Entergy

Entergy Operations, Inc.
Waterloo Road
P.O. Box 756
Port Gibson, MS 39150
Tel 601 437 6470

Jerry C. Roberts
Director
Nuclear Safety Assurance

June 1, 2000

U.S. Nuclear Regulatory Commission
Washington, D.C. 20555

Attention: Document Control Desk

Subject: Grand Gulf Nuclear Station
Unit 1
Docket No. 50-416
License No. NPF-29
Report of 10CFR50.59 Safety Evaluations and Commitment Changes
– June 1, 1998 through December 31, 1999

GNRO-2000/00041

Gentlemen:

In accordance with the requirements of 10CFR50.59(b), Entergy Operations, Inc. is reporting those changes, tests, and experiments, performed at Grand Gulf Nuclear Station during the period of June 1, 1998 through December 31, 1999. A summary of the safety evaluation for the changes, tests, and experiments is contained in the attachment. Also attached are the summaries of commitment changes made in accordance with the guidelines of NEI 95-07 for the same period. If further information is required, please contact this office.

Yours truly,

JCR/ACG/amt
attachment: AG. Table of Contents of 10CFR50.59 Safety Evaluations and Commitment Change Evaluation Summaries
cc: (See Next Page)

G200529

cc: Ms. J. L. Dixon-Herrity, GGNS Senior Resident (w/a)
Mr. D. E. Levanway (Wise Carter)
Mr. L. J. Smith (Wise Carter) (w/a)
Mr. N. S. Reynolds (w/a)
Mr. H. L. Thomas (w/o)

Mr. E. W. Merschoff (w/a)
Regional Administrator
U.S. Nuclear Regulatory Commission
Region IV
611 Ryan Plaza Drive, Suite 400
Arlington, TX 76011

Mr. S. P. Sekerak, NRR/DLPM/PD IV-I (W/2)
U.S. Nuclear Regulatory Commission
One White Flint North, Mail Stop 04-D3
11555 Rockville Pike
Rockville, MD 20852-2378

Serial Number: 98-078-NSRA Document Evaluated: LDC 98-053

BRIEF DESCRIPTION OF CHANGE, TEST OR EXPERIMENT:

This change provides an allowance to UFSAR/TRM TR3.3.6.1 for intermittently unisolating Reactor Recirculation Sample Isolation Valves (B33-F019-B and B33-F020-A) with inoperable Main Steam Line Radiation Monitor (MSLRM) isolation instrumentation. The current ACTION for inoperable MSLRM instrumentation is to isolate the affected penetration flow path(s). The penetration flow paths associated with UFSAR/TRM TR3.3.6.1 ACTION D.1 (with inoperable MSLRM instrumentation) are the Reactor Recirculation Sample Isolation Valves (B33-F-19-B and B33-F020-A) and the Main Condenser Mechanical Vacuum Pumps. These valves auto-close upon exceeding a Main Steam Line Radiation setpoint of ≈ 3X full power background. The proposed change adds a "NOTE" to UFSAR/TRM TR3.3.6.1 ACTIONS section to allow opening of these valves under administrative control. This change is similar to an allowance provided in Grand Gulf Technical Specifications which allows inoperable Containment and Drywell Isolation valves to be opened intermittently under administrative controls. The administrative controls used for inoperable Drywell isolation valves will be used when the valves are required to be open.

REASON FOR CHANGE, TEST OR EXPERIMENT:

The current REQUIRED ACTION specified in UFSAR/TRM TR3.3.6.1 requires isolation of affected penetration flow path(s) (Recirculation Sampling Valves (B33-F019-B and B33-F020-A) or condenser mechanical vacuum pump) if the Main Steam Line Radiation Monitor (MSLRM) isolation instrumentation is inoperable. Verbatim compliance with this UFSAR/TRM REQUIRED ACTION (with inoperative MSLRM instrumentation) does not provide an allowance to open these valves to obtain samples of reactor coolant either for routine sampling or during emergency conditions. Both samples could provide indication of gross fuel element failure.

SAFETY EVALUATION SUMMARY AND CONCLUSIONS:

Intermittent opening and closing of valves B33-F019-B and B33-F020-A with inoperable MSLRM instrumentation will not prevent closing these valves either manually or by other automatic means. The auto-close safety function, at Reactor Vessel Water Level-Low Low, Level 2, for these valves ensure that steam and water releases to the Drywell are channeled to the suppression pool to maintain the pressure suppression function of the drywell. This pressure suppression function is unaffected by this change. During the time that the valves are open, auto-closure (safety function) of the valves on a Reactor Vessel Water Level-Low Low, Level 2 at –41.6 inches and on Manual Initiation of Main Steam Line Isolation (NSSS push buttons) will be maintained. Opening of valves B33-F019B and B33-F020-A with inoperable isolation instrumentation under administrative controls will require alternative monitoring of the MSLRM's (if operable) or other radiation monitors (such as Offgas pre-treatment) for indication of a fuel element failure. All operator actions (closure of the valves B33-F019-B and B33-F020-A) can be taken from the Control Room by turning two hand switches on Panel 1H13-P680. B33-F020-A can be operated from Panel 1H22-295 in the 119' elevation of the Auxiliary Building. Although an operator action is credited in this evaluation, non-action by the operator is (i.e., not shutting the valves) also assumed in this evaluation. Leaving the valves open under administrative control does not prevent or cause any accidents nor does it mitigate any accidents. An allowance for opening of inoperable Drywell or Containment isolation valves currently exists in the current Grand Gulf Technical Specifications and, based on this evaluation, is an acceptable practice for the valves in question with inoperable MSLRM isolation instrumentation as long as the administrative controls for inoperable Drywell isolation valves are followed.

Serial Number: 98-079-NPE

Document Evaluated: ER 97/0588-00-00, MC-Q1P81-97034, SCN 98/0001 to GGNS-MS-39,Rev. 0, LDC 97-085

BRIEF DESCRIPTION OF CHANGE, TEST OR EXPERIMENT:

This changes does not make any physical changes or alterations to any Plant Systems, Structures or Components. The change is to revise the design documentation and UFSAR to reflect the heat rejection capability of the Division III jacket water heat exchanger with the design fouling factor applied to the tube side of the heat exchanger.

REASON FOR CHANGE, TEST OR EXPERIMENT:

Based upon the improperly applied fouling factor, the design heat rejection capability of the HPCS jacket water heat exchanger is overstated in the design documents and the UFSAR. The proposed change is to properly apply the fouling factor to the tube side of the Division III Diesel Generator jacket water heat exchanger design documentation and correct the design heat rejection capability as listed in design output documents.

It is concluded that non-conservative application of the fouling factor to shell side of the Division III Diesel Generator jacket water heat exchanger over estimates the design heat rejection capability of the heat exchanger. The fouling factor should be applied to the tube side of the Division III Diesel Generator jacket water heat exchanger.

SAFETY EVALUATION SUMMARY AND CONCLUSIONS:

The design heat removal capability which results from applying the fouling factor to the tube side of the Division III Diesel Generator jacket water heat exchanger is sufficient to satisfy its design basis requirements, does not prevent the Division III Diesel Generator jacket water heat exchanger or the Division III Diesel Generator from performing their design safety function nor does it require the Division III Diesel Generator jacket water heat exchanger or the Division III Diesel Generator to be operated in an abnormal manner.

This change makes no physical changes to the plant; it corrects a misapplication of the design fouling factor in the determination of the design heat rejection capability of the Division III Diesel Generator jacket water heat exchanger.

The GE Specification 21A9236, Revision 5 for the HPCS Diesel engine water jacket heat exchanger specifies that the heat exchanger shall have the capacity required to reject 110% of the engine rated name plate power with a fouling factor of 0.002. This condition is still satisfied as shown in Calculation MC-Q1P81-97034, Revision 0.

The design heat rejection capability of the HPCS Diesel engine jacket water heat exchanger as reported in VPF-3636-013 was used as an input to Calculation MC-Q1P41-86054, Revision 0, Standby Service Water Ultimate Heat Sink Performance. By properly applying the fouling factor to the tube side of the HPCS Diesel engine jacket water heat exchanger, it is realized that there is additional conservatism in Calculation MC-Q1P41-86054, Revision 0 because of the Calculation MC-Q1P41-86054, Revision 0 assumed the cooling load duty of the HPCS Diesel engine jacket water heat exchanger to be the overstated jacket water heat exchanger design capability. Therefore, based on this discussion, Calculation MC-Q1P41-86054, Standby Service Water Ultimate Heat Sink Performance Revision 0 is conservative and bounding.

The Division III Diesel Generator was qualification tested at GGNS similar to the testing described in the amended Topical Report NEDO 10905-3 and documented in AECM 82/152, dated April 14, 1982. The results of the qualification test demonstrated that the HPCS power supply can meet all the design requirements at 110% of engine rated name plate load without exceeding the Diesel engine manufacturer's design limits. Although this testing was performed with a clean jacket water heat exchanger, the Division III Diesel Generator water jacket heat exchanger thermal performance is monitored by a periodic testing program to ensure the heat exchanger continues satisfy its design basis cooling duty load requirement.

Since the HPCS Diesel engine jacket water heat exchanger has sufficient heat rejection capability with the fouling factor applied to the tube side of the heat exchanger to satisfy its design basis heat rejection requirement and still satisfies the OEM heat exchanger sizing criteria for the engine, NPE has concluded that this change does not adversely impact or alter the HPCS Diesel engine systems ability to perform its safety function. There are no seismic effects, environmental qualifications and no adverse impact on interfacing safety related systems. SSW is an interfacing safety related system; however, there is no adverse impact on SSW since the capability of the Ultimate Heat Sink is based on the design heat rejection capability of the serviced heat exchangers.

The acceptance criteria for the ECCS systems which includes the HPCS system as stated in the Bases for the Technical Specifications are a) Maximum fuel element cladding temperature is ≤2200°F; b) Maximum cladding oxidation is ≤ 0.17 times the total cladding thickness before oxidation; c) Maximum hydrogen generation from zirconium water reaction is ≤ 0.01 times the hypothetical amount that would be generated if all of the metal in the cladding surrounding the fuel, excluding the cladding surrounding the plenum volume were to react; d) the core is maintained in a coolable geometry; and e) Adequate long term cooling capability is maintained. The application of the fouling factor to the tube side of the jacket water heat exchanger does not cause any of these acceptance limits to be approached or exceeded; therefore, there is no reduction in the margin of safety.

The application of the design fouling factor and heat rejection capability of the Division III jacket water heat exchanger is not addressed by any of the Technical Specification Bases; therefore, NPE has concluded that the application of the fouling factor to the tube side of the HPCS Diesel engine jacket water heat exchanger and the resultant heat rejection capability of the heat exchanger does not reduce a margin of safety as defined in the basis for any Technical Specification. Therefore, based on the above discussion, NPE had concluded that this change does not constitute an unreviewed safety question.

Serial Number: 98-080-NPE Document Evaluated: ER 96/0487-00-00

BRIEF DESCRIPTION OF CHANGE, TEST OR EXPERIMENT:

This ER (ER 96/0487-00-00) is part of an overall design objective to install new pumps and motors for the Plant Service Water (PSW) Radial Well System (P47 System). The new pumps and motors will be used as replacements for the existing pumps and motors for Radial Well #3. The new stainless steel pumps (existing well #3 pumps are carbon steel) will be oil lubricated enclosed line-shaft vertical pumps rated at 5,000 gpm and discharge head of 360 ft of water. The existing 500 hp pump motors will be replaced with new 600 hp motors. As part of the modification a lubricant reservoir and associated equipment will be installed to provide lubrication for the enclosed line shafts.

Specifically, this ER (ER 96/0487-00-00) replaces the pumps and motors in Radial Well #3 (pumps NSP47C001C & D). This ER can be worked during any operational condition as long as sufficient Plant Service Water capacity is available to support the plant condition which exists during the period of implementation, which is in agreement with UFSAR Section 9.2.10.2, System Description, which states: "During normal operations, as many wells and pumps as required will be operating to meet the plant demand."

The new pumps will mount in the existing pump support structure and will connect to the existing piping with no major piping modifications required. Pump lubrication, using biodegradable oil, will be supplied by gravity from 140 gallon tanks (M-929.0-NS-1.1-3-0) mounted in the individual well houses. The tanks are constructed to the standard requirements of the manufacturer. There will be an interlock between the lube oil tank and pump trip circuit to trip the pump on low-low lube oil tank level. The equipment is being procured in accordance with Specification GGNS-M-929.0.

LDC 98-014 makes the following changes:

- UFSAR Section 2.4.13.1.3.1 – The total dynamic head of the new Radial Well pumps is now 360 ft at 5000 gpm, however, the total dynamic head of the pumps is not germane to the discussion in this section of the UFSAR, therefore, this information is being deleted. This same information is contained in Table 9.2-13.

- UFSAR Figure 8.1-1 – The horsepower for the new pump motors for Radial Well #3 is 600 hp. (Note that this Figure is Main Single Line Diagram E-0001 and will be updated automatically by Configuration Management.)

- UFSAR Table 9.2-13 – The information for the PSW pumps and lube oil system in Table 9.2-13 is being revised to reflect the new equipment for Radial Well #3.

- UFSAR Figure 9.2-27, Sheet 1- The base drawing, M-0052A, for this UFSAR Figure was revised to show the installation of the new Radial Well #3 pumps and associated equipment, therefore, the UFSAR Figure was revised accordingly. (Note that this drawing is P&ID M-0052A and will be updated automatically by Configuration Management.

REASON FOR CHANGE, TEST OR EXPERIMENT:

There has been an apparent decrease in efficiency and reliability for the PSW Radial Well #3 pumps caused by extensive pump column wear, impeller obsolescence, well #3 pump prelube abandonment, and system operational characteristics.

SAFETY EVALUATION SUMMARY AND CONCLUSIONS:

The intended purpose for the installation of the new pumps and motors is to increase the reliability, capability and availability of the PSW system. This change does not degrade below the current design basis 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 cause a greater reliance to be placed on any specific system, structure or component to perform a safety function. A failure of one of the new PSW Radial Well pumps or motors will have the same result as the failure of the existing PSW Radial Well pumps and motors. The loss of PSW and flooding as a result of a break in the PSW header in the Auxiliary Building have been analyzed in the UFSAR. The installation or failure of the new pumps or motors will not increase the probability or consequences of these analyzed failures. No increase of either the offsite or the onsite radiation dose would result because of a failure of a new pump or motor.

When the new pumps have been installed, the capability will exist to increase the operating pressure of the PSW system. This ER does not authorize changes to the operating pressure of the PSW system as described in system operating instruction 04-1-01-P44-1. Such an increase in operating pressure would adversely affect the leakage rate and water accumulation rate in the Auxiliary Building in the case of a postulated moderate energy crack in the 36" PSW line. GGCR1998-0701-00 will correct the leakage and accumulation rates currently in UFSAR Section 3C.4 and 3C.4.2.1 to reflect the normal PSW system pressure for the 36" PSW line. This condition report documents that the additional leakage due to the normal operating pressure of the PSW system would still be acceptable based on current methodology and reasoning used in the GGNS UFSAR. There would be no increase in probability of a leak or the consequences of a leak as a result of this ER, since the system will still be operated within the constraints of the system operating instruction and the operability evaluation for GGCR1998-0701-00. The flooding of the control building is bounded by the circulating water line break (Note 5 of drawing M-1575, Rev. 0) and the flooding in RHR Room C is bounded by the RHR Pump C suction line (see UFSAR Section 3C.4.2.6).

The installation of the new pumps and motors has been analyzed for its impact on the Radial Well Pumphouse HVAC system's effectiveness and found not to be adversely impacted. Because the new motors are more efficient than the existing motors, the heat load in the pumphouse will be less when the new pump motors are in operation than with the existing motors.

An evaluation was performed to determine the environmental impact of the addition of the new lube oil systems added to support the new Radial Well pumps. The evaluation concluded that the design of the new PSW pumps and the associated lube-oil system minimizes any potentially negative environmental impacts. The lube-oil reservoir, piping, and tubing are designed to preclude any direct leakage to the environment, and the pump design restricts all but incidental oil migration to the pump caisson rather than the pump effluent and downstream system piping. The specified oil is "environmental friendly" such that any trace amounts that might ultimately reach the plant piping or the river will bio-degrade without impact to the environment. The chemistry department performed a control room habitability screen for this lubricating oil per procedure 01-S-08-18, GGNS Chemical Control Program, and documented its acceptability.

This change does not adversely affect the overall performance or reliability of a safety system in a manner that could lead to an accident occurring. This change does not cause a safety system to be operated outside of its design basis limits. The new PSW Radial Well pumps and motors cannot affect any system interface in a way that could lead to an accident. The new PSW Radial Well pumps and motors will not result in degradation of safety systems. The change is intended to improve the reliability, capability and availability of the PSW system by providing a mechanism to reduce the possibility of system unavailability. Additionally, the margin of safety as defined in the bases for the Technical Specifications has not been reduced.

Serial Number: 98-081-PSE Document Evaluated: CEWO No. 98-0008

BRIEF DESCRIPTION OF CHANGE, TEST OR EXPERIMENT:

ER 96/0984-01-00 installed flow transmitter 1N21-FT-N061 in the Zinc Injection Skid in order to provide a remote flow signal to the PDS computer system. This flow signal is needed to allow correction of feedwater flow input into the reactor heat balance calculation to properly account for diversion of feedwater flow downstream of the Leading Edge Flow Meter (LEFM). CEWO 98-0008 provides the necessary software changes to the Plant Data System (PDS) to correct feedwater flow as measured by the LEFM using the flow signal from FT-N061.

Additional software will also be added to the processing of the feedwater flow information in PDS so that appropriate checks are present to ensure a conservative power calculation in the event of instrument failure. In particular, a check on excessive feedwater flow (Hi-Hi) will be done so that a zinc skid flow which is higher than the expected upper range will result in disabling of the LEFM input. This causes feedwater flow to default to the higher venturi readings, which is conservative in that is results in an indicated CTP which is too high, causing the plant operators to reduce power until indication is below rated. Also, should the zinc injection flow instrumentation fail downscale or the zinc injection skid be shutdown, it is possible for the zinc skid flow value to become negative. In this event, a minimum value of zero will be substituted, which ensures a conservative CTP calculation regardless of the amount of flow actually being diverted by the skid.

Physical installation of the zinc skid flow transmitter is addressed separately in References 2 and 3.

REASON FOR CHANGE, TEST OR EXPERIMENT:

After installation and testing of the zinc injection skid, it was realized that the design had a detrimental (although conservative) effect on the reactor heat balance calculation. A small amount of feedwater flow is diverted downstream of the LEFM and passed through the skid where zinc injection takes place, and then returned to the suction of the reactor feed pumps. Thus, feedwater flow to the vessel is lower in actuality than that being measured by the LEFM and used in the reactor heat balance calculation for core thermal power determination. This resulted in an indicated core power which was artificially high, causing gross electrical output to be unnecessarily limited. A loss of about 5 MWe was seen. A design change (Ref. 3) was initiated to install a flow transmitter to feed a new computer point (N21-N061). The value of this computer point, after appropriate unit conversions, can then be subtracted from the measured LEFM value to account for the diversion of flow to the skid and away from the vessel. This will result in a reactor heat balance calculation which uses the correct feedwater flow. Core thermal power will no longer be indicating higher than actual, thus allowing operators to bring the plant to a higher electrical output consistent with true rated thermal power.

SAFETY EVALUATION SUMMARY AND CONCLUSIONS:

No Technical Specification (TS) changes are required since feedwater flow measurement related to the plant heat balance is not directly discussed in the TS, nor need it be. No TRM changes are required. The feedwater flow instrumentation is non-safety related and is not needed for mitigation or prevention of any transient or accident. No modifications to the permanently installed feedwater flow instruments, panel indications, circuitry, or vessel level control system are being made. Changes are in computer software only. UFSAR changes necessary to reflect the installation of the zinc injection skid and the additional flow element have already been made in conjunction with Reference 3.

Feedwater flow measurement uncertainty is considered in the determination of the MCPR Safety Limit (SL), however analyses and test results confirm that the overall LEFM uncertainty is much less than that assumed in the SL determination even at 55% of rated power. This conclusion is not changed by the addition of the proposed software changes to account for the zinc injection skid flow. The accuracy of the plant heat balance used to calculate core thermal power also depends significantly on the accuracy of the feedwater flow values. Transient and accident analyses allow for some uncertainty in the initial power level for postulated events, and the use of the LEFM with the zinc skid correction remains bounded by this assumed uncertainty. The proposed change does not introduce new types of events or make the likelihood or consequences of any analyzed event or equipment failure worse. No margin of safety is reduced. Thus, no unreviewed safety question is created as a result of this change.

Serial Number: 98-082-PSE Document Evaluated: CEWO No. 98-0008

(Safety Evaluation Revision)

BRIEF DESCRIPTION OF CHANGE, TEST OR EXPERIMENT:

ER 96/0984-01-00 installed flow transmitter 1N21-FJT-N061 in the Zinc Injection Skid in order to provide a remote flow signal to the PDS computer system. This flow signal is needed to allow correction of feedwater flow input into the reactor heat balance calculation to properly account for diversion of feedwater flow downstream of the Leading Edge Flow Meter (LEFM). CEWO 98-0008 provides the necessary software changes to the Plant Data System (PDS) to correct feedwater flow as measured by the LEFM using the flow signal from FT-N061.

Additional software will also be added to the processing of the feedwater flow information in PDS so that appropriate checks are present to ensure a conservative power calculation in the event of instrument failure. In particular, a check on excessive feedwater flow (Hi-Hi) will be done so far so that a zinc skid flow which is higher than the expected upper range will result in disabling of the LEFM input. This causes feedwater flow to default to the higher venturi readings, which is conservative in that is results in an indicated CTP which it too high, causing the plant operators to reduce power until indication is below rated. Also, should the zinc injection flow instrumentation fail downscale or should the zinc injection flow be reduced to the lower end of the operating range, the zinc skid flow value may be less accurate than desired. In the event that measured zinc skid flow decreases below 30 GPM, a value of 0 GPM will be substituted in the computer point, which ensures a conservative CTP calculation regardless of the amount of flow actually being diverted by the skid. The 30 GPM flow value is well below the anticipated range of normal operation for zinc injection.

Physical installation of the zinc skid flow transmitter is addressed separately in References 2 and 3.

REASON FOR CHANGE, TEST OR EXPERIMENT:

After installation and testing of the zinc injection skid, it was realized that the design had a detrimental (although conservative) effect on the reactor heat balance calculation. A small amount of feedwater flow is diverted downstream of the LEFM and passed through skid where zinc injection takes place, and then returned to the suction of the reactor feed pumps. Thus, feedwater flow to the vessel is lower in actuality than that being measured by the LEFM and used in the reactor heat balance calculation for core thermal power determination. This resulted in an indicated core power which was artificially high, causing gross electrical output to be unnecessarily limited. A loss of about 5 MWe was seen. A design change (Ref. 3) was initiated to install a flow transmitter to feed a new computer point (N21-N061). The value of this computer point, after appropriate unit conversions, can then be subtracted from the measured LEFM value to account for the diversion of flow to the skid and away from the vessel. This will result in a reactor heat balance calculation which uses the correct feedwater flow. Core thermal power will no longer be indicating higher than actual, thus allowing operators to bring the plant to a higher electrical output consistent with true rated thermal power.

SAFETY EVALUATION SUMMARY AND CONCLUSIONS:

No Technical Specification (TS) changes are required since feedwater flow measurement related to the plant heat balance is not directly discussed in the TS, nor need it be. No TRM changes are required. The feedwater flow instrumentation is non-safety related and is not needed for mitigation or prevention of any transient or accident. No modifications to the permanently installed feedwater flow instruments, panel indications, circuitry, or vessel level control system are being made. Changes are in computer

software only. UFSAR changes necessary to reflect the installation of the zinc injection skid and the additional flow element have already been made in conjunction with Reference 3.

Feedwater flow measurement uncertainty is considered in the determination of the MCPR Safety Limit (SL), however analyses and test results confirm that the overall LEFM uncertainty is much less than that assumed in the SL determination even at 55% of rated power. This conclusion is not changed by the addition of the proposed software changes to account for the zinc injection skid flow. The accuracy of the plant heat balance used to calculate core thermal power also depends significantly on the accuracy of the feedwater flow values. Transient and accident analyses allow for some uncertainty in the initial power level for postulated events, and the use of the LEFM with the zinc skid correction remains bounded by this assumed uncertainty. the proposed change does not introduce new type of events or make the likelihood or consequences of any analyzed event or equipment failure worse. No margin of safety is reduced. Thus, no unreviewed safety questions is created as a result of this change.

Serial Number: 98-083-NPE Document Evaluated: Temp Alt 98-0019

BRIEF DESCRIPTION OF CHANGE, TEST OR EXPERIMENT:

Temp Alt 98-0019 lifts the contactor coil leads for the Z51B002B Control Room Air Conditioning unit compressor starter in the local panel.

REASON FOR CHANGE, TEST OR EXPERIMENT:

The Control Room Air Conditioning unit, 51B002B, is currently declared inoperable. The leads are being lifted to allow the operation of the Z51B002B Control Room Air Conditioning unit fan. Operation of the Z51B002B fan is required for the Control Room Standby Fresh air unit to perform its design function. This fan provides the required circulation of the air through the control room envelope. The cooling portion of Z51B002B is currently declared inoperable and will remain so throughout the duration of the Temp Alt.

SAFETY EVALUATION SUMMARY AND CONCLUSIONS:

The lifting of the coil leads to the B Control Room A/C compressor starter will not affect the operation of the B Control Room fresh air system. The A/C unit isolation dampers operate in conjunction with the A/C unit fan and their ability to perform their safety function is unaffected by this modification. The filters, dampers, fan and interlocks of the B fresh air system are unaffected by this modification. No new interfaces with existing equipment are created by this change and no new equipment is introduced to create a new failure mode. No existing setpoints will require change due this modification. This modification does not adversely affect equipment important to safety and no seismic, fire protection or control room envelope concerns have been identified.

It is concluded that this modification will not degrade any important to safety systems, components or structures. The modification does not increase the probability of occurrence or increase the consequences of malfunction of equipment important to safety of a different type than evaluated in the SAR. The Technical Specifications are not affected and the margin of safety is unchanged.

Serial Number: 98-084-NPE Document Evaluated: ER 1997-0633-00-00

BRIEF DESCRIPTION OF CHANGE, TEST OR EXPERIMENT:

This change accepts-as-is specific internal conduit seals identified in ER 97-0633-00-00 which are sealed with approximately 3 inches of RTV 108 silicone. RTV 108 is not an approved seal material for fire or air-tight boundaries. Specifically, RTV-108 silicone is used as an internal seal for conduits in penetrations CE-138F, CE-139F, CE-140F, CE-141F, CE-1G, CE-39G, and CE-431G. These conduits penetrate boundaries which are required to be rated for fire and air-tight (control room envelop) boundary requirements. This change does not make a generic acceptance of the use of RTV 108 as an approved material for sealing internal conduits for fire or air-tight boundary separation requirements.

REASON FOR CHANGE, TEST OR EXPERIMENT:

GGCR1997-0431-02 identified various conduits which penetrate control room envelop barriers and are sealed internally with approximately 3 inches of RTV 108 silicone. These barriers are also required to be rated for fire. RTV 108 is not an approved material for sealing conduits which penetrate fire or air-tight boundaries. Due to the physical characteristics of the cured RTV 108 material, removal of the internal conduit seal material is difficult and would most likely lead to cable damage created during the seal removal process. Therefore, the acceptability of the existing RTV seals have been evaluated.

SAFETY EVALUATION SUMMARY AND CONCLUSIONS:

The changes made by this ER accepts-as-is RTV 108 silicone internal conduit seals for specific electrical conduits penetrating fire barriers and the CRE. These specific internal conduit seal configurations have been evaluated and determined to maintain the fire resistance rating and air tight boundary requirements of the barriers as presently analyzed in the SAR; therefore, this change will not increase the probability or consequences of accidents or malfunction of equipment important to safety previously evaluated in the SAR. Also, this change 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. “Fire Rated Assemblies” are not addressed by Technical Specifications (TS). “Fire Rated Assemblies” are addressed in the Technical Requirements Manual (TRM) Section 6.2.8. Changes made by this ER involve internal sealing of electrical conduits penetrating “Fire Rated Assemblies” addressed by the TRM; however, these changes have been evaluated and determined to provide an adequate conduit seal and to maintain the 3-hour rating for the fire barriers. Changes made by this ER involve internal sealing of electrical conduits penetrating the CRE which is addressed by TS; however, the internal conduit seal arrangement has been evaluated and determined to provide an adequate air-tight internal conduit seal. Therefore, no parameters or requirements imposed on operation of the control room found in the Technical Specifications are altered, and the capability of these conduit seals to perform their necessary safety function for air-tightness is assured. Therefore, there will be no change to TS or TRM or the Bases for any TS or TRM.

Serial Number: 98-085-TRNG Document Evaluated: PAP 01-S-04-28

BRIEF DESCRIPTION OF CHANGE, TEST OR EXPERIMENT:

Change the Chairman of the Emergency Preparedness Training form the Director, Plant Projects and Support to the Manager, Emergency Preparedness.

REASON FOR CHANGE, TEST OR EXPERIMENT:

Reporting chain for the Manager, Emergency Preparedness changed from Director, Plant Projects and Support to Director, Nuclear Training.

SAFETY EVALUATION SUMMARY AND CONCLUSIONS:

The change of reporting order is an administrative change and will not after any accident analysis in the SAR. The reporting order Emergency Preparedness is not described in the Technical Specifications and therefore will not require any change to the GGNS Technical Specifications.

Serial Number: 98-086-NPE Document Evaluated: ER 98/0553-00-00

BRIEF DESCRIPTION OF CHANGE, TEST OR EXPERIMENT:

The proposed change will add a temporary isolation valve just downstream of isolation valve N1N11F369 in the low point drain line (1" HBD-126) in the Main and Reheat Steam (N11) System. The addition of the temporary isolation valve will allow isolation of the leaking drain line. The function of the affected drain line will not be altered by the temporary change.

REASON FOR CHANGE, TEST OR EXPERIMENT:

Steam is leaking past the seat on N1N11F369. ER 98/0553-00-00 will provide instructions to temporary repair low point drain line 1" HBD-126 on line. There are no tests or experiments associated with the temporary change.

SAFETY EVALUATION SUMMARY AND CONCLUSIONS:

The addition of the temporary isolation valve will eliminate the steam leak. The change described is only a temporary change. The original design configuration for the 1" drain line HBD-126 will be restored at the first convenient opportunity such as a forced outage but no later than RFO10. The operation of the N11 system will not be affected and the drain line reliability as well as isolation function will be improved from its current condition. Piping structural integrity will remain assured with the added components. This temporary change will not alter and will not have the potential to alter the operation, function or ability to perform the function of a system, structure or component described in the SAR. Section 15.6.4.2.2 of the UFSAR discusses the potential for, and consequences of, a steam leak in the Turbine Building (outside containment). The event analyzed and presented in this section of the UFSAR adequately bounds the potential scenarios that may result from the temporary change. There are no new systems or system function added by the temporary change, thus the existing accident scenarios and analyses presented in the UFSAR will not be adversely impacted by the temporary change. The temporary change will affect UFSAR Figure No. 10.3-001-1 since the drain line currently depicts only one isolation valve in this UFSAR Figure. However, the UFSAR Figure No. 10.3-001-1 will not be revised as this is only a temporary change. Addition of the temporary isolation valve 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 temporary change is entirely contained within the confines of the power block and will not affect or impact the plant's radiological or non-radiological effluents. Thus, the temporary change will have no adverse environmental impacts. After a review of the temporary change, it has been concluded that the addition of the temporary isolation valve does not represent an Unreviewed Safety Question and will have no adverse affects on the environment.

Serial Number: 98-087-NPE Document Evaluated: LDC 1998-056

BRIEF DESCRIPTION OF CHANGE, TEST OR EXPERIMENT:

Revise UFSAR Section 7.6.1.5.5.1.1.C to accurately describe the LPRM indications available at the operator's control console.

The referenced section currently states that when a central control rod is selected, the output signals from the nearest 16 LPRM detectors are displayed on 16 separate meters at the operator's control console. This statement is contrary to the as-built configuration of the plant. When a central control rod is selected, the outputs from the adjacent four LPRM detectors are displayed on the operator's control console.

REASON FOR CHANGE, TEST OR EXPERIMENT:

Correct inaccurate information in UFSAR Section 7.6.1.5.5.1.1.C.

SAFETY EVALUATION SUMMARY AND CONCLUSIONS:

LDC 1998-056 will correct a discrepancy in the UFSAR relating to the description of LPRM indications available at the operator's control console. UFSAR Section 7.6.1.5.5.1.1.C will be revised to be consistent with UFSAR Section 7.7.1.2.5.2 and the as-built configuration of the plant. The proposed change is an UFSAR revision only; no physical change to the facility is proposed. The change will have no affect on the operation of any system, structure or component addressed in the SAR. Revision of the referenced UFSAR Section will have no affect on radionuclide population, release rate, release duration, release mechanisms or radiation release barriers.

Serial Number: 98-088-NPE Document Evaluated: LDC 1998-067

BRIEF DESCRIPTION OF CHANGE, TEST OR EXPERIMENT:

The Seismic Qualification Central File (SQCF) Index is being replaced with fields in the Component Data Base (CDB). Existing and new fields in the CDB will provide all of the information necessary to identify seismic qualification requirements which was originally specified in the SQCF Index. Enhancement of the CDB will provide seismic test report/analysis numbers.

REASON FOR CHANGE, TEST OR EXPERIMENT:

The CDB is readily available to all plant personnel through computer access, while the SQCF Index is a hard copy document which is much harder to access. The CDB is also a "living document", which is updated after the implementation of design changes while the SQCF Index is only updated at yearly intervals.

SAFETY EVALUATION SUMMARY AND CONCLUSIONS:

Replacement of the SQCF Index with the CDB is an enhancement to the design basis information system. It will be accessible to all plant personnel. The design information in the CDB will be maintained by administrative procedures to the same level as were imposed on the SQCF Index. This replacement will have no physical affect on the operation of the plant or on any accidents evaluated in the UFSAR or create any new accidents.

Serial Number: 98-089-NPE Document Evaluated: ER 97/0043-00-00

BRIEF DESCRIPTION OF CHANGE, TEST OR EXPERIMENT:

This design change provides a series of modifications to improve reliability of the refueling bridges, and consists of upgrades supplied by the original manufacturer. Modifications include a replacement of the pneumatic systems (compressor, carbon steel piping, hoses, and reels) with newer more reliable components (compressor, refrigerated dryer, stainless steel piping, hoses and reels). Modifications will also include hoist modifications including a level wind system with breakaway rollers. The level wind mechanism and breakaway rollers help prevent cable damage due to improper winding of the cable on the hoist drums. (Note that the Fuel Handling Platform Monorail Hoist is not compatible with the addition of the level wind mechanism and hose reels. Therefore, these modifications will not be made to the Fuel Handling Platform Auxiliary Monorail Hoist.) The addition of the break-away roller/switches to hoists provides stoppage of the associated hoist motor in the event of hoist cable fouling. This modification lessens the possibility of cable fouling jamming the hoist reel and leaving a fuel assembly suspended. The switches are wired electrically in series with existing load cell switch circuitry for the Frame and Monorail hoists. For the Main Hoist, the switches de-energize new relay CR-MHCK. The contacts of this relay are in series with the existing motor fault circuitry. No hoists will be fitted with an upgraded brake assembly. Items are supplied with vendor applied paints/coatings. Where these items are to be placed inside the Containment, the vendor applied coatings will either replace existing unqualified coatings or be removed and replaced with coatings meeting GGNS coating requirements.

REASON FOR CHANGE, TEST OR EXPERIMENT:

GGNS has observed unacceptable amounts of down-time for the Refueling Platform and the Fuel Handling Platform. Causes for this vary, but have included: problems resulting from corrosion products in the pneumatic lines, problems resulting from the existing cable reels, and damage to the cables.

SAFETY EVALUATION SUMMARY AND CONCLUSIONS:

Technical Specification Section 3.9.1 and 3.9.2 require refueling interlocks to be OPERABLE. The purpose of these requirements is to prevent prompt reactivity excursions or criticality by preventing loading fuel into the core with any control rod withdrawn, or by preventing withdrawal of a rod from the core during fuel loading. This basis as defined in Technical Specification Bases B3.9.1 and B3.9.2 is not affected by these modifications and remains valid as written. The Refueling Platform Main Hoist interlocks described in the TRM interface with and provide input to the interlocks described in TS 3.9.1 and TS 3.9.2; however the specific values are not addressed in the TS. OPERABILITY of the interlocks will be restored prior to in-vessel fuel movement. Therefore, no change to the TS is required and, the margin of safety as defined in B3.9.1 and B3.9.2 is unchanged.

The Control Rod Removal Error During Refueling as described in UFSAR 15.4.1.1 as referenced in TS Bases B3.9.1 and B3.9.2 does not refer to specific limits or setpoints. This ER provides instructions to insure that the Refueling Platform main hoist cable camera, and hose reels are tensioned in a manner insuring that the calculational basis for set point limitations given in TRM Section 6.9.3.2 and 6.3.9.6 are not invalidated. All administrative controls and interlocks currently required for the operation of the main hoist will remain in effect. Therefore the probability and consequence of this accident have not increase. Additionally, fuel handling accidents in the Auxiliary Building and in the containment are described in UFSAR Sections 15.7.4 and 15.7.6, respectively. The breakaway roller and level wind modifications are both measures taken to reduce the potential for damage to hoists cables due to improper

winding onto the hoist drum. The changes to the pneumatic system will provide for a cleaner, more reliable air supply which will in turn improve the reliability and performance of air actuated tools and equipment. All hoists will receive an upgraded safety brake which has better mechanisms for absorbing the shock in the event that they are required to take up the hoist load. As a result, the probability and consequence of this accident have not increased. Also, modifications of the bridges will not result in potential for the hoists to provide uplifting forces greater than those considered for the fuel storage racks in UFSAR 9.1.1.3.2.h and 9.1.2.3.1.m. Therefore, the probability and consequence of accidents evaluated in the FSAR and associated with the Refueling and Fuel Handling platform have not increased.

Installed components have been determined to meet Seismic II/I criteria, and do not have an adverse impact on the structural integrity of the platform as a whole. The overall process for fuel movement and handling using the Refueling Platform or Fuel Handling Platform Main Hoists remains and is controlled by administrative controls, interlocks, and LCOs. Therefore, a malfunction resulting in a radionuclide release, different from those previously evaluated in the SAR, is not introduced. Therefore, the probability of occurrence and consequence of a malfunction of safety related equipment, as discussed in the UFSAR have not increased.

Reg. Guide 1.37, and sections of the UFSAR discuss the need to limit the amount of contact between halogens and stainless steel surfaces to minimize the potential for intergranular stress corrosion cracking. The addition of the refrigerated air dryer to the pneumatic system will introduce approximately 7 oz. Of R22 freon contained in a sealed closed loop system into the containment. Leakage from the system would vent as a gas to the containment atmosphere forming a negligible, non-toxic concentration. Decomposition of Chlorodifloromethane into its byproducts does not occur below ≈555°F, therefore, no explosive byproduct would be produced in the event of leakage. Unlike a spill of a liquid containing high concentrations of halogens, (e.g., lubricants, etc.) a gas leak would have no direct mechanism to enter the suppression pool or other fluid boundaries. Therefore, the addition of this small amount of a halogen into the containment does not create the possibility of a malfunction of equipment resulting in a radionuclide release, which is different from those already evaluated in the FSAR.

Serial Number: 98-090-NPE Document Evaluated: ER 98/0558-00-00

BRIEF DESCRIPTION OF CHANGE, TEST OR EXPERIMENT:

The proposed temporary change (or temporary repair) will affect drain lines of the main and reheat steam (N11) system. These lines are not performing a drain function for the N11 system during normal plant operation since the valves remain closed. These valves may be open during plant shut down for draining the affected component. The drain lines downstream of valves N11 F312, N11 F316A and N11 F316C will be isolated by the outboard isolation valve(s) kill and/or inboard isolation valve(s) kill and/or crimp/kill the line and injecting furmanite compound to stop the leak.

REASON FOR CHANGE, TEST OR EXPERIMENT:

Plant Staff has reported (GGCR1998-0882-00) a steam leakage coming from a drain hub located in high pressure (HP) feedwater heater 5A & 6A room. The source of leakage is unknown and may be one or all of the lines which tie into the drain hub. The CR requested that NPE provide a disposition to crimp/kill any one of these lines or all if necessary. However, Plant walk down has identified the steam leakage source from isolation valve seat leakage for one or all three of the ½”-DBD-140 blow down drain lines of Y type strainers N1N11D011, -D001A and D001C. These drain lines include N11 F311/F312, N11 F315A/F316A and N11 F315C/F316C isolation valves respectively. Therefore, a valve kill or pipe crimp/kill is required for the problem line(s) supplying steam to the drain hub. The proposed change will kill isolation valve(s) or crimp/kill the line and will inject furmanite compound to stop the leak.

SAFETY EVALUATION SUMMARY AND CONCLUSIONS:

The proposed temporary change will affect drain lines of the N11 system. These lines are not performing a drain function for the affected system during normal plant operation since the valves remain closed. These valves may be open during plant shut down for draining the affected component. The proposed change will kill outboard isolation valve(s) and/or inboard isolation valve(s) and/or crimp/kill the line and will inject furmanite compound to stop the leak. The change will have no effect on normal operating function of the N11 system described or implied in UFSAR. UFSAR Section 3.2 classifies the affected N11 system as ”Other” which means that a loss of system function would not affect the safe shutdown of the plant. UFSAR Table 3.2-1 classifies this system and their associated components as non-safety related, non-seismic, quality group D and ANSI B31.1. The turbine stop & control valve parameters and overspeed protection function are not affected by this modification and therefore do not represent a change to the Technical Specifications. The added weight of the Furmanite shutoff adapter, shutoff adapter plug and sealant compound is negligible and will have no adverse effect on the valve, process piping, or piping system supports. The crimping of the approved piping system(s) if required will not create any catastrophic failure mechanism. The piping will still maintain its load bearing capability. The temporary repair made by this ER change will not impose a change to the criteria listed in UFSAR Table 3.2-1.

Serial Number: 98-091-PSE Document Evaluated: Temp. Directive 04-1-01-M41-Temp 4

BRIEF DESCRIPTION OF CHANGE, TEST OR EXPERIMENT:

This safety evaluation is for temporary operation of a portion of the Drywell/Containment purge system normal supply air piping in a temporary drywell vent path line-up.

REASON FOR CHANGE, TEST OR EXPERIMENT:

This alternate drywell vent path allows for a temporary alternate flow path from the drywell back to containment in the event that the normal drywell vent path becomes operable.

SAFETY EVALUATION SUMMARY AND CONCLUSIONS:

Grand Gulf's primary containment is a General Electric Mark III design utilizing a drywell structure inside primary containment. The function of the primary containment is to isolate and contain fission products released from the reactor primary system following a design basis accident (DBA) and to confine the postulated release of radioactive material to within acceptable limits. The function of the drywell is to maintain a pressure boundary that channels steam resulting from a Loss-of-Coolant Accident (LOCA) to the suppression pool, where it is condensed.

Grand Gulf's design utilizes a safety-related containment spray mode of the Residual Heat Removal system (RHR) for cooling the containment area following a DBA LOCA. Also as part of plant design, Grand Gulf has two non-safety-related systems which are available to cool the drywell and primary containment areas during normal operation: (1) the Drywell Cooling system, and (2) the Containment Cooling system. Each of these systems is comprised of fan/coil unit coolers with a cooling water supply. The cooling water sources for the drywell and Containment Coolers are the Drywell Chilled Water system (DCW) and the Plant chilled Water system (PCW), respectively. PCW piping to/from the Containment Coolers penetrates only primary containment while the DCW piping to/from the Drywell Coolers penetrates both primary containment and drywell.

Neither the Drywell Coolers nor the Containment Coolers are required nor credited for accident mitigation. Valves M41-F013 and M41-F015 are the only safety related valves that will be opened for the vent path. The only safety function associated with M41-F013 and M41-F015 is drywell isolation.

To meet primary containment isolation requirements of General Design Criterion (GDC) 56, each Drywell/Containment Cooler containment penetration utilizes two containment isolation dampers which will remain close during the drywell ventilation. In addition to primary containment isolation, the containment cooling system drywell penetration utilizes two drywell isolation valves. The systems use a combination of safety-related air-operated valves (AOVs) that are Safety Class 2, Seismic Category I. The AOVs automatically isolate on a LOCA/primary containment isolation signal. Also, the containment isolation valves are leak tested per the requirements of 10CFR50, Appendix J, "Primary Reactor Containment Leakage Testing for Water-Cooled Power Reactors," Penetration piping is Safety Class 2, Seismic Category I fabricated and installed in accordance with ASME Section III. The remaining piping and components are non-seismic.

The existing procedures do not specify use of the containment cooler drywell isolation valves for drywell venting path during normal power operation. The operators may decide to use the containment cooler drywell isolation valves as an option for drywell venting to the containment for compensatory action purposes.

In the case of the Containment Coolers and Containment/Drywell Purge system, there is no procedure that allows operators to line up the containment cooler drywell isolation damper and close the Containment/Drywell Purge Containment isolation dampers. This safety evaluation concludes that this lineup allows reasonable action for a temporary departure from the current plant procedures but is not a departure from the Technical Specifications to provide an alternate drywell vent path. In an emergency when action is immediately needed to protect the public health, the dampers will automatically isolate.

Serial Number: 98-092-PSE Document Evaluated: Temp Alt 98-0023

BRIEF DESCRIPTION OF CHANGE, TEST OR EXPERIMENT:

This safety evaluation is for a temporary alteration of the discharge piping from two Turbine Building Cooling Water (TBCW) relief valves. The original plant design had routed the relief valve discharge to the storm drain system. The temporary alteration will re-route the relief valve discharge piping to a clean chemical waste sump inside the water treatment building.

REASON FOR CHANGE, TEST OR EXPERIMENT:

Leakage past the relief valve during routine plant operation resulted in a release of TBCW water into the storm drain system. Re-routing the relief valve discharge line to the clean chemical waste sump inside the water treatment building will allow any TBCW leakage to be captured inside the water treatment building and routed to radwaste. Re-routing the relief valve discharge line will allow approximately one half of the total Standby Service Water (SSW) discharge volume to be captured inside the water treatment building and routed to radwaste in the event of the relief valves lifting. Any remaining SSW discharge would overflow the sump and enter the water treatment building oily waste sump or exit the water treatment building through the building doors.

SAFETY EVALUATION SUMMARY AND CONCLUSIONS:

The TBCW system provides the normal source of cooling water for the instrument air compressors and service air compressors. The air compressors are located in the Water Treatment Building. Both TBCW and instrument air are non-safety related systems. The safety related SSW system provides a backup source of cooling water for the instrument air compressors to maintain operation of instrument air following a loss of all offsite power (LOP).

TBCW relief valves P43-F460A and P43-F460B are located in parallel and provide the necessary capacity to prevent over-pressure of instrument air compressor coolers when SSW is providing backup cooling water. TBCW system pressure is limited and is not capable of over-pressuring the coolers. A special case of a dead-headed SSW pump can however over-pressure the coolers (Ref. GGNS Bechtel calc. 2.2.65, Rev. 1). Relief valves P43-F460A/B are sized such that the flow capacity of the valves will lower SSW pressure downstream of the TBCW relief valve location to the point that the non-safety related instrument air compressor coolers do not exceed their rated pressure.

The Temp Alt will route the relief valve discharge to a nearby clean chemical waste drain header where the discharge can be controlled. The existing drain piping (3”-JBD-1083) is governed by ANSI B31.1, and has a maximum design piping pressure rating of 125 PSIG. The Temp Alt will use fire hoses, pipe, flanges, and fittings with a rating at or above 125 PSIG. The use of fire hose is not approved per the ANSI B31.1 Power Piping Code. However this Temp Alt will meet the functional and safety intent of ANSI B31.1 based on pressure rating for a brief period of time until permanent piping can be installed and supported per ANSI B31.1. This Temp Alt is not considered to be later approved as permanent plant design.

Serial Number: 98-093-NPE Document Evaluated: ER 97/0117-00-00

BRIEF DESCRIPTION OF CHANGE, TEST OR EXPERIMENT:

This ER will install four new flow dampers in the ductwork associated with the Control Cabinet Room HVAC System, Z51. These new flow dampers will be located upstream/downstream of existing fire dampers associated with this ductwork. This ER will also modify the control logic of CO_2 Fire Suppression System N1P64D216 which provides protection for the area served by this HVAC system. This control logic modification will automatically close all associated dampers upon manual initiation instead of heat detection. It will also change the time delay associated with closing the existing fire dampers from 5 seconds to 30 seconds after manual initiation and will close the new flow dampers 5 seconds after manual initiation. The time delay in the automatic closure of the existing dampers will allow the Z51 system air flow to be stopped by the new flow dampers prior to the existing fire dampers attempting to close.

All wiring changes associated with this modification are Non-Class 1E circuits of the P64 Fire Protection System only. The wiring modifications will be performed in such a manner which will not defeat established separation criteria between safety and non-safety related circuits or external interfaces. Conduits and cables will be routed to maintain proper separation per the requirements of Reg. Guide 1.75 and GGNS's Fire Protection Plan. The new dampers will be fabricated, tested and mounted per applicable standards to ensure the ductwork's ability to perform its safety related function before, during and after applicable design basis accidents. All new terminal boxes and conduit are Non-Class 1E and will be mounted seismic II/I.

REASON FOR CHANGE, TEST OR EXPERIMENT:

GGCR19970726-00 (MNCR 96/0047) identified a non-conforming condition that could degrade the ability of the CO_2 fire suppression system from performing its function within, among other locations within the plant, the Control Cabinet Room (OC703). This non-conformance identified that the Z51 dampers are not designed to close with air flow through their ducts. The Z51 HVAC system supplies both the control room and the control cabinet room OC703 and is not designed to shutdown upon a detected fire thereby ensuring closure of the Z51 dampers. Therefore, fire suppression system N1P64D216 is in an indeterminate state with regard to its ability to contain an adequate concentration of extinguishing agent. This modification will ensure Z51 ventilation system's associated fire dampers for the control cabinet room will close upon manual initiation of the fire suppression system.

SAFETY EVALUATION SUMMARY AND CONCLUSIONS:

This change will not affect the GGNS Technical Specifications, Technical Specification Bases, TRM or reduce any margin of safety. This modification will not increase the probability of occurrence or consequences of an accident previously evaluated in the SAR. This modification will not increase the probability of occurrence or consequences of a malfunction of equipment important to safety previously evaluated in the SAR. This modification will not create the possibility for an accident or malfunction of a different type than any previously evaluated in the SAR. No unreviewed safety question will result from this modification.

The installation of the four new flow dampers in combination with the existing fire dampers will ensure that a total flood by the CO_2 fire suppression system is effective and provides an adequate concentration of the extinguishing agent within the protected area. Based on Panel N1P64D216's pre-op test time for the CO_2 to enter the area of protection, (OC703 – Control Building, El. 189') from the CO_2 storage tank

(Yard, El. 133'), the change in fire damper closure will remain prior to discharge and will not adversely affect the concentration level in the area. Since the functions of each system, Z51 and P64, is not dependent upon the number of dampers, their functions will not be altered or change. The fire suppression system currently interfaces with the dampers associated with the Z51 ductwork. No new interfaces with other components, structures or systems will be created or removed. The wiring modifications performed by this ER will only affect non-safety related circuits of the P64 Fire Protection System. The wiring modifications will be performed in a manner which will not defeat established separation criteria between safety and non-safety related circuits. The design meets the requirements of the GGNS's Fire Protection program and Reg. Guide 1.75 for separation and isolation between Class 1E and Non-Class 1E equipment. The new electrical equipment will be powered via the existing BOP power supply at the N1P64D216 and TBN1P64D216 panels. New conduit and terminal boxes will be mounted Seismic II/I to ensure no loose items will become hazards during a seismic event. The new dampers will be mounted to existing supports utilized to retain the existing spools of duct to be replaced by the dampers. These supports were evaluated and found to be adequate for properly securing the new dampers. The new dampers were fabricated and tested per applicable standards and processes to provide assurance of their ability to perform their intended safety function.

GGNS Fire Hazards Analysis has been reviewed and will not be affected by this modification. Section 9.5.1.2.2.5 of the UFSAR describes the Gaseous Extinguishing Systems. The test of this section with regard to sequence of damper closure and gaseous agent injection will remain accurate. UFSAR Figure 9.4-001 (P&ID M-0049) requires revision to reflect the new additional dampers for the control cabinet room OC703.

Serial Number: 98-094-NPE Document Evaluated: ER 98/0567 (all supplements)

BRIEF DESCRIPTION OF CHANGE, TEST OR EXPERIMENT:

Entergy Mississippi will remove two (2) 500 kv switchyard breakers (J5212 and J5220) and install permanent bus work in their place. The breakers' associated motor operated switches (J5211, J5213, J5219 and J5221) will remain in place, but remote operation will be defeated. They will also relocate their protective relaying taps to opposite side of remaining breakers. GGNS staff will modify SH13-P807 and other panels to reflect new switchyard configuration. This safety evaluation is performed in support of the Design Change and document update only.

REASON FOR CHANGE, TEST OR EXPERIMENT:

The J5212 and J5220 500 kV breakers are high maintenance due to continuous requirement to replace escaped arc suppression gas. These two breakers were originally required to provide a "breaker-and-a-half" configuration for the GGNS Unit II Gen/SVC Xformer 21 "string" and the Ray Braswell/Baxter Wilson "string" between the East and West busses of the 500 kV switchyard. The "Breaker-and-a-half" configuration allowed isolation of either the East or West 500 kV bus while permitting the remaining bus to supply loads at GGNS and maintain the 500 kV grid integrity. Since the GGNS Unit II and the Ray Braswell line were cancelled and never installed, the "breaker-and-half" configuration for these two "strings" and these breakers are not required. These two "strings" will now be a "two-breaker-two-bus" configuration, which still permits proper isolation, fault protection and power supplying for their associated string. Once these breakers are removed and replaced with permanent bus material, the associated control room panel, SH13-P807, will be modified to reflect switchyard configuration and other associated terminal panels will be modified to secure spared, abandoned and other unused cabling and equipment. Although breakers are not GGNS equipment and GGNS does not have control of their operation, they are reflected in GGNS design documents.

SAFETY EVALUATION SUMMARY AND CONCLUSIONS:

The removal and replacement of these breakers with bus work and the modification of the associated indication/panels will not adversely affect the reliability of off-site power sources to GGNS. Should these breakers remain and sufficient gas was lost, these breakers would auto-trip open for self-preservation purposes. The removal of these unnecessary breakers will reduce the probability of isolation from a switchyard bus and reduce operator challenges. Any credits or failures associated with the motor operated switches will remain as-is. Degraded grid determinations have been reviewed and will not be impacted since these breakers are not credited for mitigating or eliminating any consequences or effects. The requirements of 10CFR50, Appendix A, GDC 17 will not be adversely affected and acceptance criteria for off-site power reliability will remain as-is. The modification to associated indication and panels will ensure all separation criteria of Reg. Guide 1.75 and GGNS standards/specifications will be maintained (physically and electrically). The seismic qualification of affected panels has been reviewed and will not be affected. This change will not affect the GGNS Technical Specifications, Technical Specification Bases, TRM or reduce any margin of safety. This modification will not increase the probability of occurrence or consequences of an accident previously evaluated in the SAR. This modification will not increase the probability of occurrence or consequences of a malfunction of equipment important to safety previously evaluated in the SAR. This modification will not create the possibility for an accident or malfunction of a different type than any previously evaluated in the SAR. No unreviewed safety question will result from this modification. All affected drawings/documentation will be updated to remove reference to these breakers and breaker J5244 that has been removed previously. Additionally, text and drawings in the GGNS UFSAR Chapter 8 will be updated to remove reference to the non-existent GGNS Unit II generator and the 500 kV Ray Braswell Line and provide general update of Unit II references.

Serial Number: 98-095-NPE Document Evaluated: ER 97/0948-00-00

BRIEF DESCRIPTION OF CHANGE, TEST OR EXPERIMENT:

The proposed change will add valve numbers to six existing valves installed in the Domestic Water System. The function of the Domestic Water System (P66) will not be altered by this change.

REASON FOR CHANGE, TEST OR EXPERIMENT:

The addition of the valve numbers will eliminate a duplicate valve number found in the plant. ER 97/0948-00-00 will provide instructions to tag six valves in the HP lab installed per MWO M21440 with new valve numbers. There are no tests or experiments associated with this change.

SAFETY EVALUATION SUMMARY AND CONCLUSIONS:

Section 9.2.4 of the UFSAR discuss the Domestic Water System. The Domestic Water System has no safety-related function. Failure of the system will not compromise any safety-related equipment or component and will not prevent safe shutdown of the plant. Adding valve number tags to existing valves will not affect the piping structural integrity. Adding valve numbers to existing valves will affect UFSAR Figure No. 9.2-014. However, the addition of the valve numbers will not result in the operation of any plant system or component in a manner that is inconsistent with information contained in the UFSAR. This change will not alter and will not have the potential to alter the operation, function or ability to perform the function of a system, structure or component described in the SAR. There are no new systems or system functions added by this change. Thus the existing description presented in the UFSAR will not be adversely impacted by this change. The change is entirely contained within the confines of the Health Physicist Lab area and will not affect or impact the plant's radiological or non-radiological effluents. Thus, the change will have no adverse environmental impacts. After a review of the change, it has been concluded that the addition of the valve numbers does not represent an Unreviewed Safety Question and will have no adverse affects on the environment.

Serial Number: 98-096-NPE Document Evaluated: ER 96/0494-00-00

BRIEF DESCRIPTION OF CHANGE, TEST OR EXPERIMENT:

ER 96/0494-00-00 mutes the speakers for the Public Address (PA) System in the Main Control Room such pages from the plant will only be heard if party line 1 is selected. This is accomplished by the installation of a Selective Page Unit, GAI-Tronics MODEL E96004 in the system. The Selective Page Unit is connected to PA Handset HAJ-OC504-1. Ceiling speakers SC-OC504-1, SC-OC504-2 and the small speakers at the Shift Superintendent console and the Control Room Operator console are connected to the Selective Page Unit. When the Selective Page Unit detects an "off-hook" condition, i.e. party line 1 selected and page pushbutton depressed, it energizes a timing relay which connects the associated speakers to the PA Handset for 1-60 seconds. After the timing relay times out, the speakers are again disconnected.

REASON FOR CHANGE, TEST OR EXPERIMENT:

The intent of this modification is to reduce the noise level in the Main Control Room. With this modification installed, routine pages on party line 2-5 will not be heard in the Main Control. Thus, the noise level will be reduced and the operators can better concentrate on their duties.

SAFETY EVALUATION SUMMARY AND CONCLUSIONS:

As described in SAR Section 9.5.2, the Public Address system is provided for intra plant voice communications. The system has no safety related functions. Failure of the system will not compromise any safety related system or component, and will not prevent the safe shutdown of the plant. The Public Address system does not interface with any safety related system. The Selective Page Unit which weighs 9.5 pounds, is mounted on the side of panel 1H13-P864 which is rated seismic category I. The additional weights is small compared to the overall panel weight and no seismic II/I concern is created. This modification improves the operation of the plant by muting routine pages and thereby reducing distractions to the operators in the Main Control Room. Paging of the Main Control Room will be maintained on PA system channel 1. This evaluation is required because FSAR Figure 9.5-009E is updated.

Serial Number: 98-097-NPE Document Evaluated: ER 98/0558-01-00

BRIEF DESCRIPTION OF CHANGE, TEST OR EXPERIMENT:

The proposed temporary change (or temporary repair) will affect drain lines of the extraction steam (N36) system. These lines are not performing a drain function for the N36 system during normal plant operation since the valves remain closed. These valves may be open during plant shut down for draining the affected component. The drain lines downstream of valves N36F017A and N36F017B will be isolated by valve(s) kill or/or crimp/kill the line and injecting furmanite compound to stop the leak.

REASON FOR CHANGE, TEST OR EXPERIMENT:

Plant Staff has reported (GGCR1998-0882-01)a steam leakage coming from a drain hub located in high pressure (HP) feedwater heater 5A & 6A room. The CR requested that NPE provide a disposition to crimp/kill any one of these lines or all if necessary. However, Plant walk down has identified the steam leakage source from isolation valve seat leakage for one or both of the ½" – GBD-127 blow down drain lines of Y type strainers N1N36D004A and N1N36D004B. Therefore, a valve kill or pipe crimp/kill is required for the problem line(s) supplying steam to the drain hub. The proposed change will kill isolation valve(s) or crimp/kill the line and will inject furmanite compound to stop the leak.

SAFETY EVALUATION SUMMARY AND CONCLUSIONS:

The proposed temporary change will affect drain lines of the N36 system. These lines are not performing a drain function for the affected system during normal plant operation since the valves remain closed. These valves may be open during plant shut down for draining the affected component. The proposed change will kill isolation valve(s) and/or crimp/kill the line and will inject furmanite compound to stop the leak. The change will have no effect on normal operating function of the N36 system described or implied in UFSAR. UFSAR Section 3.2 classifies the affected N36 system as "Other" which means that a loss of system function would not affect the safe shutdown of the plant. UFSAR Table 3.2-1 classifies this system and their associated components as non-safety related, non-seismic, quality group D and ANSI B31.1. The turbine stop & control valve parameters overspeed protection function are not affected by this modification and therefore do not represent a change to the Technical Specifications. The added weight of the Furmanite shutoff adapter, shutoff adapter plug and sealant compound is negligible and will have no adverse effect on the valve, process piping, or piping system supports. The crimping of the approved piping system(s) if required will not create any catastrophic failure mechanism. The piping will still maintain its load bearing capability. The temporary repair made by this ER change will not impose a change to the criteria listed in UFSAR Table 3.2-1.

Serial Number: 98-098-NPE Document Evaluated: ER 96/-0976-00-00

BRIEF DESCRIPTION OF CHANGE, TEST OR EXPERIMENT:

The barring devices for the Division I and the Division II diesel generators are used to manually rotate the diesel engines during maintenance. ER 96/0976-00-00 removes these barring devices from the Division I and the Division II diesel generators and locates them on the wall of the diesel generator buildings during normal plant operation. In order to support the removal of the barring device from the engines, the barring device interlock valves are disconnected from the pneumatic control logic and the tubing serving these interlocks is capped, additionally the pressure switches that illuminate the barring device engaged annunciator windows are abandoned.

REASON FOR CHANGE, TEST OR EXPERIMENT:

The barring devices for the Division I and the Division II diesel generators are used to manually rotate the diesel engines during maintenance. There have been incidents where the barring device interlock valves have leaked causing false indications of the barring device being engaged during power operations. The removal of the devices and the their associated interlocks will preclude the possibility of receiving erroneous “barring device engaged” annunciator signals during plant operation. The appropriate maintenance procedures will be revised to treat the barring device as the tool it has become.

SAFETY EVALUATION SUMMARY AND CONCLUSIONS:

As discussed in UFSAR Sections 8.3 and 9.5, the Division I and Division II diesel generators are designed to start and obtain rated voltage and frequency within 10 seconds of receipt of a start signal. The GGNS Technical Specifications Sections 3.8.1, 3.8.2 and 3.8.3 addresses the operability requirements of the diesel generators. The removal of the barring device from the engine and its storage on the wall of the diesel building during normal operation and the disconnecting of the barring device from the pneumatic control logic and plugging of this tubing and the abandoning of the pressure switches that illuminate the barring device engaged annunciator windows do not affect the operation of the diesel generators and will not affect their ability to load and attain rated voltage frequency. The modifications do not invalidate any analyses or assumptions contained in the UFSAR regarding the diesel generators. The changes do not compromise any safety related system or prevent safe reactor shutdown. The ability of equipment important to safety to perform its safety function is not altered by this modification. The Technical Specifications are not affected and the margins of safety are unchanged.

Serial Number: 98-099-PSE Document Evaluated: TA 98-0027

BRIEF DESCRIPTION OF CHANGE, TEST OR EXPERIMENT:

The Floor Drain Collector Pump Discharge Isolation Valve, SG17F113, is to be reassembled without the valve discs installed to facilitate the operation of the Floor Drain subsystem of the Liquid Radwaste System.

REASON FOR CHANGE, TEST OR EXPERIMENT:

Parts of the valve are obsolete. An excessive amount of time will be required to install a new valve. The system is needed to be operational prior to the availability of another valve. The valve is normally open; therefore, reassembly of the valve without the valve discs will not pose an unusual operational condition.

SAFETY EVALUATION SUMMARY AND CONCLUSIONS:

The Floor Drain Collector Pump Discharge Isolation Valve, SG17F113, will have its valve discs removed to facilitate operation of the Floor Drain portion of the Liquid Radwaste System. This change will render the valve inoperative but will have no affect on the operation or function of the Liquid Radwaste System because the valve is normally in the open position. This change to the Liquid Radwaste System will require a change to the figure of the system as displayed in the SAR. Although temporary, this change requires performance of this safety evaluation.

Removal of the valve seat discs will render SG17F113 inoperable. However, several valves will provide isolation for the Floor Drain Collector Pump and Tank, normally provided by this valve, in the system. The accident evaluated in the SAR associated with the Liquid Radwaste System is the spill accident. This valve will have no affect on the portions of the system involved in either scenario of the evaluated spill accident. Therefore, the change posses no increase in the probability of neither an accident nor the consequences of an accident previously evaluated in the SAR.

Because the normal position of the valve is open, this change will have no affect on the function of the Liquid Radwaste System. The Liquid Radwaste System is not a safety-related system and does not have any direct or indirect affect on systems or equipment important to safety. Thus, the change will neither increase the probability of causing a malfunction of equipment important to safety evaluated in the SAR nor increase the consequences from such a malfunction. Additionally, the change will neither cause an accident nor cause the malfunction of equipment different from any type previously evaluated in the SAR.

As stated earlier, the operation of the Liquid Radwaste System is unaffected by the change. This system is not safety related and does not maintain a reactor coolant boundary. Therefore, the margin of safety as defined in the BASES for any Technical Specification is not affected. In conclusion, there are no unreviewed safety questions created by this change.

Serial Number: 98-100-NPE Document Evaluated: TRM Section 6.7.3 and UFSAR Pages 16B.1-149 and 16B.1-150

BRIEF DESCRIPTION OF CHANGE, TEST OR EXPERIMENT:

This proposed change will alter the GGNS Technical Requirements Manual (TRM), LCO 6.7.3, "Area Temperature Monitoring," Action B.1. It presently states:

> **Initiate action to prepare and submit a Special Report to the Commission within the next 30 days providing a record of the amount by which and the cumulative time the temperature in the affected area exceeded its limit and an analysis to demonstrate the continued OPERABILITY of the affected equipment.**

The new wording will be as follows:

> Initiate action to provide a record of the amount by which and the cumulative time the temperature in the affected area exceeded its limit and an analysis to demonstrate the continued OPERABILITY of the affected equipment.

This same change will apply to the identical wording contained in TLCO 6.7.3, Required Action C.1.

Additionally, the page in the UFSAR that corresponds to these TRM pages – 16B.1-149 and 16B.1-150 – will also be changed in the same manner.

REASON FOR CHANGE, TEST OR EXPERIMENT:

The requirement in LCO 6.7.3 to provide a special report to the NRC should area temperature exceed the 6.7.3 requirements for more than 8 hours or exceed the limits in Table 6.7.3-1 by > 30 degrees is an unnecessary administrative burden to both GGNS and the NRC. Deletion of this requirement does not affect nuclear safety at all, but the deletion will enhance efficiency by getting rid of an unnecessary requirement. The intent and safety function of TLCO 6.7.3, which is to ensure operability of affected equipment, will remain the same.

SAFETY EVALUATION SUMMARY AND CONCLUSIONS:

This change to LCO 6.7.3, Required Action B.1 and C.1, deletes the requirement to provide a special report to the NRC should area temperature exceed the 6.7.3 requirements for more than 8 hours or exceed the limits in Table 6.7.3-1 by > 30 degrees. The intent of TLCO 6.7.3 is to ensure operability of affected equipment. The requirements for recording temperatures and evaluating equipment operability will remain intact. The intent of this LCO will remain the same, only the additional requirement to provide a special report to the NRC documenting the results will be deleted. In addition, any conditions that seriously affect the performance of a safety function will continue to be evaluated for reportability in accordance with 10 CFR 50.72 and 50.73; consequently, nuclear safety and safety system performance will remain unaffected by these changes.

Changing this reporting requirement is purely an administrative change and cannot affect the initiation of any accident evaluated in the SAR, nor can it contribute to or act as an initiator of any accident described in the SAR. In addition, the changes proposed do not add, change, or delete any physical components in the plant and therefore cannot create the possibility of a new type of accident or a new type of malfunction of safety related equipment. Since the change proposed is merely administrative in nature

and only affect a special reporting requirement, it cannot affect any plant equipment, both directly or indirectly. Additionally, because this reporting requirement does not contribute to the initiation and mitigation of any accidents and does not have an active role in the initiation of or response to any accidents evaluated in the SAR, this change does not affect the consequences of a malfunction of equipment important to safety.

Consequently, the changes that are proposed do not constitute an Unreviewed Safety Question and are therefore acceptable for implementation.

Serial Number: 98-101-NPE Document Evaluated: ER 96/0885-00-00

BRIEF DESCRIPTION OF CHANGE, TEST OR EXPERIMENT:

Revision 01 of this safety evaluation is prepared due to a typographical error in Revision 00. The word "not" was omitted regarding postulation of a new accident in the last sentence of question #5. Revision 00 of this safety evaluation evaluates ER 96/0885-00-00, which authorizes the permanent installation of the Leading Edge Flow Monitor (LEFM) system hardware originally installed by Temporary Alteration (T/A) 96-0008. Computer software and use of the LEFM system to provide feedwater flow input is addressed separately in safety evaluation SE 96-0030-00 and is not further evaluated in this safety evaluation. In addition to the changes made by T/A 96-0008, the ER replaces PVC jacketed cables with IEEE 383 fire rated cables, relocates the LEFM electronic unit to a more suitable environment, installs rigid conduit and related supports in place of flex conduit, and documents the permanent transducer probe installation evaluated in ER 97/0026.

REASON FOR CHANGE, TEST OR EXPERIMENT:

The purpose of ER 96/0885-00-00 is to provide details and design approval for the permanent design and installation of the LEFM system hardware, originally installed by T/A 96/0008. The permanent installation will provide a more suitable location and environment for the LEFM panel, provide documentation of permanent flow transducers (evaluated by ER 97/0026), and will upgrade the hardware installation to meet NPE design electrical standards (ES).

SAFETY EVALUATION SUMMARY AND CONCLUSIONS:

The Caldon, Inc. LEFM system is designed to provide improved accuracy feedwater flow measurements, which are then supplied as inputs to the core thermal power calculations. The software design and use of this system to provide the feedwater flow values has been evaluated and documented in SE 96-0030-00, and is not further evaluated in this safety evaluation, since the conclusion of SE 96-0030-00 was that no unreviewed safety question was raised or created as a result of use of the LEFM system. The changes in ER 96/0885-00-00 provide for permanent design in accordance with approved electrical and civil standards. The LEFM panel 1C34-P001 and hardware installed by this ER are non-safety related and so not affect any safety functions or safety systems. The external metering section (transducers and mounting collars on feedwater lines A & B) installed by T/A 96/0008 are considered "installation complete" and are not changed by this ER. Installation of the metering sections has been evaluated regarding load and pipe stress by ER 96/6047. This ER has no affect on any fire hazard analysis/safe shutdown criteria, nor does the design affect the seismic capability of any seismic class 1 component. Section 7.7.1.4.3.4 and Figure 10.4-013 of the FSAR will be revised via LDCR Change No. 97-095 to reflect the ER change. The conclusion of this safety evaluation is that no unreviewed safety question is created by this ER change, nor are any Technical Specification or TRM changes required or affected.

Serial Number: 98-102-NSA Document Evaluated: LDC 98-077

BRIEF DESCRIPTION OF CHANGE, TEST OR EXPERIMENT:

This change deletes a commitment to apply the Quality Assurance Program to the equipment and structures associated with the Emergency Support Facility.

REASON FOR CHANGE, TEST OR EXPERIMENT:

The commitment to apply provisions of the Quality Assurance Program to the non-safety related facilities is overly restrictive. The Quality Assurance Program will remain in effect for the Emergency Planning Program. Therefore maintaining a quality assurance program over the structures and equipment is unnecessary.

SAFETY EVALUATION SUMMARY AND CONCLUSIONS:

The requirement to apply the Quality Assurance Program to these facilities came from the Grand Gulf response to Question 0260.1 from the NRC on Section 3.2 of the FSAR. Our response stated that the facilities where not safety related areas but that we would apply the appropriate sections of the program as detailed in the revised Appendix B to the Q-list. This response was subsequently included in the UFSAR Table 3.2-1 Note (NN) Item 10. This evaluation concludes that the removal of this commitment will not result in any unreviewed safety questions or change to the Technical Specifications including the Technical Requirement Manual. The Operating License Condition requiring the upgrade to the ERF is not affected by this change.

Serial Number: 98-103-NPE Document Evaluated: ER 1997-0557-00-00

BRIEF DESCRIPTION OF CHANGE, TEST OR EXPERIMENT:

ER 1997-0557-00 documents the acceptability of non-standard fire barrier design utilized as part of the fire wall assembly separating Fire Zone 0C702 (Upper Cable Spreading Room, Control Building El. 189'-0") and Fire Zone 0C706 (Corridor, Control Building El. 190'-0"). In addition, structural steel fireproofing is being reworked on a portion of this non-standard fire barrier partition and a minor drawing change is being made to correct a penetration number (for a penetration through this non-standard fire barrier partition).

REASON FOR CHANGE, TEST OR EXPERIMENT:

The Upper Cable Spreading Room (Fire Zone 0C702, Fire Area 47) is separated from the Corridor (Fire Zone 0C706, Fire Area 58) on elevation 190'-0") of the Control Building by a wall section which is approximately 9 feet wide and 14 feet high. The fire protection program and UFSAR describe this wall as a 3-hour rated fire barrier and part of the fire area boundary separating Fire Areas 47 and 58. Construction of this wall section, up to approximately the 197'-4" elevation, utilizes a standard fire barrier design of reinforced concrete. The section of wall above elevation 197'-4" utilizes a non standard fire barrier design which consist of a double steel plat partition and a W27x94 I-Beam assembly which have 3-hour rated structural steel is to provide sufficient insulation from the heat of a fire such that the steel will not reach a temperature of 1100°F (sufficient to cause structural weakness). The purpose of a 3-hour rated fire barrier is to prevent the propagation of fire from one side of the barrier to the other. One of the test parameters for a fire barrier test is that the cold side temperature does not reach a temperature sufficient to ignite ordinary combustibles (250°F plus ambient is generally considered acceptable). Therefore, the test protocol for 3-hour structural steel fire proofing is very different from that required for a 3-hour rated fire barrier. Therefore, the fire proofing of the steel plant and I-Beam assembly described above does not in and of itself provide a 3-hour rated "fire barrier" configuration for the wall assembly as required by the GGNS Fire Protection Program.

In addition, 3-hour rated structural steel fire proofing material is being installed on an approximately 2 feet by 3 feet portion of the above described double steel plate assembly which was inadvertently omitted during construction. Penetration No. CE-370G on Drawing C-0626C was incorrectly numbered and this editorial change is being made.

SAFETY EVALUATION SUMMARY AND CONCLUSIONS:

License Condition 2,C(41) allows GGNS to make changes to the approved Fire Protection Program through the 50.59 process if those changes do not adversely affect the ability to achieve and maintain safe shutdown in the event of a fire. Therefore, form the fire protection standpoint the bases for evaluation is "no adverse effect on the ability to achieve and maintain safe shutdown in the event of a fire." Generic Letter 86-10, Enclosure No. 1, Interpretation No. 4 states: "Where fire area boundaries are not wall-to-wall, floor-to-ceiling boundaries with all penetrations sealed to the fire rating required of the boundaries, licensees must perform an evaluation to assess the adequacy of fire boundaries in their plants to determine if the boundaries will withstand the hazards associated with the area." As documented in Fire Protection Evaluation No. 98/0002, the non-standard fire barrier configuration separating Fire Zone 0C702 and 0C706 is capable of withstanding the hazards of either area based upon the following: 1) substantial construction of the assembly, 2) 3-hour rated structural fire proofing material applied on each side of the double steel plate and I-Beam partition assembly, 3) low combustible loading in both areas, 4) area wide fire detection in both areas, 5) automatic suppression systems (total flood $C0_2$ & sprinklers) in 0C702

And 6) accessibility to manual hose streams and portable fire extinguishers in OC702 & 0C706. Therefore, this configuration is an acceptable fire area boundary. Rework of the fire proofing material is being done in accordance with approved design as presently identified in the SAR. The penetration numbering correction is an editorial change only. Thus, the ability to achieve and maintain safe shutdown conditions in the event of a fire, as presently analyzed in the UFSAR, has not been adversely affected.

Therefore, this change will not increase the probability or consequences of accidents or malfunction of equipment important to safety previously evaluated in the SAR. Also, this change 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. "Fire Rated Assemblies" are not addressed by Technical Specifications (TS). "Fire Rated Assemblies" are addressed in the Technical Requirements Manual (TRM) Section 6.2.8. The fire barrier addressed in this change is covered by TRM Section 6.2.8; however, the change only demonstrates the adequacy of the non-standard fire barrier configuration and restores missing fire proofing material. No fire barriers are being added or deleted; therefore, there will be no change to TS or TRM or the Bases for any TS or TRM.

Serial Number: 98-104-NPE Document Evaluated: LDC 1998-092

BRIEF DESCRIPTION OF CHANGE, TEST OR EXPERIMENT:

The Main Steam Line Radiation Monitors (MSLRM), which are used to detect dose rates in the MSL for possible fuel failures, have a HI-HI trip set at 3.0x Full Power Background and the allowable value is 3.6x Full Power Background. The HI alarm setpoint is at 1.5x Full Power Background. These setpoints will be limited by maximum values calculated in JC-Q1D17-K610-1, Rev. 1 based on the analytical limit provided in EAR 98-038. The existing setpoints for the HI-HI trip and the allowable value will be bounded by the values established in the setpoint calculation. The HI alarm will be bounded by a value ½ that of the HI-HI trip maximum. UFSAR Section 11.5 Table 11.5-1 and UFSAR Appendix 16B Section 1 Table TR3.3.6.1-2 will be changed by LDC 1998-092 to reflect the maximum setpoint values above.

REASON FOR CHANGE, TEST OR EXPERIMENT:

Hydrogen injection will increase the dose rate in the MSL. This change establishes a limit based on accident analyses to ensure that the setpoint is maintained conservative. The existing setpoints are based on full power background levels which are very low with respect to the limits considered.

SAFETY EVALUATION SUMMARY AND CONCLUSIONS:

The License Document Change (LDC) evaluated by safety evaluation concludes that the change does not involve an unreviewed safety question. Changing the setpoint values and allowable value to include a maximum value that can not be exceeded will ensure the analytical value is not exceeded during normal plant operation with hydrogen injection at full reactor power. The analytical value is conservative with respect to the accident conditions postulated. If the setpoint is reached, the trip will cause shutdown of the mechanical vacuum pump, offgas valve closure, and reactor water sample valve closure (Group 10 isolation). This isolation will protect the licensing acceptance dose limits.

The radiation monitors do not function to preclude the occurrence of any accident and none are created by inadvertent actuation. Therefore, the setpoint change to include a maximum value will not increase the probability or consequences of an accident previously evaluated in the SAR. Limiting the Setpoint of the HI-HI trip will not compromise the safety and non-safety related functions of the MSLRM system. The HI alarm and HI-HI trip setpoints will also be maintained high enough above background rates to preclude inadvertent actuation. Therefore, changing the setpoint to include a maximum value will not increase the probability of occurrence or consequences of a malfunction of equipment important to safety previously evaluated in the SAR and will not create the possibility for a malfunction of equipment important to safety of a different type than any previously evaluated in the SAR.

The MSLRM setpoint values are not mentioned in the Technical Specifications but are listed in the Technical Requirement Manual Table TR3.3.6.1-2. The HI-HI setpoint value is stated as being 3.0x Full Power Background. Setting a maximum value o the HI-HI setpoint will ensure that the analytical limit is not exceeded during normal plant operation with hydrogen injection before the radiation monitor trip occurs. This trip will protect the licensing acceptance dose limits for the applicable non-limiting accidents and infrequent events (e.g., rod drop accident). Therefore, the maximum value placed on the MSLRM setpoints will not reduce the margin of safety as defined in the basis for any Technical Specifications.

Serial Number: 98-105-NPE Document Evaluated: ER 97/0114-00-00

BRIEF DESCRIPTION OF CHANGE, TEST OR EXPERIMENT:

Changes authorized by ER 97/0114-00-00 are associated with changing the designation of N71 System bulk chemical storage tanks 1N711A005 and 1N71A006 from "Biocide Tank" to "Chemical Tank". Concurrent with this designation change, these tanks will be authorized for use as bulk chemical storage tanks for use in storing biocides as well as bulk chemical other than biocides. The changes proposed by ER 97/0114-00-00 do not authorize any specific change to the type or quantity of chemicals stored or used on site. However, the affected tanks may be used to store bulk chemical provided the storage and use of the specific bulk chemical has been reviewed and approved in accordance with applicable plant procedures or programs. Tanks 1N71A005 and 1N71A006 are currently installed equipment, thus there is no field work required to implement the changes authorized by this ER. However, tanks 1N71A005 and 1N71A006 will require labeling or placarding to identify what chemicals, if any, are stored in these tanks. The two tanks are and will continue to be connected to a permanently install piping header that allows the contents of the storage tanks to be added (metered) into the N71 System. The bulk chemicals stored in the tanks will be added into the N71 System, using the concurrently installed piping header, to improve the overall effectiveness of the water treatment program in use to minimize the potential for corrosion and fouling of N71 System piping and components.

REASON FOR CHANGE, TEST OR EXPERIMENT:

The GGNS Chemistry Department maintains a water treatment programs for the purpose of minimizing the corrosion and fouling of N71 System piping and components. The overall effectiveness of this program can be improved by using storage tanks 1N71A005 and 1N71A006 as bulk storage tanks for those chemical currently being added/injected from Intermediate Bulk Containers (IBC's). While the on-site use of IBC's is authorized and acceptable, the IBC's represent a logistical concern with respect to keeping an adequate supply of chemical on location to meet the plant's demand. The use of IBC's involves excessive personnel resources and represents a potential safety concern with respect to frequency handling of the IBC's. Thus, the proposed change is part of a comprehensive water treatment program plan which is intended to provide long term protection for the N71 System piping and components. In addition, the proposed action will reduce the Chemistry Department resources required for implementation of the N71 System water treatment program while minimizing potential personnel safety concerns associated with use of the IBC's.

SAFETY EVALUATION SUMMARY AND CONCLUSIONS:

This safety evaluation has concluded that the changes proposed by ER 97/0114-00-00 do not represent an Unreviewed Safety Question. The designation of 1N71A005 and 1N761A006 as "Chemical Tanks" in lieu of "Biocide Tanks" will necessitate a revision to UFSAR Figure 10.4-007A. While the proposed action authorizes these two tanks to be used as bulk chemical storage tanks, ER 97/0114-00-00 does not specify what chemicals will be stored in the tanks. In Section 2.2.3.1.2, the UFSAR states that "potentially hazardous chemicals stored on the GGNS site are under administrative controls, including the quantities and locations, and are evaluated as necessary to ensure no adverse affect on the safe operation of Unit 1". Prior to the on-site storage or use of water treatment chemicals, the necessary reviews are conducted by appropriate plant personnel. Prior to using 1N71A005 or 1N71A006 as storage tanks for chemicals other than biocides, the specific chemical(s) to be stored or used will be evaluated to ensure compliance with this UFSAR statement. Use of these tanks for N71 System treatment chemicals will minimize the potential for corrosion and fouling of the N71 System piping and components. The chemical(s) will be added to the N71 System at a rate which produces the desired

chemical concentration in the N71 System. The N71 System is a non-safety-related system and storage tanks 1N71A005 and 1N71A006 are non-safety-related components located outside the plant, near the plant's cooling tower. The proposed action does not represent any new types of failure mechanisms for plant equipment nor will the use of these tanks impact the probability of occurrence of any previously evaluated accident. Due to the location of these tanks and the function of the N71 System, the proposed action will not impact the (radiological) consequences associated with accidents previously analyzed. The on-site storage of bulk chemicals will be reviewed to verify there are no Main Control Room habitability concerns resulting from storage or use of the proposed chemical(s). The storage and use of bulk chemicals will be controlled in a manner that does not represent a conflict with requirements contained in the GGNS National Pollutant Discharge Elimination System (NPDES) permit. The proposed change associated with ER 97/0114-00-00 will not impact the contents of the GGNS Technical Specifications (TS) or the Technical Requirements Manual (TRM). Based on review of this subject, it has been concluded that the proposed change does not represent an Unreviewed Safety Question.

Serial Number: 1999-001-PSE Document Evaluated: TA # 99-0001

BRIEF DESCRIPTION OF CHANGE, TEST OR EXPERIMENT:

This safety evaluation addresses the operability concerns associated with supplying Temporary power from ESF Bus 16AB and BOP Bus 11HD to loads normally supplied by Bus 15AA. Temporary power is being supplied to loads as required by SOI 04-1-01-R21-15, see attached Table 1. For the duration of the temporary alteration, controlled drawings will be issued to show the temporary power feeds.

REASON FOR CHANGE, TEST OR EXPERIMENT:

ESF Bus 15AA, System R21, provides power to safety and non-safety related components and instrumentation. Required maintenance and cleaning of the 15AA ESF Bus requires that it be deenergized for approximately 24 - 48 hours. This work will be conducted while the reactor is in Mode 4.

SAFETY EVALUATION SUMMARY AND CONCLUSIONS:

The additional power requirements being placed on Bus 11HD are within the loading capabilities and no loading calculations are required. The addition of battery charger 1DK5 (approximately 55 kW) to ESF Bus 16AB translates to an increase of 3.5% during normal operation, no increase in a forced shutdown (LOP) condition, and no increase in a Loss-of-Coolant Accident (LOCA). The additional load will not adversely affect the reliability due to loading since the load profiles have accounted for additional load values. No components being supplied temporary power will be considered operable. In all cases power is being supplied as a matter of convenience and not plant safety. LCOs will be entered where applicable. For the duration of this temporary alteration, the following information in the UFSAR will be inaccurate: Table 8.3-9, Figure 8.3-010, Figure 8.3-010A, Figure 8.3-10B, Paragraph 8.3.2.1.1, and Paragraph 8.3.2.1.6. The conclusion of this safety evaluation is that no unreviewed safety question exists and that the Technical Specifications are not impacted or changed by the proposed work.

Serial Number: 1999-002-PSE Document Evaluated: TA # 99-0005

BRIEF DESCRIPTION OF CHANGE, TEST OR EXPERIMENT:

Use BOP powered LCC/MCC to supply temporary power to other BOP loads during Plant outages to meet additional power requirements or to keep necessary equipment running during bus outages for electrical maintenance. Temporary power is being supplied to loads as required by SOI 04-1-01-R21-14, see attached Table 1. For the duration of the temporary alteration, controlled drawings will be issued to show the temporary power feeds.

REASON FOR CHANGE, TEST OR EXPERIMENT:

BOP Bus 14AE, System R21, provides power to non-safety related components and instrumentation. Required 14AE BOP Bus current transformer inspections and replacements, as necessary, require that it be de-energized for approximately 24 to 36 hours. As a matter of convenience, BOP power is being provided to select BOP loads which constitutes a change to the facility since BOP LCC/MCC are identified on plant drawings. This work will be conducted while the reactor is in Mode 4.

SAFETY EVALUATION SUMMARY AND CONCLUSIONS:

Temporary power from BOP LCC or MCC breakers is required to supply additional power needs during refueling outages or system outages, and to provide power to necessary BOP loads during bus maintenance and inspection activities. During the upcoming BOP Bus 14AE outages, additional power requirements will be supplied temporarily from BOP Buses 12HE and 13AD. This safety evaluation only addresses BOP power and does not address power supplied from or to Class 1E circuits.

Service transformers #11 and #21 supply loads to buses 11R and 21R. ESF power is distributed to vital distribution Load Control Centers (LCC) and Motor Control Centers (MCC) through ESF transformers ESF 11 and ESF 21. BOP power from 11R and 21R is distributed to the LCC and MCC level via BOP transformers 11A, 11B, 12A, 12B, 13 and 23.

Each BOP LCC supply breaker has a long-time over-current delay trip and an instantaneous over-current trip (except radial well switchgear house) to protect the distribution system from fault conditions. Each transformer neutral has a long-time over-current relay for ground fault backup protection. The feeder breakers to the MCCs and to the individual loads off the LCC have a long-time and instantaneous over-current trips. The distribution system is therefore adequately protected from a fault that might occur from either a designed load or a temporary load. The conclusion of this safety evaluation is that no unreviewed safety question exists and that the Technical Specifications is not impacted or changed by the proposed work. NOTE: For the duration of this temporary alteration, the following information in the UFSAR will be inaccurate: Figure 8.1-001 - Main One Line Diagram.

Serial Number: 1999-003-PSE Document Evaluated: TA # 99-0002

BRIEF DESCRIPTION OF CHANGE, TEST OR EXPERIMENT:

This safety evaluation addresses the operability concerns associated with supplying temporary power from ESF Bus 15AA, and BOP Buses 11HD, 12HE and 14AE to loads normally supplied by Bus 16AB. Temporary power is being supplied to loads as required by SOI 04-1-01-R21-16, see attached Table 1. For the duration of the temporary alteration, controlled drawings will be issued to show the temporary power feeds.

REASON FOR CHANGE, TEST OR EXPERIMENT:

ESF Bus 16AB, System R21, provides power to safety and non-safety related components and instrumentation. Required maintenance and cleaning of the 16AB ESF Bus requires that it be deenergized for approximately 24 - 48 hours. This work will be conducted while the reactor is in Mode 4.

SAFETY EVALUATION SUMMARY AND CONCLUSIONS:

The additional power requirements being placed on Buses 11HD, 12HE and 14AE are within the loading capabilities and no loading calculations are required. The addition of battery charger 1DL5 (approximately 55 kW) to ESF Bus 15AA translates to an increase of 3.5% in normal operation, no increase in a forced shutdown (LOP) condition, and no increase in a Loss-of-Coolant Accident (LOCA). The additional load will not adversely affect the reliability due to loading, since the load profiles have accounted for additional load values. No components being supplied temporary power will be considered operable. In all cases power is being supplied as a matter of convenience and not plant safety. LCOs will be entered where applicable. For the duration of this temporary alteration, the following information in the UFSAR will be inaccurate: Table 8.3-9, Figure 8.3-010, Figure 8.3-010A, Figure 8.3-10B, Paragraph 8.3.2.1.1, and Paragraph 8.3.2.1.6. The conclusion of this safety evaluation is that no unreviewed safety question exists and that the Technical Specifications are not impacted or changed by the proposed work.

Serial Number: 1999-004-PSE Document Evaluated: TA # 99-0005

(Safety Evaluation is Revised)

BRIEF DESCRIPTION OF CHANGE, TEST OR EXPERIMENT:

Use BOP powered LCC/MCC to supply temporary power to other BOP loads during Plant outages to meet additional power requirements or to keep necessary equipment running during bus outages for electrical maintenance. Temporary power is being supplied to loads as required by SOI 04-1-01-R21-14, see attached Table 1. For the duration of the temporary alteration, controlled drawings will be issued to show the temporary power feeds. Revision 01 of this Safety Evaluation allows circuit breakers 02, 07, and 20 in addition to circuit breaker 29 identified per table I of Revision 00 to be energized while temporary power is applied to BOP power panel 14B12. The temporary power source identified per Revision 00 of this Safety Evaluation is adequate to energize the additional loads. No changes to Temporary Alteration 99/0005 with the exception of closing power panel 14P12 circuit breakers 02, 07 & 20 will be allowed by Revision 01 of this Safety Evaluation.

REASON FOR CHANGE, TEST OR EXPERIMENT:

BOP Bus 14AE, System R21, provides power to non-safety related components and instrumentation. Required 14AE BOP Bus current transformer inspections and replacements, as necessary, require that it be de-energized for approximately 24 to 36 hours. As a matter of convenience, BOP power is being provided to select BOP loads which constitutes a change to the facility since BOP LCC/MCC are identified on plant drawings. This work will be conducted while the reactor is in Mode 4.

SAFETY EVALUATION SUMMARY AND CONCLUSIONS:

Temporary power from BOP LCC or MCC breakers is required to supply additional power needs during refueling outages or system outages, and to provide power to necessary BOP loads during bus maintenance and inspection activities. During the upcoming BOP Bus 14AE outages, additional power requirements will be supplied temporarily from BOP Buses 12HE and 13AD. This safety evaluation only addresses BOP power and does not address power supplied from or to Class 1E circuits.

Service transformers #11 and #21 supply loads to buses 11R and 21R. ESF power is distributed to vital distribution Load Control Centers (LCC) and Motor Control Centers (MCC) through ESF transformers ESF 11 and ESF 21. BOP power from 11R and 21R is distributed to the LCC and MCC level via BOP transformers 11A, 11B, 12A, 12B, 13 and 23.

Each BOP LCC supply breaker has a long-time over-current delay trip and an instantaneous over-current trip (except radial well switchgear house) to protect the distribution system from fault conditions. Each transformer neutral has a long-time over-current relay for ground fault backup protection. The feeder breakers to the MCCs and to the individual loads off the LCC have a long-time and instantaneous over-current trips. The distribution system is therefore adequately protected from a fault that might occur from either a designed load or a temporary load. The conclusion of this safety evaluation is that no unreviewed safety question exists and that the Technical Specifications is not impacted or changed by the proposed work. NOTE: For the duration of this temporary alteration, the following information in the UFSAR will be inaccurate: Figure 8.1-001 - Main One Line Diagram.

Serial Number: 1999-005-PSE Document Evaluated: TA # 99-0007

BRIEF DESCRIPTION OF CHANGE, TEST OR EXPERIMENT:

This Temp Alt will connect Construction Water to the supply of a temporary mobile demineralized trailer to provide a demineralized water source for filling the Demineralized Storage Tank. A fire hose will be connected from a Construction Water connection in the Makeup Water Treatment Building to the supply of a mobile demineralized trailer. The outlet of the trailer will be connected to the SP21F077 valve (shown on SAR Figure 9.2-11) which will be disassembled and adapted to accept a fire hose connection.

REASON FOR CHANGE, TEST OR EXPERIMENT:

Normally, demineralized storage water is supplied by a permanent mobile makeup water trailer. A larger capacity unit will temporarily be required to support plant startup after a forced outage.

Safety Evaluation summary and conclusions:
The changes made by this Temp Alt will not compromise any existing safety-related system, structure, or components. The proposed changes will not affect the ability to maintain the reactor in a safe shutdown condition.

As stated in Section 9.2.3.3, the Makeup Water Treatment 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. The P21 system has been evaluated in Section 3.6 of the SAR for moderate energy pipe breaks. The hose connections and temporary hose will be in the Water Treatment Building and will not be in the vicinity of safety-related equipment. Additionally, the routing of the fire hose will not create any II/I seismic concerns. Because the mobile makeup water trailer contains a 50 gallon propane tank, the trailer will be located such that it will not pose a fire or explosion hazard to a safety related component or facilty nor have a detrimental affect on Control Room Habitability.
Section 9.2.3.3 states that the P21 system has been designed to preclude the entry of potentially radioactive water in the system. This Temp Alt will use Construction Water as a makeup source to the Demineralized Storage Tank. Construction water is a non-radioactive source and poses no threat of contaminating the system with radioactive water.

Serial Number: 1999-006-NPE Document Evaluated: ER 1998-0391-00-00

BRIEF DESCRIPTION OF CHANGE, TEST, OR EXPERIMENT:

As identified in ER 98-0391-00-00, this change accepts-as-is or provides repair instructions for the following: (1) Three specific internal conduit fire seals which utilize RTV-108 silicone (Accept-As-Is) and (2) Thirty-five specific conduit configurations requiring internal conduit fire or smoke seals where the first opening from the penetration is inaccessible and the next accessible opening is used to determine the need for a fire/smoke internal conduit seal based on criteria in Note 9 on Drawing M-0800D (31 were Accept-As-Is & 4 were Repair). 1) RTV-108 silicone is not an approved seal material for fire boundaries. Specifically, RTV-108 silicone has been evaluated and determined to be acceptable as an internal conduit fire seal in the following conduits/penetration: 1BDRNS56/CE-78C on the 0C302 side of the fire barrier, 1BBRWQ02/CE-272CA on the 0C307 side of the fire barrier and 1BBRWQ01/CE-273CA on the 0C307 side of the fire barrier. This change does not make a generic acceptance of the use of RTV-108 as an approved material for sealing internal conduits for fire separation requirements. 2) Thirty-five specific locations are evaluated where the first conduit opening is inaccessible due to being wrapped with or obstructed by a fire barrier wrap system (Thermo-Lag) or a physical obstruction (structural steel fireproofing or other plant equipment) prevents access to the first opening. The specific inaccessible conduit/penetrations are as follows: 1BDRNS514/CE-12C; 1BBRNR42, 1BBRNR43, 1BBRNR45, 1BDRN61R, 2BDRO600 & 4 unscheduled conduits / CE-35C; Unscheduled 2″ Conduit / CE-113C; 1BBRNR28/CE-234CA; 1BBRNR07/CE-235CA; 1BBRNR08/CE-242CA; 1BBRNR24/CE-245CA; 1BBRNR25/CE-248CA; 1BBRNR06/CE250CA; 1BBRNS04 & ¾″ Unscheduled Conduit / CE-252CA; 2″ Unscheduled Conduit / CE-261CA; 1BDX675/CE-267CA; 1BARN630, 1BBRNS04 & 1BERS6ZB/CE-267D; 1BBRNR34/CE-271CA; 1BARNS22 & 1BDRNS53/CE-278D; 1BDRNS68/CE-281D; 1BARNQ16/CE-282CA; 1BDRS128/CE292CA; 1BDRS616, 1BDRS627 & 1BDRS628/CE-296CA; 1BARNQ41/CE-307CA and 1BARNQ41/CE-308CA. Thirty-one of these configurations did not require an internal conduit seal at the next opening because the next opening was outside the distance identified in Note 9 on Drawing M-0800D that would require an internal conduit seal (Accept-As-Is). Four of the configurations require an internal conduit seal at the next opening because the opening was located within the distance from the barrier that would require a seal per Note 9 on Drawing M-0800D (Repair). This change does not make a generic acceptance for not sealing the first opening from the penetration based on inaccessible issues.

REASON FOR CHANGE, TEST, OR EXPERIMENT:

GGNS design for internal sealing of conduits for fire and smoke is provided in Note 9 on Drawing M-0800D. The need and material type of an internal conduit seal is based on the conduit diameter, cable fill, and the distance between the fire barrier and the first opening in the conduit. RTV 108, which is installed in the three conduit configurations identified above, is not an approved material for sealing conduits which penetrate fire barriers. Due to the physical characteristics of the cured RTV 108 material, removal of the internal conduit seal material is difficult and would most likely lead to cable damage created during the seal removal process. Therefore, the acceptability of the existing RTV seals has been evaluated. In the thirty-five locations identified above, the first opening was found to be inaccessible due to either physical interferences with other equipment or the fact that the conduit is enclosed in or obstructed by a fire barrier wrap system (Thermo-Lag). This fire barrier wrap system is installed to meet Appendix R to 10CFR50 separation requirements. The requirement in Note 9 on Drawing M-0800D to seal the first opening was not verified in these inaccessible conduit configurations. Therefore, the acceptability of these inaccessible configurations has been evaluated.

SAFETY EVALUATION SUMMARY AND CONCLUSIONS:

The changes made by this ER accepts-as-is or provides repair instructions for: (1) three specific internal conduit fire seals which utilize RTV-108 silicone and (2) thirty-five specific conduit configurations

requiring internal conduit fire or smoke seals where the first opening from the penetration is inaccessible and the next accessible opening is used to determine the need for a fire/smoke internal conduit seal based on criteria in Note 9 on Drawing M-0800D. These specific internal conduit seal configurations have been evaluated and determined to maintain the fire resistance rating requirements of the barriers as presently analyzed in the SAR. Therefore, this change will not increase the probability or consequences of accidents or malfunction of equipment important to safety previously evaluated in the SAR. Also, this change 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. Also, this change 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. "Fire Rated Assemblies" are not addressed by Technical Specifications (TS). "Fire Rated Assemblies" are addressed in the Technical Requirements Manual (TRM) Section 6.2.8. Changes made by this ER involve internal sealing of electrical conduits penetrating "Fire Rated Assemblies" addressed by the TRM; however, these changes have been evaluated and determined to provide an adequate conduit seal and to maintain the 3-hour rating of the fire barriers. Therefore, there will be no change to TS or TRM or the Bases for any TS or TRM.

Serial Number: 1999-007-NPE Document Evaluated: ER 1996-0571-00

BRIEF DESCRIPTION OF CHANGE, TEST, OR EXPERIMENT:

Revision 1 to this safety evaluation is being issued to address Safety Review Committee Subcommittee #1 comments as documented in GIN 97/02392. Specifically, be brief and concise in the Executive Summary, provide more descriptive information in answers to the questions, and address questions 5, 6, & 7 for continued use of specific electrical cables as identified below.

ER 96-0571-R00 provides the final disposition to MNCR 92-0221, Supplement 1 & 2 (Thermo-Lag fire barrier design). The two changes addressed in this evaluation are: (1) acceptability of the fire barrier design, which utilize Thermo-Lag materials, for protection of two openings through 3-hour rated fire area boundary barriers, and (2) acceptability of continued use of electrical cables 1BB661111, 1BB661121, 1BB641011, 1BB661011, 1BB641012, 1BB661012 which, prior to RF08, were enclosed in a Thermo-Lag fire barrier enclosure that required ampacity derating of enclosed cables slightly greater than the available ampacity margin for the above listed cables.

REASON FOR CHANGE, TEST, OR EXPERIMENT:

Thermo-Lag materials are used at GGNS for two purposes: (1) provide fire rated enclosures for redundant safe shutdown electrical cables & (2) provide protection for two openings through 3-hour rated fire area boundary barriers required for compliance with 10CFR50, Appendix R. All changes necessary to restore compliance with 10CFR50, Appendix R for Thermo-Lag assemblies enclosing electrical cables are addressed in MCP's 94-1062 (SE# 95-0073-R00) and 94-1063 (SE# 96-0022-R00). The only two remaining issues associated with Thermo-Lag fire barriers are addressed in this safety evaluation.

Fire area boundary barriers, described in the SAR as 3-hour rated barriers, separating Fire Zone 0C214 (Fire Area 30) from Fire Zone 0C217 (Fire Area 26) and Fire Zone 0C217 from Fire Zone 0C303 (Fire Area 42) have openings which are protected with non-standard fire barrier configurations that utilize Thermo-Lag materials. Since these non-standard fire barrier configurations do not have a quantifiable fire resistance rating, they have been evaluated for acceptability based on the hazards in the area. In addition, electrical cables 1BB661111, 1BB661121, 1BB641011, 1BB661011, 1BB641012, 1BB661012 (cable tray 1BBTNR60) were wrapped in a 1-inch thick Thermo-Lag fire barrier enclosure until RFO8. These electrical cables were determined to require an additional ampacity derating of 48% due to the fire barrier wrap system. These cables do not have a 48% ampacity margin. The 1 inch thick Thermo-Lag fire barrier system on these cables was replaced in RFO8 with a wrap system which has an ampacity derating factor which is less than the ampacity margin for the listed cables. Therefore, the acceptability of these cables for continued use due to their pre-RFO8 configuration has been evaluated.

SAFETY EVALUATION SUMMARY AND CONCLUSIONS:

License condition 2.C.41 states GGNS shall implement and maintain in effect all provision of the approved Fire Protection Program. It goes on to state changes to the approved Fire Protection Program can be made without prior approval of the Commission if those changes would not adversely affect the ability to achieve and maintain safe shutdown in the event of a fire. Therefore, from the fire protection standpoint the base for evaluation is "no adverse effect on the ability to achieve and maintain safe shutdown".

As documented in Fire Protection Evaluation No. 96/0001, the non-standard fire barrier configuration separating Fire Zone 0C214 (Fire Area 30) from Fire Zone 0C217 (Fire Area 26) and Fire Zone 0C217 from Fire Zone 0C303 (Fire Area 42) are capable of withstanding the hazards of either area. Therefore, this configuration is an acceptable fire area boundary for the areas listed above.

Evaluations documented in ER 96/0571-R00 assert that the electrical cables identified above are acceptable for continued use based on the following: 1) no visible cable damage, 2) past insulation resistance testing for all cables except cable 1BB661071 (no previous insulation resistance test found) provides no indication of insulation degradation or potential insulation failure, & 3) cable 1BB661071 has only a slightly deficient margin (46.7% ampacity derating margin vs. 48%) which is considered to be bounded by conservatisms in the derating methodology. Therefore, these cables are acceptable for continued use.

Therefore, this change will not increase the probability or consequences of accidents or malfunction of equipment important to safety previously evaluated in the SAR. Also, this change 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. No change to the TS or bases for any TS is necessary.

Serial Number: 1999-008-NPE Document Evaluated: ER 1999-0034-00-00

BRIEF DESCRIPTION OF CHANGE, TEST, OR EXPERIMENT:

Perform crimping and furmanite of drain bypass line (1”-GBD-1145 and 1”-HBD-1759) for separator N62D009A to control leakage due to a detected hole in bypass valve N62F146B body.

REASON FOR CHANGE, TEST, OR EXPERIMENT:

A hole has been detected in valve N62F146B. The hole permits air to enter the system even with bypass valve closed. This inleakage should be controlled to prevent unwanted challenge to condenser vacuum.

SAFETY EVALUATION SUMMARY AND CONCLUSIONS:

This is a temporary change to control detected leakage. The original design configuration for the drain bypass line will be restored at the first convenient opportunity such as a forced outage or during RF10. The operation of the condenser air removal system (N62) will not be affected as the bypass line remains closed during normal operation and is not required for safe shutdown of the plant. Normal flow is through the restriction orifice and the purpose of the bypass line is only to allow for potential on-line maintenance of the drain orifice. The drain bypass line leakage condition will be eliminated by this temporary change. Piping structural integrity will remain assured with the modified configuration. The information in the SAR (Figure 10.4-001) will not be affected as this is a temporary change. This temporary change will not alter and will not have the potential to alter the information, operation, function or ability to perform the function of a system, structure, or component described in the SAR.

Serial Number: 1999-009-CHM Document Evaluated: ODCM Revision 22

BRIEF DESCRIPTION OF CHANGE, TEST OR EXPERIMENT:

ODCM Revision 22 is required to implement an upgrade to the offsite dose calculation and management software. The safety evaluation addresses two areas where the updated software differs from current software: **1**. Methodology for calculation of a liquid radwaste (LRW) discharge radiation monitor trip setpoint in the absence of gamma emitting nuclides and **2**. Upgrade of the nureg 0133 methodology for calculating offsite dose from iodines, particulates and tritium in gaseous releases to the methodology of RG. 1.109.

REASON FOR CHANGE, TEST OR EXPERIMENT:

The ODCM must describe the methodology used for effluent radiation monitor setpooint calculations and for the calculation of offsite doses.

The ODCM presently does not contain methodology for the cacluation of the LRW trip setpoint in the absence of gamma emitting nuclides. The current methodology for these cases was supplied as part of the original software package implementing radioactive releases and ODCM calculations and is administratively controlled. The change to setpoint methodlogy increases the conservatism associated with the discharge of LRW where no gamma emitting nuclides are detected. In the analysis of waste water for discharge, a minimum detectable activity (MDA) value is assigned to specified nuclides if they are absent from the gamma spectrum. The MDA value is a concentration (μCi/ml), above which, with a 95% confidence level, you are assured the nuclide does not exist. The current method for setpoint caculation in the absence of gamma emitters uses MDA concentrations as the basis for the setpoint, calculating a setpoint as if the specified nuclides were present at their MDA levels. The method is valid in that it utilizes actual sample parameters to generate a setpoint. The setpoint can range from 5,000 to 20,000 cpm above background. The proposed change will insert a fixed value of 2,700 cpm above background in the absence of gamma emitters. The basis for 2,700 cpm is 90% of the countrate associated with the most limiting of the principal gamma emitters specified in ODCM Table 6.11.1-1. Cesium 137, at 90% of its effluent concentration limit, yields 2,700 cpm above background. The proposed change will provide consistency in the setpoint used when gamma emitters are absent and will fulfill the requirement to restrict effluent nuclide concentrations to the limits of ODCM LCO 6.11.1. The proposed change will not affect remaining setpoints associated with LRW releases : waste tank maximum, and dilution minimum, flow rate setpoints .

The ODCM currently utilizes NUREG 0133 methodolgy for the calculation of offsite organ doses resulting from the release of tritium, iodines and particulates in gaseous form. The organ dose calculated is a composite of the highest dose to any organ. This methodology is simple to implement in that dose factors are pre-sorted to ensure the highest organ dose is assigned to each nuclide but does not report true individual organ doses. The proposed change is made possible by availability of more sophisticated dose calculation software . The change will calculate the dose to each organ and report the organ with the highest dose. The resultant doses will be lower than those currently calculated but will be more accurate. The trend towards increased accuracy in dose calculations is desirable from a regulatory and liability standpoint. The dose calculation algorithm still retains features of conservatism including use of highest annual average meterological parameters.

SAFETY EVALUATION SUMMARY AND CONCLUSIONS:

The proposed activity will not result in a reduction in the margin between calculated doses and 10CFR50 Appendix I limits (as listed in TS 5.5.4). The change to LRW setpoint methodology in the absence of Gamma Emitters is a conservative.

Serial Number: 1999-010-NSR Document Evaluated: LDC 98-060

BRIEF DESCRIPTION OF CHANGE, TEST OR EXPERIMENT:

The Area Radiation Monitor (ARM) alarm setpoints, as described in UFSAR (Updated Final Safety Analysis Report) Table 12.3-3, for the Dryer Storage Pool (1 D21-K626), Separator Storage area (1 D21-K627), Containment Fuel Area-North (1 D21-K628), and Containment Fuel Area-South (1 D21-K629) are being changed from ≤2.5 to ≤15 mR/hr. This will also involve a change to the setpoint for Function 9.a.3) (Dryer Storage Area Monitor) from ≤2.5 to ≤15 mR/hr as outlined in UFSAR/TRM Table 6.3.1-1. Detail is also being added to UFSAR/TRM Table 6.3.1-1 via a new Note (g) and is being viewed as an operator enhancement. Note (g) will describe ARM detector nNumbers (1D21-K626, K627, and K629) which can be credited to meet the criticality accident monitoring requirements of 10CFR70.24(a)(1). This crediting of specific ARMs as meeting 10CFR70.24(a)(1) requirements addresses deficiencies noted in Condition Report GGCR 1998-0365-00. The deficiency involved blocking of ARM detector 1D21-K628 during refueling outages; it is currently a 10CFR70.24(a)(1) credited criticality accident monitor. ARM Detectors 1D21-K626, K627, and K629 will be designated as 10CFR70.24 criticality accident monitors since they are not blocked during refueling outages and are identical to detector 1D21-K628.

REASON FOR CHANGE, TEST OR EXPERIMENT:

As documented on Grand Gulf Condition Report GGCR 1998-0365-00, this change is necessary due to increased background radiation levels which has caused at least one detector to be in continuous alarm and to address a detector inoperability issue related to storage of reactor components (Drywell Head) during refueling outages.

Setpoint Change:

As stated in the Condition Report (CR), an ARM in continuous alarm can desensitize workers to alarms. Also, an ARM in alarm is no longer an effective warning device if it is alarming due to increased background radiation levels and not abnormal radiation levels. To alleviate this concern, it is proposed that the radiation alarm setpoints be raised from ≤2.5 to ≤15 mR/hr for ARM detectors 1D21-K626, K627, K628, and K629. Raising the setpoint of the affected ARMs will not prevent the detectors from meeting the requirements outlined in UFSAR Section 12.3.4.1 or the criticality accident monitoring requirements of 10CFR70.24 (detectors 1D21-K626, K627, and K629 meet the criticality monitor requirements). The setpoint chosen is high enough to minimize spurious alarms due to background fluctuations yet low enough to alert personnel to abnormal or increasing radiation levels at which time Health Physics personnel would perform investigative surveys.

Redesianation of Criticality ARMs:

Another deficiency noted in the CR was that a currently credited 10CFR70.24 criticality monitor (1D21-K628) could be rendered inoperable (blocked) when the Drywell Head is stored in its refueling position. This then requires operations personnel to enter LCO 6.3.1 since blocking of an ARM detector renders it INOPERABLE. Undesignating this detector as one that meets criticality monitor requirements will alleviate this concern. Another detector (1D21-K627 -Separator Storage Pool), that does meet 10CFR70.24 requirements, will replace detector 1D21-K628. Use of detector 1D21-K627 , as a criticality monitor, is acceptable since it is identical to 1D21-K628 and meets 10CFR70.24 requirements. Redesignating which ARMs can be credited as meeting 10CFR70.24 requirements will avoid the blocking deficiency as noted in Condition Report 1998-0365-00.

SAFETY EVALUATION SUMMARY AND CONCLUSIONS:

Changing the UFSAR Table 12.3-3 radiation alarm setpoint from ≤2.5 to ≤15 mR/hr for detectors 1D21-K626, K627, K628, and K629 will not prevent the detectors from meeting the requirements outlined in UFSAR. Review of Chapter 15 accidents indicates that increasing ARM setpoints will not cause, create, or affect accidents as outlined in Chapter 15 nor affect the NRC requirements as outlined in UFSAR Section 12.3.4.1. Designating ARM Detector 1D21-K627 as a 10CFR70.24 criticality accident monitor is acceptable since the detector meets the criticality monitoring detection limits as outlined in 10CFR70.24(a)(1). These ARMs do not limit radiological releases and are not needed to ensure 1OCFR100 offsite radiological dose limits are preserved. The ARM system is not essential for safe shutdown of the plant, and it serves no active emergency shutdown function during plant operation. The ARM system has no ties to any systems important to safety and will not increase the probability of occurrence of a malfunction of equipment important to safety or affect any margins of safety. These changes are acceptable based on the evaluation performed.

Serial Number: 1999-011-NPE Document Evaluated: ER 1998-0615-00-00

BRIEF DESCRIPTION OF CHANGE, TEST, OR EXPERIMENT:

ER 1998-0615-00 documents the acceptability of a non-standard fire barrier design utilized as part of the fire wall assembly separating Fire Zone 0C702 (Upper Cable Spreading Room. Control Building El. 189'-0") and Fire Zone 0C712 (HVAC Room. Control Building El. 1 89'-0"). In addition, openings through this nonstandard fire barrier configuration are being sealed with steel plate or steel angle and 3-hour rated structural steel fireproofing is then applied to both sides of the steel.

REASON FOR CHANGE, TEST, OR EXPERIMENT:

The fire protection program and UFSAR describe the wall separating the Upper Cable Spreading Room (Fire Zone 0C702. Fire Area 47) from the HVAC Room (Fire Zone 0C7 12. Fire Area 47) on elevation I 89'-0" of the Control Building as a 2-hour rated fire barrier. The construction of the portion of this wall above the 200'-7" elevation utilizes a nonstandard fire barrier configuration that does not have a quantifiable fire resistance rating. In accordance with Generic Letter 86-10 an evaluation of this barrier has been performed and documented in Fire Protection Evaluation No. 98/0003 to determine if the existing barrier is adequate for the hazards in the area. This change documents that evaluation and makes necessary Fire Protection Program changes to reflect the non-standard fire barrier configuration.

In addition, gaps were left between the steel angle (installed between the bottom of the I-Beam and the top of the concrete wall) and the adjoining fire barriers on each side of the non-standard fire barrier configuration. This construction created a through hole in the non-standard fire barrier assembly. This ER installs steel plate and/or steel angle at these locations to seal the through hole and installs 3-hour structural steel fire proofing material on each side of repaired area.

SAFETY EVALUATION SUMMARY AND CONCLUSIONS:

License Condition 2.C.41 allows GGNS to make changes to the approved Fire Protection Program through the *50.59* process if those changes do not adversely affect the ability to achieve and maintain safe shutdown in the event of a fire. Therefore. from the fire protection standpoint the bases for evaluation is "no adverse effect on the ability to achieve and maintain safe shutdown in the event of a fire." Repair of the holes through this barrier and rework of the fire proofing material is being done in accordance with approved design as presently identified in the SAR. Generic Letter 86-10 Enclosure No.1, Interpretation No. 4 states:

"Where fire area boundaries are not wall-to-wall, floor-to-ceiling boundaries with all penetrations sealed to the fire rating required of the boundaries, licensees must perform an evaluation to assess the adequacy of fire boundaries in their plants to determine if the boundaries will withstand the hazards associated with the area." As documented in Fire Protection Evaluation No. 98/0003, the nonstandard fire barrier configuration separating Fire Zone 0C702 and 0C712 is capable of withstanding the hazards of either area. Therefore, this configuration is an acceptable fire barrier. Thus, the ability to achieve and maintain safe shutdown conditions in the event of a fire, as presently analyzed in the UFSAR, has not been adversely affected.

Therefore, this change will not increase the probability or consequences of accidents or malfunction of equipment important to safety previously evaluated in the SAR. Also, this change 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. "Fire Rated Assemblies" are not addressed by Technical Specifications (TS). "Fire Rated Assemblies" are addressed in the Technical Requirements Manual (TRM) Section 6.2.8. The fire barrier addressed in this change is covered by TRM Section 6.2.8; however, the change only demonstrates the adequacy of the non-standard fire barrier configuration. No fire barriers are being added or deleted; therefore, there will be no change to TS or TRM or the Bases for any TS or TRM.

Serial Number: 1999-012-PSE Document Evaluated: PAP 17-S-02-30

BRIEF DESCRIPTION OF CHANGE, TEST OR EXPERIMENT:

Neutron Transmission Testing (known as “Blackness Testing”) is conducted periodically at GGNS to monitor the spent fuel pool fuel storage racks for evidence of degradation in the boraflex neutron absorber. Boraflex is a polymer material laced with boron which is sandwiched between structural stainless steel plates creating individual storage cell panels. Cells are honey-combed together to create the fuel storage racks. In high density storage racks such as those at GGNS, spacing alone is not sufficient to maintain subcriticality. The boraflex ensures that rack k-eff remains <.95 as required by Technical Specifications when fuel is stored in the racks and flooded with unborated water.

In Blackness Testing, a fast neutron source and 4 neutron detectors are inserted into an empty storage cell which has been previously exposed to a high radiation dose from freshly discharged spent fuel. The fast neutrons leak from the cell where some are thermalized and attempt to re-enter the cell. In the absence of gaps in the boraflex, virtually all of the thermal neutrons will be captured by the boron absorber atoms. If gaps are present, neutrons may then pass into the cell. The neutron detectors in the cell will register a fraction of those that successfully re-enter. Using this process, the size and distribution of any gaps in the boraflex material can be measured. The source and detectors are moved vertically in every alternate cell to survey each panel in a specified rack test area. In addition, a special simulated storage cell has been fabricated with boraflex gaps of known sizes and locations intentionally built into the cell panels. This cell will be suspended in the pool and used to calibrate the neutron instruments for improved accuracy in gap measurement resolution.

Testing will be conducted by a qualified contractor as it has been for 6 previous test campaigns under the supervision of GGNS personnel. The contractor will provide test equipment and personnel, with GGNS providing Radiation Protection and other needed support. Testing will be done under GGNS programs and approved procedures. The test is non-destructive and does not affect storage rack integrity or qualification as described and assumed in the FSAR.

REASON FOR CHANGE, TEST OR EXPERIMENT:

Boraflex testing has been conducted once per cycle at GGNS since 1989 in response to NRC concerns over industry evidence of boraflex degradation. An exception to this (with NRC approval) was Cycle 9 during which the GGNS criticality analysis was updated to ensure previous test results were bounded. Testing must resume in Cycle 10 and beyond until satisfactory evidence of gap equilibrium is found. The Blackness Testing process has been previously evaluated and approved per References 1 and 2. The purpose of this evaluation is to update and clarify the previous evaluations to reflect some minor changes to the testing procedure and incorporate the use of the calibration test cell. This evaluation therefore supercedes all previous evaluations. Other than this, no significant changes to the process or approach used in the previous campaigns are being made. The issues of the previous evaluations will also be reviewed for completeness.

SAFETY EVALUATION SUMMARY AND CONCLUSIONS:

Blackness Testing does not involve the movement of fuel or the use of the fuel handling platform hoists. It does not impact the ability of the fuel storage racks to maintain fuel in a subcritical condition, and Technical Specification load restrictions will be followed. No changes to Technical Specifications are necessary. The use of the test equipment is bounded by existing fuel handling accident analyses. No heavy loads are being lifted, and the light loads restrictions contained in plant procedures will be followed. No systems needed to mitigate the consequences of an accident or equipment malfunction are

being affected. None of the events previously analyzed are more likely to occur due to Blackness Testing. No new types of accidents or equipment malfunctions are created by the introduction or use of the vendor's test equipment. All assumptions used in establishing Technical Specification margins of safety remain valid. Thus, there are no unreviewed safety questions related to this test.

Serial Number: 1999-013-NPE Document Evaluated: ER 99-0105-00-00

BRIEF DESCRIPTION OF CHANGE, TEST, OR EXPERIMENT:

Design Engineering evaluation of the scaffolding left in place in place at Elevation 185' Azimuth l40°-150° inside containment.

REASON FOR CHANGE, TEST, OR EXPERIMENT:

The scaffolding is used for a quarterly surveillance **for** G41 and is located in **a** high radiation and contamination area.

SAFETY EVALUATION SUMMARY AND CONCLUSIONS:

Design Engineering has evaluated the scaffolding currently in place at Elevation 185' Azimuth l40°-150° inside containment. Specific hazard evaluations are as follows: High Energy Line Break, Moderate Energy Line Break, Internal Missiles, Hydrogen Control, Hydrogen Generation, and Foreign Material Exclusion. Design Engineering has evaluated the scaffolding currently in place at Elevation 185' Azimuth Azimuth l40°- 150° inside containment. The scaffolding is not in any zone of influence for any postulated high-energy line break, is not in the path of internally generated missiles and cannot become a missile. The scaffolding is in accordance with the appropriate Seismic II/I criteria. The size and location of this scaffolding would not significantly affect flow paths or interfere with the function of hydrogen igniters. The total contribution of the scaffolding material to the total area of galvanized material in containment as reported in the FSAR is conservatively 0.011 percent, therefore, any contribution to hydrogen generation from this scaffold is negligible. Based on the absence of an HELB zone-of-influence and the fact that the scaffolding is made of hot-dip galvanized steel, the scaffolding material does not contribute to potential suction strainer clogging. Hot-dip galvanized steel is a plating material and not a typical paint coating. The failure mechanism for the hot-dip galvanized steel would not be equivalent to the failure of paint coatings. The hot-dip galvanized steel is chemically bonded with the scaffolding material and would not disassociate from the metal surface during design basis accident conditions. It is acceptable for this scaffolding to remain in its current position until the end of RFO11. The scaffolding must be removed or replaced with a permanent platform prior to startup from RFO11. There will be no adverse impact on the safe operation of GGNS by this scaffolding remaining in place until the end of RFO11.

Serial Number: 1999-014-NPE Document Evaluated: ER 96-0224-00-00

BRIEF DESCRIPTION OF CHANGE, TEST, OR EXPERIMENT:

ER 96/0224-00-00 authorizes installation of reduced bore size orifice plates in the recirculation line associated with Plant Service Water (PSW) pumps SP47C00lA and SP47C00lB installed at Radial Well No. 1. The recirculation line is used during well start-up and allows recirculation of well contents to the Mississippi River. The recirculation line is equipped with two orifice plates installed in series which prevent PSW pump "runout" (excessive flow rate). The equipment affected by the change is flow restricting orifice plates SP47D005A and SP47D007A, which are installed in the Radial Well No. 1 recirculation line. In addition to the orifice bore size change, minor piping changes are proposed for drain piping associated with Radial Well No. 1 recirculation header drain pipe (2"-JXD-2) and the PSW pump low flow setpoint (SP47R009A/B) will be reduced from 2000 gpm to 1600 gpm.

REASON FOR CHANGE, TEST, OR EXPERIMENT:

The existing orifice sizes were specified using 100% of the anticipated "new" well capacity and the maximum design PSW pump capacity. However, the well capacity is expected to slowly deteriorate with time. The orifice plate bore size change proposed by ER 96/0224-00-00 is due to this gradual loss in well capacity at Radial Well No. 1. Considering the deteriorated capacity of Radial Well No. 1, the PSW pump capacity can exceed the well capacity resulting in overpumping of the well. This concern is most prevalent during recirculation (start-up) of Well No. 1. To resolve this concern, it is proposed to change (reduce) the bore size of the Well No. 1 recirculation line orifice plates, thus minimizing the flow rate from this well during periods of well recirculation. The drain piping associated with the recirculation line at Radial Well No. 1 needs to be shortened to be "above grade level " to minimize soil erosion concerns. The low flow trip setpoint for SP47C001A/C001B will be reduced to minimize potential for unnecessary low flow trips of PSW pumps installed at Radial Well No. 1.

SAFETY EVALUATION SUMMARY AND CONCLUSIONS:

UFSAR Section 2.4.1.1 specifies a 4000-5000 gpm flow rate for the Radial Well Pumps while these pumps are operated in the "recirculation" mode. ER 96/0224-00-00 provides authorization to install orifice plates of reduced bore size which will result in recirculation flow rates for Radial Well No. 1 which are less than this previously specified flow rate. The affected UFSAR section will be revised to reflect installation and use of the new orifice plates at Well No. 1, thus a Safety Evaluation is required to support the proposed change and subsequent UFSAR change. The changes proposed in ER 96/0224-00-00 do not alter the function of the Radial Well (P47) System. As stated in UFSAR Section 9.2.10.3, the radial well system has no safety related function and failure of this system will not compromise any safety related structure, system, or component. The changes proposed by ER 96/0224-00-00 are not safety-related and will not alter the plant's response to anticipated transients or accidents as analyzed in the UFSAR. The radial well system is non radioactive and the proposed change will not impact the plants radiological effluents, on site or off site. Thus, based on the conclusions reached by this Safety Evaluation, the changes proposed by ER 96/0224-00-00 do not represent an Unreviewed Safety Question. An Environmental Review has been completed in conjunction with ER 96/0224-00-00 and based on that review, there are no Unreviewed Environmental Questions associated with the proposed changes.

Serial Number: 1999-015-NSR Document Evaluated: Amendment 11 of PSTG

BRIEF DESCRIPTION OF CHANGE, TEST OR EXPERIMENT:

Amendment 11 to the Grand Gulf PSTG revises the plant specific technical guidance as required to incorporate the generic BWROG Emergency Procedure and Severe Accident Guidelines, Revision 1. The PSTG is implemented through the Grand Gulf Emergency Procedures (EPs) and Severe Accident Procedures (SAPs). Use of two sets of procedures allows the EPs to provide guidance for conditions where adequate core cooling is believed to be assured and the SAPs to provide guidance for conditions where adequate core cooling cannot be assured.

REASON FOR CHANGE, TEST OR EXPERIMENT:

In GNRO-95/00043 and GNRO-98/00016 Grand Gulf committed to assess current capabilities to respond to severe accident conditions using Section 5 of NEI 91-04, Revision 1, "Severe Accident Issue Closure Guidelines" and to implement appropriate improvements identified in the assessment, within the constraints of existing personnel and hardware, no later than December 31, 1998. In GNRO-92/00157 Grand Gulf committed to actively pursue revisions to the BWROG Emergency Procedure Guidelines concerning MSIV venting to address Grand Gulf IPE insight 6.2.6. GNRI-96100135 documented NRC approval of BWROG EPG changes needed to address thermal hydraulic instability concerns during the EPG ATWS strategy. All of these issues are addressed in the BWROG EPG/SAG, Revision 1.

SAFETY EVALUATION SUMMARY AND CONCLUSIONS:

The EPG/SAGs impose various limits within which continued safe operation of the plant is ensured and beyond which certain actions may be required. While conservative, these limits have been derived using best-estimate engineering analysis rather than licensing models. Consequently, these limits are generally not as conservative as the limits specified in the plant's technical specifications and conformance with these guidelines does not necessarily ensure strict conformance with technical specifications or other licensing bases. This does not imply, however, that operation beyond technical specification limits is recommended. Rather, such operation may be required, and is now permitted, to mitigate certain degraded conditions. Certain interlocks and initiation logic must sometimes be bypassed to permit the execution of EPG/SAG steps. Such actions are a necessary part of the EPG/SAG mitigation strategy, but are generally authorized only when conditions may exist for which the interlocks or logic features were not designed. This concept has been reviewed and approved through generic SER by the NRC for all BWROG EPG revisions up to and including BWROG EPG, Rev. 4. The generic SERs have formed part of the bases for previous EOP implementation at Grand Gulf. As such this does not represent a change in the implementation of the Grand Gulf Emergency Procedures.

The EPG/SAGs are symptomatic in their approach and as such various strategies are utilized as observed plant conditions degrade. The actions prescribed in the EPG/SAGs attempt to establish and maintain long term adequate core cooling by submerging the core or operating below a limit that ensures peak cladding temperature remains below 1500° F. In general, actions to operate systems or equipment in a manner not assumed in the UFSAR are not prescribed until after the system or equipment has failed to perform its intended function. The symptomatic approach utilized by the EPG/SAGs requires that every effort be made to address any mechanistically possible condition, irrespective of the probability of occurrence, with appropriate guidance to minimize the impact on public health and safety. Thus under extremely degraded plant conditions entry into a strategy that would intentionally flood the primary containment may be required. Intentional flooding of the primary containment could require containment venting to maintain containment parameters within prescribed limits and therefore must be evaluated.

Containment venting is not assumed in any UFSAR Chapter 15 evaluation nor is it assumed in the plant design to meet General Design Criterion 50, Containment Design Basis. The actions prescribed in the EPG/SAGs were evaluated to determine if the intentional primary containment flooding strategy would

be entered under any UFSAR Chapter 15 condition and/or under GDC 50 conditions. If containment flooding strategy was required then an evaluation to determine if intentional containment venting was required would be necessary. If intentional containment venting was required within the UFSAR Chapter 15 or GDC 50 assumptions, an unreviewed safety question may exist.

Intentional flooding of the primary containment is only directed in the SAGs after assurance of adequate core cooling can no longer be maintained. An engineering analysis was performed to determine if the SAGs would be entered under any UFSAR Chapter 15 condition. The bounding condition was determined to exist after long term adequate core cooling is established by minimum ECCS following the large break LOCA (double ended recirculation line break). Entry into the SAGs is required when RPV water level cannot be restored and maintained above the Minimum RRV Steam Cooling Water Level (MSCRWL) (-192 in.). Engineering Report GGNS-98-0058 documents that under conditions described in the UFSAR Chapter 15, RPV water level, as indicated on the Fuel Zone Level Instrument, will indicate greater than the MSCRWL (-192 in), therefore the SAGs will not be entered. Thus under UFSAR Chapter 15 conditions prescribed guidance will be limited to that found in the EPGs and intentional flooding of the primary containment will not be directed. This ensures that no intentional primary containment venting will be prescribed thus the actions remain within the assumptions of the UFSAR Chapter 15.

GDC 50 requires that the reactor containment structure, including access openings, penetrations, and containment heat removal system shall be designed so that the containment structure and its internal compartments can accommodate, without exceeding the design leakage rate and with sufficient margin, the calculated pressure and temperature conditions resulting from any loss of coolant accident. This margin shall reflect consideration of energy from metal-water and other chemical reactions that may result from degradation down to the minimum as described in the UFSAR Chapter 15 but not total failure of emergency core cooling functioning. GDC 50 design considerations exceed the single failure criteria of UFSAR Chapter 15, as indicated by the required substantial core damage assumption requirement, and are representative of conditions where the SAG's guidance would be applicable and intentional flooding of the primary containment directed. An evaluation was performed to determine if primary containment venting would be required due to the intentional primary containment flooding guidance contained in the SAGs. An initial primary containment pressure of 9.9 psig (UFSAR Table 6.2-6, Loss of Coolant Accident Long Term Primary Containment Response Summary) was assumed. The containment was then flooded from the Technical Specification minimum suppression pool water level of 18.34 ft. to the upper primary containment flooding limit of 72 ft. Under these conditions primary containment pressure increased to 14.1 psig. This maximum value is less than the point where the SAG's guidance would direct intentional primary containment venting and is also less than the primary containment design pressure of 15 psig. Thus under the conditions of GDC 50 no intentional venting of the primary containment will be prescribed and compliance with GDC 50 is maintained.

Based on this evaluation it can be concluded that operation in accordance with the guidance found in the Grand Gulf PSTG will not cause operation of systems or equipment in a manner other than described in the UFSAR thus no unreviewed safety question exists.

Serial Number: 1999-016-NPE Document Evaluated: ER 97/0352-01-00

BRIEF DESCRIPTION OF CHANGE, TEST, OR EXPERIMENT:

The design value of the airflow of the LPCS Room Cooler is being increased from 8,000 cfm to 10,000 cfm based on airflow data taken in support of ER 97/0352-00-00.

REASON FOR CHANGE, TEST, OR EXPERIMENT:

Condition Report GGN-1997-0309 documented that the LPCS Room cooler Q1T51B002 failed its thermal performance test. As part of the corrective action of this CR, ER 97/0352-00-00 was issued to increase the airflow of the cooler to 9,100 cfm. Prior to implementing this modification, the as found airflow of the cooler was measured. This was found to be approximately 10,600 cfm based on TSTI 1T51-98-001-0-S test results. This exceeded the maximum design airflow for the cooler. ER 97/0352-00-00 was cancelled based on the flows being in excess of the flows required by that ER.

This ER, ER 97/0352-01-00, is being issued to revise the design airflow of the LPCS room cooler to 10,000 cfm and accept the as found airflow as the new design airflow for the LPCS room cooler.

SAFETY EVALUATION SUMMARY AND CONCLUSIONS:

This modification accepts the as found airflow of the LPCS Room coolers and establishes a new design airflow for the cooler that encompasses this value. The fan and motor RPM are not increased. No new components and no new failure modes are introduced. The additional heat removal capabilities of the LPCS room cooler will result in lower post accident temperatures for the LPCS room and in adjacent areas. The impact of the additional heat that will be removed by the cooler on the ultimate heat sink has been evaluated and determined to be acceptable.

It was concluded that increasing the design airflow of the LPCS room cooler will have no adverse impact upon the plant and will not compromise any safety related system or prevent safe reactor shutdown, prevent such equipment from operating as designed, or cause equipment important to safety to operate outside its design requirements.

Serial Number: 1999-017-NPE Document Evaluated: ER 96/0003-01-00
LDC 1999-054

BRIEF DESCRIPTION OF CHANGE, TEST OR EXPERIMENT:

The Upper and Lower Containment Personnel Airlocks will be modified to provide reinforcement to the Same Door seal mechanism pawl support rod to address the original nonconformance (CR 1997-0630). As determined in the original MNCR disposition, the electrical limit switch will be relocated from the door hinge to the upper right hand corner of the door to provide a positive means to signal the seal mechanism that the door is fully closed. CR 1999/0482 identified a potential concern for using a temporary mechanical test flange during airlock testing. This ER will provide a permanent mechanical test connection for use in airlock testing during plant operation. In addition, plant identified operational and maintenance enhancements to the airlocks will be provided based on past airlock performance. These enhancements include modifications to the airlock pneumatic system, Other Door mechanical interlock system, door seal pressure switch, and the airlock control panel system.

ER 96/0003-00-01 will provide the mechanical portion of the airlock modifications and ER 96/0003-01-00 will provide the electrical portion of the airlock modifications.

REASON FOR CHANGE, TEST OR EXPERIMENT:

This ER will provide the final recommended design disposition to CR 1997-0630 (Formerly Material Nonconformance Report (MNCR) No. 95/0256). This design will consist of reinforcement of the Same Door seal mechanism pawl support rod and relocating the door limit switch from the door hinge to the upper right hand corner of the door. An interim disposition of the MNCR repaired damage to the pawl support rod to full function in accordance with the requirements of the detailed design drawings. This ER will also provide a disposition to install a permanent mechanical test connection to be used for on line airlock testing, which will address the concerns in CR 1997/0482. The balance of the airlock modifications will enhance and improve airlock door operation and mitigate future maintenance work.

SAFETY EVALUATION SUMMARY AND CONCLUSIONS:

The modification of the Upper and Lower Containment Personnel Airlocks will meet the original design requirements of Specification 9645-C-153.0, Design Specification for Design, Furnish, Detail, Fabricate, and Deliver Personnel Locks, Equipment Hatch, Drywell Penetrations, and Drywell Head. The function of the airlocks is to provide required ingress and egress into and from containment during normal plant operations, while maintaining integrity of the containment boundary and to maintain containment integrity during accident conditions. This function will not be altered, affected or diminished in any fashion with the implementation of the modifications.

Airlock service, performance and operation will be improved. These improvements include additional controls on door seal inflation and deflation to mitigate the potential for seal damage and wear, the ability to conduct pressure tests and selective maintenance within the airlock chamber without the need to enter a Limited Condition of Operation (LCO) Condition C (TS 3.6.1.2), the capability to isolate the non-safety related pneumatic pressure differential airlock system from the air supply system, the ability to provide additional adjustment to the mechanical interlocking system, the incorporation of electrical delay relays to extend the time intervals between sequential door operations, and relocate the signal for the door status indication lights in the Control Room from the airlock door limit switch to the airlock seal pressure switch.

The enhancements made will continue to ensure the door seals are maintained at the required pressure (Ref. TRM TR3.6.1.2) and will still be capable of maintaining the 30 day required air supply (UFSAR 9.3.1.3). The modification of relocating the seal flow control valves only control the rate at which the

seals are pressurized to mitigate potential damage from rapid inflation. Furthermore, the seals will be given additional time to deflate, thus mitigating friction between the seals and door frame during opening operations and potentially causing damage to the seals. The addition of the mechanical test connections in the blind flange meets the original airlock specifications and will have 3/8”tubing seal welded to the flange and have manual isolation valves on both sides of the bulkhead, thus, having two means of isolation at all times during normal operation. The isolation valves will be maintained in the locked closed position at all times unless maintenance is being performed. The four isolation valves for each airlock will be containment isolation valves used when performing airlock testing, and these valves will be addressed appropriately in the UFSAR and the TRM. Tables 6.2-44, 6.2-49 and TR3.6. 1.3-1 have been revised to include eight new containment isolation valves (four per airlock). Under normal airlock operation the containment airlock’s inner door/bulkhead is the inboard containment isolation barrier and the airlock’s outer door/bulkhead is the outboard isolation barrier. During testing of the inner door pneumatic tubing, while the inner door is inoperable, or at any time the inboard airlock door/bulkhead is breached, the four new valves perform a containment isolation function along with the airlock outer door. The four containment isolation valves fulfill the requirements of GDC 56 for containment isolation. These valves are part of the outboard containment airlock bulkhead and will be controlled by the Technical Specifications and the TRM and will be administratively controlled in accordance with LCO 3.6.1.2 and LCO 3.6.1.3, when the valves are required to be intermittently open for airlock testing. During normal operation the isolation valves will be locked closed. Therefore, the pressure integrity of the Containment will not be degraded. The ultimate pressure capacity of the Containment as defined in UFSAR Section 3.8.1.8 will not be affected, altered or diminished by the modifications. All of the electrical enhancements are non-safety related and do not interface with safety related electrical systems and will not affect the integrity of the airlock.

No surveillance or testing requirements for the airlocks in the Technical Specifications will be changed based on these modifications. The existing airlock barrel test in Surveillance Requirement 3.6.1.2.1 will be used to ensure the pressure integrity of the airlock and the modified mechanical test connection. The probability or consequences of an accident or malfunction of equipment previously evaluated in the UFSAR will not increase with the installation or operation of the modified airlocks. The possibility of an accident or malfunction of equipment important to safety different from that previously evaluated in the UFSAR will not occur with the implementation of the airlock modifications.

The following Technical Specifications considerations are necessary for the implementation of the mechanical test connection with the primary containment isolation valves and for the use of this mechanical test connection in the future.

1. The existing 3” blind will be removed from the outboard bulkhead of the airlock. Tech Spec. LCO 3.6.1.2, Condition C, should be entered to perform this removal. The new mechanical test flange consisting of the 3/8” diameter tubing and manual valves is then installed and then an airlock barrel test is completed in accordance with Surveillance Procedure 06-ME-1M23-V-0002, “Personnel Airlock Local Leak Rate Test”, which will include the flange and the isolation valves to verify the integrity of the barrel and new mechanical test flange. Local Leak Rate Testing of the new isolation valves will be required to be completed. After successful completion of the barrel test and Local Leak Rate Testing, Tech Spec. LCO 3.6.1.2, Condition C, is exited.

2. To perform the testing of the safety related tubing and air accumulators in accordance with Surveillance Procedure 06-ME-1M23-R-0001, “Personnel Airlock Door Seal Air System Leak Test”, one door of the airlock is inoperable. Tech. Spec. LCO 3.6. 1.2, Condition A, is entered and the opposite door is locked closed. Test connection valves located on the outer bulkhead flange are Primary Containment Isolation Valves (PCIV) and fall under Tech Spec 3.6.1.3. During testing, the

inboard test connection valves are opened and left unattended requiring entry into LCO 3.6.1.3, which requires closing the outboard valves to satisfy the action statement. During testing the outboard test connection valves are also required to be open, creating a flow path through the outer bulkhead. The opening of these valves is allowed in accordance with the LCO Basis of Tech Spec 3.6.1.2, which is being revised to read “For each airlock to be considered OPERABLE, the airlock interlock mechanism must be OPERABLE, the airlock must be in compliance with the Type B airlock leakage test, both airlock doors must be OPERABLE and the test connection valves must be OPERABLE in accordance with LCO 3.6.1.3. These normally closed manual isolation valves are considered OPERABLE when closed or intermittently opened under administrative controls.” Therefore during the testing, a dedicated person must be stationed at the outboard open isolation valves at all times and must be in communication with the control room, in order to close the manual outboard isolation valves if required. This administrative control provides appropriate compensatory action to preclude entry into LCO 3.6.1.2, Condition C, for the outer bulkhead. After successful completion of the testing, the valves will be closed and the airlock door will be returned to service and Tech Spec LCO 3.6.1.2, Condition A, is exited.

Serial Number: 1999-018-NPE Document Evaluated: TA 99-013

DESCRIPTION OF CHANGE, TEST OR EXPERIMENT:

Mechanical Gags will be placed on the Plant Chilled Water system valves 1P71F298 and 1P71F019 which are isolation valves to the turbine building secondary chill water loop. The function of these valves is to isolate the turbine building portion of the chilled water system from the rest of the chilled water s stem when initiating the Turbine Building Heating Mode. The mechanical gags which will be placed will prevent 1P71F298 and 1P71F019 from being closed either by operator action or by loss of power or loss of air pressure.

REASON FOR CHANGE, TEST OR EXPERIMENT:

1P71F298 and 1P71F019 are both “Fail Closed” valves and will close in the case of loss of power or loss of air pressure. If either valve fails closed it isolates flow to the Turbine Building Secondary loop which supplies chilled water to the Circulating Water Pumps and other important heat loads such as cooling for the TVR and Feed Pump Rooms. Therefore, loss of Plant Chilled Water to the Turbine Building Secondary Loop would present a challenge to plant operation.

In January 1999, 1P71F298 failed closed on loss of air pressure due to air leaking at the seals. The plant was shut down at the time so no operational impact was experienced. CR 1999-007 9 was initiated and immediate corrective action was carried out to rework the 1P71F298 and 1P71F019.

In June 1999 a significant air leak at 1P71F019 was discovered and documented on CR 1999-0617. The valve was gagged open temporarily, then reworked and returned to service.

ER 1999-0162 was initiated as an additional corrective action to CR 1999-0079 and recommends that permanent gags be placed on valves 1P71F298 and 1P71F019 to prevent inadvertent closure the Turbine Building Heating Mode capability has been disabled and much of the associated equipment such as the Auxiliary Steam System has been de-commissioned or abandoned. Gagging valves 1P71F298 and lP71F019 open will not affect the function of the chilled water system and will prevent a challenge to plant operation.

Installing this Temporary Alteration will gag both valves open until a final solution can be implemented under ER 1999-0162.

SAFETY EVALUATION SUMMARY AND CONCLUSIONS:

Installation of the mechanical valve gags called for in the temporary alteration will prevent closure of valves 1P71F298 an 1P71F019. Gagging these valves open will further disable the Heating Mode of the Turbine Building Secondary Loop of the Plant Chilled Water System. A previous design change, DCP 88-021, decommissioned, removed or abandoned in place many of the components associated with the Heating Mode such as the Auxiliary Steam System. The valves to be Gagged open under this temporary alteration have no other function. In the normal, open position, valves 1P71F298 and 1P71F019 have no adverse effect on the ability of the Plant Chilled Water system to perform its function.

Primary containment isolation valves are discussed in Technical Specification Section 3.6.1.3 and TRM Table TR 3.6.1.3-1. Secondary containment isolation valves are discussed in section 3.6.4.2 and TRM Table 3.6.4.2-1. Building temperature limits are discussed in TRM section 6.7.3 and table 6.7.3-1. Valves 1P71F298 and 1P71F019 are not addressed in the Technical Specifications. Other than containment isolation valves in the system, the Plant Chilled Water System has no safety related function as defined in Section 3.2 of the SAR.

The valves and associated piping affected by this temporary alteration are entirely within the turbine building and are outside the boundary of the Plant Chilled Water containment isolation valves. Gagging these valves open will only further disable the Heating Mode of the Turbine Building Secondary Loop and will not affect any other function of the Plant Chilled Water System.

For the duration of this temporary alteration, the following information in the FSAR will be inaccurate: Figure 9.2-21A (P&ID M-l109F). The conclusion of this safety evaluation is that no unreviewed safety question exists and that the Technical Specifications is not impacted or changed by the proposed temporary alteration.

Serial Number: 1999-019-OPS Document Evaluated: PAP 04-1-01- C11-1, Rev. 108

BRIEF DESCRIPTION OF CHANGE, TEST OR EXPERIMENT:

The CRD Hydraulic System SOI is being revised to close the CRD pump minimum flow isolation valves and valve in the standby drive water and pump suction filters when CRD flow is maximized during emergency conditions. This will increase CRD flow to the vessel by up to 40 gpm and restore compliance with CRD flow requirements of NUREG-0619. Procedural controls will be added to prevent inadvertent isolation of the forward flowpath of the CRD pumps while the minimum flowpath is isolated.

REASON FOR CHANGE, TEST OR EXPERIMENT:

Response to GGNS CR 1997-1100 (Action #6)

SAFETY EVALUATION SUMMARY AND CONCLUSIONS:

The change is required to restore compliance with CRD flow requirements of NUREG-0619. Closing the CRD pump minimum flow isolation valves and valving in the standby drive water and pump suction filters when CRD flow is maximized during emergency conditions is an acceptable change since no credit is taken for CRD flow in any accident evaluated in the SAR.

Serial Number: 1999-020-NPE Document Evaluated: ER 98/0397-00-00

BRIEF DESCRIPTION OF CHANGE, TEST, OR EXPERIMENT:

As identified in ER 98-0397-00-00, this change evaluates only the fire barrier functions of the boundaries identified. The only other boundary function for the boundaries evaluated is “Air” (air tight control room envelope) and this boundary function is addressed in CR-GGN-1997-1348. Specifically, this change: (1) Accepts-As-Is seventeen specific electrical conduit configurations and provides “Repair” instructions for five specific electrical conduit configurations that utilize RTV-108 silicone as an internal conduit seal for fire and smoke; (2) Accepts-As-Is two specific electrical conduit configurations that require internal conduit fire or smoke seals where the first opening from the penetration is inaccessible and the next accessible opening is used to determine the need for a fire/smoke internal conduit seal based on criteria in Note 9 on Drawing M-0800D; (3) Accepts-As-Is eleven specific electrical conduit configurations where the internal conduit seal on one side of the barrier can not be physically verified because access to that side of the penetrations and the area around the related conduits is not physically possible without removal of major plant equipment or wall sections; and (4) provides “Repair” instructions for one specific electrical conduit configuration where the sensitive nature of the cable in the conduit prevented installation of an approved internal conduit seal. For the items identified above, the following are the specific conduits/penetrations affected: Item 1) (Accept-As-Is for) Conduit 1 BARNS20 on the 0C702 side of Penetration CE-11G, Conduit 1BDRNR45 on the 0C702 side of Penetration CE-13G, Conduit lBARNSl8 on the 0C703 side of Penetration CE-33G, Conduit lBDRNRO5on the 0C702 side of Penetration CE-178G, Conduit lBDRNR25 on the 0C702 side of Penetration CE-180G, Conduit
1BARWS0 on the 0C702 side of Penetration CE-181G, Conduit 1BDRT975 on the 0C703 side of Penetration CE-351G, Conduit 1BARWS14 on the 0C702 side of Penetration CE-360G, Conduit lBARN69E on the 0C702 side of Penetration CE-362G, Conduit lBARWSl2 on the 0C702 side of Penetration CE-363G, Conduit lBARNS23 on the 0C702 side of Penetration CE-364G, Conduit 1BERN68E on the 0C703 side of Penetration CE-365G, Conduit lBDRNR32 on the 0C703 side of Penetration CE-369G, Conduit 1BARNS0 on the 0C703 side of Penetration CE-382G, Conduit 1BDRNR32 on the 0C702 side of Penetration CE-400G, Conduit 1DRN67B on the 0C702 side of Penetration CE-416G, & Conduit 1BERW619 on the 0C703 side of Penetration CE-420G; (Repair for) Conduit 1BARNR10on the 0C702 side of Penetration CE-27G, Conduit 1BDRNR50on the 0C702 side of Penetration CE-126GA, Conduit 1BDRS690on the 0C703 side of Penetration CE-366G, Conduit 1BDR0698 on the 0C702 side of Penetration CE-368G, & Conduit 1BARNS0 on the 0C707 side of Penetration CE-375G. Item 2) Conduits lBDRNS42 & 1BERNS34 on the 0C702 side of Penetration CE-1. Item 3) Conduit 1BERN6AC, an unscheduled ¾” & 1/2” conduits on the 0C702 side of Penetrations CE-110GA, an unscheduled ¾” conduit on the 0C702 side of Penetrations CE-ll1A, Conduit 1BBRNR34 on the 0C702 side of Penetration CE-117GA, Conduits 1BDRWR25, 1BDRZ90l, & an unscheduled ¾”conduit on the 0C702 side of Penetration CE-120GA, Conduit 1BBRNR34 on the 0C702 side of Penetration CE-142 GA, an unscheduled 1 /2” conduit on the 0C702 side of Penetration CE-147GA, & an unscheduled ¾” conduit on the 0C702 side of Penetration CE-148GA. Item 4) Conduit lBDRM655 on the 0C706 side of Penetration CE-369G.

REASON FOR CHANGE, TEST, OR EXPERIMENT:

GGNS design for internal sealing of conduits for fire and smoke is provided in Note 9 on Drawing M-0800D. The need and material type of an internal conduit seal is based on the conduit diameter, cable fill, and the distance between the fire barrier and the first opening in the conduit. RTV 108, which is installed in the twenty-two conduit configurations identified above, is not an approved material for sealing conduits that penetrate fire barriers. Due to the physical characteristics of the cured RTV 108 material, removal of the internal conduit seal material is difficult and would most likely lead to cable damage created during the seal removal process. Therefore, the acceptability of the existing RTV seals has been evaluated. Two conduit configurations were identified that require internal conduit fire or smoke seals where the first opening from the penetration is inaccessible and the internal conduit seal required by the present GGNS design could not be verified. These configurations were evaluated for

acceptability. Eleven conduit configurations were identified that pass through a fire rated penetration and the internal conduit seal on one side of the penetration can not be physically verified because access to that side of the penetrations and the area around the related conduits is not physically possible without removal of major plant equipment or wall sections. One specific electrical conduit configuration was identified where the sensitive nature of the cable in the conduit prevented installation of an approved internal conduit seal. Acceptability of this installation has been evaluated.

SAFETY EVALUATION SUMMARY AND CONCLUSIONS:

The changes made by this ER (1) Accepts-As-Is seventeen specific electrical conduit configurations and provides "Repair" instructions for five specific electrical conduit configurations that utilize RTV-108 silicone as an internal conduit seal for fire and smoke; 2) Accepts-As-Is two specific electrical conduit configurations that require internal conduit fire or smoke seals where the first opening from the penetration is inaccessible and the next accessible opening is used to determine the need for a fire/smoke internal conduit seal based on criteria in Note **9** on Drawing M-0800D; (3) Accepts-As-Is eleven specific electrical conduit configurations where the internal conduit seal on one side of the barrier can not be physically verified because access to that side of the penetrations and the area around the related conduits is not physically possible without removal of major plant equipment or wall sections; and (4) provides "Repair" instructions for one specific electrical conduit configuration where the sensitive nature of the cable in the conduit prevented installation of an approved internal conduit seal. With the exception of the eleven conduit configurations where the internal conduit seal on one side of the barrier can not be physically verified because access to that side of the penetrations and the area around the related conduits is not physically possible without removal of major plant equipment or wall sections, the remainder of the internal conduit seal configurations have been evaluated and determined to maintain the fire resistance rating requirements of the barriers as presently analyzed in the SAR. For these eleven internal conduit seal configurations, the evaluation determined that the fire rating of the barrier may be adversely affected; however, the as-built configuration was determined to provide a sufficient level of protection to ensure that redundant safe shutdown components would not be damaged as a result of a fire on either side of the identified penetrations. Therefore, this change will not increase the probability or consequences of accidents or malfunction of equipment important to safety previously evaluated in the SAR. Also, this change 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. "Fire Rated Assemblies" are not addressed by Technical Specifications (TS). "Fire Rated Assemblies" are addressed in the Technical Requirements Manual (TRM) Section 6.2.8. Changes made by this ER involve internal sealing of electrical conduits penetrating "Fire Rated Assemblies" addressed by the TRM; however, these changes have been evaluated and determined to provide an adequate conduit seal and to maintain the hourly rating of the fire barriers involved. Therefore, there will be no change to TS or TRM or the Bases for any TS or TRM.

Serial Number: 1999-021-NPE Document Evaluated: ER 99/0066-00-00

BRIEF DESCRIPTION OF CHANGE, TEST, OR EXPERIMENT:

This ER will, as an interim repair, abandon in place the existing **3/4"** dia. carbon steel piping (3/4"-HBC-l88) and associated components between 20"-HBC-171 and SP41-FI-R009A; and between 20"-HBC-171 and SP41-FI-R009B up to and including root valves 1P41FX223, FX224, FX225, and FX226 (approximately 8" of 3/4**"-**HBC-188 will remain attached to 20"-HBC-171). These sensing lines are designed to permit periodic surveillance testing of the Standby Service Water (SSW) system pumps QlP4LC00lA and QlP4LC00lB as required by GDC 46. The differential pressure sensed across flow elements QSP4lN08lA and QSP4lN08lB is measured by local indicators SP4lR009A and SP4lR009B, respectively. GGCR19990218 documents failure (i.e. leakage) of the submerged high and low pressure sensing lines for flow element SP41N08lB. NDE data generally indicated below minimum wall thickness in the heat affected zone of each welded fitting in both of these pressure sensing lines. An interim repair to these sensing lines is necessary to restore the structural integrity of the ASME Section III Class 3 pressure boundary. This interim repair to cut, cap, and abandon in place the existing carbon steel piping will require administrative controls (not evaluated herein) to connect the resultant differential pressure taps across flow elements QSP4lN08lA and QSP4lN08lB to the local flow indicators SP4lR009A and SP4lR009B, respectively, for the purposes of surveillance testing. SCN 99-0003A to GGNS MS-02 will revise the standard to reflect this interim repair of line service number HBC-l88.

REASON FOR CHANGE, TEST, OR EXPERIMENT:

These sensing lines are designed to permit periodic surveillance testing of the Standby Service Water (SSW) system pumps QlP41C00lA and QlP4LC00lB. GGCR19990218 documents failure (i.e. leakage) of the submerged high and low pressure sensing lines for flow element SP41N08lB. NDE data generally indicated below minimum wall thickness in the heat affected zone of each welded fitting in both pressure sensing lines. An interim repair to these sensing lines is necessary to restore the structural integrity of the ASME Section III Class 3 pressure boundary. This ER will abandon in place the existing ¾" dia. carbon steel pipings (3/4"-HBC-l88) and associated components between 20"-HBC-171 and SP41-FI-R009A; and between 20"-HBC-171 and SP41-FI-R009B up to and including root valves 1P41FX223, FX224, FX225, and FX226 (approximately 8" of 3/4"-HBC-188 will remain attached to 20"-HBC-171).

SAFETY EVALUATION SUMMARY AND CONCLUSIONS:

The interim abandonment of the existing ¾"diameter carbon steel piping and root valves will not adversely impact plant safety. This interim repair does not degrade below the current design basis the performance of a safety system assumed to function in the accident analyses and does not decrease the reliability of safety systems assumed to function in the accident analyses. The change does not cause a greater reliance to be placed on any specific system, structure or component to perform a safety function. Although the operation and function of the interfacing flow indicators SP41R009A and SP41R009B will be disabled, the structural integrity of the ASME Section III Class 3 pressure boundary will be restored. SSW system operation with or without the ¾" ANSIB31.1 carbon steel pipe cap installed by this interim repair ER will have no adverse effect on the functionality of the SSW system or UHS since any resultant leakage does not represent an UHS inventory loss, UHS heat load, or significant system flow diversion path. The primary function of the abandoned pressure sensing lines is to support surveillance testing of the SSW pumps QlP4LC001A and QlP41C001B. A secondary function of the abandoned pressure sensing lines is to support positioning of manual globe valves QlP41F002A and QlP41F002B (SSW pump minimum flow protection throttle valves). The inability to perform surveillance testing of the SSW pumps will require administrative controls (not evaluated herein) to connect the resultant differential pressure taps across flow elements QSP4lN08lA and QSP4lN081B to the local flow indicators. In addition to support of surveillance testing, the function of the basin recirculation line 20"-HBC-171 is to provide a flow path for SSW pump minimum flow protection. Interim cutting, capping and abandonment of the pressure sensing lines during any and all postulated events would not prevent the basin

recirculation line 20”-HBC-171 from performing it’s minimum flow protection function. Furthermore, UHS basin inventory losses/leakage would not result as a result of the interim repair based upon the physical location of the pressure sensing lines (i.e. leakage would be contained by the UHS basin). The interim repair of the ¾”-HBC-188 piping at the interface with the 20”-HBC-171 will continue to satisfy the requirements of the ASME Code, Section III, Class 3, Seismic Category 1 and meet the support spans requirements of User Manual M-l 8. The interim repair and the resulting update of GGNS-MS-02 will meet all of the original design requirements for the piping system.

Serial Number: 1999-022-NPE Document Evaluated: ER 99/0145-00-00

BRIEF DESCRIPTION OF CHANGE, TEST, OR EXPERIMENT:

ER **99/0145-00-00** encloses the wire cage surrounding inverters 1Y97 and 1Y98 with an insulated metal building. HVAC, lights and an emergency light are provided for the new enclosure. This area is located on elevation 166 of the turbine building. The resulting room is assigned room number 1T411. UFSAR Figure 1.2, the equipment location drawing of the Turbine Building El. 166, is revised to reflect the insulated wire cage. UFSAR Figure 12.3, radiation zones of the Turbine Building El. 166, is revised to reflect the presence of the insulated wire cage.

REASON FOR CHANGE, TEST, OR EXPERIMENT:

Inverters 1Y97 and 1Y98 have been overheating due to the high ambient temperatures that are being experienced on the turbine deck. The enclosure provided will be supplied with HVAC which will provide a means to maintain the temperature of the inverters at a more acceptable value.

SAFETY EVALUATION SUMMARY AND CONCLUSIONS:

The enclosure of the existing wire cage with an insulated metal building, the addition of lights and HVAC to the new structure conforms to all design requirements. The equipment is being added in accordance with accepted design standards. The modification is non safety-related, does not affect any safety related equipment and does not introduce any new failure modes for equipment important to safety. It is concluded that enclosing the existing wire cage with an insulated metal building and providing HVAC and lights does not adversely affect plant safety and is acceptable.

Serial Number: 1999-023-NPE Document Evaluated: ER 97/0659-00-02

BRIEF DESCRIPTION OF CHANGE, TEST, OR EXPERIMENT:

Breaker 72-11D67 is being changed from a 15 Amp breaker to a 30 Amp breaker, and Spare breaker 72-11D87 is being changed from a 30 Amp breaker to a 15 Amp breaker. BK7 type cable is being used instead of BK5 as specified in drawing E-1022. A new note (Note 12) will be added to drawing E-1022 which states, "Cable from 30 Amp breaker 72-11D67 is BK7 and was evaluated as acceptable in ER 97/0659-00-02." Additionally, ER 97/0659-00-R02 replaces MIN 20 Amp fuses with BAN 20 Amp fuses; however, this change does not affect the SAR, and therefore, it is not evaluated per the seven questions presented in this safety evaluation.

REASON FOR CHANGE, TEST, OR EXPERIMENT:

20 Amp fuses 1N41F003 and 1N41F008 are not in coordination with the present 15 Amp upstream breaker 72-11D67. Replacing the breaker with a 30 Amp rated breaker provides proper coordination such that the fuses will blow on a downstream fault prior to the upstream breaker tripping. Also, since the cabling was originally sized for the 15 Amp breaker, BK7 cable was used vice BK5 cabling.

SAFETY EVALUATION SUMMARY AND CONCLUSIONS:

The change of fuse type from MIN to BAN type (Bussman manufacturer) only increases the voltage rating of the fuses which enhances the ability of the affected component to perform their intended function. This fuse replacement in itself does not present a change to the facility which alters, or has the potential to alter, the information, operation, function or ability to perform the function of a system, structure or component in the SAR. This portion of ER 97/0659-00-R02 does not affect the SAR, and therefore, it is not evaluated per the seven questions presented in this safety evaluation.

The breaker rating change does affect SAR Figure 8.3-10B and is evaluated per 10CFR50.59 in this safety evaluation. While the breaker ratings in UFSAR Figure 8.3-l0B will be changed for this design modification, the changes being made ensure that proper coordination of the downstream fusing with the upstream breaker will be maintained. This change enhances the overall reliability of the N41 system (Generator and Main Transformer Protection). Because of the replacement of the 15 Amp breaker with a 30 Amp breaker, BK7 type cabling is used instead of BK5 cabling as specified in Note 8 of drawing E-1022; however, the 30 Amp breaker still provides full protection for the BK7 cabling as shown in Attachment 8 to ER 97/0659-00-R02. While this system is non-safety related, it is always desirable to ensure proper coordination in any electrical distribution system and is an IEEE recommended practice (IEEE Std. 242). The change only serves to enhance system reliability and does not affect any accidents in the SAR, nor does it affect the radiological consequences of any accident previously evaluated in the SAR. The breakers affected are not Class 1e and do not affect any safety related AC distribution system. This change has no impact on the basis of any technical specification at GGNS. Therefore, this change does not involve an unreviewed safety question.

Serial Number: 1999-024-NPE

Document Evaluated: DRN 6745; 6747 through 6758 (CR 1998-0152)

BRIEF DESCRIPTION OF CHANGE, TEST, OR EXPERIMENT:

This change adds a note to each of the affected drawings denoting that the non-accident and Unit 2 radiation levels are operational estimates and that the information is not updated based on plant conditions. This information is considered historical.

REASON FOR CHANGE, TEST, OR EXPERIMENT:

This change is in accordance with the disposition of CR-1998-0152. Due to the dynamic nature of plant radiological conditions, it is impractical to capture these conditions on an architectural drawing

50.59 EVALUATION SUMMARY AND CONCLUSIONS:

This change is in accordance with the disposition of CR-1998-0152. This disposition is to add a note to the various radiation zone drawings to clearly state that they are pre-operational estimates of general area radiation levels and are not updated based on plant conditions. One of the following notes will be added to each drawing. Note A: All non-accident dose rates for Unit 1 and all dose rates for Unit 2 identified on this drawing are pre -operational estimates and are not updated based on actual plant radiological conditions. This information should be considered historical. Note B: All non-accident dose rates identified on this drawing are preoperational estimates and are not updated based on actual plant radiological conditions. This information should be considered historical. As discussed in the GGNS Safety Evaluation Report (SER), the rad zone drawings were included in the SAR to provide the NRC with information on radiation protection measures applicable to the design and operation of GGNS. The objective of the NRC review was to ensure that internal and external exposures to station personnel, contractors and the general public are within applicable limits of 100FR2O, and as low as reasonably achievable. These drawings were never intended nor can they be used to report current operating radiological conditions in the plant. These conditions are monitored and reported by routine radiological surveys. The addition of the proposed notes do not affect the NRC's review and approval of the GGNS design and the radiation zone drawings are not required to ensure compliance with the requirements of
10CFR2O or ALARA principles. No accident analyses are affected by the proposed change. Based on this evaluation, an unreviewed safety question does not exist.

Serial Number: 1999-025-PLS Document Evaluated: MAI 257406

Brief description of change, test or experiment:

A flatbed trailer in the auxiliary building railroad bay holds a cask (an 8'x8'6"Footprint and a weight of 7,000 lbs.) and two stands/motor mounts (each with an 8'x8'6" footprint and a weight of 26,000 lbs.) used for recirculation pump impeller and shaft changeout. The cask and stands will be transferred to the Unit 1 condenser bay on the east side (Area 2a) for semi-permanent storage. A second cask is within the auxiliary building secondary containment. It will be transferred at a separate date (post RF10 during cycle 11) after it is removed from within secondary containment during the refueling outage. This safety evaluation is for transfer of the casks and the stand/motor mounts from the auxiliary Building railroad bay to the Unit 1 condenser bay.

Transfering the casks and stand/motor mounts is contingent upon approval of ER 99-0254. This ER will contain an evaluation of the 87' condenser bay floor loading for all four items and an evaluation of the transfer route with regard to possible damage to under or in-ground piping, bus ducts, etc. Requirements specified in the ER will be followed with regard to protecting the travel path.

REASON FOR CHANGE, TEST OR EXPERIMENT:

Storage of the items has been problematic since they were first used. They are large and heavy and have high levels of loose and fixed contamination. The stands/motor mounts create a high radiation area to within several feet. The purpose of this evolution is to transfer the items from the railroad bay, around Unit 2 to behind the Unit 1 condenser bay on the east side (Area 2a), and to place them into the Unit 1 condenser bay under the "patio" hatches, which is a suitable long term storage location. This transfer will be made during power operations.

Storing the items in the condenser bay will free up space in the Auxiliary Building and the railroad bay. The items have been in the way during normal, re-occuring activities.

SAFETY EVALUATION SUMMARY AND CONCLUSIONS:

The conclusion reached in the safety evaluation supports the transfer of the items to the Unit 1 condenser bay during power operations. The transfer requires no Technical Specification change, poses no new unreviewed safety question, will not require a change to the environmental protection plan, and poses no new unreviewed environmental question.

Serial Number: 1999-026-NPE Document Evaluated: MAI 257406

(Revision to the SE)

BRIEF DESCRIPTION OF CHANGE, TEST OR EXPERIMENT:

A flatbed trailer in the auxiliary building railroad bay holds a cask (an 8'x8'6"Footprint and a weight of 7,000 lbs.) and two stands/motor mounts (each with an 8'x8'6" footprint and a weight of 26,000 lbs.) used for recirculation pump impeller and shaft changeout. The cask and stands will be transferred to the Unit 1 condenser bay on the east side (Area 2a) for semi-permanent storage. A second cask is within the Auxiliary Building secondary containment. It will be transferred at a separate date (post RF10 during cycle 11) after it is removed from within secondary containment during the refueling outage. This safety evaluation is for transfer of the casks and the stand/motor mounts from the auxiliary Building railroad bay to the unit 1 condenser bay.

Transfering the casks and stand/motor mounts is contingent upon approval of ER 99-0254. This ER will contain an evaluation of the 87' condenser bay floor loading for all four items and an evaluation of the transfer route with regard to possible damage to under or in-ground piping, bus ducts, etc. Requirements specified in the ER will be followed with regard to protecting the travel path.

REASON FOR CHANGE, TEST OR EXPERIMENT:

Storage of the items has been problematic since they were first used. They are large and heavy and have high levels of loose and fixed contamination. The stands/motor mounts create a high radiation area to within several feet. The purpose of this evolution is to transfer the items from the railroad bay, around Unit 2 to behind the Unit 1 condenser bay on the east side (Area 2a), and to place them into the Unit 1 condenser bay under the "patio" hatches, which is a suitable Long term storage location. This transfer will be made during power operations.

Storing the items in the condenser bay will free up space in the auxiliary building and the railroad bay. The items have been in the way during normal, re-occuring Activities.

SAFETY EVALUATION SUMMARY AND CONCLUSIONS:

The conclusion reached in the safety evaluation supports the transfer of the items to the Unit 1 condenser bay during power operations. The transfer requires no Technical Specification change, poses no new unreviewed safety question, will not require a change to the environmental protection plan, and poses no new unreviewed environmental question.

Serial Number: 1999-027-NPE Document Evaluated: ER 98/0085-01-00

BRIEF DESCRIPTION OF CHANGE, TEST OR EXPERIMENT:

Thermal analysis increased pipe loads on the RHR support system and resulted in exceeding the allowable stresses for a support member, a base plate, five cantilevered branch connections and two branch fitting connections. The overstressed support member and the base plate will be qualified for Code allowables by physically modifying the supports. The five overstressed cantilevered branch connections will be qualified for Code allowables by installing tieback supports for two-cantilevered branch lines, removing an existing tieback support from one cantilevered branch line and the other two overstressed cantilevered branch lines were resolved by a revision to the calculation. Two overstressed branch-fitting connections will be qualified by adjusting spring can settings.

REASON FOR CHANGE, TEST, OR EXPERIMENT:

This modification will restore stresses within code allowables for piping and supports in the RHR and FPCCU systems. (Reference CR 97/1281, CR 98/1184 and CR 98/1320).

SAFETY EVALUATION SUMMARY AND CONCLUSIONS:

The support system modifications will bring the piping stresses within Code allowables. The support systems will be designed safety related in accordance with ASME B&PV Code Sec. III, Class 2, seismic category 1. The modifications will not impact the operability, function or integrity of the subject systems. No change in the operation or function of the RHR or the FPCCU systems will be created by this modification and no change to GGNS Technical Specifications or UFSAR will be required. Deleting Support QlEl2G176R0l from the test connection will not create a missile hazard. The implementation of ER 98/0085-01 will not increase the probability or consequences of a previously evaluated accident, nor will it increase the probability of occurrence or consequences of a malfunction of equipment important to safety previously evaluated in the SAR. The implementation of ER 98/0085-01 will not create the possibility for an accident or the possibility for a malfunction of equipment important to safety of a different type than any previously evaluated in the SAR. The RHR and FPCCU systems piping integrity will be assured with the new piping stress analysis and support systems modifications. There is no change to the pipe configuration, design/quality requirements, system operation, function, integrity or other system parameters. The constant SIF value of 1.3 provided by EPRI Report # TR-106415 for Circumferential Fillet-Welded or Socket - Welded Joints is based on reliable experimental data which show that this value is valid for all Code acceptable configurations. Therefore, the margin of safety as defined in the Technical Specification Basis is not reduced.

Serial Number: 1999-028-NPE Document Evaluated: ER 98/0623-00-00

BRIEF DESCRIPTION OF CHANGE, TEST OR EXPERIMENT:

In accordance with GGNS commitments to NUREG-0612, "Control of Heavy Loads at Nuclear Power Plants", the Head Strongback Carousel, Dryer/Separator Strongback, and the Drywell Head Lifting Frame (Strongback) are classified as special lifting devices which are used to handle heavy loads in the containment. As such,. GGNS commitments to the NUREG include the provisions of ANSI N14.6-1978, "American National Standard for Special Lifting Devices for Shipping Containers Weighing 10,000 Pounds or More for Nuclear Materials" as referenced by the NUREG and as described in UFSAR Appendix 9D.

In part, the ANSI standard requires lifting devices to be load tested annually, or when conditions permit they can be dimensionally tested, visually inspected, and nondestructively examined (NDE) at critical load carrying locations. However, as described in AECM-82/415, GGNS took exception to selected portions of the ANSI standard and performs the load test only if evidence of deformation is detected or if repairs are performed. Additionally, in lieu of performing the inspections (visual, dimensional, and nondestructive) annually as prescribed in the ANSI standard, they are performed prior to each 5th refuel outage.

To perform the NDE (liquid penetrant or magnetic particle) specified by the ANSI standard requires removal and replacement of the protective coating. For the dryer/separator strongback, ER 98-0623-00-00 replaces the liquid penetrant and magnetic particle examinations with a detailed visual inspection that may be performed without coating removal.

REASON FOR CHANGE, TEST, OR EXPERIMENT:

The dryer/separator strongback (as well as the other lifting devices) is stored in the Containment at elevation 208'. To perform the NDE described in the ANSI standard, liquid penetrant (PT) or magnetic particle (MT) requires complete removal of the coatings in the area to be examined. Coating removal is an intrusive process that requires motorized brushing and grinding or sand blasting. Because the removal process may cause airborne contamination and loose material that could enter the suppression pool, extensive enclosures are required if the work is to be performed in Modes 1, 2, or 3. However, the enclosures are also fabricated from materials that, under accident conditions could also be transported into the suppression pool becoming an ECCS suction strainer blockage hazard. The changes described in ER 98-0623-00-00 provide alternative examination methods that may be performed without coating removal and are adequate to ensure structural integrity of the subject lift devices.

SAFETY EVALUATION SUMMARY AND CONCLUSIONS:

Compliance with NUREG-0612 and ANSI N14.6 as currently described in GGNS commitments provides a defense-in-depth approach for controlling the handling of heavy loads so that load handling accidents have a very low probability of occurrence.

The approaches taken in the NUREG and ANSI standard provide a high level of confidence that lifting devices manufactured of engineered materials can be depended upon to perform their design function reliably. Similarly, ASME Section XI also provides for the same assurances for safety related pressure boundaries and their structural supports. Both industry standards address design, materials, fabrication, repairs, and utilize inservice inspection as the mechanism to detect service-induced degradation. Typical inservice inspection techniques consist of visual inspections, NDE surface methods and NDE volumetric methods. Within the visual and NDE methods, numerous techniques are available and are selected based on their ability to detect degradation in its earliest stages. Improper selection or misapplication of inspection/examination techniques may permit degradation to occur or progress to a point of failure without detection.

The purpose of the inservice inspection is to detect evidence of service induced degradation or the precursor to failure. In selecting the appropriate inspection technique, it is important to know the type of failure that might be expected. Different failure mechanisms may be preceded by indications specific to that mechanism which may be detectable by one inspection technique and not another. There are two generally accepted types of material failure: one is the easily recognized fracture or separation into two or more parts; the second is the less easily recognized permanent deformation or change of shape and/or position. Although fracture is unmistakable, there are multiple mechanisms that may initiate the fracture and they all have their own characteristics that may affect their detectability.

Material fracture can typically be attributed to either environmentally assisted cracking, service induced cyclic loading (fatigue), or over stress. The lift devices are stored in the upper containment and are only subjected to loading when they are used during refueling outages. Their storage environment precludes concerns with environmentally assisted cracking and their infrequent use precludes concerns with fatigue failures. However, failure due to overstress caused by excessive or improper loading is credible.

Failure (fracture) caused by excessive stress will only occur when some critical tension-stress value is exceeded in the material. However, prior to exceeding this tensile value, the material must exceed its elastic limit and assume some permanent deformation (plastic flow) that is not recovered when the load is removed. The amount of plastic flow permitted before exceeding the ultimate material strength is dependent on the ductility of the material. Material ductility is expressed by several material properties such as: reduction in area, hardness, or elongation.

The principal members of the lifting devices are fabricated from ASTM A36 steel welded with an E60 series weld filler material. The ASTM A 36 steel has a minimum yield of 36 ksi and a minimum elongation of 20% in 2 inches; the E60 series filler material has a minimum yield of 50 ksi and a minimum elongation of 22% in 2 inches. As typical with structural design, the weld is the strongest point of the connection and in this case also exhibits a higher elongation for improved ductility. The material used in the lifting devices are very ductile and will endure significant plastic flow in both the base material and weld material before exceeding the material's ultimate tensile strength. Based on the distances between connections, this translates into sufficient deformation to allow for visual detection long before the initiation of material fractures.

Unlike ASME Section XI, the NUREG and the ANSI standard gave little consideration to the service induced failure mechanisms and simply imposed a non-descriptive visual inspection and the NDE (PT or MT) requirements contained in ASME Section III, Subsection NF for the fabrication of pipe supports. While this criteria is appropriate for examining new welds used in the construction of structural elements, it is not very amenable to serviced structures and may not adequately detect service induced degradation before a fracture initiates.

The PT and MT methods are considered surface examination techniques and provide for the detection of surface connected discontinuities. In the case of the lift devices, GGNS has already taken exception to the construction code acceptance criteria and only requires the locations to be examined for cracks. However, because the lift devices are of a structural nature, fabricated from ductile materials, and are limited to an over stress failure mechanism, the inspection program will be enhanced by focusing on detecting deformation (plastic flow) which is a precursor to fracture. Service induced deformation in the lifting device material will actually occur long before actual surface fractures are initiated. Therefore, early detection of deformation in the lifting devices is the most practical approach to preventing failure in its earliest stages.

Deformation can be identified by general observation, dimensional measurements or by inspecting for brittle fracture of coatings. Detecting deformation by general observation and dimensional measurements is effective and is appropriately adequate. However, for added assurance and increased sensitivity to small amounts of deformation, inspection of the coated surfaces in the major load bearing areas for coating failure is necessary. The dryer/separator strongback is coated with Amerlock 400 which has an elongation factor of 5% compared to the elongation factor of 20% for the structural steel used in the lift device members. Relatively, the coating is a factor of 4 more brittle than the base material meaning that it would develop cracks and chipping while the base material was well within its plastic flow range. The use of Brittle Coating Tests is a recognized method for detecting responses to strain in structures beneath the coating. Section 52 of the Nondestructive Testing Handbook describes the use of brittle coatings as a method for detecting the distribution, direction, location, sequence, and magnitude of tensile strains while the material is within its elastic limits. Using the condition of the coating as described in ER 98-0623-00-00 for evidence of material behavior in the plastic range provides adequate assurance that deformation is detected in its earliest stages. Detecting the early stages of deformation as a precursor to failure provides a margin of safety greater than that obtained by waiting until deformation results in material fracture so that it can be detected by PT or MT.

ASME Section XI has taken a similar approach for inservice inspections of structural elements used to support the safety related pressure boundary. Although Section XI does not recognize the inspections of coatings as an indication of deformation, Section XI does require visual inspections for deformation. Conditions that Section XI finds unacceptable for structural elements include deformation or structural degradation of fasteners, springs, clamps, or other support members. Surface examination with the PT or MT method of critical areas of the support is not required because deformation would have occurred long before fracture occurs.

Because of the described inspections there is no increase in the probability of an accident or malfunction of equipment different than an previously evaluated in the UFSAR. Additionally, because of the nature of the lift devices and their use, they are not considered in any existing accident analysis contained in Chapter 15 of the UFSAR. Therefore, there is no increase in probability of occurrence, consequence of an accident, or probability of malfunction of equipment previously evaluated in the UFSAR.

The changes provided by the ER have been demonstrated by the discussion in this evaluation to provide an adequate level of safety and quality, and are also consistent with other industry standards (ASME Section XI) which provide for similar inspections of structural elements that have direct safety impact to GGNS.

Serial Number: 1999-029-NSA Document Evaluated: LDC 99-061

BRIEF DESCRIPTION OF CHANGE, TEST OR EXPERIMENT:

This evaluation is for LDC-99061 and involves the addition of a new NOTE (f) to ODCM/TRM Table 6.3.10-1. The change is necessary to provide clarification for the REQUIRED ACTION COMPLETION TIME for an inoperable radioactive gaseous effluent monitoring instrumentation high or low volume flow device. The new NOTE will clarify that the 30-day completion time (ODCM/TRM LCO 6.3.10 REQUIRED ACTION D.2) for an inoperable high or low volume flow device is based on a cumulative release time which for these devices is only during high or low volume purge of containment.

REASON FOR CHANGE, TEST OR EXPERIMENT:

A Condition Report (GG-CR-1999-027) was written on 1/8/99 documenting the difference in interpretation of the REQUIRED ACTIONS A.2 specified in ODCM/TRM Section 6.3.10 as it related to inoperable high or low volume flow devices for radioactive gaseous effluent monitoring (ODCM/TRM Table 6.3.10-1, 2.d). For an inoperable high or low volume flow device, as specified in Section 2.d of ODCM/TRM Table 6.3.10-1, 30 days is allowed to restore the flow device back to an OPERABLE status after which point either high or low volume purge operations (release) would be terminated. Purging would only be allowed after the inoperable flow device was repaired and returned to an OPERABLE status. Notes (d) and (e) of ODCM/TRM Table 6.3.10-1 only require the flow devices whenever containment is in low or high volume purge. Therefore, the LCO and REQUIRED ACTION for inoperable flow devices are only applicable during containment purge and could be exited as allowed by Technical Specification Section 1.3, Completion Times. Without the additional note to Table 6.3.10-1, high or low volume flow devices instrumentation could be inoperable for periods far exceeding the current 30 days allowance since there is an allowance to exit the LCO if the applicability conditions are no longer met. Preservation of the 30 days is important for two reasons. The first reason is that, as part of the original licensing basis (see original TS 3.3.7.12 which existed prior to Amendment 87), we may only release for 30 days with inoperable instrumentation, after which time the release would be required to be terminated. Secondly, the ODCM currently requires an explanation as to why the inoperability of liquid or gaseous effluent monitoring instrumentation was not corrected within the 30-day time specified in LCOs 6.3.9 or 6.3.10 to be reported in the annual Radioactive Effluent Release Report. This change is needed to take full credit of the 30-day effluent release allowance with inoperable flow devices which was allowed in the previous OLD Technical Specification in Table 3.3.7.12-1, Action 123 and to ensure the appropriate actions are taken with inoperable flow devices.

SAFETY EVALUATION SUMMARY AND CONCLUSIONS:

Based on the evaluation provided below under 50.59 and the evaluation required by Technical Specification 5.5.1 this change is acceptable. This change maintains the levels, of radioactive effluent control required by 10 CFR 20.1302, 40 CFR 190, 10CFR5O.36a, and 10CFR50, Appendix I, and does not adversely impact the accuracy or reliability of effluent, dose, or setpoint calculations.

Serial Number: 1999-030-NSA Document Evaluated: LDC 99-062

BRIEF DESCRIPTION OF CHANGE, TEST OR EXPERIMENT:

The changes proposed will make the Safety Review Committee (SRC) charter and composition consistent with that of River Bend, allowing the use of a common (SRC) between the two sites. The changes include elimination of specific titles of members and remove the commitment for mandatory utilization of consultants as voting members. The changes also include the Plant Safety Review Committee (PSRC) and the elimination of specific titles of members on this committee. Additionally the change will add detail to the UFSAR which was previously in the Technical Requirements Manual, but subsequently removed after approval of the Quality Assurance Program Manual. The additional details provide methods for implementing the requirements from American National Standard ANSI N18.7-1976.

This change further accommodates the unification of the operation of Entergy's Nuclear Units into a single set of common practices.

REASON FOR CHANGE, TEST OR EXPERIMENT:

Entergy Operations has been moving toward a unified approach for operations of its nuclear units in order to move towards competition in the future.

SAFETY EVALUATION SUMMARY AND CONCLUSIONS:

The commonality of these practices and management philosophies will improve the flow of information and will facilitate uniform conduct of operations between the sites. The overall objective of this effort is to improve operations and nuclear safety at each of the sites. This change will facilitate the making of the safety oversight activities consistent between River Bend and Grand Gulf.

Serial Number: 1999-031-NPE Document Evaluated: LDC 99-063

BRIEF DESCRIPTION OF CHANGE, TEST, OR EXPERIMENT:

Section 9.5.3.1.1(a) of the FSAR is being changed to support the use of Remote Ocean Systems' (ROS) High Pressure Sodium (HPS) nuclear grade lights in the reactor vessel during refueling activities as well as the upper containment and spent fuel pools.

REASON FOR CHANGE, TEST, OR EXPERIMENT:

Section 9.5.3.1.1(a) currently precludes the use of switches and light fixtures containing mercury in the containment and fuel handling areas. The ROS HPS nuclear grade light fixtures, which contain a small quantity of mercury, offer significant advantages over incandescent lights including longer bulb life (24,000 hours vs 4,000 hours) and increased light output (140,000 lumens vs 17,500 lumens). The longer bulb life reduces the doses associated with frequent bulb replacements and the increased light output aids in underwater visibility, which reduces the
potential for fuel movement errors.

SAFETY EVALUATION SUMMARY AND CONCLUSIONS:

As described in Reference 1, the nuclear grade HPS fixtures are constructed such that for normal use, the probability of bulb failure and subsequent mercury release is extremely low. Additionally, a postulated break and subsequent release of mercury will not have an immediate effect on any important design features or margins. Therefore, it is concluded that the normal use of the HPS lights will not increase the probability of accidents currently analyzed in the SAR and no new accidents will be introduced. Further, the consequences of an accident previously evaluated in the SAR will not be increased, nor will the consequences of a malfunction of equipment important to safety previously evaluated in the SAR be increased. Malfunctions of equipment necessary for safety are no more probable nor are any additional malfunctions introduced. No reduction of any Technical Specification margin of safety as described in the bases will occur.

Serial Number: 1999-032-NPE Document Evaluated: ER 98/0414-00-00

BRIEF DESCRIPTION OF CHANGE, TEST, OR EXPERIMENT:

This Safety Evaluation is being completed to review and evaluate previously completed changes and provide an update to the UFSAR. Certain actions were previously completed or implemented at GGNS in accordance with the applicable approved procedures or policies, however, all of the affected UFSAR sections were not immediately revised to reflect these plant changes. As identified by GGCR1997-0279-00 (QDR 96/0154), certain sections of the UFSAR are discrepant or outdated due to the previously completed actions. Current policies and procedures are adequate to ensure all plant design changes are reviewed and evaluated in advance, and applicable documents (i.e, UFSAR) are updated concurrent with the design change implementations. There are no physical plant changes associated with this Safety Evaluation. This evaluation is being performed to support updating the UFSAR to reflect plant alterations that were previously completed. Specific changes or alterations include cancellation of the Unit 2 reactor, abandonment of Radwaste System evaporators, and discontinuing the practice of chemically regenerating Condensate System resin beads. In addition, certain UFSAR Tables and Figures need revision to reflect impacts of these changes and certain plant modifications such as installation of the Advanced Resin Cleaning System or the Reactor Water Clean-Up System filter demineralizer septa upgrade. Each of these previously completed actions had a measurable impact on the accuracy of UFSAR information describing operation and expected conditions of equipment associated with the Liquid/Solid Radwaste (G17/G18) Systems. Applicable design documents have been reviewed and revised as necessary in support of ER 98/0414-00-00 and this Safety Evaluation, as indicated by the documents referenced above. To ensure accuracy of the UFSAR, the affected UFSAR text, figures, and tables need to be updated to reflect conclusions reached or conditions shown by the revised calculations and drawings.

REASON FOR CHANGE, TEST, OR EXPERIMENT:

The UFSAR update is necessary to ensure the UFSAR accurately reflects current, as-built conditions of plant equipment design and operational practices. There are no physical plant changes authorized by or associated with this Safety Evaluation, nor are any tests or experiments associated with this Safety Evaluation. This Safety Evaluation is being conducted to review the impact previously completed actions have had on the UFSAR information, specifically as this information relates to Radwaste Systems, and to incorporate these plant/system changes into the appropriate UFSAR sections.

SAFETY EVALUATION SUMMARY AND CONCLUSIONS:

The conclusion reached by this Safety Evaluation is that although multiple previously completed plant changes have affected the accuracy of information presented in the UFSAR, this condition does not represent an Unreviewed Safety Question or an Unreviewed Environmental Question. Primarily, the current UFSAR information pertaining to the Liquid/Solid Radwaste (G17/G18) System is not reflective of actual as-built design and operation of the affected systems or equipment. As documented in the above referenced documents, the UFSAR discrepancies identified in GGCR1997-0279-00 have been reviewed and evaluated. The information noted to be discrepant or inaccurate has been determined to be conservative in certain instances, thus actual plant conditions are adequately bounded by existing UFSAR information. However, other UFSAR information was determined to be inaccurate and non-conservative. This review has determined that while certain UFSAR information was inaccurate and non-conservative, there were no instances where the extent of the inaccuracy resulted in operation of the plant outside of the limits or conditions imposed by the GGNS operating license. Since the UFSAR information discrepancies were primarily associated with operation and performance of the Liquid and Solid Radwaste (G17/G18) Systems, the identified discrepant condition did not affect safety related systems or balance of plant systems. Therefore, while certain UFSAR information was identified to be outdated, inaccurate, or nonconservative, the identified condition does not represent a decrease in the plants ability to safely operate or to conduct a safe and orderly plant shutdown. The identified discrepancies did not adversely impact any of the margins to safety as defined in the GGNS Technical Specifications. Although the identified

UFSAR discrepancies were related to the liquid and solid radioactive waste treatment systems, all of the plant's radiological releases and waste processing practices were conducted in a manner that ensured public safety and complied with all applicable regulations and requirements (i.e., the GGNS ODCM). Therefore, although inaccuracies existed in the UFSAR information, it has been determined that this condition does not represent an Unreviewed Safety Question and has no adverse impacts on the environment.

Serial Number: 1999-033-NPE Document Evaluated: ER 98/0681-00-00

BRIEF DESCRIPTION OF CHANGE, TEST, OR EXPERIMENT:

As identified in ER 98-0681-00-00, this change: (1) Accept-as-is seven specific electrical conduits and provides "Repair" instructions for two specific electrical conduits that utilize RTV-108 silicon as an internal conduit seal for fire and smoke; (2) Provides "Repair" instructions for one specific electrical conduit where a standard internal conduit seal could not be installed and an alternate design is established; (3) Accepts-As-Is three conduit penetrations through a fire rated boundary that were previously not documented on design drawings; (4) Accepts-As-Is two specific electrical conduits that pass through a fire barrier and the internal conduit seal on one side of the barrier can not be verified because the conduits pass through an inaccessible pipe chase; and (5) Accepts-As-Is eighteen specific electrical conduits that require internal conduit fire or smoke seals where the first opening from the penetration is inaccessible and the next accessible opening is used to determine the need for a fire/smoke internal conduit seal based on criteria in Note 9 on Drawing M-0800D. For the items identified above, the following are the specific conduits/penetrations affected: Item 1) (Accept-As-Is for) Two unscheduled 6"Conduits on the Ocl0l side of the Penetration CE-293B, Conduit 1BERN637 on the 0C403 side of Penetration CE-508DA. One unscheduled ¾" Conduit on the 0C604 side of Penetration CE-289FA, Conduit 1BWRM607 on the 0C703 side of Penetration CE-358G, and Conduits 1BARWS11 & 1BWRM609 on the 0C702 side of Penetration CE-359G; (Repair for) Conduit 1BDRT691 on the 0C403 side of Penetration CE-508DA & Conduit 1BDRNS70 on the 0C604 side of Penetration CE-106FA. Item 2) Conduit 1BERNS28 on the 0C403 side of Penetration CE-283DA. Item 3) Conduits 1BERN68J, 1BERN67T & 1BERM6OE through Penetrations CE-453G, CE-454G & CE-455G respectively. Item 4) Conduits 1BERM639 & 2BARQQ03 on the 0C214 side of Penetration CP-006C. Item 5) Conduit 1BERNS35 on the 0C408 side of Penetration CE-323DA, Nine unscheduled 1/2" conduits & six unscheduled 3/4" conduits on the 0C504 side of Penetration CE-007G, Conduit 1BERS61O on the 0C611 side of Penetration CE38G & Conduit 1BDRNRO4 on the 0C706 side of Penetration CE-370G.

In addition to the above, this ER makes drawing changes that are considered "software changes only" that involve seven penetrations. Four of these penetrations are in walls that are not fire barriers and were incorrectly shown as requiring fire rated penetration seals. The remaining three were penetrations that were shown on drawings but were never installed in the field. The appropriate drawings were changed to correctly show the as-built configurations.

REASON FOR CHANGE, TEST, OR EXPERIMENT:

GGNS design for internal sealing of conduits for fire and smoke is provided in Note 9 on Drawing M-0800D. The need and material type of an internal conduit seal is based on the conduit diameter, cable fill, and the distance between the fire barrier and the first opening in the conduit. RTV 108, which is installed in the nine conduit configurations identified above, is not an approved material for sealing conduits which penetrate fire barriers. Due to the physical characteristics of the cured RTV 108 material, removal of the internal conduit seal material is difficult and would most likely lead to cable damage created during the seal removal process. Therefore, the acceptability of the existing RTV seals has been evaluated. In one conduit configuration identified above, an approved seal could not be installed due to insufficient space between the electrical cables and the inside portion of the conduit. In this case an alternate method for protecting this penetration was identified. Three penetrations were identified in the field and were not documented on design drawings. The adequacy of these penetrations was evaluated and they were documented on design drawings. Two conduits were identified that pass through a fire rated penetration and the internal conduit seal on one side of the penetration can not be verified because that side of the penetration is located inside a pipe chase that has no normal means of access. Eighteen conduits were identified that require internal conduit fire or smoke seals where the first opening from the penetration is inaccessible and the internal conduit seal required by the present GGNS design could not be verified. These configurations were evaluated for acceptability.

SAFETY EVALUATION SUMMARY AND CONCLUSIONS:

The changes made by this ER (1) Accepts-As-Is seven specific electrical conduits and provides “Repair” instructions for two specific electrical conduits that utilize RTV-108 silicone as an internal conduit seal for fire and smoke; (2) Provides “Repair” instructions for one specific electrical conduit where a standard internal conduit seal could not be installed and an alternate design is established; (3) Accepts-As-Is three conduit penetrations through a fire rated boundary that were previously not documented on design drawings; (4) Accepts-As-Is two specific electrical conduits that pass through a fire barrier and the internal conduit seal on one side of the barrier cannot be verified because the conduits pass through an inaccessible pipe chase; and (5) Accepts-As-Is eighteen specific electrical conduits that require internal conduit fire or smoke seals where the first opening from the penetration is inaccessible and the next accessible opening is used to determine the need for a fire/smoke internal conduit seal based on criteria in Note 9 on Drawing M-0800D. These specific internal conduit seal configurations have been evaluated and determined to maintain the fire resistance rating requirements of the barriers as presently analyzed in the SAR. Therefore, this change will not increase the probability or consequences of accidents or malfunction of equipment important to safety previously evaluated in the SAR. Also, this change 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. “Fire Rated Assemblies” are not addressed by Technical Specifications (TS). “Fire Rated Assemblies” are addressed in the Technical Requirements Manual (TRM) Section 6.2.8. Changes made by this ER involve internal sealing of electrical conduits penetrating “Fire Rated Assemblies” addressed by the TRM; however, these changes have been evaluated and determined to provide an adequate conduit seal and to maintain the hourly rating of the fire barriers involved. Therefore, there will be no change to TS or TRM or the Bases for any TS or TRM.

Serial Number: 1999-034-NPE Document Evaluated: Calc. MC- Q1P81-90188, Revision 2

BRIEF DESCRIPTION OF CHANGE, TEST, OR EXPERIMENT:

Design Engineering calculation MC-Q1P81-90188, Revision 2, calculates new 7 / 6 day fuel oil storage requirements for the Division 3 emergency diesel generator. The new requirements are based upon the Tech. Spec. rating for this diesel generator (3300 KW) rather than the reduced loading previously used which was based upon evaluation of maximum expected post-LOCA loading of this diesel generator.

REASON FOR CHANGE, TEST, OR EXPERIMENT:

The previous calculation was based upon assumptions concerning reduced ESF loading being required after a LOCA. While such operation may be considered to be the design basis loading during first 7 days post LOCA it is felt to be a prudent course of action to not have to assume such limited loading to ensure adequate fuel oil supply. Therefore, the fuel oil storage requirement for the Division 3 emergency diesel is being increased so as to alleviate this concern.

SAFETY EVALUATION SUMMARY AND CONCLUSIONS:

Increasing the Division 3 emergency diesel generator 7 / 6 day fuel oil storage requirements to the new (higher) limits identified in this calculation revision will have no adverse effect on plant safety. The greater storage requirements are within rated capacities and provide additional fuel oil margin to allow greater operational flexibility in the event of an accident.

Serial Number: 1999-035-NPE Document Evaluated: LDC 1999-037

BRIEF DESCRIPTION OF CHANGE, TEST OR EXPERIMENT:

UFSAR Section 9B.8.3 is being revised to clarify that all personnel, whether permanent employees or contractors (i.e. referred to as temporary personnel in the UFSAR), with unescorted access to the GGNS Protected Area receive instruction in evacuation procedures and procedures for reporting fires as part of the Plant Access Training.

Section 9B.8.4 is being revised to delete the reference to the Plant Fire Chief Assistants. The GGNS Plant Fire Chief (i.e. Fire Protection Coordinator) no longer has any assistants.

Section 9B.8.5 is being revised to: 1) specify that the Fire Protection Coordinator or his designee will coordinate the training/instruction of the fire brigade members, 2) clarify the statement which implies that the instructor providing the fire training to the fire brigade members is actually a member of the brigade such that it is clear the instructor is not qualified to operate the GGNS equipment but is knowledgeable of the type and operation of the fire protection equipment installed at GGNS, 3) clearly state that the Director, Nuclear Training is responsible for coordinating the training on evacuation procedures and procedures for reporting fires for all plant personnel with unescorted access to the GGNS Protected Area and 4) clearly state that the Superintendent, Plant Security is responsible for coordinating the training of security personnel on procedures for entry of offsite fire departments and crowd control for persons exiting the plant.

REASON FOR CHANGE, TEST OR EXPERIMENT:

LDC 1999-037 is being generated to incorporate the current GGNS terminology, organizational structure and organizational responsibilities, as they pertain to training associated with the GGNS Fire Protection Program, into the UFSAR.

SAFETY EVALUATION SUMMARY AND CONCLUSIONS:

No existing Fire Protection Training practices are being changed by LDC 1999-037. LDC 1999-037 is being generated to incorporate and clarify the current GGNS terminology, organizational structure and organizational responsibilities, as they pertain to training associated with the GGNS Fire Protection Program, into UFSAR. The incorporation and clarification of current GGNS terminology, organizational structure and organizational responsibilities, as they pertain to training associated with the GGNS Fire Protection Program, into UFSAR Sections 9B.8.3, 9B.8.4 and 9B.8.5 does not impact the Technical Specifications in any way, increase the probability of occurrence of an accident previously evaluated in the SAR, increase the consequences of an accident previously evaluated in the SAR, increase the probability of occurrence of a malfunction of equipment important to safety previously evaluated in the SAR, increase the consequences of a malfunction of equipment important to safety previously evaluated in the SAR, create the possibility for an accident of a different type than any previously evaluated in the SAR, create the possibility for a malfunction of equipment important to safety of a different type than any previously evaluated in the SAR or reduce the margin of safety as defined in the BASIS: for any Technical Specification.

Serial Number: 1999-036-NPE Document Evaluated: ER 97/0390-00-00

BRIEF DESCRIPTION OF CHANGE, TEST, OR EXPERIMENT:

As identified in ER 98-0390-00-00, this change accepts-as-is the following: (1) two specific internal conduit fire seals which utilize RTV-108 silicone and (2) nine specific conduit configurations requiring internal conduit fire or smoke seals where the first opening from the penetration is inaccessible and the next accessible opening is used to determine the need for a fire/smoke internal conduit seal based on criteria in Note 9 on Drawing M-0800D. 1) RTV-108 silicone is not an approved seal material for fire boundaries. Specifically,. RTV-108 silicone has been evaluated and determined to be acceptable as an internal conduit fire seal in the following conduits/penetration: 1BARN691 & 1BARN692 in penetration CE-404BA on the 0C208A side of the fire barrier. This change does not make a generic acceptance of the use of RTV-108 as an approved material for sealing internal conduits for fire separation requirements. 2) Eight specific locations are evaluated where the first conduit opening is inaccessible due to being wrapped with a fire barrier wrap system and one location is evaluated where a physical obstruction prevents access to the first opening. The specific conduit/penetrations are as follows: 1BBRWQ04/CE-122B, 1BARNR33 & 1BARNR34/CE-186BA, 1BARWQ3l/CE-283BA, 1BARWQ36/CE-284BA, C148TELEA/CE-285BA, 1BARNR36/CE-384BA, 1BBRWQ36/CE-382BA, 1BARD6O5 & 1BARD6O6/CE-004B, and 1BDRW679/CE-193BA. This change does not make a generic acceptance for not sealing the first opening from the penetration based on inaccessible issues. In addition, ER98-0390-00-00 makes changes to penetration schedule drawings to clarify three penetration nNumbers which were issued but were not installed.

REASON FOR CHANGE, TEST, OR EXPERIMENT:

GGNS design for internal sealing of conduits for fire and smoke is provided in Note 9 on Drawing M-0800D. The need and material type of an internal conduit seal is based on the conduit diameter, cable fill, and the distance between the fire barrier and the first opening in the conduit. RTV 108, which is installed in the two conduit configurations identified above, is not an approved material for sealing conduits which penetrate fire barriers. Due to the physical characteristics of the cured RTV 108 material, removal of the internal conduit seal material is difficult and would most likely lead to cable damage created during the seal removal process. Therefore, the acceptability of the existing RTV seals has been evaluated. In the nine locations identified above, the first opening was found to be inaccessible due to either physical interference's with other equipment or the fact that the conduit is enclosed in a fire barrier wrap system (Thermo-Lag, Flammastic, or Kaowool). These fire barrier wrap systems are installed to meet either Appendix R to 1OCFR5O separation requirements or Regulatory Guide 1.75 electrical separation requirements. The requirement in Note 9 on Drawing M-0800D to seal the first opening was not verified in these inaccessible conduit configurations. Therefore, the acceptability of these inaccessible configurations has been evaluated.

SAFETY EVALUATION SUMMARY AND CONCLUSIONS:

The changes made by this ER accepts-as-is: (1) two specific internal conduit fire seals which utilize RTV-108 silicone and (2) nine specific conduit configurations requiring internal conduit fire or smoke seals where the first opening from the penetration is inaccessible and the next accessible opening is used to determine the need for a fire/smoke internal conduit seal based on criteria in Note 9 on Drawing M-0800D. These specific internal conduit seal configurations have been evaluated and determined to maintain the fire resistance rating requirements of the barriers as presently analyzed in the SAR. Therefore, this change will not increase the probability or consequences of accidents or malfunction of equipment important to safety previously evaluated in the SAR. Also, this change 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. "Fire Rated Assemblies" are not addressed by Technical Specifications (TS). "Fire Rated Assemblies" are addressed in the Technical Requirements Manual (TRM) Section 6.2.8. Changes made by this ER involve internal sealing of electrical conduits penetrating "Fire Rated

Assemblies" addressed by the TRM; however, these changes have been evaluated and determined to provide an adequate conduit seal and to maintain the 3-hour rating of the fire barriers. Therefore, there will be no change to TS or TRM or the Bases for any TS or TRM.

Serial Number: 1999-037-NPE Document Evaluated: ER 99/0049-00-00

BRIEF DESCRIPTION OF CHANGE, TEST, OR EXPERIMENT:

The UFSAR, Section 3.2 provides the methods used at GGNS to classify the systems, components and structures. It also correlates the industry Codes to the system Safety Class. For systems assigned a Safety Class of 1, 2, or 3, the piping, pumps, valves, vessels, and tanks are required to comply with ASME Section III, Subsections NB, NC, or ND respectively. UFSAR Table 3.2-4 lists the specific Editions and Addenda of ASME Section III that are applicable to the various systems/components at GGNS. In summary, GGNS safety related piping systems were designed to various editions/addenda of ASME Section III, but all installation and "NA" symbol stamping was performed to the requirements of ASME Section III, 1974 Edition with the Summer 1974 Addenda.

CR 98-0180 was initiated during a post work review of WO 171918. The CR indicates that the material documentation supplied with the studs issued under Stock Code GG89093026 does not meet the requirements of ASME Section III.

Bechtel purchased the studs as part of the original construction scope as either Unit 1 or Unit 2 material. According to Bechtel receiving documentation (MRR 84388), the studs were purchased by purchase order 9645-F-48511 in accordance with Specification 9645-M-207.0, Revision 14. The studs were turned over to Entergy as either surplus Unit 1 material or as part of the Unit 2 transfer.

Bechtel was the Architectural Engineer (AE) and ASME Section III Certificate Holder responsible for the design and installation of safety related piping systems at GGNS. As stated in the UFSAR and Specification M-204.0, the GGNS Code of Record for installation is 1974 with the 1974 Summer Addenda (hereinafter referred to as the Code). In accordance with the code (NA-3720), materials used in the construction of ASME Section III, Class 1, 2 or 3 systems are required to be obtained from Material Manufacturers/Suppliers (MM/MS) that have been determined acceptable by one of two methods:

1. The MMJMS may have a Quality System Certificate that has been issued by ASME verifying the adequacy of the suppliers/manufacturers Quality System Program, or

2. The MMJMS may have their Quality System Program surveyed and qualified by the Installer who is responsible for Code Stamping the installation of the completed piping system.

When material is obtained from an MM/MS that has been qualified as described in (1) above, the documentation transmitted by the MM/MS shall include reference to a Quality System Certificate NNumber issued by ASME, and its expiration date. When material is obtained from an MM/MS that has been qualified as described in (2) above, the documentation transmitted by the MM/MS shall include reference to the program, revision and date, used to supply/manufacturer the material and which was surveyed and approved by the Installer. Additionally, when the material is obtained from a MMJMS meeting the provisions of (2), the Installer assumes responsibility for ensuring that the material meets all Code requirements and is responsible for final certification of the material. With Bechtel, certification is accomplished when the piping system is NA Symbol Stamped and the N5 Data report is signed. In summary, somewhere in the material supply process, an ASME accredited organization with the correct scope of activities in their certificate must certify that the material meets all the requirements of the specified ASME Code.

Contrary to these requirements, the documentation for the subject studs does not contain any reference to a Quality System Certificate or a Quality Program that had been surveyed and approved by Bechtel, and since the material was not installed, Bechtel certification was also not obtained.

REASON FOR CHANGE, TEST, OR EXPERIMENT:

The nonconforming conditions documented in CR 1998-0180-00 are evaluated in ER 99-0049-00. The ER evaluation has concluded that the nonconforming material is acceptable for continued use and that the condition could exist in other situations where Bechtel purchased material has been left for Entergy use. This disposition is a change to the facility as described in the UFSAR because it permits a deviation to the stated Codes.

SAFETY EVALUATION SUMMARY AND CONCLUSIONS:

Even though strict adherence of the Code has not been complied with, the intent has clearly been met. The fact that Bechtel had received and stocked the material indicates that their QA program requirements had been fulfilled. The purchase specification clearly indicated that the material was required to meet ASME Section III requirements and was received as such. If Bechtel had not accepted the material because of not complying with the ASME Code it would not have been stocked for use during construction. The only anomaly that causes the nonconformance is that an agreement had not been reached between Entergy (MP&L) and Bechtel to have the un-used materials certified by Bechtel before their departure from GGNS. This nonconformance is a compliance issue with administrative rules to document the use of Quality Programs and does not affect the subject materials ability to perform their designed safety function. As such, there is not an increase in the probability of occurrence or consequence of an accident or malfunction previously evaluated in the UFSAR. Also, the deviation will not result in an accident or malfunction of a different type than those previously evaluated in the UFSAR. All subject materials still meet their mechanical and quality requirements thus their performance during normal and accident conditions are not diminished, thus there is not a reduction in any margins of safety as defined in the basis for the GGNS Technical Specifications.

Serial Number: 1999-038-NPE Document Evaluated: TA 99-016

DESCRIPTION OF CHANGE, TEST OR EXPERIMENT:

The Div 3 D/G jacket water and lube oil systems are maintained at standby temperatures by the jacket water immersion heater which is controlled by temperature switch 1P8lN044A/B. The jacket water heater provides direct heat to the jacket water which in turn transfers heat to the lube oil through the lube oil cooler. Since there is no forced flow of jacket water during standby, all heat transfer is accomplished by convective flow.

The low/high jacket water temp alarm (1P81L013) is actuated by temperature switch 1P8IN006A/B which is located after all heat transfer points.The current setpoint of this switch is 95ⓒF. The current 1P81N044A/B setpoints are 1 35ⓒF(CF) and 155ⓒF(OR), which turns the heater on at 135ⓒF and cuts the heater off at 155ⓒF jacket water temperature.

This temporary alteration will raise the lower setpoint of 1P81N044A from 135ⓒF (CF) to 140ⓒF (CF) to help prevent the low/high jacket water temp alarm from coming in unnecessarily on local panel 1H22P118 during standby conditions. Temperature switch 1P81N044B is not being affected by this temporary alteration.

REASON FOR CHANGE, TEST OR EXPERIMENT:

The current lower setpoint of the heater control switch 1P81N044A allows the low temperature alarm switch 1P8 IN006A to reach and briefly remain at it's alarm setpoint before the heater is able to return the temperature above 95ⓒF at the alarm switch.

SAFETY EVALUATION SUMMARY AND CONCLUSIONS:

No materials/components are being added or removed by this temp alt. No physical changes are being made. This temp. alt. will raise the lower setpomt of Engine A JW Htr Temp Cntrl switch, 1P8IN044A, from 13 (CF) to 140ⓒF (CF) to help prevent low/high jacket water temp alarm from coming in on local panel 1H22P118 during standby conditions. The current 1P81N044A setpoints are 135ⓒF (CF) and 155ⓒF (OR), which turns the heater on at 135ⓒF and cuts the heater off at 155ⓒF. The upper setpoint of 155ⓒF remains the same. This temp alt is consistent with maintaining the desired Div 3 D/G jacket water and lube oil standby temperatures per vendor recommendations. This temp alt does not affect operation, or operating parameters, of the Div 3 D/G.

Serial Number: 1999-039-NPE Document Evaluated: CR 1999-0801

BRIEF DESCRIPTION OF CHANGE, TEST, OR EXPERIMENT:

The purpose of this change is to evaluate leaving a Turbine Building Roof Hatch open during maintenance activities in the upper area elevations of the Turbine Building. The open roof hatch would be monitored for flow and isotopic releases during the period the hatch is opened.

REASON FOR CHANGE, TEST, OR EXPERIMENT:

Due to elevated temperatures in the ceiling area of the turbine building, work crews experience significant heat stress. It is proposed to leave a Turbine building roof hatch open during these work activities to provide cooler temperatures in the ceiling area of the turbine building. This reduces the temperature that work crews would be exposed to in the turbine building ceiling therefore reducing the heat stress experienced by the workers.

SAFETY EVALUATION SUMMARY AND CONCLUSIONS:

Maintenance work crews experience significant heat stress while working in the turbine building ceiling area. This evaluation addresses leaving a hatch open to reduce worker heat stress in the ceiling area. The open roof hatch would be monitored for flow and isotopic releases during the period the hatch is opened. The turbine building roof hatches were designed to provide additional ventilation in the turbine building in case of fire. Although no immediate safety threat exists, an unmonitored release pathway is created by inadvertently leaving a roof hatch opened and unmonitored. The proposed change modifies the function of the turbine building roof hatches by allowing a roof hatch to be opened to provide cooling to the ceiling area of the turbine building. The open roof hatch would be monitored by an isokinetic probe and flow element (similar to a pitot tube). This would be a planned release that will be included in the Annual Effluent Release Report.

The modification would require placing a flow element and isokinetic probe in the opening. Upon opening the turbine building roof hatch, there will be a “puff” release due to the high temperature initially in the turbine building ceiling area. The time between the opening the hatch and locating the flow element and isokinetic probe may be assessed using flowrate information from Calculation XC-Q1U22-930003, “Turbine Building Flowrates”. This provides a representative assessment of the activity released during the time of installing the instrumentation. Likewise for closing the roof hatch, the time period between removing the instrumentation and closing the turbine building roof hatch could be assessed using the same calculation. For opening or closing the hatch, the temperature in the area needs to be taken so that the calculation can be used. The outside temperature assumed in the calculation is 80ⓑF in order to maximize temperature differential between inside the turbine building and outside. The minimum 2.5% summer design temperature for Vicksburg is 78ⓑF and which provides less than 0.4% error between the absolute temperatures. This is the basis for using 80ⓑF outside temperature. The calculation may be revised based on plant conditions in order to remove temperature limitations concerning assuming 80ⓑF outside temperature and provide a more accurate representation of flowrates through the open hatch. It is recommended to use actual flowrates where possible to assess radiological conditions and use the calculation only when flowrate measurement cannot be performed. Flowrates are anticipated to be approximately 30,000 cfm based on 80ⓑF outside temperature and 130ⓑF turbine building ceiling temperature. This flow would decrease as the turbine building ceiling temperature decreases due to the open roof hatch. The calculation can be used if outside temperature is greater than 80ⓑF.

Shortly after the hatch is opened a continuous isokinetic sampler will be installed so that actual radioactive particulate and iodine released can be later quantified and included in the annual release report. Periodic grab samples will be taken for noble gases. Periodic measurements of the air flow rate

through the open hatch will also be taken to quantify the actual release. The sampler will be periodically checked to verify that it is functioning. Continuous monitoring and alarms are not required because the annual release through an open hatch will be administratively limited to 10% of the annual release through the monitored turbine building vent. This is considered insignificant.

The Chemistry department will establish radiological limits based on 10% of the annual release. A time limit will be developed for opening the turbine building roof hatch. This limit will be based on a certain percentage of the Turbine Building Ventilation Exhaust flowrate versus roof hatch flowrate and the radiological concentration expected in the Turbine Building ceiling area. The time limit and radiological limit will be established by Chemistry.

Outside air is induced into the turbine building above the operating floor and then drawn down to the areas below the floor by the exhaust system. A greater volumetric flow is exhausted from the turbine building than is supplied to it to assure that no outleakage of air will occur. A pressure control system maintains a slight negative pressure in the turbine building with respect to atmosphere by modulating dampers located in the discharge duct of the building and in the duct return to the operating levels. Opening a turbine building roof hatch and exhausting hot air to atmosphere from the turbine ceiling should have no adverse effect on the pressure control system.

Air flow control is from areas of low potential radioactivity to areas of high potential radioactivity. In this way, clean area passageways are kept free of radioactive contaminants. Opening a turbine building roof and exhausting hot air to atmosphere from the turbine ceiling area will not affect the passageways because hot air is being removed from the turbine ceiling area and allowing outside air to come in through the turbine building roll up door.

The turbine building exhaust air system exhausts air from the condenser area, turbine building equipment compartments, and the turbine building equipment drain sumps. Air is drawn out and exhausted to the vent.

Anytime the turbine building roof hatch is to be opened so that work in the turbine building ceiling area may begin; permission needs to be obtained from the Operations Shift Superintendent. Anytime the roof hatch is closed the Operations Shift Superintendent is to be notified.

Workers that will be in the area of the open roof hatch will periodically monitor the release point. Anytime the workers leave the work area, the instrumentation should be removed and the turbine building roof hatch should be closed. In the event of an emergency that would require workers to leave the ceiling area of the Turbine building, the instrumentation would be removed and the turbine building roof hatch closed. The accident dose analyses do not credit the turbine building roof or roof hatch as a radiological barrier. Therefore, the temporary use of radiological and flowrate instrumentation is only required for meeting the normal operating effluent monitoring requirements.

The proposed activity involves locating an isokinetic probe and flow element (i.e., pitot tube) at the discharge point in the turbine building roof hatch. This meets the requirements of 10CFR50 Appendix A, GDC 64, for monitoring effluent discharge paths, and the plant environs for radioactivity that may be released from normal operations including anticipated operational occurrences. Similarly Regulatory Guide 1.21 specifies requirements for monitoring for releases. Regulatory Guide 1.21, Appendix A would describe this as a “Batch Release” because this release point (i.e., roof hatch) will not be continuously open. The Reg. Guide states “For reactors which release gases intermittently, an analysis should be made of a representative sample of each planned release prior to discharge to determine the

identity and quantity of the principal radionuclides released. Continuous monitoring should also be conducted at appropriate points to obtain information on the quantity and pattern of abnormal releases." For the turbine building roof hatch release path, a "grab" sample, isokinetic probe and pitot tube which is periodical monitoring will be collectively considered as continuous monitoring of the release point.

Serial Number: 1999-040-NPE Document Evaluated: Calc. MC-Q1P75-90190, Revision 2

BRIEF DESCRIPTION OF CHANGE, TEST, OR EXPERIMENT:

Design Engineering calculation MC-Q1P75-90190, Revision 2, calculates new 7 / 6 day fuel oil storage requirements for the Division 1 / 2 emergency diesel generators. The new limits are based on post LOCA loads that are more realistically determined, based on actual vendor and startup test data, rather than the reduced loading used which was based upon limited operation of ECCS pumps post-LOCA. The Tech Spec testing rating of 5740 KW is greater than the calculated post LOCA load plus 10%. Since the useable fuel oil storage requirement is directly proportional to the assumed electrical loading the 10% margin requirement specified by Reg. Guide 1.137 / ANSI N195-1972 (for calculations which do not use the nameplate rating of the diesels) is met.

REASON FOR CHANGE, TEST, OR EXPERIMENT:

The previous calculations were based upon assumptions concerning loading such as limited ECCS pump operation (i.e. less than seven full days) being required after a LOCA. While continuous operation during the first 7 days post LOCA may not be required (as discussed in UFSAR Sections 6.2 and 6.3) it is felt to be a prudent course of action to not have to require such limited operation to ensure adequate fuel oil supply. Therefore the fuel oil storage for the Division 1 and 2 emergency diesels is being increased so as to alleviate this concern.

SAFETY EVALUATION SUMMARY AND CONCLUSIONS:

Increasing the Division 1 and 2 emergency diesel generator 7 / 6 day fuel oil storage requirements to the new (higher) limits identified in this calculation revision will have no adverse effect on plant safety. The greater storage requirements are within rated capacities and provide additional fuel oil margin to allow greater operational flexibility in the event of an accident.

Serial Number: 1999-041-NPE Document Evaluated: ER 97/0949-00-00

BRIEF DESCRIPTION OF CHANGE, TEST, OR EXPERIMENT:

The changes addressed in this ERR are to the Floor and Equipment Drainage System (P45). They involve the updating of design drawings to reflect the existing piping configuration for a DRW drain that was found closed and covered in Health Physics area of the Control Building. Instructions will also be given for closing and covering a DRW drain outside of the CAA, and a sanitary drain inside the CAA. The Drains are located in Areas 25 A & B, Elevation 93'-0" of the Control Building

REASON FOR CHANGE, TEST, OR EXPERIMENT:

Per a field walkdown by Design Engineering there is a sanitary drain (as documented in CR 97-0216-01 now 1997-1337-00) located in the CAA just north of the Control building door to the Turbine building on floor elevation 93'-0'. This drain could allow radioactive particles into the sanitary drainage system. Per discussions with HP the sanitary sewage system is monitored prior to release and some raw sewage has been contained due to it being contaminated. This sanitary drain should be plugged because it is located in an area where there is major foot traffic and personnel have not been monitored for contamination. Also the CR documented a DRW floor drain that has been plugged on elevation 93'-0" of the control building inside the CAA next to the portal monitors. Drawing M-0075 zone G3 does not reflect a plugged floor drain. The Design Engineering walkdown confirmed the drain is covered/plugged. Health Physics would like this floor drain to remain plugged. And they would like an additional DRW floor drain that is just outside of the CAA in the same area of the Control Building covered/plugged. The ER response will update the affected drawings to reflect existing conditions in the plant, and provide instructions for the closure and covering of the affected DRW and sanitary drains.

SAFETY EVALUATION SUMMARY AND CONCLUSIONS:

Section 9.3.3.1.1.C of the FSAR states that the only portion of the Floor and Equipment Drainage System identified as safety-related is the Containment and Drywell penetrations. This portion of the Floor and Equipment Drainage System is not being modified or affected by the modifications proposed in ER 97/0949-00-00. Failure of the components modified by this ER will not compromise any safety-related equipment or component, and will not prevent safe shutdown of the plant. The modifications documented by ER 98/0949-00-00 will in no way impact any of the accident analyses presented in the FSAR. The response to this engineering request provides for the updating of design drawings to reflect existing conditions, and instructions for closing and covering two additional drains. The existing configuration and the proposed modifications creates no new failure modes, thus no possibility of an accident or malfunction of a different type than previously analyzed is possible. Failure of these components will not cause a system failure, therefore this modification will not compromise any safety-related system or component and will not prevent safe reactor shutdown, thus no margin of safety will be reduced.

Serial Number: 1999-042-NPE

Document Evaluated: ER 96/0421-01-00
ER 97/0283-00-00
ER 97/0288-02-00
ER 96/0350-00-00

DESCRIPTION OF CHANGE, TEST, OR EXPERIMENT:

The Nuclear Change Response to the following ERs, enhances the operating torque margin (i.e., the difference between the torque required to operate the valve and the torque limit of the actuator) and the thrust capabilities for the indicated motor operated valves (MOVs):

ER 96/0421-01-00 - Operating Margin Improyement for Component Cooling Water MOVs
QlP42Fl14-B - CCW Supply To Containment Drywell Outboard Isolation Valve
QlP42F116-A - CCW Return from Containment Drywell Inboard Isolation Valve
QlP42F117-B - CCW Return from Containment Drywell Outboard Isolation Valve

ER 97/0283-00-00 - Operating Margin Improvement for Nuclear Boiler System MOVs
Ql B21F016-B - Main Steam Line Inboard Drain Valve
Q1B21F019-A - Main Steam Line Outboard Drain Valve

ER 97/0288-02-00 - Operating Margin Improvement for Reactor Water Cleanup MOVs
Q1G33F028-B - RWCU to Main Condenser Inboard Isolation Valve
Q1G33F039-A - RWCU to RHR System Outboard Isolation Valve
QlG33F040-B - RWCU to RHR System Inboard Isolation Valve

ER 97/0350-00-00 - Operating Margin Improvement for Low Pressure Core Spray and RHR MOVs
Q1E2IF011-A - LPCS Min Flow to Suppression Pool Isolation Valve
Q1El2F009-B - RHR Shutdown Cooling Inboard Isolation Valve

This will be accomplished by replacing the motor pinion gear and worm shaft gear sets in each actuator to increase the overall actuator ratio (OAR), and by replacing the actuator motor with an actuator motor capable of increased torque output. The valves listed above will have their yoke legs stiffened and/or a new valve yoke assembly with legs stiffened will be installed. An increased Kalsi thrust rating will be applied to the RWCU valve actuators. The inactive leakoff nipples for valves Q1G33F028, Q1G33F039, Q1B21F019 and Q1E21F011 will be plugged. New actuators will be installed for the Nuclear Boiler, CCW and LPCS system MOVs. Valves QlB21F019 and QlE12FOO9 will have new valve stems installed. Valve Q1E12F009 will have high strength yoke/actuator bolts installed and valve QlE21F011 will have high strength body/bonnet bolting installed. Also, the instantaneous breaker settings for the valve motors are being increased and the appropriate thermal overloads will be installed due to the larger horsepower motors.

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 the selected valves, the margin between required torque/thrust versus maximum available torque/thrust can be increased.

SAFETY EVALUATION SUMMARY AND CONCLUSIONS:

The changes made by these ERs will increase the degraded voltage actuator capability (DVAC) torque for the Limitorque motor operators installed on the subject valves. The changes made by these ERs will increase the operating margin for each of the motor operated valves listed above, while addressing industry concerns related to gearbox “run” efficiency.

The UFSAR does not address the operating torque/thrust requirements or capabilities of the subject motor operated valves. The operator, stem, bolting, leak off port, motor and gear changes (as applicable to the individual valves) do not alter the function or operation of these MOVs. The slower stem speed resulting from the OAR change on these valves causes the nominal stroke time for the valves to increase. Based on the fact that the post-modification calculated stroke times for each of the valves which have analytical stroke times are bounded by the maximum isolation time contained in TRM Table TR3.6.1.3-1 and the analytical stroke times contained in UFSAR Table 5.2-5 and UFSAR Table 6.2-44, as applicable, the new valve stroke times will not affect the safety analysis, function or operation of the valves (Reference calculation MC-Q1111-98002, Rev. 1).

These ERs do not adversely impact the electrical penetration assembly, and reactor containment electrical penetrations are not described in the Technical Specification; however, they are described in the Technical Requirements Manual (TRM) Section 6.8.1. TRM Section 6.8.1 states that each of the primary and backup overcurrent protective devices associated with each primary containment electrical penetration circuit, as shown in TRM Table 6.8.1-1, shall be operable. TRM (and UFSAR) Table 6.8.1-1 lists the breaker settings for 52-1511-44 (Q1P42F116-A), 52-1611-25 (Q1P42F117-B), 52-1611-31 (QIP42F114-B), 52-1631-20 (QlB21F016-B), 52-1631-37 (Q1E12F009-B), 52-1631-50 (Q1G33F028-B) and 52-1631-52 (Q1G33F040-B) and is being updated to reflect the breaker setting changes due to the ERs. Breakers 52-1511-40 (Q1G33F039-A), 52-1531-10 (Q1B21F019-A) and 52-1511-34 (Q1E21F011-A) are not listed in TRM Table 6.8.1-1 since these valves are outside containment and, therefore, do not utilize a reactor containment electrical penetration. The protective settings for the breakers are set such that the breakers will trip before the penetration can be damaged by fault currents. Fault protection is still provided using the new settings by breakers and fuses which protect them as demonstrated in calculation EC-Ql111-99003, Rev. 0. The ERs do not change the requirements to test these breakers.

UFSAR Tables 8.3-1 and 8.3-2 lists the MOV load as 185 kW for the Division I and II ESF Loads respectively. Note d in the tables for the MOV load indicates that the load is an intermittent load and not included in the long-term loading. Additionally, it states that the loads are considered in the voltage drop calculations. Calculation EC-Q1111-90028, AC Electrical Power Systems, does include the MOV loads even though it does not categorize MOV loads as a single grouping. The kW loading for MOVs in UFSAR Tables 8.3-1 and 8.3-2 are being revised to reflect the increase in load resulting from each of the ERs evaluated by this safety evaluation.

The increased Kalsi thrust rating for the RWCU MOVs (Q1G33F028, Q1G33F039 and Q1G33F040) will be applied by increasing the bolt torque on the connections between the actuator and the valve yoke legs. These changes will allow the valve to withstand the higher actuator torque and thrust capability in the closing direction initiated by the motor and gear changes for these valves.

The replacement of the existing SMB-000 actuators with SMB-00 on valves Q1B2lF0l6, QlB2lF0l9, QlP42Fl14, Q1P42F116, QlP42Fl17, and QlE21F01l is necessary to allow the use of higher torque motors for these valves (i.e., 25 ft lbf in place of the existing 5 ft lbf motors). The yoke legs will be strengthened for valves Q1G33F028, Q1G33F039, Q1G33F040, Q1B21F0l6, Q1B2lF0l9 and QlEl2F009 to accommodate the increased thrust capabilities of the higher torque motors. Additionally, new yokes will be required for valves QlB21F016, QlB2lF019, Q1E21F011, QlP42Fl14, QlP42F116 and Q1P42F117 to permit the mounting of the SMB-00 actuators and to accommodate the increased thrust capabilities of the higher torque motors/actuators.

The leakoff nipples for valves Q1G33F028, Q1G33F039, QlB21F0l9 and QlE21F0l1 are being removed and the hole plugged. This is being done since the nipples must be removed to remove the yokes on this

type valve and then welded back if they are retained. However, since these leak off nipples are not utilized they will not be welded back on, instead the hole will be plugged. This is an ASME code pressure boundary modification, therefore, it will be performed in accordance with ASME code requirements.
Two plant valves (feedwater isolation valves Q1B21F06SA and B) have previously experienced stem cracking. It was determined that the failures were a result of hydrogen embrittlement cracking. Valves Q1B2lF0l9 and Q1E12F009, which are similarly exposed to high temperature reactor chemistry water, could also be susceptible to hydrogen embrittlement cracking. The valve stems for valves Q1B21F019 and Q1El2F009 are being removed and replaced. These two valve stems removed will be subjected to examination and analysis to determine if they have any indication of cracking similar to that experienced by the feedwater valves. New valve stems with materials resistant to this type effect, will be used to replace the valve stems removed.

The Q1E21F011 valve will have its body/bonnet bolting replaced with high strength bolting. It was determined that the body/bonnet bolting was the weak link for the MOV. Additionally, valve Q1E12F009 will have high strength yoke/actuator bolts installed for the same reason. Replacement of the bolting will allow the MOVs to accommodate the revised thrust rating for the valves following the replacement of the valves actuators and motors.

No new interfaces with equipment important to safety are created and no new failure modes which would alter existing accident analyses are introduced. There will be no increase in 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 changes made by these ERs will not create the possibility of an accident or malfunction of a different type than any other evaluated previously in the safety analysis report. There will be no reduction in the margin of safety as defined in the bases for any technical specifications. Therefore, these changes will not introduce an unreviewed safety question.

Serial Number: 1999-043-NPE Document Evaluated: ER 98/0129-00-00

BRIEF DESCRIPTION OF CHANGE, TEST, OR EXPERIMENT:

LDC 98-011 was written to change-the conductivity high alarm values and the range in UFSAR page 5.4-78, UFSAR table 5.2-8, and UFSAR Section 10.4.6.5 page 10.4-29. The high alarm values for Reactor water recirculation loop and RWCU Inlet will be changed to 0.2 µmho and the range will be changed to 0-0.5 µmho per ER 98/0129-00-RO to reflect the alarm setpoint for conductivity switches 1G33N602A/B. The range for the RWCU Outlet and Drive Water Filter Discharge Control Rod Drive System (CRD) for conductivity monitoring will be changed to 0-0.2 µmho per ER 98/0129-00-RO in UFSAR table 5.2-8. The values for the "Alarm High High" will be deleted from the UFSAR table 5.2-8 since the Technical Specifications do not require a HIGH HIGH conductivity value and the alarm was never in the plant. Condensate Cleanup Inlet and Outlet conductivity setpoints will be changed to 0.1 µmho and 0.065 µmho on page 10.4-29 in section 10.4.6.5 of the UFSAR. UFSAR Section *5.4.15* (references) will be revised to include the EPRI Guidelines reference.

REASON FOR CHANGE, TEST, OR EXPERIMENT:

ER 98/0129-00-RO was written to disposition CR 97/0444-00 to correct the CDB alarm setpoint value for conductivity switches 1G33N602A/B and direct the plant maintenance staff to update calibration procedure 07-S-53-G33-8. The new setpoint will be set at 0.2 µmho so that the alarm in the control room will alert operators of a high conductivity level in the Reactor Recirculation and the RWCU inlet. The basis for 0.2 µmho is per EPRI TR-103515-R1 BWR Water Chemistry Guidelines-1996 Revision. EPRI recommends that the conductivity level in the reactor water not exceed 0.3 µmho during mode 1 to mitigate Intergranular Stress Corrosion Cracking (IGSCC). The proposed change to 0.2 µmho is more conservative than the 1.0 µmho limit specified in Technical Requirements Manual (TRM) section 6.4 and will allow plant staff to control conductivity levels in the reactor coolant before the EPRI recommended limit of 0.3 µmho is exceeded or the TRM limit is exceeded. The change is also more conservative than the "alarm high" value of 0.7 µmho that is now stated in table 5.2-8 of the UFSAR. The ranges for the conductivity monitoring instruments will be lowered to allow for better resolution of the conductivity recorder scales. The EPRI Guidelines also recommend that the conductivity levels in the Condensate Cleanup Inlet and outlet do not exceed 0.1 µmho and 0.065 µmho.

SAFETY EVALUATION SUMMARY AND CONCLUSIONS:

The License Document Change (LDC) and Engineering Response (ER) evaluated by this safety evaluation concludes that the change does not involve an unreviewed safety question. The UFSAR and CDB changes made per ER 98/0129-00-RO will not compromise any existing safety related system, structure, or component nor will they prevent safe reactor shutdown. Failure of the instruments to which setpoint and range changes will be made will not initiate any evaluated transient or accident. The changes made will only affect the conductivity alarm setpoints and recorder ranges for Reactor Recirc., RWCU Inlet and Outlet, CRD, and Condensate Cleanup Inlet and Outlet sampling. Other means of chemistry control related to conductivity will not be affected. The new setpoints are more conservative than the TRM limit of 1.0 µmho and the UFSAR setpoints of 0.7 µmho. 0.5 µmho. and 0.1 µmho. Therefore, the change made to the UFSAR and CDB will not increase the probability of occurrence of an accident previously evaluated in the SAR.

The change made to the setpoint values do not degrade the design basis performance of a safety related system assumed to function in the accident analysis. The change to the UFSAR and CDB will not increase the probability of occurrence of a malfunction of equipment important to safety previously evaluated in the UFSAR. There is no accident evaluation in the UFSAR for conductivity monitoring instruments. The change to the UFSAR and CDB per ER 98/0129-00-RO will not alter, degrade, or prevent actions described or assumed in an accident discussed in the UFSAR nor will it create the possibility for an accident of a different type than previously evaluated in the UFSAR. The levels at

which GGNS monitors conductivity do not affect the consequences of an accident previously evaluated in the UFSAR. This change will not change radionuclide population, the release rate or duration, create new release mechanisms or impact radiation release barriers but will be unique to the conductivity alarm units and annunciators for each sample point. Therefore, the change to the UFSAR and CDB will not increase the consequences of a malfunction of equipment important to safety previously evaluated in the UFSAR nor will it create the possibility for a malfunction of equipment important to safety of a different type than any previously evaluated in the SAR.

Reactor Recirc., RWCU Inlet and Outlet, CRD, and Condensate Cleanup Inlet and Outlet conductivity instrumentation or alarm values are non-safety related and are not mentioned in the Technical Specifications or Reg. Guide 1.97. However the maximum values for conductivity in the reactor coolant and the actions to take if the limits are exceeded are mentioned in the technical requirements manual (TRM). The changes made to the CDB and UFSAR table 5.2-8, section 5.4.8.2 on page 5.4-78, and section 10.4.6.5 on page 10.4-29 on conductivity alarm setpoints will not affect the maximum values listed in the TRM. The Conductivity alarm setpoints will be conservative compared to the TRM limit of 1.0 μmho. Therefore, the margin of safety as defined in the basis for any Technical Specification will not be reduced. The LCO's are based on Reg. Guide 1.56**.** GGNS administratively uses the EPRI Guidelines which have much stricter limits on conductivity. There have been no NRC requirements issued that require GGNS to incorporate EPRI Guidelines into the LCO's.

Serial Number: 1999-044-NPE Document Evaluated: ER 98/0091-00-00

BRIEF DESCRIPTION OF CHANGE, TEST, OR EXPERIMENT:

The changes addressed in this ER are to the Domestic Water System (P66). They involve the updating of design drawings to reflect the existing piping configuration for what once was an installed electrical water cooler (EWC-1) and isolation valve (P66F912). These components no longer exist, the supply line has been capped and the drain line plugged. The water cooler and isolation valve were located in the Control Room Corridor, Room OC509, Elevation 166'-0" of the Control Building.

REASON FOR CHANGE, TEST, OR EXPERIMENT:

On 01/06/96, CI 53285 was initiated by Operations to document a failure of the sweat fitting connecting valve SP66F912. The CI requested a compression fitting and cap be installed on the line to stop the leak since the valve was no longer attached to the 3/8" copper pipe. CI 53285 was later converted to WO 159492 with instructions to re-install the valve. The WO "Work Performed" section stated "Investigated valve and fining. Found that work had already been done by Bechtel. OPS informed us that the job was performed previously." WO 159492 was RTO'd on 3/20/96. The work order was then closed prior to restoring the valve and piping to the proper design configuration. Upon further investigation it was determined that the electric water cooler (EWC-1) supplied by valve SP66F912 was not installed as depicted on the FSK, P&ID, Floor Plan and Plumbing drawings. Design Engineering and System Engineering discussed with Operations the reinstallation of the water cooler and isolation valve to conform with plant documents. Operations requested that the valve and water cooler be removed permanently. The ER response will update the affected drawings to reflect existing conditions in the plant.

SAFETY EVALUATION SUMMARY AND CONCLUSIONS:

The Domestic Water System has no safety-related function as defined in Section 3.2 of the FSAR. Failure of the system will not compromise any safety-related equipment or component and will not prevent safe shutdown of the plant. The modifications documented by ER 98/0091-00-00 will in no way impact any of the accident analyses presented in the FSAR. The response to this engineering request provides for the updating of design drawings to reflect existing conditions. The existing configuration creates no new failure modes, thus no possibility of an accident or malfunction of a different type than previously analyzed is possible. Failure of the system will not compromise any safety-related system or component and will not prevent safe reactor shutdown, thus the margin of safety will not be reduced.

Serial Number: 1999-045-NPE Document Evaluated: ER 96/0166-01-00

BRIEF DESCRIPTION OF CHANGE, TEST, OR EXPERIMENT:

The existing pumps in the liquid rad monitor sample panels 1D17J005 (SSW A-P41), 1D17J006 (SSW B-P41), SD17J007 (Radwaste-G17), 1Dl7J008 (CCW-P42) and 1Dl7J015 (PSW-P44) will be replaced with new pumps with a discharge pressure of 52 psig at 15 gpm and a 1/2 gpm recommended minimum flow. The pressure rating of the new pumps (excluding seals) is 400 psig. The seals will have a 1250 psig hydrostatic rating and a 250 psig 20,000 hr rating at 3450 rpm. The shutoff head of the new pumps is 87 psig. With the new pumps, a flow of greater than 15 gpm is achievable for all panels. Since 15 gpm is the current design flow rate and the max recommended flow for the new pumps, this will be the new high flow alarm setpoint. The low alarm setpoints of the flow meters will be raised from 3 to 8 gpm. A flow this low may result in fouling and is indicative of other problems (possibly a leak or mispositioned valve). With the new pumps the max expected pump discharge pressure is increased from 220 to 250 psig at 15 gpm. The pressure rating of the panel was increased from 150 to 260 psig based on a letter from GE (GEXI 97/0117) and a independent analysis. The MS-02 design pressure of the discharge piping was raised from 180 to 260 to allow for pump discharge pressure. If valves 1D17F305 (F311), 1D17F306 (F312), or 1P4lF198A (F198B) on the sample pump discharge are closed while the SSW sample pump is running the qualified pressure rating of the panel could be briefly exceeded when the SSW A (B) system pump is shutdown because of the high shutoff head of the new sample pumps. Administrative controls will be placed on the positioning of affected valves to ensure that the 260 psig qualification rating is not exceeded. Based on vendor data the minimum hydrostatic pressure rating of the panel components is 300 psig. This pressure will not be exceeded even if the sample pumps are deadheaded.

A seismic/pressure qualification analysis and a limited scope in situ commercial grade dedication will justify upgrading the classification of the SSW rad monitors from non safety, nonseismic to safety related, seismic category I.

REASON FOR CHANGE, TEST, OR EXPERIMENT:

The pumps in the liquid radiation monitor sample panels are not adequately suited for the intended application. The discharge pressure of the existing pumps (24 psig at 15 gpm) is so low that they are not capable of achieving the 12 gpm recommended minimum flow for the pumps. Low flow also causes the sample lines to foul which further reduces flow. Flow rates for the SSW rad monitors frequently drop below the 3 gpm low flow setpoint and activate the control room annunciator. The pump seals and gaskets have had a high failure rate because of low pressure ratings. The type 21 pump seals are rated for 275 psig hydrostatic and 100 psig at 3450 rpm per John Crane. The shutoff head of the existing pumps is 26 psig. The pressure rating of the pumps (excluding the seals) is 200 psig. The pressure rating for the existing panels is 150 psig per GE drawing 828E132. Per MS-02, the design pressure for the associated suction and discharge piping is 180 psig. Per CR 98/0287, the seals and discharge piping will be exposed to 220 psig (195 psig per the CR plus 25 psi pump discharge pressure)

The SSW system piping is seismic category I. The rad monitor panels were supplied non-safety related by GE and were identified as non safety related in the Bechtel GGNS instrument index and the GGNS CDB. Per NRC Reg Guide 1.26 “Quality Group Classifications and Standards” and NRC Reg Guide 1.29 “Seismic Design Classification”, non seismic piping can be connected to seismic category 1 piping only if it is isolated by a normally closed valve or a valve capable of automatic closure. GGNS does not take exception to this requirement in UFSAR appendix 3A. Also FSAR section 9.2.1.1.1 (e) states that two valve isolation is provided between the nonseismic - seismic interfaces of SSW. Contrary to this requirement the manual root valves for the SSW rad monitor panels lD17J005 & 6 are normally open.

SAFETY EVALUATION SUMMARY AND CONCLUSIONS:

The subject panels monitor the radiation levels of liquid process and effluent streams. An increase in radiation levels is indicative of heat exchanger leakage or other equipment malfunction that could result in a radioactive release. The rad levels measured by the SSW monitors Dl7J005 & 6 are recorded in the control room and rad levels measured by the radwaste effluent monitor SD17J007 are recorded in the radwaste control room. When the high alarm setpoint of the monitors is exceeded a control room annunciator is activated. The radwaste discharge monitor also has a high-high alarm setpoint that will close the valve SG17F355 to prevent the release of water that has higher than allowed concentration of radioactive material to the Miss. River (via the plant discharge basin). This isolation is not considered a safety related function. It is a backup to the laboratory analysis of grab samples. Each rad monitor has a flow meter with high and low flow setpoints that also activate control room annunciators. A low flow alarm is indicative of a stopped pump, a mispositioned valve or a leak in the suction piping. A high alarm is indicative of a improperly set flow control valve or a leak in the discharge piping. None of the control room indications and alarms are safety related. These monitors are not required by reg guide 1.97 and are non IE powered. The SSW rad monitors must be seismically qualified because the sample panels Dl7J005 & 6 are part of the SSW system pressure boundary and are 40 ft below ground level (133-93). A catasfrophic failure of these panels (a complete line break) could potentially drain the SSW basins below the 30 day required inventory. Upgrading the safety/seismic classification of the panels will ensure that the seismic qualification is maintained. The SSW rad monitors are located in areas where they will not be damaged by missiles, jet impingement, flooding, II/I hazards, etc. They are not considered to be ASME code items. The other panels are nonsafety related, non-seismic. The liquid radwaste monitor is an appendix B Q-list item. The operation and function of the monitors and interfacing systems is not affected. No new interfaces are created. This design change will improve the reliability of the existing rad monitoring system. This instrumentation is addressed in TRM 6.3.1. No change to the TRM is required. The function of the radiation monitors does not meet the criteria specified in 10CFR50.36 for inclusion in the current technical specifications. This change is to upgrade the components for reliability and to address the classification of the components which interface with the pressure boundary of the safety related SSW system. The basic function of the radiation monitors is not affected and there is no change to the functions of any other systems. Therefore, a change to the tech specs is not required. The failure of the affected instruments is not evaluated in chapter 15 of the FSAR. The changes will not affect any other systems or components whose failures are evaluated. The FSAR safety evaluations for the SSW-P41 system (sect 9.2.1.3), CCW-P42 system (sect 9.2.2.3) and PSWP44 system (sect 9.2.8.3) are not affected by these changes. The required 30 day inventory for SSW includes an allowance/margin for 57 gpm total leakage from pump and valve seals (ref. calculation MC-Q1P41-86007 rev 0). The new pump seals will be more reliable than the existing seals. Any leakage will be negligible.

Serial Number: 1999-046-NPE Document Evaluated: ER 97/0089-00-00

BRIEF DESCRIPTION OF CHANGE, TEST, OR EXPERIMENT:

The design change installs a new large passive strainer (Q1M24D001) that rests on the floor of the suppression pool and encircles the suppression pool. One of the two existing strainers connected to each emergency core cooling system (ECCS) and reactor core isolation cooling (RCIC) system suction penetration piping tee will be removed and the new strainer will be connected to the tee where the strainer was removed. The other existing strainer, for the ECCS and RCIC, will remain connected to the suction tee. The strainers connected to the Suppression Pool Cleanup (SPCU) System suction penetration piping tee will not be affected by this change.

REASON FOR CHANGE, TEST, OR EXPERIMENT:

This design change will assure that the ECCS at GGNS will meet the regulatory requirements of 10 CFR 50.46 (Ref. 6.2.2) under the conditions postulated in NRC Bulletin 96-03 (Ref. 6.2.8). This design change installs a new, large passive strainer that rests on the floor of the suppression pool and encircles the suppression pool and is connected to each ECCS and RCIC in place of one of the two existing strainers at each suction penetration piping tee. The new strainer is designed to achieve a low approach velocity (~ 0.020 fps) at the surface of the strainer. A low approach velocity will minimize compaction of debris at the strainer surface, thereby allowing greater flow with less head loss through the debris and strainer. Due to the new strainer's large size and the resultant low approach velocity, the available NPSH will exceed the required NPSH needed for ECCS to function in the short-term to maintain peak clad temperature less than 2200"F and to provide long-term core and containment cooling capability. This satisfies the requirements of RG 1.82, Rev. 2 (Ref. 6.2.11), and as a result ensures the ECCS will meet the acceptance criteria of 10CFR50.46**.**

This Safety Evaluation evaluates the effects of the new strainer on the ability of the ECCS to satisfy the requirements of 10 CFR 50.46, and the analytical methods used to evaluate hydrodynamic loads on the strainer. Additionally, this evaluation covers installation issues including UFSAR changes, Technical Specification Bases changes, and provides guidance for evaluation of ECCS operability during installation and after tie-in to the new strainer. Appendix A (starting on page 23) provides a detailed discussion of issues related to the analysis of the new ECCS/RCIC suction strainer. Appendix B (starting on page 71) is a detailed discussion of the effect of this design change on the original evaluation of Humphrey concerns and the Humphrey concerns effect on the design of the new ECCS/RCIC strainer. Appendix C (starting on page 142) provides a detailed evaluation of the change with respect to requirements presented in RG 1.82, Rev. 2 "Water Sources for Long Term Recirculation Cooling Following a Loss of Coolant Accident". Appendix D (starting on page 153) provides a comparison of strainer analysis methodology with that currently described in the UFSAR. LDCR 97-074 (not part of this Safety Evaluation) is a discussion of UFSAR and Technical Specification Bases changes.

SAFETY EVALUATION SUMMARY AND CONCLUSIONS:

The intended purpose for the installation of the new large passive suction strainer is to alleviate the concern that the current suction strainers are marginally sized for the postulated LOCA-generated debris loading. The new strainer has been designed to maintain the approach velocity very low and has significantly more strainer surface area than the existing suction strainers. The new strainer has been designed to withstand postulated seismic, hydrodynamic and other applicable loads and to minimize clogging under postulated operational and post-accident conditions. It will be built using the same code and construction requirements as the original strainers, and has been designed to exceed the functional requirements of the original strainers. Therefore, there will be no adverse effect on the design basis of the ECCS or RCIC systems and their ability to mitigate the consequences of the accidents/events for which they were designed.

The physical separation criterion has been met for each functional group of the ECCS, each division of the containment heat removal system. Additionally, the RCIC system suction is separated from the HPCS suction. The ECCS are divided into three functional groups for mechanical separation considerations outside the drywell to ensure that critical safety functions will be fulfilled under the most limiting conditions involving a single failure in conjunction with the initiating break that results in a LOCA (Refs. 6.3.1.58, 6.1.3.21). The three functional groups (of pumping systems) are:

Low Pressure Core Spray and one Low Pressure Coolant Injection subsystem (Division I)

Two Low Pressure Coolant Injection subsystems (Division II)

High Pressure Core Spray (Division III)

Equipment in each group is independent from the other groups. In addition, the HPCS and RCIC systems are independent from each other to provide additional diversity for high pressure water sources, and to provide single failure protection for the control rod drop accident (Refs. 6.3.1.32, 6.3.1.34, 6.3.1.82).

Materials have been chosen which are qualified for the environment accounting for water chemistry, radiation, and applicable loading. To guard against single failure effects, the new strainer has been divisionalized using physical separator plates between functional groups of the ECCS, and between the RCIC and HPCS systems; and the new strainer has been designed such that there are no credible failure mechanisms that would render the entire strainer inoperable as a result of a single event. If strainer failure occurs in one division the others are unaffected. Additionally, the effects of missiles and high energy line breaks have been evaluated as having no impact on the new strainer.

This change maintains/improves the current design basis 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 cause a greater reliance to be placed on any specific system, structure or component to perform a safety function. A failure of a section of the new strainer would result in the eventual failure of the associated ECCS functional group pump(s) (and the RCIC system with Division 2). These failures (i.e., loss of ECCS or RCIC systems) have been analyzed in the UFSAR. The installation or failure of the new strainer will not increase the probability or consequences of these analyzed failures. No increase of either the expected offsite or the onsite radiation dose would result because of a failure of a section of the new strainer.

This change does not adversely affect the overall ECCS or RCIC systems performance or reliability in a manner that could lead to an accident occurring. 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 have been considered in the new strainer design. The new strainer cannot affect any system interface in a way that could lead to an accident. The new strainer will not result in degradation of safety systems. To the contrary, it is intended to improve the availability of the ECCS and RCIC systems by providing a mechanism to reduce the possibility of system unavailability. Because the new strainer is passive, no operator actions are required, therefore, no increase in the possibility for operator error has been introduced. Additionally, the margin of safety as defined in the bases for the Technical Specifications has not been reduced.

Serial Number: 1999-047-NPE Document Evaluated: ER 96/0403-00-01

BRIEF DESCRIPTION OF CHANGE, TEST, OR EXPERIMENT:

Replace the existing GE supplied EPA breaker units with new breaker units that do not use GE logic cards. The new EPA units will use discrete trip relays to sense any abnormal power quality conditions that exist on the RPS buses. The use of the discrete trip relays will improve the overall reliability of the EPA breaker units. GE's logic cards have a documented history of failure while the discrete relays being used are solid state and have tighter trip point tolerances. This difference will increase EPA reliability and reduce the potential for unnecessary challenges to safety systems. ER 96/0403-00-01 will also replace the RPS Bus A/B Alternate source voltage regulators.

REASON FOR CHANGE, TEST, OR EXPERIMENT:

Multiple half scrams have occurred due to RPS EPA breaker (1C71S003A-H) trips. In each case an EPA logic card malfunction was found to be the cause of the breaker trip. Maintenance history on the EPAs indicates a generic reliability concern due to premature failures of various GE logic card components. The RPS Bus A/B alternate source voltage regulators will be replaced due to an incompatibility found between the existing regulators and the new EPA units during RFO9. The new regulators will provide a "clean", sinusoidal output that will be acceptable for use with the new EPAs.

SAFETY EVALUATION SUMMARY AND CONCLUSIONS:

ER 96/0403-00-01 will replace the existing EPA breakers and the RPS Bus A/B alternate source voltage regulators. The existing EPAs utilize GE logic cards to monitor the Reactor Protection System for undervoltage, overvoltage, and underfreguency conditions. The replacement EPAs will utilize solid state relays instead of the logic cards to sense the abnormal power conditions. The trip devices will actuate and deenergize the undervoltage coil of a molded case circuit breaker housed in the EPA enclosure; with the undervoltage coil deenergized, the molded case breaker will trip open. This feature is consistent with the current design. The setpoint allowable limits for the EPAs are listed in the Technical Specifications and these limits are not being changed per ER 96/0403-00-01. The new EPAs are designed as Class 1 E. Seismic Category I components to ensure that they will perform as required under the required design basis conditions. The replacement of the EPAs by ER 96/0403-00-01 will not alter the ability of the Reactor Protection System to perform its required functions due to the fact that if the RPS power supply system fails, that portion of the distribution system will deenergize and a half scram signal will be created. This is considered a fail-safe design and is not impacted by the replacement of the EPAs. The new EPAs are functionally equivalent to the existing EPAs and are designed to meet the performance requirements of the existing EPAs. The new regulators to be installed for the RPS alternate sources will meet the power requirements of the existing units, however the output voltage will be a "cleaner" waveform than the existing units. This "clean" output will ensure compatibility between the regulators and the EPAs. The proposed change to the EPAs and the RPS alternate source regulators does not create an Unreviewed Safety Question. Changes to UFSAR Section 8.3.1.1.5.2 and Tech Spec Bases Section 3.3.8.2 are required in order to update the description of the EPAs. These changes will be incorporated per Licensing-Document Change Request 97-048 revision 1. The implementation of ER 96/0403-00-01 will enhance the reliability of the EPAs. This modification does not introduce any activity that will adversely impact the safe operation of the plant.

Serial Number: 1999-048-NPE Document Evaluated: ER 97/0939-00-00

BRIEF DESCRIPTION OF CHANGE, TEST, OR EXPERIMENT:

The purpose of this modification and supporting documents is to ensure the temperature limits for the concrete surrounding containment penetrations 83, 87, and 88 are maintained within acceptable limits. This modification (ER 97/0939-00-00) will provide inspection requirements for the braided metal hose to extend the operation of the current configuration of jacket water cooler for penetration 83 for one more cycle and remove the jacket cooler from penetration 87 and replace the jacket water cooler with a fin on penetration 88. Statement 3.10 of the Reactor Water Cleanup SOI will be revised to ensure RWCU does not operate in PRE-PUMP mode with a reactor water temperature above 3500F. As part of this modification, the PSW piping in the Auxiliary Building steam tunnel from the cooler to the penetration will be removed and the hangers abandoned in place. The PSW piping outside the Auxiliary steam tunnel will be cut and capped from the main PSW header piping and abandoned in place. The abandoned piping will maintain its seismic II/I capability. The fin assembly is Seismic II/I qualified and does not effect the structural integrity of the penetration. These modifications will ensure that the concrete temperature will be maintained within acceptable limits.

REASON FOR CHANGE, TEST, OR EXPERIMENT:

In July 1997, GGNS issued a Condition Report to document Plant Service Water (PSW)(P44) system flow rates below design limits for jacket coolers on containment penetrations 83 (RWCU Return Line), 87 (RWCU Combined Supply Line), and 88 (RWCU Pump Discharge Line). This reduction in flow was attributed to fouling of the cooler and piping and led to localized heating of the containment wall surrounding the penetrations. In light of this, cooling water supply to Penetration 83 cooler was changed to Plant Chilled Water. Penetration 88 cooling water supply was not changed but it was evaluated for zero flow condition at maximum temperatures during RWCU Pre-Pump operation. Penetration 87 did not require any cooling water flow to maintain containment wall temperature.

SAFETY EVALUATION SUMMARY AND CONCLUSIONS:

This modification will remove the jacket water cooler for penetration 87 and replace the cooler on penetration 88 with a fin. Calculation MC-Q1G33-98015 Rev. 1 determined that penetration 87 does not require any cooling water and calculation MCQlG33-99002 Rev. 0 evaluated penetration 88 with a fin installed to maintain the temperature of the concrete within acceptable limits. Section 6.2.1.1.10 of the SAR will be updated based on the current requirements for penetration cooling for these high temperature lines modified by this ER. In conclusion, the removal of the jacket water cooler from 87, replacing the cooler on penetration 88 with a fin, and continuing to operate another cycle with braided metal hose to penetration 83 will not increase the probability of occurrence or increase the consequences of an accident evaluated in the SAR nor will the removal of the aforementioned jacket water coolers increase the probability of occurrence or increase the consequences of a failure of equipment important to safety. Since the strength of concrete in the CTMT wall surrounding the penetrations are within acceptable limits, no mechanism exists to create the possibility for an accident or a malfunction of equipment important to safety of a different type than previously evaluated in the SAR. This modification will not affect the RWCU system operation or decrease the performance capability of the containment concrete around the penetration.

Serial Number: 1999-049-NPE Document Evaluated: ER 97/0546-00-02

BRIEF DESCRIPTION OF CHANGE, TEST, OR EXPERIMENT:

This Engineering Response approves the use of an alternate battery for the Diesel Driven Fire Pumps NSP64C003A/B identifies surveillance procedure requiring updating and initiates LDC for updating TRM SR 6.2.2.7.

REASON FOR CHANGE, TEST, OR EXPERIMENT:

GGCR19980113 identified that surveillance procedure 06-EL-SP64-Q-0001 did not get properly updated when an alternate battery was approved via revisions 0 and 1 of Engineering Response 97/0546. These two revisions failed to properly identify this surveillance procedure as requiring updating and also failed to change Appendix 16A of the UFSAR (TRM SR 6.2.2.7), which specifies the specific gravity and temperature for correction factor for the original battery.

This ER evaluates and approves an alternate battery, since the current battery utilized can not be ordered directly from the factory and, as experienced by Electrical Maintenance, has poor capacity for re-charging. This replacement battery has been determined to be superior to the original battery supplied with the components and posses adequate ampacity ratings for these applications. This ER also permits adjustment to the associated battery racks to ensure proper restraint and housing requirements are maintained.

SAFETY EVALUATION SUMMARY AND CONCLUSIONS:

The use of this alternate battery type will not affect the GGNS Technical Specifications. Nor will it create an unreviewed safety question or reduce any margin of safety. These batteries are discussed in TRM SR 6.2.2.7, which will require revision. The use of the alternate battery will not create an electrical separation concern since all requirements of Reg. Guide 1.75 are maintained. The replacement battery will perform the same function in the same manner as the original and will not create any new interfaces with other equipment. Nor does the use of this replacement create any new hazards, such as hydrogen generation, that is not currently present with the use of the original or any other type of battery. GGNS's Fire Protection Program will not be affected by the use of this approved alternate battery. This equipment is located in a Non-Seismic building (Fire Water Pump House) and does not adversely affect any safety related or non-safety related equipment. The use of this alternate battery will not increase the probability of occurrence or consequences of an accident previously evaluated in the SAR. The use of this alternate battery will not increase the probability of occurrence or consequences of a malfunction of equipment important to safety previously evaluated in the SAR. The use of this alternate battery will not create the possibility for an accident or malfunction of a different type than any previously evaluated in the SAR. No unreviewed safety question will result from the use of this alternate battery.

This Safety Evaluation is performed primarily as a result and in support of the required revision to the TRM. As previously stated, TRM SR 6.2.2.7 will require revision to remove the specific gravity and temperature reference from the text. The intent of this TRM SR of ensuring the battery has adequate capacity to perform its function and the frequency at which it is performed will remain unchanged. These characteristics will be maintained in the applicable surveillance procedure. Surveillance procedure 06-EL-SP64-Q-0001 will require updating to utilize/specify characteristics of the replacement battery.

Serial Number: 1999-050-NPE Document Evaluated: ER 98/0615-00-01

BRIEF DESCRIPTION OF CHANGE, TEST, OR EXPERIMENT:

Revision 1 to this safety evaluation is being issued to reflect changes that resulted from the issuance of Revision1I to ER 1998-0615. Specifically, the only changes in the revised ER were to details of the steel angle on each end of the fire wall assembly where the ceiling I-Beam is connected to the top of the firewall. These changes necessitated drawing changes and these drawings were referenced in various documents identified in Revision 0 to this SE. The only change to this safety evaluation are the references to the following documents: 1) UFSAR Change Request No. 1998-0091. Rev. 1, 2) Fire Hazards Analysis Revision Request No. 98/0003, and 3) Fire Protection Evaluation No. 98-0003. Rev. 1.

ER 1998-0615-01 documents the acceptability of a non-standard fire barrier design utilized as part of the fire wall assembly separating Fire Zone 0C702 (Upper Cable Spreading Room, Control Building El. 189'-0") and Fire Zone 0C712 (HVAC Room, Control Building El. I 89'-0"). In addition, openings through this nonstandard fire barrier configuration are being sealed with steel plate or steel angle and 3-hour rated structural steel fireproofing is then being applied to both sides of the steel.

REASON FOR CHANGE, TEST, OR EXPERIMENT:

The fire protection program and UESAR describe the wall separating the Upper Cable Spreading Room (Fire Zone 0C702, Fire Area 47) from the HVAC Room (Fire Zone 0C712, Fire Area 47) on elevation 189'-0" of the Control Building as a 2-hour rated fire barrier. The construction of the portion of this wall above the 200'-7" elevation utilizes a nonstandard fire barrier configuration that does not have a quantifiable fire resistance rating. In accordance with Generic Letter 86-10, an evaluation of this barrier has been performed and documented in Fire Protection Evaluation No. 98-0003. Rev. 1. to determine if the existing barrier is adequate for the hazards in the area. This change documents that evaluation and makes necessary Fire Protection Program changes to reflect the non-standard fire barrier configuration.

In addition, gaps were left between the steel angle (installed between the bottom of the I-Beam and the top of the concrete wall) and the adjoining fire barriers on each side of the non-standard fire barrier configuration. This construction created a through hole in the non-standard fire barrier assembly. This ER installs a steel plate and/or a steel angle at these locations to seal the through hole and installs 3-hour structural steel fire proofing material on each side of the repaired area.

SAFETY EVALUATION SUMMARY AND CONCLUSIONS:

License Condition 2.C.41 allows GGNS to make changes to the approved Fire Protection Program through the 50.59 process if those changes do not adversely affect the ability to achieve and maintain safe shutdown in the event of a fire. Therefore, from the fire protection standpoint the basis for evaluation is "no adverse effect on the ability to achieve and maintain safe shutdown in the event of a fire." Repair of the holes through this barrier and rework of the fire proofing material is being done in accordance with approved design as presently identified in the SAR. Generic Letter 86-10. Enclosure No. 1, Interpretation No. 4 states:

"Where fire area boundaries are not wall-to-wall, floor-to-ceiling boundaries with all penetrations sealed to the fire rating required of the boundaries, licensees must perform an evaluation to assess the adequacy of fire boundaries in their plants to determine if the boundaries will withstand the hazards associated with the area". As documented in Fire Protection Evaluation No. 98/0003, Rev. 1, the non-standard fire barrier configuration separating Fire Zone 0C702 and 0C7 12 is capable of withstanding the hazards of either area. Therefore, this configuration is an acceptable fire barrier. Thus, the ability to achieve and maintain safe shutdown conditions in the event of a fire, as presently analyzed in the UFSAR, has not been adversely affected.

Therefore, this change will not increase the probability or consequences of accidents or malfunction of equipment important to safety previously evaluated in the SAR. Also, this change 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. “Fire Rated Assemblies” are not addressed by Technical Specifications (TS). “Fire Rated Assemblies” are addressed in the Technical Requirements Manual (TRM) Section 6.2.8. TRM Section 6.2.8 covers the fire barrier addressed in this change; however, the change only demonstrates the adequacy of the non-standard fire barrier configuration. No fire barriers are being added or deleted; therefore, there will be no change to TS or TRM or the Bases for any TS or TRM.

Serial Number: 1999-051-NPE Document Evaluated: ER 96/0499-00-01

BRIEF DESCRIPTION OF CHANGE, TEST, OR EXPERIMENT:

ER 96/0499-00-01 provides the required design information to replace the existing Turbine Building Cooling Water Heat Exchanger Outlet Temperature air operated butterfly control valve (N1P44F513) with an air operated ball control valve. The replacement valve has been obtained from stock and has been refurbished by the manufacturer in accordance with the valve specification and Design Engineering instructions. The modification includes the addition of a position transmitter (P44N094) to provide a signal to a BOP computer point. The piping will be modified to allow for the installation of the new valve which is longer than the existing valve. The ball valve installed per this ER will have its actuator orientated in a different direction than the existing valve, therefore the existing air tubing will be field routed and supported from its present location to the new location of the actuator. A larger volume tank (accumulator) will be installed to maintain the fail-open capability of the larger air actuator. Electrical supply cable and instrument signal cable for the position transmitter will be installed. This modification will retain the existing valve nNumber (N1P44F513); however, the old valve drawing will be voided, vendor manuals applicable to the old valve will be amended and the Component Database will be changed to reflect the data for the new valve. Valve design specification 9645-J-607.0 will be revised via Specification Change Notice (SCN) 99/000lA. The system Piping and Instrumentation Diagram will be changed to reflect the modification. This will affect the associated UFSAR Figure (9.2-022).

REASON FOR CHANGE, TEST, OR EXPERIMENT:

Turbine Building Cooling Water (TBCW) temperature is maintained by regulating the flow of Plant Service Water (PSW) through the tubeside of the TBCW heat exchangers. The PSW flow is regulated by the TBCW Heat Exchanger Outlet Temperature Control Valve, which is a 16 inch pneumatic Fisher butterfly valve. A nNumber of flow control problems and actuator part failures have been reported for the TBCW Heat Exchanger Outlet Temperature Control Valve since 1992. As a response to Corrective Action (CA)-003 of Condition Report CR-GGN-1997-0620-00, Design Engineering has determined that a ball valve will preclude the problems associated with the existing temperature control valve.

SAFETY EVALUATION SUMMARY AND CONCLUSIONS:

The TBCW Heat Exchanger Outlet Temperature Control Valve is a component in the Plant Service Water System which is the normal service water system providing cooling for various plant systems including the component cooling water and turbine building cooling water heat exchangers, plant chillers, drywell chillers, mechanical vacuum pump water jacket coolers, Engineered Safety Features electrical switchgear room coolers, control room air conditioners, Steam Jet Air Ejector intercondensers, a containment leak rate test system compressor aftercooler and alternate decay heat removal heat exchangers and air conditioner. Additionally, the Plant Service Water system provides lube water for the circulating water pumps cutless rubber bearings. According to UFSAR 9.2.8.3, the Plant Service Water system has no safety design basis as defined in Section 3.2 of the UFSAR. Upon accident initiation, the Standby Service Water system replaces the Plant Service Water system in providing cooling water to plant components and systems important to safety. The TBCW Heat Exchanger Outlet Temperature Control Valve does not perform an Alternate Decay Heat Removal System function as delineated in UFSAR Table 3.2-1 LIV. 1. Therefore, the modification authorized by ER 96/0499-00-01 does not require a change to the GGNS Unit 1 Technical Specifications.

The TBCW Heat Exchanger Outlet Temperature Control Valve and its associated components do not perform any function that would involve it as a precursor or initiator of any accident discussed in the UFSAR; therefore, the modification authorized in ER 96/0499-00-01 does not increase the probability of occurrence of an accident previously evaluated in the SAR

The valve change does not alter the flow or operation of PSW or any other system. Valve operation is unchanged. The PSW system is not credited in the accident analyses; therefore, there is no impact to radiological conditions from this change. Therefore, the modification authorized in ER 96/0499-00-01 does not increase the consequences of an accident previously evaluated in the SAR.

The Plant Service Water system has no safety design basis as defined in Section 3.2 of the UFSAR The TBCW Heat Exchanger Outlet Temperature Control Valve interfaces with the TBCW Heat Exchangers. UFSAR 9.2.9.3 describes the TBCW system as having no safety related function as discussed in Section 3.2 of the UFSAR and further states that failure of the system will not compromise any safety related system or component and will not prevent safe reactor shutdown. The TBCW Heat Exchanger Outlet Temperature Control Valve is not located adjacent to or above any safety related SSC (i.e., there are no seismic II/I concerns). Conduit and supports added by the modification will also pose no II/I concerns. Signal cable from the new position transmitter will be connected in a non-safety related fuse and relay panel (N1H22P175*)*. Because it has no interface with equipment important to safety, the modification authorized in ER 96/0499-00-01 does not increase the probability of occurrence of a malfunction of equipment important to safety previously evaluated in the SAR.

The valve change does not alter the flow or operation of PSW or any other system. Valve operation is unchanged. The PSW system is not credited in the accident analyses; therefore, there is no impact to radiological conditions from this change. Therefore, the modification authorized in ER 96/0499-00-01 does not increase the consequences of a malfunction of equipment important to safety previously evaluated in the SAR.

The modification authorized in ER 96/499-00-01 complies with, or is reconciled with, the Codes, Specifications and Standards applicable to the original installation of the TBCW Heat Exchanger Outlet Temperature Control Valve. These include the valve/actuator assembly, piping installation, welding/Non-Destructive Examination and instrument air supply and volume tank installation. Electrical components, cable/conduit and conduit supports to be added in the modification are designed and installed to approved Codes, Specifications and Standards. Testing/calibration and inspection for the modification will be in accordance with approved Plant procedures. It is concluded that the replacement valve will perform its system function in a manner that will preclude the previously identified problems (i.e., flow control, cavitation erosion, parts failure) with the TBCW Heat Exchanger Outlet Temperature Control Valve. For these reasons, The modification authorized in ER 96/0499-00-01 does not create the possibility for an accident of a different type than any previously evaluated in the SAR.

There are no Technical Specifications that govern the TBCW Heat Exchanger Outlet Temperature Control Valve nor does it appear in any TRM Tables; thus, there is no margin of safety relating to the TBCW Heat Exchanger Outlet Temperature Control Valve established in the bases for the Technical Specifications. For this reason, the modification authorized in ER 96/0499-00-01 does not reduce the margin of safety as defined in the bases for any Technical Specification.

Serial Number: 1999-052-NPE Document Evaluated: ER 97/0103-00-00

BRIEF DESCRIPTION OF CHANGE, TEST, OR EXPERIMENT:

The proposed change will increase the analytical stroke time limit associated with the High Pressure Core Spray injection valve (Q1E22F004) from the current 16 seconds to 21 seconds. As a result of this change, the High Pressure Core Spray system initiation, which includes Diesel Generator start time, will be extended from 27 seconds to 32 seconds. This evaluation does not consider any physical modifications to any of the High Pressure Core Spray equipment.

REASON FOR CHANGE, TEST, OR EXPERIMENT:

The reason for the increase in valve stroke time limit is to recover the margin to the analytical stroke time limit that existed prior to the implementation of two system modifications which provided additional capability and reliability for the High Pressure Core Spray system. ER 96-0002 enhanced the operating thrust margins for the injection valve by replacing the motor pinion gear set; however, the change increased the nominal valve stroke time. ER 96-0078 added a half-second time delay (one second maximum) to the Level 2 initiation signal to prevent spurious HPCS initiations on fast noise transients. To accommodate this delay without increasing the overall HPCS initiation time, the ER reallocated one second from the injection valve stroke time. The injection valve stroke time still meets the current 16 second analytical stroke time limit; however, additional margin is desired.

50.59 EVALUATION SUMMARY AND CONCLUSIONS:

The proposed delay of the HPCS injection valve stroke time limit does not involve any physical modifications to the valve or any other component of the HPCS system. The overall function of the valve and HPCS system is not affected by this change. Therefore, the probability of an accident or malfunction of equipment previously analyzed in the SAR is not increased. Additionally, a new accident or malfunction not previously analyzed in the SAR is not created. A complete ECCS performance (LOCA) analysis including the 32 second HPCS initiation time has been performed using NRC approved methodology in accordance with required analyses and assumptions given in 10CFR50. The results of this analysis are well within the acceptance criteria given in 10CFR50.46. As such, the consequences of accidents and malfunctions previously evaluated in the SAR are not increased. Further, the margin of safety as defined in the basis for any Technical Specification is not reduced by this change. Therefore, an unreviewed safety question does not exist.

Serial Number: 1999-053-NPE Document Evaluated: ER 99/0424-00-00

BRIEF DESCRIPTION OF CHANGE, TEST, OR EXPERIMENT:

ER 99/0424-00 will allow the use of the Polar Crane 125 Ton Main Hoist to install and remove the Portable Radiation Shield (Cattle Chute) in lieu of the 35 Ton Auxiliary Hoist during RF1O. UFSAR Section 9.1.4.2.10.2.3.3 states the Portable Radiation Shield is lowered into place by the auxiliary hook of the containment polar crane.

REASON FOR CHANGE, TEST, OR EXPERIMENT:

The Polar Crane 35 Ton Auxiliary Hoist is currently inoperable.

50.59 EVALUATION SUMMARY AND CONCLUSIONS:

The use of the 125 Ton main hoist in lieu of the 35 Ton auxiliary hoist for movement of the portable radiation shield during RF10 is acceptable. The main hoist is rated for a higher capacity than the auxiliary hoist and the original spreader bar along with plant approved slings/shackles for the required design loads will be utilized. The only load drop accident previously evaluated in the UFSAR in Section 9.1.4.2.2.1 is for the portable radiation shield dropping into the open vessel with the dryer and separator installed and use of the main hoist hook for this evaluation will not change this drop analysis. The safe load path of the transport will be slightly changed to allow the portable radiation shield to move west away from the reactor vessel to miss an existing valve prior to being moved north to its final location. The actual safe load path to transport the portable radiation shield from its storage location to the gate does not place the portable radiation shield directly over the open vessel. No new consequences will be created in the potential drop due to this slight change in the load path. At no time will it travel over equipment more important to safety than previously evaluated. No increase in radiation exposure will occur with the potential drop of the portable radiation shield by the main hoist hook compared to the auxiliary hoist hook. The lift will be made with approved plant procedures by qualified operators and will use the originally designed spreader beams and sling/shackle, which meet plant procedures and NUREG 0612. The possibility and consequences of the accident previously evaluated (i.e. load drop) will not be increased, the possibility and consequences of malfunction of equipment important to safety previously evaluated (i.e. from a load drop) will not be increased and the possibility or consequences of accident different than previously evaluated (i.e. load drop) will not be created.

Serial Number: 1999-054-NPE Document Evaluated: ER 1999-0418-00-00

BRIEF DESCRIPTION OF CHANGE, TEST, OR EXPERIMENT:

This safety evaluation reviews the one time use of the Containment Hatchway Crane for replacing the Aux. Hoist Motor on the Containment Polar Crane which requires the crane to be operated outside the current limits of the applicable procedures and the UFSAR. The motor weighs 1650# which classifies it as a heavy load in accordance with GGNS to NUREG 0612. ER 99/0418 permits a small increase in crane rated capacity and waives load height restrictions over the Drywell Head storage slab at elevation 208, Containment, based on a calculation CC-Q1M31-86024, Supplement 3 and reconciliation of crane standard ANSI B30.5 with ANSI B30.2 and CMAA-70. These actions ensure that the change meets GGNS commitments to NUREG 0612 for handling heavy loads.

REASON FOR CHANGE, TEST, OR EXPERIMENT:

The only other hoisting system which could be used to replace the motor is a manual 3 ton chainfall mounted on the Containment dome. However, use of this chainfall requires multiple handling steps and a lengthy preparation process that introduces greater inherent process risk. With minor uprate of the Containment Hatchway Crane load rating and an evaluation of its hoisting system this operation could be performed in a single lift.

50.59 EVALUATION SUMMARY AND CONCLUSIONS:

The crane structural integrity under the increased load rating is assured as demonstrated in calculation CC-Q1M31-86024, Supplement 3. The hoisting system is designed and fabricated to meet the requirements of ANSI B30.5 which contains requirements commensurate with standards CMAA-70 and ANSI B30.2 referenced in NUREG-0612. Testing and maintenance is performed under GGNS procedures which ensure compliance with NUREG-0612. The ER contains safe load paths and demonstrates compliance with each of the seven requirements of section 5.1.1 of NUREG-0612. Therefore, a load drop need not be postulated for this lift and the load/height restrictions of Procedure 07-S-05-300 need not be applied for this lift. In addition, the revised load chart in the ER may be used without jeopardizing crane structural integrity.

Serial Number: 1999-055-NPE Document Evaluated: ER 1999-0424-00-01

BRIEF DESCRIPTION OF CHANGE, TEST, OR EXPERIMENT:

ER 99/0424-00-01 will allow the use of the Polar Crane 125 Ton Main Hoist to install and remove the Portable Radiation Shield (Cattle Chute) in lieu of the 35 Ton Auxiliary Hoist during RF10. UFSAR Section 9.1.4.2.10.2.3.3 states the Portable Radiation Shield is lowered into place by the auxiliary hook of the containment polar crane.

REASON FOR CHANGE, TEST, OR EXPERIMENT:

The Polar Crane 35 Ton Auxiliary Hoist is currently inoperable.

EVALUATION SUMMARY AND CONCLUSIONS:

The use of the 125 Ton main hoist in lieu of the 35 Ton auxiliary hoist for movement of the portable radiation shield during RFl0 is acceptable. The main hoist is rated for a higher capacity than the auxiliary hoist and the original spreader bar along with plant approved slings/shackles for the required design loads will be utilized. The only load drop accident previously evaluated in the UFSAR in Section 9.1.4.2.2.1 is for the portable radiation shield dropping into the vessel with the dryer and separator installed and use of the main hoist hook for this evaluation will not change this drop analysis. Due to the limited height the cattle chute will be raised several interferences necessitate the safe load path of the transport to be altered slightly. The cattle chute will be required to move either west or east of its storage location prior to being moved to the North to its final location. Moving the cattle chute to the East has the potential that the cattle chute will be moved over the reactor vessel with the separator install resulting in a Safety Class 3B move (Ref. UFSAR Section 9.1.4.2.2.5). Moving west does not place the portable radiation shield directly over the vessel. No new consequences beyond those considered in UFSAR Section 9.1.4.2.2.1. are created since the cattle chute weight is unchanged. At no time will it travel over equipment more important to safety than previously evaluated. The lift will be made with approved plant procedures by qualified operators and will use the originally designed spreader beams and sling/shackle, which meet plant procedures and NUREG 0612. Therefore, no load drop is required to be postulated . The possibility and consequences of the accident previously evaluated (i.e. load drop) will not be increased, the possibility and consequences of malfunction of equipment important to safety previously evaluated (i.e. from a load drop) will not be increased and the possibility or consequences of accident different than previously evaluated (i. e. load drop) will not be created.

Serial Number: 1999-056-NPE Document Evaluated: ER 1999-0391-00-00

BRIEF DESCRIPTION OF CHANGE, TEST, OR EXPERIMENT:

The change is to add a branch line off of 24" JBD-782, which will include a 12" normally closed valve (N71F416) and a blind flange. 24" JBD-782 supplies makeup water (PSW) for the circulating water (N71) system. The new 12" branch off will be in the N71 piping that supplies makeup water (PSW) to the circulating water pump house. However the use of this line, other than to supply PSW water for the SSW basin is not evaluated herein.

REASON FOR CHANGE, TEST, OR EXPERIMENT:

The new 12" branch line will be used to connect temporary piping during refueling/extended plant outages when needed to supply additional (PSW) water to the SSW basins to allow quick refill of the drained basins during refueling/extended plant outages. However the use of this line, other than to supply PSW water for the SSW basin is not evaluated herein.

EVALUATION SUMMARY AND CONCLUSIONS:

A makeup water system is provided to replace the circulating water losses due to evaporation, blowdown, and drift. Makeup water for the circulating water system is taken from the plant service water system. Approximately 21,500 gpm of makeup is required. The Circulating Water System (N71) and or the Plant Service Water (P44) system serves no safety function. Systems analysis has shown that failure of the Circulating Water System (N71) or the Plant Service Water (P44) 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 10.2-003 and 10.4-005. However, installation of valve N71F416 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 12" branch connection may be used to supply make-up water during all operational conditions with sufficient Plant Service Water capacity available to support the plant condition that exist during the period of use. Use of the 12" branch connection in this manner is in agreement with SOI 04-1-01-P44-1 Section 4.1.2.t. This section states "Start/Stop Radial Well pumps to maintain header pressure at approximately 90 psig as plant loads change". UFSAR Section 9.2.10.2 states "During normal operation, as many wells and pumps as required will be operating to meet plant demand." However the use of this line, other than to supply PSW water for the SSW basin is not evaluated herein.

The proposed change is located in the Yard at the circulation water pump house and will not affect or impact the plant's radiological effluents. The area behind the circulating water pump house is within the tie back wall and therefore is structural backfill. The function of the impermeable membrane and structural backfill is discussed in GGNS UFSAR Section 2.5.4.6 and 2.5.4.5.5 but, the impermeable membrane and structural backfill is not governed by any Technical Specifications. The proposed work activity of connecting a 12" pipe to the existing 24" JBD-782 line will require that the adjacent area be excavated. However, after completion of work activities, the area will be restored to the original design requirements. See UFSAR Section 2.4.13.5 for a discussion on ground water levels.

The temporary piping that will be attached to the new 12" branch line will be raised above grade elevation using blocks to eliminate PMP concerns during the use of the temporary piping. The anticipated size of the excavation will not affect local ground water level in the area in the event of a PMP type rainfall. The FSAR does not consider PMP to have an appreciable affect on site ground water levels. The clay cap functions to limit surface water filtration as discussed in UFSAR section 2.5.4.5.5.

Due to the limited scope of excavation, this work activity will not adversely affect the PMP evaluation. For PMP requirements see SER2.4.4 and TRM 6.7.5.

The proposed change to the N71 system will have no adverse environmental impacts. After reviewing the proposed change, it has been concluded that installation of the valve does not represent an Unreviewed Safety Question and will have no adverse affects on the environment. The GGNS Technical Specifications do not address the Circulating Water System (N71). Thus the proposed change will not result in the need to change or revise the GGNS Technical Specifications or the Technical Requirements Manual for the Circulating Water System (N71). This change does not adversely affect the overall performance or reliability of the Plant Service Water (P44) system 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 12” branch line cannot affect any system interface in a way that could lead to an accident. The new 12” branch line will not result in degradation of safety systems. Additionally, the margin of safety as defined in the bases for the Technical Specifications has not been reduced.

Serial Number: 1999-057-NPE Document Evaluated: ER 1996-0485-00-00

BRIEF DESCRIPTION OF CHANGE, TEST, OR EXPERIMENT:

ER 96/0485-00-00 will provide the justification and documentation to replace the Bailey 771 Narrow Roll recorders (1C51R603A/B/C/D) used in the IRM/APRM monitoring application with the Westronics Series 1200B miniature hybrid recorder. The Westronics recorder is a direct replacement for the Bailey 771. It slides directly into the Bailey 762 Rack Mounted Shelf and plugs directly into the existing Bailey wiring connector. The new recorders, unlike the existing Bailey 771, are programmable recorders.

The IRM/APRM recorders are non-safety related and are used as Reg. Guide 1.97 instruments even though they do not meet all the requirements for Reg. Guide 1.97 instruments. GGNS's neutron monitoring system does not have to meet Reg. Guide 1.97 requirements per NEDO-31558-A. This NEDO document was presented by the BWROG to the NRC and the NRC agreed (SER to Docket No. 50-416) that GGNS's neutron monitoring system meets the requirements of NEDO-31558-A, which provides an alternative to the guidance in Reg. Guide 1.97.

UFSAR Table 7.5-2 sheet 2 of 18, Post-Accident Monitoring Instrumentation, and sheet 7 of 18, NOTES, will need to be corrected based on the acceptance of NEDO-31588-A by NRC. This correction to the table should have been done per LDC 96/133. The change to the table will include note 8 under the column labeled "GGNS TYPE/Cat" for the measured variable row "Neutron Flux". The "QA" column for Neutron Flux will also be change from "YES" to "NO". Sheet 7 of 18 will be changed to show the statement for NOTE 9 moved to NOTE 8 and have NOTE 9 labeled DELETED. This change will be done per LDC# 99/066 with the close out of ER 96/0485-00-00.

REASON FOR CHANGE, TEST, OR EXPERIMENT:

The existing Bailey 771 Narrow Roll recorders are no longer manufactured by the Bailey company nor are the spare parts. This situation has led the Bailey 771s to become obsolete. GGNS has nearly exhausted all available spare recorders and parts. Because of the age of the recorders, the frequency of maintenance on the recorders has increased causing GGNS to nearly exhaust all available spare recorders and spare parts in stock. Westronics is currently the only manufacture of a direct replacement for the Bailey 771. Being a direct replacement, the Westronics Series 1200B recorder allows the implementation of this package with no control room panel modifications required.

EVALUATION SUMMARY AND CONCLUSIONS:

This safety evaluation of ER 1996/0485-00-00 concludes that replacing the existing Bailey 771 recorders with Westronics Series 1200B recorders for the 1C51R603A/B/C/D (IRM/APRM) application and changing the UFSAR table 7.5-2 to reflect compliance with NEDO-31558-A does not involve an unreviewed safety question. The IRM/APRM recorders are not mentioned in the Technical Specifications since there are no setpoints or allowable values associated with the recorders. The upscale and down scale trip values provided by the neutron monitoring system to the RPS are mentioned in the Technical Specifications but will not be affected by the replacement of the recorders. The IRM/APRM recorders are used to monitor and record neutron levels in the reactor from shutdown to 125 percent power. The recorders are non-safety related instruments and can be used for post accident monitoring even though they are not required per the NRC's safety evaluation in Docket No. 50-416 related to Amendment 112 of GGNS's Operating License No. NPF29. There are no SAR documents that evaluate accidents based on the failure or malfunction of the neutron monitoring recorders. The recorders will neither cause nor prevent an accident since they only function to provide indication of reactor power level nor will any new release paths be created. Therefore, the replacement of the Bailey 771 recorders with Westronics Series 1200B recorders for the IRM/APRM application will not increase the probability or occurrence or consequences of an accident previously evaluated in the SAR.

The neutron monitoring system (the in-core detectors and circuitry) is a safety related system but the IRM/APRM recorders are non-safety related. The detectors and circuitry provide a signal to the recorders, the RC&IS, and the RPS trip circuitry that represents the reactor power level. The recorders only function is to record the reactor power level. Failure of the recorders will not effect the signal to the RC&IS or the RPS and therefore will not inhibit a reactor trip/scram. The new recorders will increase the loading on the existing power supplies but will not exceed the output rating of the power supplies as analyzed in attachment 8 or ER 96/0485. The new recorders also weigh approximately 0.5 lbs more than the existing recorders. Per the Seismic Qualification Assessment Disposition (attachment 3 pg. 2 of 4) of ER 96/0485, the additional 2 lbs. of weight to the 1H13P680 panel will not adversly affect the seismic qualification of the panel. Therefore, the replacement of the IRM/APRM recorders will not increase the probability of occurrence or consequence of a malfunction of equipment important to safety previously evaluated in the SAR.

The replacement IRM/APRM recorders will function to record and provide reactor power level indication just as the existing recorders function. There will be no wiring or panel modifications involved with the implementation of this ER. Failure of the recorders will not cause any type of accident and will not prevent the RPS from initiating a scram signal. Therefore, the implementation of this package will not create the possibility for an accident of a different type than any previously evaluated in the SAR.

The IRM/APRM recorders do not provide signals to any other system or component. A signal, from each channel, is received from the neutron monitors (IRM's and APRM's) and provided to the recorders. This signal represents the reactor power level. Failure of the recorders will not inhibit any design function of the neutron monitoring system. The new recorders function will be the same as the existing recorders. Therefore, the replacement of the IRM/APRM recorders will not create the possibility for a malfunction of equipment important to safety of a different type than any previously evaluated in the SAR.

The low and high reactor power level setpoints, as described in the Tech. Specs., are provided by the neutron monitoring system to the RPS. The IRM/APRM recorders are used for recording and indication only. The recorders receive a signal from the neutron monitoring system that represents the reactor power level. The recorders are not used for any type of trip function and do not interface with the signal from the neutron monitoring system and the RPS. Therefore, the margin of safety as defined in the basis for any Tech. Spec. will not be reduced by the replacement of the IRM/APRM recorders.

Serial Number: 1999-058-NPE Document Evaluated: ER 1999-0066-00-01

BRIEF DESCRIPTION OF CHANGE, TEST, OR EXPERIMENT:

ER 99/0066-00-01 allows for the replacement of the ¾”-HBC-188 carbon steel open end pipe stubs at 20”-HBC-171 recirculation line in SSW Basin “A” with new pipe nipples (HBC) and threaded caps (HBD). The open ended pipe configuration was originally approved per CN 99/0048 to ER 99/0066-00-00. The ¾”-HBC-188 instrument sensing lines in SSW Basin “B” were cut, threaded and capped per ER 99/0066-00-00 interim repair instructions. This ER Revision provides for permanent acceptance of the cut and capped sensing lines in both SSW Basins. The piping downstream of the capped stubs through root valves Q1P41FX223 & 224 in SSW Basin “A” will be removed during RFO10. The piping downstream of the capped stubs through root valves Q1P41FX225 & 226 in SSW Basin “B” will be removed before startup from RFO11. The pipe supports for the removed piping are to remain in place. The tubing downstream of the root valves in both SSW Basin Pump Rooms will be capped allowing Flow Indicators SP41R009A & B to remain for future use, if desired. Elimination of the flow indicators will require that the throttle valves QSP41F002A & B be positioned based on pressure rather than flow.

REASON FOR CHANGE, TEST, OR EXPERIMENT:

CR-GGN-1999-0218 documented failure (i.e. leakage) of the submerged high and low pressure sensing lines for flow element SP41N081B in SSW Basin “B”. CR-GGN-1999-0329 documented pipe wall thinning of the pressure sensing lines for flow element SP41N081A in SSW Basin “A”. The repair to the pressure sensing lines in SSW Basin “A” is necessary to restore and maintain recirculation flow discharge at the original single discharge point at the end of lines 20”-HBC-171. The 3/4” HBC-188 sensing lines were designed to permit periodic surveillance testing of the Standby Service Water (SSW) system pumps QIP41C001A and Q1P41C001B and to support positioning of manual globe valves QSP41F002A & B (SSW pump minimum flow protection throttle valves). The differential pressure sensed across flow elements QSP41N081A and QSP41N081B is measured by local flow indicators SP41R009A and SP41R009B, respectively. Administrative controls (not evaluated herein) will be required to make temporary connections between the differential pressure taps across flow elements QSP41N081A and QSP41N081B and the local flow indicators SP41R009A and SP41ROO9B, respectively, if desired in the future.

EVALUATION SUMMARY AND CONCLUSIONS:

Removing SP41-F1-R009A & B from service will not impact plant safety. This repair does not degrade, below the current design basis, the performance of a safety system assumed to function in the accident analyses and does not decrease the reliability of safety systems assumed to function in the accident analyses. The change does not cause a greater reliance to be placed on any specific system, structure or component to perform a safety function. The repair will restore and maintain recirculation flow discharge at the original single discharge point at the end of lines 20-HBC-171. Minor leakage past the 3/4” ANSI B31.1 carbon steel non-safety related threaded pipe caps will have no effect on the functionality of the SSW system or UHS since any resultant leakage does not represent an UHS inventory loss, additional UHS heat load, or significant system flow diversion path. For this reason, the use of non-safety related pipe caps is acceptable. The primary function of the out of service pressure sensing lines is to support surveillance testing of the SSW pumps Q1P41C001A and B. An alternate method to perform surveillance testing without input from Flow Indicators QSP41R009A & B has been developed as documented in CR-GGN-1999-0218, CA-002. Surveillance Procedures 06-OP-1P41-Q-0004 & 0005 have been revised accordingly. A secondary function of the out of service pressure sensing lines is to support positioning of manual globe valves QSP41F002A and QSP41F002B (SSW pump minimum flow protection throttle valves). P&SE flow balance procedures 17-S-06-22 & 23 will require revision to reflect a new throttle valve positioning process. Attachments IA and lB of SOI 04-1-01-P41-1 will require revision to reflect changes in the throttled range for valves1QSP41F002A and QSP41F002B. In addition to support of surveillance testing, the function of the basin recirculation line 20”-HBC-171 is to provide a flow path

for SSW pump minimum flow protection. The cutting and capping of the pressure sensing lines during any and all postulated events would not prevent the basin recirculation line 20”-HBC-171 from performing it’s minimum flow protection function. Furthermore, UHS basin inventory losses/leakage will not occur as a result of the repair based upon the physical location of the pressure sensing lines (i.e. leakage would be contained by the UHS basin). The repair of the 3/4“-HBC-188 piping will continue to satisfy the ASME Code, Section III, Class 3, Seismic Category 1 support span requirements of Bechtel User Manual M-18.

Serial Number: 1999-059-NPE Document Evaluated: ER 1997-0022-00-01 & ER 1997-0022-01-01

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 0 and CR 1999-1147.

Table – 1

Pen. No.	Line No. to be Protected from Over pressur-ization	Drill Disc(s) Of Valve No.	Add Relief Device on Line No.	Replace Bolts with New High Strength Bolts for Valve No.	Implement New Closure Time for Valve No.	Valve Loc.	Rev. Ops. Proc
36	4”-HBB-40	Q1P72F123 (Gate Drill Inboard Disc)	4”-HBD-520 (Relief Valve)	Q1P72F122. Gate		Ctmt Aux	
38	4”-HBB-44		None-Utilizes Pen. 39 Rupture Disc				
39	4”-HBB-43	Q1P72F149 (Gate Drill Inboard Disc)	4”-JBD-734 (Rupture Disc)	Q1P71F148. Gate		Ctmt Aux	
47	¾”-DCB-50	NO MODIFICATION REQUIRED					
49	4”-HBB-152	Q1G36F106 (Gate Drill Inboard Disc)	4”-HBD-1010 (Rupture Disc)	Q1G36F101, Gate		Ctmt Aux	
50	6”-HBB-102				Q1P45F067, Gate Q1P45F068, Gate	Ctmt Aux	(1)
51	6”-HBB-101				Q1P45F061, Gate Q1P45F062, Gate	Ctmt Aux	(1)
58	8”-HBB-6			Q1G41F044, Gate Q1G41F029, Gate		Ctmt Aux	
81	¾”-DCB-51	NO MODIFICATION REQUIRED					
84	3”-HCB-19			Q1P45F098, Gate Q1P45F099, Gate		Ctmt	
331	4”-HBB-42	Q1P72F126 Gate (Drill both Discs)	None –Utilizes Pen. 36 Relief Valve			Ctmt	
333	4”-HBB-111						(2)
348	4”-HBB-95	Q1P45F010 Gate (Drill both Discs)	4”-HBD-766 (Rupture Disc)			Ctmt	
349	4”-HBB-96	Q1P45F004 Gate (Drill both Discs)	4”-HBD-757 (Rupture Disc)			Ctmt	
364	1①”-HCB-20	Q1P45F097 Gate (Drill both Discs)	3”-HCD-31 (Rupture Disc)			Ctmt	

(1) Revision to Surveillance procedures is required.
(2) Affected Drywell isolation valves are: Q1B33F204 (inboard) & Q1B33F205 (outboard)

ER 97-0022-00 will provide instructions for the installation of the six relief devices (1 relief valve and 5 rupture discs), and ER 97-0022-01 will address the remaining required valve related modifications shown in table-1. NS&RA licensing commitment number A-34282 tracks the revision of the Operations Procedures and/or Instructions.

SCN 99/0008B for M-242.0 Rev 57 to document the holes in discs and add accident P/T information for gate valves P72F123, P71F149, G36F106, P72F126, P45F010 and P45F004. The SCN also documents bolt material and accident P/T information for valves P72F122, P71F148, G36F101, G41F044, G41F029, P45F098 and P45F099. Finally, the SCN documents accident P/T information for valves P72F125, P45F06l/F067/F003/F009/F062/F068 and P71F150/F151.

SCN 99/0001B for M-25l.0 Rev 37 to document the hole in disc/plug for globe valve P45F097. The SCN also documents accident P/T information for valves P45F096/F097 and B33F125/F126/F127/F 128.

SCN 99/0003A for M-189.3 Rev. 0 includes the new rupture discs.

SCN 99/0008B for MS-02 Rev 48 adds a reference note for GL 96-06 predicted pressures.

SCN 99/0001A for MS-44 Rev 1 includes a reference calculation of revised allowable seismic accelerations for some valves affected by GL 96-06.

SCN 99/0001A for M-141.l Rev 37 includes the new relief valve Q1P72F209.

SCN 99/G0001A for M-220.0 Rev 0 adds a note for the use of rupture discs for piping penetrations affected by GL 96-06.

Implementation of both ERs 97-0022-00 & 97-0022-01, and closure of the licensing commitment A-34282 will support the resolution of GL 96-06.

Enclosure 1 to this Safety 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/Drvwell may be heated beyond its maximum operating temperature. The concern is that water trapped in piping sections isolated during a Containment/Drywell isolation event would thermally expand and produce extremely high pressures that could potentially challenge the piping integrity.

Engineering Report GGNS-97-0002 Rev 0 and CR 1999-1147 identified some of the Grand Gulf penetrations susceptible to the over-pressurization issue of Generic Letter 96-06 and provided the related predicted pressures. The effects of the increased pressures for piping and valves associated with 10 Containment (36, 38, 39, 47, 49, 50, 51, 58, 81 & 84) and 5 Drywell (331, 333, 348, 349 & 364) penetrations have been evaluated. The results of the evaluations indicated that modifications or actions shown in table-1 above were necessary to assure the structural integrity of the penetrations piping and valves.

Note: Throughout this Safety Evaluation and ERs, 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:

These 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 ERs. The testing and design limits for Drywell bypass leakage are maintained.

A 1/8” diameter orifice will be drilled at a location approximately halfway between the disc hub and seating ring on the inboard disc for inboard Containment isolation gate valves Q1P72F123, Q1P71F149 and Q1G36F106. It is demonstrated via testing as well as engineering analysis (per Engineering Report GGNS 99-0014 Rev 0 that the outboard disc will experience significant bypass leakage at a differential fluid pressure across the disc above 400 psid, provided the disc thrust does not exceed 9,675 lbf total stem thrust limit will be higher . The modified valves meet the requirements to allow bypass leakage. The bypass leakage will occur from the pressurized side past the first outboard disc into the bonnet cavity and then through the drilled hole in the inboard disc. The pressure build-up between the isolation valves will be relieved into the portions of the piping that are non-safety related. To assure positive relief a safety related pressure relief device will be installed on the non-safety-related portion of the system piping. The non-safety piping is designed to seismic II/I criteria and is expected to remain intact during the postulated dynamic events. If failure of this piping were to occur, it would provide an additional relief path. This configuration will provide pressure relief for Containment penetrations 36, 39 and 49 piping located between the isolation valves without affecting the Containment leakage limits or requirements. (See Enclosure 1 for a detailed discussion of pressure relief scenarios)

A 1/8” diameter orifice will be drilled through both inboard and outboard valve discs at a location approximately halfway between the disc hub and seating ring for Drywell outboard isolation gate valves Q1P72F126, Q1P45F010 and Q1P45F004. Also a 1/8” orifice will be drilled in the globe valve plug/disc in Drywell outboard isolation globe valve Q1P45F097. This will allow continuous bypass leakage across the valves. A safety-related relief device will be installed past the modified valve on the non-safety-related portion of the system piping. This configuration will provide pressure relief for the Drywell penetrations 331, 348, 349 and 364 piping located between the isolation valves with some limited impact on the Drywell bypass leakage limits or requirements (See Enclosure 1 for a detailed discussion of pressure relief scenarios).

The 1/8” orifice size is considered adequate per calculation MC-Q1M24-99011. Rev 0 to provide enough flow to relieve the penetration pressures as temperature rises. Flow rates provided in the test data from engineering report GGNS 99-0014 are consistent with those required in calculation MC-Q1- M24-99011. All orifices will have a smooth, continuous inside and outside perimeter with no sharp edges to avoid introducing stress risers.

Whether one valve disc or two discs are drilled, the flow through the disc(s) from the high-pressure side to the low-pressure side will be the same. Since the penetration heat-up process is expected to occur over a period of time, the process of depressurizing the piping is essentially a throttling process. The throttling process occurs when fluid (i.e. liquid) flows from a region of higher pressure into a region of lower pressure through a valve or constricted passage (i.e., 1/8” opening). The region of higher pressure would be the penetration piping and the lower pressure region would be the system piping past the isolation valve containing the relief device. The pressure in the penetration piping between the isolation valves will equalize with the system piping via the drilled disc(s). The system piping for penetrations 36, 39, 49, 331, 348, 349 and 364 will contain a relief valve or rupture discs that will limit the penetration piping to no more than maximum rupture disc burst pressure + ΔP required for bypass leakage. At the time when the new relief valve opens or disc rupture occurs, the process fluid is calculated in engineering report 97-0002 to be at a temperature well below the boiling point of 212©F. Therefore no flashing is expected to occur.

The Drywell is a pressure-containing envelope that channels steam from a postulated loss-of-coolant accident through the horizontal vents and into the suppression pool for condensation. The UFSAR permits the tolerance of a certain amount of bypass leakage through the Drywell structure via

penetrations into the Containment. While the safety function of the Mark III primary Containment during a LOCA in the Drywell is contingent on the integrity of the Drywell, the primary Containment's ability to perform its safety function is relatively insensitive to the amount of Drywell bypass leakage. The Drywell allowable leakage rate is large enough that penetration flow paths about 1/8 inch in diameter have only a negligible impact on the total bypass leakage. Calculation MC-Q1M24-99004, Rev 0, evaluates the effect of leakage through .225" holes for 5 Drywell penetrations and demonstrates compliance with the current GGNS allowable Drywell bypass leakage limit presented in UFSAR Section 6.2.1.1.5. For the purpose of this modification, a tenth of an inch has been added to the opening diameter size to provide a tolerance in drilling the openings. The required Drywell Bypass Leakage test limit will consequently be reduced in the related testing procedures to accommodate this change. Surveillance procedure 06-ME-1M10-O-0003, "Drywell Bypass Leakage Rate", will therefore be revised to lower the allowable Drywell bypass leakage rate by 274 scfm. The overall leakage limit will be maintained within the ten percent allowed by Technical Specifications SR 3.6.5.1.1.

The Containment leakage limit is not affected since only one disc is drilled in Containment isolation valves. The drilled inboard disc on the inboard isolation valve is not considered part of the Containment isolation boundary. Only the non-drilled outboard disc of the inboard isolation valve is considered as a pressure-retaining disc and is effective as a Containment isolation boundary.

The installation of the relief valve will consist of installing a non-safety related 1" branch connection off the main pipe with a set of non-safety related flanges and a safety related relief valve attached at its end. Since the relief valve is intended to protect the safety related piping between the isolation valves, the installed relief valve is procured to ASME Section III requirements.

The installation of the rupture discs will consist of installing a non-safety related 1" branch connection off the main pipe with a set of non-safety-related flanges and a safety related rupture disc attached at its end. Since the rupture disc is designed to fail when line pressure reaches a specified value (maximum 126 psig or 650 psig) to protect the safety related piping between the isolation valves, the installed rupture discs are procured to ASME Section III requirements.

ASME Section III NC-7000 implies that the use of a rupture disc as a primary pressure relief device without an associated relief valve is not permitted. However, the use of a rupture disc to provide a relief path for safety related pipe through adjacent non-safety related pipe is not an application anticipated by the ASME Code. Section III always assumes that the over pressure protection device serves two functions (1) active function to prevent over pressurization, and (2) passive function to also provide pressure boundary function. The limitation contained in Subsection NC that requires a rupture disc to always be used with a relief valve is to ensure that a Code item (the valve disc) is performing the safety related pressure boundary function. However, the application of rupture discs as used in the modifications for GL 96-06 is different than that addressed by ASME Section III. The rupture discs are not performing a passive safety related pressure boundary function and therefore the Section III limitation is not specifically applicable.

The rupture discs are installed to protect the safety-related portion of piping from catastrophic failure under post LOCA conditions only. Contrary to a relief valve, the rupture discs are not intended to continuously preserve the safety-related piping pressure boundary during normal or maximum transient conditions. When called upon to perform their safety function, i.e., rupture during a LOCA condition, the rupture and loss of the pressure boundary for the applicable nonsafety-related systems is desirable to preserve Containment and Drywell pressure boundaries. Therefore the use of a rupture disc alone is considered acceptable for these applications. The systems affected have no function required or desired

post LOCA and the disc's rupture and subsequent loss of the non-safety related piping pressure boundary do not create undesired effects on safety related equipment.

The rupture discs are not considered part of the isolation mechanism for the affected Containment and Drywell penetrations. They only ensure failure of the non-safety piping pressure boundary early enough during a LOCA to preclude a potential similar failure occurring due to over-pressurization in the safety-related piping portion. This is consistent with the requirement that all GGNS non-safety-piping systems be designed to not cause a failure of the connected safety-related piping. This will meet the requirements of GL 96-06 regarding protecting penetration piping from thermal over-pressurization.

The newly added relief valve and rupture discs are procured ASME components to ensure they will perform the required function when needed. In addition, the relief valve is seismically qualified. However, seismic qualification of the rupture discs is not required, as the safety function of the discs is to fail under accident conditions. When called upon to perform their safety function, the rupture discs are no longer considered pressure boundary components. Any failure, although unlikely, due to a seismic event will not adversely affect the integrity of the associated safety-related piping or the isolation function of the isolation valves. Therefore, the Drywell and Containment isolation functions will not be altered. System leakage of the piping due to a rupture disc failure under normal operating conditions is unlikely due to margin and maintenance.

A relief valve instead of a rupture disc was selected for installation on the Drywell Chilled Water System to ensure the availability of this system after a small break LOCA event as Grand Gulf emergency procedures restore the system to help mitigate accident consequences.

Safety relief valve rating and size are selected to provide the required pressure relief when needed and to ensure that no relief would occur during normal operating conditions. The rupture discs' rating and size are determined in calculation MC-Q1M24-99011, Rev 0, and are installed to provide the required pressure relief when needed while ensuring that no disc rupture will occur due to worst case operating transients. The additions of the small bore branches (including relief valve, flanges and rupture disc) have been evaluated along with the existing piping in stress calculation MC-Q1111-99012, Rev 0, for all plant conditions (including the elevated rupture pressure) to meet the design requirements of ASME Section III, Subsection NC-3600, Code Case 1606-1, ANSI B31.l, M-18 and drawing 9645-M-1398.

There are no pipe break jet impingement cones postulated in the area of the newly added relief devices. Therefore, the 1" lines cannot fail due to jet impingement caused by an adjacent main line break. Failure due to suppression pool swell is not expected since the 3 rupture disc lines (P45) installed in the swell zone are shielded by existing piping and are located at least 20 ft above the normal suppression pool elevation (111'-10"). This will ensure that insignificant pool swell loads are acting on the rupture disc connections.

Isolation valves Q1P72F122, Q1P71F148, Q1G36F101. Q1G41F044, Q1G41F029, Q1P45F098 and Q1P45F099 for Containment penetrations 36, 39, 49, *58* and 84 will require replacement of the existing SA-193 Grade B (allowable stress value S = 25 ksi) body to bonnet flange bolts with similar 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 NPE P41F007A-B/F015A, B/F016A-B/ P45F273/F274/ P72F121-F126. NPE-G41F029/F044/P42F114/116/117, NPE-E51F076, CC-Q1 111-93017 and CC-Q1111-99001 demonstrate the structural integrity of the valves with this change.

Closure of penetrations 50 and 51 outboard Containment isolation valves Q1P45F068 and Q1P45F062 will be delayed at least three and half (3½) seconds over the closure time of the associated inboard

isolation valves Q1P45F067 and Q1P45F061. This will provide enough time to allow the piping to partially drain to the floor and equipment drain tanks if closure occurred during a sump pump down cycle. The resultant air void in the penetration piping will preclude its pressurization due to temperature increase in the Containment. The 3 seconds delay includes a maximum 2 seconds potential time differential between the electrical Divisions (Division 1 and Division 2) isolation LOCA signals. However IST tolerance must be added to valve stroke time and the time delay of 3 ½ seconds for
penetrations 50 and 51 must be increased to accommodate this tolerance. The addition of the IST tolerance along with the limited ability of the valves to demonstrate adequate repeatability will cause the valve stroke time to exceed the current limit of 7 seconds presented in TRM table TR3.6.l.3-l. GGNS has proposed to revise the licensing basis for these Containment isolation valve closure times from 7 seconds to 110 seconds. This submittal is currently under NRC review and approval may be granted prior to RFO10. Therefore. this modification for valves Q1P45F061/F062/F067/F068 (penetrations 50 and 51) will not be implemented until the NRC approval is granted and the TRM revised.

The modification for valves Q1P45F061/F0621F067/F068 is contingent upon NRC approval of the stroke time increase (Reference Entergy letter GNRO-98/00085).

These staggered isolation times are needed to assure the integrity of Containment penetrations 50 and 51. The determination and evaluation of the required time delay is presented in calculation MC-Q1P45-99014, Rev. 0. The acceptance criteria presented in Surveillance procedure 06-OP-1P45-Q-0002, “Floor, Equipment, and Chemical Drain Isolation Valve Operability Check”, will be revised to accommodate this change.

The equivalent thrust force to close the affected air operated valves 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. The largest internal pressure predicted for AOV’s evaluated in ERs 97-0022-00-01 and 97-0022-01-01 is 1190 psig. An evaluation has been performed to ensure that stem ejection forces cannot exceed valve required unwedging forces during accident conditions. Therefore. undesired opening of AOV isolation valves will not occur due to stem ejection forces induced by valve internal pressure (reference response to EAR MC-99-0014).

An analysis has also been performed in calculation PC-Q1111-99004 for AOV’s P71F149 and G36F106 to demonstrate the maximum possible seating force for 110 psig cylinder pressure is not large enough to prevent disc flexure needed to vent the associated penetrations. The maximum thrust limit in MS-25 for MOV P72F123 is also not high enough to prevent disc flexure.

All the 1” rupture disc branch connections will be connected to the run pipes through l/8” openings in the run pipes to limit leakage rate in case of rupture disc failure. Pool level increase due to disc rupture during a LOCA condition is insignificant when compared to level increase from other sources. No adverse chemistry concerns are createdpost LOCA by the introduction of small quantities of water from these s sytems into the supression pool. The discs are designed for burst pressures well above maximum operating pressures for the systems to prevent inadvertent disc failures. These discs will be periodically disassembled and inspected, and also replaced with new discs every five years. The systems on which the rupture discs are installed are not required post LOCA and therefore s sytem inventor loss is not a concern. The new relief valve will be tested in accordance with procedure 07-S-14-395 “General Maintenance Instruction — Safety and Relief Valve Program — Safety Related”.

To minimize the potential for debris blocking the 1/8” holes in the valve disc/piping. the rupture discs are located on the top or sides of the run pipes. The valve discs are normally positioned outside the flow

stream where they should not be susceptible to debris accumulation. The P72, P71 and G36 systems are closed systems that are relatively free of debris. Any accumulation of debris in the P45 system would be expected to be blown free by the high fluid pressures.

The pressure increase described in GL 96-06 will cause the systems to experience pressures above 275 psig. However this will occur for less than 2% of the total s sytem operation time. Therefore the classification of these lines as moderate energy remains applicable, and there is no change to the original HELB/MELB evaluation.

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.

Operations Procedures for Drywell penetration 333 will be revised to require draining the line prior to isolation during normal power operation since the line may only be used during plant outages. These procedure changes are tracked b NS&RA licensing commitment no. A-34282, and are also required to return the system to service post modification.

Serial Number: 1999-060-PSE Document Evaluated: TSTI 1N19-99-002-0-N

BRIEF DESCRIPTION OF CHANGE, TEST, OR EXPERIMENT:

TSTI 1N19-99-002-0-N will drain the HP Condenser water trough, re-fill the water trough with a dye solution through the water trough drain line, and will drain the dye solution into drums after the plant is shut down for RFO10.

REASON FOR CHANGE, TEST, OR EXPERIMENT:

In April of 1999, the HP Condenser expansion joint seal developed a leak. The leakage rate through the seal has averaged approximately 6 gallons per day. Since there is a water trough running around the perimeter of the expansion joint, there is no way to inspect the seal for the location of leakage. Draining the water trough and re-filling it with a fluorescent dye solution will allow a means of identifying the location of the leak during RFO10. After the plant is shut down and the HP Condenser is entered, shinning a black light on the expansion joint will cause the dye that has leaked through to fluoresce and identify the location of the leak.

EVALUATION SUMMARY AND CONCLUSIONS:

SAR Section 10.4.7.3 states “the condensate and the feedwater system serve no safety function”. The components directly affected by the TSTI are the water trough drain line (shown on SAR Figure 10.4-010) and the HP condenser expansion joint. Neither the drain line nor the expansion joint serves a safety function. The TSTI will use a temporary pump and hoses to add a dye solution to the water trough to facilitate finding a leak in the expansion joint. The temporary pump, the hoses, and the storage drums will not be in the vicinity of any piece of equipment important to safety. Therefore, the potential failure of a hose or spillage of the dye from a drum will not compromise any safety-related system or prevent safe reactor shutdown. Use of the condenser trough drain line to add the dye solution will be similar to the normal makeup of water to the trough, but the temporary pump pressure and flow will be much less than the normal makeup flow and pressure. The dye solution will be composed of demineralized water and 200 ppm Sherwin lnc. A-416 fluorescent additive. The dye manufacturer evaluated the use of the dye solution and determined that the solution will have no adverse affect on the seal’s service life. Draining of the HP Condenser water trough was previously evaluated and approved by Safety Evaluation 99-0033-R00. The Safety Evaluation analysis allows draining the water trough if expansion joint leakage is less than 0.7 gpm which would correspond to an increase in Off Gas flow of 0.32 scfm. Since the HP expansion joint leakage was discovered, the leakage has averaged below 10 gallons per day or 0.007 gpm which corresponds to a potential increase in Off-Gas flow of 0.02 scfm. The current leakage is much less than the leakage evaluated by Safety Evaluation 99-0033-R00. Therefore, draining the trough is bounded by the safety evaluation. Off Gas flows will be monitored during the draining of the trough, and if a significant increase in flow occurs, Operations should restore the water seal back to the joint to limit air in-leakage. Chemistry approved use of the dye (Sherwin Inc. A-416) per Plant Administrative Procedure 01-S-08-18: GGNS Chemical Control Program. Based on Chemistry’s analysis of the dye, the quantity of the solution has been limited to 200 ppm. The analysis assumed that the full volume of the water trough (135 gallons) leaked through the seal and entered the condensate system. A 200 ppm solution of the dye can be effectively removed by the RWCU system and through normal decomposition in the system although it would cause an increase in the condensate/feedwater/reactor water conductivity. The increase is expected to be small in the condensate and feedwater systems since the organic chemical may not be removed efficiently by the condensate demineralizers. Upon entering the reactor, the organic chemical will decompose due to radiation and produce decomposition products such as Carbon, Hydrogen, and Oxygen. These products will primarily be in gaseous form and exit the reactor with the steam. Portions of the by-products will react with reactor water to produce ionic species that result in conductivity increases. The conductivity increase will be of short duration and due to the small amount of chemicals used will not exceed Technical Specification limits. Therefore, the dye solution used during the TSTI will have no detrimental affect on the reactor vessel and its associated systems, structures, components

(SSCs) as described in SAR Section 4.0. The quantity of the dye will be controlled by the TSTI. At the end of the TSTI, the water trough will be drained into drums and flushed with demineralized water. The resultant waste water will be processed through the Radwaste system. The Safety Evaluation concluded that the TSTI will not create the need to change the GGNS Technical Specifications, increase the probability or consequences of an accident previously evaluated in the SAR or create any new accident scenarios for the plant not previously analyzed in the SAR. The TSTI will not increase the probability of occurrence of a malfunction of equipment important to safety or the consequence of a malfunction of equipment important to safety previously evaluated in the SAR. Lastly, the Safety Evaluation concluded that the TSTI will not decrease the margin of safety as defined in the basis for any Technical Specification.

Serial Number: 1999-061-NPE Document Evaluated: ER 98/0358-01-01

BRIEF DESCRIPTION OF CHANGE, TEST, OR EXPERIMENT:

The proposed change is the final repair disposition to GGCR1998-0673 and will re-install root valve 1N11FX301. The new root valve will be an instrument type valve (1/2" J3-N) installed in the pressure line from the turbine 1st stage header to pressure transmitters C71N052B, C71N052D, 1C34N007, 1N11N038B, C11N054B and C11N054D.

REASON FOR CHANGE, TEST, OR EXPERIMENT:

The interim repair response per ER 98/0358-00-00 deleted root valve 1N11FX301. The final repair disposition to CR 98/0673 per ER 98/0358-01-01 is to reinstall a root valve for the purpose of isolating the associated instrument sensing line to allow for maintenance or isolation of a possible steam leak in the sensing line or connected instruments. Per Plant Staff request ER 98/0358-01-01 will provide instruction to install a 1/2" instrument isolation valve for root valve 1N11FX301 in lieu of a 3/4" carbon steel valve.

EVALUATION SUMMARY AND CONCLUSIONS:

The reinstallation of root valve 1N11FX301 is to allow isolation of the instrumentation tubing from the main header in the event of a tubing leak, rupture or maintenance. Section 15.6.4.2.2 of the UFSAR discusses the potential for, and consequences of, a steam leak in the Turbine Building (outside containment). The event analyzed as presented in this section of the UFSAR adequately bounds this potential scenario. There are no new systems or components 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 No. 10.3-001-1 since 1N11FX301 was deleted from this UFSAR Figure during the interim repair disposition of CR 98-0673 per ER 98/0358-00-00. Reinstallation of the instrument root valve (1N11FX301) 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 entirely contained within the confines of the power block and will not affect or impact the plant's radiological or non-radiological effluents. After a review of the proposed change, it has been concluded that the reinstallation of root valve 1N11FX301 does not represent an Unreviewed Safety Question and will have no adverse affects on the environment.

Serial Number: 1999-062-NPE

Document Evaluated: ER 97/0487-01-00 & LDC 1999-076

BRIEF DESCRIPTION OF CHANGE, TEST, OR EXPERIMENT:

This safety evaluation addresses a change in the amount of water coverage necessary for control blade movements using the auxiliary hoists on the refueling and fuel handling platforms. The actual reduction of water coverage and normal-up interlock setpoint changes will be determined by P&SE, but the total water coverage reduction may not exceed 10 inches (6' 2" water coverage). This change does not approve operation of any equipment in a manner inconsistent with its original design.

REASON FOR CHANGE, TEST, OR EXPERIMENT:

Various control blades currently being used in the reactor are approaching their end of usable life and replacement of these blades is necessary in order to maintain blade reactivity requirements. The process by which these blades are replaced involve the use of the auxiliary hoists on the refueling and fuel handling platforms.

The current water coverage requirement for these hoists' loads make the blade movements very difficult as determined during RFO8 when eight control blades were replaced. Specifically, difficulties were experienced while loading and unloading the blades in the upender due to close tolerances. Additionally, the auxiliary hoist's cable and load has to be physically pulled close to the main grapple in order to pass through the cattle chute. This results in a reduction of water coverage, over the blade. To compensate for this reduction and maintain the coverage requirement, the blade was stopped a distance equal to this reduction before reaching the normal up limit switch as indicated by a piece of tape placed on the cable. Compliance with the coverage limit was the sole responsibility of the operator(s) involved instead of the intended limit switches.

By reducing the water coverage limit for the blades, the overall blade moving process will be facilitated and made more efficient. Also, compliance with the reduced water coverage limit will be controlled by the redundant normal up interlocks instead of a piece of tape and operator actions.

EVALUATION SUMMARY AND CONCLUSIONS:

A 10 inch reduction in the water coverage (shielding) above the control rod blade will produce exposure rates approximately *5* times greater than normal. These higher rates will only be experienced during the brief time when the blades are being moved through the cattle chute.

The basic functions and equipment used in control blade movements will remain unchanged. When operated as designed, structural and seismic adequacy of the platforms are not compromised by a change of blade coverage. The additional impact energy of a dropped blade due to the increase in height is well bounded by current analyses and administrative controls. Accidents currently analyzed in the FSAR (15.7.4 and 15.7.6) are no more likely to occur and no new accidents are introduced. Radiological consequences of these accidents are within regulatory requirements as determined by the accident analyses. Additionally, malfunctions of equipment necessary for safety are no more probable nor are any additional malfunctions introduced. No reduction of any Technical Specification margin of safety as described in the bases will occur.

Serial Number: 1999-063-PSE Document Evaluated: TA 99-025

BRIEF DESCRIPTION OF CHANGE, TEST OR EXPERIMENT:

This safety evaluation addresses the operability concerns associated with supplying temporary power from ESF Bus l6AB and BOP Bus 11HD to loads normally supplied by Bus 15AA. Temporary power is being supplied to loads as required by SOI 04-1-01-R21-15, see attached Table 1. The duration of this Temporary Alteration is scheduled for less than 7 days, and will have personnel dedicated to perform an oversight function. For the duration of the temporary alteration, an updated load list document for power supply changes will be supplied to the Main Control Room, Work Control Center, Equipment Clearance Group, Maintenance Planners and Radwaste Control Room (if equipment is under Radwaste Control) by oversight personnel.

REASON FOR CHANGE, TEST OR EXPERIMENT:

ESF Bus 15AA, System R21, provides power to safety and non-safety related components and instrumentation. Required maintenance, inspection and cleaning of the 15AA ESF Bus requires that it be deenergized for approximately 24 - 48 hours. This work will be conducted while the reactor is in Mode 4 or *5*.

SAFETY EvALUATION SUMMARY AND CONCLUSIONS:

The additional power requirements being placed on Bus 11HD are within the loading capabilities and no loading calculations are required. The addition of battery charger 1DK5 (approximately *55* kW) to ESE Bus 16AB translates to an increase of 3.5% during normal operation, no increase in a forced shutdown (LOP) condition, and no increase in a Loss-of-Coolant Accident (LOCA). The additional load will not adversely affect the reliability due to loading, since the load profiles have accounted for additional load values. No components being supplied temporary power will be considered operable. In all cases power is being supplied as a matter of convenience and not plant safety. LCOs will be entered where applicable. For the duration of this temporary alteration, the following information in the UFSAR will be inaccurate: Table 8.3-9, Figure 8.3-010, Figure 8.3-010A, Figure 8.3-010B, Paragraph 8.3.2.1.1, and Paragraph 8.3.2.1.6. The conclusion of this safety evaluation is that no unreviewed safety question exists and that the Technical Specifications are not impacted or changed by the proposed work.

Serial Number: 1999-064-NPE

Document Evaluated: CR 1997-1433 & LDC 1999-071

BRIEF DESCRIPTION OF CHANGE, TEST OR EXPERIMENT:

GGCR 1997-1433 (formerly GGCR 1997-1230-04), identified a discrepancy in that the secondary containment penetration in-leakage analysis for the Standby Gas Treatment System (SGTS) did not account for the penetration used for the hydrolasing flow path (penetration AJ-1F). As Part of the corrective actions for the CR, the applicable calculations were revised to incorporate the secondary containment inleakage pathway associated with penetration AJ-1F.

CR 1999-0360 was initiated in the review process and questioned the validity of the TS surveillance criteria since the TS value, as calculated in revision 0 of calculation 3.9.12 [Ref. 2❁, did not include the contribution from unqualified lines 2” and under. Revision 1 of calculation 3.9.12 was updated to determine the testing requirements for the secondary containment. The revised calculation and the resolution of this CR will be ultimately used to modify the GGNS Technical Specifications.

This safety evaluation evaluates the revised calculations as well as chanqes to the Technical Specifications Bases 3.6.4.2 and UFSAR section 6.2.3.2, 6.5.3.2 and Table 6.2-43. The TS bases and UFSAR changes clarify that the SGTS design analyses demonstrate the system can meet the drawdown time and maintain the required negative pressure assuming the failure of all non-qualified small bore piping (i.e., smaller than 2 1/2” nominal dia.). The failure of a non-isolated 4” dia. line is clarified as a non-limiting sinqle failure.

REASON FOR CHANGE, TEST, OR EXPERIMENT:

Technical Specification Bases and FSAR for SGTS are revised to reflect the auxiliary building drawdown analysis described in Bechtel calculations 3.9.3, Rev. 1, 3.9.8, Rev.1, and 3.9.12, Rev.1. These calculations were revised to account for open penetration AJ-1 F. The change further clarifies that the analysis allows for failure of all nonqualified lines smaller than 2 1/2 inches. Penetration AJ-1F is included in the population of penetrations less than 2½ inches.

SAFETY EVALUATION SUMMARY AND CONCLUSIONS:

Condition Report (CR) GGCR 1997-1433 (formerly GGCR 1997-1230-04) was issued to address an uncontrolled release of contaminated water from the auxiliary building that occurred during hydrolasing activities on the refueling floor. The CR identified a discrepancy in that the secondary containment penetration in-leakage analysis for the Standby Gas Treatment System (SGTS) did not account for the penetration used for the hydrolasing flow path (penetration AJ-1F). As Part of the corrective actions for this CR, the applicable calculations were revised to incorporate the secondary containment inleakage pathway associated with penetration AJ-1F.

This safety evaluation evaluates the revised calculations as well as chanqes to the Technical Specifications Bases 3.6.4.2 and UFSAR sections 6.2 and 6.5. The TS bases and UFSAR changes clarify that the SGTS design analyses demonstrate the system can meet the drawdown time requirements of the Technical Specifications. In addition, revised calculation establishes the secondary containment boundary testing requirement to ensure that the testing acceptance criteria demonstrate that the boundary will perform as required under accident conditions (i.e., maintain the required negative pressure assuming the failure of all non-gualified small bore piping). The evaluation determines that althouqh a change to the GGNS Technical Specifications is required, an unreviewed safety question does not exist.

Serial Number: 1999-065-NPE Document Evaluated: ER 1998/0314-00-00

BRIEF DESCRIPTION OF CHANGE, TEST OR EXPERIMENT:

ER 98/0314 provides evaluation and justification for the Cooling Tower degradation issues identified in CR 1998-0514-00 related to removal of the De-icing Ring, repair of concrete, basin floor joint repair and repair of structural steel.

REASON FOR CHANGE, TEST, OR EXPERIMENT:

The condition of the de-icing ring supports have deteriorated to a condition that may result in failure of the support, and dropping of cement asbestos pipe into the Cooling Tower Basin. Other structural repairs are required due to degradation from normal service.

SAFETY EVALUATION SUMMARY AND CONCLUSIONS:

The Circulating Water (CW) system is not addressed in the GGNS Technical Specifications; however, the condenser vacuum setpoint is addressed. Loss of the CW system may result in a turbine trip through the loss of condenser vacuum. However, implementation of the evaluated change will not adversely affect the CW supply to the condenser and will therefore not adversely affect condenser vacuum. The CW system serves no safety related function. The change will not compromise any safety related system or prevent safe shutdown since no new interface with equipment important to safety is created. Therefore the existing evaluations are considered bounding for the system. The technical specifications do not contain any margins of safety for operation or design of the CW system. Implementation of the described change will not affect or prevent safe shutdown of the reactor.

Serial Number: 1999-066-NPE

Document Evaluated: ER 1999/0066-00-01
CN 99/0079

BRIEF DESCRIPTION OF CHANGE, TEST OR EXPERIMENT:

Remove 1-JGD-42 with check valve NSP41F216 from SSW basin A and the SSW A valve room.

REASON FOR CHANGE, TEST, OR EXPERIMENT:

Chlorine injection line 1-JGD-42 was abandoned in place in 1989 per MCP 89/1095. The line has since deteriorated until it may be structurally unsound.

EVALUATION SUMMARY AND CONCLUSIONS:

The subject line is no longer in use and its removal will have no effect on the SSW system or any other safety related system. Its removal will prevent possible future problems but will require a drawing update. Nine hangers in good condition will be left on the south wall of the SE qudrant of SSW A basin. This change does not represent an unreviewed safety question because no safety related components, systems or procedures are affected.

Serial Number: 1999-067-NPE

Document Evaluated: ER 96/0421-01-00
ER 97/0283-00-00
ER 97/0288-02-00
ER 96/0350-00-00

(Revision to the Safety Evaluation)

DESCRIPTION OF CHANGE, TEST, OR EXPERIMENT:

The Nuclear Change Response to the following ERs, enhances the operating torque margin (i.e., the difference between the torque required to operate the valve and the torque limit of the actuator) and the thrust capabilities for the indicated motor operated valves (MOVs):

ER 96/0421-01-00 - Operating Margin Improyement for Component Cooling Water MOVs
QlP42Fl14-B - CCW Supply To Containment Drywell Outboard Isolation Valve
QlP42F116-A - CCW Return from Containment Drywell Inboard Isolation Valve
QlP42F117-B - CCW Return from Containment Drywell Outboard Isolation Valve

ER 97/0283-00-00 - Operating Margin Improvement for Nuclear Boiler System MOVs
Q1B21F016-B - Main Steam Line Inboard Drain Valve
Q1B21F019-A - Main Steam Line Outboard Drain Valve

ER 97/0288-02-00 - Operating Margin Improvement for Reactor Water Cleanup MOVs
Q1G33F028-B - RWCU to Main Condenser Inboard Isolation Valve
Q1G33F039-A - RWCU to RHR System Outboard Isolation Valve
Q1G33F040-B - RWCU to RHR System Inboard Isolation Valve

ER 97/0350-00-00 - Operating Margin Improvement for Low Pressure Core Spray and RHR MOVs
Q1E2IF011-A - LPCS Min Flow to Suppression Pool Isolation Valve
Q1El2F009-B - RHR Shutdown Cooling Inboard Isolation Valve

This will be accomplished by replacing the motor pinion gear and worm shaft gear sets in each actuator to increase the overall actuator ratio (OAR), and by replacing the actuator motor with an actuator motor capable of increased torque output. The valves listed above will have their yoke legs stiffened and/or a new valve yoke assembly with legs stiffened will be installed. An increased Kalsi thrust rating will be applied to the RWCU valve actuators. The inactive leakoff nipples for valves Q1G33F028, Q1G33F039, Q1B21F019 will be plugged. New actuators will be installed for the Nuclear Boiler, CCW and LPCS system MOVs. Valves QlB21F019 and QlE12F009 will have new valve stems installed. Valve Q1E12F009 will have high strength yoke/actuator bolts installed and valve QlE21F011 will have high strength body/bonnet and body/yoke bolting installed. Also, the instantaneous breaker settings for the valve motors are being increased and the appropriate thermal overloads will be installed due to the larger horsepower motors. CN 1999-0049 changes ER 97/0350-00-00 as follows: corrections to the materials list stud size for valve Q1E21F011 and thermal overload relay testing procedure nNumber; additions to the level of detail of information for valve QlE21F011 in SCN 99/0005 (MS-25.0): and changes in fabrication method to fit the yoke stiffeners for valve QlEl2F009 to the actual configuration of the yoke. The leakoff connection plug that was to be installed on valve QIE2IF0l I is deleted from ER 97/0350-00-00. None of the changes authorized in CN 1999-0049 alters any conclusions discussed in the original SE 99-0028-00. CN 1999-0054 changes ER 97/0283-00-00 as follow's: corrects the thermal overload relay testing procedure number, adds to the level of detail of information for valves QlB21F016 and Q1B21F019 in SCN 99/0003 (MS-25.0) and changes from the use of a field-strengthened valve yoke to use of a strengthened yoke supplied by the valve manufacturer for valves QIB2lF0I6 and QlB2lF0l9. None of the changes authorized in CN 1999-0054 alters any conclusions discussed in the original SE 99-0028-00.

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 the selected valves, the margin between required torque/thrust versus maximum available torque/thrust can be increased.

EVALUATION SUMMARY AND CONCLUSIONS:

The changes made by these ERs will increase the degraded voltage actuator capability (DVAC) torque for the Limitorque motor operators installed on the subject valves. The changes made by these ERs will increase the operating margin for each of the motor operated valves listed above, while addressing industry concerns related to gearbox “run” efficiency.

The UFSAR does not address the operating torque/thrust requirements or capabilities of the subject motor operated valves. The operator, stem, bolting, leak off port, motor and gear changes (as applicable to the individual valves) do not alter the function or operation of these MOVs. The slower stem speed resulting from the OAR change on these valves causes the nominal stroke time for the valves to increase. Based on the fact that the post-modification calculated stroke times for each of the valves which have analytical stroke times are bounded by the maximum isolation time contained in TRM Table TR3.6.1.3-1 and the analytical stroke times contained in IJFSAR Table 5.2-5 and UFSAR Table 6.2-44, as applicable, the new valve stroke times will not affect the safety analysis, function or operation of the valves (Reference calculation MC-Q1111-98002, Rev. 1).

These ERs do not adversely impact the electrical penetration assembly, and reactor containment electrical penetrations are not described in the Technical Specification; however, they are described in the Technical Requirements Manual (TRM) Section 6.8.1. TRM Section 6.8.1 states that each of the primary and backup overcurrent protective devices associated with each primary containment electrical penetration circuit, as shown in TRM Table 6.8.1-1, shall be operable. TRM (and UFSAR) Table 6.8.1-1 lists the breaker settings for 52-1511-44 (Q1P42F116-A), 52-1611-25 (Q1P42F117-B), 52-1611-31 (Q1P42F114-B), 52-1631-20 (Q1B21F016-B), 52-1631-37 (Q1E12F009-B), 52-1631-50 (Q1G33F028-B) and 52-1631-52 (Q1G33F040-B) and is being updated to reflect the breaker setting changes due to the ERs. Breakers 52-1511-40 (Q1G33F039-A), 52-1531-10 (Q1B21F019-A) and 52-1511-34 (Q1E21F011-A) are not listed in TRM Table 6.8.1-1 since these valves are outside containment and, therefore, do not utilize a reactor containment electrical penetration. The protective settings for the breakers are set such that the breakers will trip before the penetration can be damaged by fault currents. Fault protection is still provided using the new settings by breakers and fuses which protect them as demonstrated in calculation EC-Q1111-99003, Rev. 0. The ERs do not change the requirements to test these breakers.

UFSAR Tables 8.3-1 and 8.3-2 lists the MOV load as 185 kW for the Division I and II ESF Loads respectively. Note d in the tables for the MOV load indicates that the load is an intermittent load and not included in the long-term loading. Additionally, it states that the loads are considered in the voltage drop calculations. Calculation EC-Q1111-90028, AC Electrical Power Systems, does include the MOV loads even though it does not categorize MOV loads as a single grouping. The kW loading for MOVs in UFSAR Tables 8.3-1 and 8.3-2 are being revised to reflect the increase in load resulting from each of the ERs evaluated by this safety evaluation.

The increased Kalsi thrust rating for the RWCU MOVs (Q1G33F028, Q1G33F039 and Q1G33F040) will be applied by increasing the bolt torque on the connections between the actuator and the valve yoke legs. These changes will allow the valve to withstand the higher actuator torque and thrust capability in the closing direction initiated by the motor and gear changes for these valves.

The replacement of the existing SMB-000 actuators with SMB-00 on valves Q1B2lF0l6, Q1B21F019, Q1P42F114, Q1P42F116, Q1P42F117, and Q1E21F011 is necessary to allow the use of higher torque motors for these valves (i.e., 25 ft lbf in place of the existing *5* ft lbf motors). The yoke legs will be strengthened for valves Q1G33F028, Q1G33F039, Q1G33F040, Q1B21F0l6, Q1B2lF0l9 and Q1E12F009 to accommodate the increased thrust capabilities of the higher torque motors. Additionally, new yokes will be required for valves Q1B21F016, Q1B21F019, Q1E21F011, Q1P42F114, Q1P42F116 and Q1P42F117 to permit the mounting of the SMB-00 actuators and to accommodate the increased thrust capabilities of the higher torque motors/actuators.

The leakoff nipples for valves Q1G33F028, Q1G33F039, and Q1B21F019 are being removed and the hole plugged. This is being done since the nipples must be removed to remove the yokes on this type valve and then welded back if they are retained. However, since these leak off nipples are not utilized they will not be welded back on, instead the hole will be plugged. This is an ASME code pressure boundary modification, therefore, it will be performed in accordance with ASME code requirements.

Two plant valves (feedwater isolation valves Q1B21F065A and B) have previously experienced stem cracking. It was determined that the failures were a result of hydrogen embrittlement cracking. Valves Q1B21F0l9 and Q1E12F009, which are similarly exposed to high temperature reactor chemistry water, could also be susceptible to hydrogen embrittlement cracking. The valve stems for valves Q1B21F019 and Q1El2F009 are being removed and replaced. These two valve stems removed will be subjected to examination and analysis to determine if they have any indication of cracking similar to that experienced by the feedwater valves. New valve stems with materials resistant to this type effect, will be used to replace the valve stems removed.

The Q1E21F011 valve will have its body/bonnet and body/yoke bolting replaced with high strength bolting. It was determined that the body/bonnet bolting was the weak link for the MOV. Additionally, valve Q1E12F009 will have high strength yoke/actuator bolts installed for the same reason. Replacement of the bolting will allow the MOVs to accommodate the revised thrust rating for the valves following the replacement of the valves actuators and motors.

No new interfaces with equipment important to safety are created and no new failure modes which would alter existing accident analyses are introduced. There will be no increase in 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 changes made by these ERs will not create the possibility of an accident or malfunction of a different type than any other evaluated previously in the safety analysis report. There will be no reduction in the margin of safety as defined in the bases for any technical specifications. Therefore, these changes will not introduce an unreviewed safety question.

Serial Number: 1999-068-NPE Document Evaluated: ER 98/0099-00-00

DESCRIPTION OF CHANGE, TEST, OR EXPERIMENT:

ER 98/0099 will modify the digital feedwater control system software to eliminate nuisance alarms that occur when reactor level decreases below the range of the upset range level transmitter. ER 98/0099 will also modify the feedwater control software such that the system will automatically transfer to single element mode at Level 3, the initiation point of the setpoint setdown function.

REASON FOR CHANGE, TEST, OR EXPERIMENT:

The modifications to the upset range level transmitter supervision logic will eliminate nuisance 'DFCS Trouble' alarms under post scram conditions when level drops below 0". The modifications to the single element transfer logic will lessen post scram level overshoot by eliminating an observed control output step (and associated feedpump speed increase) which occurs as the system transfers from three element mode to single element mode at the completion of the setpoint setdown function.

EVALUATION SUMMARY AND CONCLUSIONS:

The feedwater control system is an operational power generation system with no safety related functions. There are no existing interconnections with safety related systems and no specific regulatory requirements are imposed on the system. No new interfaces with safety related systems will be created as a result of the proposed modifications. The proposed modification to the single element / three element transfer logic will not affect system performance during the 'setup' portion of the setpoint setdown function (i.e., if reactor water level remains below Level 3, the feedwater flow demand signal will remain at maximum for 10 seconds as with the existing system). The single element/three element transfer logic modification will lessen post scram level overshoot after the setpoint setdown function is completed. The proposed alarm logic change will have no affect on the control functions of the system. Thus, no new challenges to safety related systems will be created as a result of the modifications. The probability of malfunctions of equipment important to safety will not be increased, and the modifications will not create the possibility for a malfunction of equipment important to safety of a different type than previously evaluated.

The proposed software modifications will have no affect on the reliability or fault tolerance of the feedwater control system. All postulated failure modes of the feedwater control system will remain bounded by previously analyzed events. Therefore, the proposed modifications will not increase the probability of occurrence of a previously analyzed accident or create the possibility of an accident of a different type than previously analyzed.

The proposed modifications to the feedwater control system will have no impact on radionuclide population, release rate, release duration, release mechanisms or release barriers. The feedwater control system is not required during or after accident conditions and is not required for safe shutdown of the plant. Therefore, the radiological consequences of accidents or malfunctions of equipment important to safety will not be increased.

Serial Number: 1999-069-NPE

Document Evaluated: ER 98/0099-00-01
& ER 1999-0466-00-01

(Revised Safety Evaluation)

BRIEF DESCRIPTION OF CHANGE, TEST, OR EXPERIMENT:

ER 99/0466-00-01 will replace valve Q2P41F006A with a blind spool assembly. The basis for this valve in the Standby Service Water (SSW) System is to provide an isolation boundary. The removal of this valve will allow this valve to be reused in order to replace valve
Q1P41F006A-A that has excessive leakage.

Valve Q1P41F006A-A was enhanced prior to plant start up to increase the trunnion bolt holes from 5/8" to ¾"**.** The replacement valve (formerly Q2P41 F006A) will have the trunnion holes increased from 5/8" to ¾" holes to match valve Q1P41F006A-A.

The USAR figure 9.2-001 identified valve Q2P41F006A on the continuation piping. This figure will be revised to show the valve removed and show one blind flange of the blind spool assembly, which was installed.

REASON FOR CHANGE, TEST, OR EXPERIMENT:

Q1P41F006A-A valve has excessive leakage across the seat and must be replaced. Unit 2 valve Q2P41F006A-A is used only for isolation and is an acceptable like for like replacement for the leaking Unit 1 valve. This modification is replacing valve Q2P41F006A-A with a blind spool assembly consisting of two blind flanges and a 24" section of pipe welded on center on the outside of each flange to maintain piping system structural integrity. The existing Unit 2 Q2P41F006A-A valve will be reused to replace Unit I valve Q1P41F006A-A.

EVALUATION SUMMARY AND CONCLUSIONS:

Since all valves have the potential for leakage, a blind spool consisting of two blind flanges and a 24" section of pipe welded on center on the outside of each flange is inherently more reliable than the existing valve. The blind spool assembly meets the original design requirements for the SSW system to provide an isolation boundary.

Valve Q2P41F006A-A will be structurally enhanced by increasing the trunnion boltholes from 5/8" to 3/4". This is the same enhancement that was performed on the original Q1P41F006A-A valve and is qualified and documented in calculation CC-Q1111-91037. This enhancement will not increase the probability of a system failure.

Serial Number: 1999-070-NPE Document Evaluated: ER 99/0452-00-00

BRIEF DESCRIPTION OF CHANGE, TEST, OR EXPERIMENT:

ER 99/0452-00-00 replaces carbon steel piping and components in selected areas of the Condenser Air Removal System with stainless steel piping and components. The piping modified by this ER is in the continuous drain line between water separator N1N62D009B and nozzle #80 on the L.P. Condenser N1N19B007C. The piping line class is being changed from GBD & HBD to ECD. The existing pipe routing and existing pipe supports are to be utilized for the stainless steel replacement piping.

The USFAR figure identified above will be revised to show the following new piping line nNumbers:

Existing Line #	New Line#
1”-GBD-50	1”-ECD-26
1”-GBD-1145	1” -ECD-26
1”-HBD-551	1”-ECD-25
1”-HBD-1759	1”-ECD-25
1 ½”-HBD-551	1 ½”-ECD-25

REASON FOR CHANGE, TEST, OR EXPERIMENT:

This modification is changing piping and component material to stainless steel for better erosion/corrosion resistance. These small drain lines see two phase steam flow at high velocities during normal operation, which creates the erosive environment.

EVALUATION SUMMARY AND CONCLUSIONS:

Two items of the Flow Accelerated Corrosion (FAC) Program are experiencing significant levels of wear as documented in CR-GGN-1999-1421. The components are located in the 1” & 1 1/2” diameter continuous drain line between Water Separator N1N62D009B and the L. P. Condenser. The stainless steel replacement piping and components will provide better erosion resistance and decrease the probability of a piping failure. The replacement piping and components meet the original design requirements for the Condenser Air Removal System. As indicated in UFSAR Sections 3.2 & 10.4.2 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. A manual search was performed of the UFSAR and no other sections were identified.
Serial Number: 1999-071-NPE Document Evaluated: Standard GGNS MS-48.0

Serial Number: 1999-071-NPE Document Evaluated: CN 99-0096 to ER 99-0285-01-00

BRIEF DESCRIPTION OF CHANGE, TEST, OR EXPERIMENT:

This safety evaluation assesses the reload-related changes associated with Cycle 11 operation as presented in Revision 7 to the Core Operating Limits Report, Mechanical Standard GGNSMS-48.0. Cycle 11 has been designed for 485 Effective Full Power Days with a core consisting of 228 fresh GEl1 fuel bundles, 268 once-burnt GEl1 bundles, 268 twice-burnt GEl1 bundles, and 36 thrice-burnt SPC 9x9-5 bundles. SAR, TS, TS Bases, TRM, and COLR changes are reguired to operate with this new core. Individual design changes on GGNS systems are assessed in the safety evaluation associated with the specific change package and are not addressed in this evaluation. Attachment 1 provides a detailed description of the Cycle 11 reload analysis and the issues considered in this evaluation.

REASON FOR CHANGE, TEST, OR EXPERIMENT:

Cycle 11 operation will reguire new core operating limits and the Core Operating Limits Report has been revised to include these new limits. These limits include flow-, power-, and exposure-dependent LHGR, MAPLHGR, and MCPR limits. Other changes are required in the TRM, SAR, and TS Bases.

EVALUATION SUMMARY AND CONCLUSIONS:

This evaluation concludes that the reload-related changes associated with Cycle 11 operation will not constitute an unreviewed safety question; however NRC approval of the revised MCPR safety limits in GNRO-99/00037 is required for implementation.

Serial Number: 1999-072-NPE Document Evaluated: CN 99-0096 to ER 1999-0285-01-00

BRIEF DESCRIPTION OF CHANGE, TEST, OR EXPERIMENT:

CN 99-0096 to ER 99-0285-01-00 wi11 replace valve Q2P41F006B-B with a blind spool assembly. The basis for this valve in the Standby Service Water (SSW) System is to provide an isolation boundary. The removal of this valve will allow this valve to be put into inventory and reused as a replacement valve in a similar application.

The UFSAR figure 9.2-001 identified valve Q2P41F006B-B on the continuation piping. This figure will be revised to show the valve removed and show one blind flange of the blind spool assembly, which was installed.

REASON FOR CHANGE, TEST, OR EXPERIMENT:

This modification is to replace valve Q2P41F006B-B with a blind spool assembly consisting of two blind flanges and a 20” section of pipe welded on center on the exterior side of each flange to maintain piping system structural integrity. The existing removed Unit 2Q2P41F006B-B valve will be placed in inventory and listed as spare after necessary modification and refurbishment is performed.

EVALUATION SUMMARY AND CONCLUSIONS:

Since all valves have the potential for leakage, a blind spool consisting of two blind flanges and a 20” section of pipe welded on center on the outside of each flange is inherently more reliable than the existing valve. The blind spool assembly meets the original design requirements for the SSW system to provide an isolation boundary and maintain structural integrity of the piping system.

Serial Number: 1999-073-NPE Document Evaluated: ER 99/0388-00-00

BRIEF DESCRIPTION OF CHANGE, TEST, OR EXPERIMENT:

ER 99/0388-00-00 will replace the existing Drywell Purge Compressor Standby Service Water (SSW) Isolation Stop Check Valves. The existing valves are 2-inch, y-pattern stop check valves. The new valves are 2-inch, in-line, spring-and-poppet, soft-seat check valves. Minor piping changes will be made to fit up the new valves.

REASON FOR CHANGE, TEST, OR EXPERIMENT:

As documented in Condition Reports Numbers CR-GGN-l998-0740, CR-GGN-l 998-0878, CRGGN-l999-0526. CR-GGN-1999-0802, CR-GGN-1999-1038, CR-GGN-1999-1040, CR-GGN-1999-1052, CR-GGN-1999-1132, CR-GGN-1999-1177, CR-GGN-1999-1180 and CR-GGN-1999-1279, the Drywell Purge Compressor SSW Isolation Stop Check Valves have repeatedly failed to close promptly during surveillance testing. The failure mechanism appears to be a combination of problems with the valve design configuration, the orientation of the valve in the piping system, the properties of the process fluid and manufacturing imperfections.

EVALUATION SUMMARY AND CONCLUSIONS:

The Safety Evaluation concludes that neither the probability nor the consequences of an accident or malfunction of equipment will be increased by replacing the existing Drywell Purge Compressor SSW Isolation Stop Check Valves with in-line, spring-and-poppet, soft seat check valves. The new valves are designed and fabricated to the same ASME Ill Class 2 code requirements as the existing valves. The stop valve function of the existing valves has been evaluated and determined not to have a safety design basis. Design configuration differences have been evaluated and determined not to create the possibility of an accident or a malfunction of equipment important to safety of a different type than any previously evaluated in the SAR.

Serial Number: 1999-074-NPE Document Evaluated: CR 1999-1684

BRIEF DESCRIPTION OF CHANGE, TEST, OR EXPERIMENT:

The proposed change is the final repair disposition to GGCR1999-1684 and will replace root valve 1N11FX300. The new root valve will be an instrument type valve (1/2" J3-N) installed in the pressure line from the turbine 1st stage header to pressure transmitters C71N052A, C71N052C, 1N11N038A, Cl1N054A and Cl1N054C.

REASON FOR CHANGE, TEST, OR EXPERIMENT:

The final repair disposition to CR 1999/1684 per ER 98/0358-02-00 is to replace the root valve with an instrument valve. Per the generic implications instructions associated with ER 98-0358-01-01 1N11FX300 was to be inspected for cracks. The PT examination performed revealed three cracks at the weld on the upstream side of the root valve. CR 1999-1684 was written to document the flaws in the piping material. The repair instructions for ER 98-0358-02-00 will remove the flawed piping and prevent the recurrence of the nonconformance. The welds made between the nip-o-let and the cap are P5A to P1 which is more compatible material than previously used. The previous repair for this valve had the inaccurate material types considered in the welding process and this is considered to be a contributing factor for the weld failures.

EVALUATION SUMMARY AND CONCLUSIONS:

The replacement of root valve 1N1lFX300 with an instrument valve will reduce the weight attached on the 30 inch steam piping thus reducing the probability of a steam leak from a vibration concern. The tubing will be more flexible than the fittings being replaced therefore the tubing will be less susceptible to failure in this application. Section 15.6.4.2.2 of the UFSAR discusses the potential for and consequences of, a steam leak in the Turbine Building (outside containment). The event analyzed as presented in this section of the UFSAR adequately bounds this potential scenario. There are no new systems or components 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 No. 10.3-001-1 since the Q boundary will be moved to the pipe to tubing interface. The new valve used for 1Nl1FX300 will be an instrument valve and will be included in the Q boundary. The proposed change is entirely contained within the confines of the power block and will not affect or impact the plant's radiological or non-radiological effluents. After a review of the proposed change, it has been concluded that the replacement of root valve 1N11FX300 does not represent an Unreviewed Safety Question and will have no adverse effects on the environment.

Serial Number: 1999-075-NPE Document Evaluated: CN 99-0097 to ER 99-0285-01-00

BRIEF DESCRIPTION OF CHANGE, TEST, OR EXPERIMENT:

CN 99-0097 to ER 99-0285-01-00 will remove valve Q2P41F0l1-C from service along with a section of upstream and downstream piping. The valve downstream SSW “B” Basin Loop “C” return line piping will be removed up to the 24”X10” branch fitting located on SSW Basin “B” LOOP “A” return line (24”-HBC-82) and capped on the 10” side of the branch fitting. The valve upstream SSW “B” Basin LOOP “C” return piping 10”-HBC-84 will be removed back to just inside the basin wall and capped. The basis for this valve in the Standby Service Water (SSW) System is to provide an isolation boundary. The removal of this valve will allow this valve to be put into inventory as a replacement valve in a similar application for Unit 1.

The UFSAR figure 9.2-001 identified valve Q2P41F011-C on the Unit 2 continuation piping. This figure will be revised to show the valve removed and show the installed 10” welded pipe cap.

REASON FOR CHANGE, TEST, OR EXPERIMENT:

This modification is to remove valve Q2P41F0l1-C from service, remove sections of the piping located upstream and downstream of the valve and install welded pipe caps on the remaining upstream and downstream piping to maintain piping system pressure and SSW Basin inventory. The removed Unit 2 Q2P41F0l1-C valve will be placed in inventory and listed as spare after any required Unit 1 modifications and refurbishment is performed.

EVALUATION SUMMARY AND CONCLUSIONS:

Since all valves have the potential for leakage, the removal of a portion of the downstream and upstream piping and installing welded caps on the remaining piping will meet the original design requirements for the SSW system to provide an isolation boundary between the Unit 1 piping and the out of service Unit 2 Piping.

Serial Number: 1999-076-NPE Document Evaluated: CN 99-0094 to ER 96-0882-00-00

BRIEF DESCRIPTION OF CHANGE, TEST, OR EXPERIMENT:

CN 99-0094 to ER 96/0882-00-00 will revise P&ID M-1081A to show air supply regulator N1C11D401. ER 96/0882-00-00 was issued to relocate one air supply regulator for AOVs Q1Cl1Fl80 and Q1Cl1Fl81. SDV Drain and Vent Valves Q1C11F180 and Q1C11F181 are associated with the Scram Discharge Volume portion of the Cl1 System. Each valve currently has an air supply regulator located at the valve (N1C11-0401 for F180 and N1C11-D402 for F181). The regulator locations require exhaust airflow from Q1C11F18O and Q1C11F181 to pass through the regulators in the reverse direction. The valves are spring closed and air opened. It has been determined that in order to improve the valve stroke time and increase the reliability of AOVs Q1C11F18O and Q1C11F181, the air supply regulators need to be relocated upstream of solenoid valve 1Cl1-SV-F182.

P&IDs normally do not show the valve air regulators, however this air regulator will now serve as a common regulator for both the Cl1Fl80 and Fl81 valve, and it is being moved off the valve yoke. In order to provide a better understanding of the new air regulator location and to avoid future confusion the P&ID will be revised. The UFSAR figure 4.6-007 is the piping and instrumentation diagram for the Control Rod Drive Hydraulic System. This figure will be revised to show the new location of air supply regulator N1C11D4O1.

REASON FOR CHANGE, TEST, OR EXPERIMENT:

Diagnostic testing was performed on valves Q1C11F18O and QIC1IF181 to determine the cause of random slow strokes when performing 06-OP-1C11-Q-0009, “Surveillance Procedure Scram Discharge Volume Vent and Drain Valves Operability Test”. The testing showed that there is approximately a 5-second time delay from solenoid actuation to regulated pressure reduction. There is an additional 5-second delay until valve travel commences, and valve travel takes an additional 10 seconds to full close.

The current location of the air regulators requires exhaust airflow from Q1Cl1Fl80 and Q1Cl1Fl81 to pass through the regulators in the reverse direction. The valves are spring closed and air opened. The Conoflow, model GF H60/65, no bleed style regulator being used is dependent on leakage to allow valve closure. The air regulators restrict the exhaust flow and appear to be the predominate factor in Q1Cl1Fl80 and Q1C11F181s close stroke times. The current routing of the air supply tubing and the position of the regulator seems to hinder desired operation of the AOVs. System Engineering requested Design Engineering provide corrective action as applicable. In order to improve the valve stroke time and increase the reliability of AOVs Q1Cl1Fl80 and QlCl1Fl81, the air supply regulators need to be relocated upstream of solenoid valve 1C1l-SV-F182.

EVALUATION SUMMARY AND CONCLUSIONS:

The relocation of the air regulator will improve the valve stroke time and increase reliability of the valves. This modification does not alter the design or function of the SDV Drain and Vent Valves QlCl1F180 and QlCl1Fl81, or the Control Rod Drive Hydraulic System.

Serial Number: 1999-077-NPE Document Evaluated: CN 99-0103 to ER 99/0466-00-01

BRIEF DESCRIPTION OF CHANGE, TEST, OR EXPERIMENT:

ER 99/0466-00-01 replaced valve Q2P41F006A with a blind spool assembly. The basis for this valve in the Standby Service Water (SSW) System is to provide an isolation boundary. The removal of this valve will allow this valve to be reused if required in order to replace valve Q1P41F006A-A that has excessive leakage. If the valve is not utilized as a replacement for the Unit 1 valve it will be entered into the GGNS spare part inventory.

Valve QlP41F006A-A was enhanced prior to plant start up to increase the trunnion bolt holes from 5/8” to 3/4”. The replacement valve (formerly Q2P41F006A) will have the trunnion holes increased from 5/8” to 3/4” holes to match valve QlP41FOO6A-A. This modification will be made prior to entering this valve into GGNS spare part inventory.

Note: The Unit 2 valve is not a like for like replacement until the trunnion bolt holes are modified. The UFSAR figure 9.2-001 identified valve Q2P4lF006A on the continuation piping. This figure will be revised to show the valve removed and show one blind flange of the blind spool assembly, which was installed.

REASON FOR CHANGE, TEST, OR EXPERIMENT:

Q1P41F006A-A valve was reported to have excessive leakage across the seat under CR 1999-1608 and may require replacement. Unit 2 valve Q2P41 F006A-A is used only for isolation and is an acceptable like for like replacement, with the exception of the trunnion bolt hole size, for the Unit 1 valve if required. This modification is replacing valve Q2P41 FOO6A -A with a blind spool assembly consisting of two blind flanges and a 20” section of pipe welded on center on the outside of each flange to maintain piping system structural integrity. The existing Unit 2 Q2P41F006A valve if required will be reused to replace Unit 1 valve QlP41 F006A or returned to GGNS spare part inventory.

EVALUATION SUMMARY AND CONCLUSIONS:

Since all valves have the potential for leakage, a blind spool consisting of two blind flanges and a 20” section of pipe welded on center on the outside of each flange is inherently more reliable than the existing valve. The blind spool assembly meets the original design requirements for the SSW system to provide an isolation boundary. Valve Q2P41F006A-A will be structurally enhanced by increasing the trunnion boltholes from 5/8” to 3/4” prior to being installed as the Unit 1 valve if required or prior to becoming a spare part in the GGNS spares inventory. This is the same enhancement that was performed on the original Q1P4IF006A-A valve and is qualified and documented in calculation CC-Q1111-91037. This enhancement will not increase the probability of a system failure.

Serial Number: 1999-078-NPE Document Evaluated: ER 99/0545-00-00

BRIEF DESCRIPTION OF CHANGE, TEST, OR EXPERIMENT:

Alternate methods of monitoring the drywell floor drain sump in-leakage are required. The preferred alternative will utilize the existing computer point P450001. This computer point monitors the starter auxiliary contacts of the sump pumps. A new computer point P45 3001 will be created to trend the sump in-leakage. The trend values will be calculated based on the sump fill times. A new digital point P45 4001 will be created to alarm if the leakage indicated by P453001 exceeds 4.2 gpm. This will be added as an input to the LDS trouble annunciator 1E31L609.

If PDS fails, the sump fill times (time between pump stop and pump start) may be monitored by any approved M&TE (a stop watch or recorder). The in-leakage can then be calculated. A recorder may be temporarily installed per MAI to monitor the status of the sump pump starter auxiliary contacts by measuring the voltage across the coil of relay 1E31KOl6. This relay coil is energized by the starter auxiliary contacts if either pump is started.

R.G. 1.45 regulatory position 5 requires a system sensitivity and response to detect a 1 gpm leak in less than one hour. The FSAR was changed to take exception to this requirement per FSAR CR 94-025 (GIN 94/3251) and safety evaluation 94-080-ROO based on the assumption that an alarm was required for a 1gpm leakage increase per QOR 92-0282. This exception may not have been necessary for the level transmitter loop since the sensitivity and response of this loop is capable of detecting a 1 gpm leak in less than 1 hour. RG 1.45 does not require an alarm for a 1 gpm leakage increase.

The alternate methods of monitoring sump in-leakage will have the sensitivity to detect a 1 gpm leakage increase. The response time will be slower with low sump in-leakage because of longer fill times. The worst case response time to a 1 gpm increase (including the delay introduced by the computer) will be less than 3 hrs with zero initial leakage. This response time is acceptable per the existing exception to RG 1.45. Either the primary or alernate method will alert the operators of a 1 gpm leak increase within the 12 hour surveillance

REASON FOR CHANGE, TEST, OR EXPERIMENT:

These changes are being made to allow more flexibility within LCO 3.4.7 and to clarify the GGNS License Basis. The changes to the Bases for LCO 3.4.7 allow for an additional means of determining leak detection while still meeting the full intent of the LCO.

GGNS Technical Specification 3.4.7 allows the drywell floor drain sump monitoring system to be inoperable for 30 days. Technical Specification 3.4.5 requires reactor coolant system leakage be verified within limits once per 12 hours (SR 3.4.5.1). The leakage must be less than 5 gpm with less than a 2 gpm increase in 24 hours. Until now the leakage and change in leakage was determined by looking at the trend of drywell floor drain sump level on recorder 1E31R618 and determining the sump fill rate. The Barton level transmitter 1E31N093 that monitors sump level and provides input to this recorder has apparently failed (reference CR 99/1957). This transmitter also provides input to I E31 K606 which converts the change in the level signal to a gpm value that is also recorded on 1 E31 R61 8 and monitored by flow switch 1 E31 N693. If the 4.2 gpm setpoint of 1E31N693 is exceeded the annunciator 1E31L625 is activated. Operations has requested an alternate method to determine RCS leakage when 1E31N093 is not available.

The drywell floor drain sump level is also monitored by the FCI level sensor 1P45N217 which provides input to the FCI level switches 1P45N223 & N224 and FCI level sensor 1P45N218 which provides input to FCI level switch 1P45N225. The high level switch 1P45N223 starts one pump when the sump level reaches approximately 20”. If the level continues to rise, the high-high level switch 1P45N225 starts the

second pump at approximately 24” and activates the high level annunciator 1P45L617. The low level switch 1P45N224 stops the pump(s) when the sump level drops to approximately 10”.

The pump down and fill times are monitored by Eagle timers 1E31R603 & 604. If the time limits are exceeded the in-leakage is excessive and the annunciator 1E31L627 is activated. The current technical specification basis implies that the timers are the instruments credited for RG 1.45 monitoring of the drywell floor drain sump level. They can not be used for this purpose because a trend is needed to determine the change in leak rate. The tech. spec. bases are being updated to provide additional clarification.

EVALUATION SUMMARY AND CONCLUSIONS:

The drywell floor drain monitoring instrumentation is not safety related, is not required by RG *1.97* and is not credited for any accident mitigating functions. The changes being made to the UFSAR and the Bases in this LDC do not affect the initiation of any accident described in the SAR. The intent of LCO 3.4.7 is still maintained. This intent is that a valid drywell floor drain sump monitoring system is available to determine RCS leakage. Both the level indications supplied from the drywell floor drain sump transmitter and the floor drain sump level switches and associated instrumentation are capable of providing this Technical Specification function. Since the intent of this LCO is adhered to and the intended function is not changed, nuclear safety and safety system performance will remain unaffected by these changes. Since these Technical Specification Bases and UFSAR changes do not change any function, but rather provide an additional means of maintaining that Technical Specification function, these changes do not affect the initiation of any accidents previously evaluated in the SAR.

These changes do not affect the radiological consequences of any accident, do not affect any fission product barriers, and do not alter any assumptions previously made in evaluating the radiological consequences of an accident described in the SAR, because the changes only provide an alternate method for maintaining the already established Technical Specification function of determining RCS leakage. Additionally, the consequences of an accident cannot be increased by this change, because the drywell floor drain monitoring instrumentation is not safety related, is not required by RG 1.97 and is not credited for any accident mitigating functions. The resulting accident is a line break which is already bounded by existing analyses within the UFSAR.

The leakage detection system is used as a means for determining potential RCPB degradation before the integrity of the RCPB is significantly impaired. The subsequent accident would be a breach of the RCPB (a LOCA). The changes being made do not add, change, or delete any physical components in the plant. The changes only add an additional means for determining the RCS leakage. Consequently a new type of accident as previously identified cannot possibly be created and the changes are already bounded by existing UFSAR analyses.

This change does not affect the Technical Specifications, but rather, they are changes to the Technical Specification Bases and UFSAR. The changes allow the use of alternate instrumentation for the purposes of determining drywell leakage. This alternate method will be performed in accordance with the Technical Specifications and will meet the full intent of LCO 3.4.7.

Serial Number: 1999-079-NPE Document Evaluated: CR 1998-1269

BRIEF DESCRIPTION OF CHANGE, TEST, OR EXPERIMENT:

Bailey INFI-90 firmware modules used for control and communication functions. IMMFP01, IMMFP02, IMMFP03, INNIS01, INICT01, and INNPM01, are being upgraded to Year 2000 (Y2K) compliant revision levels. These modules are located in the 1H22P171, 1H22P172, 1H13P612, 1N21P001A, and lN21P00lB panels.

REASON FOR CHANGE, TEST, OR EXPERIMENT:

GGCR 1998-1269-00 documents that the Bailey INFI-90 control and communication firmware modules installed in 1H13-P612, 1H22-P171, 1H22-P172, 1N21-P00lA, and 1N21-P001B are Year 2000 (Y2K) indeterminate. ER 98/0642-00-01 approves Y2K compliant upgrades for these firmware modules.

SAFETY EVALUATION SUMMARY AND CONCLUSIONS:

Obsolete Bailey Controls Series 7000 analog control equipment was replaced with INFI-90 digital control equipment per the following:

DCP 91/0088-1, DCP 91/0088-2, and DCP 91/0088-5. DCP 91/0088-1 upgraded panels 1H22P171 and 1H22P172 with digital controls. DCP 91/0088-2 upgraded the Reactor Feed Pump Turbines with digital controls and DCP 91/0088-5 approved the Feedwater Control System Digital Upgrade. Safety Evaluations 93-0073-R01, 93-0062-R00, and 95-0013-R00 were performed for these DCPs respectively. The limiting conditions evaluated per these approved safety evaluations are not impacted by the Y2K firmware upgrade approved per ER 98/0642-00-01. Documentation of firmware revision level changes was reviewed and the upgraded firmware revision levels are determined to be functionally equivalent to the existing modules. Due to this functional equivalency, all previously analyzed limiting conditions are still bound and no new limiting conditions are created. The INFI-90 distributed control system at Grand Gulf Nuclear Station (GGNS) is non-safety related and not required for safe shutdown of the plant. Neither the GGNS Technical Specifications, the Technical Specification Bases, or the TRM address this equipment; therefore, no changes are required. The firmware upgrade will not increase the probability of occurrence of accidents or malfunction of equipment important to safety previously evaluated in the SAR because equipment performance remains within the limits currently assumed in the existing analyses. This change will also 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 because limiting system failure modes are unchanged. This change will not reduce the margin of safety as defined in the basis for Technical Specifications since limiting and non-limiting events, which may affect fission product barriers, remain clearly bounded by existing analyses.

Serial Number: 1999-080-NPE Document Evaluated: ER 97/0732-00-00

BRIEF DESCRIPTION OF CHANGE, TEST, OR EXPERIMENT:

ER 97/0732-00-00 adds to plant, on a permanent basis, the free standing air-conditioning units located in the elevator machinery room (El 186’3”Turbine Building) and the radio repeater room (El 186’3” Turbine Building) as well as the floor mounted fan located in the elevator equipment room and their associated construction water piping, power supplies and drain lines. The air-conditioning units are free standing commercial grade water-cooled condensing units, the fan is constructed of welded angle sized to hold filters and a fan and a backdraft damper. The cooling water to the units is supplied from construction water and returned to the existing unit 2 drainage system, the condensate drain lines are copper tubing that join together and become a non-metallic hose that is routed to a drain. The power is supplied by extension cords. The existing penetration that housed the window air conditioning unit that has been removed is closed using plywood.

REASON FOR CHANGE, TEST, OR EXPERIMENT:

GGCR 1997-0583 documents that the window air conditioner for the elevator machinery room located on El 186’3” of the turbine building is not present and the penetration is closed with plywood. It also states that free standing air conditioning units are located in the elevator equipment room and the radio repeater room.

SAFETY EVALUATION SUMMARY AND CONCLUSIONS:

The addition of the air conditioning units and fan units and their associated construction water piping, drain lines and power cords is acceptable. There is no safety-related equipment located in these areas that will be affected by the failure of any of this equipment. There are no combustible heat load calculations that will be affected by the presence of the plywood, non-metallic hose and power cords. The other indirect effects (missiles from the rotating components, flooding from the water lines, etc.) have been reviewed and no increased probability of failure of equipment important to safety due to those concerns were identified. The addition of these components will not degrade any system, structure or component important to safety nor will they degrade or prevent actions described in the SAR accident analysis. These modifications do not result in a new pathway for release of radioactive material and do not affect offsite dose. No assumptions utilized in evaluating the consequences of an accident are altered. The changes do not affect equipment important to safety and will not cause any systems or components important to safety to be operated outside design limits. Failure of the fan, air-conditioning units, construction water piping, drain lines and power cords will not compromise any safety related system or component and will not prevent safe reactor shutdown. The changes do not increase the probability of occurrence or increase the consequences of malfunction of equipment important to safety or of a different type than previously evaluated in the SAR. The Technical Specifications are not affected and the margin of safety is not reduced. It is concluded that the modifications made by this ER are acceptable.

Serial Number: 1999-081-NPE Document Evaluated: ER 98/0583-00-00

BRIEF DESCRIPTION OF CHANGE, TEST, OR EXPERIMENT:

This modification will alter the trip logic/circuitry for Turbine Building Fan Coil Units, N1U41B004 & N1U41B005, such that its fan will stop if its associated room's CO_2 fire deluge panel initiates on automatic or manual actuation. The CO_2 panel design logic is based on automatic actuation via rate-of-rise heat detectors in the area of protection or a manual initiation. The Electro Thermal Links (ETL) for the associated rooms' HVAC fire rated dampers have an existing time delay of 5 seconds before actuation by their associated CO_2 panel. This time delay will be slightly increased to allow the associated Turbine Building Fan Coil Unit to come to a complete stop before they are actuated. This time delay will be coordinated with the CO_2 panel "dump" of suppression gas in the affected room.

REASON FOR CHANGE, TEST, OR EXPERIMENT:

GGCR1997-0726-00 (MNCR 96/0047) identified a non-conforming condition that could degrade the ability of several CO_2 fire suppression systems from performing their function by not permitting the required concentration of extinguishing agent in their respective area of protection due to HVAC fire dampers installed in their area's associated ventilation systems not being designed to close under air flow conditions. This CR listed, among other areas of the plant, the Turbine Building BOP Switchgear rooms 1T219 & 1T323. For a gaseous fire suppression system to be totally effective, an adequate concentration of the extinguishing agent must be contained in the protected area. Therefore, fire dampers associated with these rooms must completely close to ensure adequate concentration of extinguishing agent is contained in the area.

Currently, Turbine Building room 1T219 (El. 113', Area 4) contains BOP Switchgear 11HD & 13AD, is provided HVAC via N1U41B004-N and is protected by fire suppression system N1P64D203. Additionally, Turbine Building room 1T323 (El. 133', Area 4) contains BOP Switchgear 12HE & 14AE, is provided HVAC via N1U41B005-N and is protected by fire suppression system N1P64D205. As presently designed, fire suppression system N1P64D203 has no interface with associated room HVAC unit N1U41 B004-N and fire suppression system N1P64D205 has no interface with associated room HVAC unit N1U41B005-N to automatically stop the fan to permit the associated fire dampers to close under no flow conditions.

SAFETY EVALUATION SUMMARY AND CONCLUSIONS:

This modification will not affect the GGNS Technical Specifications. Nor will it create an unreviewed safety question or reduce any margin of safety. These CO_2 panels are not listed in TRM Section 6.2.4 nor should they be added. These affected areas are not listed in TRM Section 6.7.3 and no temperature limit is associated with either of these two areas which requires the equipment within these rooms to be considered inoperable should the fan be stopped via the CO_2 panel. This modification maintains requirements of Reg. Guide 1.75 for required electrical separation and isolation between Class 1E and Non-Class 1E and between Class 1E divisions. This modification occurs in a Non-Seismic Category building (Turbine Building) and affects non-safety related equipment that can not adversely affect the operation of any equipment important to safety. Supports for all raceway components and boxes are installed per standard design guidance and are not required to be seismically rugged. FSAR Figures 9.4-006 and 9.5-005 require revision to reflect signal input from the CO_2 panels to their associated Turbine Building Fan Coil Unit.

Serial Number: 1999-082-NPE Document Evaluated: ER 97/0823-00-00

BRIEF DESCRIPTION OF CHANGE, TEST, OR EXPERIMENT:

This ER (ER 97/0823-00-00) is part of an overall design objective to install new pumps and motors for the Plant Service Water (PSW) Radial Well System (P47 System). The new pumps and motors will be used as replacements for the existing pumps and motors for Radial Wells #4. The new stainless steel pumps will be oil lubricated enclosed line-shaft vertical pumps rated at 5,000 gpm and a discharge head of 360 ft of water. The existing 500 hp pump motors will be replaced with new 600 hp motors. As part of the modification a lubricant reservoir and associated equipment will be installed to provide lubrication for the enclosed line shafts.

Specifically, this ER (ER 97/0823-00-00) replaces the pumps and motors in Radial Well #4 (pumps NSP47C00lJ & K). This ER can be worked during any operational condition as long as sufficient Plant Service Water capacity is available to support the plant condition which exists during the period of implementation, which is in agreement with UFSAR Section 9.2.10.2, System Description, which states: "During normal operation, as many wells and pumps as required will be operating to meet the plant demand."

The new pumps will mount in the existing pump support structure and will connect to the existing piping with no major piping modifications required. Pump lubrication, using biodegradable oil, will be supplied by gravity from 140 gallon tanks (M-929.0-NS-l.l-3-0) mounted in the individual well houses. The tanks are constructed to the standard requirements of the manufacturer. There will be an interlock between the lube oil tank and pump trip circuit to trip the pump on low-low lube oil tank level. The equipment is being procured in accordance with Specification GGNS-M-929.0.

LDC 98-006 makes the following changes:

UFSAR Section 2.4.13.1.3.1 - The total dynamic head of the new Radial Well pumps is now 360 ft at 5000 gpm, however, the total dynamic head of the pumps is not germane to the discussion in this section of the UFSAR, therefore, this information is being deleted. This same information is contained in Table 9.2-13.

UFSAR Sections 3C.4 and 3C.4.2.1 - These sections, which address compartment flooding, are being revised to reflect the new leakage flow and accumulation rate postulated for the 36" plant service water line in the auxiliary building as a result of the increased operating pressure that may be encountered due to the new pumps installed.

UFSAR Table 9.2-13 - The information for the PSW pumps and lube oil system in Table 9.2-13 is being revised to reflect the new equipment for Radial Well #4.

UFSAR Figure 9.2-27, Sheet 2 - The base drawing, M-0052B, for this UFSAR Figure was revised to show the installation of the new Radial Well #4 pumps and associated equipment, therefore, the UFSAR Figure was revised accordingly. (Note that this drawing is P&ID M-0052B and will be updated automatically by Configuration Management

REASON FOR CHANGE, TEST, OR EXPERIMENT:

There has been an apparent decrease in efficiency and reliability for the PSW Radial Well #3 and #5 pumps caused by extensive pump column wear, impeller obsolescence, well #3 and #5 pump prelube abandonment, and system operational characteristics. The well #4 pumps and motors, which are a different design than wells #3 and #5, have not experienced the problems experienced by wells #3 and #5, however, they are being replaced so they will be similar to wells #3 and #5 in capacity and functionality.

SAFETY EVALUATION SUMMARY AND CONCLUSIONS:

The intended purpose for the installation of the new pumps and motors is to increase the reliability, capability and availability of the PSW system. This change does not degrade below the current design basis 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 cause a greater reliance to be placed on any specific system, structure or component to perform a safety function. A failure of one of the new PSW Radial Well pumps or motors will have the same result as the failure of the existing PSW Radial Well pumps and motors. The loss of PSW and flooding as a result of a break in the PSW header in the Auxiliary Building have been analyzed in the UFSAR. The installation or failure of the new pumps or motors will not increase the probability or consequences of these analyzed failures. No increase of either the offsite or the onsite radiation dose would result because of a failure of a new pump or motor.

When the new pumps have been installed, the capability will exist to increase the operating pressure of the PSW system. This increase in operating pressure will increase the leakage rate and water accumulation rate in the Auxiliary Building in the case of a postulated moderate energy crack in the 36” PSW line. GGCRl998-0701-00 documented that the original leakage and accumulation rate used to determine the values contained in UFSAR sections 3C.4 and 3C.4.2.1 were based on a pressure of 75 psig which is non-conservative even for the existing PSW Radial Well pumps. LDC 98-006 revises the leakage and accumulation rates currently in UFSAR section 3C.4 and 3C.4.2.1 to reflect a maximum PSW system pressure of 120 psig for this 36” PSW line based on the results of calculation 195.0-1. This is conservative in that calculation MC-N1P47-97039 determined that the normal operating pressure will be 103 psig after the installation of the new PSW Radial Well pumps. As stated in condition report GGCR1998-0701-00, the additional leakage due to a maximum operating pressure of 120 psig of the PSW system would still be acceptable based on current methodology and reasoning used in the GGNS UFSAR since the increase in postulated leakage is not substantial enough to prevent detection and isolation of the leak in a timely manner. There would be no increase in probability of a leak or the consequences of a leak as a result of this ER, since the system will still be operated within the constraints of the system design requirements. The flooding of the control building is bounded by the circulating water line break (Note *5* of drawing M-1575, Rev. 0) and the flooding in RHR Room C is bounded by the RHR Pump C suction line (see UFSAR Section 3C.4.2.6).

The installation of the new pumps and motors has been analyzed for its impact on the Radial Well Pumphouse HVAC system’s effectiveness and found not to be adversely impacted. Because the new motors are more efficient than the existing motors, the heat load in the pumphouse will be less when the new pump motors are in operation than with the existing motors.

An evaluation was performed to determine the environmental impact of the addition of the new lube oil systems added to support the new Radial Well pumps. The evaluation concluded that the design of the new PSW pumps and the associated lube-oil system minimizes any potentially negative environmental impacts. The lube-oil reservoir, piping, and tubing are designed to preclude any direct leakage to the environment, and the pump design restricts all but incidental oil migration to the pump caisson rather than the pump effluent and downstream system piping. The specified oil is “environmental friendly” such that any trace amounts that might ultimately reach the plant piping or the river will bio-degrade without impact to the environment. The chemistry department performed a control room habitability screen for this lubricating oil per procedure 01-S-08-18, GGNS Chemical Control Program, and documented its acceptability.

This change does not adversely affect the overall performance or reliability of a safety system in a manner that could lead to an accident occurring. This change does not cause a safety system to be operated outside of its design basis limits. The new PSW Radial Well pumps and motors cannot affect any system interface in a way that could lead to an accident. The new PSW Radial Well pumps and motors will not result in degradation of safety systems. The change is intended to improve the reliability, capability and availability of the PSW system by providing a mechanism to reduce the possibility of system unavailability. Additionally, the margin of safety as defined in the bases for the Technical Specifications has not been reduced.

Serial Number: 1999-083-NPE Document Evaluated: ER 97/0900-00-00

BRIEF DESCRIPTION OF CHANGE, TEST, OR EXPERIMENT:

The proposed change will replace the original ten-disc (five discs per flow, 44-inch last-stage blades) low pressure (LP) No. 3 turbine rotor with an eight-disc (four discs per flow, 46-inch laststage blades) advanced design turbine rotor and associated components to provide efficiency improvements and reduced maintenance requirements which result in cost savings. Also, the first two discs per flow of the original LP No. 3 turbine rotor are combined into a single disc for the upgrade LP No. 3 turbine rotor. The upgrade rotor discs are made from 3.5% NiCrMoV steel material. This design improvement will increase disc inspection interval up to 100,000 equivalent operating hours. The change is limited to the replacement of the turbine rotor, the inner - inner casing, the stationary blade rings. the diffusers and associated components. Also, the proposed change will increase the nNumber of bolts used in LP shaft seal compensator joints (38 in lieu of 20) and replace the original gasket with a thicker gasket. Minor modification of the coupling on generator rotor (turbine end) will also be conducted to ensure adequate clamping force is provided for the upgraded turbine configuration.

REASON FOR CHANGE, TEST OR EXPERIMENT:

The design of the 1970 vintage nuclear LP steam turbine was based on extensive experience gained with disc-type rotors of fossil turbines built in the 1950's. In the meantime. Siemens and Kraftwerk Union (KWU) began manufacturing turbines that improved thermal performance while maintaining and enhancing the already high degree of reliability and availability of their turbines. Siemens Power Corporation (SPC) will supply and install the LP No. 3 turbine upgrade components to increase the efficiency of the turbine and to increase the interval between inspections. The proposed change will replace the original LP No. 3 rotor with an advanced design rotor to provide additional electrical megawatts for the same reactor thermal output (increase efficiency of the turbine-generator). The increased turbine efficiency is due to a more efficient integrally shrouded T4 blade profile, more twisted blading to better utilize the steam flow velocity, an advanced free standing blade section, one additional stage and more efficient inter-stage seal strips. The efficiency improvements convert more thermal energy into mechanical energy, rather than being lost to friction, steam velocity changes, and blade bypass. In order to minimize air inleakage in the shaft seal compensator joints, additional bolts will be added and the original gasket design will be replaced with a thicker gasket design. The additional bolts are required to achieve the required gasket compression.

Also, the original generator rotor coupling (Turbine End) bolt holes were bored before shrinking the coupling on the rotor. Subsequent to the shrinking process, the shape of the prebored coupling bolt holes changed. In order to resolve this discrepancy, the original coupling bolt holes were bored to a larger size to correct the hole shape. Modifications to the generator rotor coupling will be made to ensure that proper bolt and nut pressure contact surfaces are maintained between upgraded LP No. 3 and original generator rotor. Due to this concern, a sleeve will be installed inside each coupling bolt hole of the generator rotor coupling (turbine end).

SAFETY EVALUATION SUMMARY AND CONCLUSIONS:

The LP No. 3 turbine (N31D002C) is a part of the turbine-generator (N31 system). The proposed change will affect turbine-generator design parameters listed in UFSAR Sections 1.3.10.1.10.2 and 10.4.7. The proposed change will also update information provided in UFSAR Section 3.5.1.3. The change will affect the design information provided for turbine cycle heat balances by increasing generator output as shown on UFSAR Figures 10.1-1 & 10.1-2. However, the change will have no significant affect on interfacing UFSAR Sections for main and reheat steam (N1l). heater vents & drains (N23), main and R.F.P. turbine seal steam and drains (N33), moisture separator-reheater vents & drains (N35), extraction steam (N36) and turbine bypass (N37) systems. UFSAR Section 3.2 classifies the affected systems (N1l, N19, N21, N23, N33. N35, N36 and N37) as "Other" which means that a loss of system function would

not affect the safe shutdown of the plant. UFSAR Table 3.2-I classifies these systems and their associated components as non-safety related, non-seismic, quality group D and ANSI B31.1. The proposed change will replace the original LP No.3 rotor, inner - inner casing and associated components with an improved design that will increase the efficiency of the LP No.3 turbine. No other changes to the original LP No.3 turbine configuration will be required. The upgraded components supplied by SPC have been designed in accordance with the German standards used to construct the original turbines.

SPC has submitted a missile analysis report for the upgraded LP rotors (FS 4/1018/1995 and 98044J). The report compares disc 1 and disc 4 (end stage disc) of the upgraded LP rotor with disc 5 (end stage disc) of the original LP rotors. The report indicates that disc 1 of the upgraded LP rotor has additional mass, lower average temperature and an additional row of blading. Also, this report shows that the fragment with the maximum translation energy is considered to be the most dangerous since it is subject to the minimum loss due to friction and hence the translation energy is the deciding criterion for the penetration of safety barriers. As stated in the report, the translation energy for disc 1 of the upgrade LP rotor is 5.7×10^6 Joules which is lower than the translation energy (10.8×10^6 Joules) of disc 5 of the original LP rotor. The results show that each of the upgrade LP rotors can be operated for 12 years or 100,000 hours between inspections and the external missile probabilities are well below design requirements. Also. the burst probabilities are within SPC allowable values. Therefore. the LP turbine missile analysis addressed in UFSAR section 3.5.1.3 is not affected by the LP turbine upgrade. The turbine stop & control valve parameters and overspeed protection function are not affected by this modification and therefore do not represent a change to the Technical Specifications or LP turbine missiles analysis (UFSAR Section 3.5.1.3). Also, the function of the generator rotor coupling (Turbine End) is to transfer torque from the HP turbine, LP No.1, LP No. 2 & LP No. 3 turbine rotors to the generator rotor and to withstand maximum short-circuit torque (without major coupling bolt damage) is not affected by this design modification. SPC has submitted a design report (DG 96/006) which evaluates the coupling of the original LP rotors and the upgraded LP rotors. The results of the evaluation show that the coupled Turbine Generator rotor system torsional natural frequencies are free from excitable torsional frequencies in the range of 57 Hz to 63 Hz and / or 114 Hz to 126 Hz. Also, the proposed LP No. 3 turbine upgrade components have excellent erosion corrosion (EC) resistant material properties. The design has been evaluated against the applicable design criteria, installation and operational requirements, and all necessary requirements and commitments are met. The change will not affect any equipment important to safety. The modifications made by this design change will not impose a chance to the criteria listed in UFSAR Table 3.2-1. Based on information provided by SPC for the LP No. 1 turbine, this ER will not affect any parameters specified in the cycle11i reload safety analysis (i.e. HP turbine first stage pressure, HP control valve positions, etc.). Other HP & LP turbine parameters, such as extraction steam pressures, will remain within the ranges specified for the original turbine design. The modification will enhance the turbine efficiency without affecting the operation of the reactor pressure control system. Therefore, there is no increase in probability of occurrence an accident or increase in consequences of an accident or malfunction of any equipment important to safety by this change.

Serial Number: 1999-084-NPE Document Evaluated: ER 99/0540-00-00

BRIEF DESCRIPTION OF CHANGE, TEST, OR EXPERIMENT:

The proposed change will remove two instrument tap root isolation valves (1N11FX300 and 1N11FX301) from the Main Steam (N11) System. A section of stainless steel tubing will be installed to replace the root valves and re-establish continuity of the instrument lines. Tubing compression fittings at the first pipe interface will be replaced with a welded connection. The individual instruments served by lNllFX300 and INllFX30l have individual isolation valves located near the instruments, thus removal of the instrument tap root valves will not prevent isolation of the affected instruments. The overall routing and function of the affected instrument tubing, or the associated instruments, will not be altered by the proposed change.

REASON FOR CHANGE, TEST, OR EXPERIMENT:

The instrument root valves (lNllFX300 and lNllFX30l) being removed by the proposed change are connected to Main Steam Lines "A" and "B" and are used as isolation valves for instrument sensing lines. The sensing lines are used for several instruments including QlC71N052A/ B/C/D and QlC11N054A/B/C/D, which provide Main Steam Line pressure signals (Turbine 1st stage pressure —low) for the Reactor Protection System and Control Rod Drive System and various other non-safety related instruments. To eliminate the potential source of steam leaks and minimize the associated operational challenges, the proposed change removes these root valves from the N11 System. There are no tests or experiments associated with the proposed change.

EVALUATION SUMMARY AND CONCLUSIONS:

The removal of the root valves lNllFX300 and lNllFX30l is an exemption from the B31.l code Section 122.3.3, however it still meets the licensing basis for safety-related equipment, which are installed in the Turbine Building. Safety-related equipment installed in the Turbine Building is designed such that any failure will initiate the associated safety related trip functions. The removal of root valves lN11FX300 and 1N11FX301 from the associated instrument sensing lines will eliminate the possibility of locally isolating the associated instrument lines in the event of a tubing leak or rupture. Thus, should a steam leak occur due to a leak or rupture of the associated instrument tubing, other methods will be required to isolate the steam leak. In extreme conditions, the Main Steam Line Isolation Valves could be closed to depressurize and isolate the postulated leak. Section 15.6.4.2.2 of the UFSAR discusses the potential for, and consequences of, a large steam line break outside containment. The event analyzed and presented in this section of the UFSAR adequately bounds the potential scenarios that may result from the proposed change. There are no new systems or components added by the proposed change, thus the existing accident scenarios and analyses presented in the UFSAR will not be adversely impacted. The proposed change will affect UFSAR Figure No. 10.3-001-1 since lNllFX300 and lNllFX30l are currently depicted on this UFSAR Figure. However, removal of instrument root valves 1N11FX300 and 1N11FX301 does not represent an Unreviewed Safety Question and will have no adverse affects on the environment.

The GGNS Technical Specifications do not address root valves 1N11FX300 and 1N11FX301 or the associated instrument tubing lines. However the instruments served by these sensing lines are associated with the Reactor Protection System (RPS) and Control Rod Drive System, which are discussed in Sections 3.3 and 3.10, respectively, in the GGNS Technical Specifications. The function and availability of the associated RPS instruments will not be altered by the proposed change. Thus the proposed change will not result in the need to change or revise the GGNS Technical Specifications or Technical Requirements Manual.

Serial Number: 1999-085-NPE Document Evaluated: ER 1997-0822-00-00

BRIEF DESCRIPTION OF CHANGE, TEST, OR EXPERIMENT:

This ER (ER 97/0822-00-00) is part of an overall design objective to install new pumps and motors for the Plant Service Water (PSW) Radial Well System (P47 System). The new pumps and motors will be used as replacements for the existing pumps and motors for Radial Wells #5. The new stainless steel pumps (existing well #5 pumps are carbon steel) will be oil lubricated enclosed line-shaft vertical pumps rated at 5,000 gpm and a discharge head of 360 ft of water. The existing 500 hp pump motors will be replaced with new 600 hp motors. As part of the modification a lubricant reservoir and associated equipment will be installed to provide lubrication for the enclosed line shafts.

Specifically, this ER (ER 97/0822-00-00) replaces the pumps and motors in Radial Well #5 (pumps NSP47C001E & F). This ER can be worked during any operational condition as long as sufficient Plant Service Water capacity is available to support the plant condition which exists during the period of implementation, which is in agreement with UFSAR Section 9.2.10.2, System Description, which states: "During normal operation, as many wells and pumps as required will be operating to meet the plant demand."

The new pumps will mount in the existing pump support structure and will connect to the existing piping with no major piping modifications required. Pump lubrication, using biodegradable oil, will be supplied by gravity from 140 gallon tanks (M-929.0-NS-l.1-3-0) mounted in the individual well houses. The tanks are constructed to the standard requirements of the manufacturer. There will be an interlock between the lube oil tank and pump trip circuit to trip the pump on low-low lube oil tank level. The equipment is being procured in accordance with Specification GGNS-M-929.0.

LDC 98-005 makes the following changes:

UFSAR Section 2.4.13.1.3.1 - The total dynamic head of the new Radial Well pumps is now 360 ft at 5000 gpm, however, the total dynamic head of the pumps is not germane to the discussion in this section of the UFSAR, therefore, this information is being deleted. This same information is contained in Table 9.2-13.

UFSAR Table 9.2-13 - The information for the PSW pumps and lube oil system in Table 9.2-13 is being revised to reflect the new equipment for Radial Well #5.

UFSAR Figure 9.2-27, Sheet 1 - The base drawing, M-0052A, for this UFSAR Figure was revised to show the installation of the new Radial Well #5 pumps and associated equipment, therefore, the UFSAR Figure was revised accordingly. (Note that this drawing is P&ID M-0052A and will be updated automatically by Configuration Management.

REASON FOR CHANGE. TEST. OR EXPERIMENT:

There has been an apparent decrease in efficiency and reliability for the PSW Radial Well #5 pumps caused by extensive pump column wear, impeller obsolescence, well #5 pump prelube abandonment, and system operational characteristics.

SAFETY~EVALUATIONSUMMARYAND CONCLUSIONS:

The intended purpose for the installation of the new pumps and motors is to increase the reliability, capability and availability of the PSW system. This change does not degrade below the current design basis 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 cause a greater reliance to be placed on any specific system, structure or component to perform a safety function. A failure of one of the new PSW Radial Well pumps or motors will have the same result as the failure of the existing PSW Radial Well pumps and motors. The loss of PSW and flooding as a result of a break in the PSW header in the Auxiliary Building have been analyzed in the UFSAR. The installation or failure of the new pumps or motors will not increase the probability or consequences of these analyzed failures. No increase of either the offsite or the onsite radiation dose would result because of a failure of a new pump or motor.

When the new pumps have been installed, the capability will exist to increase the operating pressure of the PSW system. This ER does not authorize changes to the operating pressure of the PSW system as described in system operating instruction 04-l-01-P44-1. Such an increase in operating pressure would adversely affect the leakage rate and water accumulation rate in the Auxiliary Building in the case of a postulated moderate energy crack in the 36” PSW line. GGCRl998-0701-00 will correct the leakage and accumulation rates currently in UFSAR section 3C.4 and 3C.4.2.l to reflect the normal PSW system pressure for this 36” PSW line. This condition report documents that the additional leakage due to the normal operating pressure of the PSW system would still be acceptable based on current methodology and reasoning used in the GGNS UFSAR. There would be no increase in probability of a leak or the consequences of a leak as a result of this ER, since the system will still be operated within the constraints of the system operating instruction and the operability evaluation for GGCRl998-0701-00. The flooding of the control building is bounded by the circulating water line break (Note 5 of drawing M-1575, Rev. 0) and the flooding in RHR Room C is bounded by the RHR Pump C suction line (see UFSAR Section 3C.4.2.6).

The installation of the new pumps and motors has been analyzed for its impact on the Radial Well Pumphouse HVAC system’s effectiveness and found not to be adversely impacted. Because the new motors are more efficient than the existing motors, the heat load in the pumphouse will be less when the new pump motors are in operation than with the existing motors.

An evaluation was performed to determine the environmental impact of the addition of the new lube oil systems added to support the new Radial Well pumps. The evaluation concluded that the design of the new PSW pumps and the associated lube-oil system minimizes any potentially negative environmental impacts. The lube-oil reservoir, piping, and tubing are designed to preclude any direct leakage to the environment, and the pump design restricts all but incidental oil migration to the pump caisson rather than the pump effluent and downstream system piping. The specified oil is “environmental friendly” such that any trace amounts that might ultimately reach the plant piping or the river will bio-degrade without impact to the environment. The chemistry department performed a control room habitability screen for this lubricating oil per procedure 0l-S-08-18, GGNS Chemical Control Program, and documented its acceptability.

This change does not adversely affect the overall performance or reliability of a safety system in a manner that could lead to an accident occurring. This change does not cause a safety system to be operated outside of its design basis limits. The new PSW Radial Well pumps and motors cannot affect any system interface in a way that could lead to an accident. The new PSW Radial Well pumps and motors will not result in degradation of safety systems. The change is intended to improve the reliability, capability and availability of the PSW system by providing a mechanism to reduce the possibility of system unavailability. Additionally, the margin of safety as defined in the bases for the Technical Specifications has not been reduced.

Serial Number: 1999-086-NPE Document Evaluated: CN 99-0022 to ER 98/0600-00-00

BRIEF DESCRIPTION OF CHANGE, TEST, OR EXPERIMENT:

CN 99-0022 was issued to provide clarification of details 5 specified in the ER package, to provide alternate methods of performing work activities, and to address material substitutions. With the exception of Item 3 of the CN, the specific items included in the Change Notice do not require any additional review. These items are encompassed by the evaluation performed for ER 1998-0600-00-00, Safety Evaluation 99-0022-ROO.

Item 3 of the CN requested the use of fiber optic cables which do not have black jackets as specificallv required by the ER. The black jacketed fiber optic cables were going to require a special order with an excessive minimum purchase for each of the cable types specified. Recommended use of the orange jacketed cables is not in compliance with UFSAR Section 8.3.1.2.3e, and must be evaluated as an exception to the requirement.

REASON FOR CHANGE, TEST, OR EXPERIMENT:

Black jacketed fiber optic cables are not manufactured as a standard industry item. The fiber optic cables specified in the CN are commercially supplied with orange jacketing . Use of orange jacketed fiber optic cables does not meet the color coding requirements of UFSAR Section 8.3.1.2.3e. To address the color coding requirements for the fiber optic cables specified in the CN, Section 8.3.1.2.3e will be revised to reflect these applications as exceptions to the color coding requirement.

SAFETY EVALUATION SUMMARY AND CONCLUSIONS:

Item 3 of CN 99-0022 will not compromise any Safety Related system, structure or component. The changes included will not prevent a safe reactor shutdown. The Security Monitoring System (C83) is Non-Safety Related and performs no safety function important to plant operation. This CN does not change the functions of the Monitoring System, but provides an evaluation of the use of atypical cable colors for its non-divisional fiber optic installations.

The fiber optic cables for this CN will be installed to provide non-divisional networking and communications functions. Existing GGNS Standards will be utilized and the separation requirements of Reg. Guide 1.75 will not be violated. Only non-divisional raceways will be utilized for installation of these cables. The plenum rating of these cables exceeds the requirements of UFSAR Section 8.3.3.1 and Table 9.5-11, D.3.f and g.

There is no UFSAR accident evaluation affected by the communications/networking cables for the Security Monitoring Stations. These functions are not required to mitigate the consequences of any transient or accident. There are no new interfaces created with Safety Related equipment and no new failure modes are introduced. These modifications will not introduce any unreviewed safety issue. The Security Monitoring System (C83) is not addressed in the GGNS Technical Specifications.

CCE-99-0001

Commitment Number:	R24349, R24351, R24352, R24353, R24339, R24225	Source Document Number:	AECM-87/0011, MAEC-85/0224, AECM-88/0102, AECM-88/0161

COMMITMENT CHANGE TITLE: Incident Review Board

COMMITMENT DESCRIPTION: GGNS would establish an IRB, convene the IRB with-in 24 hours of a human error and the results of the IRB and event would be provided to supervision and management would meet with effected sections to discuss the event. The IRB process would be proceduralized and timely.

JUSTIFICATION FOR CHANGE OR DELETION: The IRB process was to address a concern with GGNS performance in the area of human performance in 1985-1988 time frame. In the last 10 years significant changes have been made to the corrective action process as well as supporting processors such as Root Cause that make these detailed commitments unnecessary.

CCE-99-0002

Commitment Number: 9197 Source Document Number: NOV50-416/83-17

COMMITMENT CHANGE TITLE: Failure to Provide Positive Access Control to a Vital Area 50-416/83-17-1

COMMITMENT DESCRIPTION: In accordance with 10 CFR 73.55(d)(7)(i) access to a vital area must be controlled during non emergency conditions to allow only those personnel with unescorted access to the area. A guard post was established at the lower containment personnel air lock to provide positive access control to the area.

While making a routine check of facilities, the Resident NRC Inspector alleged the guard posted at containment personnel air lock (Elevation 119' of the Auxiliary Building) controlling access was asleep. As a result of the allegation, a Severity Level III violation was issued by the Region II NRC Office of Enforcement.

The licensee committed to the following corrective actions as result of the violation:

- Immediate termination of guard employment,
- The frequency of radio checks with fixed vital area posts was increased to every 15 minutes on evening and midnight shifts and every 30 minutes during daylight hours,
- The frequency of patrol checks at vital area security posts was increased to once every 60 minutes versus once every 90 minutes,
- Post rotation durations were established to a maximum of four (4) hours for individual vital area posts, and
- Desks/tables and chairs at vital area posts were replaced with stools and podiums.

JUSTIFICATION FOR CHANGE OR DELETION: 10 CFR 73.55(d)(7)(i) states in part:
The licensee shall:

Security Department is eliminating these requirements as they do not serve as a deterrent against inattentiveness, and are a burden to security operations. Also, elimination would not have any adverse impact on the effectiveness of the physical protection system at GGNS.

(i) Establish an access authorization system to limit unescorted access to vital areas during non-emergency conditions to individuals who require access in order to perform their duties. To achieve this, the licensee shall:

(A) Establish a current access authorization list for all vital areas.

(B) Positively control all points of personnel and vehicle access to vital areas.

(C) Revoke unescorted access of any individual involuntarily terminated.

(D) Lock and Protect by an activated intrusion alarm system all unoccupied vital areas.

Requirements (A), (C), and (D) were in place at time of violation. The requirement for (B) was in place; however, the air lock was blocked open and a guard was posted to ensure positive access control. Access controls failed as a result of the guard falling asleep on post, a human performance error.

10 CFR 73.55(d)(7)(i) remains in force today and no relief from existing regulatory requirements will result from elimination of the above commitments. Presently, all entries to reactor containment are within a vital area and access is no longer controlled by security personnel. Additionally, our vital area doors have decreased by more than 50 percent; thus, reducing the number of fixed vital area posts requiring access controls.

Making radio checks every 15 minutes (nights) and 30 minutes (days) is burdensome and interferes with the Alarm Station Operator's ability to adequately monitor intrusion detection equipment and CCTV to detect intrusions at the protected area barrier. It is presumptive to believe that compliance with the violation commitments will preclude an individual from sleeping or being inattentive at a fixed post location.

Having the Alarm Station Operator monitor a clock to enforce commitments set forth in the cited violation where the process has been overcome by events is unnecessary and burdensome. We have continuous vital area patrols committed to making periodic checks of active fixed vital area posts.

Normally, personnel on fixed post will be rotated every four hours. If rotation cannot be made due to extenuating circumstance, roving patrols and fixed post personnel will alternate post in order to prevent inattentiveness to duty. Periodic radio checks will continue to be made each time a fixed vital area post is activated.

CCE-99-0003

Commitment Number: 16363 Source Document Number: GNRO-91/00169
LER 91-005-01

COMMITMENT CHANGE TITLE: Delete analysis of Siemens Breaker compressor oil

COMMITMENT DESCRIPTION: Perform analyses of the Siemens breakers gas compressor oil remove during each oil change in Response to LER 91-005-01.

JUSTIFIC**ATION 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 tracing 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 SD1203 which includes performing compressor maintenance. Commitment A-4434 is for inspecting, maintaining, and testing all 500kV circuit breakers.

CCE-99-0004

Commitment Number: A-15363 Source Document Number: AECM-88/0135

COMMITMENT CHANGE TITLE: EOP Audits moving From Separate Audit to Scope Inclusion in Operations Audits

COMMITMENT DESCRIPTION: The original commitment was made in a response to Inspection Report 50-416/88-06, concerning GGNS' Emergency Operating Procedures. The results of the inspection concluded that the EOPs being used at GGNS were weak. An action plan for strengthening the EOPs was submitted to the NRC under AECM-88/0135. Part of this action plan was for QP to "establish appropriate measures to insure proper QP involvement in EOP program in the future." No frequency for this item was specified.

JUSTIFICATION FOR CHANGE OR DELETION: Entergy Operations has gone through a Renewal Effort to better standardize organization and functions at all the Nuclear stations. Part of this renewal effort was the standardization of the Audit Programs at all the facilities. In comparing audit areas between the facilities, it was recognized that Grand Gulf had a separate audit of the EOPs, where the other stations did not. They evaluated the EOPs and associated program at the same time they reviewed the Operations Department's functions. The Operations Department at Grand Gulf owns the EOP Program and associated procedures. Moving this subject as a scope to be considered when the Operations Department is evaluated/audited is in keeping with the original intent of keeping some QA type oversight on the changes in EOPs and EOP Process. The operations Area is on a 24-month audit frequency. This will still keep proper QA involvement in the EOP Program in the future.

CCE-98-0005

Commitment Number: 33262 Source Document Number: GNRO-91/00092

COMMITMENT CHANGE TITLE: Procedurally controlled mechanisms that constitute periodic procedure reviews in lieu of required biennial reviews

COMMITMENT DESCRIPTION: The proceduralized Craft Feedback Form provides an avenue for deficiencies or problems which prevent procedure implementation to be documented and provided to Maintenance Management for evaluation and incorporation.

JUSTIFICATION FOR CHANGE OR DELETION: The proceduralized Craft Feedback Form was replaced with an equivalent process and the commitment is being revised to apply ANSI 18.7 section 5.2.15, which requires utilization of a method for systematic review and feedback of information based on procedure use, at all applicable plant departments.