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

Attention: Document Control Desk

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

GNRO-2001/00075

Ladies and Gentlemen:

Pursuant to 10CFR50.59(d)(2), Entergy Operations, Inc. hereby submits the Summary of the 10CFR50.59 Evaluations for the period May 01, 2001 through August 31, 2001. Also attached is the SUMMARY of the commitment change made in accordance with guidelines of NEI 95-07 for the same period.

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

This letter does not contain any commitments

Yours truly,

Charles A. Bottemiller
Manager, Plant Licensing

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Enclosure: 10CFR50.59 Evaluations and Commitment Change Evaluation Summary
cc: (See Next Page)

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10CFR50.59 SUMMARY FOR THE PERIOD
STARTING MAY 01, 2001 AND ENDING AUGUST 31, 2001

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

Evaluation Number: 1999-0086-R01

Document Evaluated: ER 99/0545-00-00 (LDC-1999-088)

BRIEF DESCRIPTION OF CHANGE, TEST OR EXPERIMENT:

Revision 1 of safety evaluation #99-0086 is being issued to correct errors identified by CR 00/1 525.

Alternate methods of monitoring the drywell floor drain sump in-leakage are required. The preferred alternative will utilize the existing computer point P45C001. 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 P45 3001 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 stats of the sump pump starter auxiliary contacts by measuring the voltage across the coil of relay 1E31K016. 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 ESAR was changed to take exception to this requirement per FSAR CR 94-025 (GIN-94/3251) and safety evaluation 94-080-R00 based on the assumption that an alarm was required for a 1 gpm leakage increase per QDR 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 alternate method will alert the operators of a 1 gpm leak increase within the 12 hour surveillance intervals. The FSAR and tech spec basis will be appropriately updated.

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

1E31K606 which converts the change in the-level signal to a gpm value that is also recorded on 1E31R618 and monitored by flow switch 1E31N693. If the 4.2 gpm setpoint of 1E31N693 is exceeded the annunciator 1E31L625 is activated. Operations have 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.

50.59 EVALUATION SUMMARY AND CONCLUSIONS

The drywell sump level instrumentation is classified in Reg. Guide 1.97 as a type B category 1 which would require seismic and environmental qualification. GGNS took exception to this in AECM 85/0059, and classified the equipment as category 3 with no environmental or seismic qualification. The justification was that the drywell sump systems are deliberately isolated at the primary containment penetration upon receipt of a LOCA isolation signal. Once the sumps are isolated and full, they can not provide any information which could be useful for accident mitigation or long term surveillance. This exception to Reg. Guide 1.97 was approved by the NRC in MAEC 87/0013 and is noted in FSAR table 7.5-2.

The changes being made to the Technical Specification Bases and the UFSAR 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 changes to the Technical Specification Bases and UFSAR 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 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.

Therefore, no unreviewed safety questions are created as a result of this change.

Evaluation Number: 2001-0042-R00 Document Evaluated: ER 2001-0172-00-00

BRIEF DESCRIPTION OF CHANGE, TEST OR EXPERIMENT:

Design Engineering evaluation of scaffolding currently in place at Elevation 185' Azimuth 1400 — 1500 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 1400 — 1500 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. 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 for scaffolding erection in a seismic category 1 structure. 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 UFSAR 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 RF 12 . The scaffolding must be removed or replaced with a permanent platform prior to startup from RF 12 . There will be no adverse impact on the safe operation of GGNS by this scaffolding remaining in place until the end of RE 12. ER-GG-2001-0173-000-0 has been assigned to cover installation of the permanent platform.

Evaluation Number: 2001-0043-R00 Document Evaluated: MS – 48.0, Rev. 9

BRIEF DESCRIPTION OF CHANGE, TEST OR EXPERIMENT:

This safety evaluation assesses the reload-related changes associated with Cycle 12 operation as presented in Revision 9 to the Core Operating Limits Report, Mechanical Standard GGNS MS-48.0. Cycle 12 has been designed for 459 Effective Full Power Days with a core consisting of 204 fresh ATRIUM-10 assemblies, 228 GE11 once-burnt assemblies, 268 GE11 twice-burnt assemblies, and 100 GE11 thrice-burnt assemblies. This is the first use of the ATRIUM-10 fuel design at GGNS. In addition, Framatome-ANP's (FRA-ANP) POWERPLEX core monitoring system will be used to monitor the core for compliance to the operating limits. SAR, TS, TS Bases, TRM, and COLR changes are required 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 12 reload analysis and the issues considered in this evaluation.

REASON FOR CHANGE, TEST OR EXPERIMENT:

Cycle 12 operation will require new core operating limits and the Core Operating Limits Report has been revised to include these new limits. These limits include flow-, power-, and exposure-dependent LHGR. MAPLHGR, and MCPR limits. Other changes are required in the Tech Specs, TRM, SAR, and TS Bases. Cycle 12 core operation will also require a change in the core monitoring system to the PowerPlex -III (PPX-III) system supplied by FRA-ANP.

SAFETY EVALUATION SUMMARY AND CONCLUSIONS:

This evaluation concludes that the reload-related changes associated with Cycle 12 operation in GIN-2001/00461 are required for implementation. UFSAR will not constitute an unreviewed safety question: however the changes to the EOPs transmitted in GIN-2001/00461 are required for implementation.

Evaluation Number: 2001-0044-R00   Document Evaluated: ER GG–1997–0939-002–00

BRIEF DESCRIPTION OF CHANGE, TEST OR EXPERIMENT:

The jacket water cooler for penetration 83 (RWCU) is supplied cooling water by the Plant Chilled Water system through a ¾" stainless steel braided flex hose. ER 97/0939-00-00 approved the operation of the current configuration of the hoses for jacket water cooler for penetration 83 until RF11. The 3/4 inch braided flex hoses are all metal construction and have a working pressure and temperature well in excess of the system service conditions. They were evaluated and considered acceptable per Safety Evaluation 98-0034-R01. ER-GG-1997-0939-002 makes the installation of the currently installed flex hoses permanent and provides inspection criteria in order to maintain the flex hoses in an operable condition.

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, supplied to the Penetration 83 jacket water cooler through a ¾" flexible braided metal hose. Inspections of the currently installed flex hoses in RF10 and RF11 revealed that they are in good condition. No leakage or deterioration of the hoses was observed. The flex hoses are all metal and of rugged construction. ER-GG-1997-0939-002 makes the installation of the current flex hoses permanent and provides inspection criteria to insure that the hoses are kept in an operable condition.

SAFETY EVALUATION SUMMARY AND CONCLUSIONS:

ER 97/939-02-00 evaluates making the installation of the flex hoses installed on the jacket water cooler for penetration 83 permanent. The original installation of the flex hose was considered acceptable per Safety Evaluation 98-0034-R01. The temperature limits of the concrete surrounding the penetrations will be maintained. All design bases remain unchanged. In conclusion, continuing to operate 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 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.

Evaluation Number: 2001-0045-R00 Document Evaluated: MAI-294012

BRIEF DESCRIPTION OF CHANGE, TEST OR EXPERIMENT:

This evaluation is in support of back filling the Steam Jet Air Ejector flow transmitters while maintaining open breaker 52-124120 for the 1N62F003B valve. This MAI will be performed in order to provide reliable operation of the Steam Jet Air Ejectors during the next operating cycle. Per UFSAR Section 10.4.2, valve 1N62F003B is provided with an automatic closure signal to isolate condenser flow to the B offgas system if low steam flow to the second stage Steam Jet Air Ejector B is sensed by flow transmitters 1N62N01 3B and D. These are the transmitters that require backfilling. The purpose of this automatic closure signal is to prevent excessive Hydrogen buildup in the condenser air removal system, because this system is not designed to accommodate a Hydrogen detonation. Therefore, during the period when the breaker is open, Operations will monitor Offgas flow and hydrogen concentration to assure that levels are maintained in a safe operating condition. An operator will be stationed at the breaker in order to close the breaker if high hydrogen concentrations or a large fluctuation in Offgas flow is noted. Instrumentation will ensure that these transmitters are reading greater than 8.0 klbs/hr prior to returning the 1N62F003B Valve to service to reduce the potential for a spurious closure of the 1N62F003B valve.

REASON FOR CHANGE, TEST OR EXPERIMENT:

The Steam Jet Air Ejector flow transmitters must be back filled for proper operation. The circuit breaker for the 1N62F003B valve will be opened to prevent inadvertent isolation of the condenser suction to the air ejector during this evolution.

SAFETY EVALUATION SUMMARY AND CONCLUSIONS:

This screening is to support a maintenance activity required to back fill the Steam Jet Flow transmitters. This activity will be performed in a timely manner and compensatory actions will be taken by operations to assure that the function of these transmitters is maintained. Operations will monitor Offgas flow and hydrogen concentrations to assure that levels are maintained in safe operating conditions. An operator will be stationed at the breaker for 1N62F003B to close the breaker for Operations to close the isolation valve if an unacceptable increase in hydrogen concentrations or a large fluctuation in Offgas flow is noted. Instrumentation will ensure that these transmitters are reading greater than 8.0 klbs/hr prior to returning the 1N62F003B to service. The Condenser Air Removal System is described in Section 10.4.2 of the UFSAR. Offgas System Leak or Failure is the limiting accident described in Section 15.7.1 of the UFSAR. Although the automatic functions described in Section 10.4.2 will be temporarily disabled when the breaker for 1N62F003B is opened, these functions are not relied upon to mitigate the accident described in Section 15.7.1. Manual isolation of the system from the Main Condenser is the mitigating action to prevent release of radioactive gases. Therefore, the compensatory measures described above will ensure that adequate administrative controls are used so that the protective features normally provided by the automatic isolation are maintained using direct manual control by Operations personnel. This change is temporary and administrative controls will ensure the manual protective actions described in UFSAR 15.7.1 can be performed in the unlikely occurrence of a system failure during the period that the breaker is open.

Evaluation Number: 2001-0046-R00 Document Evaluated: LDC-2001-073

BRIEF DESCRIPTION OF CHANGE, TEST OR EXPERIMENT:

The TIP system operability requirements of TRM 6.3.4 are revised based on the capabilities of the new POWERPLEX core monitoring computer system. The revised operability requirements will allow the operation of the TIP system with one TIP machine inoperable, or an equivalent number of individual TIP channels out of service (OOS). In addition, the applicability section is revised to remove the monitoring of power distribution limits (APLHGR, LHGR, and MCPR) using the TIP system. UFSAR and Bases sections are also changed as necessary to reflect the capabilities of the new core monitoring system for TIPs OOS and up to 50% LPRMs failed. The uncertainty associated with 50% LPRMs failed and one TIP machine OOS for the new core monitoring system is bounded by the uncertainty assumed in the MCPR Safety Limit analysis (as shown in Reference 3). The MCPR safety limit submittal (which includes the increased uncertainties for equipment out of service) was reviewed and approved by appropriate plant staff before submission to the NRC. The NRC subsequently approved the MCPR safety limit as shown in Reference 3.

REASON FOR CHANGE, TEST OR EXPERIMENT:

This change updates the requirements for the TIP system operability and failed LPRMs consistent with the capabilities of the new core monitoring system.

SAFETY EVALUATION SUMMARY AND CONCLUSIONS:

The TIP system is not safety-related, has no direct safety function, has no automatic actions on any plant equipment, is not an accident initiator, and is not credited as being required for mitigation of any accidents described in the SAR. No physical modifications are being made to the TIP system. The proposed change to the TIP operability requirements is consistent with the capability of the new core monitoring system (POWERPLEX III) used in calibration of the nuclear instruments (LPRMs) and monitoring of power distribution limits. The number of LPRM inputs required for the core monitoring system is not addressed in the SAR. Operation of the TIP system with the revised operability requirements and up to 50% LPRMs failed is bounded by the uncertainties assumed in the licensing basis analyses (References 2 and 3). No margins of safety contained in the Technical Specification Bases are affected by this change. Therefore, this change does not result in the creation of an Unresolved Safety Question.

Evaluation Number: 2001-0047-R00 Document Evaluated: LDC 2000-074

BRIEF DESCRIPTION OF CHANGE, TEST OR EXPERIMENT:

Section 9.5.6 of the UFSAR discusses the Diesel Generator Starting Systems. In section 9.5.6.3, "Safety Evaluation", page 9.5-37 of the UFSAR states that "The system piping is installed at an elevation lower than the engine inlet, and is provided with a drip leg to provide for removal of any water which may be present in the lines". The statement is applicable to DG division I and II. The description of the starting air piping configuration is misleading. Not all of the starting air system piping is below the engine inlet. Only the starting air system piping immediately upstream of the engine inlet is below the elevation of the engine inlet. The section of piping immediately upstream of the engine inlet does contain a drip leg. The drip leg or drain leg is a design requirement per SDC-P75: System Design Criteria Standby Diesel Generator System. The SDC requires that drains shall be provided for piping low points in order to remove moisture from the system. The design bases of the system are still met even though the UFSAR description is inaccurate. Moisture is prevented from reaching the starting components of the DG engine by removing it from the system by means of the drip leg located in the system piping immediately upstream of the engine inlet which is at a lower elevation than the engine inlet.

Section 9.5.8 of the UFSAR discusses the Diesel Generator Combustion Air Intake and Exhaust System. On page 9.5-45, in section 9.5.8.3, the 2nd paragraph states "The enclosures are designed to protect the enclosed equipment against fibers, flyings, dust and dirt, lint, and light splashing, seepage..." The sentence contains the word "flyings". It is a term specific to the electrical field that means fibers or lint, and is therefore considered technically correct; however, the word is confusing without knowledge of the specific electrical definition. The word "flyings" is also redundant. The sentence, which contains the word "flyings", also contains the word fibers and lint, which means the same as "flyings." The word should be deleted because it causes confusion and is redundant in the sentence.

LDC 2000-074 makes corrections to the UFSAR in order to clarify the starting air system piping configuration description with respect to the diesel engine inlet and removes the word "flyings" from section 9.5.8.3 to reduce the confusion and to remove the redundancy in the sentence.

REASON FOR CHANGE, TEST OR EXPERIMENT:

CR-GGN-1999-0290 was written to document the misleading starting air system piping description and a potential typo. LDC 2000-074 will clarify the statement in the UFSAR to reflect the starting air system piping with respect to the diesel engine inlet as it exists in the plant and deletes the word "flyings" from 9.5.8.3 to reduce the confusion and to remove the redundancy within the sentence.

SAFETY EVALUATION SUMMARY AND CONCLUSIONS:

The change to the Standby Diesel Generator Starting Air System Piping description made per LDC 2000-074 clarifies the UFSAR to reflect the starting air system piping configuration as it exists in the plant. The design bases of the system are currently met even though the UFSAR description is inaccurate. Moisture is still prevented from reaching the starting components of the DG engine by removing it from the system by means of the drip leg located in the system piping immediately upstream of the engine inlet. The drip leg is at a lower elevation than the engine inlet. The Diesel Generator (DG) starting air systems are safety related. They provide the motive force for starting

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the Division I and II DG engines. The DG Generator system still operates inside its design bases. The change does not make any physical changes to plant configuration, design bases, design operating limits, or plant operating practices. The changes made per LDC 2000-074 do not increase the probability or consequences of an accident or malfunction of equipment important to safety.

The removal of the word “flyings” from section 9.5.8.3 of the UFSAR made per LDC 2000-074 reduces the confusion and removes the redundancy from within the sentence. This change is strictly editorial. The intent of the sentence does not change with the removal of the word “flyings”. This change to section 9.5.8.3 does not make any changes to plant configuration, design bases, design operating limits, or plant operating practices.

Evaluation Number: 2001-0048-R00 Document Evaluated: LDC 2001-041

BRIEF DESCRIPTION OF CHANGE, TEST OR EXPERIMENT:

The FSAR, tech spec basis, E3 1 SDC and E3 1 design spec will be updated per CR 00/0055 & CR 00/0056 based on the discussion below.

REASON FOR CHANGE, TEST OR EXPERIMENT:

FSAR 7.6.1.4.3.9.2.1 states that a 1 gpm leak into the drywell floor drain sump equates to a 4.7 inch/hr change in sump level and that the instrumentation is capable of detecting a 1.8 inch/hr change in sump level. Per CR 00/0056, both values are incorrect. The 4.7 inch/hr value should be 3.96 and the 1.8 inch/hr value should be 0.58 inches /hr. The sump fill-up rates and calculated instrument sensitivity values will not be corrected. They will instead be removed from the UFSAR. The statement that GGNS meets the 1 gpm/hr sensitivity/response time requirements of Reg. Guide 1.45 with the exceptions listed in appendix 3A is acceptable. This is considered a FSAR simplification that is acceptable per NEI 98-03 rev 1. The incorrect values were added to the FSAR by FSAR change notice 140 per Q&R 211.1. The values originated from calculation J-E3 1-1 which contained several errors. This calculation has been superceded by MC-N1P45-99034 & J-E31-6 which are correct.

Position 6 of Reg. Guide 1.45 specifies that the drywell floor drain sump level/flow monitoring equipment and the drywell air cooler condensate flow rate monitoring equipment shall be seismically-qualified for an OBE. An Exception is taken to the seismic qualification of the air cooler condensate flow rate monitoring equipment in FSAR section 7.6.2.4.2.1. Per CR 00/0055, the justification for the exception is incorrect because it assumes that the drywell floor drain sump level/flow monitoring equipment is seismically qualified. The SDC and design spec still specify that both the Sump level/flow monitoring equipment and the Air Cooler condensate flow rate monitoring equipment are seismically qualified. The documents will be revised to take exception to the seismic qualification requirement for both.

There are two Sump level/flow monitoring systems. The primary system consists of a level transmitter, rate converter, recorder and high flow alarm switch. The backup system consists of level sensors / level switches that control the operation of the sump pumps, a digital computer point that monitors the on/off status of the sump pumps, a leakage rate computer point that is calculated based on the time between pump downs and a corresponding high alarm computer point. The primary system was added by FDI WAPJ (10/19/81) to meet the qualification and 1 gpm/hr sensitivity requirements of Reg. Guide 1.45. However, the qualification documentation was not reviewed by GGNS personnel and the equipment was never added to the SQCF and it has not been maintained as qualified. The statement in UFSAR 7.6.2.4.2.1 regarding the OBE qualification of the level measurement system was added by FSAR change notice 142 (2/13/79) to address Q&R 211.3. Use of the backup system was approved by ER 99/0545 but the backup system equipment was never qualified.

The LDC 99/0088 issued per ER 99/0545, added the following statement to the FSAR and the tech spec B3.4-7 basis:

"Additionally approved M&TE equipment can be used to monitor the fill times and the leakage and change in leakage can be manually calculated." Licensing has requested that this statement be removed from B3.4-7 since it could be misinterpreted as meaning the tech spec LCO for 3.4.7 can bypassed if M&TE is used when the permanent plant equipment fails.

The M&TE can be used for the determination of leakage per tech spec 3.4.5 but can not be used to by pass the equipment operability requirements of the tech spec 3.4.7 LCO. This is considered an editorial change.

SAFETY EVALUATION SUMMARY AND CONCLUSIONS:

The equipment used for the drywell floor drain sump level/flow monitoring and drywell air-cooler condensate flow rate monitoring are not seismically qualified for an operating basis earthquake (OBE) as specified in position 6 of Reg. Guide 1.45. This deviation is acceptable based on the following.

Failure of the primary or backup sump level/flow monitoring instrumentation or the drywell air cooler condensate flow rate monitoring instrumentation because of an earthquake below the OBE alarm setpoint is unlikely because at GGNS the OBE alarm setpoint is very low. The levels are low enough that standard nonsafety related equipment would be expected to continue functioning. A failure because of an earthquake that would cause all three loops to appear operable but read non-conservative is very unlikely.

Per off normal event procedure 05-S-02-VI-3, if an OBE alarm or safe shutdown earthquake (SSE) alarm occurs, the operators are required to go to cold shut down at the maximum rate allowed per 101 03-1-01-3. Before the plant can be restarted, an evaluation must be made by the General Manager. For earthquakes less than the OBE alarm setpoint, operators are required to evaluate several plant parameters (including the leak detection system) for abnormal changes or conditions. The OBE limit is 0.075 g and the OBE alarm setpoint is 0.01 g. The SSE limit is 0.15 g and the SSE alarm setpoint is 0.13 g. Tech. spec 3.4.5 requires operators to confirm that unidentified reactor coolant leakage is within specified limits every 12 hours. A failure of the subject instrumentation would not go undetected.

The drywell sump level instrumentation is classified in Reg. Guide 1.97 as a type B category I which would require seismic and environmental qualification. GGNS took exception to this in AECM 85/0059, and classified the equipment as category 3 with no environmental or seismic qualification. The justification was that the drywell sump systems are deliberately isolated at the primary containment penetration upon receipt of a LOCA isolation signal. Once the sumps are isolated and full, they can not provide any information which could be useful for accident mitigation or long term surveillance. This exception to Reg. Guide 1.97 was approved by the NRC in MAEC 87/0013 and is noted in FSAR table 7.5-2.

Tech spec 3.4.5 unidentified leakage limit is 5 gpm and less than a 2 gpm increase in 24 hrs. Per the tech spec basis, limits on reactor coolant system (RCS) leakage is required to ensure appropriate action is taken before the integrity of the reactor coolant pressure boundary (RCPB) is impaired. Per FSAR 5.2.5.5.3, the analysis is based on a 5 gpm leak but a leakage rate of hundreds of GPM would precede unstable crack development. The analysis and tech spec limits are not affected by this exception to Reg. Guide 1.45.

Tech spec 3.4.7 allows both the Sump level/flow monitoring instrumentation and the Air Cooler condensate flow rate monitoring instrumentation to be inoperable for up to 30 days if either the drywell atmospheric particulate or gaseous monitoring systems are available. These LCO's are not affected by this exception to Reg. Guide 1.45. There are no unreviewed safety questions.

Evaluation Number: 2001-0049-R00 Document Evaluated: ER GG – 2001–0193–00-00

BRIEF DESCRIPTION OF CHANGE, TEST OR EXPERIMENT:

The RHR "A" minimum flow line (4"-HBB-120) has developed a pinhole leak in one of the lines' elbows. Online weld repair of pinhole leak and wall thinning is required.

REASON FOR CHANGE, TEST OR EXPERIMENT:

CR GGN-2001-0955 documented that the RHR "A" minimum flow line (4"-HBB-120) has a pinhole leak which requires online repair. The pinhole is located in the Auxiliary Building elevation 104'-3". The current leakage rate is 0.0024 gpm (9 ml/mm) as measured by plant personnel. This repair requires that the RHR piping in Containment Penetration #23 be drained during the weld repair. This piping cannot be isolated from the suppression pool and therefore the effects of performing this online repair on primary containment must be evaluated.

SAFETY EVALUATION SUMMARY AND CONCLUSIONS:

The weld overlay repair of the pin hole leak will not impact the operability, function or integrity of the subject system. No change in the operation or function of the RHR system will be created by this repair and no change to GGNS Technical Specifications or UFSAR will be required. The implementation of ER GG-2001-0193-000-00 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 GG-2001-0193-000-00 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 system piping integrity will be assured with the weld overlay repair of the pin hole leak. There is no change to the pipe configuration, design/quality requirements, system operation, function, integrity or other system parameters. The margin of safety as defined in the Technical Specification Basis is not reduced.

Evaluation Number: 2001-0050-R00 Document Evaluated: LDC-2001-080

BRIEF DESCRIPTION OF CHANGE, TEST OR EXPERIMENT:

The LDC revises UFSAR table 3.9-2c to reflect the results of resolved Design Basis Discrepancies OIN D-3.9-3, Rev. 1 and D 3.9-14.

REASON FOR CHANGE, TEST OR EXPERIMENT:

Design Basis Discrepancies were found in UFSAR table against design documents.

SAFETY EVALUATION SUMMARY AND CONCLUSIONS:

The RWCU non-regenerative Heat Exchanger is discussed in various sections of the UFSAR including Table 3.9-2c which contains the stress analysis for the heat exchanger. UFSAR Table 3.2-1 indicates that the RWCU heat exchangers are Safety Class "Other", Quality Group C constructed to ASME section III-3 & TEMA C. The changes made to Table 3.9-2c reflect analyzed actual loads and corrects a typographical error to match design documents. The proposed change is within the existing licensing basis of the Grand Gulf Nuclear Station. This Safety Evaluation documents the fact that the proposed change does not result in an Unreviewed Safety Question for the following reason:

The change made is an administrative type and updates the UFSAR to reflect latest actual data from the design document. The change does not affect any specific system, structure or component that performs a safety function. The change does not degrade the performance of a safety system assumed to function in an accident analysis and does not decrease the reliability of a safety system assumed to function in an accident analysis; the change does not put plant operation in an unanalyzed region; the change herein is bounded by the Technical specification, the TRM and the SAR; and the change does not adversely affect the overall performance or reliability of a safety system in a manner that could lead to an accident.

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

Evaluation Number: 2001-0051-R00 Document Evaluated: ER–GG–2000–0792–006–00; LDC 2001-065

BRIEF DESCRIPTION OF CHANGE, TEST OR EXPERIMENT:

ER-GG-2000-0792-006 is one of a series of Engineering Requests that will design and install an Auxiliary Cooling Tower (ACT) in the Circulating Water System. ER-GG-2000-0792-006 provides for design, fabrication and installation of approximately 1200 feet of 96-inch piping that connects the circulating water system discharge piping to the ACT. The new piping will be connected to the circulating water system piping tee's that were installed under ER-GG-2000-0792-003 and will be routed both above and below ground as it travels to the ACT location. Piping design includes ports for temporary and permanent (future) flow monitoring instrumentation, manways, and chemical injection quills. A cross-connect manifold with butterfly valve is provided at the ACT manifolds to allow ACT operation from either header, or both headers simultaneously. In addition, an 8-inch drain line will be routed from each ACT supply line to the existing 36-inch natural draft cooling tower drain line. The ER installs an electrical handhole and manhole and the ductbanks that connect them, and installs a portion of the ductbank that travels under the security perimeter fence toward the ACT. The ER also provides for re-route of a site power loop ductbank and a security system ductbank whose existing locations interfere with the piping run.

REASON FOR CHANGE, TEST OR EXPERIMENT:

The natural draft cooling tower design is such that a significant loss of condenser and turbine efficiency occurs during periods of high ambient air temperatures combined with high relative humidity. These conditions typically occur during periods of peak electrical demand. Installation of the ACT will help lower the circulating water temperature at the condenser inlet, thereby increasing the unit efficiency and electrical output. Future planned power uprates further accentuate the issue.

SAFETY EVALUATION SUMMARY AND CONCLUSIONS:

The circulating water system and site power loop serve no safety functions and are not safety-related. Upon completion of ER-GG-2000-0792-006, the new piping will not be placed in service, but will remain isolated by valves installed under ER-GG-2000-0792-003. Use of the new piping will be evaluated separately under ER 2000-0792-000 that installs the ACT and licenses it for use. Therefore, the functions of the circulating water system, including the natural draft cooling tower, are unchanged by this ER. The site power loop and security monitoring system functions also remain unchanged as a result of ductbank re-routing. The activities associated with this ER do not create any new failure modes or accident initiators. The proposed changes are within GGNS licensing basis. This Safety Evaluation documents the fact that the proposed changes do not result in an Unreviewed Safety Question for the following reasons:

1) It does not put the plant operation in an unanalyzed region. The changes herein are bounded by analyses in the existing Technical Specifications (including Bases), TRM and SAR.
2) The proposed changes do not degrade the performance of any structure, system or component as defined in the SAR, TRM or Technical Specifications (including Bases).

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

Evaluation Number: 2001-0052-R00 Document Evaluated: LDC 2001-088

BRIEF DESCRIPTION OF CHANGE, TEST OR EXPERIMENT:

The purpose of this evaluation is to eliminate the TRM Required Actions to prepare and submit Special Reports to the NRC for TRM sections 6.2.1, Fire Detection Instrumentation; 6.3.3, Meteorological Monitoring Instrumentation; 6.3.5, Loose-Part Detection System; and, 6.3.6, Main Condenser Offgas — Explosive Gas Monitoring Instrumentation.

REASON FOR CHANGE, TEST OR EXPERIMENT:

There is no regulatory requirement for preparing these Special Reports, and the corrective action program will address the deficiencies and any corrective actions taken. The elimination of these Special Reports will not affect operating this facility in a safe manner. There is no requirement for the Nuclear Regulatory Commission to approve the Special Reports.

SAFETY EVALUATION SUMMARY AND CONCLUSIONS:

Reviewed LBDs (TRM, UFSAR, QAPM, 10CFR, ODCM, E Plan) to find references to Special Reports. Conducted searches using 'Special Report', 'Special Reports', and 'Reports within 30'. The results of the search indicated several sections of the TRM / UFSAR that referenced preparation of a special report. This evaluation addressed elimination of those special reports referenced in TRM sections 6.2.1, Fire Detection Instrumentation; 6.3.3, Meteorological Monitoring Instrumentation; 6.3.5, Loose-Part Detection System; and, 6.3.6, Main Condenser Offgas —Explosive Gas Monitoring Instrumentation. However, this evaluation does not address those Special Reports required in the Radiological Effluents Section nor Radiological Environmental Monitoring Section because of their ties to the ODCM, the reportability procedure, and 10CFR.

There is no regulatory requirement for preparation and submittal of these Special Reports, and there is no requirement for the Nuclear Regulatory Commission to approve the Special Reports. The site's Corrective Action Program will address the deficiencies and any corrective actions taken. This Corrective Action Program is routinely reviewed by resident NRC inspectors and visiting inspectors. This proposed change is administrative in nature and the elimination of these Special Reports will not affect operating this facility in a safe manner nor does it create any Unreviewed Safety Questions.

Evaluation Number: 2001-0053-R00 Document Evaluated: LDC-2000-041

BRIEF DESCRIPTION OF CHANGE, TEST OR EXPERIMENT:

The Turbine Stop and Control Valve Operability surveillance is used to demonstrate the operability of the four high pressure stop valves, four high pressure control valves, six low pressure stop valves, and six low pressure control valves at least once per 31 days by cycling each of the valves through at least one complete cycle from the running position using the manual test or the Automatic Turbine Tester (ATT). This safety evaluation will change the required frequency from 31 days to 84 days. This change has been approved by Siemens/Westinghouse Power Corporation. Documentation supporting this transition from monthly to quarterly can be found in GEXI–2000/00060.

REASON FOR CHANGE, TEST OR EXPERIMENT:

Generator output must be less than 90% of rated load (1223 MWe) and reactor power less than 90% of rated (3449 MWT) to perform Attachment 1 of 06-OP-1N32-V-0001. This surveillance is presently on a monthly schedule. This change will extend the allowed testing frequency to 12 weeks (maximum). This change should allow for improvement in unit capability factor, and will provide significant cost savings over the remaining life of the plant.

SAFETY EVALUATION SUMMARY AND CONCLUSIONS:

Siemens! Westinghouse Power Corporation has conducted studies to determine if it is permissible to extend the recommended interval between turbine valve testing beyond the present specified time. The calculated probability of occurrence of impermissible overspeed, as a function of the interval between tests, shows that a an interval of 12 weeks between tests is permissible in view of the desired reliability level. However, this extension of the interval between tests is permissible only if time-dependent defects do not develop. Current procedures at GGNS incorporate the methods and instruments needed to monitor for time dependent defects.

UFSAR Chapter 15.2.3, Turbine Trip was reviewed. There are no changes to the operation of the Automatic Turbine Tester that impact the finding in this chapter. This Safety Evaluation addresses the safety limits, boundary performance during normal and accident conditions, and the impact of assumptions of system performance made in the UFSAR. It is determined there were no Unreviewed Safety Questions that emerged during the process of this Safety Evaluation.

Evaluation Number: 2001-0053-R01 Document Evaluated: LDC-2000-041

BRIEF DESCRIPTION OF CHANGE, TEST OR EXPERIMENT:

The Turbine Stop and Control Valve Operability surveillance is used to demonstrate the operability of the four high pressure stop valves, four high pressure control valves, six low pressure stop valves, and six low pressure control valves at least once per 31 days by cycling each of the valves through at least one complete cycle from the running position using the manual test or the Automatic Turbine Tester (ATT). This safety evaluation will change the required frequency from 31 days to 92 days. This ATT testing interval change from monthly to quarterly is in accordance with Siemens/Westinghouse Power Corporation. Recommendation provided in GEXI 2000-00060.

REASON FOR CHANGE, TEST OR EXPERIMENT:

Generator output must be less than 90% of rated load (1223 MWe) and reactor power less than 90% of rated (3449 MWT) to perform Attachment 1 of 06-OP-1N32-V-000I. This surveillance is presently on a monthly schedule. This change will extend the allowed testing frequency to quarterly (92 days). This change should allow for improvement in unit capability factor, and will provide significant cost savings over the remaining life of the plant.

SAFETY EVALUATION SUMMARY AND CONCLUSIONS:

Siemens/Westinghouse Power Corporation has conducted studies to determine if it is permissible to extend the recommended interval between turbine valve testing beyond the present specified time. The calculated probability of occurrence of impermissible overspeed. as a function of the interval between tests, shows that a quarterly interval between tests is permissible in view of the desired reliability level. However, this extension of the interval between tests is permissible only if time-dependent defects do not develop. Current procedures at GGNS incorporate the methods and instruments needed to monitor for time dependent defects.

UFSAR Chapter 15.2.3, Turbine Trip was reviewed. There are no changes to the operation of the Automatic Turbine Tester that impact the finding in this chapter. This Safety Evaluation addresses the safety limits, boundary performance during normal and accident conditions, and the impact of assumptions of system performance made in the UFSAR. It is determined there were no Unreviewed Safety Questions that emerged during the process of this Safety Evaluation.

Evaluation Number: 2001-0054-R00 Document Evaluated: LDC 2001-090

BRIEF DESCRIPTION OF CHANGE, TEST OR EXPERIMENT:

UFSAR Sections 18.1.13 and 13.1.2.3.10 are being revised to eliminate specific references to plant procedures by name and numbers.

REASON FOR CHANGE, TEST OR EXPERIMENT:

This change is necessary to allow for flexibility in the future as Entergy standardizes plant processes between nuclear sites. This change is specifically being implemented for the standardization of the Protective Tagging Process under Nuclear Management Manual OP-102 and the deletion of the Grand Gulf Nuclear Station Administrative Procedure 01-S-06-1, “Protective Tagging System”.

SAFETY EVALUATION SUMMARY AND CONCLUSIONS:

The change to the UFSAR is basically administrative in nature as only specific details in the UFSAR dealing with procedure numbers and names are being removed. Basic plant processes are still being implemented consistent with current standards and practices. This change does not involve an Unreviewed Safety Question or Technical Specification changes.

Evaluation Number: 2001-0055-R00 Document Evaluated: ER GGN-2001-0189-00-00

BRIEF DESCRIPTION OF CHANGE, TEST OR EXPERIMENT:

ER GGN-2001-0189-00-00 is providing on-line leak repair instructions for the Reactor Water Clean Up System (RWCU) drywell outboard isolation valve Q1G33F253.

REASON FOR CHANGE, TEST OR EXPERIMENT:

Q1G33F253 has pressure seal leakage and is allowing leakage from some of the eight-pressure seal ring knockout holes in the valve body. This repair will consist of drilling and tapping the existing segment ring knockout holes on the Q1G33F253 valve and injecting Furmanite sealant compound downstream of the leakage. This repair method will allow the Furmanite compound to dam on the downstream side of the pressure seal ring and fill the void areas near the valve bonnet and should provide a means of controlling the leakage.

These repair instructions governed by this 50.59 will be valid until the first forced outage of sufficient duration to allow the final rework or replacement of the valve, but no later than the completion of RFI2 (i.e. Fall 2002).

The repair described in this ER is different than that recommended by EPRI, valve manufacturer and Furmanite. This method of controlling the leakage is being utilized because the attempts on other pressure seal valves to inject the seal ring from below (as recommended by EPRI) have been unsuccessful at GGNS. Backside injection requires reevaluation of the bonnet bolt stresses for additional loads.

SAFETY EVALUATION SUMMARY AND CONCLUSIONS:

Valve Q1G33F253 is the RWCU Return Outboard Drywell Isolation Valve and is located in the Containment Building. This valve currently has pressure seal ring leakage and is exhibiting leakage from at least one of the eight segment ring knockout holes located downstream of the pressure seal ring. The valve will be injected on-line with sealant compound via the valve segment ring knockout holes on the downstream side of the pressure seal ring. This location for the shutoff adaptors should allow the sealant compound to fill the valve body voids between the bonnet and downstream side of the pressure seal ring and control the current leakage. This repair will be valid until the first forced outage of sufficient duration to allow the final replacement or rework of the valve or until the completion of RFI2 (i.e. Fall 2002).

Calculation NPE-1E12F394/G33F001 /F004/F250/F251/F252/F253, Rev. 13, Supplement to the Powell Seismic Calculation D-67763 was performed utilizing a maximum system pressure of 1220 psig to evaluate the bonnet and yokearm bolting stresses. This evaluation has shown that, based on the maximum operating pressure of 1220 psig and utilizing the highest conservative as left valve maximum stem thrust force of 31693 lb., for valve QI G33F253, this repair will maintain the valve safety related operation and drywell isolation function following the on-line injection of sealant through the segment ring holes. A limited amount of the sealant will be injected into the valve to seal around the pressure seal ring. Injection of the sealant compound on the downstream side of the pressure seal ring is an alternate location, which is not usually used for performing on-line leak repairs, but has been evaluated and found to be acceptable for use on valve Q1G33F253. This allows confining the sealant by utilizing the voids between the valve body and valve bonnet. The installation of the shutoff adaptors and injection of the Furmanite sealant compound will not adversely affect the structural integrity of the valve.

Calculation 9645-M-242.0-Q1-8.0-5-4, Supplement 3, Rev. 0, has shown that the valve structural integrity is maintained with the larger 5/8" holes with no reinforcement required. Also, a Furmanite calculation has evaluated the valve body stresses for adding the shutoff adaptors and for the injection pressures shown in Furmanite Procedure No. N-2001109. The Furmanite and GGNS calculations have shown all stresses in the valves will remain within ASME Section III code allowable for a Class 1 valve. Also, the actual injection pressure of the Furmanite compound inside each valve body will be held to 1220 psig, which is less than the design pressure of 1250 psig for the valves as specified on vendor drawing M-242.0-Q1-1.2-188, Rev. 5. After the injection of the sealant, the valve will be partially stroked to verify the stem movement in the close safety direction. The Q1G33F253 valve will therefore be capable of performing its safety-related function as Outboard Drywell isolation valve and will not increase the possible offsite radiation dose, and therefore not affect the health and safety of the public.

While the valve repair may change the operational load path and affect the non-Code portions of the valve, it does not alter the original valve design because the load path that prevents disassembly of the valve's pressure boundary during operation or accident conditions remains the same. As in the original design, the segmented thrust ring still provides the positive locking mechanism that retains the bonnet inside the valve body. The pressure seal ring (a gasket) is being partially or completely replaced with an injected sealant that depends on non-Code portions of the valve to retain the sealant in position similar to a packing gland assembly. Based on evaluations it is the position of Central Engineering Programs and GGNS Design Engineering that the Code boundary is unaffected by the described repair. However, the repair does alter the stresses on the non-Code portions of the valve and a thorough evaluation of that effect on the integrity of the valve's actuator assembly has been performed per calculation NPE1E12F394/G33F001/F004/F250/F251/F252/F253, Rev. 13, Supplement to the Powell Seismic Calculation D-67763.

Calculation NPE-1E12F3 94/G33F001/F004/F250/F251 /F252/F253, Rev. 13, Supplement to the Powell Seismic Calculation D-67763 assumed that the on-line leak sealant would apply uniform loads to the body to yokearm bolting. This assumption is acceptable since any potential non-uniform filling will result in minor stress variation that will not challenge margins provided in code allowable stress.

The final replacement or rework of the valve will consist of replacing the valve with a like for like component or if not possible removing the shutoff adaptors from the segment ring holes and removing the Furmanite compound from the valve body and other components. Also, if the valve is replaced, this ER response allows for tapping out the new segment rings knockout holes for future installation of Furmanite shutoff adaptors. Tapping these holes will not change the valve function since the holes are downstream of the pressure seal ring and are normally used to knockout the segmental ring during valve rework.

This on-line repair will not affect the pipe break accidents identified in UFSAR Appendix 3C, Section 3C.2.2, since the valve will maintain its original design function. Also, this repair will not affect the missile evaluations identified in UFSAR Section 3.5. This repair is not creating any new missiles, since the shutoff adaptors are similar to the nut/bolt combination discussed in UFSAR 3.5.1.1.2, subsection h. UFSAR 3.5.1.1.2., subsection h concludes that nuts/bolts only have a small amount of stored energy and thus are of no concern as potential missiles. One risk involved in performing a leak repair is injecting too much sealant into a valve to seal a leak. ER-GG-2001-0189-000-0 will administratively control the amount of sealant as well as the pressure being injected into the valve. Controlling the amount of sealant and pressure ensures valve component stresses will not be increased to values higher than code allowable stresses and that the sealant will not be introduced into the piping, in a manner that could cause the piping to be plugged or excessive sealant to be injected into the reactor vessel.

Evaluation Number: 2001-0056-R00

Document Evaluated: ER 1999-0242-00-00 (LDC 2000-046 & 047)

BRIEF DESCRIPTION OF CHANGE, TEST OR EXPERIMENT:

ER 1999-0242-00-00 makes necessary modifications and documents the acceptability of two non-standard ventilation duct penetration configurations. ER 1999-0242-00 addresses penetration CV-115D-A, which separates the Unit 1 Support Area (0C404, Elev. 148') from the Computer and Control Panel Room (0C403, Elev. 148'), and penetration CV-59G, which separates the Upper Cable Spreading Room (0C702, Elev. 189') from the Instrument Motor Generator Room (0C707, Elev. 189').

REASON FOR CHANGE, TEST OR EXPERIMENT:

The fire protection program and the UFSAR describe the barriers containing penetrations CV-115D-A and CV-59G as 3-hour rated fire barriers. All penetrations through rated fire barriers are required to be sealed at least to the rating of the barrier. Penetrations CV-115D-A and CV-59G utilize a non-standard penetration closure that does not have a quantifiable fire resistance rating. Installation of qualified ventilation duct penetrations is not possible for these penetrations due to the inaccessible nature of one side of the penetrations. In accordance with Generic Letter 86-10, the adequacy of these penetration closures has been evaluated and documented in Fire Protection Evaluations 2000-0004 and 0005.

SAFETY EVALUATION SUMMARY AND CONCLUSIONS:

License Condition 2.C.41 allows GGNS to make changes to the approved Fire Protection Program through the 50.59 process if those changes do not adversely affect the ability to achieve and maintain safe shutdown in the event of a fire. Therefore, from the fire protection standpoint the basis for evaluation is "no adverse effect on the ability to achieve and maintain safe shutdown in the event of a fire". Generic Letter 86-10, Enclosure No.1, Interpretation No.4 states: "Where fire area boundaries are not wall-to-wall, floor-to-ceiling boundaries with all penetrations sealed to the fire rating required of the boundaries, licensees must perform an evaluation to asses 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 Evaluations 2000-0004 & 0005, the non-standard duct penetration closures for penetrations CV-59G and CV-115D-A have been evaluated to be capable of withstanding the hazards associated with areas on either side of each penetration. Thus, the ability to achieve and maintain safe shutdown conditions in the event of a fire, as presently analyzed in the UFASR, has not been adversely affected.

Evaluation Number: 2001-0057-R00

Document Evaluated: ER 2000-0166-00-00 (LDC 2000-049)

BRIEF DESCRIPTION OF CHANGE, TEST OR EXPERIMENT:

ER 2000-0166 makes necessary modifications and documents the acceptability of a non-standard penetration seal separating 0C702 (Unit I Upper Cable Spreading Room) and 0C706 (Corridor). ER 2000-0166 also documents the acceptability of existing portions of non-standard fire barrier configurations separating 0C702 from 0C703 (Control Cabinet Area) and 0C703 from 0C706. The existing non-standard barrier configurations consist of areas (approximately 2' x 3.5' x Wall thickness) where structural steel members pass through the wall. The structural steel members have three hour structural steel fire proofing material applied on both sides of the barriers to provide the fire separation between the areas.

ER 2000-0166 documents the acceptability of removing the fire rating from the North, South and East walls of 0C617 and 0C619; the floor of 0C601, 0C617 and 0C619; and the East walls of 0C618 that interface with 0C601. The above described barriers were evaluated against GGNS fire barriers requirements and determined to not be required to be fire rated, therefore, their fire ratings have been removed. With the barriers no longer fire rated, previously identified non-conformances are acceptable as is with no additional fieldwork. The West walls and ceilings of 0C617 and 0C619 were evaluated and determined to be required to be fire rated barriers. While the West walls of 0C617 and 0C619 are still considered fire rated barriers, ER 2000-0166 provides a basis for no longer applying TRM operability requirements to the West wall of 0C617 and 0C619.

REASON FOR CHANGE, TEST OR EXPERIMENT:

The fire protection program and the UFSAR describe the barriers separating 0C702(FA47)/0C703(FA53) and 0C703/0C706(FA58) as rated fire barriers. The fire protection program and the UFSAR describe the barrier separating 0C702/0C706 as a non-standard barrier that has been evaluated and been found acceptable for the hazards involved in the areas. All penetrations through rated fire barriers are required to be sealed at least to the rating of the barrier or be evaluated to establish adequacy for the hazards in the area. A penetration through the barrier separating 0C702/0C706 utilizes a non-standard penetration seal that does not have a quantifiable fire resistance rating. Additionally, a portion of the barriers separating 0C702/0C703 and 0C706/0C703 are non-standard (i.e. not tested) configurations that do not have quantifiable fire resistance ratings. In accordance with Generic Letter 86-10, the adequacy of the non-standard penetration closure, as repaired by ER 2000-0166, and the non-standard barrier configurations has been evaluated and documented in Fire Protection Evaluation 2000-0006. This change documents that evaluation and makes necessary Fire Protection Program Changes to reflect the non-standard fire barrier configuration.

A number of deficiencies were identified in electrical chases 0C617 and 0C619. The chases are physically inaccessible, therefore, field actions could not be taken to resolve any of the identified issues. Fire Protection Engineering reviewed the deficiencies and fire barriers involved and determined that the West wall and ceiling were the only barriers in each chase that were required to be maintained as fire rated. In order to maintain the West wall of each chase as a two hour rated fire barrier and the ceiling as a three hour fire barrier, larger hatches will be required to be installed for personnel entry into the chases. With access gained, the non-conformances in the West wall and ceiling of each chase will be reworked. Due to the removal of the fire rating of the identified barriers in 0C617 and 0C619, other identified non- conformances were accepted as is with no additional field action required. This change documents the barrier rating removal evaluations and makes necessary Fire Protection Program Changes to reflect the removal of the fire rating of the identified barriers.

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SAFETY EVALUATION SUMMARY AND CONCLUSIONS:

License Condition 2.C.41 allows GGNS to make changes to the approved Fire Protection Program through the 50.59 process if those changes do not adversely affect the ability to achieve and maintain safe shutdown in the event of a fire. Therefore, from the fire protection standpoint the basis for evaluation is "no adverse effect on the ability to achieve and maintain safe shutdown in the event of a fire". Generic Letter 86-10, Enclosure No.1, Interpretation No.4 states: "Where fire area boundaries are not wall-to-wall, floor-to-ceiling boundaries with all penetrations sealed to the fire rating required of the boundaries, licensees must perform an evaluation to asses the adequacy of fire boundaries in their plants to determine if the boundaries will withstand the hazards associated with the area". As documented in Fire Protection Evaluation 2000-0006, the non-standard penetration seal for penetration CE-456G and the identified barrier configurations separating 0C706 from 0C703 and separating 0C703 from 0C702, have been evaluated to be capable of withstanding the hazards associated with areas on either side of the non-standard barrier configurations and penetration CE-456G.

ER 2000-0166 provides the basis for removing the fire rating from the North, South and East walls of 0C617 and 0C619; the floor of 0C601, 0C617 and 0C619; and the East walls of 0C618 that interface with 0C601. The barriers were compared to applicable regulatory and site insurance requirements and found not required to be fire rated. The removal of the fire rating of the identified barriers allows previously identified non-conformances to be accepted as is with no field action required to be taken. As a result of the 0C617 and 0C619 barrier evaluation, it was determined that the West wall and ceiling of each chase will be required to be maintained as fire rated barrier configurations. However, the West walls of each chase, which are currently considered TRM, were reviewed against the TRM fire barrier criteria and determined to longer be considered TRM required.

On the basis of the evaluations provided in ER 2000-0166 and FPE 2000-0006 the proposed changes are considered acceptable. Thus, the ability to achieve and maintain safe shutdown conditions in the event of a fire, as presently analyzed in the UFASR, has not been adversely affected.

Evaluation Number: 2001-0058-R00

Document Evaluated: ER 2000-0093-00-00 (LDC 2001-098)

BRIEF DESCRIPTION OF CHANGE, TEST OR EXPERIMENT:

As indicated in ER 2000-0093-00-00, this change: 1) Accepts-As-Is the potential for not having internal conduit fire/smoke seals in specific conduits located internal to inaccessible HVAC Chase 0C217 and Electrical Chase 0C609 and in the 3-hour rated fire wall separating the Instrument Motor Generator Room 0C407 from HVAC Chase 0C409A. 2) De-rates from 3-hour to 2-hours the fire resistance rating of the firewalls separating individual fire pump rooms located in the Fire Pump House. 3) Completely de-rates and removes from the Fire Protection Program various fire barriers located in the Control Building that were originally designed for Unit 2 and are not required for Unit I operation or was originally designated as fire rated structures but are not required for regulatory or insurance purposes and serve no fire protection purpose. 4) Accept-as-is specific locations where small portions of the 3/4" thick Portland cement plaster was not installed on concrete block walls, as the design required to obtain a 3-hour rated configuration. 5) Revised various design drawings to eliminate references to Unit 2 construction and more clearly show the as-built plant configurations. As a result of updating documentation to reflect as-built configurations with respect to Unit 2, combustible loading in these areas were updated to reflect as-built configurations. Combustible loading in a total of 17 fire zones was revised. In all cases the total severity of the combustible loading (i.e. Low, Moderate or Severe) did not change and the in situ combustible loading in all but 3 areas actually decreased. In Fire Zones 0C204, 0C214 and 0C406 the in situ combustible loading increased above that identified in the Fire Hazards Analysis (FHA).

In addition, CA10 to CR-GGN-1997-1120 required revising the heat load calculation and FHA for the Health Physics Area (Fire Zone 0C101) located on elevation 93'-0" in the Control Building. Combustible loading increased in severity hazard from Low to Moderate as a result of this change.

REASON FOR CHANGE, TEST OR EXPERIMENT:

HVAC Chase 0C217 and Electrical Chase 0C609 are totally enclosed chases with no personnel access openings. The walls of these chases are fire rated barriers as described by the Fire Protection Program at GGNS and have electrical conduits that penetrate them. These conduits are required to have intemal conduit fire seals installed at the first available opening on each side of the barrier to maintain the fire resistance rating of the barrier. Since these are totally enclosed chases, personnel access into the chases to verify proper fire sealing of the conduits is not possible without removal of a portion of the wall. A similar condition exist in the Instrument Motor Generator Room 0C407 where access to the end (first available opening) of a 2" conduit is not possible without removal of the dry type transformer to which it is attached. Therefore, this evaluation provides the justification for accepting-as-is the potential for not having the required internal conduit fire seals in these specific locations.

Numerous locations on the surface of the 3-hour rated fire walls separating Fire Pump Rooms 0M101, 0M102 and 0M103 were identified where the required 3/4" thick plaster does not exist due to obstructions from electrical components and other structural members. The design of these firewalls consists of a 2-hour rated concrete block wall with 3/4" thick Portland cement plaster applied to both sides to provide the 3-hour rating. A similar but less extensive condition exists on 3-hour rated fire barriers enclosing the Division I, II & III Battery Rooms (0C207, 0C211 & 0C209 respectively) that are located on the 111 '-0" elevation in the Control Building. These deficiencies are not easily reworked without removal of permanently installed plant equipment. Therefore, this evaluation provides the justification that the base 2-hour rated concrete block walls in the Fire

Pump House are adequate for the hazards in these areas and the deviations with the plaster on the fire walls of the Division I, II & II Battery Rooms are so minimal that they do not adversely affect the 3-hour rating of the barriers.

GGNS was originally designed to be a 2-unit site. The Control Building was to be a shared structure for both units. Fire rated barriers located on the Unit 2 side of the Control Building and designed for Unit 2 were indicated on design drawings and described in the Fire Hazards Analysis (FHA) and thus incorporated into the Fire Protection Program at GGNS. Maintenance of these nonessential fire barriers takes resources away from fire barriers that are necessary. These barriers are not required for Unit I operation and serve no fire protection purpose. Therefore, they are being derated for fire protection purposes. In addition, design drawings and the FHA are being revised to eliminate non-essential references to Unit 2 design features and more clearly show the as-built plant configuration.

Combustible Loading Calculation MC-QSP64-86058 and the FHA are being updated to reflect as-built configurations with respect to Unit 2. In addition, this calculation and the FHA are being revised to reflect increased combustible loading in the Health Physics Area (Fire Zone OCIOI) resulting from CA10 to CR-GGN-1997-1120.

SAFETY EVALUATION SUMMARY AND CONCLUSIONS:

License Condition 2.C.41 allows GGNS to make changes to the approved Fire Protection Program through the 50.59 process if those changes do not adversely affect the ability to achieve and maintain safe shutdown in the event of a fire. Therefore, from the fire protection standpoint the basis for evaluation is "no adverse effect on the ability to achieve and maintain safe shutdown in the event of a fire". Generic Letter 86-10, Enclosure No.1, Interpretation No.4 states: "Where fire area boundaries are not wall-to-wall, floor-to-ceiling boundaries with all penetrations sealed to the-fire rating required of the boundaries, licensees must perform an evaluation to asses the adequacy of fire boundaries in their plants to determine if the boundaries will withstand the hazards associated with the area".

The degraded fire barriers associated with HVAC Chase 0C217, Electrical Chase 0C609 and in the 3-hour rated fire wall separating the Instrument Motor Generator Room 0C407 from HVAC Chase 0C409A have been evaluated and documented in Fire Protection Evaluation No. 2000-010. This evaluation concluded the existing fire barriers, with the deficient internal conduit seals, are capable of preventing propagation of fire across these fire barriers; therefore, the existing fire barriers are adequate for the hazards in the areas. This conclusion was based on the following: 1) low combustible loading in areas adjacent to the deficient barriers. 2) No combustible loading in Chases 0C217 and 0C608. 3) Area wide smoke detection and accessibility to manual hose streams and extinguishers in fire zones presenting an exposure fire potential to these barriers and an automatic suppression system in 0C407.

ER 2000-0093 provides the basis for de-rating from 3-hours to 2-hours the fire resistance rating for the firewalls separating fire pumps at GGNS. Combustible load calculations show that the combustible loading in these areas is less than the 2-hour rating of concrete block walls. Therefore, the fire barriers are adequate for the hazards in the area. Fire Protection Evaluation 2000-011 provides the basis for accepting-as-is the minor deficiencies with the plaster fire walls enclosing the Division I, II & III Battery Rooms. This evaluation concluded the deficiencies noted are minimal and do not adversely affect the 3-hour fire rating of the barriers. This conclusion is based on the fact that the percentage of the wall surface area affected was extremely small (maximum of <0.8%) and the discrepancies were limited to the surface of the barrier and were not through openings.

ER 2000-0093 provides the basis for removing the fire rating for various barriers located in the Control Building on elevations 111'-0", 148'-0" and 189'-0". The need for these barriers to be fire rated was compared to applicable regulatory and site insurance requirements. The barriers de-rated for fire purposes are not required or needed.

In situ combustible loading in Fire Zones 0C204, 0C214 and 0C406 loading increased above that identified in the Fire Hazards Analysis (FHA). However, the overall fire load severity as described in the FHA did not change. Therefore, fire barriers, detection and suppression provided for these area remain adequate.

The total fire severity in the Health Physics Area (Fire Zone 0C101) located on elevation 93'-0" in the Control Building increased from Low to Moderate. However, the rating of the fire barriers separating this area from other areas containing safe shutdown components remains greater than the loading in the area. Therefore, a fire in this area will be contained and can not propagate to damage redundant safe shutdown components.

Based on the above and evaluations provided in ER 2000-0093, Fire Protection Evaluations 2000-0010 and 2000-0011 and CA 10 to CR-GGN-1997-1120, the proposed changes are considered acceptable. Thus, the ability to achieve and maintain safe shutdown conditions in the event of a fire, as presently analyzed in the UFASR, has not been adversely affected.

Evaluation Number: 2001-0059-R00 Document Evaluated: LDC 2001-089

BRIEF DESCRIPTION OF CHANGE, TEST OR EXPERIMENT:

SDC accuracy review and the UFSAR consistency review for the Fuel Pool Cooling and Cleanup (FPCCU) & FPCCU Filter/Demineralizer systems identified several discrepancies with design documents. The proposed change will revise UFSAR applicable sections to resolve these discrepancies. The changes include correcting typo in SSW system load table, correcting valve locations on P & ID, revising P & ID to delete input to alarm from pressure switches, correcting pipe length from containment to isolation valve, correcting system capacity in Btu/hr, clarifying system description, clarifying system operation, clarifying system safety evaluation and clarifying environmental conditions of the system for accident conditions.

REASON FOR CHANGE, TEST OR EXPERIMENT:

These discrepancies were due to previous UFSAR changes, P & ID revision and incorrect information. All changes have been reviewed with design documents (SDC, P & IDs and drawings) and are software related only. These changes will not affect equipment function or performance of the FPCCU system.

SAFETY EVALUATION SUMMARY AND CONCLUSIONS:

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

Evaluation Number: 2001-0060-R00

Document Evaluated: ER 2000-0240-00-00
(LDC 2000-015 & 016)

BRIEF DESCRIPTION OF CHANGE, TEST OR EXPERIMENT:

ER 2000/0240-00-00 is being issued to install an additional Standby Liquid Control System Waste Station in the Auxiliary Building in Area 10, at Elevation 166'-0". The waste station with the required pipe and pipe supports will be installed as Suspended Floor and Equipment Drain System (P48) piping.

There currently exist several spare ¾" pipes penetrating the Containment wall into the Auxiliary Building at Penetration M-111C. A ¾"" branch connection off the existing 4" SLCW drain line for the Standby Liquid Control Waste Station will allow one of the ¾" pipes to be used as a drain line for the new Auxiliary Building Standby Liquid Control System Waste Station. The existing Standby Liquid Control Waste Station located in Containment, Area 11, El. 1 73'-2", is to remain functional for continued use in the event it is needed. However, the new Auxiliary Building Standby Liquid Control Waste Station will be the primary SLCW Station.

The Standby Liquid Control Waste Station is the collection point for the Boron Solution generated after surveillances are completed on the SLC System. There is no physical tie-ins between the SLC System and the Standby Liquid Control Waste Station in either the existing Containment SLCW Station or the new Auxiliary Building SLCW Station. The modification as given in ER 2000/0240-00-00 will not alter the operation of the SLC System or any of its components.

REASON FOR CHANGE, TEST OR EXPERIMENT:

Currently Boron barrels (55 gallon drums) from the Standby Liquid Control System Waste Station located in Containment, Area 11, El. 173'-2", have to be moved in and out of Containment each time the barrels are filled and then drained. This constitutes a safety hazard, as the barrels have to be lifted with the jib crane located on El. 208-10" or carried up and down stairs while full of Boron Solution. Personnel and plant equipment safety could be compromised, or put at unnecessary risk during these barrel moves. The proposed solution to this problem is the installation of an additional Standby Liquid Control Waste Station located in The Auxiliary Building, Area 10, El. 166-0". Removal of the barrels would be accomplished using the Auxiliary Building and/or Turbine Building elevators.

SAFETY EVALUATION SUMMARY AND CONCLUSIONS:

The Standby Liquid Control System Waste Station located in Containment, Area 11, El. 173-2" will be installed as Suspended Floor and Equipment Drain System (P48) piping. The Auxiliary Building Standby Liquid Control Waste Station will utilize isolation valves to ensure minimum design requirements for Containment isolation. These valves will be locked closed when the SLCW System is not in operation as a means of providing sealed closed containment isolation valves. Per Section 2 of ANS-56.2/N271 -1976, a sealed closed isolation valve is "a valve that is in a closed position by administrative controls by any of the following methods;

A mechanical device sealing or locking the valve in the closed position.

A normally closed valve with a seal or lock on any manual over-ride if present and a seal or lock on the power source in a manner that prevents power from being supplied to the valve.

The containment isolation valves installed per this ER will be manual valves (Q1P48F009, and Q1P48F01 0) with mechanical devices locking the valve in the closed position when the SLCW System is not operational.

As previously stated there are no physical tie-ins between the SLC System and the Standby Liquid Control Waste Station in either the existing Containment Station or the new Auxiliary Building Station. Boron is physically drained via a detachable hose to 55-gallon drums (reference Section 9.3.5.2 of the UFSAR). The modification as given in ER 2000/0240-00-00 will not alter the operation of the SLO System or any of its components. This modification will utilize an existing spare ¾" pipes penetrating the Containment wall into the Auxiliary Building at Penetration M-111C. A branch connection off the existing 4" SLCW drain line for the Standby Liquid Control Waste Station will allow this 3/4" pipe to be used as a drain line for the new Auxiliary Building Standby Liquid Control System Waste Station.

This modification will require a change to an UFSAR figure (Figure Number 9.3-016, P&ID Diagram Embedded & Suspended Drains Auxiliary Building). No components or descriptions of components mentioned in the SAR for either the Floor and Equipment Drainage Systems, or the Standby Liquid Control System will be changed because of this modification (reference UFSAR Sections 9.3.3 and 9.3.5). Manual isolation valves Q1P48F009 and Q1P48F010, added per this modification, have been added to Table 6.2-44 and Table 6.2-49 the UFSAR, and Table TR3.6.1 .3.1 of the TRM (reference LDCR 2001-097).

The modifications made by ER 2000/0240-000-00 will in no way impact any of the accident analysis presented in the FSAR. No new failure modes are being created thus any possibility of an accident or malfunction of a different type than previously analyzed is not possible. Failure of the P48, Suspended Floor and Equipment Drain System, will not compromise any safety-related system or component, and will nor prevent safe reactor shutdown. The existing margin of safety is not reduced. The piping and fittings installed by this design change meet ASME B&PV Code Section III, Safety class 2, seismic category I, and ANSI B31.1, seismic category II/I requirements.

Evaluation Number: 2001-0061-R00   Document Evaluated: LDC-2001-083

BRIEF DESCRIPTION OF CHANGE, TEST OR EXPERIMENT:

Revises the bases of Technical Specification surveillance requirement 3.6.4.1.4 to reflect the testing criteria developed in Calculation 3.9.12, rev. 2 and incorporates these into the associated surveillance procedures. The proposed TS Bases also include an editorial change to clarify the discussion on the staggered test basis. No additional surveillance testing is required as a result of these changes since the results of 06-OP-1T48-R-0002, Att. II, was completed on 4/8/01. During this test the recorded values ranged from 0.55 in. w.g. vacuum to 0.78 in. w.g vacuum with a max flow rate of 3800 cfm. These values exceed the revised surveillance criteria.

REASON FOR CHANGE, TEST OR EXPERIMENT:

This change is necessary to resolve the condition identified in GGNS CR-1999-0360.

SAFETY EVALUATION SUMMARY AND CONCLUSIONS:

CR 1999-0360 questioned the validity of the TS surveillance criteria for SR 3.6.4.1.4. Calculation 3.9.12 (Revision 2) was revised to establish the basis for the TS surveillance value and to determine the definitive testing requirements for the secondary containment boundary. These requirements are incorporated into the TS bases and the associated implementing procedures by the proposed change. The proposed TS Bases also include an editorial change to clarify the discussion on the staggered test basis.

This safety evaluation evaluates the revised calculation as well as changes to the Technical Specifications Bases 3.6.4.1.4. The proposed TS bases changes reflect the revised calculation by establishing that the SGTS steady state drawdown requirement is based on an allowance for building degradation between performances of the surveillance. The proposed changes also establish that a configuration-specific allowance for assumed failures is required to ensure that the system testing is consistent with the calculated inleakage associated with these assumed failures. These adjustments ensure that the system will perform as designed in response to postulated failures following a design basis accident. The evaluation concluded that the proposed changes are consistent with existing accident assumptions for the secondary containment performance and that an unreviewed safety question does not exist.

Evaluation Number: 2001-0062-R00 Document Evaluated: ER 2001-0158-00-00: (LDC 2000-078)

BRIEF DESCRIPTION OF CHANGE, TEST OR EXPERIMENT:

ER 2001-0158-000-0 provides a modification for alteration of the Standby Liquid Control (SLC) system for system operation without actively functional pulsation dampers (a.k.a., damper or accumulator) and with system pump discharge piping design pressure (and relief valve setpoint) changed to 1700 psig. The ER also provides new high point vents for the SLC system. Out of necessity, the ER requires that the modifications made to each SLC train be done separately. Operation with one train with a relief valve setpoint of 1700 psi and the other with a setpoint of 1500 psi has been evaluated and found to be acceptable.

REASON FOR CHANGE, TEST OR EXPERIMENT:

The SLC system is required to be capable of injecting the neutron absorber into the reactor via the High Pressure Core Spray (HPCS) sparger with both SLC pumps running simultaneously. To assure that the relief valve set pressure will not be exceeded during pump operation with the pump discharge path open, pulsation dampeners were installed by Reference 5 on the discharge piping of each SLC pump.

CR GGN-2001-0565 documented the failure of the "A" SLC system pulsation dampener internals. Subsequent inspection of the "B" dampener internals determined that their material condition was unacceptable for continued service. The "B" dampener internals were replaced and there are now no spare parts available. This ER response provides the required configuration changes and the justification for a modification to the SLC system (which increases the pump discharge piping design pressure and relief valve setpoint to 1700 psig) which would result in system operation without actively functional pulsation dampeners.

Also, recent events have highlighted the difficulties involved with adequately filling and venting the SLC system after maintenance. This ER response includes the addition of high point vents on the system to facilitate SLC system post maintenance fill and vent.

SAFETY EVALUATION SUMMARY AND CONCLUSIONS:

The original design function of the SLC system was to serve as an independent backup reactivity control system to the Control Rod Drive system (but not the "SCRAM" function). The system was designed to ensure reactor shutdown from full power operation to cold subcritical, without control rod movement. This is achieved by pumping 5803 pounds of sodium pentaborate into the reactor in 90 to 120 minutes. This is sufficient boron to keep the reactor sub-critical during the most limiting conditions including a cold moderator, the absence of xenon and other poisons and at the most reactive point in core life. The SLC system was designed to accomplish this function with a single active component failure.

Subsequently, IOCFR5O.62 (the "ATWS" rule) imposed additional requirements on the SLC system (i.e. simultaneous two pump operation at 86 gpm system flow with 13% sodium pentaborate concentration or equivalent). This function was not required to be single failure proof.

The ability of the SLC system to meet its design functions with these modifications is within the capabilities of the modified/rerated equipment. The guidance for modifying the dampeners and other components/piping assures complete system operability. The system will continue to be capable of injecting the neutron absorber into the reactor (with either one or two pumps operating) and will impose no additional challenges to any other equipment. The SLC system design functions will continue to be met. The operation of the SLC system without functional pulsation dampeners (or accumulators) was evaluated and a minimum margin of 257 psi will exist between the maximum theoretical SLC pump operating pressure during an ATWS and the revised minimum relief valve setpoint which assures that the SLC pump discharge relief valves will not open during SLC system operation with ATWS (and other) conditions. The components and piping within the affected portion of the SLC system have been evaluated and rerated as necessary for this new design pressure (1700 psig).

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

Evaluation Number: 2001-0063-R00 Document Evaluated: ER 2001-0228-00-00

BRIEF DESCRIPTION OF CHANGE, TEST OR EXPERIMENT:

To access the Reactor Water Cleanup (RWCU) Filter/Demineralizer (F/D) plugs to perform maintenance on the RWCU F/D, the Reactor Vessel Head (RPV) Head Carousel (Strongback), !F13E009, requires temporary relocation. ER-GG-2001-0228-00 has evaluated and provides instructions for lifting and moving it the strongback and F/D plugs to a temporary location on the concrete refuel floor (containment elevation 208'-l0").

REASON FOR CHANGE, TEST OR EXPERIMENT:

In its current location, the Strongback straddles the floor plugs directly above the RWCU F/D, thereby blocking access to the F/Ds for maintenance functions. In order to remove the floor plugs, the Strongback is required to be temporarily moved. The strongback weighs approximately 19 tons, that is greater than 1,140 lbs, falling under the auspices of NUREG 0612 for heavy loads.

SAFETY EVALUATION SUMMARY AND CONCLUSIONS:

The maintenance of the RWCU F/D involves moving the Strongback from its permanent storage location to a temporary storage location for allowing access to the floor plugs that must be removed to allow access to the RWCU F/Ds. The UFSAR specifies the storage location during reactor operation. ER-GG-2001-0228 evaluates temporary relocation of the strongback that will be moved to a temporary location on the refueling floor for the required time period to perform the maintenance on the RWCU F/D. Movement of the strongback has been postulated in the UFSAR but only during safe shutdown of the reactor (Modes 4 & 5). The accident evaluated during movement of the strongback, UFSAR Section 9.1, assumes the RPV head, strongback; nuts and washers are dropped from the highest lift point, onto the refueling floor during transport. This drop has been determined to cause no significant damage to the floor and no damage to any other plant equipment. The proposed move of only the strongback to the approved temporary location has been evaluated in ER-200 1-0228-00 during Mode 1. The height has been limited during the movement to a maximum of six feet to assure compliance with the GGNS Heavy Rigging requirements. Therefore, the proposed move does not represent either a new or greater risk to plant safety, via postulated or non-postulated accident, equipment failure, or a change in Technical Specification margins.

Evaluation Number: 2001-0064-R00

Document Evaluated: ER 2000-0077-00-00 (LDC-2000-046)

BRIEF DESCRIPTION OF CHANGE, TEST OR EXPERIMENT:

The length of fire hose will be increased on hose stations HS-16E, HS-18B, HS-18C, HS-19B, HS-19C, HS-22A and HS-53C. Hose station HS-19C will be relocated to remove a seismic II/I concern.

REASON FOR CHANGE, TEST OR EXPERIMENT:

As documented in CR-GGN-1 999-0947, there is inadequate fire hose on hose stations HS-16E, HS-18B, HS-18C, HS-19B, HS-19C, HS-22A and HS-53C to reach all areas covered by the individual hose stations. Additionally, CR-GGN-2001-0457 documents that hose station HS19C presents a seismic II/I hazard to pressure transmitter E32N059.

SAFETY EVALUATION SUMMARY AND CONCLUSIONS:

The addition of fire hose to HS-16E, HS-18B, HS-18C, HS-19B, HS-19C, HS-22A and HS-53C will not introduce any undocumented seismic II/I issue. Note that during completion of this ER, it was determined that hose station 19C created a seismic II/I concern with E32N059. This is documented on OR 2001-0457 and this ER will relocate the hose station. The additional hose on HS-22A will not adversely affect the ECCS suction strainer. The modifications do not adversely affect any safety related system and will introduce no new failure modes nor will they increase any previously identified failure modes. No safety related equipment is compromised by this modification and it will not prevent safe reactor shutdown. The changes do not affect the operation of any safety related component. These changes do not 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 of an accident of a different type than any previously evaluated in the SAR, create the possibility of 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.

Evaluation Number: 2001-0065-R00

Document Evaluated: ER 2000-0193-00-00 (LDC-2000-015)

BRIEF DESCRIPTION OF CHANGE, TEST OR EXPERIMENT:

ER 2000-0193 replaces the existing Reactor Water Cleanup (RWCU) resin metering pump, 1G360003, with a smaller pump. In order to accomplish this, the pump supply and discharge lines as well as the air supply line require rerouting. The inlet and outlet size for the pump is different than the previous model. This necessitates a P&ID change, M-1080A - SAR Figure 5.4-025, which results in a SAR figure change.

REASON FOR CHANGE, TEST OR EXPERIMENT:

Both the existing pump and the replacement pump are air-powered pumps. The existing pump is sized for a minimum air pressure of 45 psig. The pump's air supply is currently throttled to 8 psig to maintain the water/resin mixture flowrate within desired limits. The replacement pump has been sized to provide the desired flowrate using 60 psig supply air.

SAFETY EVALUATION SUMMARY AND CONCLUSIONS:

There are no accidents evaluated in the SAR related to the RWCU resin metering pump or its appurtenances. Secondary and indirect effects of this modification (fire protection, area fire loading, potential missiles, pipe break) have been reviewed and no increased probability of failure of equipment important to safety due to these concerns has been identified. No changes are being made to the basic system design functions. The changes meet the original design criteria of the system. The modification does not affect system integrity, capacity, fire protection, fire hazards, room temperature during a DBA, area temperature monitoring or missile protection. The change does not result in a new pathway for release of radioactive material and does not affect offsite dose. No assumptions utilized in evaluating the consequences of an accident are changed. No new failure modes are created and there is no increase in previously identified failure modes for equipment which is important to safety. This design change will not compromise any safety related system or component and will not prevent safe reactor shutdown since the design does not create any new interface with equipment important to safety, prevent such equipment from operating as designed, or cause equipment important to safety to operate outside design requirements. No assumptions utilized in evaluating the consequences of a malfunction of equipment important to safety will be altered by this modification. The changes will not compromise the function of any safety related system or component or prevent safe reactor shutdown. It is concluded that the change does not increase the probability of occurrence or increase the consequences of malfunction of equipment important to safety of a different type than previously evaluated in the SAR. The Technical Specifications are unaffected and the margin of safety remains unchanged.

Evaluation Number: 2001-0066-R00 Document Evaluated: ER GG-1999-0342-00-00

BRIEF DESCRIPTION OF CHANGE, TEST OR EXPERIMENT:

Replace existing Riley temperature switches currently installed in tire Control circuits of the T46 ESF Electrical Switchgear Room Coolers with switches manufactured by S. Levy, Incorporated. The new switches utilize the switch housing supplied with the original Riley switches but are microcontroller-based with digital readout. These switches will provide improved adjustment control for the cooler circuits.

REASON FOR CHANGE, TEST OR EXPERIMENT:

CR-GGN-1999-0518-00 identified a condition with the circuit breaker for ESF Switchgear Room Cooler 1T46B004A. The breaker was found tripped open and excessive cycling of the unit was determined to be the cause of the problem. Further, the condition was attributed to the temperature switch control for the unit. It is believed that the narrow, fixed hysteresis adjustment was causing the unit to cycle more frequently than required. This ER replaces each of the existing T46 Riley temperature switches with a new switch made by S. Levy, Inc. The S. Levy switch has an adjustable hysteresis between I and 50 degrees Fahrenheit. This replacement will allow better switch adjustment and is initially being performed for all T46 ESF switchgear room cooler circuits.

SAFETY EVALUATION SUMMARY AND CONCLUSIONS:

Based on the information included in the Technical Specifications and UFSAR, no LBD revisions are required. Information included in these documents is not specific to the type of switch utilized or to exact configurations or part numbers. This information includes the requirements of the system to maintain room temperatures at particular values and defines specific operability requirements for the coolers. ER modifications will not impact the precise setpoints or performance requirements and system operating logic will not be changed. Area temperature and cooler operability requirements are not changed. Though both Divisions I and 2 will be modified, recovery from any switch failure is achievable by use of manual control functions. Should the automatic control function be lost, the manual RUN logic of the control circuits remains available for use. No adverse environmental impact is created by this package. Software Quality Assurance requirements of Procedures 01-S-17-46 and IT-104 have been met, and EPRI TR-102348 digital upgrade guidelines considered.

Evaluation Number: 2001-0067-R00 Document Evaluated: LDC 2001-100

BRIEF DESCRIPTION OF CHANGE, TEST OR EXPERIMENT:

The UFSAR consistency review for the Standby Diesel Generator System identified a discrepancy in Section 9.5.6.3 of the UFSAR. The propose change will revise the applicable UFSAR section to clarify the function and the relationship between the air strainer and the pressure switches in the starting air piping at the DG engine inlet.

REASON FOR CHANGE, TEST OR EXPERIMENT:

Section 9.5.6.3 of the UFSAR states that the performance of the DGSS filters and strainers in each of the starting air supply lines is monitored by a pressure sensor located just upstream of the starting air solenoid valves and downstream of the final strainer in the piping. The actual configuration in the plant, as depicted on the P&ID (M-1070A-D & M-1093A-O), is such that pressure sensors monitor pressure upstream of both the starting air solenoid valves and the final strainer in each air supply line. The existing configuration provides local alarms in accordance with Regulatory Guide 1.97, Rev 2 when/if the starting air pressure in the supply piping between the air storage tanks and the engine inlet is low. It does not monitor the condition of the strainers in the supply piping due to the physical location of the pressure sensor in relation to the strainer. The performance of the individual strainers is routinely demonstrated as part of the monthly surveillance procedures, which verify, at least once per quarter, that individual starting air solenoid valves are capable of starting the engine. The air strainers are also periodically checked/ cleaned as part of the routine maintenance program.

SAFETY EVALUATION SUMMARY AND CONCLUSIONS:

The existing configuration with respect to the air strainers and the pressure sensors of the DG starting air system is considered to be adequate and has no adverse impact on the ability of the starting air system to perform its design safety function. This conclusion is based the fact that the strainers are periodically checked/cleaned as part of the routine maintenance program and the fact that the performance of the individual strainers is demonstrated at least once per quarter as part of the routine surveillance program. Changing section 9.5.6.3 of the UFSAR will clarify the function and the relationship between the pressure sensors and the air strainer. The change does not make any physical changes to plant configuration, design bases, design operating limits, or plant operating practices. The change made per LDO 2001-100 does not increase the probability or consequences of an accident or malfunction of equipment important to safety.

Evaluation Number: 2001-0068-R00 Document Evaluated: ER GG-2000-0086-02-00

BRIEF DESCRIPTION OF CHANGE, TEST OR EXPERIMENT:

Relocate a portion of the SSW “A” & “B” Basin PSW 8” makeup piping from below grade to above grade. Also, the piping entrance point for east wall of each basin will change to above grade and will enter the basins above the I 33’-0” valve room slab, which will entail adding a new penetration in the east wall of each basin and sealing with grout. In order to perform the modification on-line a hot tap isolation will be required for both basin 8” PSW make up lines. In order to perform the on-line isolation a vendor supplied split clamshell tee with a flanged branch will be required for each line to provide a means for installing the on-line isolation equipment. This clamshell tee will be welded in place and remain installed following the completion of the ER and will be shown on FSAR Figure 9.2-001. Also, remaining will be a blind flange installed on the flanged branch side of the clamshell tee and a friction type line plug will remain directly under the blind flange. This friction type line plug allows for removing on-line isolation equipment and for installing the blind flange on the flanged branch of the clamshell tee. This vendor supplied clamshell tee will also be designed to the same code requirements as the existing piping system (Ref. ANSI B31 .1). All underground piping including the clamshell tee will be coated to prevent the loss of piping and piping component structural integrity from corrosion and to prevent underground leakage from occurring.

The abandoned submerged piping will be cut as close as possible to the outside of the east basin wall and then will be filled with a minimum of 2’-0” of grout. In the valve room the existing pipe will be cut as close as possible to the floor penetration sleeve and filled with a minimum of 2’-0” of grout. This will assure that the piping exiting the SSW “A” & “B” basin east wall will not be a leak path in the future if the exterior of the submerged pipe was to corrode to the extent that a through wall hole would exist.

REASON FOR CHANGE, TEST OR EXPERIMENT:

This piping system re-route will eliminate a portion of the P5W 8” makeup line for basins A and B from being submerged in the basin inventory. This minor piping system re-route will eliminate the previously documented corrosion problem for the piping system, which resulted in minor pipe wall erosion (Ref. CR-GGN-1999-1505).

SAFETY EVALUATION SUMMARY AND CONCLUSIONS:

The minor piping system re-route for the SSW 8” PSW makeup piping will include the following. Relocate approximately 15 feet of underground and submerged piping for each basin to enter their respective valve room above grade, abandon a portion of the old submerged PSW 8” makeup piping for each basin and filling the pipe with grout to obtain a seal, adding a new wall penetration in the east wall of each basin for the re-routed piping to pass through and installing a vendor supplied Hot Tap On-line Flanged Isolation Split Tee on the underground PSW makeup yard piping to provide adequate isolation for each basin independent from the other basin. These minor changes will not have any adverse affect on the SSW “A” &“B” Basins to perform their safety related function nor will the changes keep the PSW makeup line from performing its required SSW “A” & “B” makeup function during an emergency condition if required. The piping system re-route and material requirements will be per the same design code as the original design. The modified small portion of the PSW makeup piping being located above grade will remain the same size. This change will eliminate a small portion of the PSW makeup 8” piping from being submerged in the basin water, which can result in the

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exterior of the piping system corroding and thus causing a reduction in the pipe wall. The aforementioned work will only be performed when the applicable basin is out of service and is in a scheduled workweek.

Evaluation Number: 2001-0069-R00 Document Evaluated: ER GG-1996-0086-00-00 (LDC 2001-086 & 2001-025)

BRIEF DESCRIPTION OF CHANGE, TEST OR EXPERIMENT:

Eberline Control Terminals SD17JIOO (TSC) and 5D17J600 (Control Room) used for effluent radiation monitoring as part of the Process Radiation Monitoring System (DI 7) are being removed from plant service. The monitoring and control functions performed by these terminals will be re-instated as part of the PDS computer system. This will allow greater access to monitored information, as well as better control capabilities and operation for system operators. Modifications to the system will include new man-machine interface, PDS data acquisition and new data server functions. The man-machine interface will perform Eberline data request functions. The PDS data acquisition mux will serve as the physical interface to communicate with Eberline SPING and AXM units. The data server will function to receive data requests and commands. The capabilities of this modification have been demonstrated by prototype testing with actual plant data and equipment.

REASON FOR CHANGE, TEST OR EXPERIMENT:

Eberline control terminals SD17J100 and 5D17J600, originally supplied in accordance with Design Specification J-366.0, are obsolete and have become unreliable because replacement parts are not readily available. The inoperable status of these terminals has often impeded plant system performance.

SAFETY EVALUATION SUMMARY AND CONCLUSIONS:

The modifications being implemented by this Engineering Request are Non-Safety Related, Non-seismic and Low Safety Significant, but applicable Q-List Appendix B requirements are specified. There will be no compromise of any Safety Related system, structure or component. There will be no changes made by this package that would prevent safe reactor shutdown. The modifications being made to this system will be reflected by the revisions of UFSAR Section 7.5.1.2.16 (Table 7.5-2), Section 11.5.2.2.4.1 and Section 18.1.27 (Table 18.1-3) and Figures 11.5-002 through 1.5-007. Modifications being made change neither the methods of field monitoring nor associated analytical components, but provide a more reliable location for compiling system data and controlling commands for system performance. Modifications include changes to control room annunciation that will expedite the response times of plant operators if high radiation conditions exist for SGTS parameters monitored to Technical Specification requirements.

Evaluation Number: 2001-0070-R00

Document Evaluated: LDC 2000-058
ER 2000/0073-01-00

BRIEF DESCRIPTION OF CHANGE, TEST OR EXPERIMENT:

Installation of two new fans (one supply, one exhaust) that bypass the existing supply and exhaust fans and one new refrigerant cooling unit with associated ductwork, controls, etc. in the safeguard switchgear and battery rooms ventilation system will be accomplished by ER 2000/0073-00-00, including Supplement I (identified as 2000/0073-01-00). Modification activities that are to be performed during plant operation, prior to refueling outage activities, are included in the base ER. Modification activities that are to be performed during a refueling outage to complete the installation are included in Supplement I. A separate *50.59* Evaluation will be performed for the base ER to provide a complete evaluation of the final system configuration. This evaluation addresses the activities of Supplement 1 only. As indicated on page 1, the validity of this evaluation is dependent on subsequent issuance of the base ER package with all necessary supporting document changes, analyses, and *50.59* evaluation. This is acceptable since the relationship between the base ER and the Supplement evaluated herein is such that the base ER must be implemented prior to or concurrent with Supplement 1.

LDC No. 2000-058 to revise the appropriate UFSAR sections, tables and figures to reflect the final system configuration will be issued with ER 2000-0073-00-00.

Calculation C-T151.0 Rev. 1 Supplement I is issued to address modifications to seismic duct supports. Calculations IB-325B-243; 1B-325B-394; IB-325B-486; IB-325B-501; 1B-325B-535; and IB-325B-536 are issued to address changes to seismic conduit supports. JC-Q I Z77-K600- 1 is revised to remove the outside air temperature setpoint for fan start and to revise the setpoint for heater activation. JC-Q 1Z77-N606- I Supplement 1 is issued to reflect the circuitry modifications that provide an automatic start of the safety-related fans if the high HVAC equipment room temperature limit is exceeded.

REASON FOR CHANGE, TEST OR EXPERIMENT:

The proposed changes included in ER 2000/0073-00-00 and Supplement 1 are made to reduce the wear on the existing safety-related fans resulting from continuous operation and speed changes. The modifications will allow the existing safety-related fans to be placed in a standby configuration with normal ventilation being supplied by the new equipment. A reduction in power consumption by the operating equipment is an additional benefit of the changes.

SAFETY EVALUATION SUMMARY AND CONCLUSIONS:

A complete evaluation of the modified system will be included with the base ER. The scope of this evaluation is limited to the modification activities that are to be accomplished by Supplement 1 (ER 2000/0073-01-00) during a refueling outage while the plant is shutdown and the affected portions of the Safeguards Switchgear and Battery Rooms Ventilation System and Plant Chilled Water System can be removed from service for an extended period of time. The ER requires that room temperature limits and hydrogen concentration limits be satisfied-during equipment out-of-service times. With regard to control room envelope and secondary containment integrity, opening of penetration seals will be controlled in a manner that insures the maximum opening size for control room envelope is not exceeded, and secondary containment integrity is maintained at all times when operability is required. The activities of Supplement I cannot be performed until after the base ER (2000/0073-00-00) is issued and implemented (or they must be implemented concurrently).

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Calculations have been performed, as necessary, to evaluate the impact of the changes listed above. The calculations have determined that the modified damper/duct configuration, conduit configuration and chilled water piping configuration (including associated conduit/duct/pipe supports) implemented under Supplement I satisfy all design requirements such as seismic qualification, stress analyses, etc. as necessary to maintain the integrity of the system. In addition, changes inside the safety-related panels have been evaluated and a determination made that the changes do not adversely affect qualification of the panels. Modification activities evaluated herein are to be performed only when plant conditions allow the affected systems to be removed from service for the period of time necessary to complete the changes (i.e.. during a refueling outage).

The changes made by Supplement I (ER 2000/0073-01-00) do not cause a greater reliance to be placed on any specific system, structure or component to perform a safety function. The changes do not degrade the performance or reliability of a safety system assumed to function in the accident analysis. The changes do not put the plant operation in an unanalyzed region. The changes herein are bounded by existing Technical Specification and TRM requirements. And, the changes do not adversely affect the overall performance or reliability of a safety system in a manner that could lead to an accident occurring.

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

Evaluation Number: 2001-0071-R00

Document Evaluated: LDC 1999-083
ER 1999/0311-01-00

BRIEF DESCRIPTION OF CHANGE, TEST OR EXPERIMENT:

The Tech Spec and TS Surveillance requirements for the Division I, II, and III DG lube oil storage requirements are being changed to require additional lube oil inventory so as to compensate for higher than nominal oil consumption rates. The Tech Spec Bases and UFSAR Section are being revised to clarify that, for Division III, the entire oil inventory required to meet the seven day operating requirement does not have to be contained within the engine sump. The amount of useable oil in the Division III engine sumps shown on UFSAR Table *9.5-6* is being corrected. A lube oil storage skid is being established in the Division III DG room to provide the additional dedicated and easily accessible additional oil inventory for the Division III engines. The new lube oil storage skid is being situated in the northernmost portion of the existing Division III DG room safe storage area on the west side of the room. The existing safe storage area is being reduced in size (SCN 0O-002A to CS-17.0) to allow for installation of the lube oil storage skid.

The addition of three *55* gallon barrels of lube oil and *75'* of rubber hose increases the fire duration in Fire Zone 1D306 from <*45* minutes to <60 minutes.

REASON FOR CHANGE, TEST OR EXPERIMENT:

CR GGN-1998-0733 identified the concern that the Division 3 diesel engine lube oil consumption rates were significantly greater than the nominal consumption rates provided by the engine manufacturer (0.6 gallons per hour). The existing Division III engine lube oil sumps are barely large enough to contain sufficient lube oil for seven full days of operation but provide no margin to allow for an increase in the consumption rate.

SAFETY EVALUATION SUMMARY AND CONCLUSIONS:

The increase to the Division I, II, and III DG engine lube oil storage requirements provides additional margin to ensure engine operability during design basis accidents. The additional required oil storage does not adversely impact the operation of any safety-related equipment. The lube oil storage skid is designed to meet the requirements of Standard CS- 17 and therefore will not create a Seismic III I hazard. Based on the low seismic loads in the diesel building at this location coupled with the stable configuration of the three barrels when tied together, adequate assurance exists that the barrels and steel containment pan will maintain their integrity during a seismic event. Additionally, there are no concerns with adverse interactions with adjacent structures or equipment during a seismic event.

Since the fire duration with the addition of 165 gallons of lube oil and 75' of rubber hose is less than 1 hour, the three hour rated walls between Fire Zone 1D306 and the adjacent areas of the diesel generator building, which contain safe shutdown equipment, will be more than adequate to prevent propagation of a fire from 1D306 to the adjacent areas. Therefore, the addition of 165 gallons of lube oil to Fire Zone 1D306 will not adversely affect the ability to achieve and maintain safe shutdown. The additional oil storage required for the Division I and II engines is within the capacity of existing engine sumps.

Evaluation Number: 2001-0072-R00 Document Evaluated: LDC 2001-141

BRIEF DESCRIPTION OF CHANGE, TEST OR EXPERIMENT:

This change deletes Section 6.3.5 of the TRM and UFSAR Appendix 16B.1. which provides operational and surveillance requirements for the LPMS. This change does not address LPMS modification or any other licensing basis document change. No physical change is being proposed by this 50.59 evaluation. It removes TRM and UFSAR Appendix 16B.1 LPMS requirements but does not affect any other UFSAR LPMS equipment requirements.

REASON FOR CHANGE, TEST OR EXPERIMENT:

Boiling Water Reactor Owners Group (BWROG) Topical Report NEDC—32975P, Revision 0, provides justification for the elimination of the requirements for the LPMS. The NRC approved the topical report in a letter to the BWROG dated January 25. 2001.

SAFETY EVALUATION SUMMARY AND CONCLUSIONS:

The LPMS provides a monitoring function only and is not involved in the initiation of any automatic system actuations. It is not safety—related and is not designed to mitigate or monitor offsite doses. BWROG Topical RePort NEDC-32975P and EPRI Topical Report EPRI TR—105707 both concluded that a LPMS does not provide safety significant benefits. Thus, this licensing basis document change will not result in more than a minimal increase in the frequency of occurrence or the consequences of an accident previously evaluated in the FSAR: result in more than a minimal increase in the likelihood of occurrence or the consequences of a malfunction of a structure, system, or component important to safety previously evaluated in the FSAR: create a possibility for an accident of a different type or a malfunction of a structure, system, or component important to safety with a different result than any previously evaluated in the FSAR: result in a design basis limit for a fission product barrier as described in the FSAR being exceeded or altered: nor result in a departure from a method of evaluation described in the FSAR used in establishing the design basis or in the safety analysis.

This licensing basis document change eliminates the TRM and UFSAR Appendix 16B.1 operational and surveillance requirements associated with the LPMS. No physical change is being proposed by this 50.59 evaluation. It removes TRM and UFSAR Appendix 16B.1 LPMS requirements but does not affect any other UFSAR LPMS equipment requirements. These requirements are contained in Regulatory Guide 1.133. An NRC letter to the BWROG documenting NRC review and approval of Topical Report NEDC-32975P states that the NRC finds the subject topical report acceptable for referencing in licensing applications. The letter also states that the staff will not repeat its review of the matters described in the subject report. There are no technical specifications associated with the LPMS and this change does not meet any of the criteria specified in 10CRF50.59(c)(2). Therefore, NRC approval via a license amendment pursuant to 10CFR50.90 prior to removal of LPMS operational and surveillance requirements from the TRM and UFSAR Appendix 16B.1 is not required.

CCE-2001-0004

Commitment Number: 34951 Source Document Number: GNRO-2000/00052

COMMITMENT CHANGE TITLE:

Monitoring of suppression pool pH

COMMITMENT DESCRIPTION:

Revise the applicable emergency procedures to call for a pH test in the event the suppression pool water contains substantial amounts of iodine in the late phases of the accident.

JUSTIFICATION FOR CHANGE OR DELETION:

As described in GIN 2001/00367, GGNS has analytically demonstrated that SLC injection is sufficient to maintain the post-accident suppression pool pH in an alkaline chemistry for the duration of the event, ensuring iodine dissolved in the suppression pool will not re-evolve. The NRC has agreed with this assessment in GNRI-2001/00032. Therefore, there is no impact of re-evolved iodine on the safety functions of SSCs and the pool pH does not need to be monitored.