be postulated concurrently with the GL96-06 over-pressurization effects. The criteria and rigor normally used for design are not applicable when Emergency Procedures are active and alternate Fire Protection injection is initiated.

The reactor head spray line is not used at GGNS and isolation valve opening is not expected to occur at normal RHR pressure. The only time valve E12F023 may be stroked is during testing or the EP mode when the RHR line pressure is below the Fire Water injection pressure of 150 psig or less. Therefore water hammer in the line cannot occur. In the unlikely event where valve E12F023 is opened inadvertently, water hammer is still not expected to occur in the line. The condensed water upstream of the valve will be below 150 °F and the valve opening speed is slow and will take a minimum of 61 seconds (maximum is 94 seconds per TRM table TR3.6.1.3-1, Page 3.6-17-II). Therefore, no flashing into steam will occur and no significant loading due to unbalanced fluid transient forces will be present.

All the above modifications and changes will assure piping systems and Containment integrity under over-pressurization conditions post LOCA.

Evaluation Number: 2001-0010-R00

Document Evaluated: LDC-2000-015
ER 2000-0887-00-00

BRIEF DESCRIPTION OF CHANGE, TEST OR EXPERIMENT:

ER 2000/0887-00-00 is being issued to install vendor recommended check valves with the required piping and pipe supports on Makeup Water Treatment System (P21) piping. Installation of these check valves is required to prevent contamination of the desiccant used in the Dri-Breathers located on Acid Storage tanks SP21AOO3A, B, & C. ER 2000/0887-00-00 will also serve as a response to Corrective Action 3 to CR-GGN-2000-0940.

This modification will bring the field installation into compliance with the manufacture's instructions for installation as described in vendor manual 460002919. The Dri-Breather absorbent has been contaminated on several occasions because of flow reversal experienced during inventory replenishment. After implementation of this ER Temporary Alteration 2000-0007 may be removed.

REASON FOR CHANGE, TEST OR EXPERIMENT:

The Dri-Breathers attached to Acid Storage Tanks SP21A003A, B, and C are improperly configured in that the manufacture's instructions for installation as delineated in vendor manual 460002919 does not agree with the actual field installation. CR-GGN-2000-0940 was written to document this condition. The vendor of the Dri-Breathers (The Kemp Company) provided instructions for a check valve arrangement that would not only prevent moisture from entering the storage tanks upon draw-down of inventory, but would also prevent contamination of the Dri-Breather absorbent by corrosive vapors upon flow reversal during inventory replenishment. A phone conversation with the Kemp Company, revealed that the absence of the check valve arrangement to prevent corrosive vapors (from the sulfuric acid inventory) from entering the Dri-Breather would also account for the deterioration of the Dri-Breather nozzle screens (documented on CR-GGN-2000-0746).

SAFETY EVALUATION SUMMARY AND CONCLUSIONS:

ER 2000/0887-00-00 is being issued to install the vendor recommended check valves with the required pipe and pipe supports on the Makeup Water Treatment System (P21) piping. This will bring the field installation into compliance with the manufacturer's instructions for installation as described in vendor manual 460002919. FSAR Figure 09.2-011 (P&ID M-0033G for the Makeup Water Treatment System) will be revised to show this arrangement. The installation of these check valves will have no adverse affect on the Make-up Water Treatment System. Installation of these valves will not alter present system operations as described in the FSAR.

USFAR section 9.2.3.3 states the Makeup Water Treatment System serves no safety-related function as defined in Section 3.2 of the UFSAR. Installation of these check valves will not compromise any safety-related systems or components, and will not prevent safe reactor shutdown. The existing margin of safety will not be reduced. No new failure modes are being created thus no possibility of an accident or malfunction of a different type than previously analyzed is possible. Installation of these check valves will help ensure successful operation of the affected system.

Evaluation Number: 2001-0011-R00 Document Evaluated: ER 2000-0847-00-00

BRIEF DESCRIPTION OF CHANGE, TEST OR EXPERIMENT:

Carbon steel piping 1" DBD-94 (approximately nineteen feet) will be replaced with Chrome-moly piping 1" DAD-5 from downstream of valve NIN33F300C up to and including the one-inch branch fitting at the 18"-HCD-667 stub tube (previously identified as 18" HBD-1132), at connection 140 of H.P. Condenser Shell N1N19-B007A. Also to minimize the pressurized portion of the abandoned Auxiliary Steam piping, this ER will install a set of flanges with a permanent blank in 6"-HBD-318 at the interface of the Auxiliary Steam piping and the Seal Steam Generator piping.

REASON FOR CHANGE, TEST OR EXPERIMENT:

CR-GGN-1999-1927 and CR-GGN-2000-1366 identified steam leaks through welds in the piping downstream of valve N1N33F3000. The subject piping is on the bypass stop and control valve drain line and is subject to a two-phase flow and flashing into steam phenomenon. The Auxiliary Steam System was highly susceptible to FAC. Most system degradations are caused by two-phase flow as a result of past usage. FAC System Susceptibility Analysis has identified the Auxiliary Steam system as non-susceptible based on the usage since it is permanently out of service. In order to minimize risk of possible new failures and reduce number of unnecessary FAC inspections a set of flanges with a permanent blank will be installed in 6"-HBD-318 to isolate the Seal Steam Generator piping from the Auxiliary Steam System.

SAFETY EVALUATION SUMMARY AND CONCLUSIONS:

The Condensate System (N19) nor the Main and RFP Turbine Seal Steam and Drain System (N33) system serves a safety function. Systems analysis has shown that failure of the Condensate System (N19) or the Main and RFP Turbine Seal Steam and Drain System (N33) system will not compromise any safety-related systems or prevent safe shutdown. There are no new systems added by the proposed change, thus the existing accident scenarios and analyses presented in the UFSAR will not be impacted by the proposed change. The proposed change will affect UFSAR Figure Numbers 10.4-003 and 10.4-012. Installation of the chrome moly piping and the permanent blank will not result in the operation of any plant system or component in a manner that is inconsistent with information contained in the UFSAR. The proposed change is located in the Turbine building and will not affect or impact the plant's radiological effluents. The proposed change to the N19 and N33 systems will have no adverse environmental impact. After reviewing the proposed change, it has been concluded that installation of the chrome moly piping and the permanent blank does not represent an Unreviewed Safety Question and will have no adverse affects on the environment. The requirements specified in the Technical Specifications are not impacted by the implementation of this ER. Thus the proposed change will not result in the need to change or revise the GGNS Technical Specifications or the Technical Requirements Manual. This change does not adversely affect the overall performance or reliability of the Condensate System (N19) or the Main and RFP Turbine Seal Steam and Drain System (N33) in a manner that could lead to an accident occurring. This change does not cause the systems to be operated outside of their design basis limits. The new chrome moly piping and permanent blank will not affect any system interface in a way that could lead to an accident. The new chrome moly piping and permanent blank will not result in degradation of safety systems. The piping and fittings installed by this design change meet ANSI B31 .1 code requirements and is supported for the appropriate dead weight and thermal loads. Therefore, the margin of safety as defined in the bases for the Technical Specifications has not been reduced.

Evaluation Number: 2001-0012-R00

Document Evaluated: Temp. Alt. 2001-0002

BRIEF DESCRIPTION OF CHANGE, TEST OR EXPERIMENT:

This temporary alteration will remove the lower portion of the "B" offgas regenerative skid dryer/chiller loop seal piping between the 1N64F207B connection to the loop seal and the reducer to the four (4") piping and replace it with a clear tygon tubing.

REASON FOR CHANGE, TEST OR EXPERIMENT:

To have capability to monitor and/or remove loop seal sediment to establish a performance trend.

SAFETY EVALUATION SUMMARY AND CONCLUSIONS:

This temporary alteration will not affect operability, functionality or radiation monitoring of the offgas system but it will provide capability to monitor or remove restricting sediment and/or restriction of the "B" drain loop seal. Neither function and operability of this system nor any other (safety or non-safety related) system will be affected. The standard operating procedure of this system will not be affected by this temporary alteration. Failure of the system will not impact normal plant operation. The portion of the system used in this temporary alteration is not part of normal offgas flow and is lined up to be part of the regenerative cycle for an out of service bed (SAR sec. I5.7.1).

Evaluation Number: 2001-0013-R00

Document Evaluated: LDC-2001-003, ER 2000-0792-03-00

BRIEF DESCRIPTION OF CHANGE, TEST OR EXPERIMENT:

This ER will address the design and installation of the Auxiliary Cooling Tower (ACT) circulating Water System (N71) tie-ins. The modification will include the installation of 2 reducing tees, 4 throttling valves (2-120" valves and 2-96" valves), 2 expansion joints, instrumentation and miscellaneous hardware supporting these installations. In addition, fill lines, appropriate vent valves and supports will be installed for the new 96" ACT tie-ins.

This ER authorizes the installation of these tie-ins in the circulating Water System condenser outlet lines located near the natural draft cooling tower inside the Protected Area. The excavation and installation of various valve pits and the installation of the ACT are outside the scope of this ER and are covered in ER's 2000-0792-001, 2000-0792-002, 2000-0792-005 and 2000-0792-014.

Existing flow elements N1N71-N037A & B and flow transmitters N1N71-N038A & B were removed by Temp Alt 00-004, and the boundary of the alteration was shifted by Temp Alt 00-011. This ER will make these temporary changes permanent. Therefore, the UFSAR will be revised to eliminate the requirement to isolate acid feed on low circulating water flow.

This 50.59 Evaluation supports LDC 2001-003 which revises UFSAR Section 10.4.5.2 and UFSAR Figure 10.4-005.

REASON FOR CHANGE, TEST OR EXPERIMENT:

The natural draft cooling tower is designed for a wet bulb temperature of 79^0 F. The design range and the design approach to wet bulb are 30.4^0 F and 18^0 F respectively (Specification No. 9645-M-015.5). This results in a cooling water inlet temperature to the condenser of 97^0 F at the design wet bulb.

The GGNS condenser, however, is designed for 572,000 gpm flow at a design inlet temperature of 85^0 F (Specification No. 9645-M-004.0). Rated turbine/generator output cannot be achieved at higher inlet water temperatures due to the inability to maintain sufficient condenser vacuum. Because elevated wet bulb temperatures typically occur during periods of peak electrical demand, this mismatch in performance parameters results in degraded generator electrical output coincident with periods of peak-electrical demand. Based on regional data, the design condenser inlet water temperature is exceeded slightly over 50% of the year.

Future nuclear unit thermal power uprates will further accentuate this issue; with out the capability to maintain condenser cooling water inlet temperatures near the design 85^0 F when ambient conditions are at or below the design wet bulb temperature, projected power uprate benefits will be limited.

The reason for the change in this ER 2000-0792-003 is to install piping required to tie-in the future ACT.

SAFETY EVALUATION SUMMARY AND CONCLUSIONS:

The natural draft cooling tower (W20), including the circulating water system (N71) serve no safety functions and are not safety-related systems. Failure of these systems will not compromise safety-related systems and will not prevent safe reactor shutdown. The primary concern with these systems is the possibility of flooding safety-related- structures. This change will not increase the probability of flooding, change the location of flooding, or increase the severity of flooding. The failure of water piping located in the yard is not an analyzed event at GGNS since the resultant flooding in the yard would not affect safety related-structures, components or systems. This change does not adversely impact the existing flooding analysis for the failure of the circulating water piping in the Turbine Building. All construction activities will be completed during a station outage when the Circulating Water system is not required to be operational. The impact of the open excavation on site PMP was addressed in ER 2000-0792-001. The design and installation of the new piping will be performed to the same quality as the original piping system, therefore affects on ground water will not be adversely changed. Therefore, the activities associated with the implementation of this ER create no failure modes or accident initiators (e.g., flooding or plant trip due to loss of circulating water).

This ER will remove flow elements and flow transmitters which provide automatic isolation of the acid feed to the CW system on a low flow condition and indication to the control room of CW low flow. According to the alarm response instruction (Ref. 6), the only action required by the low flow condition is to secure acid injection. The only known cause of low CW flow is system trip or system shutdown, during which the CW SOI (Ref. 5) requires that acid injection be secured. Furthermore, the system is currently operated in this manner without adverse results. Therefore this change will not increase the probability of CW malfunction.

The proposed change is within the licensing basis of GGNS. This Safety Evaluation documents the fact that the proposed change does not result in an Unreviewed Safety Question for the following reasons:

1) It does not put the plant operation in an unanalyzed region. The changes herein are bounded by the analysis in the Technical Specifications, the TRM and the SAR.

2) The proposed change does not downgrade the performance of any structure, system, or component as defined in the SAR, the TRM or the Technical Specifications.

Because the changes described above will meet or exceed the requirements of the original design (component integrity, capacity, functionality, etc.) and existing analyses, they will not degrade any important to safety systems, components, or structures nor will they degrade or prevent actions described in the SAR accident analysis. The changes do not increase the probability of occurrence or the consequences of an accident or malfunction of equipment important to safety previously evaluated in the SAR, and do not create the possibility for an accident or malfunction of a different type than any evaluated previously in the SAR. The Technical Specifications and the Technical Requirements Manual are not affected, and the margin of safety remains unchanged. Therefore, this change does not constitute an Unreviewed Safety Question.

Evaluation Number: 2001-0014-R00

Document Evaluated: Specification M-500.00 (FHA), LDC-2000-043

BRIEF DESCRIPTION OF CHANGE, TEST OR EXPERIMENT:

Rev. 12 of the FHA revises the method used in the Fire Hazards Analysis Report to describe combustible loading in safety related fire zones/areas. Currently the FHA refers to combustible loading in a given safety related fire zone in 15 minute increments (e.g. :≤15 min., ≤ 30 min. etc.). The 15 minute increment includes all insitu (permanent) combustibles and small amounts of transient combustibles. The new method will: 1) Quantify the fire duration for in situ combustibles using the existing 15-minute increments and 2) Quantify the total (includes in situ and transient combustibles) fire severity as "Low", "Moderate", or "High" (e.g. Low - ≤60 minutes, Moderate - >60 minutes but ≤120 minutes, & High - ≥120 minutes but ≤180 minutes). This method of describing total fire severity as Low, Moderate, & High is similar to the British Fire Loading Studies classification (Ref. NFPA Fire Protection Handbook, 18th Ed., pg.7-80). This methodology for describing combustible loading has no impact in safety related fire zones where no combustible loading is postulated. The existing 15 minute loading increments will continue to be used but will describe only the insitu combustible loading in each safety related fire zone. Additional changes to be incorporated include various editorial corrections, editorial changes associated with the deletion of text associated with "Unit 2 construction" and incorporation of previously approved NPE FHARRs - 98/0001 (SE 98-0088 R00), 99/0002 (SE 99-0040 R00), 99/0003 (SE 98-0093 R0l), 99/0005 (ER 98-0861-01-00) & 2000/0003 (SE 2000-0029 R00). The editorial corrections/changes are considered "editorial" as defined by section 5.4.1.1.d & h of procedure LI-I0l.

REASON FOR CHANGE, TEST OR EXPERIMENT:

The FHA is incorporated into UFSAR App. 9A by reference, therefore, changes require a safety evaluation. The proposed FHA combustible loading description change is being initiated to provide more flexibility for the use and control of combustible materials required during routine maintenance and work activities. The amount of transient combustibles to be used/stored in a safety related fire zone is administratively controlled by procedure 10-S-03-4. Since combustible loading is described in the FHA and thus the UFSAR, administrative procedure 10-S-03-4 controls the amount of transient combustibles that are used in maintenance and work activities to ensure that the combustible loading described in the FHA is not exceeded without taking additional compensatory measures. Considering that the current combustible loading is described in 15-minute increments and this fire loading includes both in situ and transient combustibles, there is very little margin for transient combustibles before the loading described in the FHA is exceeded. This causes an excessive administrative burden to control these transients when in actuality, considerable margin exists because of the fire ratings of fire zone and fire area boundaries. While the FHA may describe the combustible loading as ≤30 minutes, the fire rating of the area barriers may be as much as 3-hours. Therefore, a considerable margin exists in many cases between the rating of the barriers surrounding an area and the combustible loading contained in that area. By changing the method for describing total combustible loading to Low, Moderate, and High additional margin will be provided to relieve the excessive administrative burden while keeping the total combustible loading in areas well below the ratings of the fire barriers surrounding the areas. The FHA controls/restricts the amounts of transient combustibles allowed in a safety related fire zone by virtue of the combustible loading description (i.e.≤ 15 min., ≤30 min. etc.) for each fire zone. The combustible loading for a given safety related fire zone is maintained at or below its combustible loading limit as described in the FHA, per the administrative guidance provided in procedure 10-S-03-4 for transient fire loads and the ER process which addresses the addition of permanent combustibles. The revision of the FHA fire severity descriptions for each safety related fire zone will be as follows: the total fire severity (insitu plus any transients) will be described as Low (less than or equal to 1 hour), Moderate (less than or equal to 2 hours but greater than 1 hour) or High (less than or equal to 3 hours but greater than 2 hours) and the existing 15 minute loading increment in the FHA will be used to describe the insitu combustible loading only. The revised method of describing the combustible loading in the FHA will produce a combustible heat load margin in each safety related fire zone. This combustible loading margin will then allow reasonable amounts of ordinary transient combustible material for routine maintenance and work activities to be used/stored in a particular safety related fire zone without exceeding the loading as described in the FHA. This increased combustible loading margin would also facilitate the ability to store limited quantities of combustible materials in specially designated storage areas without exceeding the loading as described in the FHA. This change, in addition to subsequent 10-S-03-4 procedure changes, will greatly decrease the administrative burden for plant staff while performing routine maintenance or work activities in safety related areas. The changes being made to the combustible loading descriptions in the FHA do not diminish or circumvent the administrative controls that have been established to control/track transient combustible material in the plant. The FHA changes will provide additional flexibility in the administration of the combustible control program. The combustible control program addresses/governs the use/transport of all combustible material in the plant; however, based on the current program there is little flexibility to support routine operating/work/maintenance activities that require limited amounts of ordinary combustible material. The change in the combustible loading descriptions in the FHA will enhance the ability of the combustible control program to accommodate materials required for routine maintenance and work activities.

SAFETY EVALUATION SUMMARY AND CONCLUSIONS:

Rev. 12 of the FHA revises the method used by the Fire Hazards Analysis Report to describe combustible loading in safety related fire zones. Currently the FHA refers to combustible loading in a given fire zone in 15 minute increments (e.g. ≤15 min., ≤ 30 min. etc.). The 15 minute increment includes all insitu (permanent) combustibles and small amounts of transient combustibles. The proposed change will quantify the total fire severity description of safety related fire zones as Low, Moderate and High (e.g. Low ≤60 min., 60 min < Moderate ≤120 min., 120 min < High ≤ 180 min.) similar to the British Fire Loading Studies classification (Ref. NFPA Fire Protection Handbook, 18th Ed., pg.7-80). This change will have no impact on areas where no combustible loading is currently postulated. The existing 15 minute loading increment will continue to be used, but will describe only the insitu combustible loading present in each safety related fire zone. The FHA fire severity descriptions provide the basis for transient combustible material use/storage limits in safety related areas of the plant. This change will provide Fire Protection Engineering greater flexibility in administering the transient combustible control program. The proposed changes in addition to subsequent changes to procedure 10-S-03-4, will reduce the administrative burden on all plant personnel performing routine maintenance/work activities in safety related areas. The described changes revise the methodology used to describe combustible loading in safety related fire zones in the FHA, makes editorial changes and incorporate previously approved FHARRs. Therefore, the proposed changes will not increase the probability or consequences of accidents or malfunction of equipment important to safety previously evaluated in the SAR. Also, the proposed changes will not create the possibility for an accident or malfunction of equipment important to safety of a different type than any previously evaluated in the SAR. Combustible loading descriptions are not addressed by Technical Specifications (TS) or the Technical Requirements Manual. Therefore, there will be no change to TS or TRM or the Bases for any TS or TRM.

Evaluation Number: 2001-0015

Document Evaluated: ER 2000-0792-004-00-00

BRIEF DESCRIPTION OF CHANGE, TEST OR EXPERIMENT:

The objective of ER-GG-2000-0792-004, Revision 0, is to provide details for the design and construction of the new flume, connecting the future auxiliary cooling tower (ACT) basin to the natural draft cooling tower (NDCT) basin. Specifically, the following structures are addressed in this ER link:

- New reinforced concrete flume, connecting the (future) ACT basin to the existing NDCT basin. The flume is designed to be constructed in two phases, prior to and after RF-11
- Removal of existing NDCT basin wall at new flume tie-in. This work will be performed during RF-11.
- Fabrication of water stop-logs for installation at the three flume branches. Fabrication will be prior to RF-11. Installation will be prior to or during RF-11.
- Fabrication and installation of the catwalk across the three flume branches, and handrails on top of the flumewalls. This work will be performed after RF-11.
- Constructing a 4-inch thick concrete slab at the south end of the two NDCT basin drain sump pits, to provide a clean run-off surface for storm water into the pits.
- Removal of a small section of the NDCT basin dividing wall at the inlet tie-in of the new flume center branch. This work will be performed during RF-11.

ER-GG-2000-0792-004, Revision 0, includes only the design and construction of the above listed reinforced concrete structures and associated structural steel works (e.g. catwalks, stop-logs, etc.). This ER link does not address relocation of existing lamp posts and existing duct banks (if required), and soil excavation. These activities are covered by ER-GG-2000-0792-001. This ER link does not address the construction of the ACT basin. This activity is covered by ER-GG-2000-0792-005. However, the location of the ACT Basin will be added to the UFSAR Figures (based on the 50.59 evaluation for ER-GG-2000-0792-000). All license changes associated with the impact of the flume on system operation (after completion of the ACT and stop log removal), environmental impact, meteorology, flooding, groundwater, site drainage, site geology, site grading, etc., will be addressed by ER-GG2000-0792-000. This 50.59 Evaluation considers the possibility of an Unreviewed Safety Question introduction due to construction activities associated with this ER and operation of the cooling tower with the stop-logs in place.

This 50.59 Evaluation supports LDC 2001-006 which revises UFSAR Figures 1.2-1, 1.2-15, 2.1-1, 2.1-2 and 3.4-1 to show the new ACT flume and basin.

REASON FOR CHANGE, TEST OR EXPERIMENT:

The natural draft cooling tower is designed for a wet bulb temperature of 79^0F. The design range and the design approach to wet bulb are 30.4^0F and 18^0F respectively (specification No. 9645-M-015.5). This results in a cooling water inlet temperature to the condenser of 97^0F at the design wet bulb. The GGNS condenser, however, is designed for 572,000 gpm flow at a design inlet temperature of 85^0F (Specification No. 9645-M-004.0). Rated turbine/generator output cannot be achieved at higher inlet water temperatures due to the inability to maintain sufficient condenser vacuum. Because elevated wet bulb temperatures typically occur during periods of peak electrical demand, this mismatch in performance parameters results in degraded generator electrical output coincident with periods of peak electrical demand. Based on regional data, the design condenser inlet water temperature is exceeded slightly over 50% of the year.

Future nuclear unit thermal power uprates will further accentuate this issue; without the capability to maintain condenser cooling water inlet temperatures near the design 85^{0}F when ambient conditions are at or below the design wet bulb temperature, projected power uprate benefits will be limited.

To address these issues, a new ACT is being designed and constructed. Part of the new auxiliary cooling system requires construction of a reinforced concrete flume, connecting the auxiliary cooling tower (ACT) basin to the existing NDCT basin.

SAFETY EVALUATION SUMMARY AND CONCLUSIONS:

The natural draft cooling tower (W20), including the circulating water system (N71) serve no safety functions and are not safety-related systems. Due to the installation of stop-logs during the outage, the NDCT post-outage function and operation will be the same as that before the outage. Therefore, the activities associated with the implementation of this ER create no failure modes or accident initiators (e.g., flooding or plant trip due to loss of circulating water).

The proposed change is within the licensing basis of GGNS. This Safety Evaluation documents the fact that the proposed change does not result in an Unreviewed Safety Question for the following reasons:

1) It does not put the plant operation in an unanalyzed region. The changes herein are bounded by the analysis in the Technical Specifications, the TRM and the SAR.

2) The proposed change does not downgrade the performance of any structure, system, or component as defined in the SAR, the TRM or the Technical Specifications.

Because the changes described above will meet or exceed the requirements of the original design (component integrity, capacity, functionality, etc.) and existing analyses, they will not degrade any important to safety systems, components, or structures nor will they degrade or prevent actions described in the SAR accident analysis. The changes do not increase the probability of occurrence or the consequences of an accident or malfunction of equipment important to safety previously evaluated in the SAR, and do not create the possibility for an accident or malfunction of a different type than any evaluated previously in the SAR. The Technical Specifications and the Technical Requirements Manual are not affected, and the margin of safety remains unchanged. Therefore, this change does not constitute an unreviewed safety question.

Evaluation Number: 2001-0016-R00 Document Evaluated: ER 2000-0792-07-00

BRIEF DESCRIPTION OF CHANGE, TEST OR EXPERIMENT:

The new ACT System will consist of a mechanical draft cooling tower provided with motor driven fans, therefore, the new ACT requires electrical power. The power for the new ACT fans and other associated components will be provided by existing BOP Transformer No. 14 via 13.8 kV BOP Bus 19UD located in the Water Treatment Building. This ER makes the connection to BOP Bus I9UD in the compartment for breaker 352-1901; this connection consists of a portion of the cable that will eventually supply the new ACT transformer, and installation of a new cable tray. This cable will remain in the cable tray in the Water Treatment Building and breaker 352-1901 will remain removed from its cubicle until such time the new ACT transformer is installed and the final connection between the new ACT transformer and BOP Bus 19UD is made (outside the scope of this ER Supplement). Additionally, labeling/mimic changes will be performed on panel SH13-P854 for the change of system/function of the Circuit Breaker Handswitch.

BOP Bus 19UD and BOP Transformer No. 14 are required to be de-energized to perform the work within the scope of this ER; this may require de-energization and will require re-energization of 34.5kV Bus 12R during plant shutdown.

UFSAR Figure 8.1-001 (E-0001), Main One Line Diagram, is being revised to reflect that breaker 352-1901 will be used for the future ACT and note 8 is being added to indicate that breaker 352-1901 is being removed until the new ACT transformer is installed.

REASON FOR CHANGE, TEST OR EXPERIMENT:

This ER makes an electrical tie to BOP Bus 19UD to provide an electrical source for the future ACT transformer which will supply the electrical requirements for the ACT.

The UFSAR change is made to reflect the impact on the licensing basis documents for GGNS.

SAFETY EVALUATION SUMMARY AND CONCLUSIONS:

Construction activities will be performed with BOP transformer Number 14 and Bus 19UD de-energized. No new failure modes or accident initiators will be created by this change.

The breaker associated with BOP Bus 19UD (352-1901) will be physically removed and stored in a suitable location until installation of cable to the new ACT transformers is completed by a future Supplement.

The proposed change is within the licensing basis of GGNS. This Safety Evaluation documents the fact that the proposed change does not result in an Unreviewed Safety Question for the following reasons:

1) It does not put the plant operation in an unanalyzed region. The changes herein are bounded by the analysis in the Technical Specifications, the TRM and the SAR.

2) The proposed change does not downgrade the performance of any structure, system, or component as defined in the SAR, the TRM or the Technical Specifications.

Because the changes described above will meet or exceed the requirements of the original design and existing analyses, they will not degrade any important to safety systems, components, or structures nor will they degrade or prevent actions described in the SAR accident analysis. The changes do not increase the probability of occurrence or the consequences of an accident or malfunction of equipment important to safety previously evaluated in the SAR, and do not create the possibility for an accident or malfunction of a different type than any evaluated previously in the SAR. The Technical Specifications and the Technical Requirements Manual are not affected, and the margin of safety remains unchanged. Therefore, this change does not constitute an unreviewed safety question.

Evaluation Number: 2001-0017 Document Evaluated: LDC 2000-043

BRIEF DESCRIPTION OF CHANGE, TEST OR EXPERIMENT:

The change to UFSAR Table 3.9-3C corrects the valve size for P53F006 from 1.0" to 0.75". The change to the UFSAR is based on GGNS-M-189.1 and Drawing M-1067A. The change to UFSAR Table 9.3-1 corrects the figure number and location for valve P53F026A from Figure 9.3-1, location C-6 to Figure 9.3-1 B, location G-6 and for valve P53F026B from Figure 9.3-1, location D-6 to Figure 9.3-2D, location G-8. The change to UFSAR Table 9.3-2 corrects the figure number and location for valve P53F001 from Figure 9.3-1, location B-5 to Figure 9.3-1 B, location F-5, for valve P53F026A from Figure 9.3-1 location C-6 to Figure 9.3-1 B, location G-6 and for valve P53F026B from Figure 9.3-1 location D-6 to Figure 9.3-2D, location G-8. The change to the UFSAR is based on drawings M-1067A, M-1067E, and M-1067M. The manner in which the system is operated is not changed and will not affect the Technical Specifications. The proposed change does not change plant procedures and does not involve any test or experiments.

REASON FOR CHANGE, TEST OR EXPERIMENT:

The reason for the change to UFSAR Table 3.9-3C is to correct the valve size for P53F006 from 1.0" to 0.75". The reason for the change to UFSAR Tables 9.3-1 and 9.3-2 is to correct the figure number and location for valves P53F001, P53F026A, and P53F026B. These changes ensure the UFSAR is consistent and accurately reflects other design basis documentation.

SAFETY EVALUATION SUMMARY AND CONCLUSIONS:

The change to UFSAR Table 3.9-3C revises the valve size for P53F006 from 1.0" to 0.75". The change to UFSAR Table 9.3-1 corrects the figure number and location for valve P53F026A from Figure 9.3-1, location C-6 to Figure 9.3-1 B, location G-6 and for valve P53F026B from Figure 9.3-1, location D-6 to Figure 9.3-2D, location G-8. The change to UFSAR Table 9.3-2 corrects the figure number and location for valve P53F001 from Figure 9.3-1, location B-5 to Figure 9.3-1 B, location F5, for valve P53F026A from Figure 9.3-1, location C-6 to Figure 9.3-1 B, location G-6, and for valve P53F026B from Figure 9.3-1,location D-6 to Figure 9.3-2D, location G-8. The revision to UFSAR Tables 9.3-1 and 9.3-2 corrects the figure number and location for valves P53F001, P53F026A, and P53F026B. These changes do not change the manner in which the system is operated or the limits placed on containment isolation as described by the Technical Specifications. Therefore, the change will not increase the probability of occurrence or the consequences of an accident or malfunction of equipment important to safety previously evaluated in the safety analysis report. The revision to UFSAR Tables 3.9-3C, 9.3-1 and 9.3-2 does not affect the equipment installed in the plant, how the equipment is operated, or the potential types of malfunctions of the equipment. Therefore the change does not create the possibility of an accident or malfunction of a different type than any evaluated previously in the safety analysis report. The revision to UFSAR Tables 3.9-3C, 9.3-1 and 9.3-2 does not change the limits placed on containment isolation by the Technical Specifications or the associated required actions. Therefore, the change does not reduce the margin of safety as defined in the basis for any Technical Specification.

Evaluation Number: 2001-0018 Document Evaluated: LDC 1998-050

BRIEF DESCRIPTION OF CHANGE, TEST OR EXPERIMENT:

UFSAR Table 9.2-7 provides a listing of the Component Cooling Water (P42) component descriptions. The description for the CCW Surge Tank design pressure is being corrected to indicate that the design pressure of the tank is atmospheric.

REASON FOR CHANGE, TEST OR EXPERIMENT:

The design pressure information in the Table is not consistent with the design documentation. This change will correct the design pressure listed in the Table for this component.

SAFETY EVALUATION SUMMARY AND CONCLUSIONS:

The change in the design pressure for the COW surge tank indicated in the FSAR will not affect any present requirements or add any new requirements to the Technical Specifications. The CCW surge tank is not safety related. The change in the FSAR information will not affect the actual system design or operation and it will not affect any system that interfaces with the CCW system. A failure of the surge tank would not compromise any safety related system or component and would not prevent a safe reactor shutdown.

Evaluation Number: 2001-0019-R00 Document Evaluated: LDC 2000-032

BRIEF DESCRIPTION OF CHANGE, TEST OR EXPERIMENT:

The change to UFSAR Section 6.2.4.2.10 corrects the document associated with valve operability. The change to the UFSAR is based on Pump & Valve Inservice Testing Program, GGNS-M-189.1. The change to note (v) is an editorial change in that note (v) is confusing and will be revised. The manner in which the system is operated is not changed and will not affect the Technical Specifications. The proposed change does not change plant procedures and does not involve tests or experiments.

REASON FOR CHANGE, TEST OR EXPERIMENT:

The reason for the change to UFSAR Section 6.2.4.2.10 corrects the reference made to the document that specifies the valve operability requirements. The reason for the change to UFSAR Table 3.2-1 Note (v) is the note is confusing with regards to the outboard feedwater check valves, B21-F032A/B. The reason for the change is to ensure the UFSAR is consistent and accurately reflects design basis.

SAFETY EVALUATION SUMMARY AND CONCLUSIONS:

The change to UFSAR Section 6.2.4.2.10 corrects the reference made to the document that specifies the valve operability requirements. UFSAR Table 6.2-44 does not indicate which valves will be cycled during normal operation to assure their operability. Pump & Valve Inservice Testing Program GGNS-M-189.1 specifies the valves and their testing requirements for valve operability. The change to UFSAR Table 3.2-1 Note (v) is a revision to clarify the portion of the note affecting the outboard feedwater check valves, B21-F032A/B. The revision to UFSAR Section 6.2.4.2.10 corrects the reference made to the document that specifies the valve operability requirements and does not change the manner in which the system is operated or the limits placed on containment isolation as described by the Technical Specifications. The revision to UFSAR Table 3.2-1 Note (v) revises the portion of the note affecting the B21-F032A/B. Therefore, these changes will not increase the probability of occurrence or the consequences of an accident or malfunction of equipment important to safety previously evaluated in the safety analysis report. The revisions to UFSAR Section 6.2.4.2.10 and UFSAR Table 3.2-1 Note (v) does not affect the equipment installed in the plant, how the equipment is operated, or the potential types of malfunctions of the equipment. Therefore, these changes do not create the possibility of an accident or malfunction of a different type than any evaluated previously in the safety analysis report. The revision to UFSAR Section 6.2.4.2.10 does not change the limits placed on containment isolation by the Technical Specifications or the associated required actions. The revision to UFSAR Table 3.2-1 does not change the function or operation of the B21-F032A/B. Therefore, these changes do not reduce the margin of safety as defined in the basis for any technical specification.

Evaluation Number: 2001-0020-R00 Document Evaluated: LDC 2001-013

BRIEF DESCRIPTION OF CHANGE, TEST OR EXPERIMENT:

The close testing requirement for the P52-FI96 check valve is being deleted. The P52-FI96 is the inboard drywell isolation check valve for the Service Air system. The valve will be locked closed during modes 1, 2, and 3 in order to assure the penetration is isolated.

REASON FOR CHANGE, TEST OR EXPERIMENT:

Testing the valve is operator intensive and requires resources that could be used elsewhere. Deleting the close test on the P52-F196 will free up resources.

SAFETY EVALUATION SUMMARY AND CONCLUSIONS:

The close testing requirement for the P52-F196 check valve is being deleted. The P52-F196 is the inboard drywell isolation check valve for the Service Air system. The valve will be locked closed during modes 1, 2, and 3 in order to assure the penetration is isolated. Maintaining the valve locked closed in modes 1, 2, and 3 assure that the Service Air Drywell penetration is isolated. The penetration is not inoperative in these modes. The penetration is not required to be isolated in modes 4 and 5, therefore the penetration can be unisolated and utilized in these modes. As added assurance that the drywell penetration is isolated, the P52-F195, Outboard Isolation MOV is closed per SOI 04-S-0I-P52-1, step 3.7. With the P52-F196 closed in modes 1, 2, and 3, the drywell isolation of the penetration is assured. During modes 4 and 5 when the penetration could be unisolated, the drywell isolation function is not needed because during modes 4 or 5, a LOCA would be more of a flooding concern than pressurization of the drywell. The probability and consequences of these events would be reduced by the pressure and temperature during these modes. The accident postulated for modes 4 and 5 is a fuel handling accident that occurs either in the containment pool or the auxiliary building fuel pool. Both of these accidents are outside of the drywell and drywell isolation would not help in these cases. Per Technical Specification Bases 3.6.5.3, the drywell isolation valves are not required to be leak rate tested. The valves only have to go closed. The drywell leakage limit is addressed in the Technical Specification for the Drywell.

Evaluation Number: 2001-0021-R00

Document Evaluated: LDC-2000-015
ER 2001-0040-00

BRIEF DESCRIPTION OF CHANGE, TEST OR EXPERIMENT:

The condensing unit, SX57B025, for the chemistry lab HVAC system has failed. This ER replaces the failed unit with the factory recommended replacement unit, Trane Model TTA180C400E.

REASON FOR CHANGE, TEST OR EXPERIMENT:

The condensing unit, SX57B025, for the chemistry lab HVAC system has failed. This ER replaces the failed unit with the factory recommended replacement unit, Trane Model TTA180C400E.

SAFETY EVALUATION SUMMARY AND CONCLUSIONS:

This ER will replace the failed condensing unit of the Chemistry laboratory with a new model condensing unit. No changes are being made to the basic system design. The changes do not result in a new release pathway and do not affect offsite dose. The modification has no affect on equipment considered important to safety and does not cause any systems or components to be operated outside design limits. No new failure modes are introduced. It is concluded that the modification will not degrade any important to safety systems, components or structures and will not degrade or prevent actions described in the SAR safety analysis. The changes do not increase the probability of occurrence or increase the consequences of malfunction of equipment important to safety. The Technical Specifications are not affected and the margin of safety remains unchanged.

Evaluation Number: 2001-0022-R00 Document Evaluated: LDC 2000-083

BRIEF DESCRIPTION OF CHANGE, TEST OR EXPERIMENT:

Currently TRM (and UFSAR Appendix 16B.1) Surveillance Requirement SR TR3.8.1 .2 states:

"Subject the diesel to an inspection in accordance with procedures prepared in conjunction with its manufacturer's recommendations for this class of standby service every 18 months. Note: Inspections that call for significant engine internals disassembly or require a retest that cannot be performed on-line shall not be performed in MODE 1 or 2".

Currently UFSAR Section 8.3.1.1.4.1.2, Maintenance, states:
"Preventive maintenance will be performed in accordance with manufacturer's recommendations based on engine run time and calendar time and in accordance with approved maintenance procedures. Upon completion of repairs or maintenance, complete system and equipment checks will be performed to ensure proper system operability. The diesel generator will then be returned to standby service under the control of the control room operator".

The manufacturer developed preventive maintenance recommendations for DIV I/II SDG engines based primarily on engine run time in commercial applications since there was no experience base for determining an appropriate inspection frequency for engines in the nuclear plant environment (that ran for a few hours a year - typically less than 100 hours a year versus 8760 hours in the commercial environment). Consequently, a conservative approach was initially taken, and many inspections were specified at every refueling outage.

In the 15 years since this program was first developed, a tremendous amount of operating experience has been accumulated, and the program has been periodically revised to reflect this experience. In most cases, inspections/maintenance have been relaxed, but in cases where problems were found, inspections/maintenance have been intensified. Additionally, during the early years of operation, these SDGs have had numerous modifications that have improved performance and availability.

REASON FOR CHANGE, TEST OR EXPERIMENT:

The purpose of this 50.59 Evaluation is to evaluate LDC 2000-083, which revises TRM Surveillance Requirement SR TR3.8.1.2 (and UFSAR Appendix 16B.1) to require that the Division I and II diesels be subjected to an inspection, commensurate for nuclear standby service, that takes into consideration the following factors: the manufacturer's recommendations, diesel owners group recommendations, engine run time, calendar time, and the GGNS comprehensive maintenance inspection program (the change does not impact the HPCS DG).

Likewise, UFSAR Section 8.3.1.1.4.1.2 is being revised to state: "Preventive maintenance will be performed in accordance with an approved maintenance program, commensurate for nuclear standby service, that takes into consideration the following factors: the manufacturer's recommendations, diesel owners group recommendations, engine run time, calendar time, and the GGNS comprehensive maintenance inspection program."

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The current UFSAR and TRM statements could be interpreted to mean that the Division I and II diesel generator maintenance program must be in verbatim compliance with manufacturer recommendations. The changes are being made so that maintenance requirements and inspection frequencies can be determined based on a combination of the latest recommendations of the manufacturer, the Cooper-Enterprise Owners Group recommendations, engine run time, calendar time, and the GGNS comprehensive maintenance inspection program. Many of the recommendations of the manufacturer will continue to be followed; however, where the other above-listed inputs support deviation from these recommendations, the maintenance requirements may be adjusted accordingly. This could result in either more or less stringent maintenance requirements depending on inspection findings. It is intended that the maintenance program for the Division I and II SDGs be a dynamic program that is refined as more experience is gained from future inspections, and methodologies for determining the condition of the engines are advanced. This change will make the SDG maintenance program comparable to the programs of other safety-significant systems.

SAFETY EVALUATION SUMMARY AND CONCLUSIONS:

A review of past inspection results, license commitments, CRs, MNCRs, and manufacturer's/owners group recommended maintenance program is contained in Engineering Report No. GGNS-01-0001, Rev 0. The review resulted in proposed changes to the Divisions I and II diesel maintenance program that will provide for improved diesel generator availability by reducing the frequency of certain invasive inspections, thereby reducing the total cumulative time required to perform maintenance and inspections.

The changes made to the TRM and UFSAR do not alter the operation of the Standby Diesel Generator System or the response to an accident that the Diesel Generators are required to help mitigate.

The changes do not cause a greater reliance to be placed on any specific system, structure, or component to perform a safety function. The changes do not degrade the performance or reliability of a safety system assumed to function in the accident analysis. The changes do not put the plant operation in an unanalyzed region. The changes do not adversely affect the overall performance or reliability of a safety system in a manner that could lead to an accident occurring.

Evaluation Number: 2001-0023-R00

Document Evaluated: LDC 2000-080 (ER 2000-0859-00-00)

BRIEF DESCRIPTION OF CHANGE, TEST OR EXPERIMENT:

ER-GG-2000-0859-000 installs a LEFM Check Plus flow measuring instrument network in the Feedwater System. The LEFM Check Plus network consists of two 24-inch diameter spool pieces, each containing sixteen transducers, and an electronics cabinet. One spool piece is to be installed in each of the two 24-inch feedwater lines located inside the turbine building steam tunnel. The transducers mounted on the spool pieces provide feedwater flow and temperature measurement signals for each individual feedwater line to the electronics-cabinet that is to be located outside the turbine building steam tunnel on elevation 133'. A total feedwater flow signal will be transmitted by modem from the electronics cabinet to the PDS MIJX cabinet but will not be electrically connected to the NSSS computer by this ER. The cable from the LEFM electronics cabinet is connected to the PDS MUX components mounted in Panel 1C91P860. The cables will be terminated by a separate ER supplement which will also remove the temporary PDS MLJX and will tie the LEFM system to the NSSS computer MUX in panel 1C91P866. The LEFM Check Plus network is classified as non-safety related. However, the requirements of I0CFR50 Appendix B have been invoked by Specification J-912.0 for the LEFM Check Plus software and laboratory calibration tests to insure reliability of the equipment. The electrical load is being supplied from BOP inverter 1Y98 which is powered from BOP 125 Vdc bus 11DK which is connected to BOP Battery Chargers 1K4 and 1K5 which are fed from Class IE Load Centers 15BA1 and 15BA2, which are fed by the Division I diesel generator. ER-GG-2000-0859-00-00 also installs two pressure transmitters to measure feedwater line pressure at the upstream side of feedwater flow elements 1C34N001A/B, and transmit the pressure signal to the LEFM Check Plus electronics cabinets for use in calculating total feedwater flow.

LDC 2000-080 updates UFSAR Tables 8.3-1 and 8.3-2 to reflect the added load during a loss of offsite power, there is no impact to the LOCA load, therefore, there is no impact on diesel loading during a LOCA and therefore, no impact on the Diesel Fuel Oil Storage Requirements as reflected in calculation MC-Q1P75-90190, Rev. 2 or in the Technical Specifications LCO 3.8.3, Diesel Fuel Oil, Lube Oil, and Starting Air, or its bases. LDC 2000-080 also updates Figure 10.4-13 to reflect equipment addition and UFSAR Section 8.3.1.2.3.e to reflect a deviation from color coding of cables; the new flow measuring instrument network vendor cables do not comply with the color coding specified in the UFSAR. Calculations issued in support of these changes are: NPE-PDS-139 Supplement 4, Rev. 0, EC-N1L11-95002 Rev. 3, ECQ1111-93001, Supplement 1, Rev. 0, JC-N1C34-N105-1, Rev. 0. Associated SCN 00/0015A to MS-02, Rev. 49 was also issued. Calculation MC-N1111-93009, Revision 0, "Feedwater Hydraulic Flow/Pressure Evaluation", has been reviewed and is not impacted as a result of this ER.

REASON FOR CHANGE, TEST OR EXPERIMENT:

GGNS plans to pursue NRC approval for an increase in the plant's licensed reactor thermal power limit (currently 3833 MWt) in the near future. Installation of the LEFM Check Plus feedwater flow measurement system with it's significant accuracy improvement over the currently installed equipment is an important part of the power uprate effort. Increased instrument accuracy for feedwater flow measurement provides the primary justification-for increasing the licensed-thermal power limit. The LEFM Check Plus network will be installed and tested under ER-GG-2000-0859-00-00, but will not be used in support of plant operation until connected to the NSSS computer and separately evaluated for such use as part of the future power uprate effort. The LEFM Check Plus system, consisting of flow measurement spool pieces installed in the feedwater piping inside the turbine building steam tunnel and an electronics cabinet located in the turbine building, is also provided. The LEFM Check Plus system provides a total feedwater flow signal to the PDS computer that will be used to support a future project for implementing a power uprate. The safety analysis evaluating the actual use of the LEFM flow signal for power operations will be performed in association with the future Appendix K project.

SAFETY EVALUATION SUMMARY AND CONCLUSIONS:

The changes within the scope of this ER do not adversely impact any important to safety equipment. The LEFM Check Plus system is non-safety related, as is the feedwater system in which it is being installed. However, 10CFR50 Appendix B requirements are applied to the system software and laboratory calibration tests to insure reliability of the equipment. The new instrument network will provide a total feedwater flow signal, but will not actually be used to support plant operation in any manner until it is connected to the NSSS computer. Therefore, this evaluation is limited to the physical installation and associated testing of the new instrument network.

Structural integrity of the feedwater system piping outside containment, during a design basis earthquake, is taken credit for in the analysis of potential leakage from the feedwater system check valves (containment isolation valves) The LEFM spool pieces have been evaluated and a determination made that they will satisfy structural integrity requirements of the system piping and tubing. Cable routing between the control building and turbine building complies with all separation criteria, and penetration opening/closing will be performed in accordance with approved plant procedures.

The proposed changes do not adversely impact any accident or equipment malfunction analyses presented in the UFSAR. The changes do not degrade below the current design basis the performance or reliability of any safety system assumed to function in the accident analysis. The changes do not cause a greater reliance to be placed on any specific system, structure or component to perform a safety function. No new failure modes are created thus the possibility of an accident or malfunction of a different type than previously analyzed is not possible. The changes do not adversely affect the overall performance or reliability of any system in a manner that could lead to an accident occurring Additionally, the margin of safety as defined in the bases for the Technical Specifications has not been reduced.

Evaluation Number: 2001-0024-R00

Document Evaluated: LDC 2000-015
ER 2000-0118-00-00

BRIEF DESCRIPTION OF CHANGE, TEST OR EXPERIMENT:

This ER will remove the disc position indication switch, position switch 1E22N103, the position switch actuator rod and associated wires and conduit, from valve 1E22F005, HPCS Primary Containment isolation valve. It will also remove the indicating lights for the disc position indication of the valve from main control room panel 1H13P601-16C. The disc position indication switch is used for remote disc position indication during valve testing.

REASON FOR CHANGE, TEST OR EXPERIMENT:

Valve 1E22F005 has a history of the position switch actuator rod causing the valve disc to stick resulting in dual indication in the control room. This frequently requires valve disassembly to correct

SAFETY EVALUATION SUMMARY AND CONCLUSIONS:

No changes are being made that affect basic system design functions. The disc position indication switch is used for remote disc position indication during valve testing. The changes do not affect the ability of valve 1E22F005 to perform its function under accident conditions. These changes do not result in a new pathway for the release of radioactive materials- and do not affect offsite dose. No assumptions utilized in evaluating consequences of an accident will be altered. No new failure modes are created and there is no increase in previously identified failure modes for equipment important to safety. No assumptions utilized in evaluating the consequences of a malfunction of equipment important to safety will be altered by this change. This modification does not introduce any new failure modes and does not affect equipment other that the check valve and its associated disc position indication. Secondary and indirect effects (Fire protection, fire loading, pipe break, electrical shorts) have been reviewed and no increased probability of failure of equipment important to safety due to these concerns has been identified. These changes will not degrade any important to safety systems, components or structures nor will they degrade or prevent actions described in the SAP. analysis. They do not increase the probability of occurrence or increase the consequences of malfunction of equipment important to safety of a different type than previously evaluated in the SAR. The Technical Specifications are not affected and the margin of safety is unchanged.

Evaluation Number: 2001-0025-R00 Document Evaluated: ER 2000-0183-00-00

BRIEF DESCRIPTION OF CHANGE, TEST OR EXPERIMENT:

ER 2000/0183-00-00 installs a dc-powered pump in the Turbine Building Cooling Water (TBCW) system to supply cooling water to the generator air-side seal oil coolers during loss of ac power conditions. The new pump will be powered from non-safety related 250 Vdc bus 11DF (batteries 1 K3 and 1 L3). This will allow Operations 60 minutes to respond to a loss of AC Power to vent hydrogen from the generator.

LDC 2001-032 updates UFSAR Tables 8.3-1 and 8.3-2 to reflect the added load during a loss of offsite power. LDC 200 1-032 also updates the UFSAR to indicate the modifications to the TBCW system, 250V DC bus 11DF, and 125VDC bus 11DL.

REASON FOR CHANGE, TEST OR EXPERIMENT:

Seal oil, supplied at a higher pressure than the hydrogen gas used for main generator cooling, prevents leakage of hydrogen from the generator to the atmosphere along the generator shaft. The existing Hydrogen Seal Oil System contains a dc-powered pump that supplies seal oil to the air-side generator seals when neither of the two ac-powered seal oil pumps are available (e.g., during a loss of ac power event). However, a loss of ac power also results in the loss of TBCW cooling water to the hydrogen seal oil coolers such that the oil being circulated by the dc pump is not cooled, and the oil temperature may increase significantly above normal. Siemens, the turbine-generator OEM, has identified a concern that high seal oil temperature will allow hydrogen to escape past the shaft seals, creating a potential fire or explosion hazard in the surrounding area of the turbine building. Installation of a dc-powered TBCW pump will alleviate that concern.

SAFETY EVALUATION SUMMARY AND CONCLUSIONS:

The changes within the scope of this ER do not impact any important to safety equipment. The TBCW system does not have any safety-related functions (UFSAR 3.2 and 9.2.9). Failure of the TBCW system will not compromise any safety-related system or component and will not prevent safe reactor shutdown (UFSAR 9.2.9.3). The changes made by ER 2000/0183-00-00 do not alter the system in a way that makes these statements untrue. Additionally, the new dc pump is powered from BOP (non-safety-related) buses/batteries. The changes affect the Class 1E power systems by increasing the EDG loading during forced shutdown. However, the total load remains within the EDG rating. There is no change to the LOCA load and therefore no change to the Diesel Fuel Oil Storage Requirements as reflected in calculation MC-Q1P75-90190, Rev. 2 (which addresses fuel oil consumption for LOCA/LOP), or in the Technical Specifications LCO 3.8.3, Diesel Fuel Oil, Lube Oil, and Starting Air, or its bases.

The proposed change is within the licensing basis of GONS. This Safety Evaluation documents the fact that the proposed change does not result in an Unreviewed Safety Question for the following reasons:

1) It does not put the plant operation in an unanalyzed region. The changes herein are bounded by the analysis in the Technical Specifications, the TRM and the SAR.

2) The proposed change does not downgrade the performance of any structure, system, or component as defined in the SAR, the TRM or the Technical Specifications.

Because the changes described above will meet or exceed the requirements of the original design (component integrity, capacity, functionality, etc.) and existing analyses, they will not degrade any important to safety systems, components, or structures nor will they degrade or prevent actions described in the SAR accident analysis. The changes do not increase the probability of occurrence or the consequences of an accident or malfunction of equipment important to safety previously evaluated in the SAR, and do not create the possibility for an accident or malfunction of a different type than any evaluated previously in the SAR. The Technical Specifications and the Technical Requirements Manual are not affected, and the margin of safety remains unchanged. Therefore, this change does not constitute an unreviewed safety question.

Evaluation Number: 2001-0026-R00

Document Evaluated: LDC 2000-054
(ER 2000-0052-01-00)

BRIEF DESCRIPTION OF CHANGE, TEST OR EXPERIMENT:

This ER installs a new Division I (Loop A) Standby Service Water (SSW) pump. The new pump will be manufactured utilizing stainless steel components. The associated LDC makes the following change:

UFSAR Table 3.9-25d is being revised to reflect the results of the seismic analysis for the new pump.. Calculation C-C811, Rev. 0, Supplement 2 and vendor document M-931.0-Q1P41C001B-7.0-1-0 support this UFSAR change.

REASON FOR CHANGE, TEST OR EXPERIMENT:

The utilization of stainless steel components in the new pump will enhance reliability of the pump and minimize required maintenance compared to the existing Div I SSW pump which utilizes predominantly carbon steel components. The new pump will be manufactured such that it will have the same fit and function as the existing pump and will have the same performance characteristics as the existing pump.

SAFETY EVALUATION SUMMARY AND CONCLUSIONS:

The Div I SSW pump 1P41C001A-A is discussed in various sections of the UFSAR including Table 3.9-25d (Summary of Results, Equipment: Standby Service Water Pumps) which contains the stress analysis and limits for the pump and UFSAR Table 3.2-1 which indicates that the SSW pumps are Safety Class 3, Quality Group C, constructed to ASME Section III-3 and Seismic Category I. Installation of the new pump requires a change to the information contained in UFSAR Table 3.9-25d to reflect the seismic analysis of the new pump. Since the new pump is essentially a like-for-like replacement, except for materials of construction, there is no change to the operation of the SSW system and therefore, there is no other licensing basis impact. The Technical Specifications, Operating License, TRM and Fire Hazards Analysis do not require a change as a result of issuance of this ER for the installation of the new pump. The new pump was procured such that there will be no adverse impact on diesel generator loading, no additional heat rejected to the SSW system via pump motor cooler or pump work (i.e., will have the same performance characteristics and same power requirements as the existing pump). The existing motor will be used for the new pump.

An electronic search was performed of the GGNS UFSAR, Fire Hazards Analysis, SER (including supplements), Licensing Commitments, Technical Specifications, TRM and Operating License. Keywords (and their variations) used for the search were: SSW pump, standby service water pump, basin, and pump AND material. The following relevant UFSAR Sections were identified and reviewed for impact: UFSAR Section 1.2.2.8.1, Table 1.3-3, Table 32-1, Section 3.9.3.2.2.1.1, Table 3.9-3b, Table 3.9-25d, Table 8.3-1, Section 9.2.1, Table 9.2-3, Table 9.2-4, Table 9.2-16, Table 9.2-17, Figures 9.2-1, 9.2-2, 9.2-3, and 9.2-4. Additionally, the following licensing commitments and documents were identified and reviewed for impact: commitments A-6306, A-6309, A-7083, A-7573, A-7585, A-7846, A-7961, A-9236, Operating License Conditions C.2.40; and Technical Specifications and Bases 3.7.1

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The proposed change is within the existing licensing basis of the Grand Gulf Nuclear Station. This Safety Evaluation documents the fact that the proposed change does not result in an Unreviewed Safety Question for the following reasons: the change does not cause a greater reliance to be placed on any specific system, structure or component to perform a safety function; the change does not degrade the performance of a safety system assumed to function in the accident analysis and does not decrease the reliability of safety systems assumed to function in the accident analysis; the change does not put plant operation in an unanalyzed region; the change herein is bounded by the analysis in the Technical Specifications, the TRM and the SAR; and the change does not adversely affect the overall performance or reliability of a safety system in a manner that could lead to an accident occurring.

Because the change described above will meet or exceed the requirements of the original design (component integrity, capacity, functionality, etc.) and existing analyses, the change will not degrade the performance of any safety systems, components, or structures nor will it degrade or prevent actions described in the SAR accident analysis. The change does not increase the probability of occurrence or increase the consequences of an accident or malfunction of equipment important to safety previously evaluated in the SAR and does not create a different type of accident or malfunction than previously evaluated in the SAR. The Technical Specifications and the Technical Requirements Manual are not affected, and the margin of safety as defined in the basis for any Technical Specification remains unchanged. Therefore, this change does not constitute an unreviewed safety question.

Evaluation Number: 2001-0027-R00

Document Evaluated: LDC-2000-055
ER 2000-052--02-00

BRIEF DESCRIPTION OF CHANGE, TEST OR EXPERIMENT:

This ER installs a new HPCS Service Water pump. The new pump will be manufactured utilizing stainless steel components. The associated LDC makes the following change:

UFSAR Table 3.9-25a is being revised to reflect the results of the seismic analysis for the new pump

REASON FOR CHANGE, TEST OR EXPERIMENT:

The utilization of stainless steel components in the new pump will enhance reliability of the pump and minimize required maintenance compared to the existing HPCS Service Water pump which utilizes predominantly carbon steel components. The new pump will be manufactured such that it will have the same fit and function as the existing pump and will have the same performance characteristics as the existing pump.

SAFETY EVALUATION SUMMARY AND CONCLUSIONS:

The HPCS Service Water System pump 1P41C002-C is discussed in various sections of the UFSAR including Table 3.9-25a (Summary of Results, Equipment: HPCS Service Water Pumps) which contains the stress analysis and limits for the pump, and UFSAR Table 3.2-1 which indicates that the SSW pumps are Safety Class 3, Quality Group C, constructed to ASME Section III-3 and Seismic Category I. Installation of the new pump requires a change to the information contained in UFSAR Table 3.9-25a to reflect the seismic analysis of the new pump. Since the new pump is essentially a like-for-like replacement, except for materials of construction, there is no change to the operation of the SSW system and therefore, there is no other licensing basis impact. The Technical Specifications, Operating License, TRM and Fire Hazards Analysis do not require a change as a result of issuance of this ER for the installation of the new pump. The new pump was procured such that there will be no adverse impact on diesel generator loading, no additional heat rejected to the SSW system via pump work (i.e., will have same performance characteristics and same power requirements as the existing pump). The existing motor will be used for the new pump.

An electronic search was performed of the GGNS UFSAR, Fire Hazards Analysis, SER (including supplements), Technical Specifications, TRM and Operating License. Keywords (and their variations) used for the search were: SSW pump, standby service water pump, pump AND material, and HPCS Service Water pump. The following relevant UFSAR Sections were identified and reviewed for impact: UFSAR Section 1.2.2.8.1, Table 1.3-3, Table 3.2-1, Section 3.9.3.2.2.1.1, Table 3.9-3b, Table 3.9-25a, Table 8.3-3, Section 9.2.1, Table 9.2-3, Table 9.2-4, Table 9.2-16, Table 9.2-17, Figures 9.2-1, 9.2-2, 9.2-3, and 9.2-4. Additionally, the following licensing documents were identified and reviewed for impact: SER Supplement 2 Section 3.10, SER Supplement 4 Section 3.10; Operating License Conditions C.2.10 and 40; and Technical Specifications and Bases 3.7.2.

The proposed change is within the existing licensing basis of the Grand Gulf Nuclear Station. This Safety Evaluation documents the fact that the proposed change does not result in an

Unreviewed Safety Question for the following reasons: the change does not cause a greater reliance to be placed on any specific system, structure or component to perform a safety function; the change does not degrade the performance of a safety system assumed to function in the accident analysis and does not decrease the reliability of safety systems assumed to function in the accident analysis; the change does not put plant operation in an unanalyzed region; the change herein is bounded by the analysis in the Technical Specifications, the TRM and the SAR; and the change does not adversely affect the overall performance or reliability of a safety system in a manner that could lead to an accident occurring.

Because the change described above will meet or exceed the requirements of the original design (component integrity, capacity, functionality, etc.) and existing analyses, the change will not degrade the performance of any safety systems, components, or structures nor will it degrade or prevent actions described in the SAR accident analysis. The change does not increase the probability of occurrence or increase the consequences of an accident or malfunction of equipment important to safety previously evaluated in the SAR and does not create a different type of accident or malfunction than previously evaluated in the SAR. The Technical Specifications and the Technical Requirements Manual are not affected, and the margin of safety as defined in the basis for any Technical Specification remains unchanged. Therefore, this change does not constitute an unreviewed safety question.

Evaluation Number: 2001-0028-R00 Document Evaluated: LDC 2001-038

BRIEF DESCRIPTION OF CHANGE, TEST OR EXPERIMENT:

CR GGN-1997-0403 identified that the drywell purge compressor equipment qualification requirement is not described in the FSAR. This 50.59 evaluates changing the FSAR and Technical Specification Bases to include the drywell purge compressor equipment qualification function. The drywell purge compressor equipment qualification requirement is already credited in the analyses.

REASON FOR CHANGE, TEST OR EXPERIMENT:

CR GGN-1997-0403 identified that the drywell purge compressor equipment qualification requirement is not described in the FSAR. The drywell purge compressor equipment qualification requirement is already credited in the analyses.

SAFETY EVALUATION SUMMARY AND CONCLUSIONS:

This evaluation adds the description of the drywell purge compressor equipment qualification requirement to the FSAR. The drywell purge compressor equipment qualification requirement is already credited in the analyses. Adding the description of the drywell purge compressor to the FSAR provides additional information that the drywell purge compressor provides an equipment qualification function for drywell radiation doses. It does not increase the chances of an accident occurring nor does it change the consequences of an accident occurring. The results of the equipment qualification requirements for the drywell purge compressor are presented in FSAR Section 3.11 for equipment qualification and these results are unaffected by this evaluation. The equipment qualification results in FSAR section 3.11 provide the requirements of electrical equipment per 10CFR50.49 and these results are unaffected meaning the failure or malfunction of equipment has been evaluated for equipment qualification.

Evaluation Number: 2001-0029-R00 Document Evaluated: LDC 2001-039

BRIEF DESCRIPTION OF CHANGE, TEST OR EXPERIMENT:

The UFSAR consistency review for the Component Cooling Water (CCW) system identified several discrepancies with design documents. The proposed change will revise UFSAR applicable sections to resolve these discrepancies. The changes include clarifying classification of the CCW system, correcting postulated leakage rate of the CCW line, deleting references to subsection 7.6.2.10, deleting references to Chapter 15 and Appendix 15A, adding a reference to criterion 56 (GDC 56), correcting design temperature of containment isolation valve to 267 F, adding note to the CCW pump motor load requirements, correcting typo and correcting CCW heat exchanger description on shell side flow rate and inlet temperature. The changes bring into agreement information contained in the UFSAR with that in design calculation, design specifications and drawings.

REASON FOR CHANGE, TEST OR EXPERIMENT:

These discrepancies were due to previous UFSAR changes, incorrect reference number and incorrect design information. All changes have been reviewed with design documents (Specifications, calculation and drawings) and are software related only. These changes will not affect equipment function or performance of the CCW system.

SAFETY EVALUATION SUMMARY AND CONCLUSIONS:

The UFSAR consistency review for the CCW system identified several discrepancies with design documents (Specifications, calculation and drawings). The proposed change will revise UFSAR applicable sections to resolve these editorial type (correcting inconsistencies within UFSAR sections) discrepancies and-correct information without changing intent or scope of the CCW system. These discrepancies were due to previous UFSAR changes, incorrect reference number and incorrect information. All changes have been reviewed with design documents (Specifications, calculation and drawings) and are software related only. These changes will not affect equipment function or performance of the CCW system. The change will not alter the design, function or operation of any equipment important to safety as evaluated in the UFSAR.

Evaluation Number: 2001-0030-R00

Document Evaluated: LDC 2000-015
ER-2000-0118-003

BRIEF DESCRIPTION OF CHANGE, TEST OR EXPERIMENT:

This ER will remove the disc position indication switch, position switch 1E21N103, the position switch actuator rod and associated wires and conduit, from valve 1E21F006. It will also remove the indicating lights for the disc position indication of the valve from main control room panel 1H13P601-21C. The disc position indication switch is used for remote disc position indication during valve testing.

REASON FOR CHANGE, TEST OR EXPERIMENT:

Valve 1E21F006 has a history of the position switch actuator rod causing the valve disc to stick resulting in dual indication in the control room. This frequently requires valve disassembly to correct 50.59 Evaluation summary and conclusions

SAFETY EVALUATION SUMMARY AND CONCLUSIONS:

No changes are being made that affect basic system design functions. The disc position indication switch is used for remote disc position indication during valve testing. The changes do not affect the ability of valve 1E21F006 to perform its function under accident conditions. These changes do not result in a new pathway for the release of radioactive materials and do not affect offsite dose. No assumptions utilized in evaluating consequences of an accident will be altered. No new failure modes are created and there is no increase in previously identified failure modes for equipment important to safety. No assumptions utilized in evaluating the consequences of a malfunction of equipment important to safety will be altered by this change. This modification does not introduce any new failure modes and does not affect equipment other that the check valve and its associated disc position indication. Secondary and indirect effects (Fire protection, fire loading, pipe break, electrical shorts) have been reviewed and no increased probability of failure of equipment important to safety due to these concerns has been identified. These changes will not degrade any important to safety systems, components or structures nor will they degrade or prevent actions described in the SAR analysis. They do not increase the probability of occurrence or increase the consequences of malfunction of equipment important to safety of a different type than previously evaluated in the SAR. The Technical Specifications are not affected and the margin of safety is unchanged.

Evaluation Number: 2001-0031-R00

Document Evaluated: LDC 2000-063
Engineering Report-98-0059, Rev. 1

BRIEF DESCRIPTION OF CHANGE, TEST OR EXPERIMENT:

The change revises Engineering Report GGNS-98-0059, Rev. 1 for the acceptable debris for the ECCS and RCIC suction strainers located in the suppression pool. With revised debris distribution in the suppression pool, the suppression pool cleaning can be deferred. NEDO-32686-A (Reference 4) provides a more realistic method i.e. zone of influence method to calculate expected insulation debris generated for a LOCA event. This method results in a smaller calcium silicate and kaowool debris generation than the fifty percent total drywell insulation used in the present qualification. To evaluate the effects of the new debris quantities on suction strainer head loss, a small scale test consisting of a single section ¼ scale suction strainer was performed at Power Generation Technologies and the results are documented in references 3 and 8. Included in the testing were the effects of increased quantities of sludge, zinc oxide and epoxy coatings. The test results showed that the strainer head loss with the zone of influence insulation debris and including additional large quantities of sludge, zinc oxide and epoxy is bounded by the original test results for head loss that are currently used in the qualification of the strainers. The results of additional sludge utilized in the testing formed the basis for the previous deferment of suppression pool cleaning from RE 10 (Ref. 5).

REASON FOR CHANGE, TEST OR EXPERIMENT:

This 50.59 is to justify adding the small scale test results discussion to the FSAR section 6.2.2.2 so that credit for the additional sludge, zinc oxide and epoxy used in the testing can be taken. Another purpose for this 50.59 is to alleviate the concern that CR GGN-2000-1175 documented with the extension of suppression pool cleaning and quantification of sludge from RF10 to RF11 as not being consistent with previous Safety Evaluation 97-0016, Rev. 1. The results of the testing documented in the engineering reports (Ref. 2, 3, and 8) provide adequate justification for deferring the suppression pool beyond RFI0. A particular sludge generation rate of 300 lbm/year as required by Reference 4 allows for 6.5 years past RF08.

SAFETY EVALUATION SUMMARY AND CONCLUSIONS:

This change which revises the acceptable debris for the ECCS and RCIC suction strainer does not adversely affect the overall ECCS or RCIC systems performance. The acceptable debris including sludge, zinc oxide and epoxy quantities are based on reduced insulation quantities allowed by the NEDO document and the higher quantities of the sludge, zinc oxide and epoxy are confirmed to be acceptable by testing. This change does not cause the ECCS or RCIC systems to be operated outside of their design basis limits, i.e., the environmental conditions, seismic, hydrodynamic and other applicable loads, and system NPSH requirements. The additional quantities of material do not effect the structural integrity of the strainer. The additional material would collect on the strainer after the ECCS/RCIC system has been operating post accident. The suction strainer loading is based on DBA LOP/LOCA, SSE and SRV actuation. System performance evaluations are based on the head loss determined by the original testing. The results of the small scale testing clearly showed a lower strainer screen head loss from the alternate loading quantities. The change will allow larger quantities of sludge, zinc oxide and epoxy to be accommodated during a LOCA event Additionally, the margin of safety as defined in the bases for the Technical Specifications has not been reduced.

Design engineering was requested to evaluate the deferment of suppression pool cleaning until after RF10. Pool cleaning was planned for RF10 to quantify the amount of sludge and validate the amounts used in the testing to establish head loss in the new suction strainers installed in RF09. Subsequent, issuance of the NEDO document by the NRC resulted in a lower total debris generation. Additional strainer testing confirmed that substantial margins exist and that larger quantities of sludge are acceptable. With a 300 lbm debris generation rate (Ref. 4.0), the limit for the acceptable debris will be reached in 6.5 years after the clean up in RF08. After cleaning the suppression pool, the debris generation rate can be determined replacing the 300 lbm/yr rate.

Evaluation Number: 2001-0032-R00

Document Evaluated: LDC 2001-034
Calc. MC-Q1Z51-01003, Rev. 0

BRIEF DESCRIPTION OF CHANGE, TEST OR EXPERIMENT:

FSAR Section 6.4.4.2 reports that the maximum concentration from a release of Freon R-22 from the Control Room chillers would produce a maximum concentration of 0.13% by volume of freon within the Control Room. This disagrees with the value in the supporting calculation. Also, calculation 3.6.42 had assumed Control Room isolation by chlorine detection and used the old control room volume. This calculation was reevaluated using no chlorine isolation and the present control room volume.

REASON FOR CHANGE, TEST OR EXPERIMENT:

Update UFSAR to agree with the supporting calculations.

SAFETY EVALUATION SUMMARY AND CONCLUSIONS:

These supporting calculations are old Bechtel calculations that used the old Control Room volume. The calculation for the release of R-22 from the control room chillers used the quantity from both chillers; which is not required by R. G. 1.78. This calculation also had unclear acceptable oxygen concentration values. This calculation was updated using the new Control Room volume, the update used the release quantity as allowed by R.G. 1.78 and provided clarification as to the acceptance concentration values. The new concentration of oxygen from a release of R-22 from a control room chiller was determined to be about 19.64%; a 1.36% reduction in oxygen which is considered an acceptable oxygen concentration for performing routine duties. According to Patty's oxygen concentration down to 16% will not adversely impair the average individual. Therefore, even though this concentration amount is lower than that presently reported in the UFSAR; it is well within acceptable limits.

The calculation for chlorine assumed control room isolation upon chlorine detection. GGNS does not have a chlorine detector. Therefore, the release of chlorine from a 150 lb. bottle west of the control room was reevaluated without chlorine detection and with the new Control room volume. The determined chlorine concentrations are less than that previously determined. This is due in part to the computer program using a more through atmospheric transport model.

Evaluation Number: 2001-0033-R00 Document Evaluated: LDC-2001-040

BRIEF DESCRIPTION OF CHANGE, TEST OR EXPERIMENT:

The proposed change will
- update latest land use census data [had no impact on sampling locations]
- update a variable used in gaseous effluent radiation monitor setpoint calculations per Corrective Action [CA] 04 to Condition Report CR GGN-1999-0954
- modify LCO 6.3.10 required actions per CA 06 to CR-GGN-2000-1451
- delete a redundant on-site vegetation sampling location, retaining a conservative location
- make editorial changes to improve format of index

REASON FOR CHANGE, TEST OR EXPERIMENT:

The ODCM revision is required to update land use census results, eliminate a redundant sampling location, implement corrective actions for Condition Reports and make editorial changes for clarity.

SAFETY EVALUATION SUMMARY AND CONCLUSIONS:

The proposed changes do not decrease the effectiveness of the Radioactive Effluent Controls Program [RECP] required by Technical Specification [TS] 5.5.4. or the Radiological Environmental Monitoring Program [REMP] required by Technical Requirements Manual [TRM] 7.6.3.2.

The appropriate tables will be updated with the most recent land use census data. No changes to REMP sampling locations resulted from the last land use census.

Changes to a variable in the Gaseous Effluent Radiation Monitor Setpoint calculation are allowed by the current flexibility in the ODCM. SE 97-0054-ROO evaluated a change to the variable. CR-GGN-1999-0954 identified the need to update this variable in the ODCM. Revision 23 will update the variable.

TRM 6.3.10 and the associated section of the ODCM are being modified to incorporate an additional action to TRM 6.3.10. TRM 6.3.10 deals with Radioactive Gaseous Effluent Monitoring Instrumentation. The additional action being added to TRM 6.3.10, Condition C is only clarifying a practice which has always been in place at GGNS. The added action requires the alternate sampling equipment flow rate to be estimated once per 8 hours whenever the alternate sampling equipment is required to collect particulate and iodine samples per TRM 6.3.10, Action C.1. The change to the action of TRM 6.3.10 is required because the alternate sample flow rate was not estimated as documented in CR GGN 2000-1451. Currently, the Senior Reactor Operator must rely on his memory to ensure that the alternate sampling flow rates are estimated. Adding this action to TRM 6.3.10 Condition C provides a positive trigger, which will help reduce the possibility of missing the flow rate estimation requirement.

NRC Branch Technical Position, Environmental Technical Specifications for Nuclear Plants, Rev 1,1979 allows modifications to the REMP after three years of commercial power operation. The proposed change will eliminate one of two onsite vegetation sampling locations and retain he more onservative of the two. ODCM 6.12.1 requires vegetation sampling at an offsite location with the highest anticipated annual average deposition. Use of an onsite vegetation sampling location with a higher annual average deposition will continue to exceed the sampling requirement of ODCM 6.12.1.

The remaining changes are editorial.

Evaluation Number: 2001-0034-R00

Document Evaluated: LDC 2000-002
ER 1998-0426-00-00

BRIEF DESCRIPTION OF CHANGE, TEST OR EXPERIMENT:

UFSAR Section 4.6.1.1.2.4.2.3 describes the flow through the stabilizing valves 1C11-F007A-D. A value of 16 gpm is given as the flow through the stabilizing valve during normal operation (all valves open). Figure 4.6-010 gives specific values for stabilizer valve flow for normal operation, single rod insert and withdrawal and gang mode. ER-GG-1998-0426-000, Rev 0, substitutes a description for the specific values for stabilizer valve flow. This ER also established higher stabilizer valve flows. TSTI 1C11-96-001-0-S temporarily increased stabilizer valve flow to the new values and demonstrated that RC&IS gang mode was effectively restored. This TSTI was performed in April 1998.

REASON FOR CHANGE, TEST OR EXPERIMENT:

The Control Rod Drive Mechanism (CRDM) piston has seals which limit the flow of drive water past the piston during rod movement. In a new CRD, the flow required to insert a rod at 3 inches/sec is approximately 4 gpm and to withdraw a rod is approximately 2 gpm. However, with use the flow past the piston seals increases and the flow required to maintain rod speed at the desired value increases. With a mature CRD population, the average required flow to insert a rod is 5.2 gpm and to withdraw a rod is 4.2 gpm.

The flow through each set of stabilizing valves is set at 4 gpm for insert and 2 gpm for withdrawal. Because of the difference in flow between the CRDs drive flows and the stabilizing valves, the stability of drive pressure is degraded. This condition is especially apparent when moving rods in gang mode. Rods do not reliably move and latch together. Presently, operators generally ignore gang mode because of this problem. This results in additional time required to startup and shutdown the reactor.

SAFETY EVALUATION SUMMARY AND CONCLUSIONS:

No change to Technical Specifications nor the TRM is necessary as stabilizer flows are not specifically addressed in these documents. Increasing the stabilizer valve flows increases drive pressure during CRD motion. For CRDM's with low drive flow requirements, the drive pressure increases above steady state drive pressure. The greatest increase observed during the performance of the TSTI, was 19 psid. The expected drive pressure increase and resulting increase in rod speed are not addressed in these documents either. The expected pressure increase is bounded by currently approved maximum allowable drive pressure of 475 psid (see SE 95-0072-ROO). The increased chance of inadvertent over-notching a rod past its intended position during withdrawal is bounded by the "Rod Withdrawal Error (RWE) analysis (UFSAR section 15.4.2). Engineering experience and estimates based on proposed change indicate the worst case over-notching will be below the Rod Withdrawal Limiter (RWL) allowed range of 4 notches below the HPSP which is based on the RWE analysis (UFSAR section 15.4.2).

The control rods will possess a higher stroking speed after the change has been made; however, control rod drive speed is not an analysis basis parameter. Rod insertion capability has been adequately addressed during and after the time the changes are made. After the adjustments are complete, but before the test conditions are established to test the changes, the rods will still insert via normal drive mode just as before though at a slightly increased speed. During the time the changes are being made, should plant conditions require rod insertion, there exists the potential that the stabilizing valves would be disconnected and closed and therefore unable to maintain drive pressure (especially in gang mode). Guidance is provided to compensate for this potential sluggish insertion capability (normal drive mode only, scram is unaffected) by increasing drive pressure per existing plant procedures. Further, UFSAR section 4.6.2.3.2.2.8 addresses the worst case scenario of the pressure control valve being fully closed or having total flow blockage while withdrawing a rod with the reactor a 0 psig. This would result in the drive pressure increasing to the CRD Pump shutoff pressure of approximately 2000 psig. The nominal drive speed of 3 in/sec would increase to 7 in/sec. This would completely bound the proposed test condition in the ERT: hence there is no increased probability or consequences of any accident or malfunction previously analyzed. The scram function of the CRD system is unaffected by this change. The Control Rod Drop Accident (CRDA) is also unaffected by this change. No margin of safety is affected by this change since there is no affect on MCPR Safety Limit, plastic strain limit, or radiological dose limits. Hence, no unreviewed safety question is created by changing the CRD stabilizer valve flow settings.

Evaluation Number: 2001-0035 Document Evaluated: ER 2001-0093-00-00

BRIEF DESCRIPTION OF CHANGE, TEST OR EXPERIMENT:

ER 2001-0093 reviewed the performance history of 1P41C00IB to determine if the inspection/overhaul of the pump by repetitive task 13529 could be extended beyond 4/15/2001 to the end of March 2002. MNCR 94-0036 established the six year frequency for the SSW pump inspection/overhauls based on a previous eight year span of operation before its failure in 1994.

REASON FOR CHANGE, TEST OR EXPERIMENT:

It was requested to evaluate extending the due date for the inspection of SSW B to beyond RF11 to make better use of plant resources. MNCR 94-0036 and Safety Evaluation 94-0091-ROI set the interval of the inspection of the SSW pump to evaluate the pump with respect to corrosion and this evaluation will allow an exception to the six year inspection requirement for SSW B. The failure in 1994 of the B SSW pump was due to the line shaft coupling bolts and washers corroding allowing the shaft to become loose and allowing the impellers to impact the pump bowls. To prevent that type of failure from occurring, both SSW A&B pumps were modified in 1995 by changing the line shaft coupling bolts from carbon steel to Monel and by changing the lock washer to stainless steel.

SAFETY EVALUATION SUMMARY AND CONCLUSIONS:

This 50.59 has reviewed the six year inspection frequency established by MNCR 94-0036 and SE 94-0091-R0I and has evaluated an extension for the B SSW pump to 3/30/2002. The review of the SSW B performance history does not give any trends that would indicate that extending the inspection frequency would degrade the system performance or its ability to perform its safety function. The IST pump flow trend is steady and not indicating any signs of changing pump performance. Although the vibration data for the SSW pump is collected from the motor not the pump impellers, there is no changing trend in the data to indicate degrading conditions. The modifications made previously to upgrade the bolting and washer material has removed the failure mechanism that contributed to the "B" pump failure in 1994. The extension of repetitive task 13529 past its due date of 4/15/2001 to the end of March 2002 does not require a change to the GGNS Technical Specification, will not increase the consequences nor the probability of a malfunction of equipment important to safety previously evaluated in the SAR, will not create the possibility for an accident of a different type than any previously evaluated in the SAR, will not create the possibility for a malfunction of equipment important to safety of a different type than any previously evaluated in the SAR, and will not reduce the margin of safety as defined in the basis for any Technical Specification.

Evaluation Number: 2001-0036-R00 Document Evaluated: LDC-2001-047

BRIEF DESCRIPTION OF CHANGE, TEST OR EXPERIMENT:

ER 2001-0019-00-00 provides an increased setpoint for the feedwater control (C34) high water level instrumentation as well as an increased setpoint for the existing C34 time delay relays. The increase in the high water level setpoint results in the creation of a new vessel water level (Level 9).

REASON FOR CHANGE, TEST OR EXPERIMENT:

There are two functions associated with a high vessel water level. The reactor protection function, which receives its input from the B21 level instrumentation, provides an RPS scram to reduce reactor power. The turbine protection function, which receives its input from the C34 level instrumentation, provides a main turbine trip and feedwater pump trip to prevent gross moisture carryover and overfilling the vessel.

CR GGN-2000-1810 identified that the two sets of high water level instrumentation (C34&B21) have the same Nominal Trip Setpoint (Level 8). However, application of the required drift and uncertainty characteristics for each set of instrumentation in its conservative direction results in a configuration where the non-safety related main turbine/feedpump trips may occur prior to or without the corresponding safety related RPS scram. The Level 8 RPS scram is credited in the mitigation of high water level events including the limiting feedwater controller failure-maximum demand. The increased setpoints for the C34 level instrumentation and the time delay relays ensure that the Level 8 RPS scram will be initiated and the turbine/feedpump trips will occur no sooner than this scram.

SAFETY EVALUATION SUMMARY AND CONCLUSIONS:

The proposed setpoint changes are within the capabilities of the current equipment and no system modifications or equipment replacement is necessary. The system will continue to provide the same trips when high water level is reached and will impose no additional challenges to any other equipment. The turbine protection and vessel overfill design functions will continue to be met. The delay in the main turbine trip following a high water level event will delay the vessel pressurization due to main turbine stop and control valve fast closure, which will lead to less severe thermal and pressurization transients. The consequences of these events will continue to be bounded by current analyses.

As such, the probabilities and consequences of previously evaluated accidents or equipment malfunctions are not increased. No new accidents or malfunctions will be introduced and the margins of safety for the Technical Specifications will not be reduced. Therefore, the proposed setpoint changes do not result in an Unreviewed Safety Question.

Evaluation Number: 2001-037-R00 Document Evaluated: LDC 2001-053

BRIEF DESCRIPTION OF CHANGE, TEST OR EXPERIMENT:

Presently, the UFSAR states that core verification is done "when refueling is completed." LDC 2001-053 has been prepared to clarify that core verification may begin in areas of the core where fuel is in its final position for restart while core alterations continue elsewhere as long as all fuel is verified prior to vessel reassembly. This evaluation does not address any particular type of equipment to be used for verification. Such equipment must be evaluated separately per plant procedures and processes if it differs from normal viewing equipment attached to the refuel platform by design.

REASON FOR CHANGE, TEST OR EXPERIMENT:

This change is needed so that core verification may be conducted in one area of the core simultaneously with core alterations continuing in other areas. This will allow improvements in outage efficiency without detracting from the safety, effectiveness, or intent of the core verification.

SAFETY EVALUATION SUMMARY AND CONCLUSIONS:

The timing and method of core verification is not described in the Technical Specifications. The clarification provided by this LDC does not impact the probability or results of any accident analysis. It does not remove the requirement to conduct a core verification, it only allows verification to be done in parallel with other activities. It does not change the characteristics of the core which must be verified (correct orientation, seating, and location.) The change does not modify the fuel handling equipment or any fuel handling procedures. No new types of events are created by this change and no Tech. Spec. Basis margins of safety are affected. Therefore, this change does not present an unreviewed safety question.

Evaluation Number: 2001-038-R00 Document Evaluated: LDC 2001-015

BRIEF DESCRIPTION OF CHANGE, TEST OR EXPERIMENT:

ER 2001-0029-00 installs a door between the Decontamination area (OC101) and the H. P. Counting Room (OC110) on elevation 93'-0" of the Control Building.

REASON FOR CHANGE, TEST OR EXPERIMENT:

INPO has identified that there should be a door between these two areas in order to improve contamination control.

SAFETY EVALUATION SUMMARY AND CONCLUSIONS:

It is concluded that the addition of a door on EL 93' of the control building will not increase the probability of occurrence of any accidents previously evaluated in the SAR. The modification does not result in a new pathway for the release of radioactive material and does not affect onsite dose in a way that restricts access to vital areas or impedes mitigating actions. The modification will have no affect on any equipment considered important to safety and does not cause any system or components to be operated outside design limits. No new failure modes are created and there is no increase in previously identified failure modes for equipment which is considered important to safety. The changes will not compromise the function of any safety related system or prevent safe reactor shutdown since the changes do not create any new interface with equipment designed, or cause equipment important to safety to operate outside of design requirements. System analysis has shown that failure of the door will not compromise any safety related system or component and will not prevent safe reactor shutdown. The changes will not affect the mode of operation of any equipment important to safety or Technical Specification associated equipment, will not create a system or operating condition such that a Tech Spec limiting condition for operation (LCO) or surveillance requirement is no longer adequate. Nor will it bypass or invalidate automatic actuation features required to be operable by Tech Specs.

Evaluation Number: 2001-039-R00

Document Evaluated: ER-2001-0113-00-00
Temp. Alt. 2001-12

BRIEF DESCRIPTION OF CHANGE, TEST OR EXPERIMENT:

ER 2001-0113-00-00 provides an evaluation to support temporary alteration of the Standby Liquid Control (SLC) system for system operation with a single operable pulsation dampener (i.e., damper or accumulator).

REASON FOR CHANGE, TEST OR EXPERIMENT:

The function of the SLC system shall be capable of injecting the neutron absorber into the reactor via High Pressure Core Spray (HPCS) spargers with both SLC pumps running simultaneously. To assure that the relief valve set pressure will not be exceeded during pump operation with the pump discharge path open, a pulsation dampener is installed on the discharge piping of each SLC pump.

CR-GGN-2001-0565 documented the failure of the "A" SLC system pulsation dampener with no spare parts available. This evaluation is justification for a Temporary Alteration to the SLC system which would result in system operation with only one operable pulsation dampener.

SAFETY EVALUATION SUMMARY AND CONCLUSIONS:

Proper SLC system operation with the proposed temporary alteration is within the capabilities of the current equipment. The guidance for temporarily modifying the failed accumulator assures complete system operability. The system will continue to be capable of injecting the neutron absorber into the reactor and will impose no additional challenges to any other equipment. The SLC design functions will continue to be met. The operation of the SLC system with one pulsation dampener (or accumulator) has been evaluated in Ref. 2. The calculation (i.e., Ref. 2) determined the maximum pressure which would occur in the event of a failure of a single SLC system pulsation dampener during the most limiting design conditions to verify that the system relief valves would not lift and result in short cycling or reduced capability of the SLC system. The calculation results verify that a single dampener failure would not result in system pressures that challenge the relief valve setpoint. The consequences of these events will continue to be bounded by current analyses.

As such, the probabilities and consequences of previously evaluated accidents or equipment malfunctions are not increased. No new accidents or malfunctions will be introduced and the margins of safety for the Technical Specifications will not be reduced. Therefore, the proposed temporary alteration of the SLC system does not result in an Unreviewed Safety Question.

Evaluation Number: 2001-040-R00 Document Evaluated: LDC 2001-066

BRIEF DESCRIPTION OF CHANGE, TEST OR EXPERIMENT:

The proposed change to the UFSAR is the modifying the requirement to perform complete inspections of all three Condensers during each refueling outage. Full inspections of the Condensers involves inspection of the condenser shell, waterboxes, hotwell, and leak checks on the tubes. The proposed change will make these inspections "train" based, i.e. only one condenser will have a full inspection, the other two condensers will have a visual inspection. The visual inspection will identify loose lagging and other potentially degraded components. These items will be evaluated and corrected in accordance with our corrective action program. The full inspection for each condenser will have a 3R (4.5 year) frequency.

REASON FOR CHANGE, TEST OR EXPERIMENT:

It has been determined that it is no longer necessary to perform a full inspection of all the condensers during each refueling outage. This change is possible due to a lack of significant degradation found during past outage inspections, changes in operating philosophy in the industry, and acceptable risk.

SAFETY EVALUATION SUMMARY AND CONCLUSIONS:

The FSAR describes Tests and Inspections for the Main Condenser in Section 10.4.1.4. These include full inspections on all three Main Condensers (High Pressure, Intermediate Pressure, and Low Pressure) each refueling outage. The proposed change will cause each condenser to have a full detailed inspection on a 3R (every third refueling outage) schedule with a visual inspection of all the condensers to be performed each outage. The full inspection includes access to the bootseal and hood area. This area will not be inspected during the visual inspections. This change does not modify any equipment or the manner in which the equipment is operated. Accidents previously evaluated in the SAR do not take credit for the operation of the main condenser *(i.e.,* active function) for mitigation. The control rod drop accident described in USAR Section 15.4.9 does credit the condenser structure *(i.e.,* passive function) for mitigating the radiological release by a limited holdup and plateout of the radionuclides (USAR para. 15.4.9.5.2). However, this passive function is neither maintained nor verified by the condenser inspections described in USAR Section 10.4.1.4. Thus, changing the frequency of the major inspections of the main condenser will not increase the probability of occurrence or the consequences of an accident previously evaluated in the SAR.

No changes are being made to the equipment or how the equipment operates. Therefore, no increase in the probability of occurrence of a malfunction of equipment important to safety, nor the consequences from a malfunction of equipment important to safety previously evaluated in

the SAR, nor the possibility of an accident of a different type than any previously evaluated in the SAR, nor the possibility of a malfunction of equipment important to safety of a different type than any previously evaluated in the SAR will occur.

The main condenser is not mentioned in the basis for any Technical Specification except in the condition where it is unavailable. Therefore, changing the frequency of condenser inspections will not affect the margin of safety as defined in the basis for any Technical Specification.

Evaluation Number: 2001-041-R00 Document Evaluated: LDC 2001-059

BRIEF DESCRIPTION OF CHANGE, TEST OR EXPERIMENT:

Extend specified TRM surveillance frequencies from monthly to quarterly for selected trip units based on evaluation of failure history utilizing Weibull Data Analysis and Risk Assessment data listed as first item under reference. The change is limited to monthly functional test TRM surveillance frequencies for Rosemont trip unit models 510DU2, 510DU7, 7I0DU and combination of models 510DU2, 510DU7, and 7I0DU. The referenced analysis of historical data and risk assessment shows no significant impact in the failure rate between surveillances by extension of these surveillance frequencies. The 18-month testing and calibration surveillances are unaffected by these changes.

REASON FOR CHANGE, TEST OR EXPERIMENT:

To reduce the number of functional tests or surveillances on highly reliable devices based on historical data and risk assessment which will reduce the risk of plant challenges and transients.

SAFETY EVALUATION SUMMARY AND CONCLUSIONS:

This change does not increase risk to the public health and safety. These surveillances were removed from the Technical Specification at various times because they did not meet the below listed selection criteria of I0CFR50.36:

(A) *Criterion 1.* Installed instrumentation that is used to detect, and indicate in the control room, a significant abnormal degradation of the reactor coolant pressure boundary.

(B) *Criterion 2.* A process variable, design feature, or operating restriction that is an initial condition of a design basis accident or transient analysis that either assumes the failure of or presents a challenge to the integrity of a fission product barrier.

(C) *Criterion 3.* A structure, system, or component that is part of the primary success path and which functions or actuates to mitigate a design basis accident or transient that either assumes the failure of or presents a challenge to the integrity of a fission product barrier.

(D) *Criterion 4.* A structure, system, or component which operating experience or probabilistic risk assessment has shown to be significant to public health and safety.

The bases of this change is that extension of the surveillance frequencies for the selected TRM surveillances will not adversely effect the failure rate of the trip units.

The subject equipment, Rosemount trip unit models 510DU2, 510DU7, 710DU, and combination of models 710DU, 510DU7, 510DU2, has been found to be highly reliable. This change only extends the surveillance interval and based on analysis will not credibly impact failure of these highly reliable instruments. This conclusion is based on analysis of the failure detection history.

Additionally, a risk assessment for the devices was also performed to verify no significant impact to public health and safety. These evaluations are documented in GGNS Engineering Report No. GGNS-01-0008, Rev. 1.

CCE-2001-0001

Commitment Number: 16361

Source Document Number: GNRO-91/00169
LER 91-005-01

COMMITMENT CHANGE TITLE:

Delete subsequent to RFO5, the Siemens breakers will be inspected and cleaned every six years

COMMITMENT DESCRIPTION:

Mississippi Power and Light, who has responsibility for performing maintenance on switchyard equipment, has agreed to inspect and clean the Siemens breakers during Refueling Outage 5. Subsequent to RFO5, the Siemens breakers will be inspected and cleaned every six years.

JUSTIFICATION FOR CHANGE OR DELETION:

LER 91-005-01 attributes an electrical fault on generator output breaker J5232, which caused a plant scram to a high particle content in the compressor oil sample that allowed tracking to ground. In 1996, the generator output breakers, J5228 and J5232, were replaced with a Mitsubishi puffer style breaker which is not susceptible to this degradation mechanism. Furthermore, the remaining Siemens breakers, J5216 and J5224, are inspected and maintained according to Entergy MS Standard SD 1203 which includes performing compressor maintenance. Commitment A-4434 is for inspecting, maintaining, and testing all 500kV circuit breakers.

CCE-2001-0002

Commitment Number: P-23912 Source Document Number: AECM-82/0012

COMMITMENT CHANGE TITLE:

Independent Verifier Qualifications

COMMITMENT DESCRIPTION:

Independent verifiers shall be classified to a position higher than Trainee. This position will be in accordance with certification per ANSI 18.1-1971. Independent verifiers shall be of Journeyman level for the their respective maintenance disciplines; e.g., engineer, a person certified to perform duties in an engineering discipline, a qualified health physicist or a chemist per the above noted ANSI requirements. Operations personnel shall be either licensed operators or qualified Nuclear Operator B's for the system or systems being independently verified.

JUSTIFICATION FOR CHANGE OR DELETION:

The intent of this commitment is that independent verifiers must be task qualified and certified to perform actions required per ANSI standard. Removing the classification requirements for maintenance personnel does not remove the requirement that maintenance personnel certified to perform the independent verification task must be qualified and certified to do so per the ANSI standard.

Independent verifiers shall be certified per ANSI 18.1-1971 and task qualified by their respective maintenance disciplines; e.g., engineer, a person certified to perform duties in an engineering discipline, a qualified health physicist or a chemist per the above noted ANSI requirements to perform the actions of an independent verifier Operations personnel shall be either licensed operators or qualified Nuclear Operator B's for the system or systems being independently verified.

CCE-2001-0003

Commitment Number: 23826 Source Document Number: 08-S-05-2

COMMITMENT CHANGE TITLE:

Shipping Radioactive Materials.

COMMITMENT DESCRIPTION:

In January of 1986 the Southeast Compact Commission for Low Level Waste Management established a ban on the export of radioactive waste out of the southeast region for compact party states.

JUSTIFICATION FOR CHANGE OR DELETION:

The Southeast Compact Commission for Low Level Waste Management has been abolished.