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Serial Number: 97-001-NPE   Document Evaluated: DCP 93/0050, R01

DESCRIPTION OF CHANGE: DCP 93/0050, Revision 1 will add one cell to the Division I (1A3) and Division II (1B3) batteries for the purpose of maintaining voltage level and capacity when one cell is taken out-of-service during maintenance. This DCP will also provide a new battery monitoring device to accommodate an odd number of cells. This monitor will be isolated and analyzed per failure modes and effects report No. GGNS-96-0073 to meet the requirements of Reg. Guide 1.75.

REASON FOR CHANGE: Calculations have shown that due to aging of the battery cells the voltage level and capacity of the 1A3 and 1B3 batteries are placed in a marginal condition when one cell is taken out-of-service for maintenance. A reduction in voltage and available ampere hours could cause the batteries to fall below required levels. Credit is taken for the voltage based battery monitoring device in Section 8.3.2 of NUREG-0831 in lieu of a current based battery monitoring device for determining battery condition.

SAFETY EVALUATION: The installation of an additional cell will help maintain the Division I and Division II batteries. Limiting conditions of operation as defined in the GGNS Technical Specifications will not be affected. The basis for evaluation of any accident as defined in the UFSAR will not be affected. No new conditions are created which may affect any system or equipment important to safety as previously evaluated in the UFSAR. Reviews of electrical calculations associated with the operations of these batteries have shown that the addition of a cell will not create a degrading overvoltage for systems or equipment associated with the batteries. The maximum voltage permitted by UFSAR Section 8.3.2.1.6.2 of 140VDC will not be exceeded. Calculation EC-Q1L21-90026 (125 VDC Division I and II Batteries Short Circuit Evaluation) has been revised to include the added cell. Additional hydrogen evolution has been evaluated and determined to be acceptable based on 10 air changes per hour by the associated ventilation systems. The associated battery charges possess suitable excessive margin to maintain the battery recharge time of less than 12 hours. The new cell and associated hardware will be purchased safety related and qualified per applicable standards/specifications. The new battery monitoring device will provide the same annunciator currently provided, therefore, any indication requirements will not be affected by this change. The new monitor will be "Associated" and will be installed per Reg. Guide 1.75 requirements. Engineering Report No. GGNS-96-0073 "Failure Modes and Effects Analysis for Battery Monitors 11DA-96 and 11DB-96" provides the analysis required to ensure that any postulated failure of the battery monitor will not adversely affect the Class 1E batteries. All new equipment will be mounted to withstand seismic loading. Calculation EC-Q1L21-96004, performed in support of DCP 93/0050, Revision 1, evaluated the setpoint information for the new battery monitoring device. Calculations EC-Q1L21-90032 and EC-Q1L21-90047 have been revised to reflect the actual cell type and number utilized for these batteries. UFSAR Section 9.4.5.5.5 and Figure 8.3-10 require revision per CR 96-019. The addition of these cells will not adversely affect the combustible heat load calculation for the affected battery rooms (OC207 & OC211) as described in calculation MC-QSP64-86058. This calculation has been revised to account for the additional combustible material to these rooms, therefore, combustible loading as described in the UFSAR has not changed.

Serial Number: 97-002-NSRA Document Evaluated: LDCR 96-136

DESCRIPTION OF CHANGE: Changes to UFSAR Section 18.1.22 and the EP training procedure (01-S-04-21) are being made to allow operations management personnel that are not part of a shift operating crew to have the mitigating core damage training either in a plant-specific form or be credited for similar training from other nuclear facilities.

REASON FOR CHANGE: The UFSAR states that operator-oriented training is required of shift technical advisors and operations personnel from the GGNS General Manager to the licensed operators. However, the UFSAR does not specify position specific training requirements.

NUREG 0737 item II.B.4 (Training for Mitigating Core Damage) states that STAs and operations personnel from the plant manager through the operations chain shall receive all the training in Enclosure 3 to H.R. Denton's letter dated 3/28/80. This letter provided revised criteria to be used by the NRC staff for evaluating reactor operator training and licensing programs. Enclosure 3 stated that a program was to be developed to ensure that all operating personnel are trained in the use of installed plant systems to control or mitigate an accident in which the core is severely damaged. The main aspects of the training were focused on: 1) detection and determination of the extent of core damage, 2) recognition of faulty indications and alternate methods of monitoring vital parameters, 3) expected chemistry results and the importance of leak tight systems, 4) expected response of radiation monitors and determination of doses, and 5) an understanding of hydrogen generation and potential consequences.

Although it is important for upper management in the operations chain to understand the principles of mitigating core damage, training on the direct manipulation or use of installed plant-specific equipment is not warranted. This level of training is included in the licensed operator training program for personnel who will be part of an operation's shift crew and for the Operations Supt. who is currently required to maintain an active SRO license. The Operations Manager and the GGNS General Manager are not required to maintain an active SRO license and will not be involved in the direct use or manipulation of system controls. Therefore it is acceptable to credit training on core damage mitigation from other nuclear facilities.

SAFETY EVALUATION: The safety evaluation concludes that plant specific in-depth training is not necessary for those operations management personnel that are not part of a shift operating crew. The training for mitigating core damage will be provided to operations management personnel either in a plant-specific form or credit will be given for similar training from other nuclear sites. The shift technical advisors and operations personnel who are directly responsible to maintain the critical safety functions of the plant structures, systems, components will continue receiving plant-specific training for mitigating core damage.

Serial Number: 97-003-NPE Document Evaluated: ER 96/0207-00-00

DESCRIPTION OF CHANGE: Permit the temporary routing of a camera cable between secondary containment door 1A501 and its threshold. The presence of the cable is required until fire protection sprinklers and thermolag are added to room 1A539 per MCP 94/1063, and can be expected to have a very minor impact on the door's ability to serve as an airtight boundary.

REASON FOR CHANGE: MNCR 95-0262 was written to document air leakage through Secondary containment doors 1A401 and 1A501. Leakage through door 1A401 was attributed to a misalignment of the door seal mechanisms. Leakage through door 1A501 was attributed to a missing section of gasket material and the presence of a camera cable routed between the bottom of the door and threshold. The System Engineering disposition attached to the MNCR required the missing section of gasket material on door 1A501 to be replaced per existing requirements. Additionally, the NPE disposition provided in ER 96/0207-00-00 requires the misalignment at door 1A401 to be reworked per existing requirements. However, it is necessary to leave the camera cable at door 1A501 for fire watch observations until fire protection sprinklers and thermolag are added to room 1A539 per MCP 94/1063.

SAFETY EVALUATION: With the exception of an "interim Accept-As-Is" disposition concerning the camera cable under door 1A501, all items in the MNCR are being dispositioned as "Rework" returning them their documented design condition. Final disposition concerning the camera cable is also "rework". Dispite the potential for a small temporary increase in air leakage through door 1A501 due to localized effects around the cable, (i.e., failure to obtain a perfectly airtight seal between the door and the gasket at the sides of the cable) the integrity of the secondary containment is maintained. Per the Operability/Reportability Resolution and the P&SE disposition for MNCR 95-0262, monthly Standby Gas Treatment System (SGTS) drawdowns have been successfully completed demonstrating the ability to maintain the secondary containment despite observed leakage through several doors, including door 1A501. The majority of the leakage through door 1A501 was attributable to the missing gasket on the lockset side of the door. With this gasket replaced and doors 1A308, 1A310, 1A401, and 1A504 reworked, the drawdown margin associated with SGTS testing will have improved. Additionally, the existing design requirements for the doors given in Material Requisition 9645-A-002.4 would allow leakage of approximately 5 cfm at 0.3" water gauge differential pressure, whereas, SGTS calculations 39.3 and 3.9.8, Supplement 1 consider the equivalent of approximately 40 cfm of leakage at the same differential pressure. Although the presence of the cable may result in more than the 5 cfm allowable, it would not exceed the 40 cfm considered in the SGTS calculations.

Further, the auxiliary building tornado depressurization analysis contained in calculation C-H-006.0 is not affected by the presence of the cable. Door 1A501 is not a fire barrier, and the presence of the cable does not invalidate any UL listing or fire hazard analysis. The cable does not affect the seismic design of the door.

Section 3.6.4.1 of the GGNS Tech. Specs. requires that doors accessing the secondary containment be verified as closed to insure secondary containment integrity; and per the basis for Section 3.6.4.1, the doors are closed to ensure that the rate at which air leaks into the secondary containment is not great enough to prevent the desired negative pressure from being maintained. As noted in the basis for Section 3.6.4.1.1, "...the term 'sealed' has no connotation of leak tightness." Exact leakage rates are not specified in the Technical Specifications, and the temporary leakage around door 1A501 due to the camera does not prevent a successful drawdown of the secondary containment. Therefore, changes to the Technical Specifications are not required, and the margin of safety associated with Tech. Spec. Section 3.6.4.1 is not reduced.

This change does not interface with any other system, or directly affect any parameter such that it could alter radionuclide population, release rate, or duration; or create new release mechanisms.

Serial Number: 97-004-NPE Document Evaluated: ER 96/0525, R00

DESCRIPTION OF CHANGE: ER 96/0525 requested the installation of a road bed from the west face of the control building to about 250 feet west of the control building.

REASON FOR CHANGE: The UFSAR states that use of material other than clay for the moisture barrier is an exception. The clay seal does not have the strength to support heavy loads. This has caused problems in traffic areas and material staging areas. The Plant has increasingly requested using a more stable material for the moisture barrier. Therefore, the use of other materials is becoming less of an exception and more of the normal method of repair.

SAFETY EVALUATION: The purpose of the clay seal is to minimize surface water infiltration into the sand backfill. The UFSAR allows the use of alternate material but refers to such use as an exception. However, plant use requires a material with higher load bearing capacity. The ER allows the replacement of a section of clay seal in front of the Control Building with clay gravel and a concrete overlay. The use of alternate materials will not affect the moisture barrier's ability to minimize surface water infiltration into the sand backfill. The alternate moisture barrier will also minimize the surface water infiltration into the sand backfill. Grade elevation restrictions have been placed in the ER to insure that PMP levels are not increased. The staging of sand bags for protection of PMP doors and for temporary berming of the work area is required if inclement weather occurs. Since the grade elevations are maintained the protection of underground facilities has no impact on radionuclide release rate, duration, nor will create new release mechanisms or radiation release barriers. The possibility for a malfunction of equipment important to safety of a different type has not been created because all work performed is outside of an existing building. The only item the moisture barrier comes in contact with is the structural sand backfill. The new cap will not affect the form, fit or function of the structural sand backfill. Additionally, the margin of safety as defined in the basis for any Tech. Spec. has not been reduced. TRM 6.7.5 does address flood protection against PMP rainfall, and changes in the yard can impact flood water levels. The 45% blockage restriction to Culvert No. 1 which is given in the TRM, is based on the PMP door seals' ability to provide protection from PMP floodwater levels with a freeboard of 6" above the calculated water levels. The ER requires the final elevations not to exceed those evaluated for the PMP event. Therefore, these changes will not elevate PMP water levels above 133'-3" and since these changes do not have the potential to block Culvert No. 1 or affect the basis for the TRM they will not impact this or any other Tech. Specs.

During construction activities the normal protection provided against tornado missiles and surface runoff into the structural backfill will be reduced. Therefore, certain restrictions have been provided to insure that protection is provided by an alternate method. These methods include the staging of sandbags at affected flood doors, the use of a temporary berm around the exposed area to minimize any adverse affects upon the plant from a postulated PMP event and the use of steel plates to insure adequate protection from tornado generated missiles for electrical duct banks that become exposed.

Serial Number: 97-005-NPE Document Evaluated: N/A (RFO8 Core Shuffle)

DESCRIPTION OF CHANGE: During RFO8, 272 depleted Siemens 9x9-5 fuel assemblies will be replaced with fresh GE11 fuel assemblies. These new assemblies have been designed and built specifically for the Grand Gulf Cycle 9 core and are currently stored in the Auxiliary Building. This safety evaluation addresses the RFO8 core shuffle and considers the following items:

a. movement of light loads, fresh and irradiated fuel in the containment and the Auxiliary Building;

b. storage of fresh and irradiated fuel in the containment pool during RFO8;

c. compatibility of the GE11 fuel with the fuel handling equipment, the reactor internals and the SPC fuel; and

d. shutdown margin for all interim RFO8 core configurations and the final Cycle 9 core loading in Modes 4 and 5.

REASON FOR CHANGE: Cycle 9 operation requires the addition of fresh fuel assemblies and the removal of depleted assemblies from the reactor vessel. Actual Cycle 9 operation with these fuel assemblies will be assessed in an upcoming safety evaluation.

SAFETY EVALUATION: This evaluation concludes that (I) a fuel handling accident during RFO8 will not result in doses above the allowable limits, (ii) Fresh fuel can be moved into containment upon entering Mode 4 and irradiated fuel can be removed from the reactor 3 days after subcriticality is achieved, (iii) the GE11 fuel compatible with the fuel handling equipment, the reactor internals and the remaining SPC fuel, and (iv) the acceptance criterion for shutdown margin is satisfied for interim RFO8 and final Cycle 9 core configurations during Modes 4 and 5.

Serial Number: 97-006-NPE   Document Evaluated: MCP 94/1058-R00 & SCN 96/0021A

DESCRIPTION OF CHANGE: The change will provide a new ground detection circuit for the BOP 125 Volt DC buses 1DD1, 1DE1, 1DG1, 2DG1, 1DK1, and 1DL1. The new circuits are located in the following panels; 11DD, 11DG, 2DG, 11DK and 11DL. Computer points shall be provided for identification of grounds on these BOP DC buses.

The change will also provide details for the replacement of non-class 1E battery 1L3. The replacement cells are C&D Power Systems Type LCR-33. The replacement of the 1L3 batteries are necessary to support the relocation of the 250 VDC bus 11DF. The MCP will change the DC power and control sources for 250 VDC bus 11DF from 125 VDC buses 11DD and 11DE to buses 11DK and 11DL.

REASON FOR CHANGE: MNCR 0166-94 documented a scram that occurred on 11/1/94 due in part to bus 1DA1's ground detection circuit and an existing ground fault on backup scram valve 1C11F110A. The designed ground on the ground detection circuit (1kΩ) did not limit the ground fault current to an acceptable level not to support an inadvertent actuation of 1C11F110A. Also, since the backup scram valve's solenoid is isolated from either pole of the DC bus during normal operation the existing ground was undetected.

BOP battery buses 11DD, 11DE, 11DG, 21DG, 11DK, and 11DL had similar ground detection circuits as battery 11DA. Therefore, to preclude similar inadvertent actuations on these BOP buses, these 1kΩ ground detection circuits were eliminated. MCP 94/1058 is providing a new ground detection circuit that will not be susceptible to causing an inadvertent actuation. Electrical calculation evaluated the ground detection circuit to ensure no inadvertent actuations will occur given a postulated single solid ground fault on the DC bus.

BOP battery 1L3 will be replaced to maintain the reliability of BOP bus L when the 250 VDC pump load connected to bus F is energized. This battery set will be replaced with C&D LCR-33 cells. Both batteries 1K3 and 1L3 are necessary to support the operation of bus 11DF.

SAFETY EVALUATION: The systems affected by MCP 94/1058, Revision 0 are non-safety related, L11, L21 and L51. MCP 94/1058 shall install a ground detection circuit for each of the 125 Volt DC BOP buses (11DD, 11DE, 11DG, 21DG, 11DK and 11DL) to detect grounds on the BOP buses. The ground detection circuit shall be installed such that proper isolation per Reg. Guide 1.75 is maintained, thus ensuring that a postulated failure on any of these Non-class 1E circuits shall not propagate a failure within any Class 1E circuit. All equipment and work associated with this design shall be within Non-class 1E enclosures, thus ensuring that the equipment installed per this design shall not damage any Class 1E equipment during a postulated seismic event. Electrical calculation EC-N1L21-95004, Revision 1 verifies the acceptance of the new ground detection circuits.

Electrical calculation EC-N1L11-95002, Revision 1 verifies the acceptance of the capacity and voltage ratings of the new 1L3 batteries and the existing 1K3 batteries. Other loads for batteries 1K3 and 1L3 to supply DC power to shall be BOP Inverters. These new batteries are acceptable, with the new non-Class 1E 250 VDC motor load, because they will meet existing requirements for design and operation of the "K", "L", and "F" VDC systems as evaluated by battery sizing calculation EC-N1L11-95002. Electrical calculation EC-N1L21-91014, Revision 1 verifies that the feeder breakers for buses 11DK, 11DL, and 11DK are properly coordinated and will provide the necessary protection for each circuit. The only association of Batteries 1K3 and 1L3 to Class 1E equipment is the source for the battery chargers. Battery chargers, 1DK4, 1DK5, 1DL4, and 1DL5 are fed by Class 1E load Centers 15BA1, 15BA2, 16BB1, and 16BB2, respectively. The feeder breakers for these charges are tripped upon an accident signal per the requirements of Reg. Guide 1.75. Replacement of Battery 1L3 will not have an impact on the isolation of Battery 1L3 from the Class 1E bus 16AB per the requirements of Reg. Guide 1.75.

The original design and operational requirements for VDC bus 11DF will be maintained with the relocation of the power and control source for all loads associated with this bus. The 250 VDC system serves four large non-class 1E auxiliary loads that are vital to the plant's main generator when needed. This system is created by a series connection of the two 125 VDC non-Class 1E batteries K and L. Electrical calculations EC-N1L11-95002, Revision 1, EC-N1L21-91012, Revision 1, EC-N1L21-91014, Revision 1, and EC-N1L21-96016, Revision 0, support the design change to move the dc power feeds for bus 11DF from 11DD and 11DE to buses 11DK and 11DL and replace battery 1L3 with a large battery set.

Serial Number: 97-007-NPE Document Evaluated: DCP 92/0006

DESCRIPTION OF CHANGE: This change makes the following changes to the non-safety related portion of the Plant Chilled Water (PCW) system: (1) the three existing 850 ton Plant Chillers (N1P71B001A,B&C), which use R-12 refrigerant (CFC refrigerant), are being replaced with three 1200 ton chillers which use R-134a refrigerant (HFC refrigerant); (2) the Auxiliary Plant Chiller is being deleted; and (3) a refrigerant recovery system is being installed.

The replacement of the chillers requires revision to the Fire Hazards Analysis, Specification M-500.0, to reflect the change in quantity of lube oil and to verify that the current analysis is valid.

REASON FOR CHANGE: Refrigerant R-12 will not be available after January 1, 1996 due to environmental reasons, therefore, the Plant Chillers will be replaced with chillers that utilize a refrigerant that should be available through the plant's remaining life. The Auxiliary Plant chiller is being deleted because the three new Plant Chillers will have more than adequate capacity for design will have more than adequate capacity for design conditions and the new chillers will be capable of operating at reduced capacities down to 15% (i.e., 180 tons). The refrigerant recovery system is being installed for maintenance purposes, with the capability to store the full refrigerant charge of one of the new chillers. The service water outlet temperature for the individual chillers will increase due to the increased rating of the chillers (i.e., 850 ton versus 1200 ton).

SAFETY EVALUATION: As stated in UFSAR Section 9.2.7.3, the PCW system provides no safety function other than function of the Auxiliary Building and Containment penetration isolation valves. Additionally, there is a safety related rupture disc (Q1P71D011), in the common condenser refrigerant relief valve discharge piping exhaust to atmosphere, which provides auxiliary building isolation to the exterior. The changes do not affect the operation of these isolation valves and do not affect the integrity of rupture disc. System analysis has shown that a failure of the non-safety related portion of the system will not compromise any safety-related system or component and will not prevent safe reactor shutdown. Moderate energy line breaks as a result of the failure of the system is analyzed in UFSAR Section 3.6. Missile generation as a result of the failure of the system is analyzed in UFSAR Section 3.5. The change will not invalidate any of the analyses or assumptions contained in the UFSAR concerning the failure of the PCW system.

Because refrigerant R-134a falls within the same ASHRAE Standard 34 Safety Group as R-12 (i.e., A1), the change in refrigerant does not create the potential for a chemical of higher toxicity or higher flammability than the existing refrigerant. In fact, Safety Group A1 classification represents refrigerants with the lowest flammability and lowest toxicity. Should a release of refrigerant from the new Plant Chillers occur, the consequences would be the same as with the existing chillers as related to its effect on the operability of the Standby Gas Treatment System (i.e., Technical Requirements Manual Administrative Controls, Procedure 7 Programs, Section 7.6.3.4 "Filter Testing Program" requires testing in accordance with Technical Specification 5.5.7 "following painting, fire or chemical release in any ventilation zone communicating with the subsystem). The refrigerant detectors would alert operators of a release.

The new chillers result in an additional heat load at elevation 139' of the Auxiliary Building due to the larger motors and the motors not being hermetically sealed as were the original chiller motors. This area is cooled by Aux. Building Ventilation System Fan Coil Unit N1T41B003. The fan coil unit is sufficiently sized with design chilled water flow and 44°F chilled water temperature, to maintain the area within the design limit of 104°F (per Technical Requirements Manual Table 6.7.3-1) and at the normal temperature of 80°F. The temperature in the area would be unaffected during a DBA since the chillers are not assumed to be operating during a LOCA or a LOCA coincident with a Loss of Offsite Power, therefore, the room temperature of 131°F assumed during post LOCA conditions is unaffected by the increased load due to the new chillers.

The new chillers result in an additional connected electrical load on the BOP 4.16 kV buses. The additional connected load on Bus 13AD will be 700 hp and the additional connected load on Bus 14AE will be 350 hp. The additional load will not invalidate any of the voltage analyses or protective device coordination for either the BOP or ESF buses.

The relocation of the fire protection line and sprinkler nozzle (#55) will not degrade the effectiveness of the fire protection system since the location of the line and sprinkler have been chosen to provide equivalent fire protection and meets NFPA 13.

The relocated HVAC duct register are part of the non-safety related portion of the Auxiliary Building HVAC system, for area unit cooler N1T41B003, and will not affect cooling during a DBA since no credit is taken for the system during a LOCA or a LOCA with Loss of Offsite Power. Additionally, the registers are being located such that they will facilitate the necessary cooling required during normal plant operation. The relocated temperature element, N1T41N037, is part of the Auxiliary Building HVAC system and provides no safety function. The element provides area temperature monitoring via computer point N1T41N037; its relocation will not change its function or effectiveness.

As stated in UFSAR Section 9.2.8.3, the PSW system has no safety design basis. Break of the PSW system inside the Auxiliary Building has been analyzed. Non-safety related piping for the PSW system is routed so that a pipe break will not flood or damage any safety related equipment.

During maintenance activities which require the use of the refrigerant recovery system for removing refrigerant from a chiller, the system is mechanically connected only to the chiller being serviced and electrically connected t a convenience outlet supplied from BOP 120/240 VAC panel 14P12. Performance of this maintenance activity, including during normal plant operation, will have no additional impact on plant operation, since the worst case situation (i.e., loss of refrigerant from the recovery tank) would be no worse than the loss of refrigerant from a chiller. The recovery tank is manufactured to ASME and ARI (Air Conditioning & Refrigeration Institute) criteria. The unit will be permanently mounted in the location where the Auxiliary Plant Chiller was mounted and will be appropriately restrained. The recovery system consists of a tank, sized to hold approximately one chiller charge of refrigerant, a compressor and associated controls. The new line from the recovery system tank relief valves outlet that ties into the existing common condenser refrigerant relief valve discharge piping exhaust to atmosphere has been seismically supported. Failure of the line would not affect the integrity of the rupture disk. The new power supply for the monitor for the rupture disc in the common condenser refrigerant relief valve discharge piping exhaust to atmosphere is being added to ensure that the rupture disk monitor has power regardless of which chiller is in operation.

Because the changes described above will meet or exceed the requirements of the original design and existing analyses, they will not degrade any function important to safety systems, components, or structures nor will they degrade or prevent actions described in the SAR accident analysis. The piping installed by this design change meets ANSI B31.1 code requirements and is supported for the appropriate dead weight and thermal loads. The electrical power for these components is from non-1E sources and the effect on Class 1E power supplies has been considered.

Serial Number: 97-008-NPE Document Evaluated: MCP 96/1006, R00

DESCRIPTION OF CHANGE: The existing metal bellows on the Standby Liquid Control (SLC) Pump "B" will be replaced with a braided flexible metal hose by MCP 96/1006. The cause of failure of the expansion joint for the "B" pump has been determined to be the result of high cycle fatigue induced by vibration from the pulsation of the pump during operation. The "A" pump has not experienced a failure of the Metal Bellows. The results of the monitoring performed on the configurations for both "A" and "B" piping system (expansion joint to the anchor support) gives lower vibration and movement for the "A" piping than the "B" piping. Therefore, the Engineering evaluation concludes that the "A" Expansion Joint is expected to meet the 40 year design life. Therefore, the MCP addresses replacement of the "B" Expansion Joint, only.

REASON FOR CHANGE: During surveillance testing of the SLC Pump "B" cracks were discovered in the pump discharge expansion joint resulting in failure and the initiation of MNCR 0234-95 to document the failure. The cause of the cracking was determined to be a result of fatigue induced by pulsation vibrations from operation of the pump.

SAFETY EVALUATION: The existing metal bellows on the "B" SLC Pump has had reoccurring failures, therefore, MCP 96/1006 will replace the existing metal bellows with a braided flexible metal hose assembly. The design of the new assembly has a very high fatigue life margin and will meet the specified design requirements for repetitive movements and vibration. The design of the braided metal flexible hose has an increase length, reducing the lateral spring rates to accommodate pipe and pump movements. Additionally, the body thickness is increased for increased pressure capability. Therefore, the braided flexible metal hose assembly meets the design requirements and has an expected design life of 40 years. The movements of the new hose assembly will be checked at installation for movements and vibration. The function of the piping system will not be affected with the change to the braided flexible metal hose.

Serial Number: 97-009-NPE Document Evaluated: ER 96/0082-00-00

DESCRIPTION OF CHANGE: Replace Process Sampling System Containment/Drywell Atmospheric and Condensate Suction Flow Indicator 1P33-FI-R145 with approved alternate replacements. All modifications will be performed at Panel 1P33P010, Turbine Building Elevation 93'0".

REASON FOR CHANGE: The existing MI60 series indicator currently installed is obsolete. The existing indicator is a multi-function, microprocessor based unit utilized for rate indicator and totalizer functions. Because the MI60 indicator is outmoded, an alternate replacement model has been approved for installation in the system. The MI60 is a dual channel indicator which provides processing capabilities for both the atmospheric and suction flow functions. The new FC70A model is a single channel device which cannot function to process both inputs. Therefore, two of the FC70A indicators inputs will not be identified as 1P33-FI-R145A and 1P33-FI-R145B, respectively. Installation of the two replacement indicators will be done in existing Panel 1P33P010.

SAFETY EVALUATION: The changes required for replacement of this obsolete indicator do not affect the content of the GGNS Technical Specifications or UFSAR, except for revision of UFSAR Figure 9.3-007a (P&ID M1069D) to reflect two single input indicators instead of the one dual channel indicator currently shown. This modification does not change the intended design function of or the information supplied by the current Process Sampling System configuration. There is no change to the design bases of the Process Sampling System.

Serial Number: 97-010-NSRA Document Evaluated: LDCR 96-106

DESCRIPTION OF CHANGE: The purpose of this safety evaluation is to answer the question "Does the open Turbine Building roll up door represent an actual or potential release path with respect to 10 CFR 20 or 10 CFR 100?" The evaluation will be used to support the attached change to the FSAR which clarifies the role of the Turbine Building rollup door.

There are three basic operating conditions of the plant related to the Turbine Building ventilation system to address: Accident conditions, normal operation, and malfunctions. It is important to understand that the Turbine Building roll up door is really a question of Turbine Building ventilation system operation. The roll up door represents a suction path for the ventilation system and under normal operation air flows in through the door. This was verified by a review of the ventilation system design. Additionally to see if any unique problems occurred, a demonstration was conducted to verify inflow always occurs during normal operation.

The demonstration consisted of three streamers (short, medium, and long) which were hung in the opening. A periodic check was done until 100 data points had been gathered over a period of 18 days. The data revealed that under normal Turbine Building configuration* air always flowed into the building when the Turbine Building ventilation system was operating. It should be noted that 5% of the time the shortest streamer indicated some outflow. However this is due to local currents or "flow back" caused by changes in building ventilation flow. One hundred percent of the time the long streamer indicated inflow.

ACCIDENTS

Based on review of the UFSAR the Turbine Building and hence the Turbine Building ventilation system is not credited under accident conditions. The only consideration given for the Turbine Building is that its failure will not compromise other safety related structures. The Turbine Building roll up door position has no effect on this assumption. Additionally the Turbine Building roll up door is mentioned in flood protection. However as noted in Section 3.4.1.3 ample time would be available to close the door before flooding conditions occur. Therefore, the position of the Turbine Building roll up door is irrelevant under accidents conditions.

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* The demonstration suggested that when turbine building floor plugs are removed, air flow patterns within the turbine building may be altered sufficiently to cause outflow through the turbine building rollup door. Should such unusual conditions occur it will be identified and corrected (i.e., turbine building door closed) through observation such as during twice daily turbine building rounds.

NORMAL OPERATION

There are two cases for normal operations. Either the plant is operating in Modes 1 or 2, or the plant is shutdown in Modes 3, 4, or 5. During shutdown, the use of normal radiological controls work practices protect against the occurrence of uncontrolled releases. Therefore, the scope of this evaluation only addresses the position of the Turbine Building rollup door during Modes 1 or 2.

As shown from the demonstration during normal operation of the Turbine Building ventilation system, air flows in the open door. As such the roll up door does not represent a release path.

MALFUNCTIONS

The Turbine Building and the Turbine Building Ventilation System were built to the required design standards. The Turbine Building Ventilation System does contain some limited redundancy. The position of the Turbine Building roll up door is not addressed in any of the design information with the exception of the previously mentioned flooding. It is clear, however, that the Turbine Building roll up door is a large opening to atmosphere, and that should a malfunction of normal Turbine Building Ventilation occur it represents an unquantified potential (but minor) effluent pathway. While no specific requirement exists it is prudent to close the Turbine Building roll up door should an extended Turbine Building Ventilation System malfunction occur.

REASON FOR CHANGE: This evaluation was prepared to support an FSAR change which clarifies the role of the Turbine Building rollup door.

SAFETY EVALUATION: There is no unreviewed safety question concerning operation with the Turbine Building rollup door open.

Serial Number: 97-011-NPE Document Evaluated: ER 96-0748 & LDCR 96-127

DESCRIPTION OF CHANGE: Technical Requirements Manual and UFSAR Appendix 16B limits for grapple-engaged loaded interlock and primary and redundant fuel load interlocks are being changed from 750 lb. to 660 lb.

REASON FOR CHANGE: There is an upward force on the Refueling Platform Main Hoist resulting from cable reel tension from air hose, electrical cable and camera cable which were not considered in original calculation for establishing limits. Revision of this calculation to consider this upward force resulted in a need to change the TRM limits and corresponding setpoints. The new GE11 fuel being loaded in RFO8 is approximately 40 pounds lighter than any previous fuel. This change in minimum fuel weight also necessitated a change in the TRM and UFSAR limit.

SAFETY EVALUATION: The change in the Technical Requirements Manual and UFSAR Appendix 16B limits for grapple-engaged loaded interlock and primary and redundant fuel load interlocks from 750 lb. to 660 lb. are acceptable because (a) the overall process for fuel movement and handling using the Refueling Platform Main Hoist remains unchanged (only the specific limit and setpoint values are being changed), (b) the process is adequately controlled by administrative controls, interlocks, and LCOs, and (c) the new limits and setpoints provide for more accurate controls during fuel movement.

The applicable accident event for this change in limits is the Control Rod Removal Error During Refueling as described in UFSAR 15.4.1.1 as referenced in TS Bases B 3.9.1 and B 3.9.2. All administrative controls and interlocks currently required for the operation of the main hoist will remain in effect and only the specific values for limits and setpoints are being changed with the functions remaining the same; therefore, the probability of occurrence of the applicable accident is unchanged.

No modifications to the hoist are being made other than the set point change, and the basic functioning of the hoist remains unchanged.

The bases as defined in Technical Specification Bases B 3.9.1 and B 3.9.2 are not affected by this TRM limit change and remain valid as written. Therefore, the margin of safety as define in these Bases is unchanged.

Serial Number: 97-012-NPE Document Evaluated: ER 96/0803

DESCRIPTION OF CHANGE: During RFO8, the drywell monorail with two 15 Ton hoists was used to lift and move the ≈30 ton reactor recirculation "A" pump motor in preparation for maintenance on the pump. While moving the suspended motor to the temporary storage location, the leather disc brake on one of the 15 Ton monorail hoists failed. The main hoist chain then "free-wheeled" allowing one side of the motor to drop. As the load was dropping, the hand chain became knotted and jammed, stopping the hoist after the load had dropped approximately 7". Personnel then secured the motor using two 10 Ton hoists attached to the failed 15 Ton hoist and using a 80 Ton Sling double wrapped around the monorail beam and attached to the motor. It is presently located near azimuth 170° in the Drywell.

The purpose of this safety evaluation is to assess the method for moving the motor to a safe storage area where it can be lowered and decoupled from the hoists. The details of this method are described in ER 96/0803.

Redundant rigging will consist of a minimum of 2 slings, each capable of supporting 30 tons. One sling to be positioned directly over each of the 15 ton hoists. The redundant rigging will be provided by passing the sling over the top of the 15 Ton hoist and hoist monorail and attaching both ends of the sling to the motor. Sufficient slack should be left in the slings to allow them to slide along the monorail. However, excessive slack which would allow the load to drop an appreciable distance if the primary load path failed will be avoided. Rigging angles on the slings will be maintained small enough that the rated capacity of the sling is not exceeded.

As the motor is moved along the monorail towards the platform and the hoist approaches a monorail support, one additional sling (third sling) must be installed to the motor as described above on the opposite side of the monorail support to redundantly support the motor. After the additional sling (third sling) is in place the sling on the other side of the support will be removed and the motor moved along the monorail until the trailing hoist reaches the monorail support. At this time the additional sling is attached to the motor again in order to move the motor past the monorail support. At all times during the travel, at least one sling will be attached to each end of the motor and draped over the associated 15 Ton hoist.

When the motor is positioned over the platform steel at the Drywell Equipment Hatch, four 10 Ton Hoists will be installed on the monorail beam and slings will be used to redundantly support the motor from the four 10 Ton hoists. Then the two slings which were used as redundant support during the movement of the motor will be removed and the motor will then be lowered down to the platform by lowering all six hoists simultaneously.

REASON FOR CHANGE: The motor needs to be moved to a secure location in the safest manner possible to repair the hoists. Movement in accordance with ER 96/0803 and placing the motor on the platform is considered much safer than leaving the motor suspended from the monorail beam while performing hoist maintenance and testing. Therefore, this action will require the motor to be moved from Azimuth 170° to the Drywell Equipment Hatch platform at Azimuth 220° where it will be lowered to the platform and disconnected from the hoists.

SAFETY EVALUATION: The motor will be moved and stored using redundant heavy load lifting and handling equipment designed with sufficient margins of safety to ensure that a credible failure mechanism does not exist that could cause a load drop. The load handling equipment will consist of one refurbished 15 Ton hoist, the other 15 Ton hoist and redundant safety slings. Although one of the hoists will be refurbished with new brakes, no new load tests can be performed in the present location. Therefore the hoist hand chains will be administratively controlled, by locking the hand chains together providing positive locking of the load on the hoists. This will remain in place during the movement of the motor along the monorail. With the one 15 Ton hoist being refurbished, if the hoist is then capable of lifting the load, it will be considered acceptable to be the primary lifting device with the slings as the redundant lifting device. To maintain prudent safety margins such that the probability of an accidental load drop is sufficiently low as to not be credible, redundant safety slings will be used in addition to the 15 Ton hoists will transfer the full load of the failed hoist onto its respective safety sling. This redundant load lifting and handling method meets the intent of single-failure-proof requirements of the applicable GGNS licensing commitments (see Reference 1, Section 9D.1). These commitments provide a defense-in-depth approach for controlling the handling of heavy loads so that load handling accidents have a very low probability of occurrence. Since the safe load handling methods will ensure that the motor can be moved and placed in the temporary storage location without creating a credible drop mechanism, no equipment important to safety will be affected. The movement of the pump motor with the monorail using redundant lifting and handling methods will not reduce the margin of safety of any basis to the Technical Specifications.

Serial Number: 97-013-NPE   Document Evaluated: ER 96/0803-01-00

DESCRIPTION OF CHANGE: During RFO8, the drywell monorail with two 15 Ton hoists was used to lift and move the ≈30 ton reactor recirculation "A" pump motor in preparation for maintenance on the pump. While moving the suspended motor to the temporary storage location, the leather disc brake on one of the 15 Ton monorail hoists failed. The main hoist chain then "free-wheeled" allowing one side of the motor to drop. As the load was dropping, the hand chain became knotted and jammed, stopping the hoist after the load had dropped approximately 7". Personnel then secured the motor using two 10 Ton hoists attached to the failed 15 Ton hoist and using a 80 Ton sling double wrapped around the monorail beam and attached to the motor. The motor will be placed in a safe storage position on the platform steel at Azimuth 220° in accordance with the instructions and requirements provided in ER 96/0803-00-01 and 50.59 Safety Evaluation 96-0110-R00.

The purpose of this safety evaluation is to assess the method for moving the motor to the storage area at Azimuth 270° where it can be stored during pump rework. The details of this method are described in ER 96/0803-01-00. This movement will not be completed until the 15 Ton hoists have been refurbished and load tested to 125% capacity.

Redundant rigging will consist of a minimum of 2 slings, each capable of supporting the load of the motor and dynamic loads with consideration of the sling configuration. One sling to be positioned over the monorail beam either above each of the 15 Ton hoist or above each of the alternate lifting lugs. Sufficient slack should be left in the slings to allow them to slide along the monorail. However, excessive slack which would allow the load to drop an appreciable distance if the primary load path failed will be avoided. Rigging angles on the slings will be maintained small enough that the rated capacity of the sling is not exceeded.

As the motor is moved along the monorail towards the storage location at Azimuth 27-° and the hoist approaches a monorail support, one additional sling (third sling) must be installed to the motor as described above on the opposite side of the monorail support to redundantly support the motor. After the additional sling (third sling) is in place the sling on the other side of the support will be removed and the motor moved along the monorail until the trailing hoist reaches the monorail support. At this time the additional sling is attached to the motor again in order to move the motor past the monorail support. At all times during the travel, at least one sling will be attached to each end of the motor and draped over the monorail beam near the associated 15 Ton hoist.

When the motor is positioned at storage position at Azimuth 270°, the load shall be left suspended from the hooks of the 15 Ton hoists and two of the slings will be left attached to the alternate lifting lugs with the slings not carrying any of the load. These slings will still be considered as the redundant rigging.

REASON FOR CHANGE: The motor needs to be moved to the storage position at Azimuth 270° suspended from the monorail beam in order to perform the pump internal activities. This movement will not be performed until specified maintenance and 125% load tests have been performed on the 15 Ton hoists.

SAFETY EVALUATION: The motor will be moved and stored using redundant heavy load lifting and handling equipment designed with sufficient margins of safety to ensure that a credible failure mechanism does not exist that could cause a load drop. The load handling equipment will consist of two 15 Ton hoists and redundant safety slings. To provide additional safety the hoist hand chains will be administratively controlled, by locking the hand chains together providing positive locking of the load on the hoists. This will remain in place during the movement of the motor along the monorail, however can be temporarily removed if the motor must be raised to avoid interferences during the travel. The 15 Ton hoists will be the primary lifting device with the slings as the redundant lifting device. To maintain prudent safety margins such that the probability of an accidental load drop is sufficiently low as to not be credible, redundant safety slings will be used in addition to the 15 Ton hoists. These slings will be installed such that a postulated failure of either of the 15 Ton hoists will transfer the full load of the failed hoist onto its respective safety sling. This redundant load lifting and handling method meets the intent of single-failure-proof requirements of the applicable GGNS licensing commitments (see Reference 1, Section 9D.1). These commitments provide a defense-in-depth approach for controlling the handling of heavy loads so that load handling accidents have a very low probability of occurrence. Since the safe load handling methods will ensure that the motor can be moved and placed in the storage location without creating a credible drop mechanism, no equipment important to safety will be affected. The movement of the pump motor with the monorail using redundant lifting and handling methods will not reduce the margin of safety of any basis to the Technical Specifications.

Serial Number: 97-014-NPE Document Evaluated: ER 96/0902-00-0

DESCRIPTION OF CHANGE: The four main steam lines feature pressure instrument connections just downstream of the outboard main steam isolation valves. Each pressure instrument connection consists primarily of a root isolation valve, tubing and associated tubing fittings, tubing supports, and another isolation valve at the pressure connection rack. Root valves of component numbers Q1B21-FX002, Q1B21-FX004, Q1B21-FX008, and Q1B21-FX011 will be removed and the resulting pipe nipples will have pipe caps welded to them. The section of tubing between each root valve and the first tubing support will also be removed and the tube end left open. The balance of the pressure instrument connections will be abandoned in place. The pressure instrument connection rack is located in the auxiliary building.

REASON FOR CHANGE: Root valves Q1B21-FX002, Q1B21-FX004, Q1B21-FX008, and Q1B21-FX011 were installed for plant startup testing and serve no useful function for power generation. The subject valves require maintenance to ensure valve integrity. Valve packing leaks can promote airborne activity and cause elevated ambient temperature in the steam tunnel.

SAFETY EVALUATION: The affected root valves and pressure instrument connections are not described in Technical Specifications. The modified piping will be subject to the same ASME inspection and testing requirements as the original configuration, thus there is no change to Technical Specifications.

The pipe caps are compatible with the piping to which they will be attached; the caps will be installed to the same code and quality requirements as the existing configuration; and the caps provide additional assurance of isolation from main steam line leakage. The piping and cap will continue to be considered as a potential missile outside containment. The existing missile protection will continue to function as a protective barrier. Circumferential weld failure postulated for the revised configuration is identical to the failure postulated for the previous configuration.

The margin of safety as defined in the Technical Specification Basis is not reduced as there is no change to design and quality requirements, system operation or other system parameter. The worst case credible failure of the connecting weld to the steam pipe is unchanged by this modification as that weld will not be impacted by this change.

Revision of the GGNS UFSAR was required to reflect the new plant configuration. Revisions the GGNS UFSAR were developed as part of the design review package.

Serial Number: 97-015-NPE Document Evaluated: ER 96/0904-00-00

DESCRIPTION OF CHANGE: ER 96/0904-00-00 will permanently install a camera in the Auxiliary Building Steam Tunnel (1A305) for the purpose of monitoring equipment operation and possible leak sources. This camera and its associated equipment (cable, monitors, control stick, etc.) were installed under Temporary Alterations 96-0009 & 96-0025. This camera is similar to a cameras currently installed to monitor turbine/generator operation. This change includes the installation of a non-flame tested cable which requires addition to UFSAR Section 8.3.3.1.

REASON FOR CHANGE: From an ALARA standpoint, entry into the Auxiliary Building Steam Tunnel for equipment observation or other non-essential purposes is not acceptable. The camera installation will permit observation without entry into the area. This 10CFR50.59 Safety Evaluation is performed in support of the required UFSAR change to list the non-flame tested cable in Section 8.3.3.1.

SAFETY EVALUATION: The installation of this camera into the Auxiliary Building Steam Tunnel is the most viable option for conformation of proper equipment operation during full power production. Limiting conditions of operations as defined in the GGNS Technical Specifications will not be affected. The basis for evaluation for any accident as defined in the UFSAR will not be affected. No new conditions are created which may affect any system or equipment important to safety as previously evaluated in the UFSAR. This camera will be installed onto a structure having suitable mass to withstand the additional loading. This camera will also be installed in a manner to prevent it from becoming a projectile during a seismic event. This camera will not be required or designed to operate during or after any design basis accident. The associated cable for the camera will be routed in a manner not to intervene between redundant safe shutdown areas to maintain divisional separation per Regulatory Guide 1.75. The cable will be adequately secured to items using material suitable for the expected environment. The penetration utilized by the cable for egress from the area will be re-sealed in accordance with approved processes utilizing acceptable specified material. The monitor/control unit for the camera will be in the auxiliary building corridor and will be properly secured. The monitor/control unit will be located in a suitable location providing adequate distance from other equipment. Combustibles associated with this change have been included in a Combustible Heat Load Calculation. These additional combustibles do not increase the fire loading above that which is presently described in the UFSAR. Consequently, this cable is a specialty cable supplied by the vendor for this camera and is not flame tested as an assembly. Therefore, Section 8.3.3.1 of the UFSAR requires revision to list/describe this specialty cable as a non-flame tested cable. The addition of this cable will not adversely affect the overall percent of non-flame tested cable at GGNS as stated in Section 8.3.3.1 of the UFSAR. This equipment will not interface/interact with any other equipment, therefore, its failure will not adversely affect the operation of any equipment. This equipment will be periodically powered by a local convenience outlet supplied by BOP power, therefore, any postulated electrical failure of this equipment will be isolated via feeder breaker at the power panel and will not degrade any Class 1E power sources. Although this equipment will be permanently installed, it is not required for any plant operation or function, therefore, it will not be considered permanent plant equipment.

Serial Number: 97-016-NPE Document Evaluated: ER 96/1022-00

DESCRIPTION OF CHANGE: Additional operational and inspection requirements for the SSW UHS during cold weather operation (notable ice storms).

REASON FOR CHANGE: Realization that additional threats to SSW UHS operability from cold weather operation (notable ice storms) exist that have not been previously addressed.

SAFETY EVALUATION: The additional operational requirements (regarding fan operation without cooling water flow before and during ice storms as well as SSW loop operation without fans to remove ice buildup on the tower fill) and inspection requirements (for ice buildup on the fill and around the main header piping) for the SSW system specified in this interim disposition provide adequate assurance for operability of the UHS in the near future. Long term solutions to the problems discussed in this safety evaluation will be addressed. Operation of the fans without cooling water flow does not impact the automatic initiation functions of the SSW system.

Serial Number: 97-017-PSE Document Evaluated: Temp Alt 97-0001

DESCRIPTION OF CHANGE: The Temp Alt will remove the blind from the flange upstream of Valve SP47F146 to allow connection of a surfactant tank to the PSW system.

REASON FOR CHANGE: Chemistry has chosen to treat the PSW system with a surfactant (PCL-361) to aid in the control of microbiological activity in the PSW system and in the reduction of fouling in plant heat exchangers.

SAFETY EVALUATION: The changes made by the Temp Alt will not compromise any existing safety-related system, structure, or component. These changes will not affect the ability to maintain the reactor in a safe shutdown condition.

As stated in Sections 9.2.8 and 9.2.10 of the SAR, neither the Plant Service Water (P44) nor the Radial Well (P47) systems have any safety-related functions. Failure of these systems will not compromise any safety-related systems or component and will not prevent safe reactor shutdown.

The P44 and P47 systems are not addressed in Chapter 15 of the SAR as necessary systems to mitigate an accident. Installation of the Temp Alt will not affect any other accident evaluated in the SAR or change requirement of the systems during an accident situation. Chemistry's Control Room Habitability screening of the surfactant (PCL-361) has concluded that the chemical poses no habitability concerns. Any leakage from the tank, hose, or flange will be contained inside of the chemical treatment facility's dike; therefore, the Temp Alt does not increase the potential of an unmonitored release to the environment.

The activities of the Temp Alt will not affect the operation of the P44 Secondary Containment Isolation valves listed in TRM Table TR3.6.4.2-1 nor will it affect the operation of PSW components that have been classified ASME Section III, Class 3 (SSW/PSW double isolation valves and PSW piping and valves serving ADHRS) as stated in Section 9.2.8.3 of the SAR.

Figure 9.2-27 of the SAR shows a "Future" tank location where the surfactant tank will be installed. The SAR addresses the use of a surfactant in the PSW system in Section 9.2.8.2 which states: "To prevent scale formation in system heat exchangers and iron oxide and suspended solids deposition, a dispersant and/or surfactant are injected on a continuous basis at a metered rate dependent on plant service water flow rate". Treatment of surfactant will be within the restrictions of NPDES permit number MS0029521. The surfactant (PCL-361) is currently being used n the Circulating Water System. Chemistry monitors corrosion rates of stainless steel, mild steel, and bronze coupons submerged in the Circ Water system and has seen no increase in corrosion rates of the coupons when feeding PCL-361. Because the Plant Service Water systems and the Circ Water Systems are constructed of similar material composition, it can be concluded that the use of the surfactant will not increase the corrosion rates in the PSW system or have an adverse affect on safety-related components in the PSW system. Additionally, PCL-361 was previously used in the PSW system (~1985-1986), and no adverse trends in corrosion rates or water chemistry were detected while using the surfactant.

The changes made in the Temp Alt will not create or introduce any unbounded failure modes for any safety-related system, structure, or component. The surfactant tank, temporary hose, and flange connection installed by the Temp Alt will not be in the vicinity of any safety-related equipment, and the potential failure of any of the equipment installed by the Temp Alt would not adversely affect the operation of a safety-related piece of equipment.

Serial Number: 97-018-NPE Document Evaluated: ER 97-0080-00-00

DESCRIPTION OF CHANGE: ER 97/0080 will allow for the optional removal of the trip unit rack tamper guard currently installed on panels 1H13-P618 and 1H13-P629.

Licensing Document Change Request (LDCR) 97-005 is required to revise the referenced UFSAR sections which describe the use of `keylocked barriers' on the affected panels as a means of compliance with IEEE-279,-1971, Section 4.18.

IEEE 279, Section 4.18 states: "The design shall permit the administrative control of access to all set point adjustments, module calibration adjustments and test points." This requirement is satisfied by existing plant administrative procedures for work control (Ref. IPC 88/4615).

REASON FOR CHANGE: As documented in Condition Reports spurious actuations of SRV trip units have been observed during the removal/installation of the trip unit rack tamper guards. These spurious actuations have been attributed to electro-static discharge which occurs during the manipulation of the tamper guards.

SAFETY EVALUATION: The removal of the trip unit rack tamper guards from the instrument racks located in panels 1H13-P618 and 1H13-P629 will not adversely affect any existing system, structure or component. The modification will have no affect on the operation of the associated instrumentation. The removal of the tamper guards will have no adverse affect on the seismic qualification of the affected panels or the instrumentation installed within the panels. Access to setpoint adjustments for the trip units located in the affected panels will be under administrative control per existing work control plant procedures in accordance with the requirements of IEEE 279, Section 4.18. The modification will have no affect on radionuclide population, release rate or duration. The modification will not create new release mechanisms or have any impact on radiation release barriers. The components to be removed serve no function to mitigate the radiological consequences of an accident or malfunction of equipment. The removal of the tamper guards from the instrument racks will have no impact on the Technical Specification operational requirements, surveillance requirements or setpoints for the associated instrumentation. The tamper guards are not required to meet any Technical Specification requirement and are not described in the Technical Specifications.

Serial Number: 97-019-NPE Document Evaluated: LDCR 97-006

DESCRIPTION OF CHANGE: UFSAR Appendix 16A, Section SR 6.2.2.11.4 and the corresponding TRM Section SR 6.2.2.11.4 state: " Verify that each required fire suppression pump starts sequentially to maintain the fire suppression water system pressure ≥ 120 psig" and require that the pumps start sequentially "upon a continued pressure drop in the fire suppression system". This change is to the UFSAR and TRM only and does not change any set points or make physical changes to the plant as presently designed.

REASON FOR CHANGE: The discrepancy identified between the minimum start pressure for any fire pump at GGNS (117 psig for "B" Diesel) and the TRM requirement that the fire pumps start sequentially to maintain the fire suppression water system pressure ≥ 120 psig. Fire pumps at GGNS consist of a small jockey pump to maintain pressure and three 100% capacity pumps (i.e. one pump can supply the largest fire water demand for water suppression systems identified in TRM Section 6.2.3 "Spray and Sprinkler Systems"). All three main fire pumps are nominal 1500 gpm at 125 psig. Set point pressures for starting these fire pumps are as follows: Jockey pump - 135 psig, Motor Driven Fire Pump - 129 psig; Diesel Driven Fire Pump "A" - 123 psig; & Diesel Driven Fire Pump "B" - 117 psig. This change will make the UFSAR and TRM consistent with the existing design of the fire water system.

SAFETY EVALUATION: This change is to the UFSAR and TRM only and does not change any set points or make physical changes to the plant as presently designed. The ≥ 120 psig" requirement presently specified in the above SAR & TRM sections is an arbitrary figure which has no basis for establishing operability of the sequential start feature of the fire pumps at GGNS. Section 6.2.2.11.2 is the section that establishes the minimum flow and pressure requirements for each fire pump. The purpose of Section 6.2.2.11.4 is to ensure the fire pumps start sequentially. Since all three main fire pumps at GGNS are 100% capacity pumps, a minimum sequential start pressure is not required for the system to perform its design function. Therefore, removing the requirement for a sequential start based on a specific arbitrary pressure and requiring a sequential start upon a continued pressure drop in the fire suppression water system will demonstrate that the sequential start feature will perform its intended design function.

The "Fire Suppression Water System", is not addressed by Technical Specification(s); therefore, no change to the TS or bases for any TS is necessary. The "Fire Suppression Water System" is addressed in the Technical Requirements Manual (TRM) Sections 6.2.2 and surveillance requirement changes made by this LDCR do adequately demonstrate operability of the system.

Serial Number: 97-020-NPE Document Evaluated: ER 96/0584

DESCRIPTION OF CHANGE: The Plant Discharge canal flow transmitter/recorder SN71R031 and sensor SN71N020 will be replaced with a Marsh-McBirney 253 flow transmitter, and sensor and a Westronics 2100 recorder. The new sensor utilizes a pressure transducer to determine level and the sensor electrodes are less prone to fouling. The existing temperature recorder SN71R032 and the new flow transmitter SN71N021 and recorder SN71R031 will be installed in a climate control portable building. A surge protector will be used to guard against lightning damage.

REASON FOR CHANGE: The Marsh-McBirney flow transmitter/recorder SN71R031 has consistently shown lower than actual water flow to the Miss river when compared to blowdown flow (computer point N716031). Flow readings are inaccurate and erratic. The Marsh-McBirney 250 uses a bubbler system for level determination that is prone to failures and the sensor electrodes are easily fouled. The instruments are also subject to degradation from environmental extremes (heat, cold and moisture) since they are mounted in a Hoffman box outside the plant.

SAFETY EVALUATION: The subject instruments are nonsafety related, nonseismic and nonseismic II/I. The operation and function of the Circulation Water (N71) system is not affected. No interfaces with other systems are created. This design change should improve the reliability of the existing flow monitoring system. This instrumentation is addressed in TRM 6.3.9 and not Tech Specs. The changes of this ER will not require that they be added to the Tech Specs. The failure of the affected instruments is not evaluated in Chapter 15 of the FSAR. The changes will not affect any other systems or components whose failures are evaluated. None of the UFSAR Section 10.4.5.3 safety evaluation for the N71 system is affected by these changes.

Serial Number: 97-021-PSE Document Evaluated: Temp Alt 97-0002

DESCRIPTION OF CHANGE: The Temp Alt will connect temporary chillers and booster pumps to the Plant Chilled Water system to provide additional cooling capacity to the Plant Chilled Water System while new chillers are installed by DCP 92/0006. Power to the temporary chillers and pumps will be provided by two 800 kW diesel generators supplied by the vendor. Each of the diesel generators will have a 2,300 gallon, double-wall fuel tank for fuel supply. All of the temporary equipment will be placed just North of the Unit 1 Turbine Building Train Bay. The temporary cooling system is expected to be on site for approximately 5 months.

The temporary chillers and booster pumps will connected to the Plant Chilled Water system via 6" flexible, braided hoses connected to abandoned Unit 1/Unit 2 PCW cross-tie piping. These flange connections are located in the Unit 2 Turbine Building. Cooling water to and from the temporary chilled water system will enter the Plant Chilled Water System through valves SP71F321 and SP71F322 (both valves are located in the Unit 1 Turbine Building).

REASON FOR CHANGE: The replacement schedule of an individual plant chiller by DCP 92/0006 is expected to have a duration of approximately 20 days. The temporary chillers will compensate for lost performance of an existing Plant Chiller due to loss of operation, due to fouling of its condenser from Plant Service Water, or due to maintenance problems.

SAFETY EVALUATION: The Temp Alt will not compromise any existing safety-related system, structure, or component and will not affect the ability to maintain the reactor in a safe shutdown condition.

As stated in Section 9.2.7.3 of the SAR, other than the Containment and Auxiliary Building isolation valves, the Plant Service Water system has no safety-related function. Failure of the system will not compromise any safety-related system or component and will not prevent safe reactor shutdown. The modification will not affect any safety-related component or affect operation of P71 Containment and Auxiliary Building isolation valves.

The supply and return hoses of the temporary chillers will be connected to abandoned in place Unit 2 piping that was to be used as a cross-tie between the Unit 1 and Unit 2 Plant Chilled Water systems. This piping is located in the Unit 2 Turbine Building. Section 10.4.5 of the SAR evaluates the effects of flooding from pipe breaks in the Circulating Water system, and the analyses assumes the security wall separating the Unit 1 and Unit 2 turbine buildings "not to be installed or fails". The volume of the water contained in the Circulating Water system is considerably larger than in the temporary cooling system. Therefore, the temporary system will have no effect on plant safety due to flooding.

Four temporary chillers utilizing freon R-22 (470 # per chiller) will be located in the yard (elevation 133') just North of the Turbine Building Train Bay. ER 97-0125 evaluated the temporary chillers in respect to Control Room Habitability in case a freon leak developed during operation on site. The ER concluded that the operation of the chil ‹rs is bounded by the previous NPE analysis "Control Room Habitability Effects of Chemical Spills" (MC-Q1Z51-920003). Thus, no significant hazard to Control Room personnel will occur due to the location of the temporary chillers and the amount of freon contained in them.

Power to the temporary chillers will be provided by diesel generators supplied by the vendor. Therefore, the temporary chillers and pump can in no way affect an electrical load of any equipment important to safety.

ER 97/0125 performed a Fire Hazards Analysis and Control Room Habitability review of the two 2,300 gallon diesel storage tanks that will be used as part of the temporary chiller skid, and concluded that the diesel storage tanks will not create conditions beyond which were not assumed in the fire hazards analysis report. The ER imposed requirements to follow to minimize the chance of fire from the temporary diesels and storage tanks. These requirements will be controlled by the Temp Alt's implementing work order. A potential fire of the temporary diesels was considered and found not to have a detrimental effect on Control Room Habitability due to the following reasons: 1) the fire's heat and intensity would be such that its smoke plume would rise above the Control Room intake, 2) the Control Room's smoke detection system would detect any smoke entering the Control Room and trip the operating air conditioning unit and initiate the Control Building Purge Fan, if not already running, to exhaust smoke from Control Room, envelope, and 3) Operations is aware of temporary diesels and storage tanks and could manually isolate Control Room ventilation in case of fire. Additionally, SAR Section 2.2.3 analyzed the effect of a potential fire from the diesel storage tanks located on site and concluded that a fire would have no detrimental effect on Control Room Habitability.

Exhaust from the temporary diesel generators will not effect Control Room Habitability due to the dissipation of the exhaust in the air and the distance the temporary diesels are from the Control Room intake. Additionally, the temporary diesel generators (800 kW) will produce much less exhaust than is produced by the Div I and Div II Diesel Generators (7000 kW) and the Div III diesel generator (3500 kW) when in operation. Therefore, the exhaust from operation of the temporary diesel generators will not exceed the diesel exhaust limits shown in FSAR Table 2.2-7 or have a detrimental effect on Control Room Habitability.

SAR Section 2.4.10 describes the evaluation of worst case flood conditions but evaluates only doors for Unit 1. The hoses of the temporary chiller will require Door 2T303 to be opened which is located in the Unit 2 turbine building on the 133' elevation. Door 2T303 is not addressed in the SAR; however, the use of the door will not affect the current SAR flooding analyses becaus> all assumed flood prevention devices will be place during the Temp Alt.

Serial Number: 97-022-NPE   Document Evaluated: Proposed Chemical Cleaning of Q1T51B002-A and Associated P41 Piping

DESCRIPTION OF CHANGE: The Standby Service Water (SSW) side of LPCS Room Cooler Q1T51B002-A unit and associated piping will be chemically cleaned. The chemical solution to be used in the cleaning process is Betz DE-1178. The chemical composition of these solutions are 40% Citric Acid, 10% Phosphate, 10% corrosion inhibitor and 40% inert ingredients. The system boundaries established for cleaning of the Q1T51B002-A Room Cooler are shown on P&ID M-1061B between valve Q1P41F037 and valve Q1P41F038. The materials of construction between the established boundaries that will be exposed (wetted) by the selected cleaning process have been identified and evaluated for compatibility with the specified chemical solution.

REASON FOR CHANGE: The coils/tubes (I.D) within the room cooler will be chemically cleaned to help enhance the heat transfer capability of the unit.

SAFETY EVALUATION: A chemical cleaning process using Betz DE-1178 has been selected to improve the cleanliness of the room cooler coils/tubes (I.D) which will enhance the heat transfer capability of the units. The materials of construction between the established boundaries of the affected P41 and T51 systems and equipment have been reviewed and it has been concluded that the materials and equipment will not be compromised or adversely impacted by performance of the selected chemical cleaning process or by the use of the specified chemical solution. Based on the reference documents, the affected materials are compatible with the selected chemical cleaning process and specified chemical solution.

Other types of corrosion (crevice, IGSCC, pitting, etc.), corrosion of welding/brazing metal or some other corrosion mechanisms possibilities were reviewed and determined not to be factors due to the nature of the selected chemical cleaning process, the specified chemical solution, the wetted materials within the established boundary and by following the prescribed process controls.

The process controls for the selected chemical cleaning process are established as (1) A 10% solution of the specified chemical solution BETZ DE-1178 (based on the volume of the established system boundary) is prepared, injected, and recirculated until one of the following process control limits are met (a) dissolved Iron level ceases to increase (b) dissolved Copper reaches 700 ppm (c) three hours maximum (2) Minimize low flow rates and stagnant conditions (3) Flush the affected system with water after the cleaning process (4) The total number of chemical cleanings is limited to 10 times without further evaluation of the available corrosion allowances or design conservatism's.

The selected chemical cleaning process using BETZ DE-1178 chemical solution on the materials of construction between the established boundaries will not compromise or have adverse impact on the affected P41 or T51 systems or equipment, provided the established process controls are followed. Thereby, the margin of safety in accordance with the design requirements and functional capabilities of the systems is maintained. Therefore, it is concluded that the use of the selected cleaning process as described in this evaluation does not increase the probability or the consequences of any accident evaluated in the SAR, does not create the possibility of a new accident or malfunction, and does not reduce any margin of safety defined in the basis for any technical specification. There are no unreviewed safety questions or issues resulting from using the selected chemical cleaning process or specified chemical solution as discussed in this evaluation.

Serial Number: 97-023-NPE Document Evaluated: GGCR1997-0082-00

DESCRIPTION OF CHANGE: The Drawing Revision Notice (DRN) 5008 provided a miscellaneous as-builting of a vendor print showing the Model number of the relief valves on the Division III starting air storage tanks P81F048A/B and P81F049A/B. This evaluation will address the material change and the UFSAR figure change.

REASON FOR CHANGE: The model number had previously been changed but the vendor drawing and associated UFSAR figure had not been updated. Issuance of the DRN ensured automatic update of the UFSAR figure via the existing UFSAR Figure update process controlled by Configuration Management.

SAFETY EVALUATION: Replacing the existing P81F048A/B and P81F049A/B relief valves on the Division III starting air storage tanks Farris model number 1875-OL with Farris model number 1855-OL will not reduce the reliability of the HPCS DG air start system. This change will ensure that the Division III starting air system operating pressure does not exceed the design pressure of the protected components by providing suitable replacement parts when required.

Serial Number: 97-024-PSE Document Evaluated: Temp Alt 96-0028

DESCRIPTION OF CHANGE: This temporary alteration (temp alt) provides a connection on the P21F341 valve for a level instrument sense line at the Demin Water Storage Tank (SP21A001), such that an automatic function will be realized for starting and stopping the vendor water treatment trailer, relevant to tank level. A sample valve for water samples to be taken by chemistry is also provided at the P21F341 valve with adequate heat tracing to prevent the pipe from freezing in inclement weather.

REASON FOR CHANGE: This temp alt will provide automatic starts/stops of vendor supplied water purification trailer for Demin Water Storage Tank Makeup with no human interfacing and allow sampling of storage tank contents by chemistry at the tank.

SAFETY EVALUATION: The changes made by the Temp Alt will not compromise any existing safety-related system, structure, or component. The changes will not affect the ability to maintain the reactor in safe shutdown condition.

The makeup water treatment system has no safety-related function. Failure of the system will not compromise any safety-related system or component and will not prevent safe reactor shutdown. Primary and secondary isolation valves identified per Tech Spec Table TR3.6.1.3-1 and TR3.6.4.2-1 are in no way hindered from performing their intended function.

The implementation of the Temp Alt will not alter the design intent of the P21 system and no potentially radioactive water will be introduced into the system, nor will the Temp Alt create any Seismic II/I concern. The Demin Water Storage Tank is located in the yard next to the Water Treatment Building and implementation of this Temp Alt will not affect, nor be located in the vicinity of, any safety related equipment which would be affected indirectly.

Serial Number: 97-025-NPE Document Evaluated: ER 96-0509-01

DESCRIPTION OF CHANGE: Material Nonconformance Report (MNCR) 96-0121 (1) reported gaps in the blackness test area that were larger than those assumed in the GGNS criticality analysis. Consequently, the blackness test area and directly adjacent cells have been temporarily restricted from holding fuel as evaluated in Safety Evaluation 96-0067-R00. The GGNS criticality analysis has recently been revised to bound the Boraflex degradation experience in the test area. (2) This safety evaluation assesses (i) removing the temporary restriction on fuel loading in the test area and the adjacent cells and (ii) extending the GGNS blackness test interval from the current 18 months to 36 months as reported to the NRC in GNRO-96/00118. (3) Consistent with the extension of the blackness testing interval, the requirement to load the test area with freshly discharged fuel during each refueling outage will be deleted. Cycle-specific analysis associated with each incoming reload batch (and documented in a safety evaluation) will confirm the continued applicability of this 36-month interval and evaluate the need for loading the blackness test area with freshly discharged fuel.

REASON FOR CHANGE: As reported in SAR Section 9.1.2.3.2.1, fuel loading in areas where excessive degradation (defined as gaps in excess of the 4% assumed in the CSA) is indicated, will be controlled administratively until a safety evaluation is performed. This assessment also supports the engineering evaluation associated with MNCR 96-0121 and the associated changes to the blackness test program.

SAFETY EVALUATION: This safety evaluation concludes that (i) loading the blackness test area and directly adjacent cells with any fuel type currently in the plan and (ii) extending the blackness test surveillance interval to 36 months will not constitute an unreviewed safety question and will not change the GGNS Technical Specifications nor reduce the margin of safety as defined in any Technical Specification Bases.

Serial Number: 97-026-PSE Document Evaluated: WO 00184798

DESCRIPTION OF CHANGE: The LPCS Pump Room Cooler Q1T51B002 failed to meet the heat removal requirements of MS-39.0 as determined by the thermal performance test conducted on 3/26/97, therefore the room cooler will need to be cleaned to increase its heat removal capability. The room cooler will be flushed with Betz DE 1178 a mild citric acid. Betz DE 1178 clearing solution is used to flush the deposits out of the cooler/piping thus restoring cooler efficiency. Betz DE 1178 is 40% critic acid, 10% phosphate, 10% corrosion inhibitor and 40% inert ingredients. The use of Betz DE 1179 cleaning solution in Q1T51B002 room cooler is approved for use by ER 97/0271 and SE 970023 R00.

REASON FOR CHANGE: FSAR Section 9.4.5.2.4 states that each safety-related pump room is provided with a full capacity fan-coil unit to prevent the room temperature from exceeding 150°F during pump operation. Because of the fouling in the piping/tubing, the heat removal capability of the LPCS Pump Room Cooler is less than the requirements of MS-39.0.

SAFETY EVALUATION: The materials of construction was reviewed and evaluated for compatibility under Safety Evaluation 97/0023 R00. The process controls established by SE 97/0023 R00 are (1) A 10% solution of the chemical solution Betz DE 1178 (Based on volume of the established system boundary) is prepared, injected, and recirculated until one of the following process control limits are met (a) dissolved Iron level ceases to increase (b) dissolved copper reaches 700 ppm (c) three hours maximum flush (2) Minimize low flow rates and stagnant conditions (3) Flush the affected system with water after the cleaning process (4) The total number of chemical cleanings is limited to 10 times without further evaluation of the available corrosion allowances. The instruction for cleaning the room cooler implements the controls given by ER 97/0271 and SE 970023 R00. This safety evaluation evaluates method of acid flushing and ensuring the process controls are implemented. It is concluded that the use of the cleaning instructions does not increase the possibility or consequences of an accident evaluated in the SAR, does not create the possibility of a new accident or malfunction and does not reduce any margin of safety defined in the basis for any technical specification. There are no unreviewed safety questions or issues from using the cleaning instruction discussed in this evaluation.

The floor drains in the surrounding area will be intentionally covered during performance of this activity to ensure that acid solution does not inadvertently enter the floor drain system. The floor drain system serves to divert gross leakage away from affected equipment in the area. During performance of the flush activity, personnel will be continuously on the job. In the event of that a flood event were to occur, necessitating uncovering the floor drain, this can be accomplished by personnel at the job site. Per FSAR Section 6.3.1.1.3, the ECCS room are constructed to be water tight to protect against mass flooding of redundant ECCS pumps therefore temporary covering of two floor drains will not increase the consequences of a flooding event.

Serial Number: 97-027-NPE Document Evaluated: CN 97/0004 to DCP 83/4070

DESCRIPTION OF CHANGE: Respan Offgas (N64) flow transmitters, 1N64N033A/B, N011, & N062 in order to change the current 4-40 scfm and 4-400 scfm non-linear ranges to the proposed 0-40 scfm and 0-400 scfm linear ranges. The scales for the corresponding flow indicating switches and recorders, 1N64R616A/B, R035, R036, R617, and R620, will also be changed to be consistent with transmitter ranges. UFSAR Section 7.7.1.10.3.4 will be changed to correctly show the low and high flow ranges for Offgas System flow measurements. UFSAR Figure 11.3-6 will be updated to show the correct mounting configuration for 1N64N011 and 1N64N062. These transmitters are locally mounted and not panel mounted. The mounting changes were approved by DCP 83/4070.

REASON FOR CHANGE: The changes proposed to these Offgas System flow instruments were requested in order to use all standard, linear ranged components.

SAFETY EVALUATION: The respanning of Offgas afterfilter discharge and adsorber discharge flow transmitters (1N64N033A/B, 1N64N011, and 1N64N062) to linear 0-40 scfm and 0-400 scfm ranges is acceptable based on the range requirements given by GE Design Specification 22A3089. The ranges given by the design specification are 3-30 scfm for low flow and 3-300 scfm for high flow. These system requirements are bound by the proposed 0-40 and 0-400 ranges, therefore the linear 0-40 scfm and 0-400 scfm ranges requested by CN 97/0004 are conservative and maintain the design requirements of the instrumentation. Rescaling the flow switches and recorders (1N61R035, R036, R616A/B R617, R620) associated with the Offgas System afterfilter discharge and adsorber discharge flow transmitters will have no affect on the intended design functions of the components. The changes proposed make the indication linear and therefore consistent with sensed flow. The use of linear Offgas flow indication on panels 1H13P845 and 1N64P002 will also improve the readability of the indication. The rescaling of flow indicating switches will not impact associated control room alarms. The proposed changes will not decrease the functionality of the components. Therefore, the proposed changes will not increase the probability of occurrence of an accident previously evaluated in the SAR, impact radiological consequences or cause any reduction to the margin of safety.

Serial Number: 97-028-PSE Document Evaluated: FSAR LDC 97-023

DESCRIPTION OF CHANGE: Revise FSAR to modify description of SRM and neutron source uncovery (withdraw one of four control rods surrounding each chamber or source) during normal startup procedures.

REASON FOR CHANGE: Continue reactor startup with control rod withdrawal in accordance with normal startup procedures if criticality is achieved before uncovery of each SRM chamber.

Neutron sources are no longer installed in the core.

SAFETY EVALUATION: Relaxation of the requirement to uncover each SRM (withdraw one of four control rods surrounding chamber) and neutron source before the reactor is critical is acceptable. It will not increase accident or equipment malfunction probabilities or consequences. It will not create risk of an accident or equipment malfunction of a type different than any previously evaluated in the SAR or result in a decrease in a margin of safety. Further, discrete neutron sources have been removed from the core since irradiated fuel loaded in the core provides sufficient neutron flux; therefore reference to them is not applicable. There are no unreviewed safety questions.

Serial Number: 97-029-NPE Document Evaluated: Q1T51B005-B and Associated P41 Piping

DESCRIPTION OF CHANGE: The Standby Service Water (SSW) side of RHR "C" Room Cooler Q1T51B005-B unit and associated piping will be chemically cleaned. The chemical solution to be used in the cleaning process is Betz DE-1178. The chemical composition of this solution is 40% Citric Acid, 10% Phosphate, 10% corrosion inhibitor and 40% inert ingredients. The system boundaries established for cleaning of the Q1T51B005-B Room Cooler are between valve Q1P41F047 and valve Q1P41F048 (Reference P&ID M-1061B). The materials of construction between the established boundaries that will be exposed (wetted) by the selected cleaning process have been identified and evaluated for compatibility with the specified chemical solution.

REASON FOR CHANGE: The coils/tubes (I.D.) within the room cooler will be chemically cleaned to help enhance the heat transfer capability of the unit.

SAFETY EVALUATION: The materials of construction between the established boundaries of the affected P41 and T51 systems have been reviewed and it has been concluded that the design margins on the affected P41 and T51 systems will not be adversely impacted by the selected chemical cleaning process using the Betz DE-1178 chemical solution or by performance of the selected chemical cleaning process. Based on the reference documents, the affected materials are compatible with the selected chemical cleaning process and specified chemical solution. Ref. 1 documents testing conducted to determine corrosion rates of Copper/Nickel (90% Cu-10% Ni) materials. The testing showed that the average metal loss from general corrosion of Copper/Nickel (90% Cu-10% Ni) was 0.08 mils based on 4 hour exposure data with the proposed chemical solution. this corrosion rate was shown to be conservative by testing documented in Ref. 2. By review of the heat exchanger data sheet in 9645-M-611.0, R/14 and Attachment 2 of Engineering Report SERI-88-0006, Rev. 0 the tubes/coil and header are 90% Cu-10% Ni with a design conservatism (corrosion allowance) of 27 mils and 12 mils respectively. The bounding design corrosion allowance used for carbon steels was the HBC piping at 80 mils in accordance with GGNS-MS-03. The average metal loss for carbon steel from general corrosion due to exposure to the proposed cleaning solution was 1.6 mils based on 4 hour exposure data s presented in Ref. 1. An evaluation was performed using this data and the methodology presented in ASME Boiler & Pressure Vessel Code Section ND-3641.1, NPE concluded that a 4 hour cleaning could be performed 10 times and still not reduce the pipe wall thickness below the design minimum wall thickness plus the design corrosion allowance. For stainless steels, the average metal loss from general corrosion was less than 0.000046 mils based on the 4 hour exposure data as presented by Ref. and Ref. 3, which is insignificant, and a bounding design corrosion allowance for stainless steels is not necessary and need not be considered further.

The process controls for the selected chemical cleaning process are established as (1) A 10% solution of the specified chemical solution Betz DE-1178 (based on the volume of the established system boundary) is prepared, injected, and recalculate until one of the following process control limits is met: (a) dissolved Iron level ceases to increase, (b) dissolved Copper reaches 700 ppm, or (c) three hours maximum; (2) Maximize flow rates and minimize stagnant conditions; (3) Flush the affected system with water after the cleaning process; (4) The total number of chemical cleanings is limited to 10 times without further NPE evaluation.

The selected chemical cleaning process using Betz DE-1178 chemical solution on the materials of construction between the established boundaries will not adversely impact the design margins of the affected P41 or T51 systems, provided the established process controls are followed. Thereby, the margin of safety in accordance with the design requirements and functional capabilities of the systems are maintained. Therefore, it is concluded that the use of the selected cleaning process as described in this evaluation does not increase the probability or the consequences of any accident evaluated in the SAR, does not create the possibility of a new accident or malfunction, and does not reduce any margin of safety defined in the basis for any technical specification. There are no unreviewed safety questions or issues resulting from using the selected chemical cleaning process or specified chemical solution as discussed in this evaluation.

Serial Number: 97-030-NPE Document Evaluated: ER 97/0091-00-00

DESCRIPTION OF CHANGE: FSAR Section 6.2.7 is being revised to accurately reflect changes made to the Suppression Pool Makeup (SMPU) system initiation set point.

REASON FOR CHANGE: Material Nonconformance Report (MNCR) 0143-89 was written to document a nonconservative Tech Spec set point for initiation of SPMU. The MNCR disposition changed the Tech Spec set point to its present conservative setting, but did not change the FSAR to reflect the new Tech Spec set point. Condition Report (CR) 960588 was initiated to document the omission of the changes to the SAR that should have been made per MNCR 0143-89.

SAFETY EVALUATION: The analytical limit for low low suppression pool level is 16'10". Therefore, in order to maintain vent coverage, the SPMU must initiate at 16'10" or above. SAR Section 6.2.7.3.1 discusses system initiation. The SAR states that there is a 1-1/2 minute delay from the LOCA signal to the suppression pool low low level signal and the resulting SPMU initiation. It also states that this 1-1/2 minute delay assures that the drywell pressure transient due to vessel blowdown has ended prior to dumping of the upper pool and corresponding increase of vent submergence. In the SAR, the 1-1/2 minute delay is called the "volume integrated delay. " Prior to MNCR 0143-89, the low low suppression pool set point was calculated by determining instrument drift and uncertainty and applying this margin to the nonconservative side of the analytical limit, i.e. the set point was set such that the low low level signal would occur at 16'10" or lower. Based on the disposition of MNCR 0143-89, the Tech Specs were changed to apply the instrument error on the conservative side of the analytical limit. When the Tech Spec change was made, it was not recognized that the volume integrated delay could be affected by the change. However, the 1-1/2 minute value is based on maximum ECCS pump flow beginning concurrently with the LOCA signal. In fact, there will be a significant delay before all of the pumps will be capable of injecting due to the time required for the valves to open and for vessel pressure to decrease to a low enough pressure. In addition, vessel inventory mass added to the pool is not considered in the calculation of the volume integrated delay time. Therefore, the decrease in the volume integrated delay will not shorten it such that the SPMU initiation would occur before the end of the drywell pressure transient. The volume integrated delay is not discussed in the Tech Spec Bases. Since the 1-1/2 minute value is not a part of the Tech Spec Bases and is actually only an estimate of the delay that would occur, the SAR is being modified to remove the specific number and to include a qualitative discussion of the volume integrated delay.

Serial Number: 97-031-NPE Document Evaluated: MCP 95/01042

DESCRIPTION OF CHANGE: The changes addressed are to the Domestic Water System (P66). They involve disconnecting, removing/abandoning in place Emergency Decontamination showers that provide no radiological or industrial safety benefit.

REASON FOR CHANGE: Emergency Decontamination showers have historically been a high maintenance item. Health Physics does not utilize these showers since (personnel) decon showers are performed only at the Health Physics decon area (93' elevation of the Control Building).

SAFETY EVALUATION: The Domestic Water System has no safety-related function as defined in Section 3.2 of the FSAR. Failure of the system will not compromise any safety-related equipment or component and will not prevent safe shutdown of the plant. The modifications made will in no way impact any of the accident analyses presented in the FSAR. No new failure modes are being created, thus no possibility of an accident or malfunction of a different type than previously analyzed is possible. Failure of the system will not compromise any safety-related system or component and will not prevent safe reactor shutdown, thus the margin of safety will not be reduced. Plant Safety has evaluated the P66 Emergency Decontamination showers to determine which showers should be retained in regard to OSHA 29CFR 1910.151 requirements and which showers may be incapacitated. System Engineering performed a review and walkdown of these showers in conjunction with Safety and concurs with their recommendation. Therefore, those Emergency Decontamination showers identified as providing no plant, personnel or HP safety benefit, and are a high maintenance item, shall be permanently decommissioned.

Serial Number: 97-032-NPE Document Evaluated: LDC 96-108

DESCRIPTION OF CHANGE: Section 7.3.1.1.1.4.12.9 of the UFSAR which discusses the Safety Relief Valve Low-Low Setpoint Logic has been clarified in regards to interdivision redundancy and the single failure criterion.

REASON FOR CHANGE: On June 6, at 11:26 a.m., GGNS was manually scrammed based on increasing suppression pool temperatures. It was noted that six Safety Relief Valves were open. These SRVs were identified as the low-low set valves. Based on the Sequence of Events (SOE) Log from the Plant Data System (PDS), all twenty SRVs received a signal to open. After approximately 200 ms, the open signal was removed from the fourteen non low-low set SRVs. The low-low set SRVs remained open past insertion of the manual scram. The duration of time which the low-low set SRVs remained open was between 2.5 to 3 minutes.

The cause for the transient has been determined to be the failure of a capacitor in trip unit 1C11N655B which monitors first stage turbine pressure and is not functionally related to the trip units that provide S/RV or low-low set relief logic input. The failure of this capacitor created a short circuit which resulted in the opening of fuse 1E12-F38 as per design. The effects of this short circuit (i.e. power supply variation), resulted in the initiation of the S/RV trip logic. Laboratory simulation indicates this disturbance influenced the S/RV trip units via response of the transmitters to the voltage fluctuation.

IEEE Standard 379-1977, "IEEE Standard Application of the Single Failure Criterion To Nuclear Power Generating Station Class IE Systems", is the primary basis document for application of single failure analysis to GGNS Class IE Systems. This standard addresses application of the criterion to ensure that the system under consideration is capable of performing protective action(s) required to accomplish the required protective function(s) in the presence of any single detectable failure within the system concurrent with all identifiable but undetectable failures, all failures occurring as result of the single failure, and all failures which would be caused by the design basis event requiring the protective function. Thus, the primary focus is achievement of required function(s).

The treatment of inadvertent system actuations is limited to determination of whether such actuations would constitute an event with unacceptable safety consequences. Only for any such actuation where the safety consequences are identified as being unacceptable must the single failure criterion be met (that is, the class IE systems must not initiate the action as a result of any single detectable failure in addition to all undetectable failures in the system). Additional guidance related to intent and application of the single failure criterion is provided in IEEE Standard 603-1980, "IEEE Standard Criteria for Safety systems for Nuclear Power Generating Stations", Section 5.1. In the language employed in this section, the standard clearly states that the single failure criterion "...does not invoke coincidence (or multiple channel) logic within a safety group; however, the application of coincidence logic may evolve from other criteria or considerations to maximize plant availability or reliability. "

GGNS design of the S/RV and low-low set point logic has incorporated two safety groups of redundant equipment (Division 1 and Division II) which can perform the system protective function, and these two safety groups have design independence with no points of common vulnerability identified (including consideration for the S/RV actuators with both safety groups (trains having actuation capability at each valve location). Thus, the single failure criterion for achievement of the required function is satisfied.

The remaining consideration for single failure design requirements is that of inadvertent system actuations. Such an actuation would likely be of short duration for the valves that are not designed to perform the low-low set function. This is because the reset pressures for the non low-low set S/RVs are much closer to the initial operating pressure at the onset of a logic circuitry/power supply circuitry component failure-induced events; therefore is an actual lift occurred these valves would quickly reach their reset setpoints and reclose. The reset point for the trip units controlling the low-low set valves is much lower than normal operating pressure and in addition, the low-low set logic seals in and requires operator action to be reset.

SAFETY EVALUATION: Evaluation of the safety consequences for inadvertent S/RV actuation, Engineering Report GGNS-96-0037, Rev. 1, has determined that an initial lift of all S/RVs is bounded by design considerations for vessel overpressure protection capabilities. A sustained inadvertent lift of the low-low set S/RVs is also bounded by design considerations. Appropriate instrumentation and control capabilities to detect and respond to this occurrence are also provided by the existing design (i.e. the condition would be indicated by plant annunciators and status indicators, the low-low set logic has manual reset capability, and the S/RVs can be manually controlled by the operator if desired).

Serial Number: 97-033-NPE Document Evaluated: MCP 95/1020

DESCRIPTION OF CHANGE: A pressure regulating valve will be installed downstream of each rough flow adjustment valve in sample panel 1G33Z020. A new pressure gage is being installed downstream of each regulator to monitor pressure. New relief valves are being installed downstream of each regulator for equipment/personnel protection in the event the regulator fails.

REASON FOR CHANGE: Pressure spikes are causing flow indicators in the panel to leak and explode creating both personnel safety and ALARA concerns. These Fischer Porter rotameters are rated for 250 psig but could potentially be exposed to pressures above 1700 psig. Rotameters rated for 3000 psig are available which would solve the problems with the flow indicators. However, the conductivity cells in the panel still have a 200 psig rating and the oxygen analyzer cells have a 50 psig rating.

SAFETY EVALUATION: The reactor sample panel is non safety related, non seismic category I. The tubing inside the panel is per ANSI, not ASME. The sample lines are connected to ASME piping but the panel is adequately isolated by isolation valves and flow restrictors (reference FSAR Section 9.3.2.3). This isolation capability is not being affected. The changes that are made will be done in accordance with applicable codes and standards including ANSI B31.1 and J-621.0. No interfaces will be created. This design change should improve the reliability of the dissolved oxygen and conductivity monitors in the panel. This instrumentation is addressed in TRM 6.4.1, not the Tech Specs. The changes will not require that they be added to the Tech Specs. The calibration/setpoints of the affected conductivity and oxygen analyzers are not being changed.

Serial Number: 97-034-NPE   Document Evaluated: ER 96/0285-00-00

DESCRIPTION OF CHANGE: Remote Shutdown System level transmitter and level indicator for the Condensate Storage Tank (CST), 1C61N102 and 1C61R102, will be respanned and rescaled in order to correspond with the CST level instrumentation for the main control room, 1P11N003 and 1P11R601.

REASON FOR CHANGE: Changes proposed to the Remote Shutdown System CST level indication will allow Operations to use the Remote Shutdown System as backup indication for the control room instruments that monitor CST level. Changes will allow indication over the entire range of CST operation.

SAFETY EVALUATION: The Remote Shutdown System Condensate Storage Tank (CST) level transmitter and level indicator are designed to monitor CST level over its anticipated range during a loss of control room habitability. As part of the Remote Shutdown System they are designed to be totally independent of control room instrumentation. Both components are safety related and are required to meet Seismic Category I requirements. Respanning the Remote Shutdown CST level transmitter and rescaling CST level indicator in order to use them as an identical backup for control room CST level indication will have no adverse effects on the components themselves or the Remote Shutdown System (C61). The proposed changes will provide Ops with an identically scaled backup to the current CST control room indication. The range change to the Remote Shutdown instrumentation is considered an enhancement because the range is being increased to include indication of the entire range of CST operation and not just the anticipated accident range. The design requirements for Remote Shutdown CST level indication are not changed. Remote Shutdown CST level indication is taken from the tap that goes to the instruments that initiate the HPCS/RCIC automatic suction transfer. The proposed changes will not decrease the functionality of the components. Therefore, the proposed changes will not increase the probability of occurrence of an accident previously evaluated in the SAR, impact radiological consequences or cause any reduction to the margin of safety.

Serial Number: 97-035-NPE   Document Evaluated: MCP 94/1061, SCN JS08-95/0040A

DESCRIPTION OF CHANGE: This Minor Change Package (MCP) replaces the controllers and adds additional temperature monitoring on the four drywell chiller skids on elevation 119'-0" of the auxiliary building in accordance with vendor recommendations. In addition, SCN JS08-95/0041a revises JS-08 or reflect the additional temperature monitoring equipment and the new controllers.

REASON FOR CHANGE: The drywell chillers require significant maintenance and troubleshooting efforts due to operational problems and lack of flexibility in operating combinations of the four chillers. The vendor has recommended changes to the control scheme which includes new microprocessor based controllers and additional process monitoring to alleviate these problems. This MCP installs these modifications.

SAFETY EVALUATION: The drywell chilled water system (P72) has no safety related function per UFSAR Section 9.2.11.3 other than the containment isolation portion of the system. The controls for the drywell chillers are discussed briefly in the UFSAR text in Section 9.2.11.5. The controls and sensing points are shown on P&IDs included as UFSAR Figures 9.2-23C, 9.2-48, 9.2-49 and 9.2-50. The modifications installed by this MCP do not affect the overall function or operation of the drywell chilled water system. None of the affected equipment is required to mitigate the consequences of an accident nor required for safe shutdown. The only portions of the interfacing systems considered safety related are those portions forming part of containment boundary which are unaffected by this change. UFSAR Section 3.2 classifies equipment affected by this modification as non-Q and non-seismic. The design, fabrication, installation, examination and testing of this modification are commensurate with the original design code. The system modifications will conform to all applicable design and material specification requirements and required construction practices. The modification will not increase the consequences or the probability of occurrence of any accident or transient analyzed in chapter 15 of the FSAR, nor will it increase the consequences or probability of a malfunction of equipment important to safety. No new accident scenarios or malfunctions of equipment important to safety are introduced as a result of this change. The Technical Specifications do not address the drywell chilled water system. As well, the affected portions of the interfacing systems are not governed by any Technical Specifications. Based on the above, this modification does not constitute an unreviewed safety question or reduction in the margin of safety.

Serial Number: 97-036-NPE   Document Evaluated: ER 97-0332-00-00

DESCRIPTION OF CHANGE: This safety evaluation will evaluate the change in FSAR Figure 9.4-004. The proposed change to the Figure will reflect the "as built" plant conditions of independent air intakes SV41Y701A/B and independent HVAC suction ductwork to the Radwaste Building Supply Air Fans SV41C002A/B. The Figure presently shows independent air intakes with common HVAC suction ductwork between the air intakes and the respective fans.

REASON FOR CHANGE: Condition Report GGCR 1997-0249-00 documented and identified a deviation between the "as built" condition of the HVAC ductwork and suction intakes SV41Y701A/B to the Radwaste Building Supply Air Fans SV41C002A/B and the configuration delineated on P&ID M-0047A. ER 97-0332-00-00 evaluated the deviation and identified that FSAR Figure 9.4-004 and SFD-0047 SH1 also failed to reflect the "as built" conditions in the plant.

SAFETY EVALUATION: Research into the relevant sections of the FSAR, the Technical Specifications, and the Offsite Dose Calculation Manual fully support a conclusion that the proposed change to Figure 9.4-004 of the FSAR (as prescribed by ER 97-0332-00-00) does not yield an unreviewed safety question. In addition, extensive research failed to find any additional changes required to design base documents outside the bounds of those already identified in ER 97-0332-00-00. Based on the findings of this safety evaluation, Figure 9.4-004 of the FSAR should be changed to reflect the "as built" conditions of independent air intakes SV41Y701A/B and independent HVAC suction ductwork to the Radwaste Building Supply Air Fans SV41C002A/B.

Serial Number: 97-037-NPE   Document Evaluated: MCP 95/1012
(EAR NO. E-96/013)

DESCRIPTION OF CHANGE: The proposed change abandons-in-place heat trace circuits associated with the GGNS Liquid (G17) and Solid (G18) Radioactive Waste Systems. The heat trace circuits were installed to support operation of radwaste system evaporators. However, UFSAR Section 11.2.2.7 states that the radwaste evaporators have never been used to process radioactive liquid wastes and there are no plans to use these evaporator units in the future. Thus, maintenance and use of equipment associated with these evaporator units, such as heat trace circuits, is not necessary for routine plant operations. The proposed change will result in electrically disconnecting the affected heat trace circuits. However, to minimize radiation exposures to plant personnel, the heat trace circuits will remain installed on the associated piping systems and will not be physically removed from the plant.

REASON FOR CHANGE: The proposed change is being conducted to resolve discrepancies between the actual configuration of plant equipment and the associated design documents.

SAFETY EVALUATION: The proposed change, abandoning-in-place heat trace circuits associated with the Liquid (G17) and Solid (G18) Radioactive Waste Systems, is necessary to design documents. The affected heat trace circuits were initially installed to support operation of the radwaste system evaporators. However, since the evaporators have not been used to process radioactive wastes, and there are no plans to use these evaporators for such purposes, the heat trace circuits are no longer required. Consequently, maintenance and use of various evaporator support equipment, such as the affected heat trace circuits, are no longer necessary. De-energizing these circuits will not adversely impact the ability to safely operate the nuclear plant, nor will the proposed change impact the ability to conduct a safe orderly plant shutdown. There are no safety related functions associated with the G17 and G18 Systems, the radwaste system evaporators, or the associated heat trace circuits. The heat trace circuits are independent of safety related electrical power distribution and instrument control systems. The changes performed will not defeat established separation criteria for safety and non-safety related circuits nor will it alter any interfaces with other equipment. Thus, the anticipated response of plant equipment during analyzed events, and the radiological consequences associated with these events, will not be impacted by the proposed change. Based on these conclusions, the proposed change does not represent an Unreviewed Safety Question. The affected heat trace circuits are not specifically addressed in the GGNS Technical Specification, nor in the Technical Requirements Manual, thus revisions to these documents will not be necessary as a result of the proposed change.

Serial Number: 97-038-NPE Document Evaluated: ER 96/0528-00-01 & SCN 97/006A To GGNS-MS-02

DESCRIPTION OF CHANGE: Plant Staff has reported an acid leak from the Sulfuric Acid Storage Tank (SP21A003A) outlet pipe (GGCR 1997-0453-00). A small pin hole has been detected at the weld. This outlet pipe was installed during the year of 1988. A REPAIR of the damaged section of pipe is necessary in order to restore the piping integrity. The scope of the REPAIR is to replace the entire section of outlet piping assembly between the tank and (SP21F042A) valve with an upgraded material (UNS N08020) for general corrosive service in lieu of carbon steel.

REASON FOR CHANGE: The sulfuric acid storage tank (SP21A003A) and supply are part of the makeup water treatment system (P21). The change will not affect parameters of the P21 system. The proposed change will enhance system reliability to prevent an acid leak without affecting of sulfuric acid supply system. The sulfuric acid storage tanks are located in a reinforced concrete dike to prevent uncontrolled release of acid to the ground. The system has no safety-related function as defined in UFSAR Section 3.2. The system is not required for safe shutdown of the plant.

SAFETY EVALUATION: The sulfuric acid storage tank (SP21A003) and supply system is a part of the makeup water treatment system (P21). The change will not affect parameters of the P21 system. The proposed change will enhance system reliability to prevent an acid leak without affecting operation of sulfuric acid supply system. The sulfuric acid storage tanks are located in a reinforced concrete dike to prevent uncontrolled release of acid to the ground. The system has no safety-related function as defined in UFSAR Section 3.2. UFSAR Table 3.2-1 classifies this system's components as non-safety related, non-seismic, quality group D, and ANSI B31.1. The system is not required for safe shutdown of the plant. The change will not affect design information provided in UFSAR Sections 9.2.3 and 10.2.5. The replacement piping assembly has been designed in accordance with original standards. The makeup water treatment system Containment and Auxiliary Building penetrations are addressed in the Technical Specifications, which are not affected by this change. The proposed change will not affect operating function of the sulfuric acid storage tanks supply or any safety related system nor will it impose any new requirements to the current Technical Specifications. The design has been evaluated against the applicable design criteria, installation, and operational requirements, and all necessary requirements and commitments are met. The change will not affect any equipment important to safety. The modifications made by this design change will not impose a change to the criteria listed in Table 3.2-1.

Serial Number: 97-039-NPE   Document Evaluated: N/A

DESCRIPTION OF CHANGE: This change evaluates increasing the maximum allowed stroke time of the RHR/LPCI test return valve Q1E12F024A&B by 54 seconds. The maximum allowed stroke time currently is 90 seconds.

REASON FOR CHANGE: NPE has reviewed/evaluated the operating capability or margin of each motor operated valve in the Generic Letter 89-10 MOV Program in an effort to identify valves which have low operating thrust and/or torque margins. Valves which have low operating margins will be modified and/or reset to achieve an acceptable operating thrust and torque margin. NPE identified the Q1E12F024A and Q1E12F024B valves as having low operating torque margins. NPE evaluated the valves/actuators and concluded that the only viable option to improve the operating torque margins for Q1E12F024A and Q1E12F024B is to replace the actuator gearing. This modification will increase the output torque capability of the actuator assembly; however, it will also reduce the output or operating speed of the actuator which will result in longer stroke times.

SAFETY EVALUATION: The proposed change increases the maximum allowed stroke time on the RHR test return valves, Q1E12F024A&B, by 54 seconds. The Technical Specifications are not affected by the change. The proposed change was determined not to impact automatic operation of the RHR system in response to design basis accidents and the proposed stroke time is consistent with other valves performing a similar isolation function for other ECCS applications (leakage into closed systems). Since no new initiators were created or affected and the consequences of accidents evaluated were not affected, the probability of occurrence or consequences of an accident evaluated in the SAR is not increased. SAR analyses that specify operation of the suppression pool cooling function of RHR are not affected by this change since the time required to establish this manually initiated mode of containment heat removal is only approximated in the analyses for these relatively slow progression events. Therefore, no new failure modes are created by this change and accidents or malfunctions of a different type are not created. The SAR credits the containment spray (CS) system actuation as a means of mitigating the effects of drywell bypass leakage for small primary system breaks. The CS is assumed to actuate 13 minutes into the event. This analysis forms the basis for the Technical Specifications surveillance requirement for testing drywell bypass leakage. The proposed change was determined to not impact the automatic performance of the CS mode of RHR as assumed in the analysis, thereby not reducing the margin of safety as defined in the basis for any Technical Specification. It was therefore concluded that this change does not create an unreviewed safety question.

Serial Number: 97-040-NPE Document Evaluated: ER 96/0425-00-00

DESCRIPTION OF CHANGE: Engineering Request (ER) 96/0425-00-00 modifies the ductwork and piping as necessary to install a backdraft damper in the turbine building exhaust ductwork to impede the airflow from the turbine building exhaust system through the mechanical vacuum pump discharge piping to the pumps. FSAR Figure 9.4-006 has been revised to show the addition of the backdraft damper.

REASON FOR CHANGE: Per ER 96/0425 the airflow in the turbine building exhaust system is approximately 120°F db and 115°F wb. Per ER 96/0425 the vacuum pumps are located in an area that is maintained at approximately 80°F. Therefore, condensation will form on the vacuum pumps. The addition of the backdraft damper will impede the flow of the exhaust air to the vacuum pumps and reduce the rate at which the condensation forms on the pumps.

SAFETY EVALUATION: As stated in Sections 9.4.4.3 and 10.4.2.3, neither the turbine building HVAC system nor the condenser air removal system provides a safety function. The modification of these systems as described in ER 96-0425 will not invalidate any assumptions contained in the SAR regarding system operation or failure.

Serial Number: 97-041-NSR Document Evaluated: 06-OP-1N32-V-0001

DESCRIPTION OF CHANGE: The Turbine Stop and Control Valve Operability surveillance is used to demonstrate the operability of the four high pressure turbine stop valves, four high pressure turbine control valves, six low pressure turbine stop valves, and six low pressure turbine control valves at least once per 14 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 14 days to 42 days.

REASON FOR CHANGE: Generator output must be less than 90% of rated load and reactor power less than 94% core thermal power (3603 MWt) to perform Attachment I of 06-OP-1N32-V-0001. This surveillance is presently on a bi-weekly schedule. This change will extend the allowed testing frequency to six weeks (maximum) and allows it to coincide with the Control Rod Operability Surveillance (06-OP-1C11-M-0001) which is currently on a four week schedule. This change should allow for improvement in unit capability factor, and will provide significant cost savings over the remaining life of the plant. The 31 day cycle plus the 25% grace period, if used, will not exceed the 42 day maximum time which Siemens Power has analyzed for the ATT frequency change. This change has been approved by Siemens Power Corporation and documentation supporting this transition from bi-weekly to every six weeks can be found in GEXI-95-00625 and GEXI-97-00102.

SAFETY EVALUATION: Siemens Power Corporation conducted a study to determine whether 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 an interval of 6 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. This information is based on operating and test data for Siemens units worldwide and uses probability analysis of this data to base its conclusions. GGNS will incorporate into existing procedures the methods and instruments needed to monitor for time-dependent defects. The activity will not be implemented until all the requested information, documentation, and procedures are in place to support the change.

UFSAR Chapter 15.2.3, Turbine Trip, was reviewed. There are no changes to the operation of the Automatic Turbine Tester that impact the findirg in this chapter. This Safety Evaluation addressed 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.

Serial Number: 97-042-NPE Document Evaluated: LDCR 97-0027 (ODCM Revision 21)

DESCRIPTION OF CHANGE: The evaluated LDCR will change the ODCM to clarify what instrumentation is required to be operable during a containment low volume purge. The proposed ODCM change will allow the flow rate instrumentation upstream of the containment ventilation exhaust fans to be used to measure flow rate during a low volume purge.

The FSAR and TRM will be updated to indicate that the low volume purge instrumentation (1M41R600) will be used to monitor containment vent discharge flow when a low volume purge is being performed.

REASON FOR CHANGE: The ODCM is presently interpreted to mean that the flow rate instrument installed in the twenty-inch duct immediately upstream of the exhaust penthouse is required to be operable during a low volume purge. However, the discharge flow rate during this mode is below the instrument's range. The exhaust fan flow meter is installed in six-inch duct work and is capable of measuring flows in the appropriate range, and during a low volume purge, the flow through the exhaust filter train is the same flow which is discharged through the penthouse.

SAFETY EVALUATION: Presently an LCO is required anytime a low volume purge is performed. During the LCO, exhaust flow rate is estimated. The proposed changed will eliminate the requirement for entering an LCO and will provide an acceptable method for measuring exhaust flow during a low volume purge.

The proposed change will allow the use of the installed containment low volume purge flow indication (1M41R600) in lieu of estimating the discharge flow. Per GGNS commitments to Reg. Guide 1.97, containment vent discharge flow is a Type E Category 3 variable. Reg. Guide 1.97 provides design requirements for instrumentation used to measure Type E Category 3 variable. The proposed instrumentation meets the design requirements of Reg. Guide 1.97 for a Type E Category 3 variable. The flow rate instrumentation is not required to mitigate the consequences of any accident or malfunction. It cannot create the possibility of an accident or malfunction of equipment important to safety, and it is not associated with any margin of safety. The proposed change does not result in an unreviewed safety question.

Serial Number: 97-043-NPE   Document Evaluated: ER 96/0360-00-00

DESCRIPTION OF CHANGE: The Computer Room Air Conditioning Subsystem operates continuously to maintain computer room relative humidity at 50% at 75°F. Humidifier SZ17D001 is installed on the common supply ductwork of the system fan coil unit to provide this function and is controlled automatically by a moisture switch. If the humidity level in the computer room falls below the set point, the system humidifier utilizes water to reestablish humidity level in the room. A new humidifier, model No. EHU-601 was installed to replace previously installed (and now obsolete) model EHU-410, both manufactured by Armstrong Machine Works. Although the new model is the most direct replacement for the previous unit, the water supply and drain line connections for the new model have a slight dimensional and location differences from the old humidifier. The existing domestic water supply and drain lines will be modified per this ER to suit the new model. Additionally, the new humidifier has a 1/2" drain on its dispersion tube that will be connected to the unit drain. The size of the existing drain line will be increased from currently 1/4" dia to 1" dia. as per vendor recommendation and to alleviate drain clogging that was experienced in the past.

REASON FOR CHANGE: A new humidifier, model No. EHU-601 was installed to replace previously installed model No. EHU-401. Because the new model water supply and drain line connections have a slight dimensional and location differences from the old humidifier, the existing domestic water supply and drain lines will be modified per this ER to fit the new humidifier. The existing drain line size will be increase from currently 1/4" dia. to 1" dia. as per vendor recommendation and to prevent pipe clogging. Also, the new humidifier has a 1/2" drain on its dispersion tube that will be connected to the unit's new 1" drain.

SAFETY EVALUATION: The Computer Room Humidifier and The Control Building Ventilation System serve no safety function and as identified in the UFSAR Sections 9.4.10.2.4 & 9.4.10.3 are Non-Safety Related. The Domestic water supply piping modification affects only a small portion(2 feet) of the existing piping that was slightly rerouted but remains adequately supported. As described in the UFSAR section 9.2.4.3, the domestic water system has no safety related function and failure of the system will not compromise and safety-related equipment or component and will not prevent safe shut down of the plant. The modified drain piping is designed and supported non-safety related and seismic category II/I. The humidifier specific design features, domestic water supply piping and control building ventilation system are not addressed by the GGNS Tech Specification. No change in the operation or function of humidifier HVAC, Domestic Water or Floor drain systems will be created by this change and No change to GGNS Technical Specification is required.

Serial Number: 97-044-NPE   Document Evaluated: ER 97-0300-00-00

DESCRIPTION OF CHANGE: The Standby Service Water (SSW) side of Q1T51B001, Q1T51B002, Q1T51B003, Q1T51B004, Q1T51B005, Q1T51B006, Q1T51B007A, Q1T51B007B (ECCS pump room coolers) and Associated P41 Piping will be chemically cleaned. The chemical solution to be used in the cleaning process is Betz DE-1178. The chemical composition of this solution is 40% Citric Acid, 10% Phosphonate, 10% corrosion inhibitor and 40% inert ingredients. The system boundaries established for cleaning of the Q1T51B001, Q1T51B002, Q1T51B003, Q1T51B004, Q1T51B005, Q1T51B006, Q1T51B007A, Q1T51B007B (ECCS pump room coolers) and Associated P41 Piping are between valves shown below in Table 1:

Table 1: ECCS Room Coolers To Be Flushed

Emergency Pump	Room Cooler	Isolation Valve Inlet	Isolation Valve Outlet	P&ID
HPCS	1T51B001	1P41F054	1P41F060	M-1061B
LPCS	1T51B002	1P41F038	1P41F037	M-1061B
RHR "A"	1T51B003	1P41F102A	1P41F103A	M-1061C
RHR "B"	1T51B004	1P41F102B	1P41F103B	M-1061D
RHR "C"	1T51B005	1P41F048	1P41F047	M-1061B
RCIC	1T51B006	1P41F105	1P41F106	M-1061C
FPC&CU	1T51B007A	1P41F292A	1P41F296A	M-1061B
FPC&CU	1T51B007B	1P41F292B	1P41F296B	M-1061B

The materials of construction between the established boundaries that will be exposed (wetted) by the selected cleaning process have been identified and evaluated for compatibility with the specified chemical solution.

REASON FOR CHANGE: The coils/tubes (I.D) within the room cooler will be chemically cleaned to restore the heat transfer capability of the unit.

SAFETY EVALUATION: A chemical cleaning process using Betz DE-1178 has been selected to improve the cleanliness of the room cooler coils/tubes (I.D) which will enhance the heat transfer capability of the units. The materials of construction between the established boundaries of the affected P41 and T51 systems have been reviewed and it has been concluded that the design margins on the affected P41 and T51 systems will not be adversely impacted by the selected chemical cleaning process using the Betz DE-1178 chemical solution or by performance of the selected chemical cleaning process.

Based on the referenced documents, the affected materials are compatible with the selected chemical cleaning process and specified chemical solution.

Reference 1 documents testing conducted to determine corrosion rates of Copper/Nickel (90% Cu-10% Ni) materials. this corrosion rate was shown to be conservative by testing documented in Reference 2. NPE conducted an evaluation of the heat exchanger data from 9645-M-611.0, R/14 and Engineering Report SERI-88-0006, Rev. 0. Based on this evaluation, NPE concluded that the material loss from the cooling coil tubes and headers due to 10 cleanings with the proposed solution and process was acceptable.

Further, Reference 1 also documents testing conducted to determine the general corrosion rates of carbon steels due to exposure to the proposed cleaning solution. NPE evaluated the average metal loss for carbon steel from general corrosion due to exposure to the proposed cleaning solution using this data and the methodology presented in ASME Boiler & Pressure Vessel Code Section ND-3641.1 and determined the material loss due to 10 cleanings was acceptable. Based on this evaluation, NPE concluded that a 4 hour cleaning could be performed 10 times and still not reduce the pipe wall thickness below the design minimum wall thickness plus the design corrosion allowance. Additionally, the impact of the wall thinning due to the above cleanings was evaluated against the piping stress analysis. Based on this evaluation, NPE concluded that the wall thinning due to a 4 hour cleaning could be performed 10 times and still not adversely impact the piping stress analysis for the affected P41 system piping. Finally, wall thickness baseline readings of the room cooler piping was taken and compared against acceptance criteria to shown the acid flushes were acceptable.

Considering all the affected materials, their associated design requirements, and the data presented by References 1, 2, and 3, the established boundary can be safely cleaned by the selected chemical process with the specified chemical solution suing the established process controls provided in the response to Engineering Request 97/0300-00-00.

The selected chemical cleaning process using Betz DE-1178 chemical solution on the materials of construction between the established boundaries will not adversely impact the design margins of the affected P41 or T51 systems provided the established process controls are followed. Additionally, two locations which are representative of all the room cooler piping will be added to QAP 9.90 to monitor the wall thickness periodically to ensure the piping wall thickness do not become less than the minimum wall thickness based on the ASME code allowables; hence, the margin of safety in accordance with the design requirements and functional capabilities of the systems are maintained and assured. Therefore, it is concluded that the use of the selected cleaning process as described in this evaluation does not increase the probability or the consequences of any accident evaluated in the SAR, does not create the possibility of a new accident or malfunction, and does not reduce any margin of safety defined in the basis for any technical specification. There are no unreviewed safety questions or issues resulting from using the selected chemical cleaning process or specified chemical solution as discussed in this evaluation.

Serial Number: 97-045-NPE Document Evaluated: GGNS-MS-48.0, Revision 5 (COLR)

DESCRIPTION OF CHANGE: This safety evaluation assesses changes made to the Cycle 9 Core Operating Limits Report (COLR), GGNS Engineering Standard GGNS-MS-48.0. These changes incorporate the GE11 rated LHGR limits, that are in the core monitoring system, into the COLR. In addition, power and flow dependent off-rated multipliers will be applied to the rated GE11 LHGR limits in the same manner that the off-rated limits are currently being applied to the GE11 MAPLHGR limits. These changes are being made as part of the corrective action for Condition Report GGCR 1997-0074-00 which identified that when Gradient Local Peaking (GLP) is applied to the high power node in a fuel bundle, the thermal margins are reduced. Including LHGR limits for GE11 in the COLR increases control of the information, established a design basis for the LHGR limits monitored by the process computer, and establishes a GGNS position that LHGR should be monitored during rated and off-rated conditions in addition to the APLHGR limits. This Safety Evaluation addresses changes to the COLR and is intended to augment SE 96-0115-R00 which describes in detail how the GGNS Cycle 9 core operating limits were calculated.

REASON FOR CHANGE: The Cycle 9 COLR is being revised to incorporate the rated and off-rated LHGR limits for GE11 as part of the corrective action for GGCR 1997-0074-00.

SAFETY EVALUATION: This evaluation concludes that the COLR changes (i) will require no additional changes to the current GGNS Technical Specifications, and (ii) will not constitute an unresolved safety question. Revision 5 of the Cycle 9 COLR has been shown to meet all requirements in the GGNS Technical Specifications, GGNS UFSAR, 10CFR, and the Standard Review Plan.

Serial Number: 97-046-NPE   Document Evaluated: ER 96/1014-00 R0

DESCRIPTION OF CHANGE: Replace the existing relief valves P81F050A/B (Division III starting air compressor relief valves) with Farris model number 1896M-OL relief valves. Modify the existing 1-1/4 inch relief valve discharge tail pipes for P81F050A/B by installing a section 3/4 inch pipe to mate with the relief ports on the new relief valves.

REASON FOR CHANGE: This change replaces the existing relief valves on both the motor driven and diesel driven starting air compressors installed in the HPCS Diesel Generator starting air subsystem. The P81F050A/B relief valves currently installed on the Division III starting air compressors are Farris model number 1855-OL valves which have a maximum set pressure of 250 psig. The design set pressure for relief valves P81F050A/B is 275 psig. The maximum set pressure for the Farris model number 1896M-OL relief valve is 300 psig. UFSAR Figure 9.5-016 lists the P81F050A/B relief valves installed on the Division III starting air compressors as Farris model number 1875-OL. Both Farris model numbers 1855-OL and 1875-OL are obsolete. The Vendor has recommended the Farris Model number 1896M-OL relief valve as a replacement for the original 1875-OL relief valve.

SAFETY EVALUATION: Replacing the existing P81F050A/B relief valves on the Division III starting air compressors with Farris model number 1896M-OL relief valves and adapting the relief valve discharge tail pipe to the 3/4 inch relief valve discharge port will not reduce the reliability of the HPCS DG air start system. The Farris model number 1896M-Ol relief valve has sufficient relieving capacity and maximum set pressure suitable for this application. This change will ensure that the operating pressure of the Division III starting air compressors do not exceed the design pressure rating of the protected components and that suitable replacement valves are available when required.

Serial Number: 97-047-NPE   Document Evaluated: ER 97/0209-00-00, GGCR1997-0205-00

DESCRIPTION OF CHANGE: The Spent Resin Tank (NSG17A007) was exposed to an overpressure condition which resulted in exceeding the allowable stresses for the tank materials. Consequently, the tank shell has yielded (deformed) and minor tank support damage has occurred. The change proposed by ER 97/0209-00-00 involves the repair of damaged welds on the Spent Resin Tank supports and returning the Spent Resin Tank to normal service conditions.

REASON FOR CHANGE: The Spent Resin Tank (SRT) experienced physical damage during performance of a hydrostatic pressure test. While repairs will be implemented on the SRT support welds per ER 97/0209-00-00, the tank's allowable stresses have been exceeded and the tank shell has yielded, resulting in permanent strain and deformation of the tank shell. The proposed change is being implemented to justify return of this tank to normal service and resume use of this tank. It should be noted that the function or use of the Spent Resin Tank is not altered by the proposed change, but the UFSAR will be updated to incorporate the new conditions.

SAFETY EVALUATION: This safety evaluation has concluded that although the Spent Resin Tank (SRT) has been deformed as a result of an overpressure condition, the tank has been inspected and is adequate to perform its intended function. Resuming use of the SRT in its current condition has been evaluated and does not represent an Unreviewed Safety Question or an Unreviewed Environmental Question. The SRT is a component installed in the Liquid Radwaste (G17) System and does not serve or provide any safety functions. The SRT is located in the Radwaste Building and as such, does not impact or interface with safety related systems, structures, or components. As such, the proposed change will not adversely affect plant or equipment response to normal or abnormal operating conditions, nor will the proposed change affect radiological or non-radiological effluent releases occurring as a result of plant operation. Resuming of the SRT in its present condition will not increase the probability or consequences of accidents previously analyzed in the UFSAR, nor will the proposed change introduce an accident or a nature different than those accidents previously analyzed in the UFSAR. The SRT is not addressed in the Technical Specifications (or TRM) and the SRT is not a factor used in the Margins of Safety as defined in the basis for any of the Technical Specifications. Thus, the proposed change will not result in a conflict with information currently contained in the Technical Specifications nor will the proposed change result in a need to revise the contents of the Technical Specifications (or TRM).

Serial Number: 97-048-ECH Document Evaluated: TS SR 3.5.1.7 & SR 3.5.2.6

DESCRIPTION OF CHANGE: This change eliminates the specific response time testing of the actuation instrumentation for the High Pressure Core Spray System.

The need to perform response time testing of the other ECCS actuation instrumentation was reviewed and deleted by the NRC in Technical Specification Amendment 20, dated October 6, 1986 (MAEC-86/0330). In the Safety Evaluation Report for this Technical Specification change the NRC concluded that the testing of the DG start time and testing of the actuated equipment response time resulted in required testing "not significantly different" from the actuation instrumentation response time testing requirements. Following this change, the response time of the HPCS System will continued to be tested by testing the DG start time and the actuated equipment response time. These times will be verified to be less than 27 seconds, thereby, ensuring the assumptions of the accident analyses are met.

REASON FOR CHANGE: Deleted unnecessary testing and the associated personnel burden.

SAFETY EVALUATION: The accident analyses assumes a 27 second time for the starting of the HPCS system to allow the associated DG to start and supply AC power, the pump to start, and system valve movement. The response time of the actuation instrumentation is very small when compared to the 27 second HPCS start time, therefore, the response time of the HPCS actuation instrumentation is not a critical parameter in the ability of the HPCS System to perform its design function.

A detailed analysis supporting elimination of this instrument response time testing requirements and the requirement for response time testing of other components is documented in the BWR Owners Group licensing topical report "NEDO-32291". As discussed in the NRC's SER for NEDO-32291, the actuation system response time for the HPCS System is much shorter than the total system response time and as a result the actual instrument response time is unimportant in meeting the system response time. In addition, the instrumentation components that may experience response time degradation will continue to respond in the millisecond range until failure.

Since the response time of these instruments is masked by the system start time (27 seconds), the remaining required surveillances for this instrumentation (e.g., CHANNEL CHECKs, CHANNEL FUNCTIONAL TESTs, and CHANNEL CALIBRATIONs) provide adequate assurance that the instrument response time has not degraded to a point that the assumed system response time is affected without declaring the system inoperable. These surveillances ensure that any instrument degradation would be identified prior to affecting system performance and, therefore, would have no adverse affect on system actuation and the system's ability to perform its safety function.

Serial Number: 97-049-NPE Document Evaluated: MCP 91/1052

DESCRIPTION OF CHANGE: At present, humidity switches monitor the relative humidity of the air entering the Control Room Standby Fresh Air Units. If the relative humidity of this air is in excess of 70%, these humidity switches provide inputs to energize the Standby Fresh Air Unit heater assemblies. when energized, these heaters reduce the relative humidity of the incoming air in order to protect the downstream filtering elements from degradation due to moisture buildup.

MCP 91/1052 will abandon in place the Fresh Air Unit humidity switches, delete the control room annunciation associated with these switches, and modify the unit heater control logic to provide for automatic energization of the heater assemblies whenever: the unit fans are running, normal flow has been established through the system, and the heater control handswitches are in their normal 'standby' position. Control room annunciation will be provided to alert operations personnel whenever the heater control handswitches are in the 'off' position to prevent inadvertent system operation with the heaters bypassed. The change will also abandon in place the humidifiers, and associated humidity switches, of the Control Room AC System.

REASON FOR CHANGE: The Standby Fresh Air Unit humidity switches to be removed from service are obsolete. Spares are no longer available from the original manufacturer.

The Control Room HVAC humidifiers have proven to be a recurring maintenance problem due to the poor quality of their domestic water supply. (Ref. MNCR 0037-94)

SAFETY EVALUATION: The modifications to be performed on the Control Room Fresh Air and Control Room AC systems will not create new release mechanisms or adversely impact radionuclide population, release rates, release duration or release barriers. No new interfaces will be created with safety systems as a result of the modifications to be performed, and the modifications are in accordance with the requirements of Reg. Guide 1.75. The integrity of the control room envelope will not be adversely affected as a result of this change. The proposed modifications to the Control Room Fresh Air system will not adversely affect the systems ability to perform its safety function as described in the Z51 System Design Criteria (i.e., to ensure the radiation exposure of control room personnel does not exceed the limits set by GDC 19 of 10 CFR 50, Appendix A).

Compensatory measures will be implemented, as required, to maintain control room relative humidity above the minimum recommended by NUREG-0700 (20% RH) to ensure a suitable environment for equipment operation and the control room operators. The proposed modifications will not increase the probability of occurrence or the consequences of an accident or malfunction of equipment important to safety previously evaluated in the SAR. Further, the proposed modifications do not create the possibility of an accident or malfunction of equipment of a different type than any evaluated previously in the SAR.

Serial Number: 97-050-Chem Document Evaluated: LDCR 97-027 (ODCM Rev. 21)

DESCRIPTION OF CHANGE: The evaluated LDCR 97-027 changes the ODCM to:

1. Modify the methodology used to determine conservative effluent gaseous radiation monitor setpoints,

2. Remove an unused table of dose factors,

3. Remove information related to liquid radwaste treatment components no longer used (reverse osmosis equipment),

4. Re-instates the radiological environmental monitoring program (REMP) fish sampling requirement at an annual frequency (prior to ODCM Revision 20, fish samples were collected on a semi-annual frequency).

5. Modify the methodology used to determine conservative effluent gaseous radiation monitor setpoints,

2. Remove an unused table of dose factors,

3. Remove information related to liquid radwaste treatment components no longer used (reverse osmosis equipment),

4. Re-instates the radiological environmental monitoring program (REMP) fish sampling requirement at an annual frequency (prior to ODCM Revision 20, fish samples were collected on a semi-annual frequency).

5. Make editorial changes (including: introduction, table of contents, corrects an example graph for an instrument response curve.)

Note: ODCM Revision 21 will also implement use of an alternate effluent flow rate measuring device for containment building low volume purge. This change was previously evaluated and approved in SE 97-002-R02.

REASON FOR CHANGE: The proposed changes are intended to improve gaseous effluent radiation monitor setpoint methodology, remove unused information from the ODCM and re-instate a sample to the radiological environmental monitoring program.

1. Modify the methodology used to determine conservative effluent gaseous radiation monitor setpoints:

Gaseous effluent radiation monitor setpoints function to provide indication to station operators of increasing release rates of noble gases, prior to exceeding the site release limit at the site boundary or at unrestricted areas within the site boundary of <3000 mrem/year skin and <500 mrem/year total body. The ODCM presently contains two methods to calculate these setpoints. The proposed change will only affect the method used in the absence of measured radioactivity, hereafter called the "conservative" method. Setpoint methodology used when noble gases are measured in grab samples is not affected by the proposed change.

2. Remove an unused table of dose factors:

The removal of a table of unused dose factors eliminates unnecessary information from the ODCM. The dose factor table is for dose rates for the child age group, P_I. The age group utilized in the dose rate calculation is the most restrictive age group (infant age group per NUREG 0133, Section 5.2.1).

3. Remove information related to liquid radwaste treatment components no longer used (Reverse osmosis equipment)

The removal of certain liquid radwaste treatment components from the test and associated diagram eliminates unnecessary information from the ODCM. The liquid radwaste treatment reverse osmosis (RO) equipment has been removed from the plant. Liquid radwaste treatment will be in accordance with existing permanent plant equipment.

4. Re-instate the radiological environmental monitoring program (REMP) fish sampling requirement at an annual frequency.

Prior to ODCM Revision 20, fish samples were collected at a semi-annual frequency. ODCM Rev. 20 (July 1996) eliminated a number of samples from the REMP, including a semi-annual fish sample from the Mississippi River. During NRC Inspection 97-10 (June 1997) the elimination of fish sampling from the REMP was identified as being inconsistent with the use of the fish ingestion pathway in the ODCM liquid dose model. Although GGNS historical data from plant operation to present has not detected any radioactivity in downstream fish samples and evidence for a resident fish population is inconclusive, the decision was made to reinstate the fist sample to ensure consistency between ODCM pathways and REMP sample media.

5. Make editorial changes (Introduction, Table of Contents, etc.)

These changes are editorial in nature and do not affect information contained in the UFSAR, Technical Requirements Manual or Technical Specifications. Figure 1.0-1 is an example instrument response curve which has a gridlines reversed on the X-axis. The figure will be corrected. The figure is not used for any calculations associated with doses or setpoints.

SAFETY EVALUATION:

1. Gaseous effluent radiation monitor setpoints function to provide indication to station operators of increasing release rates of

noble gases, prior to exceeding the site release limit at the site boundary or at unrestricted areas within the site boundary of <3000 mrem/year skin and <500 mrem/year total body. Presently, conservative gaseous effluent setpoints are calculated using a default isotope which has a three minute half life.

Although this fission gas is produced in the reactor, it has not been detected in GGNS effluents since power operation commenced. UFSAR Table 11.3-9 shows an annual release of zero for Krypton-89. Replacement of the default isotope with a historical mixture based on actual GGNS releases will provide more realistic setpoints.

2. Elimination of unused information from the ODCM does not affect the accuracy of the effluent dose or effluent radiation monitor setpoint calculations.

3. Following removal of the reverse osmosis equipment, processing of liquid radwaste continued using existing permanent plant equipment. the change to remove the reverse osmosis equipment from the ODCM text and LRW treatment system diagram reflects current system configuration and does not deviate from the processing of liquid waste as described in the UFSAR.

4. Re-instating the fish sample requirement in the REMP provides consistency between ODCM ingestion pathways and REMP sample media. Reduction from semi-annual to annual frequency is within the allowances and guidance of NRC Branch Technical Position, Environmental Technical Specifications for Nuclear Power Plants, Revision 1, November 1979 and Regulatory Guide 4.8, Environmental Technical Specifications for Nuclear Power Plants, December 1975. The reduction in frequency is based on data collected over a eleven year period of plant operation showing no positive results for plant generated radioactivity in fish and the fact the fish population in the Mississippi River is not physically confined to the location of the plant discharge. While evidence is given for the potential for resident fish population in the main river channel there is also indication of periodic displacement of the fish. UFSAR Section 2.2.3.2.1 states "during extensive flooding, such as that recorded in the spring of 1973, fish are often displaced from their normal habitat", Section 2.2.3.2.2.A.2 states "... a possible resident in the channel during these two months" (August/September). UFSAR Section 2.2.3.2.2.A.3 states that "...fish abundance in river bank habitats also appeared to fluctuate seasonally. " These statements do not conclusively define the fish population as transitory or resident. The fish sample will be re-instated at a reduced frequency to provide continuing evidence of the lack of radiological impact via the ingestion pathway.

Serial Number: 97-051-NPE Document Evaluated: ER 96-0575

DESCRIPTION OF CHANGE: This 50.59 evaluation applies only to the core spray portion of the Engineering Request (ER).

USAR Section 3.1.2.4.7 states that the core spray spargers within the vessel will be inspected each refueling outage using a remote underwater television camera. The change eliminates the specific reference to inspection frequency and methods for performing these inspections. Alternatively, this section is revised to indicate that inspection of the core spray sparger will be performed in accordance with Program Plan GGNS-M-489.7, "Vessel Internals Management Program". As part of this change, the inspections originally scheduled for RF-08 are being deferred until RF-09.

REASON FOR CHANGE: This change will permit the core spray spargers to be inspected with the core spray piping in accordance with the plant specific programs that are prepared and maintained for consistency with current industry practices. As the BWR fleet gains experience with the degradation of RPV internals and with the increase in inspection technology, frequencies and methods currently contained in the UFSAR may be inappropriate.

When the requirement to visually inspect the spargers was added to the SAR, visual inspection was the technique preferred by the industry at that time. However, based on the work of the BWRVIP, visual inspection may no longer be the method of choice for core spray sparger inspections. Technological advancements of the BWRVIP is producing new methods such as ultrasonic, eddy current and enhanced visual examination techniques that are being adapted specifically to the reactor vessel internals. The GGNS vessel internals management program will appropriately incorporate these enhanced inspection methods in future outages to ensure continued integrity of the core spray piping and sparger assembly.

Inspections during RFO8 have been deferred based on an engineered review that compares the GGNS core spray system to those of other facilities which have reported cracking. Because of the significant differences that exist in the attributes known to promote cracking, adequate basis exist to support deferral of inspections to RFO9 when enhanced techniques will be available. Inspections to be performed during RFO9 may consist of ultrasonic examinations that will interrogate the full volume of each selected location or enhanced visuals with the capability of detecting a 1/2 mil wire. The visual inspection only examines the outside surface of the selected welds, therefore, for this technique inspection frequencies will be more frequent than those required if ultrasonic examinations are performed.

SAFETY EVALUATION: This safety evaluation provides a bases for removing the specific inspection frequency and method requirements for the core spray spargers from the UFSAR and deferring inspections from RFO8 until

RFO9. Inspections of the core spray spargers will be performed at a frequency and using examination techniques listed in the GGNS vessel internals management program. This program is prepared and maintained with consideration given to BWRVIP recommendations and industry guidance (SILs, RICSILs, operating experience, etc...) for the inspection of the reactor vessel and its internals. Inspection frequencies and techniques specified in the vessel internals management program will assure that the integrity of the core spray spargers are maintained sufficiently to ensure their ability to provide their intended safety function. The requirement to perform visual inspections of the core spray spargers each outage was based on technologies available when the SAR was written. Emerging technologies are providing more effective and repeatable techniques for inspection of BWR vessel internals.

Serial Number: 97-052-NPE   Document Evaluated: LDCR 97-072

DESCRIPTION OF CHANGE: Revise the discussion of the underground fire protection yard loop in UFSAR Section 9.5.1 to clarify the fact that not all portions of the yard loop are located underground.

REASON FOR CHANGE: Condition Report GGCR1997-0184-00 identified the discrepancy between the wording in UFSAR Section 9.5.1.2.1 regarding the underground fire protection yard loop and the installed plant configuration. Specifically, Section 9.5.1.2.1 of the UFSAR states: "The fire protection system consists of an underground yard loop... ". Contrary to this statement, the installed plant configuration consists of a looped fire water main, part of which is run above ground. The northeast corner of this looped fire water main is above ground piping which runs through the Unit 2 Turbine Building. The installed looped fire main configuration is accurately shown on UFSAR Figure 9.5-008B and Design Drawings C-0035B and M-0147B. Therefore, the wording "underground yard loop" is not totally accurate and needs to be clarified.

SAFETY EVALUATION: The condition addressed by this UFSAR change is a wording clarification only and does not involve plant changes. The looped fire main as presently installed is accurately shown on UFSAR Figure 9.5-008B and applicable design drawings. Therefore, the existing fire main configuration is installed in accordance with existing plant design documents and the UFSAR. Therefore, this change will not increase the probability or consequences of accidents or malfunction of equipment important to safety previously evaluated in the SAR. Also, this change will not create the possibility for an accident or malfunction of equipment important to safety of a different type than any previously evaluated in the SAR. "Fire Suppression Water System" is not addressed by Technical Specifications (TS). "Fire Suppression Water System" is addressed in the Technical Requirements Manual (TRM) Section 6.2.2; however, the changes being made are wording clarification changes only and do not affect in any way the existing TRM Section 6.2.2 or the bases for this section.

Serial Number: 97-053-NPE Document Evaluated: ER 96/0984, R0 & SCNs 97/0005A to MS-02, 97/0001A to M-195-0

DESCRIPTION OF CHANGE: This modification installs the GE Passive Zinc Injection System as a subsystem of the Feedwater System. The passive GE Zinc Injection Passivation System is designed to continuously inject a dilute solution of ionic zinc in water into the reactor feedwater. The injected zinc ions reduce the corrosion film buildup on piping surfaces in the primary system, which will lower the radiation levels due to cobalt-60 deposition.

REASON FOR CHANGE: Hydrogen Water Chemistry (HWC) has been proven to reduce the risk of reactor vessel Inter-Granular Stress Corrosion Cracking (IGSCC), however, it cause increased radiation dose rates. When HWC is implemented at GGNS, some method of dose rate control is needed to limit the increased dose rates. The injection of depleted zinc into the primary system provides a dose reduction benefit from its ability to reduce the corrosion film buildup on piping. Hence, Zinc injection is to be implemented per this design change to off-set the dose rates incurred as a result of HWC.

SAFETY EVALUATION: This design change will require interface with the Condensate and Refueling Water Storage and Transfer (CRWST) system for the purpose of providing flush water to the zinc skid, and the Instrument Air system to provide an air supply to automatic isolation equipment upstream of the skid. The affected portions of the CRWST, Feedwater, Instrument Air Systems are non-safety related and are not postulated as initiators of accidents described in the FSAR. Failure of these systems will not compromise any safety related equipment or prevent safe reactor shutdown. The installation of the zinc injection sub-system and the associated piping and instrumentation will not affect the function or performance of the CRWST, Feedwater, and Instrument Air Systems.

The location of the zinc injection skid and piping in the Turbine Building were reviewed considering High Energy Line Breaks (HELBs) and the affects on Trip Critical and Trip Sensitive systems. The associated zinc skid modification and consideration of HELBs will have no impact on the probabilities of the accidents previously evaluated in the UFSAR. Frequency classifications reported in UFSAR Chapter 15 are not affected. No unusual failure modes of increased failure frequencies have been identified for this determination of the zinc injection skid design. On these bases, installation of the proposed zinc injection skid will not increase the probability of occurrence or consequences of an accident previously evaluated in the UFSAR.

The zinc injection design was reviewed against current accident analysis with regard to current feedwater check valve testing criteria. ER 96/0984 provides for isolating the zinc injection system and maintaining the piping pressure boundary with feedwater system depressurization.

With the incorporation of these features, the accident analysis for the feedwater check valve testing criteria is maintained.

The affected portions of the CRWST, Feedwater, and Instrument Air Systems are non-safety related and are not postulated as initiators of accidents described in the FSAR. In addition, the zinc skid and the piping, structural, and electrical modifications are designed in accordance with the applicable design codes and requirements. Therefore, there will be no adverse impact on the feedwater system or any system used to mitigate the consequences of an accident as a result of the new piping and skid. The systems will continue to function in their intended manner.

All applicable system design requirements are maintained by this modification, equipment considered important to safety is not affected, and no system or component will be operated outside of design parameters. On these bases, implementation of the proposed zinc injection skid will not increase the probability of or consequences of a malfunction of equipment important to safety previously evaluated in the UFSAR.

This modification will not cause an interaction between the affected systems with any safety-related system or component. Likewise, the malfunction of any such system or component will not be affected by these modifications, and no assumptions utilized in evaluating the consequences of a malfunction of equipment important to safety will be altered. The change will not degrade or prevent any actions required to mitigate the consequences resulting from a malfunction of equipment important to safety as previously evaluated in the UFSAR. Thus, there are no increases in the consequences of a malfunction of safety related equipment previously evaluated in the UFSAR associated with this design change.

The described changes will not alter the design, function, or operation of any equipment important to safety. Implementation of this change will not compromise any safety-related system or prevent safe shutdown since no new interface with equipment important to safety is created nor is such equipment prevented from operating as designed. Therefore, this change will not create the possibility for an accident or malfunction of safety related of a different type than any previously evaluated in the UFSAR.

The CRWST, Feedwater, and Instrument Air systems are not addressed in the Technical Specifications. The addition of the zinc injection skid and the associated piping and valves will not change the function or operation of these systems. This modification serves to off-set the high dose rates resulting from implementation of HWC. It does not change or add limiting conditions for operation, applicability, actions or surveillance requirements to the Technical Specifications. Therefore, the margin of safety as defined in the basis for any technical specification will not be reduced, and Technical Specification changes are required.

Serial Number: 97-054-NPE   Document Evaluated: ER 96/0096-00-00

DESCRIPTION OF CHANGE: Provide details for installation of Local Instrument Rack 1H22-P534 to be located on Turbine Building Elevation 166'0" in Area 5. This rack will be dedicated to Condenser Pressure Transmitters 1N19TN005A, B, and C, and associated Pressure Switches 1N19SHN006A, B and C. Vacuum Transmitter 1N19PTN036 will also be installed on this rack. These devices will be relocated from lower elevations of the Turbine Building, 113'0" and 133'0", and will utilize existing tubing runs from higher elevation condenser taps to obtain the desired signal. Current tubing configurations and locations will be maintained as test points for use by Engineering Support. This modification will require opening and closing of penetrations throut 3-hour fire barriers and 3 psi pressure boundaries in accordance with TRM Section 6.2.8, UFSAR Sections 3.8.4.1.1.5 and 9.5.1.2.2.9, and UFSAR Appendix 16A, Section 6.2.8, requirements.

REASON FOR CHANGE: Investigation and follow-up testing after several GGNS scrams identified that N19 pressure and vacuum indications more closely matched the N31 Turbine Trip Logic (HLVTD) alarm/trip points if condenser vacuum changes were steady. However, when pressure transients were experienced the N19 points reacted slower and indicated higher vacuums than the N31 points. These differences in readings have been attributed to condenser design and specific locations of instrument line taps. Control room indication for main condenser vacuum is sensed from a different tap location (approximate Elevation 106') in the main condenser than the pressures sensed by HLVTD (between Elevations 145' and 149'). Differences in these indications have been as much as 4" Hg vacuum. This condition has led to confusion to actual system status, including Control Room indication, annunciation and HLVTD alarms and trips out of setpoint sequence. This evaluation has been identified as a Scram Frequency Reduction issue and has been documented as LCTS Item 32309.

SAFETY EVALUATION: System safety analysis has shown that failure of the Condensate System will not compromise any Safety Related System nor will it prevent a safe shutdown of GGNS. The Condensate System serves no safety-related function and related equipment has been evaluated by the Graded QA Program as Low Safety Significant. The modifications required to alleviate inaccurate instrument indication for the system will not degrade any system or its related performance. Supports for new conduits and the local rack do not create any seismic or seismic II/I concerns. Opening and closing of penetrations for installation of new cabling will not affect the penetration's ability to perform as previously evaluated. Appropriate penetration seal design requirements described in UFSAR Sections 3.8.4.1.1.5 and 9.5.1.2.2.9 have been maintained and seal details ensure penetration integrity. Operational considerations have been provided for the affected penetrations to comply with Technical Specification and UFSAR requirements. This modification will improve operator response to plant conditions related to condenser pressures and vacuum. Relocation of the pressure and vacuum transmitters and switches will be reflected by the revision of UFSAR Figure 10.4-010.

Serial Number: 97-055-PSE Document Evaluated: WO 194183

DESCRIPTION OF CHANGE: This safety evaluation is related to Control Blade Replacement and addresses:

1. Remove miscellaneous radioactive hardware waste items and spent incore neutron monitor chambers from the spent fuel pool (SFP) work table and package these items in an approved shipping cask liner;

2. Set up volume reduction equipment necessary to prepare spent control blades for packaging in the shipping cask liner. This includes the Control Rod Blade Roller Bearing Punch (CRP) and the Underwater Shear Compactor (USC);

3. Process the spent control blades by removing the velocity limiters and roller bearings (stored at GGNS for later disposal) and cut the remaining portion of the control blades into approximately 10 inch segments for packing in the cask liner;

4. Prepare and load the liner into an approved shipping cask and transport the package offsite in accordance with applicable shipping requirements.

Items 1 through 4 will be repeated numerous times throughout plant life and decommissioning as necessary to dispose of all spent control blades, plus miscellaneous hardware items and used neutron detectors.

REASON FOR CHANGE: Some GE Duralife-100 control blades at GGNS are now beginning to reach the end of useable life and must be replaced. During RF08, eight blades were replaced. These are stored in the SFP control blade storage racks, which can house a maximum of 45 spent blades, although 8 of the locations are being used to store failed fuel rods and other items. This currently leaves only 29 available storage locations. During RF09, 34 control blades will be replaced, with 36 more scheduled for RF010. This replacement is slightly accelerated above the minimum required by predictions to help alleviate elevated coolant boron/tritium levels which are suspected to be at least partially due to leaching of boron from small cracks in older control blades. Periodic replacement of blades is then scheduled for various outages through the remainder of plant life.

Thus, it is necessary to process and ship spent control blades on a regular basis in order to maintain adequate storage space. Other miscellaneous radioactive items stored in the pools are also being shipped to allow movement of the SFP worktable for easier access to needed fuel storage locations, as well as to generally improve the cleanliness of the pools, lessen areas dose rates, and reduce the resource burden of on-site storage.

SAFETY EVALUATION: The activities proposed do not involve new or unique operations other than the processing of spent control blades. Blade processing has been conducted at numerous BWR sites for more than 10 years and the equipment involved has a proven track record. Events which have been analyzed are no more likely to occur and no new types of accidents or events are being introduced. Fuel pool systems will not be affected, except that a slightly different cooling flow path may be used as allowed by plant procedures to help minimize the potential for contamination of system components. "Heavy loads" program (NUREG-0612) issues will continue to be met and all loads will be within the capacity of lifting equipment to be used. Some significant contamination control issues are involved, but these are addressed by standard ALARA and RWP review requirements. No environmental issues beyond those already considered are introduced. Therefore, the processing and shipping of spent control blades, used neuron detectors, and miscellaneous radwaste hardware proposed in this project do not constitute an unreviewed safety or environmental question.

Serial Number: 97-056-NPE Document Evaluated: MCP 91/1032, R0

DESCRIPTION OF CHANGE: The old Bailey 771 analog recorders 1G33R601, 1G33R603 and 1G33R611 and the three unit Bailey 762 shelf in which they are mounted will be replaced with two new Westronics series 2100 digital recorders (1G33R612, 613). The H13P680-11B insert blank will have to be replaced with a filed fabrication that has a larger cutout for the new recorders. The new recorders will utilize special auto ranging software that will cause the scales to change (i.e 0-2 to 0-20) when required. The range of the associated transmitters P33N079A, B, N085A, B, C, will be changed to match the new ranges of the recorder. The alarm setpoints of the alarm units G33N062A, B will be changed from 0.1 to 0.2 uMHO per plant staff request.

REASON FOR CHANGE: The Bailey 771 style 1 recorders are obsolete and spare parts are difficult to obtain. Resolution of the recorders is poor because of the large span required to cover fluctuations in conductivity.

SAFETY EVALUATION: The affected transmitters and alarm units are non-seismic, non-seismic category II/I and non-safety related. The recorders are seismic category II/I. The H13P680-11B insert blank is seismic category II/I. The nonsafety related recorders will be installed so that the seismic qualification of the H13P680 panel is maintained. The new H13P680-11B insert blank will be purchased safety related and installed so that the seismic qualification of H13P680 is maintained. Reg. Guide 1.75 separation will be maintained. The affected instruments are not required to perform any active or passive safety related functions. They are not connected to class IE power. They are not required for Reg. Guide 1.97 indication. The affected instruments monitor reactor water conductivity, RWCU inlet/outlet conductivity, CRD system discharge conductivity, reactor water dissolved oxygen, CRD system discharge dissolved oxygen and RWCU inlet dissolved oxygen. The changes of this MCP will not compromise any existing safety related system, structure or component nor will they prevent safe reactor shutdown. No evaluated accident is predicated by a failure of the affected instruments. This design change will be an improvement in terms of recorder reliability and monitoring capability.

The changes of this MCP will not compromise any existing safety related system, structure or component. The failure of the affected instruments and the circuits to which they are connected will not initiate any evaluated transient or accident. The G33 (RWCU) and P33 (process sampling) system operation and function will not change. No interfaces with safety related or important to safety systems are created. This change will therefore not introduce an unreviewed safety question. The affected instruments are not required to mitigate the consequences of any evaluated transient or accident. Per Tech Spec 3/4.4.4, in-line conductivity measurements must be taken every 24 hours during modes 4 or 5 if the reactor water conductivity recorder is not operational. This requirement is not being changed. The conductivity limits specified in Tech Spec Table 3.4.4-1 are not being changed.

Serial Number: 97-057-NPE Document Evaluated: ER 96/0964-00-00

DESCRIPTION OF CHANGE: Engineering Request (ER) 96/0964 adds a 1 inch high lip to the air inlets of the ESF switchgear room coolers to capture the condensation that forms on the cooler coils and also adds a drain line to each cooler to carry the condensation from the coolers. Additionally, the doors for airhandlers Q1T46B001A-A/B-B, Q1T46B004A-A/B-B and Q1T46B005A-A/B-B be split horizontally to allow for easier handling of the door by maintenance crews. The studs on the airhandler are replaced with bolts tack welded to the inside of the unit and bolts are used to hold the door in place.

REASON FOR CHANGE: MNCR 96-066 documents that condensation from the ESF switchgear room coolers is dripping onto the switchgear and other equipment located inside the ESF switchgear rooms. Additionally ER 96/0631 requests that the access doors for airhandler Q1T46B001A-A/B-B, Q1T46B004A-A/B-B and Q1T46B005A-A/B-B be split horizontally to allow for easier handling of the door by maintenance crews. It also requests that the studs on the airhandler are replaced with nuts tack welded to the inside of the unit and bolts are used to hold the door in place.

SAFETY EVALUATION: As discussed in UFSAR Section 9.4.5, the ESF Switchgear room coolers are designed to maintain the temperature of the ESF switchgear rooms within acceptable limits both during normal power operation and during post accident conditions. TRM Section 6.7.3, Area Temperature Monitoring, addresses the temperatures that are required to be maintained in these areas. TRM Section 6.7.1 addresses the operability requirements of the ESF switchgear room coolers.

The addition of the 1" lip to room coolers to capture the condensation that forms on the cooler coils, the addition of a drain line to each cooler to carry the condensation from the coolers and the dividing of the access doors on selected coolers does not affect the operation of the coolers and will not affect the ability of the coolers to maintain the rooms within the required temperature limits.

The modifications to the units do not invalidate any of the analyses or assumptions contained in the UFSAR regarding the ESF Switchgear Room coolers. The changes do not compromise any safety related system or prevent safe reactor shutdown. The ability of equipment important to safety to perform its safety function is not altered by this modification. The Technical Specifications are not affected and the margins of safety are unchanged.

Serial Number: 97-058-NPE   Document Evaluated: LDC 97-115

DESCRIPTION OF CHANGE: This UFSAR change reflects Entergy's change in membership from the Southwest Power Pool Reliability Council (SPP) to the Southeastern Electric Reliability Council (SERC). A primary purpose of the SPP is to provide for increased operating efficiency and continued service reliability. These objectives are accomplished by each member adhering to SPP operating criteria. The objectives of the SERC are consistent with those of the SPP and are accomplished through similar operational protocols.

The FSAR changes are being made to update the proper names of the councils with which Entergy participates in.

REASON FOR CHANGE: On January 1, 1998, Entergy will leave the SPP and join the SERC. Consequently, changes to the GGNS UFSAR will need to be approved prior to this transition.

SAFETY EVALUATION: The changes being made maintain GGNS within the NRC acceptance criteria established during licensing for Section 8.2.3, "Stability," of the GGNS UFSAR. Additionally, the change in membership from the SPP to the SERC has been evaluated by Entergy's Transmission Operations department to identify similarities in operating protocols pertaining to electrical grid reliability and stability. The results of this evaluation indicate no adverse effect on grid reliability and stability in the vicinity of Entergy's nuclear facilities due to changing membership from the SPP to the SERC. Therefore, the results of previously performed grid stability analyses as referenced in SAR Chapters 8 and 15 are not impacted.

Also, this change involves neither modification to plant hardware nor change in plant operations. The capability of safety-related systems required to respond to transient or accident conditions are not adversely impacted by this change. This change has no impact on the basis for any technical specification at GGNS. Therefore, this change does not involve an unreviewed safety question.

Serial Number: 97-059-NPE Document Evaluated: ER 1997-0321-00

DESCRIPTION OF CHANGE: Fire Area 46 has only one Fire Zone (OC406) which, per the Fire Hazards Analysis (FHA), is enclosed by 3-hour rated fire barriers and contains no safety related or safe shutdown components. Fire Zone OC406 is a small area (200 sq. ft.) and was originally planned to house the Unit 2 Instrument Motor Generator Set (IMGS). The Unit 2 IMGS was never installed and OC406 was converted to a "Unit 1 Support Area". This change deletes Fire Area 46 and incorporates the are (Fire Zone OC406) into Fire Area 42. Fire Area 42 is comprised of a number of fire zones, one of which is Fire Zone OC405 "Unit 1 Support Area". Specifically, the fire boundaries (presently 3-hour rated) that separate Fire Zone OC406 from Fire Zone OC405 are being derated. In addition to the FHA and drawing changes, this change requires a revision to the Fire Pre-Plan for OC406 (show derated walls) and Combustible Heat Load Calculation MC-QSP64-86058 (show new fire area designation for OC406).

REASON FOR CHANGE: GGCR 97-0261-00 documented the fact that penetration CE-261D had no fire rated penetration seal installed. This penetration is blockout with cable tray and conduits passing through and is located in the 3-hour rated fire barrier separating Fire Area 46 from Fire Zone OC405 (Fire Area 42). Based on review of the FHA for these two fire area and walkdowns of the areas and penetration, it was determined that there is no need or requirement for fire separation between Fire Area 46 and Fire Zone OC405 (Fire Area 42). Therefore, Fire Area 46 will be deleted and the area (Fire Zone OC406) included in Fire Area 42.

SAFETY EVALUATION: License Condition 2.C.41 allows GGNS to make changes to the approved Fire Protection Program without prior approval of the Commission as long as those changes do not adversely affect the ability to achieve and maintain safe shutdown in the event of a fire. Generic Letter 86-10 states that in addition to the above requirement, two additional conditions must be met to make changes to the fire protection program without prior approval of the Commission. These two conditions are 1) under 10 CFR 50.59 such changes must not otherwise involve a change in a license condition of the TS or result in an unreviewed safety question and 2) such changes must not result in failure to complete prior commitments concerning the fire protection program which have been approved by the Commission. The changes documented in ER 97-0321-00 do not involve any commitments concerning the fire protection program previously approved by the Commission. In addition, the changes do not adversely affect the ability to achieve and maintain safe shutdown in the event of a fire, and as documented in this safety evaluation, do not involve a change in a license condition, TS, or result in an unreviewed safety question. Therefore this fire protection program change is acceptable and does not require prior approval of the Commission.

Serial Number: 97-060-NPE Document Evaluated: ER 97/0443-00-00

DESCRIPTION OF CHANGE: This modification provides for the removal of a Unit II Cardox Fire Suppression Panel, associated ETL panel and other peripheral equipment. The removal of this panel is required for its use as a test subject for Seismic qualification. The panel selected for removal is N2P64D209, originally installed for protection of Unit II Division III switchgear room (OC213). This panel was installed during Unit II construction, per A-0630, this panel has not been activated and as stated on M-0035F, it is not required for Unit 1 function. This Safety Evaluation is required to support the FSAR figure change to remove reference to this panel from Figure 9.5-6 (P&ID M-0035F).

REASON FOR CHANGE: GGCR1997-0284-00 documents a potential for inadvertent discharge of carbon dioxide (the fire suppression agent) and damper closure during a Seismic event as a result of chatter by the Cardox panels' internal initiation relays. Currently, all areas currently provided protection by this panel type are at risk of this inadvertent discharge and HVAC damper closure until it has been shown that this chatter, if any, of the initiation relays is acceptable.

SAFETY EVALUATION: This CO_2 panel has never been declared operational/activated and no credit is taken for its installation. Per M-0035F this panel is not required for the function of Unit I and per A-0630 has not been activated. Due to its status, it is not credited for fire suppression in the Unit II, Division III switchgear room. No new or additional equipment is being added via this modification nor is the function of any equipment being changed. Currently, the GGNS Fire Hazards Analysis Report (M-500) does not take credit for the presence of this panel or the suppression it was to provide. Section 6.2.5 of the GGNS Technical Specifications list the CO_2 panels required to be operational when the equipment protected is required to be operational, of which this panel is not listed. Neither the GGNS Technical Specifications nor its bases reference, credit, or determine any margin of safety on the presence of this panel. Therefore, no change to the GGNS Technical Specifications or bases is required. The removal of this panel will not increase the probability of occurrence or the consequences of an accident previously evaluated in the SAR. Nor will it increase the probability of occurrence or the consequences of a malfunction of equipment important to safety previously evaluated in the SAR. it will not create the possibility for an accident of a different type than any previously evaluated in the SAR or a malfunction of equipment important to safety of a different type than any previously evaluated in the SAR. The revision to the figure provides for correct reflection of plant configuration as described in the SAR.

Serial Number: 97-061-NPE   Document Evaluated: ER 96/1018

DESCRIPTION OF CHANGE: Tubing, a new solenoid unloading valve P81F515 and a new blocking valve P81F104 are to be added to the existing unloading port on the Division III diesel driven starting air compressor P81C002 discharge check valve P81F043. The unloading valve will be de-energized open when the diesel driven starting air compressor is not running, and it will vent that portion of piping between the compressor discharge check valve P81F043 and the compressor P81C002, and thus provide a drainage path to an Oily Waste Drain (OWD) hub for any leakage of air or water past the P81F043 compressor discharge check valve. The new solenoid will be actuated by the 12 volt DC control power for the diesel driven starting air compressor coming from the 12 volt battery power source provided for starting the diesel driven starting air compressors. When the diesel driven starting air compressor starts the solenoid will be energized shut and when the diesel driven starting air compressor stops it will be-energize to open. A blocking valve will be provided and will be normally open. The blocking valve will be installed upstream of the solenoid. The drain line will be sloped down from the solenoid to a oily waste drain hub within the Division III diesel room.

REASON FOR CHANGE: The diesel driven starting air compressor P81C002 is experiencing water buildup in the compressor oil sump. NPE evaluated the sources of water into the compressor oil sump and determined that the water buildup is due to a small amount of air leakage past the diesel driven starting air compressor discharge check valve P81F043.

SAFETY EVALUATION: The unloading solenoid and blocking valve will improve the reliability of the Division III diesel driven starting air compressor side of the diesel driven compressor discharge check valve P81F043 to be vented and de-pressurized thus providing a drainage path for any moist air leakage thus preventing water buildup in the compressor oil sump. The solenoid operated vent path will provide not only a vent path away from the compressor sump but will also provide a greater differential pressure across the compressor discharge check valve P81F043 to assist in the seating and sealing the valve. The blocking valve in the solenoid unloading line will allow for isolation and repair of the unloading solenoid valve without interruption of service from the diesel driven air start compressor. Finally, the power source to the new solenoid unloading valve is the same power supply for the compressor diesel stop solenoid. The compressors diesel stop solenoid energizes to run and de-energizes to the compressors diesel. The addition of the solenoid unloading valve on the diesel driven starting air compressor discharge check valve P81G043 and the blocking valve will not adversely impact the reliability of the HPCS DG air start system. This system is a redundant non-safety related system which has no active safety function and is defined as a nonessential component in the SAR. Based on the above information, NPE has concluded that this change will not increase the probability of occurrence of an accident previously evaluated in the SAR, nor increase the consequences of an accident previously evaluated in the SAR, nor increase the probability of occurrence of a malfunction of equipment important to safety previously evaluated in the SAR, nor increase the consequences of a malfunction of equipment important to safety previously evaluated in the SAR, nor does it create the possibility for an accident of a different type than any previously evaluated in the SAR, nor does it create the possibility for a malfunction of equipment important to safety of a different type than any previously evaluated in the SAR, nor does it reduce the margin of safety as defined in the basis for any Technical Specification. As a result of the above discussion, NPE has concluded that this change does not result in an unreviewed safety question.

Serial Number: 97-062-NPE Document Evaluated: ER 97-0249-00, R1

DESCRIPTION OF CHANGE: The changes made via this ER are associated with the Grand Gulf Nuclear Station Security Plan. These changes are being made in preparation for the Operational Safeguards Response Evaluation (OSRE). The changes made per this ER include installation of protective (defensive) shields, equipment (gun/ammunition) storage lockers, adversary delay devices (chain link fences, expanding doors, grenade netting), and power and telecommunications associated with learning centers and manned posts established by the changes. The changes result in a change in occupancy in Fire Zone OC406, Fire Area 46 in the Control Building. The fire zone will now be utilized as a security office. In addition, adversary delay devices installed by this change necessitate changes to the fire pre-plans to reflect their potential effect on fire brigade access.

REASON FOR CHANGE: These changes are being implemented to enhance the defensive strategies for protection of Grand Gulf Nuclear Station against attack by adversaries. These changes are made in preparation for the Operational Safeguards Response Evaluation (OSRE).

SAFETY EVALUATION: Items installed in the Auxiliary Building and Control Building have been designed and mounted in accordance with the plant design basis criteria, including seismic loads, as delineated in UFSAR Section 3.7. The C83 system (Security System), and the new shields, barriers, and lockers, are not governed by Technical Specifications. The changes to this system will enhance GGNS defensive strategies. All cabling and raceway modifications will be in accordance with the separation requirements of Reg. Guide 1.75. The components affected by the ER are not required to mitigate the consequences of any evaluated transient or accident. No new interfaces with equipment important to safety are created and no new failure modes which would alter existing accident analyses are introduced. Implementation of the changes described in this ER will not affect the operation of any safety-related systems.

ER 97-0249-00, Rev. 1 also installs desks, chairs, computers, and/or telephones in one area in the Auxiliary Building and two areas in the Control Building. License Condition 2.C.41 allows GGNS to make changes to the approved Fire Protection Program without prior approval of the Commission as long as those changes do not adversely affect the ability to achieve and maintain safe shutdown in the event of a fire. Therefore, from a 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." As documented in NPE FHAR-97/0002, the change in occupancy in Fire Zone OC406 has resulted in a reduction in the combustible loading in the fire zone and as such a reduction in the fire hazards in the zone. Therefore, the ability to achieve and maintain safe shutdown conditions in the event of a fire, as presently analyzed in the UFSAR, has not been adversely affected.

Therefore, these changes will not increase the probability or consequences of accidents or malfunction of equipment important to safety previously evaluated in the SAR. Also, these changes will not create the possibility for an accident or malfunction of equipment important to safety of a different type than any previously evaluated in the SAR. These changes are not addressed by Technical Specifications (TS) or the Technical Requirements Manual (TRM). Changes made by this ER involve a specific change in occupancy and a reduction in the combustible loading in Fire Zone OC406, which neither the TS nor the TRM specifically address. As a result, there will be no change to the TS or the TRM or the Bases for any TS or TRM.

Serial Number: 97-063-NPE Document Evaluated: Calculation XC-Q1G41-97007, R0

DESCRIPTION OF CHANGE: The spent fuel pool decay heat load analysis has been revised considering updated projections on GGNS batch size, reduced outage duration, and historical discharge information. This updated heat load is somewhat larger than that calculated by the most recent analysis (Ref. 1) but below the results of prior analyses (Ref. 4). SAR Section 9.1.3.1.2 and Tables 9.1-10, 9.1-12 and 9.1-13 will be revised with the results of this analysis.

REASON FOR CHANGE: The current fuel pool heat load analysis assumes 30-day refueling outages and a 284-bundle equilibrium reload batch. However, refueling outages shorter than 30 days may be achieved at GGNS and the application of advanced fuel types is predicted to eventually reduce the reload batch size to 240 bundles. Since this heat load analysis has not been revised since Cycle 5 (Ref. 1), actual operating history can be incorporated in lieu of projected performance. On these bases, the fuel pool decay heat load analysis was revised with this new information. This calculation is part of a larger effort to address NRC concerns reported in GNRI-97/00074 regarding the GGNS fuel pool heat load and heat rejection capability. Additional analyses regarding the fuel pool heat rejection capability are being addressed separately by other actions associated with LCTS 33002.

SAFETY EVALUATION: This evaluation concludes that this revision to the fuel pool heat load analysis involves no changes to the Technical Specifications and no unreviewed safety question.

Serial Number: 97-064-NPE Document Evaluated: ER 97/0084-00-R0

DESCRIPTION OF CHANGE: This design shall trip the Division 1 & 2, Unit 1 & 2, Z77 Air Handling Supply and Exhaust fans and the Remote Shutdown panel heat pump upon a detected fire from either of the three ESF Switchgear Room or Remote Shutdown Panel Room fire detection cabinets. Tripping of these loads will occur 5 seconds after a detected fire. Fusing of the associated ETLs (electric thermal links), that will cause the dampers to close, will be delayed by 30 seconds from the time a fire is detected, to ensure air flow through their associated ducts has stopped due to tripping of the Z77 fans and heat pump. After a 45 second delay from a detected fire, the local and master CO_2 control valves will be opened. Fifty seconds after a fire is detected, the Z77 fans will be automatically restarted if an automatic start signal exist. the heat pump shall remain tripped until an operator resets the logic associated with this modification at the local handswitch station 1Z77HSM001A.

REASON FOR CHANGE: MNCR 96/0047 identified a non-conforming condition that could degrade the ability of the CO_2 fire suppression system from performing its function within, among other locations within the plant, the ESF Switchgear Rooms and Remote Shutdown Panel Room. This non-conformance identified that the Z77 dampers are not designed to close with air flow through their ducts. The Z77 system that provides the cooling for the ESF Switchgear and Remote Shutdown Panel Rooms is not designed to shutdown upon a detected fire thereby ensuring closure of the Z77 dampers.

Presently, the Z77 dampers are gravity powered closed upon fusing of their associated electric thermal links (ETLs). The ETLs are provided a signal to fuse after a 5 second delay upon a detected fire. The three ESF Switchgear Rooms and one Remote Shutdown Panel Room each have a separate fire detection control cabinet and ETHYL Terminal Box to close associated dampers. The fire detection control cabinet supervises the room for a fire, provides an audible and visual alarm upon a sensed fire, provides a signal to its associated ETL TB which fuses the affected rooms ETL controlled dampers after a 5 second delay and provides a signal, after a 30 second delay, to open both the master and local valve to allow CO_2 to be dumped into the affected room. The opening of the local valve is timed to ensure that the room is filled with a minimum of 50% by volume of CO_2 concentration.

SAFETY EVALUATION: This design will not affect the GGNS Technical Specifications. Nor will create an unreviewed safety question or reduce any margin of safety. This design will not allow the temperature within any of the affected rooms to approach the temperature limit in the TRM which requires the equipment within these rooms to be considered inoperable. Per calculation MC-Q1Z77-97029, Rev. 0, the temperature rise that these rooms will experience due to this design change is bounded by the maximum temperature identified in Engineering Report GGNS 92-0002, Rev. 0, which ensures continued operation of the affected equipment. The design meets the intent of Reg. Guide 1.75 for required separation and isolation between Class 1E equipment and Non-Class 1E equipment and between Class 1E divisions. The evaluation contained in Section 4.0 of the subject ER demonstrates that any credible failure of the Non-Class 1E Fire Detection Equipment, that supports tripping and allowing restart of the Safe Shutdown ESF Switchgear and Battery Room Ventilation System, will not propagate a failure to the ESF Switchgear and Battery Room Ventilation System. This evaluation also documents that the circuits added to the Safe Shutdown Z77 equipment are routed, protected and designed to support continued compliance with GGNS's Fire Protection Plan. The design utilizes Seismic Category 1 supports for all conduits and supports for the local handswitch station. Electrical Calculations EC-Q1R28-90037, EC-Q1R28-90039, EC-Q1R28-91019 and EC-Q1R28-91020 have been reviewed and this modification will not have an adverse impact upon Division I/II power panel voltage levels or coordination.

Serial Number: 97-065-NPE Document Evaluated: ER 97/0466-00-00

DESCRIPTION OF CHANGE: Accept-As-Is disposition for GGCR-197-0393-00, which documents the connection of the control building floor drains in the upper and lower cable spreading rooms to the yard storm drain system, thereby partially venting control building during a design basis tornado event. UFSAR Sections 2.4 and 9.3 currently imply that all floor drains discharge to sumps inside the powerblock. These sections will be revised to indicate that the floor drains serving the control building upper and lower cable spreading rooms discharge into the storm drain system. Section 3.8 will be revised to discuss the possibility of depressurization of the control building due to partial venting through the floor and storm drains.

REASON FOR CHANGE: GGCR1997-0393-00 identified a condition where floor drains in the upper and lower cable spreading rooms connect to the yard storm drain system. The drains which penetrate the exterior walls of the control building are not provided with a means of isolating to prevent building depressurization in the event of a tornado. the fact that the floor drains empty into the yard storm drains is in accordance with the design drawings. However, the UFSAR and corresponding tornado depressurization calculations do not consider the floor and storm drains as a potential vent path.

SAFETY EVALUATION: Floor drains in the upper and lower cable spreading room do not breach the control room envelope boundary. Calculation C-T-455.0, Supplement 2, demonstrates that interior CMU walls in the area, including CMU walls serving as control room envelope boundaries, have sufficient structural margin to withstand loads which might result from partial venting of the control building through the floor and storm drain systems. Therefore, requirements and margins of safety for the control room HVAC system, as discussed in TS Section 3.7.3 and B3.7.3, are unaffected.

The probability of occurrence for the design basis tornado described in UFSAR Sections 3.3 and 3.5 remains unchanged. Calculation C-T-455.0, Supplement 2, was performed using the methodology set forth in base calculation C-T-4550 and Bechtel Topical Report BC-TOP-3-A and demonstrates that Interior CMU walls will not collapse, posing a II/I hazard to equipment in the control building, and CMU wall boundaries (i.e., fire and Control Room Envelope) are unaffected. Floor drains in the upper and lower cable spreading rooms do not breach the control room envelope, or represent a potential leak path affecting the control room HVAC system. Therefore, the probability of occurrence for accidents and equipment malfunction previously evaluated in the UFSAR has not increased. For these same reasons, the consequences of accidents and equipment malfunctions previously evaluated in the UFSAR have not increased.

This change does not affect any parameter which could alter radionuclide population, release rate, or duration; or create new release mechanisms. C-T-455-0, Supplement 2, determined that the maximum pressure variations in the upper and lower cable spreading rooms are expected to be 0.10 psi and 0.24 psi, respectively. Several safety related instruments were noted in the lower cable spreading room, but in all instances, the configuration of the instrumentation was such that it would not be susceptible to the expected pressure transient. The only significant items identified in the upper cable spreading rooms were the Control Room HVAC ducts. The expected differential pressure across the ducts is expected to have no noticeable effect. Therefore, the possibility of an accident or equipment malfunction not yet considered in the UFSAR has not been created.

Serial Number: 97-066-NPE Document Evaluated: Temp. Alt. 97-0009

DESCRIPTION OF CHANGE: This safety evaluation addresses the bypassing of travel limit switches on the auxiliary building Fuel Handling Platform using a Temporary Alteration. These switches provide interlocks which normally prevent the platform from traveling out of the allowed main spent fuel pool zone unless the fuel mast trolley is positioned such that the mast cannot come into contact with the pool walls. That is, the trolley and cab must be properly aligned within the allowed gate, transfer canal, or cask storage pool zones.

In order to perform some activities using only the auxiliary monorail mounted hoist (such as processing of spent control blades), it may be necessary to raise the fuel mast out of the water and stow it in its "dry" position per approved procedures. This is needed to allow auxiliary hoist access to remote areas of the pool. This also requires that the main mast trolley be left centered on the platform. In these circumstances, the interlocks cannot be met for the bridge to leave the main pool area because the trolley cannot be positioned in the required zones. These interlocks are not needed under these circumstances however, because the mast is out of the water and cannot contact pool walls regardless of the bridge position. Thus, the limit switch interlocks must be bypassed in order to allow the platform to leave the pool zone. This enables the auxiliary monorail hoist to transfer items into the transfer canal and cask storage pool with the main mast stowed, while still having access to edges and corners of the fuel pool.

In particular, the Temporary Alteration will move the position switch arms to prevent actuation of position switches LS1, LS3, and LS7.

REASON FOR CHANGE: At times, it is necessary to place the Fuel Handling Platform over the transfer canal and cask storage pool for purposes other than fuel movement as described above. This situation was not fully considered in the original limit switch/interlock design. Thus, it is necessary to bypass these interlocks to allow the platform to be moved out of the pool such that work not modified to clarify that this is acceptable since this evaluation will be performed numerous times throughout remaining plant life.

SAFETY EVALUATION: The interlocks being bypassed by this Temporary Alteration affect only the operability of the fuel platform main fuel mast. The mast will be in its stowed position and remain out of service while the alteration is in effect. Various FSAR discussions confirm that the travel platform interlocks were not intended to protect equipment other than the fuel mast. An examination of the switch logic shows that the auxiliary hoists loads can contact the pool walls whether or not these interlocks are in service. Further, it is clear from the FSAR that operation of the hoists individually was contemplated, including operation over the cask storage pool.

Therefore, provided that Fuel Handling Platform mast remains stowed out of the water, bypassing the platform travel interlocks does not affect the operability of the platform or auxiliary monorail hoist. No new types of events are created, nor is the likelihood of any previously evaluated event increased. No safety limits or other margins for the Technical Specifications are affected, and none of the limits assumed in accident analyses are changed. Modification of the interlocks has no impact on the environment.

Thus, bypassing of the Fuel Handling Platform travel limit switch interlocks does not create an unreviewed safety question.

Serial Number: 97-067-NPE   Document Evaluated: ER 94-0039, Rev. 1

DESCRIPTION OF CHANGE: This safety evaluation assesses a comprehensive study of the safety functions for the feedwater check valves (FWCVs B21F010A/B and B21F032A/B) and thorough characterization of the operational conditions for these functions. This study is documented in Engineering Report No. GGNS-94-0039, Rev. 1. The primary purposes of this report are to provide the bases for changing the leak testing requirements of the FWCVs from air to water, establishing more accurate allowable leakage limits, determining safety and design margins, and improving the overall functional performance of GGNS.

REASON FOR CHANGE: Prior to RF07, design and testing requirements were established from an original GGNS Technical Specification license condition that necessitated extremely restrictive air testing with tight allowable leakage limits. As a direct result of these requirements, the original high endurance hard seats in the FWCVs were modified with elastomeric seals to provide a sealing surface capable of meeting the stringent air leakage limits. However, due to the relatively short functional life of the elastomeric seals compared to the hard seats, the overall reliability of the sealing function actually decreased. This degraded performance was exhibited by frequent seal failures and subsequent valve repairs (typically every refueling outage). Thus, adherence to the original license condition requirements have resulted in higher operating costs, more difficult testing methods, higher station personnel radiation doses, and an overall degradation in equipment performance. The reasons for this change are to improve the long term equipment performance, operational burdens, and real effective margins associated with the FWCV safety functions.

SAFETY EVALUATION: This safety evaluation concludes that the proposed changes to the leak testing criteria for the FWCVs and other related feedwater isolation valves do not involve an unreviewed safety question. The FWCVs perform active safety-related design functions to:

1. prevent the loss of reactor coolant inventory during certain feedwater line breaks (i.e., reactor isolation function)

2. prevent the transport of radioactive material through the feedwater leakage pathway (i.e., containment isolation function) during the short-term period following a severe accident involving core damage (e.g., DBA-LOCA) and prior to operation of the Feedwater Leakage Control System (FWLCS).

The original Technical Specification requirements were based only on the containment isolation function as supported by limited analysis and the belief that all of the high energy feedwater evaporated during the LOCA blowdown. This phenomena would have resulted in completely voided feedwater lines and thus a steam environment within the feedwater leak pathway. Given this condition, the appropriate testing criteria would thus be based on air with a relatively tight allowable limit.

The analyses summarized in Engineering Report No. GGNS-94-0039, Rev. 1, evaluated all postulated accident conditions associated with the containment isolation functions in which the FWCV allowable leakage must mitigate the consequences of an accident to within the limits of 10CFR100. The primary analyses performed in support of this engineering study are:

1. a comprehensive transient thermal-hydraulic analysis (using design basis RELAP code and models) of the DBA-LOCA and the resulting feedwater line blowdown (Ref. 1)

2. an analysis of the post-accident suppression pool water inventory (Ref. 2)

3. a radiological dose analysis of a feedwater line break inside the drywell (using design basis TRANSACT code and models; Ref. 3)

4. a radiological dose analysis of the post-LOCA operation of the feedwater leakage control system (using design basis TRANSACT code and models; Ref. 4)

5. a comprehensive evaluation of potential containment leak paths (Ref. 5)

6. a seismic capabilities evaluation of the feedwater system (see Engineering Report)

The analysis results show that for the nonlimiting feedwater line break accidents, complete failure of the containment isolation function of the FWCVs (i.e., gross leakage through all feedwater leak pathway isolation valves) will not result in offsite and control room dose consequences exceeding the applicable regulatory limits. For severe accidents such as the limiting DBA-LOCA, no leak path exists through the feedwater lines during the reactor blowdown phase since the direction of flow for the steam/liquid mixture is only from the feedwater lines into the reactor. Following the blowdown phase and prior to the initiation of the FWLCS (i.e., the leak phase), sufficient subcooled water remains in various portions of the feedwater piping to form liquid water loop seals that effectively isolate this leak path. The feedwater leak paths remain isolated by these loop seals until the FWLCS has been initiated and refloods the containment portions of the feedwater lines with suppression pool water.

The results of this study demonstrate that the most appropriate leak testing requirements for the FWCVs are:

1. liquid water as a test medium (i.e., hydrostatic instead of pneumatic testing)

2. allowable leak rate for each feedwater line pathway (i.e., containment penetration) is 1 gpm @ 11.5 psid

3. valve specific allowable leak rate of 1 gpm @ 11.5 psid for each of the outboard containment and feedwater isolation valves B21F032A&B, and B21F065A&B

4. valve specific allowable leak rate of 7 gpm @ 11.5 psid for each of the inboard containment isolation valves B21F010A&B.

Serial Number: 97-068-PSE Document Evaluated: Q1T51B005

DESCRIPTION OF CHANGE: The special instructions are written to flush the Q1T51B005 room cooler if the cooler is found in a degraded condition. The room cooler will be flushed with Betz DE 1178 a mild citric acid. Betz DE 1178 cleaning solution is used to flush the deposits out of the cooler/piping thus restoring cooler efficiency. Betz DE 1178 is 40% citric acid, 10% phosphonate, 10% corrosion inhibitor and 40% inert ingredients. The use of Betz DE 1178 cleaning solution in Q1T51B005 room cooler is approved for use by ER 97/0300-01 and SE 970029 R00.

Room Cooler Q1T51B005 will be isolated from SSW by closing the inlet and outlet isolation valves. Fittings, will be connected to the low points in the system to drain as much water as possible out of the piping/cooler into 55 gallon drums. Sufficient acid will be added to water to make a 10% solution of acid to water. An injection and collection point will be established to flush the piping and cooling coil. The selection for injection and collection minimizes slow flow rates of acid solution and stagnant conditions. The acid flush will continue until one of the follow criteria is met 1) dissolved iron content ceases to increase 2) dissolved copper content exceeds 700 ppm 3) a maximum of three hours flushing time. The acid solution will be drained from the piping/cooler into the 55 gallon drums using the low point drains. The room cooler will be valved back to SSW and the residual acid solution will be washed away.

REASON FOR CHANGE: FSAR section 9.4.5.2.4 states that each safety-related pump room is provided with a full capacity fan-coil unit to prevent the room temperature from exceeding 150°F during pump operation. Because of the fouling in the piping/tubing, the heat removal capability of the RHR C Pump Room Cooler can be less than the requirements of MS-39.0. The room cooler coils will need to be cleaned to improve the heat transfer capabilities.

SAFETY EVALUATION: A chemical cleaning process using Betz DE 1178 has been selected to improve the cleanliness of the cooling coils for Q1T51B005 RHR C Pump Room Cooler which will enhance the heat transfer capability of the unit. The materials of construction was reviewed and evaluated for compatibility under Safety Evaluation 970029 R00. This safety evaluation evaluates the method of acid flushing and ensuring the process controls are or consequences of an accident evaluated in the SAR, does not create the possibility of a new accident or malfunction and does not reduce any margin of safety defined in the basis for any technical specification. There are no unreviewed safety questions or issues from using the cleaning instruction discussed in this evaluation.

The floor drains in the surrounding area will be intentionally covered during performance of this activity to ensure that acid solution does not inadvertently enter the floor drain system. The floor drain system serves to divert gross leakage away from affected equipment in the area.

During performance of the flush activity, personnel will be continuously on the job. In the event of that a flood event were to occur, necessitating uncovering the floor drain, this can be accomplished by personnel at the job site. Per FSAR section 6.3.1.1.3, the ECCS rooms are constructed to be water tight to protect against mass flooding of redundant ECCS pumps therefore temporary covering of the floor drain in the immediate area will not increase the consequences of a flooding event.

Serial Number: 97-069-PSE Document Evaluated: LDC 97-044

DESCRIPTION OF CHANGE: Technical Specification SR 3.6.3.2.3 and 3.6.3.2.4 require the hydrogen igniters to be tested once every 18 months. The fourth sentence of the Bases for the Surveillance states that the hydrogen igniter surveillance must be conducted during a plant outage to prevent an unplanned transient with the reactor at power. The change will delete this fourth sentence of the Bases for Technical Specification Surveillance Requirement (SR) 3.6.3.2.3 and 3.6.3.2.4. The fourth sentence currently states "The 18 month Frequency is based on the need to perform an unplanned transient if the Surveillance were performed with the reactor at power.

REASON FOR CHANGE: Engineering Request (ER) 96/0647 has recommended that the Bases for Technical Specification SR 3.6.3.2.3 and 3.6.3.2.4 be revised to delete the Bases description that the hydrogen igniter surveillance be conducted during a plant outage. Deleting the sentence will remove the described restriction and allow partial or complete surveillance with the reactor at power. Allowing a surveillance of all or part of the hydrogen igniters during plant operation will reduce the scope of outage work when operator manpower is heavily taxed.

SAFETY EVALUATION: No postulated scenarios were found that could lead to unplanned transients. All igniters are currently energized once per 184 days (once per 92 days prior to PCOL-93/01) per Technical Specification SR 3.6.3.2.1 for current and voltage checks. Energizing the igniters while at power has not caused any unplanned transients. Typical steam leakage rates (such as valve packing leaks) result in small total amounts of hydrogen being generated in enclosed areas. Normal room cooling requirements during plant operation provide adequate air circulation to preclude any concentrated buildup of hydrogen. The hydrogen igniters were originally designed to be periodically energized for surveillances with the plant in operation and are designed for complete submergence in water while hot. No other unplanned transients were postulated that could be initiated by conducting the surveillances. The change to the Technical Specification Bases does not change the hydrogen igniters identified as Normally Accessible and Normally Inaccessible as listed in the TRM. The change to the Technical Specification Bases does not alter the design or function of the hydrogen igniters, nor does the change alter any existing surveillance scope, frequency, or acceptance criterion.

Serial Number: 97-070-NPE Document Evaluated: ER 96-0383-00-00

DESCRIPTION OF CHANGE: The Change will provide all the necessary instructions to perform the replacement of the SSW supply and return carbon steel piping (including isolation valve P41F198B) to radiation monitor D17J006 with similar components of stainless steel material. Piping system flush connections will be added to the piping system for future use. The radiation monitor pump discharge piping system design, normal and maximum operating pressure rating is also being increased by this change.

REASON FOR CHANGE: The weekly chemical additions to the SSW Basins has not proven effective in combating the biological growth and fouling in the SSW supply and return piping for Loop "B" radiation monitor D17J006. When degraded conditions exist, the supply and return piping must be replaced or chemically cleaned to obtain acceptable flow rates. Replacing the supply and return piping components with similar stainless steel components should eliminate or greatly reduce any future fouling.

SAFETY EVALUATION: Replacing valve P41F198B as well as the supply/return piping to the SSW radiation monitor D17J006 with similar components of higher operating pressure/temperature rating and stainless steel material will have no effect on safety related systems required for plant safe shutdown. This piping system material replacement will not affect current plant operation or change the operation of the Standby Service Water System (P41) or the Radiation Monitoring System (D17) or any other plant system during normal or accident conditions. Therefore, this modification will not reduce any margins of safety discussed in the GGNS licensing documents. The SSW radiation monitor supply and return piping system integrity has been assured by using the criteria presented in documents M-18, Rev. 20. "Office and Field Engineering Users manual for Routing and Supporting Two inch and Under Piping", and Specification M-1398, Rev. 16. The design change and the resulting update of GGNS-MS-02 and SERI-MS-03 in no way affect the Technical Specifications and cause only a minor figure change to the UFSAR. This design change will not adversely impact the seismic capability of the D17 Radiation Monitoring System skid and equipment (D17J006). The implementation process of this design change including two separate phases was evaluated against the applicable design criteria, installation and operational requirements and all necessary requirements and commitments are met. Therefore, no Unreviewed Safety Question is created as a result of this change.

Serial Number: 97-071-NPE   Document Evaluated: ER 96/0678-00-00

DESCRIPTION OF CHANGE: The changes addressed in this EER are to the Turbine Building Cooling Water System (TBCW) (P43). They involve the installation of vent lines/valves on the TBCW suction and discharge lines and a manual bypass line/valve around the TBCW surge tank makeup valve (N1P43F520).

REASON FOR CHANGE: Filling the TBCW pumps (N1P43C001A/B/C) when they have drained for maintenance is an involved process requiring temporary hoses and much operator involvement. This process still leaves much air in the TBCW pump suction and discharge piping (existing vents are to low). Incident Review Board Action Item IR 91-03-06 documents a near loss of the Instrument Air Compressors while attempting to clear tags on P43-C001A (TBCW "A" Pump). Five "Corrective Actions" were identified by IR 91-03-06 and evaluated per EER 93/6004. Of the items evaluated, all but two remain open and are addressed by this ER Response. The installation of the vent lines/valves close to the TBCW pump suction and discharge valves will eliminate much of the air that is left per the present vent locations. Placing a manual bypass around the TBCW surge tank air operated makeup valve (N1P43F520) will insure a continued source of makeup for the TBCW surge tank (N1P43A001) in the event of a malfunction of the air operated makeup valve.

SAFETY EVALUATION: The Turbine Building Cooling Water System has no safety-related function as defined in Section 3.2 of the FSAR. Failure of the system will not compromise any safety-related equipment or component and will not prevent safe shutdown of the plant. The modifications made by ER 96/0678-00-00 will in no way impact any of the accident analyses presented in the FSAR. No new failure modes are being created, thus no possibility of an accident or malfunction of a different type than previously analyzed is possible. Failure of the system will not compromise any safety-related system or component and will not prevent safe reactor shutdown, thus the margin of safety will not be reduced.

Serial Number: 97-072-NPE Document Evaluated: ER 97/0222-00

DESCRIPTION OF CHANGE: The purpose of this ER response is to replace approximately 34 feet of large bore piping downstream of each heater drain pump. The selected replacement material will be a seamed carbon steel piping with a 0.08" to 0.10" stainless steel (304-L) clad to provide enhanced erosion resistance. The piping being replaced is in the GGNS/erosion corrosion program (MS-41) as items 212, 213, 254 and 273 for the "A" pump discharge pipe and item 214 for the "B" pump discharge line. This change includes the replacement of piping and fittings, and the reinstallation of an existing pressure sensing line and drain for each train (note there are no required or recommend changes to the small bore line material) and large bore process valves. The replacement piping has essentially the same form, fit, and function as the existing components since it has an equivalent internal diameter, pressure rating and minimum wall thickness as the piping that it is replacing.

REASON FOR CHANGE: The existing pump discharge piping segments for the heater drain pumps, N1N23-C001A/B, is experiencing wall thinning as a result of single phase flow accelerated erosion/corrosion based on high water velocity and line configuration. Use of a clad pipe should reduce susceptibility of the replaced pipe segment to additional erosion without requiring extensive system reevaluation and potential configuration changes that are normally required for pipe material changes (i.e. chrome-moly pipe), and will not contribute toward future iron transport.

SAFETY EVALUATION: As stated in UFSAR Section 10.4.7.3, this portion of the condensate and feedwater system (which includes heater drain pumps per 10.4.7.2.4) provides no safety function. The system analysis has shown that a failure of the system will not compromise any safety-related systems or prevent safe shutdown. Failure of this system could impact feedwater heating, which could result in a reactor trip due to reduced coolant temperature. The change is intended to reduce the possibility of a line failure of the selected heater drain pump piping due to single phase flow accelerated corrosion (FAC) induced wall thinning of the piping via the use of a piping material that is more resistive to this FAC. The interfacing piping will not be reconfigured and will continue to function as originally designed. The changes made by the ER affect each heater drain pump's respective discharge line, however, because the changes are intended to meet the requirements of the original design (component integrity, welding, pressure ratings, etc.), they will not degrade the integrity of the heater drain system nor will they degrade or prevent actions described in the accident analysis. The changes do not increase the probability of occurrence or increase the consequences of malfunction of equipment important to safety or of a different type than previously evaluated in the UFSAR. The Technical Specifications are not affected, and the margin of safety remains unchanged. The piping installed by this design change are not affected, and the margin of safety remains unchanged. The piping installed by this design change meet ANSI B31.1 code requirements and is supported for the appropriate dead weight and thermal loads.

Serial Number: 97-073-NPE   Document Evaluated: ER 97/0221-00

DESCRIPTION OF CHANGE: The purpose of this ER response is to replace approximately 30 feet of large bore process piping between check valve N1N36-F013A and an upstream tee fitting. The selected replacement material will be a seamed carbon piping having a 0.08" to 0.10" thick stainless steel clad. The piping being replaced is in the GGNS erosion/corrosion program (MS-41) as item 331. The replacement piping has essentially the same form, fit and function as the existing components since it has an equivalent internal diameter, pressure rating and minimum wall thickness as the piping that it is replacing.

REASON FOR CHANGE: Existing piping segment for the extraction steam line to feedwater heater 5, between check valve N1N36-F013A and upstream tee fitting shows pipe wall degradation due to FAC and must be replaced. use of a clad pipe will reduce susceptibility of the replaced pipe segment to additional erosion without requiring extensive system reevaluation and potential configuration changes that are normally required for pipe material changes (i.e. chrome-moly pipe), and will not contribute toward future iron transport.

SAFETY EVALUATION: As stated in UFSAR Section 10.3, the Main and Reheat Steam system (which includes extraction steam) has only power generation design bases, and per Section 10.3.3, this portion of the main steam system provides no safety function. The system analysis has shown that a failure of the system will not compromise any safety-related systems or prevent safe shutdown, but could result in a reactor trip due to reduced coolant temperature. The change is intended to reduce the possibility of a line failure of the selected extraction steam piping due to potential for two phase flow induced wall thinning of the piping. Therefore, the interfacing piping will continue to function as originally designed. The changes made by the ER affect extraction steam system piping to the fifth stage reheater, however, because the changes are intended to meet the requirements of the original design (component integrity, welding, pressure ratings, etc.), they will not degrade the integrity of the extraction steam system nor will they degrade or prevent actions described in the accident analysis. The changes do not increase the probability of occurrence or increase the consequences of malfunction of equipment important to safety or of a different type than previously evaluated in the UFSAR. The Technical Specifications are not affected, and the margin of safety remains unchanged. The piping installed by this design change meet ANSI B31.1 code requirements and is supported for the appropriate dead weight and thermal loads.

Serial Number: 97-074-NPE Document Evaluated: ER 97/0220-00

DESCRIPTION OF CHANGE: Approximately 100 feet of large bore vent and drain piping between the `B' moisture separator reheater (MSR), N1N35-B001B and the moisture separator shell drain tank `B', N1N35-A004B, including nozzle connections to these two components are experiencing wall thinning due to two phase flow. The selected carbon steel piping and equipment nozzles shall be replaced with seamed carbon steel piping having a 0.08" to 0.10" thick stainless steel clad. The piping being replaced is in the GGNS/erosion corrosion program (MS-41) as items 85, 318, 319, 320, 321, 322, 323, 347, 349 and 360 for the drain piping item 324 for the vent piping. The replacement piping and fittings have essentially the same form, fit, and function as the existing components since it has an equivalent internal diameter, pressure rating and minimum wall thickness as the piping that it is replacing.

REASON FOR CHANGE: MSR piping segments and nozzle connections are experiencing wall thinning due to erosion from two phase flow. Use of a clad pipe should reduce susceptibility of the replaced pipe segments to additional erosion without requiring extensive system reevaluation and potential configuration changes that are normally required for pipe material changes (i.e. chrome-moly pipe), and will not contribute toward future iron transport concerns.

SAFETY EVALUATION: As stated in UFSAR Section 10.2.2.1, the MSR system provides no safety function. The system analysis has shown that a failure of the system will not compromise any safety-related systems or prevent safe shutdown. The change is intended to reduce the possibility of a line failure of the MSR vents and drains due to erosion induced wall thinning of the piping. Therefore, the interfacing piping and connections to the MSR shell and MSR shell drain tank will continue to function as originally designed. The changes made by the ER affect nozzle connections to the MSR and MSR shell drain tank, however, because the changes are intended to meet the requirements of the original design (component integrity, welding, pressure ratings, etc.), they will not degrade the integrity of the MSR system nor will they degrade or prevent actions described in the accident analysis. The changes do not increase the probability of occurrence or increase the consequences of malfunction of equipment important to safety or of a different type than previously evaluated in the UFSAR. The Technical Specifications are not affected, and the margin of safety remain unchanged. The piping installed by this design change meet ANSI B31.1 code requirements and is supported for the appropriate dead weight and thermal loads.

Serial Number: 97-075-NPE Document Evaluated: ER 96/0936 Supp 0

DESCRIPTION OF CHANGE: This engineering request supplement provides all installation and startup activities for operation of bulk hydrogen and oxygen storage facilities in support of the future Hydrogen Water Chemistry (HWC) System at Grand Gulf Nuclear Station (GGNS), which will be completed by other supplements associated with this ER.

The bulk hydrogen and oxygen storage facilities are located on the plant north end of the Unit 2 cooling tower basin area. Report SL-5104, "Site Report, Hydrogen Water Chemistry System," dated January 16, 1997 provides an in depth discussion of the acceptability of this site with respect to NFPA, CGA, OSHA and EPRI requirements and recommendations. Based on the storage facility location (northern end of Unit 2 cooling tower basin, with hydrogen to the west of oxygen), the closet safety related structures are the SSW cooling tower (south-west) and the control building (south). The concrete walls for these structures were considered to be representative of all station safety related structures, and thus provide a bounding separation distance. Air intakes for the SSW pump room are on the south-western side of the structure and air intakes for the control building are on the north-western corner of the structure. Based on this configuration, the hydrogen storage facility is located approximately 810 feet from the nearest safety-related structure (SSW cooling tower) and approximately 865 feet from the nearest air pathway into a safety-related structure (SSW cooling tower pumphouse air intakes). The oxygen storage facility is located approximately 1060 feet from the nearest air pathway into a safety-related structure (SSW cooling tower pumphouse air intake). The location for these facilities was selected to minimize the potential consequence of tank rupture or excessive leakage to reactor safety, and is also in accordance with applicable NFPA and 29 CFR (OSHA) criteria for non-safety related structures and hazards. Note that catastrophic hydrogen storage facility failure could result in damage to structures and components similar to the results of a tornado. This includes venting of the enclosure building and loss of off-site power, which are currently considered in the UFSAR.

The bulk hydrogen storage facility will be capable of producing 200 scfm of gaseous hydrogen at a pressure of 225 psig. It will be comprised of a 20,000 gallon cryogenic, ASME Code stamped tank, two phase cryogenic pumps, ambient air vaporizers, an over-pressure protection system, excess flow check valves, permanent ASME certified gaseous storage tubes, and associated piping and instrumentation. Both the storage facility and the back-up supply are located in the conservative direction with respect to the prevailing wind direction at the site (i.e., from the southwest to the northeast; thereby blowing any potentially released gases away from the plant).

The bulk oxygen storage facility will be capable of producing 75 scfm of gaseous oxygen at a pressure of 120 psig. The liquid oxygen storage system will be comprised of a 9,000 gallon cryogenic, ASME Code stamped tank, ambient air vaporizers, an over-pressure protection system, excess flow check valves, and associated piping and instrumentation.

Hydrogen and oxygen supply piping, and the storage facility instrumentation cable conduit will be routed underground to a single to a point outside of the Unit 2 cooling tower basin, which will interface with conduit and piping to be installed via supplement 01 to this ER.

REASON FOR CHANGE: Operation of the HWC System is to reduce rates of intergranular stress corrosion cracking (IGSCC) in recirculation piping and reactor vessel internals. Reduction of IGSCC is achieved by injecting hydrogen in the reactor coolant thereby suppressing the formation of radiolytic oxygen, which reduces the electrochemical corrosion potential (ECP) to below -230 mV (SHE). Oxygen is injected into the offgas system as part of this process in order for it to combine with excess hydrogen prior to entering the offgas recombiners.

SAFETY EVALUATION: New UFSAR sections provide discussion of the HWC system storage facility components and their associated design considerations to limit impact on safe station operation. Revised UFSAR sections are provided to describe the revised location and capacity of the station bulk hydrogen storage facility (note that specific UFSAR sections discussing generator hydrogen will be addressed in the supplement connecting this new facility to the generator hydrogen supply) and provide general design documentation (i.e. HWC acronym listing, applicable NFPA codes). No changes to Technical Specifications are required or recommended as part of this ER supplement.

Review of the normal operational and failure conditions associated with the bulk gas storage facilities did not identify any unreviewed safety questions associated with the proposed design or impacts to other station systems. Due to potential uncontrolled external events resulting in storage facility failure (tornado, airplane) it is assumed that loss of offsite power resulting from a catastrophic storage facility failure would be considered at worst to be an event of moderate frequency, which is consistent with the currently evaluated probability of occurrence for a loss of off-site power accident. Installation and operation of the hydrogen and oxygen storage facilities will not increase accident or malfunction probabilities or consequences. It will not create any risk of a different type of accident or malfunction and does not reduce a margin of safety as described on the Technical Specifications Bases. Citing and design of these facilities, as proposed, meet the intent of the EPRI Guidelines for BWR HWC Systems, which has been generically evaluated by the NRC for all operating BWRs.

The existing tornado considerations for the enclosure building bound the overpressure wave associated with hydrogen storage facility explosion, such that there is no revised loading on the exposed enclosure building framing or its transmission to the containment dome. Therefore, this ER meets the requirements of 10 CFR 50.59 and may proceed in accordance with applicable station procedures.

Serial Number: 97-076-NPE Document Evaluated: ER 96/0056-00-00

DESCRIPTION OF CHANGE: Provide evaluation and details necessary for installation of a new, dedicated headset system for emergency communications between the Control Room and the Emergency Response Facilities (ERF). The headsets will be utilized as the primary means of communication between the Control Room and TSC, Backup TSC, OSC, Backup OSC and EOF locations during actual emergencies and will also be utilized during emergency drills for communication with personnel manning the Simulator. These headsets will utilize existing UPS power available in the Control Room, as well as in most other respective plant locations. Installation of headsets in the Backup OSC will require installation of a UPS feed to the room. This installation will require opening and closing of a penetration through a 3-hour fire barrier in accordance with TRM Section 6.2.8 and UFSAR Section 9.5.1.2.2.9 and Appendix 16A, Section 6.2.8, requirements. Headsets will be installed in the Control Room at the Plant Supervisor and Shift Superintendent desks and in other ERF locations as determined by EP. This installation will utilize spare, dedicated telephone lines between the specific locations to support upgraded communications.

REASON FOR CHANGE: The system currently used for ERF communications requires the use of telephone switches and handsets. The existing system has become somewhat unreliable and needs to be upgraded to a more current and more dependable configuration. The use of the dedicated, powered headset configurations and the dedicated phone lines will help to ensure system availability. Utilization of UPS power will also increase the reliability of the ERF communication loops.

SAFETY EVALUATION: System analysis has shown that failure of the Communications System will not compromise any Safety Related System nor will it prevent a safe shutdown of GGNS. The Communications System serves no safety-related function and has been evaluated by the Graded QA Program as Low Safety Significant. The modifications required to install the dedicated headsets will not degrade any system or its related performance. Opening and closing of the penetration for the installation of new cabling to the Backup OSC will not affect the penetration's ability to perform as previously evaluated. Appropriate penetration seal design requirements described in UFSAR Section 9.5.1.2.2.9 have been maintained and seal details ensure penetration integrity. Operational considerations have been provided for the affected penetration to comply with Technical Specification and UFSAR requirements.

Serial Number: 97-077-NPE Document Evaluated: ER 1997-0461-00

DESCRIPTION OF CHANGE: This change revises the criteria and adds a new configuration (2 inches of mineral fiber) for the sealing of internal conduits which penetrate fire rated barriers at GGNS. Changes to the internal conduit fire/smoke seal criteria and new seal configuration are based on full scale fire test conducted by Professional Loss Control, Inc. titled "Conduit Fire Test Program" and documented in PLC final report "Conduit Fire Protection Research Program" dated June 1, 1987 (commonly referred to as the" Wisconsin Electric Conduit Test". This fire test has been reviewed by the NRC and determined acceptable based on a Technical Evaluation Report dated May 12, 1989.

REASON FOR CHANGE: GGCR1997-0431-00 and MNCR1995-0274-00 identified missing internal conduit fire/smoke seals. The corrective action plan established to resolve this nonconformance included revising the existing internal conduit fire/smoke seal criteria based on the most recent fire test and reverification of all internal conduit fire/smoke seals in regulatory required fire barriers at GGNS. ER 1997-0461-00 revises the existing conduit fire/smoke seal criteria and adds a new conduit seal configuration. These changes will address the corrective action item in GGCR1997-0431-00 and MNCR1995-0274-00 pertaining to revision of internal conduit seal criteria to be used at GGNS.

SAFETY EVALUATION: Changes to the internal conduit fire/smoke seal criteria and new seal configuration are based on full scale 3-hour fire test which have been reviewed and accepted by the NRC. These test demonstrate that conduits sealed in accordance with criteria established by the test, have a 3-hour fire resistance rating and will not propagate fire and hot gases. Therefore, the fire resistance rating of the fire barrier is maintained and the ability to achieve and maintain safe shutdown conditions in the event of a fire, as presently analyzed in the UFSAR, has not been adversely affected.

Therefore, this change will not increase the probability or consequences of accidents or malfunction of equipment important to safety previously evaluated in the SAR. Also, this change will not create the possibility for an accident or malfunction of equipment important to safety of a different type than any previously evaluated in the SAR. "Fire Rated Assemblies" are not addressed by Technical Specifications (TS). "Fire Rated Assemblies" are addressed in the Technical Requirements Manual (TRM) Section 6.2.8. Changes made by this ER involve changes to internal conduit fire/smoke seal criteria and a new conduit seal configuration, which neither TS or TRM address. In addition, the 3-hour rating of the internal conduit seals are maintained. Therefore, there will be no change to TS or TRM or the Bases for any TS or TRM.

Serial Number: 97-078-NPE   Document Evaluated: ER 97/0762-00-00

DESCRIPTION OF CHANGE: The changes addressed are to the Domestic Water System (P66). They involve restoring the Water Treatment Building restroom as a functional facility. The restroom supply and drain lines still exist and have been capped and plugged per a disposition to MNCR 0134-94. The restoration would require the installation of a lavatory, water closet, and applicable plumbing fixtures.

REASON FOR CHANGE: The long distance from the chemistry Lab and the Radwaste Control Room to toilet facilities (located in the HPC area of the Control Building, El. 93') makes the reinstallation of the Water Treatment Building bathrooms a necessity.

SAFETY EVALUATION: The Domestic Water System has no safety-related function as defined in Section 3.2 of the FSAR. Failure of the system will not compromise any safety-related equipment or component and will not prevent safe shutdown of the plant. The reinstallation of the Water Treatment Building bathroom will in no way impact any of the accident analysis presented in the FSAR. No new failure modes are being created as a results of ER 97/0762-00-00, thus no possibility of an accident normal function of a different type than previously analyzed is possible. The piping and equipment involved is non-safety related and their reinstallation and operation is not and will not be required to mitigate the occurrence or consequence of a malfunction of equipment important to safety, therefore, they can have no adverse affect on safety. The margin of safety as defined for any Technical Specifications is not changed and no Technical Specifications are affected.

Serial Number: 97-079-NPE Document Evaluated: Temp Alt 97/0010

DESCRIPTION OF CHANGE: Due to repairs being performed on the Waste Neutralization Tank (SP21A002), this Temporary Alteration will provide an alternate method of maintaining low levels of sulfuric acid and water mixtures in the acid dike by connecting corrosive resistant hose from the Acid Dike Sump Pump (SP21C006A) discharge, and routing it to a vendor supplied tank. Additionally, this temp alt provides an alternate method of maintaining allowable dP's across the Carbon Filters (SP21D006A/B), by performing necessary backwashes and routing discharge to the Low Volume Wastewater Basin.

REASON FOR CHANGE: This temporary alteration will provide a contingency to facilitate carbon filter and acid dike wastewater disposal due to repairs currently being performed on the Waste Neutralization Tank (SP21A002).

SAFETY EVALUATION: The changes proposed by this temp alt will not compromise any existing safety-related system, structure, or component and will not prevent safe reactor shutdown. Primary and Secondary Containment Isolation Valves identified per Technical Specification Table TR3.6.1.3-1 and TR3.6.4.2-1 are in no way hindered from performing their intended function. This documents the temporary alteration to the P21 system design drawing identified on UFSAR Figure 9.2-011 indicating the Low Volume Waste Water Basin and P21F061 deadened valve, where a temporary hose connection will facilitate the carbon filter backwash. The implementation of this temp alt will not alter the design intent of the P21 system and no potentially radioactive water will be introduced into the system, nor will the temp alt create any Seismic II/I concern.

The Acid Dike is located in the yard next to the Unit 1 Warehouse and the connection for the Carbon Filter Backwash Discharge is located in the Water Treatment Building on the east side of carbon filter skid. Implementation of this temp alt will not affect, nor be located in the vicinity of, any safety related equipment which would be affected indirectly.

Serial Number: 97-080-PSE   Document Evaluated: Temp Alt 97-0013

DESCRIPTION OF CHANGE: Installation of a temporary discharge line from the Main Condenser Water Box Drain and Recirculation Pump 1N71-C002B to facilitate pumping down the water boxes. The change will allow pumping of circulating water to either of the circulating water pump pits and/or to storm drains to permit complete condenser water box drainage and condenser tube repairs during reduced power operations (i.e. capable of operating at reduced reactor power with one circulating water pump and one train of condenser water boxes).

REASON FOR CHANGE: The current configuration of the piping discharge from the condenser water box drainage and recirculation pumps, 1N71C002A and 1N71C002B does not permit complete condenser water box drainage during power operations without opening the cooling tower bypass valves due to the low discharge head of the pump(s) and the high static head required to discharge through the natural draft cooling tower nozzles. Opening of the bypass valve on one circulating water train causes the other train's hot side inventory to reverse flow through the open bypass valve and eliminates any significant cooling of the circulating water. The proposed temporary alteration will significantly reduce the static head required to pump the water box inventory by discharging to the circulating water pump pits, while allowing the remaining circulating water train to provide cooling.

SAFETY EVALUATION: The circulating water system, including the condenser water box drainage and recirculation subsystem, serves no safety functions as discussed in UFSAR Section 10.4.5. Loss of condenser vacuum and flooding events, analyzed in the UFSAR, envelope the occurrence of postulated events due to implementation of this temporary alteration. The location and routing for the temporary piping in the turbine building present no seismic II/I hazards. The temporary alteration does not alter or affect the condenser vacuum low setpoints.

Implementation of the temporary alteration will not create any new interface, or new failure mode which would affect any equipment, components or systems which are safety related or important to safety. Implementation of the temporary alteration will not create any new or affect existing functions which mitigate the consequences of a malfunction of equipment important to safety. Implementation of the temporary alteration will not change any function, parameter or operating characteristic which would create an interface with or affect any safety related components, equipment or systems. The location of the temporary piping outside of the turbine building does not pose any new failure modes for the generator isophase bus ducts in the general area.

Serial Number: 98-001-NPE Document Evaluated: ER 97/0644-00

DESCRIPTION OF CHANGE: The Division I and II Standby Diesel Generators have had a reoccurring leakage of the Dresser couplings on the Jacket Water piping. ER 97-0644-00-00 is replacing the Dresser couplings on the supply piping to the exhaust manifold water jacket and the turbocharger water supply piping for the Division I and Division II Diesel Generators with a flexible hose configuration. The new configuration has been evaluated for stresses due to thermal movement, vibratory and seismic loads. The piping and supports for this modification have been designed per ASME Section III, Class 3. The hose being installed by this ER is a metal reinforced polymer and is qualified for this application.

REASON FOR CHANGE: The Dresser couplings for both Division I and Division II being replaced have a reoccurring history of leaking (GGCR 1997-0727-00) and having a potential affect on the reliability of the Standby Diesel Generators. By replacing the leaking couplings with a flexible hose configuration, this change provides increased integrity and reliability of the Standby Diesel Generator Cooling Water System. The flexible hose configuration will absorb engine vibration and thermal movements and will not be susceptible to the leakage problem experienced by the Dresser Couplings and will therefore, provide enhanced engine performance.

SAFETY EVALUATION: ER 97/0644-00 replaces four leaking jacket water Dresser couplings on each of the Division I and Division II Diesels with a flexible hose configuration. By installation of the new flexible hose configurations, the problems with leaks at the Dresser couplings will be resolved for these locations.

A change to UFSAR Table 3.2-1 identifies the applicable Code governing the design and installation of the new flexible hose configuration. LDCR 97-102 has been initiated to add Note (zz10) for UFSAR Table 3.2-1 with the applicable information pertaining to the modification of the Jacket Water lines. These changes assure the classification of the Standby Diesel Generators will not be affected.

The original engine mounted piping of the jacket water system was designed and fabricated to the guidelines of the Diesel Engine Manufactures Association (DEMA) standards and the manufacturer's own standard procedure. The applicable Code for the jacket water piping is ANSI B31.1 and Quality Classification is Quality Group D (aa), Seismic Category I and Safety Class 3 (Ref. Table 3.2-1).

The ASME Section III is an approved Code for use in GGNS Safety Related piping system design and installation. The use of this Code assures that a component will meet its design requirements for design basis loading, like pressure, dead weight, thermal and seismic. The use of the ASME Section III Code allows the use of existing approved Safety Related programs for design and installation at GGNS. The use of the ASME Section III Code in lieu of ANSI B31.1 Code in the design and installation of the piping does not affect safety margins. The use of the ASME Code Section III assures the integrity of the piping system.

The fundamental frequency of the piping was above 33 Hz and the hose lengths were designed so that their natural frequencies did not coincide with those of the engine, so that no resonant condition would occur.

This changed portion of the jacket water will be subject to a pressure test at operating pressure and temperature. A hydrotest was not required. This is in keeping with the philosophy of the later ASME Section XI Code and which has been approved by the NRC (Ref. GGNS-M-489.0). Section XI Code replaced the hydrostatic test with system leakage test performed at nominal operating pressure based on reviews that leakage caused by hydrostatic test was not substantially different from leakage created at nominal operating pressure.

The operation of the Standby Diesel Generators will not be changed by this modification. System interface with other plant systems will not be directly or indirectly affected. This change will protect against leakage of this piping by providing a more suitable piping arrangement to withstand system temperature changes, thermal growth, seismic and engine vibration. The new flexible hoses are metal reinforced polyethylene-like material to accommodate system temperature expansions and contractions. There will be no increase in the probability of occurrence or the consequences of an accident or malfunction of equipment important to safety previously evaluated in the safety analysis report. This modification will not create the possibility of an accident or malfunction of a different type than any evaluated previously in the safety analysis report. The new configuration of the piping and hose is qualified by analysis and there will be no reduction in the margin of safety as defined in the bases for any Technical Specifications. The new flexible hose configurations help to ensure the stated function of the jacket water portion of the Standby Diesel Generators is not jeopardized.

Serial Number: 98-002-NPE Document Evaluated: ER 96/0203, SCN 96/0010A to GGNS-MS-02

DESCRIPTION OF CHANGE: This ER will install a lube oil sample-drain line with a valve for each of the Drywell Purge Compressors Q1E61B001A & B to facilitate oil sampling. The 1 inch dia. oil sample - drain line will replace the existing vendor supplied 1 inch dia. oil drain plug. The new line and valve will be designed to the classification of the compressor, safety related, ASME B&PV Code Section III, Class 2, seismic Category 1. The installation of this additional hardware on the compressor will not change the function of the compressor nor will it alter the integrity or the function of the compressor lube oil system.

REASON FOR CHANGE: Oil samples are required to be obtained before and after operating the Drywell Purge Compressors. These samples are taken from the reservoir by slightly loosening the drain plug momentarily (unscrewing but not completely removing the plug). If the plug was accidentally completely removed, the chance exists of losing a large amount of the lube oil from the compressor.

SAFETY EVALUATION: The compressor lubrication system supplies cool and filtered light oil to the bearings and gear. The oil reservoir is fabricated as part of the compressor base. The bottom of the reservoir is sloped toward the drain at the edge of the base. The drain plug will be replaced with a sample-drain line and isolation valve to ensure a good quality sample is obtained and to eliminate the risk of loosing or spilling oil during sampling. The new sample-drain line and associated pipe fittings and valve connections, except the threaded pipe cap, will be welded. The drain line and valve will be designed safety related and in accordance with ASME B&PV Code Section III, Class 2, Seismic Category 1, similar to the classification of the Drywell Purge Air Compressors. The addition of the sample-drain line is an enhancement to the present method of obtaining oil samples. The process of obtaining oil samples is neither addressed in the Technical Specification or the UFSAR and the sample-drain line will not impact the operability, function or structural integrity of the Drywell Purge Air Compressors.

Serial Number: 98-003-PSE Document Evaluated: Q1T51B007A/ WO 199472

DESCRIPTION OF CHANGE: The special instructions are written to flush the Q1T51B007A room cooler if the cooler is found in a degraded condition. The room cooler will be flushed with Betz DE 1178 a mild citric acid. The use of Betz De 1178 cleaning solution in Q1T51B007A room cooler is approved for use by ER 97/0300-00 and SE 970052 R00.

Room Cooler Q1T51B007A will be isolated from SSW by closing the inlet and outlet isolation valves. Fittings will be connected to the low points in the system to drain as much water as possible out of the piping/cooler into 55 gallon drums. Approximately 9 gallons of water is contained between the inlet and outlet valves of the room cooler. An additional 11 gallons of water will be added to the 55 gallon drum to provide enough water for the sandpiper pump to operate. sufficient acid (2 gallons) will be added to the water to make a 10% solution of acid to water (total volume of solution is 22 gallons). An injection and collection point will be established to flush the piping and cooling coil. The selection for injection and collection minimizes slow flow rates of acid solution and stagnant conditions. The acid flush will continue until one of the following criteria is met 1) dissolved iron content ceased to increase 2) dissolved copper content exceeds 700 ppm 3) a maximum of three hours flushing time. The acid solution will be drained from the piping/cooler into the 55 gallon drums using the low point drains and properly disposed (Reference NPDES Permit No. MS0029521). The room cooler will be valved back to SSW and the residual acid solution will be washed away.

REASON FOR CHANGE: Because of the fouling in the piping/tubing, the heat removal capability of the FPC-CU Room Cooler can be less than the requirements of MS-39.0. The room cooler coils will need to be cleaned to improve the heat transfer capabilities.

SAFETY EVALUATION: The process controls established by ER 97/0300-00 and SE 970052 R00 are (1) A 10% solution of the chemical solution Betz De 1178 (Based on volume of the established system boundary) is prepared, injected, and recirculated until one of the following process control limits are met (a) dissolved Iron level ceases to increase (b) dissolved copper reached 700 ppm (c) three hours maximum flush (2) Minimize low flow rates and stagnant conditions (3) Flush the affected system with water after the cleaning process (4) The total number of chemical cleanings is limited to 10 times without further evaluation of the available corrosion allowances. It is concluded that the use of the cleaning instructions does not increase the possibility or consequences of an accident evaluated in the SAR, does not create the possibility of a new accident or malfunction and does not reduce any margin of safety defined in the basis for any Technical Specification. There are no unreviewed safety questions or issues from using the cleaning instruction discussed in this evaluation.

The floor drains in the surrounding area will be intentionally covered during performance of this activity to ensure that acid solution does not inadvertently enter the floor drain system. The floor drain system serves to divert gross leakage away from affected equipment in the area. During performance of the flush activity, personnel will be continuously on the job. In the event of that a flood event were to occur, necessitating uncovering the floor drain, this can be accomplished by personnel at the job site.

Serial Number: 98-004-PSE Document Evaluated: Q1T51B004/ WO 199473

DESCRIPTION OF CHANGE: The special instructions are written to flush the Q1T51B004 room cooler if the cooler is found in a degraded condition. The room cooler will be flushed with Betz DE 1178 a mild citric acid. The use of Betz DE 1178 cleaning solution in Q1T51B004 room cooler is approved for use by ER 97/0300-00 and SE 970052 R00.

Room Cooler Q1T51B004 will be isolated from SSW by closing the inlet and outlet isolation valves. Fittings will be connected to the low points in the system to drain as much water as possible out of the piping/cooler into 55 gallon drums. Sufficient acid will be added to the water to make a 10% solution of acid to water. An injection and collection point will be established to flush the piping and cooling coil. The selection for injection and collection minimizes slow flow rates of acid solution and stagnant conditions. The acid flush will continue until one of the following criteria is met 1) dissolved iron content ceases to increase 2) dissolved copper content exceeds 700 ppm 3) a maximum of three hours flushing time. The acid solution will be drained from the piping/cooler into the 55 gallon drums using the low point drains. The room cooler will be valved back to SSW and the residual acid solution will be washed away.

REASON FOR CHANGE: FSAR Section 9.4.5.2.4 states that each safety-related pump room is provided with a full capacity fan-coil unit to prevent the room temperature from exceeding 150°F during pump operation. Because of the fouling in the piping/tubing, the heat removal capability of the RHR B Pump Room Cooler can be less than the requirements of MS-39.0. The room cooler coils will need to be cleaned to improve the heat transfer capabilities.

SAFETY EVALUATION: A chemical cleaning process using Betz DE 1178 has been selected to improve the cleanliness of the cooling coils for Q1T51B004 RHR B Pump Room Cooler which will enhance the heat transfer capability of the unit. The materials of construction was reviewed and evaluated for compatibility under Safety Evaluation 970052 R00. The instruction for cleaning the room cooler implements the controls given by ER 97/0300-00 and SE 970052 R00. This safety evaluation evaluates the method of acid flushing and ensuring the process controls are implemented. It is concluded that the use of the cleaning instructions does not increase the possibility or consequences of an accident evaluated in the SAR, does not create the possibility of a new accident or malfunction and does not reduce any margin of safety defined in the basis for any Technical Specification. There are no unreviewed safety questions or issues from using the cleaning instruction discussed in this evaluation.

The floor drains in the surrounding area will be intentionally covered during performance of this activity to ensure that acid solution does not inadvertently enter the floor drain system. The floor drain system serves to divert gross leakage away from affected equipment in the are.

During performance of the flush activity, personnel will be continuously on the job. In the event of that a flood event were to occur, necessitating uncovering the floor drain, this can be accomplished by personnel at the job site. Per FSAR Section 6.3.1.1.3, the ECCS rooms are constructed to be water tight to protect against mass flooding of redundant ECCS pumps therefore temporary covering of the floor drain in the immediate area will not increase the consequences of a flooding event. The total volume of the acid solution is 34 gallons (31 gallons of water and 3.12 gallons of acid), therefore the flood potential from the hoses used in the cleaning process breaking and spilling the total volume of cleaning solution is insignificant because of the small volume of solution.

Serial Number: 98-005-NPE Document Evaluated: ER 96/0383-01-00

DESCRIPTION OF CHANGE: This ER will replace the existing 1" dia. carbon steel piping and associated components between rad. monitor 1D17J005 and 24"-HBC-82 (with the exception of 3/4" root valve & piping) with stainless steel material. The ER will also add flushing connections to both the supply and return lines. Replacement of the subject piping components with stainless steel material is expected minimize the susceptibility to fouling and improve the reliability of the monitor.

REASON FOR CHANGE: The 1" dia. SSW supply and return carbon steel piping for liquid radiation monitor 1D17J005 have experienced fouling and reduced flow to and from the monitor and 24"-HBC-82. This ER will replace the existing 1" dia. carbon steel piping and associated components between radiation monitor 1D17J005 and 24"-HBC-82 (with the exception of 3/4" root valve & piping) with stainless steel materials. The ER will also add flushing connections to both the supply and return lines. Replacement of the subject piping components with stainless steel material is expected to minimize the susceptibility to fouling and maintain adequate piping flow.

SAFETY EVALUATION: The installation of replacement 1" diameter stainless steel piping and valve and the addition of flushing connections will not impact plant and function of the interfacing radiation monitor 1D17J005, which monitors the radiation level of effluent streams, is not affected by this change which will improve the reliability of the monitor. An increase in radiation level is indicative of equipment malfunction that could result in a radioactive release. These radiation levels are measured by SSW monitor 1D17005 and recorded in the Radwaste Control Room replacing the supply and return piping to the monitor, including valve P41F198A, with similar components of stainless steel material, and addition of flushing connections, will not affect current plant operation or change the operation or function of the Standby Service Water System (P41) or the Radiation Monitoring System (D17) or any of the interfacing systems during normal or accident conditions. This change will not cause the system to be operated in a new manner and will improve the performance of the piping system by reducing fouling. The piping inherent safety margin has been maintained and the structural integrity of the piping is assured. The replacement piping and existing supports will still satisfy the requirements of ASME Code, Section III, Class 3, Seismic Category 1 and meet the support span requirements of M-18. This change will not prevent any piping component from maintaining the pressure boundary of the fluid it carries. The operation reliability of Standby Service Water as defined in the UFSAR has not changed as a result of this ER. The replacement of small piping and installation of flushing connections installed by this ER will have no adverse effect on the functionality of systems required to mitigate the consequence of postulated accidents or malfunction of equipment important to safety. Furthermore, the response of the P41, D17 systems to accidents or malfunction of equipment important to safety as described in the UFSAR remain unchanged and no new failure modes outside those previously described in the UFSAR will be created. The change will not create the possibility or increase the probability of an accident or a malfunction of equipment important to safety previously evaluated in the SAR or of a different type than any previously evaluated in the SAR. The replacement of small piping and their associated components, including addition of flushing connections by this ER, will not change the operability of the Standby Service Water System as defined by the bases in Section 3.7.1 of the Technical Specifications, and therefore, the margin of safety is not reduced.

Serial Number: 98-006-NPE Document Evaluated: ER 97/0958-00-00/ GGCR1997-0343-01

DESCRIPTION OF CHANGE: Raise the retract permissive setpoint for the Neutron Monitoring System Source Range Monitors (1C51K600A-F). Also specify an allowable operating range for the setpoint.

REASON FOR CHANGE: The SRM detector retract permissive setpoint specified in surveillance procedure 06-IC-1C51-V-0003 is not in accordance with the nominal setpoint specified in GE Design Specification Data Sheet 22A3739AE or the SRM retract permissive functional requirement stated in TRM Table TR3.3.2.1-2.

SAFETY EVALUATION: ER 97/0958 approves raising the Neutron Monitoring System Source Range Monitors retract permissive setpoint. Changing the setpoint will not impact the operation of the SRMs or affect their ability to perform their intended design functions. Adequate core monitoring must be established before SRMs can be withdrawn. Changing the SRM retract permissive setpoint will not impact core monitoring at any power level. The retract permissive setpoint is not specified in the GGNS Technical Specifications but the functional requirement for this value is stated in TRM Table 3.3.3.1-2. The functional requirement for this setpoint states that this value must remain greater than 100 counts per second in order to satisfy its retract permissive design function. The change approved by ER 97/0958 will increase the setpoint to a value greater than 100 cps and therefore maintain the setpoint in a conservative direction with respect to the process of SRM neutron flux monitoring. The SRMs have no safety function and are not assumed to function during any design basis accident or transient analysis. UFSAR Table 7.6-3 will be updated per LDCR 98-001 in order to clarify the normal setpoint given for the SRM detector retract permissive. Raising the SRM retract permissive setpoint in the conservative direction does not create any Unreviewed Safety Question or introduce any activity that will adversely impact the safe operation of the plant.

Serial Number: 98-007-NPE   Document Evaluated: ER 96/0222-00-00

DESCRIPTION OF CHANGE: ER 96/0222 addresses modifications to C11 - Control Rod Drive (CRD) System and C71 - Reactor Protection System (RPS) to re-arrange the piping and valve configuration at the backup scram valves and the physical layout of the backup scram logic. The present arrangement of the backup scram valves and associated piping is such that actuation of either C11-F110A or C11-F110B will result in a plant scram. As modified by ER 96/0222 the arrangement of the backup scram valves and associated piping will be such that actuation of all the backup scram valves will be necessary to accomplish the backup scram function. C11-F110A will remain in its present location as a 3-way valve, C11-F110B will become a 2-way valve and will be relocated to the vent port of C11-F110A, and C11-F110C will be installed as a 3-way valve, replacing the existing C11-F111 check valve. The C11-F111 check valve is a by-pass around C11-F110A which presently allows for venting the scram air header through C11-F110B when C11-F110A is energized. The post-modification configuration will require energization of C11-F110A, C11-F110B, and C11-F110C to block the instrument air header and vent the scram air header. The physical layout of the electrical logic associated with the backup scram valves will be modified such that the DC "island" which is presently unmonitored for ground faults will be moved into the control room where ground faults and water intrusion are less likely. ER 96/0222 will also provide for control room annunciation upon energization of any backup scram valve.

REASON FOR CHANGE: The Backup Scram Valves are associated with Scrams 82 (11/1/94) and 83 (3/16/95). Scram 82 occurred during performance of a surveillance when a planned half-scram occurred in which the backup scram valves inadvertently energized due to a low resistance connection to ground in the ground fault monitoring circuitry. Scram 83 occurred during performance of a surveillance when a planned half-scram occurred in which the backup scram valves again inadvertently energized this time due to water intrusion and a pinched wire. In addition, water intrusion was found again at the backup scram valves during RFO8 rebuild of the backup scram valves (MNCR 0047-95) and in 1996 the Scram contactor auxiliary contacts that control the backup scram valves were found loose (MNCR 0071-96).

As presently configured, it is possible to have half of the logic for actuation of the backup scram valves completed, bypassed or short-circuited without any notification of the half tripped condition. Upon performance of a surveillance in which a half scram will be entered, the backup scram valves would become actuated inadvertently. Scrams 82 and 83 occurred when unknown ground fault conditions were present and a half-tripped condition was introduced during performance of surveillances. MNCRs 0047-95 and 0071-96 addressed conditions which could have lead to a half-scram condition that would be unknown to control room operators.

SAFETY EVALUATION: It is concluded that no unreviewed safety questions exist and no changes to the GGNS Technical Specifications are required for the modifications to be performed by ER 96/0222-00-R00.

Additionally, the function of the backup scram valves remains non-safety related; the probability of a reactor scram due to inadvertent energization of the backup scram valves is reduced (which also reduces the probability of unnecessary challenges to safety systems), and failure of the backup scram valves in any mode is not detrimental to plant safety and cannot prevent any safety function from occurring. All system design criteria for C11 - CRD and C71-RPS are maintained. The small increases in the Division 1 and Division 2 battery loading due to larger solenoid valves and new relays have been evaluated to have no adverse affect on battery capacity or availability.

Serial Number: 98-008-PSE   Document Evaluated: WO 00184798

DESCRIPTION OF CHANGE: The LPCS Pump Room Cooler Q1T51B002 failed to meet the heat removal requirements of MS-39.0 as determined by the thermal performance test conducted on 3/26/97, therefore the room cooler will need to be cleaned to increase its heat removal capability. The room cooler will be flushed with Betz DE 1178 a mild citric acid. Betz DE 1178 cleaning solution is used to flush the deposits out of the cooler/piping thus restoring cooler efficiency. The use of Betz DE 1178 cleaning solution in Q1T51B002 room cooler is approved for use by ER 97/0271 and SE 970023 R00.

Room Cooler Q1T51B002 will be isolated from SSW by closing the inlet and outlet isolation valves. Fittings will be connected to the low points in the system to drain as much water as possible out of the piping/cooler into 55 gallon drums. Sufficient acid will be added to the water to make a 10% solution of acid to water. An injection and collection point will be established to flush the piping and cooling coil. The selection for injection and collection minimizes slow flow rates of acid solution and stagnant conditions. The acid flush will continue until one of the follow criteria is met 1) dissolved iron content ceases to increase 2) dissolved copper content exceeds 700 ppm 3) a maximum of three hours flushing time. The acid solution will be drained from the piping/cooler using the low point drains into the 55 gallon drums for disposal. The room cooler will be valved back to SSW (Steps 7.4.14 and 7.4.15) and the residual acid solution will be washed away (Set 7.4.16).

REASON FOR CHANGE: FSAR Section 9.4.5.2.4 states that each safety-related pump room is provided with a full capacity fan-coil unit to prevent the room temperature from exceeding 150°F during pump operation. Because of the fouling in the piping/tubing, the heat removal capability of the LPCS Pump Room Cooler is less than the requirements of MS-39.0. The room cooler coils will be cleaned to improved the heat transfer capabilities.

SAFETY EVALUATION: A chemical cleaning process using Betz DE 1178 has been selected to improve the cleanliness of the cooling coils for Q1T51B002 LPCS Pump Room Cooler which will enhance the heat transfer capability of the unit. The materials of construction were reviewed and evaluated for compatibility under Safety Evaluation 970023 R00. The process controls established by SE 970023 R00 are (1) A 10% solution of the chemical solution Betz DE 1178 (Based on volume of the established system boundary) is prepared, injected, and recirculated until one of the following process control limits are met (a) dissolved Iron level ceases to increase (b) dissolved copper reaches 700 ppm (c) three hours maximum flush (2) Minimize low flow rates and stagnant conditions (3) Flush the affected system with water after the cleaning process (4) The total number of chemical cleanings is limited to 10 times without further evaluation of the available corrosion allowances. The instruction for cleaning the room cooler implements the controls given by ER 97/0271 and SE 970023 R00. This safety evaluation evaluates the method of acid flushing and ensures the process controls are implemented. It is concluded that the use of the cleaning instructions does not increase the possibility of occurrence or consequences of an accident evaluated in the SAR, does not increase the probability of occurrence or consequences of a malfunction of equipment important to safety previously evaluated in the SAR, does not create the possibility of a new accident or malfunction, and does not reduce any margin of safety defined in the basis for any Technical Specification. There are no unreviewed safety questions or issues from using the cleaning instructions discussed in this evaluation.

The floor drains in the surrounding area will be intentionally covered during performance of this activity to ensure that acid solution does not inadvertently enter the floor drain system. The floor drain system serves to divert gross leakage away from affected equipment in the area. During performance of the flush activity, personnel will be continuously on the job. In the event that a flood event were to occur, necessitating uncovering the floor drain, this can be accomplished by personnel at the job site. Per FSAR Section 6.3.1.1.3, the ECCS room are constructed to be water tight to protect against mass flooding of redundant ECCS pumps therefore temporary covering of two floor drains will not increase the consequences of a flooding event. The total volume of the acid solution is 27.5 gallons (25 gallons of water and 2.5 gallons of acid), therefore the flood potential from the hoses used in the cleaning process breaking and spilling the total volume of cleaning solution is insignificant because of the small volume of solution.

Serial Number: 98-009-NPE Document Evaluated: TCN 117 to 03-1-01-1, TCN 51 to 03-01-01-2, TCN 13 to 04-1-01-N11-1, and TCN 35 to 04-1-01-N23-1

DESCRIPTION OF CHANGE: This Safety Evaluation addresses the Startup and Operation of GGNS without the Moisture Separator Reheaters' (MSR-1N35B001A/B) 2nd Stage Reheaters being placed in service. Operation in this configuration is expected to occur only during the last three months of Cycle 9. This evaluation includes an analysis of the decrease in feedwater temperature due to the reheater isolation and the effect on the bypass power, high-and low-power setpoints associated with first stage turbine pressure. The current degraded condition of MSR-A (i.e., multiple tube ruptures) is also considered in this evaluation.

REASON FOR CHANGE: The 2nd Stage Reheater tubes in MSR 1N35B001A have known leaks which are affecting thermal efficiency and can pose a potential risk of additional MSR tube damage and Main Turbine damage associated with MSR outlet temperature differentials. It is undesirable to operate the MSR's in this condition until the ruptured tubes are repaired.

SAFETY EVALUATION: This evaluation concludes that the operation of GGNS at reduced power is acceptable and does not constitute an unreviewed safety question. The reload analysis credits both the bypass power, high- and low-power setpoints for mitigating the consequences of transients and accidents. The effect of the proposed change is to increase HP turbine first stage pressure for a given reactor power level which will initiate these trips at lower reactor powers than the design values. The high- and low-power setpoints are credited for limiting rod withdrawals via the RWL system. The EOC-RPT bypass function is also based on the first stage turbine pressure. Initiation of these limitations at slightly lower power levels is acceptable for protecting the core from transients. Above the bypass power setpoint the consequences of fast closure of the turbine stop/control valves are mitigated by a direct scram on valve closure (via EHC fluid pressure) and a RPT from a fast speed. Initiation of this direct scram at lower powers is conservative with respect to the current core limits. The Technical Specification analytical limit requires the rod pattern controller (RPC) to be in service at powers up to 10% of rated. The RPC is credited in the reload analysis with minimizing the rod worths associated with the control rod drop accident (CRDA). A reduction in the power at which the RPC is disabled is non-conservative; however, this change is found to be acceptable for the following reasons.

- The current GGNS administrative control is to keep the RPC system in service up to the low-power setpoint, which is well above the Technical Specification 10% requirement.

- The limiting control rod drop accidents were determined to be at power levels at or near zero percent - significantly less than the affected setpoint.

- The impact of this proposed change is expected to decrease at reduced reactor power levels.

It is therefore concluded that the Cycle 9 reload analysis bounds the Technical Specification requirements for the proposed plant operational configuration and that no changes to these requirements are needed. The plant protection systems will still meet all the requirements in the GGNS Technical Specifications and GGNS UFSAR.

Operation at a reduced feedwater temperature could occur due to an event in which certain stage(s) or string(s) or individual heaters become inoperable (i.e., isolation of MSRs). Loss of feedwater heating from the highest pressure heaters would result in the highest temperature reduction. FSAR Chapter 15 App. B justifies operation for GGNS with rated feedwater temperatures between 420°F and 370°F due to inoperable feedwater heater(s).

For the Feedwater Heater(s) out of service (FWHOS) in this operating condition, the RPS scram function on the turbine stop valve closure and the turbine control valve closure assures that the scram bypass is consistent with 40% of rated power FWHOS conditions. No operating limit MCPR changes are needed for operation with feedwater temperatures between 420°F and 370°F.

Engineering evaluation of the proposed operational configuration resulted in the following conclusions:

(a) The abnormal operating transients in FSAR Chapter 15 were reevaluated at rated feedwater temperature of 370°F to determine the required operating MCPR limits for FWHOS operation. The results show that no operating limit MCPR change is required for operation between 420°F and 370°F rated feedwater temperature.

(b) It was determined that the fuel mechanical limits are met during FWHOS operation under steady state and anticipated operational occurrences.

(c) The Loss-of-Coolant Accident (LOCA) and containment response as described in FSAR Chapter 6 were reevaluated for FWHOS operation. It was concluded that the normal feedwater temperature analysis adequately bounds those events with FWHOS conditions.

(d) Fuel integrity and thermal-hydraulic stability was previously evaluated with respect to General Design Criteria 12 (10CFR50, Appendix A). It is shown that the FWHOS operation satisfies the stability criterion and fuel integrity is not compromised.

(e) The effect of acoustic and flow induced loads on the reactor shroud, shroud support and jet pumps were analyzed to show that the design limits are not exceeded. The effect of FWHOS on feedwater nozzle and sparager fatigue usage factor was previously evaluated and is not effected. It was found that the increase fatigue usage on the feedwater nozzle adequately meets the acceptance criterion for unlimited operation up to 40 years.

(f) The turbine stop valve and the turbine control valve scram bypass setpoints in the Reactor Protection System are less than or equal to 40% rated power with reduced feedwater temperature conditions.

Additional engineering evaluations included feedwater system piping, annulus pressurization load analysis, and Anticipated Transients Without Scram (ATWS) to justify FWHOS operation. The feedwater heater temperature correction (estimated in the Appendix to this Safety Evaluation) will correspond to less than 10°F which is well within the 50°F limits of FWHOS operation. These evaluations concluded that the standard operation design is adequate for FWHOS operation.

Serial Number: 98-010-NPE Document Evaluated: ER 96/0885-00-00

DESCRIPTION OF CHANGE: This document evaluates ER 96/0885-00-00, which authorizes the permanent installation of the Leading Edge Flow Monitor (LEFM) system hardware originally installed by Temporary Alteration (T/A) 96-0008. Computer software and use of the LEFM system to provide feedwater flow input is addressed separately in Safety Evaluation SE 96-0030-00 and is not further evaluated in this safety evaluation. In addition to the changes made by T/A 96-0008, the ER replaces PVC jacket cables with IEEE 383 fire rated cables, relocates the LEFM electronic unit to a more suitable environment, installs rigid conduit and related supports in place of flex conduit, and documents the permanent transducer problem installation evaluated in ER 97/0026.

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

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

Serial Number: 98-011-NPE Document Evaluated: ER 96/0184-00-00

DESCRIPTION OF CHANGE: The changes addressed in this EER are to the G17 Liquid Radwaste System (installation of a local sample sink) and the P33 Process Sampling System (re-routing a sample drain from the chemical drain system to the equipment drain system).

REASON FOR CHANGE: EER 95/6189 identified 3/8" tubing that supplies a sample of liquid radwaste from the G17 C001, G17 C002, and G17 C008 recirculation piping. This tubing has experienced frequent clogging due to the long length of tubing between the sample points and sample collection panel SH22P127 (Radwaste Building Sample Station). System Engineering proposed as a solution to this problem the installation of local sample sinks at the 93' elevation, in Area 29 of the Radwaste Building. The shortened tubing run should eliminate the current problems experienced with the clogging of the tubing (this is identified as an Operator Work Around). EER 94/6164 (and ER 96/0298-00-00) identified effluent from dissolved oxygen meters for the Reactor Feed Pump Discharge (IP33-N012), Condensate Demineralizers Combined Effluent (1P33-N009), and Condensate Pump Discharge (1P33-N041) that is being routed to the miscellaneous waste chemical drain system. The effluent from the oxygen meters has no chemicals added to it and is recoverable water. System Engineering performed a walkdown of the problem and proposed as a solution routing the effluent for the oxygen meters to the equipment drain system (an equipment drain hub is located approximately 14" from the chemical drain hub that currently receives the effluent). System Engineering estimates that re-routing the effluent from these meters would cut down on the year to date discharges by 145,000 - 160,000 gallons of water, reduce the load on the P21 Make-up Water System, and reduce radwaste discharge in the range of 202, 356 to 279,444 gallons yearly.

SAFETY EVALUATION: The G17 Liquid Radwaste and P33 Process Sampling Systems have no safety-related classification as defined in Section 3.2 of the FSAR. Failure of the systems will not compromise any safety-related equipment or component and will not prevent safe shutdown of the plant. The modifications made by ER 96/0184-00-00 will in no way impact any of the accident analyses presented in the FSAR. No new failure modes are being crated, thus no possibility of an accident or malfunction of a different type than previously analyzed is possible. Failure of the systems will not compromise any safety-related system or component and will not prevent safe reactor shutdown, thus the margin of safety will not be reduced.

Serial Number: 98-012-NPE Document Evaluated: ER 96/0010-00-00

DESCRIPTION OF CHANGE: The ER provides instructions for installing the Reliance electric spare motor on SSW Pump "A".

REASON FOR CHANGE: The existing motor for the SSW "A" pump is a General Electric 5K6347XC135A (992C937AB) 1250 HP 1200 RPM 3Phase 60 Hz 4160 Volts which has differences in the cooling water connections and other slight differences from the Reliance Electric spare motor.

SAFETY EVALUATION: The spare motor was purchased as a direct replacement for the existing motor on Standby Service Water pump "A". The slight differences in bearing cooling water line connections require a P&ID change affecting UFSAR Figure 9.2-002. The other slight differences in orientation, service factor, insulation class, dimensions, and added intake air filters do not affect the performance of the motor/pump combination. These differences were adequately resolved in the purchasing process as documented in DMR 93/0065 and in the body of the ER providing the installation instructions. The spare motor will perform all design functions of the original motor. No changes to the basic system or automatic features are made.

Serial Number: 98-013-PSE   Document Evaluated: Temp Alt 98-0002

DESCRIPTION OF CHANGE: Temporary Alteration (Temp Alt) 98-0002 will be used to install a blind flange in place of the existing Radial Well pressure relief valve SP47F025J. The relief valve is located on the discharge line from Radial Wells #4. The blind flange will later be removed and the valve replaced with a like-for-like replacement valve. The new blind flange will mount on the existing pipe nozzle for the relief valve and will connect to the existing piping without requiring piping modifications.

The temporary blind flange is carbon steel (SA-105), commensurate with the construction of the radial well discharge piping class.

The temp alt can be worked during any operational condition as long as sufficient Plant Service Water capacity is available to support the plant condition which exists during the period of implementation, which is in agreement with UFSAR Section 9.2.10.2, System Description, which states: "During normal operation, as many wells and pumps as required will be operating to meet the plant demand."

REASON FOR CHANGE: Relief valve SP47F025J has developed a significant leak through the nozzle to body connection. A suitable replacement valve cannot be located as the existing valve body is obsolete. Procurement of a new valve has a lead time that would require the associated radial well pump to be kept out of service for an extended period of time. Operation of the pump is needed to support normal plant operation.

SAFETY EVALUATION: This temporary alteration will not adversely affect the ability of the PSW system to supply cooling water to the ADHRS heat exchanger nor will the change adversely affect the ability of the PSW system to supply makeup water to the SSW cooling tower basins. The change will not adversely affect the performance of any of the safety system HVAC systems during normal operation. During accident conditions, the safety related systems serviced by the PSW system are automatically supplied by the Standby Service Water system. the changes will not negatively affect the performance of any system governed by the Technical Specifications.

There are no "accidents" evaluated in the SAR related to PSW system failures. The change will not degrade any important to safety system, component, or structure nor will the change degrade or prevent actions described in the SAR accident analysis. No new failure modes are created and there is no increase in previously identified failure modes for equipment which is important to safety. The probability of the failure of the modified PSW piping will be no higher than that for the existing PSW piping and pumps.

Serial Number: 98-014-NPE Document Evaluated: LDCR 97-107

DESCRIPTION OF CHANGE: Feedwater (FW) is distributed through spargers in the reactor pressure vessel (RPV) that deliver the flow evenly to ensure proper jet pump subcooling and help maintain proper core power distribution. an essential part of the sparger is the thermal sleeve, which projects into the RPV nozzle bore and is intended to prevent the impingement of cold FW on the hot nozzle surface. This surface is usually heated to reactor water temperature by the returning water from the steam separators and steam dryers. However, BWRs have experienced bypass leakage past the thermal sleeves allowing relatively cold FW to impinge on the hot nozzles. The FW, when heated by extraction steam from the main turbine, is typically about 100°F to 200°F colder than the reactor water. When the FW heaters are not inservice, as during startups and shutdowns, the differential could be equal to or greater than 400°F. Bypass leakage through a loose thermal sleeve causes a fluctuation (at times severe) in the metal temperature of the FW nozzle and has resulted in metal fatigue and crack initiation at several domestic BWR facilities in the 1970's. The cracks were then driven deeper by the larger temperature and pressure cycles associated with startups, shutdowns, and certain operational transients.

The cracking was attributed to fatigue, and significant analysis was performed by General Electric (GE) to identify the problem and develop a solution [1]. The cracking problems resulted in the Nuclear Regulatory Commission (NRC) issuing NUREG-0619 [2] IN 1980. The NUREG described the cracking phenomena, identified fixes, and provided examination and plant hardware modification recommendations that were based on extensive testing, analysis, and examination technology available at that time. The NRC amendment NUREG-0619 with Generic Letter (GL) 81-11 in 1981, allowing for plant specific analysis in lieu of hardware modifications. Together, these documents specified NRC-endorsed actions to mitigate the initiation and propagation of FW nozzle cracking and provided examination guidelines to identify the onset of any further cracking.

As a result of these issues and commitments to implement the guidance of NUREG-0619, UFSAR Section 5.3.3.1.4.5.3 and GGNS-M-489.1 [5] requires:

- ultrasonic examination of each FW nozzle's inner radius, bore, and safe-end region during every second refuel outage, and
- removal of one FW sparger and performance of a dye-penetrate (PT) examination of that nozzle's bore and inner radius, and accessible areas of remaining FW nozzles every nine refueling outages (RFO9) or 135 startup/shutdown cycles, which ever occurs first, and
- performance of a visual inspection of the flow holes (spray nozzles) and welds in the sparage arms and sparger tees every fourth refueling outage.

This evaluation provides for a change to the existing examination requirements described in the UFSAR based on up-to-date field experience, industry initiatives, enhanced UT technology, and GGNS specific fracture mechanics analysis [4]. The revised examination requirements are:

- The FW nozzle inner radius and bore regions will be UT examined using either automated or manual techniques and procedures that have been demonstrated to accurately detect and characterize flaws in the area of interest with a depth of 0.250 inches. The examination interval shall be set at 1/3 of the critical flow life, as explained further in this evaluation, for Zones[1] 1 and 2 of the nozzle. For Zone 3, the examination interval may be twice the time specified for Zones 1 and 2. The examination interval commences after a complete set of baseline examinations have been completed using UT capabilities as described here.

  No change is made to examinations performed on the safe-end to nozzle welds. These locations are examined at frequencies specified by ASME Section XI.

- PT examination one nozzle bore and inner radius, and accessible areas of the other nozzles every nine refueling outages or 135 startup/shutdown cycles, which ever occurs first, is deleted.

  The visual inspection of the sparger welds and flow holes (spray nozzles) is retained.

The proposed change eliminates the PT examination and increases the time between the UT examination of the FW nozzles. The UT technology available today is capable of detecting the flaws that were assumed undetectable when the NUREG was issued. Therefore eliminating the need to perform the PT of the nozzle inner surface is acceptable since UT examinations will perform the same function. The UT examination frequency is being changed to be consistent with the BWROG recommendation except that the criteria applied to automated techniques is also applied to the manual technique. This examination frequency combined with the improved UT will identify flaws before they exceed 1/3 of the critical flaw size as determined by the GGNS site specific fracture mechanics evaluation.

REASON FOR CHANGE: At the time NUREG-0619 was issued, there was not a complete understanding of the technical issues associated with feedwater nozzle cracking and the reliability of ultrasonic examinations. As such, GGNS opted for a conservative course and adopted the augmented examination plan recommended by the NUREG. The NUREG indicated that the confidence level in the UT process available at the time the NUREG was issued (November 1980) was unacceptably low. Therefore, the augmented examination program required both UT and PT examinations as well as visual inspections based on the technology available at the time. The

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[1] Feedwater Nozzle Zones are defined in Reference 3 and are based on susceptibility to cracking and crack growth rates. Form 316.1 Revision 9

USNRC noted that continued reliance on dye penetrant (PT) examinations is not acceptable because of unnecessary radiation exposure to examination personnel.

Revising the examination requirements is acceptable because:

1. GGNS FW nozzle and thermal sleeve design provides high reliability for preventing FW bypass flow that may cause thermal cycling of the FW nozzle.

2. The GGNS FW nozzles do not contain stainless steel classing.

3. Current UT techniques are capable of detecting small flaws that have been demonstrated by site specific fracture mechanics analysis to be benign.

4. The revised examination frequency will ensure that if flaws were to initiate, that they would be detected before they reached 1/3 of the maximum permissible flaw size.

5. To pursue disassembly for performing the PT examination is a significant hardship that may result in degradation of the reactor vessel nozzle and associated FW nozzle hardware.

6. The FW sparager disassembly and PT examinations would require a significant radiation exposure to personnel.

SAFETY EVALUATION: GGNS is changing the existing examination requirements for feedwater nozzle examinations. The alternative examination requirements increases the time between UT examinations (currently all nozzles are examined every other refueling outage based on BWROG recommendations. Also, the augmented PT examination of the feedwater nozzles is eliminated.

The alternative examinations do not affect current Technical Specifications or their basis. GGNS Technical Specifications (TS) do not specifically discuss reactor coolant pressure boundary (RCPB) integrity. However, the bases for TS 3.4.5, RCS Operational Leakage does indicate that the limits for unidentified leakage allows time for corrective action before the RCPB could be significantly compromised. The proposed changes sod not affect the ability to detect or act upon detection of unidentified leakage. Additionally, TRM 6.4.2, Structural Integrity indicates that the structural integrity of ASME Class 1 2, and 3 components shall be maintained in accordance with the inservice inspection program. The inference to the "inservice inspection program" is understood to mean ASME Section XI. The FW nozzle is part of an ASME Class 1 component and therefore is required to be examined in accordance with ASME Section XI, IWB-2500. Pursuant to these requirements, the subject areas of the FW nozzle would be ultrasonically examined once every ten years irrespective of flaw behavior and crack growth rates. The alternative examinations evaluated herein are in excess of the requirements of ASME Section XI and will require examinations that have been proven by demonstration and will be performed at frequencies based on GGNS specific fracture mechanics analysis. Additionally, before the examination frequency begins, all six nozzles must be baselined using the UT processes described within. The changes proposed in this evaluation do not reduce the margin of safety as defined by any TS basis.

Because the design and operating characteristics of GGNS are not being changed by this evaluation, the probability of occurrence of an accident previously evaluated, and of a malfunction of equipment important to safety previously evaluated, are not increased. The performance of inspections/examinations do not affect the probability of occurrence, but do provide measures for the detection of laws in the unlikely event one were to initiate. The flaws discussed in this evaluation are hypothetically assumed to exist at the nozzle blend radius, nozzle bore, or safe end region of a feedwater nozzle. Failure at one of these locations would be considered loss of the reactor coolant pressure boundary and results in a loss of coolant accident (LOCA). In accordance with 10 CFR 50.46(c)(1), LOCA's are postulated accidents that would result from loss of the reactor coolant, at a rate in excess of the capability of the reactor coolant makeup system, from breaks in pipes in the reactor coolant pressure boundary up to an including a break equivalent in size to the double-ended rupture of the largest pipe in the reactor coolant system. GGNS loss of coolant accidents are described in UFSAR Sections 15.6.5, 6.2 and 6.3. This event frequency is categorized as a limiting fault (design basis accident). If cracking was permitted to progress undetected in the area of the nozzle blend radius (Zone 1) to the point of failure it could result in conditions resembling reactor vessel failure which is an event that cannot be assumed to be bound by the UFSAR LOCA discussions.

However, the proposed change offers the same level of protection against flaw propagation to an unacceptable size as previous actions, the probability of occurrence is not increased. As stated previously, the GGNS FW nozzle configuration contains those attributes acceptable to GE and the NRC for preventing the introduction of fatigue cracks in the FW nozzles. The BWR fleet has accumulated approximately 15 years of operation since implementing the changes without the discovery of additional cracking. Because appropriate examinations will be conducted that are proven to effectively detect shallow flaws (0.250 inches) and at a frequency that assures detection before the flaw exceeds 1/3 of the limiting flaw size, the probability of occurrence is not increased. Additionally, because of the fracture mechanics analysis, the examination program can be adjusted as required to respond to changes in plant operating conditions that may affect flaw growth.

The consequence of previously evaluated accidents and malfunctions of equipment important to safety are not increased by the proposed alternative examination requirements. The proposed change does not alter the ability to initiate or resist cracking, and it is extremely unlikely that cracking will initiate due to the original modifications that eliminated the clad from the nozzle surface, installed triple sleeve double piston thermal sleeves and improved operation of the feedwater system. Future UT examinations will provide a higher level of confidence than previous UT examinations because if cracking were to initiate, it will be detected in its early stages and a least before the flaw could exceed 1/3 of the maximum flaw size permitted by ASME Section XI.

As per UFSAR 6.3.3.7.4 and 3.6B.2.1.1.3, the postulated design basis LOCA initiating event is a double-ended recirculation line break with a break of over three square feet. This event would result in the most severe and limiting plant conditions possible for a LOCA. A break of the feedwater line was also considered but its break area was less significant (.362 square feet) when compared to the double-ended recirculation line break. Since this change does not alter any design or plant operating characteristics that would change the results of the LOCA analysis, the change in UT frequency and the elimination of the augmented PT requirement would not increase the consequences of a malfunction of equipment important to safety previously evaluated in the UFSAR. 10 CFR 50.46(C)(1) limits the evaluation of LOCAs to pipes in the reactor coolant pressure boundary therefore excluding the reactor pressure vessel. However, actions taken in the original design of GGNS to prevent crack initiation are unchanged by this evaluation, therefore, any previous evaluations by the NRC of GGNS actions to eliminate or minimize the potential for crack initiation are also unchanged. The change in UT examination frequencies and the elimination of the PT examination are appropriate and do not increase the consequences of a malfunction of equipment important to safety previously evaluated in the SAR. Examination of Zones, 1, 2 and 3 at frequencies determined by site specific fracture mechanics analysis provide assurance that if flaws were to initiate as described in NUREG 0619 that they would be detected before they exceeded 1/3 the maximum size permitted by ASME Section XI. Additionally, the alternative examinations do not create the possibility of an accident, or malfunction of equipment important to safety, of a different type than any previously evaluated. Feedwater nozzle cracking in Zones 2 and 3 has the potential for contributing to the loss of reactor coolant pressure boundary integrity, thus this event is a possible contributor to a LOCA inside containment. These accidents have already been evaluated in the UFSAR and there are no other accidents of a different type related to feedwater nozzle cracking.

Cracking in Zone 1 is in an area that would be considered part of the reactor pressure vessel which, by assumption, cannot be bound by current UFSAR LOCA evaluations. However, in the GGNS Safety Evaluation (NUREG-0831), it was concluded that after review of all factors contributing to the structural integrity of the reactor that no special considerations exists that make it necessary to consider potential reactor vessel failure. The design attributes for material, thermal sleeve configuration, and system operation have not changed since the initial evaluation. Therefore the possibility or probability for crack initiation has not changed since the initial evaluation. However, the proposed change does alter the examination program, but is believed to provide an equal or enhanced level of safety. Because of the better understanding obtained by the GGNS fracture mechanics, a tailored examination program can be maintained to ensure that the flaw detection threshold is maintained at 1/2 of the critical flaw size. This may exceed the level of safety that is currently offered through examinations required by ASME Section XI which are typically performed once every ten years.

Serial Number: 98-015-NPE Document Evaluated: MCP 96/1005,

DESCRIPTION OF CHANGE: This MCP will provide pressure relieving/equalization capabilities for Residual Heat Removal (E12) System Valves 1E12-F008 and 1E12-F009. pressure equalization lines will be added to these valves to eliminate pressure locking concerns. This change is in response to pressure locking concerns addressed in Engineering Report GGNS-92-0035.

REASON FOR CHANGE: To alleviate the potential concerns associated with pressure locking on 1E12F008, a 3/4" Class DBB line with a 3/4" manual globe valve 91E12-F444) will be routed from the area communicating between the 1E12-F008 valve seats to the upstream side (reactor vessel side) of the upstream valve seat. The equalization line will contain a branch connection with two 3/4" manual isolation valves to be used during construction/testing. These valves (1E12-F445 and 1E12-F446) are to be considered containment isolation test connection valves. To alleviate the potential concerns associated with pressure locking of 1E12F009, a 3/4" Class DBB line will be routed from the area communicating between the valve seats to the upstream side (reactor vessel side) of the upstream valve seat.

SAFETY EVALUATION: NPE review of MCP 96/1005 has concluded that the proposed activity, adding pressure equalization lines to RHR System Valves 1E12-F008 and 1E12-F009, does not represent an Unreviewed Safety Question. The purpose of the pressure equalization lines installed by this MPC is to perform the passive function of providing pressure equalization between the upstream piping and the valve internals to prevent pressure locking of these valves. This action will provide an enhancement to the reliability of 1E12-F008 and 1E12-F009 valve operations. The reactor coolant pressure boundary, primary containment integrity and related aspects with respect to valves 1E12-F008 and 1E12-F009 are not changed by this modification. The outboard valve disc continues to perform the sealing and isolation function. Thus, bypassing the inboard disc will have no effect on containment integrity and should not be construed as such. The modification authorized by this MCP does not result in testing which differs from testing activities currently described in the FSAR. The modified 1E12-F008 and 1E12-F009 valves will be tested for leakage consistent with the requirements of Technical Specification 3.6.1.3, FSAR Section 6.2.6 and FSAR Table 6.2-49. Additionally, the new containment isolation test connection valves associated with 1E12-F008 are designed, installed and tested to ensure reactor coolant pressure boundary and the primary containment integrity is maintained. This design change will have no adverse impacts on the operability of affected valves or the associated E12 system or components. The design has been evaluated against the applicable design criteria, installation and operational requirements, and all necessary requirements and commitments are met. This modification is intended to provide added assurance that the 1E12-F008 and 1E12-F009 valves will open, however, the ability of these valves to isolate or be closed is unaffected by this change. Since all functions and requirements discussed in the Technical Specifications remain unchanged by the implementation of this design, all margins of safety remain unaffected. After review of the proposed activity, it has been concluded that the proposed plant modification does not create any new radiological effluent release pathways, does not introduce any new radiological effluents and all associated work is within the confines of the power block. Thus, the work authorized by this MCP does not represent an Unreviewed Safety Question or an Unreviewed Environmental Question.

Serial Number: 98-016-NPE Document Evaluated: LDCR 97-096

DESCRIPTION OF CHANGE: Enhanced description of the spent fuel pool cooling and cleanup (FPCC) system in GGNS UFSAR Section 9.1.3. Revised footnotes in Tables 9.2-16 and 9.2-17 to describe that the FPCC decay heat loads from the 18 month fuel cycles were bounded by the FPCC decay heat loads from the 12 month fuel cycles. These UFSAR changes were precipitated in part by NOV 50-416/97-05. Added brief discussion of FPCC to Appendix 8A for loss of all alternating current power. Editorial changes were made to UFSAR Tables 1.9-1 and 9.2-1.

REASON FOR CHANGE: Current UFSAR descriptions for FPCC were incorrect or incomplete. Tables 9.2-16 and 9.2-17 footnotes were clarified to describe that FPCC design basis heat loads are bounded by previous FPCC decay heat load analyses. Addition to Appendix 8A complements evaluations of loss of all alternating power events performed for other systems.

SAFETY EVALUATION: The UFSAR changes proposed by LDCR 97-096 are editorial in nature. These changes are primarily intended to clarify the functional requirements of FPCC. These UFSAR changes will not adversely impact plant operating conditions previously considered. No change to plant system hardware will be implemented. An unreviewed safety question does not exist as a result of the UFSAR revisions proposed by LDCR 97-096.

Serial Number: 98-017-NPE Document Evaluated: ER GGNS-97-0050-0

DESCRIPTION OF CHANGE: This safety evaluation addresses the issues concerning receipt of the Cycle 10 reload fuel. These issues include (i) the environmental impact of its transportation, (ii) its movement into the spent fuel pool racks (fuel handling accident), and (iii) storage in either the spent fuel pool racks or new fuel vault (criticality and seismic).

REASON FOR CHANGE: Reload fuel is necessary for Cycle 10 operation.

SAFETY EVALUATION: This evaluation concludes that (i) transport of the reload batch (as fresh and spent fuel) poses no significant environmental impact, (ii) the current fuel handling accident remains applicable, and (iii) the reload batch can be safely moved to and stored in either the new fuel vault or spent fuel pool (including the blackness test area).

Serial Number: 98-018-NPE Document Evaluated: DCP 89/0069

DESCRIPTION OF CHANGE: DCP 89/0069 will replace the eight existing analog APRM flow control trip reference (FCTR) cards (two each, located in H13P669, 670, 671 and 672) with new microprocessor controlled cards containing a digital representation of the various input flow/trip reference curves. A scram will occur upon entering the exclusion region via the existing flow biased scram trip circuits. A control rod block will be activated upon entering the restricted region. The Exclusion Region is the area of the licensed core power-flow operating domain where the reactor is susceptible to reactor instabilities. The restricted region is the area of the licensed core power-flow operating domain where the reactor is susceptible to reactor instabilities in the absence of restrictions on core power distributions. The setpoint curves of the new FCTR cards must be more complex in order to enforce the required exclusion and restrictive regions but minimize the impact on plant operation. In addition to the curves for single and two loop operation there is also a setpoint setup feature that will allow required reactor maneuvering in the restricted region under administratively controlled conditions to ensure adequate stability margin (Fraction of Core Boiling Boundary < 1 as discussed in NEDO 32339). Since core flow is used in the new setpoint analysis, the new cards include a mapping function that concerts the total drive flow signal to core flow representation. The new FCTR cards are designed to be physically interchangeable with the existing cards. They will use the same +15 and -15 vdc power supplies. One of the three card out of file monitoring circuits will be modified on the new card. The one that produces an INOP trip will have a contact added that will open if a card malfunction is detected by the cards microprocessor.

Two new microprocessor based stability monitor period based detection system (PBDS) cards will be installed in existing spare LPRM card slots (one in H13P669 and one in H13P670). The PBDS cards are designed to accept a maximum of 16 input signals 0-10 vdc representing 0-125% power. The signals for each PBDS card will be obtained by tapping into the LPRM signal output of the LPRM filter card. Five of the 22 LPRMs in channel A are from "D" level detectors and 5 in channel B are from D level detectors. These LPRMs will be excluded as inputs to the PBDS cards because the "D" level LPRMs provide little useful information concerning core stability. The PBDS cards are designed to use the existing +15 and -15 vdc power supply. The new stability cards have three alarm outputs (HI, HI HI and INOP) that must be wired to annunciator circuits via the AR6 optical isolator in 1H13P669/1H13P670. These cards have no trip functions but the operators will be required to initiate a manual scram upon receipt of a instability HI HI alarm. The digital fiber optic output of the new PBDS cards will be connected to a new "Fiber-Optic to RS-232 Converter" installed in the non divisional section of 1H13P669/670. The output of this converter will be input to the plant computer as serial data. The two analog outputs (highest count, second highest count) will be added as inputs to the C88 muxes. The LPRM filter card will be deactivated by removing the capacitors from the circuit card. The stability monitor cards need unfiltered LPRM signal and they can not be connected directly to the LPRM card output. The new plant computer can perform the filtering for the thermal/hydraulic calculations. The additional noise in the RC&IS indication is acceptable to operations.

REASON FOR CHANGE: In March 1988, LaSalle Unit 2 experienced a reactor power instability event in the natural circulation region of the power/flow map because of a dual recirculation pump trip. In June 1988 the NRC issued bulletin 88-07 advising that BWR cores were susceptible to coupled neutronic/thermal-hydraulic reactor instabilities in certain potions of the core power/flow operating domain. Analytical studies later determined that for most plants, existing neutron monitoring features of the reactor protection system do not assure automatic protection of the fuel limits for these events.

In December 1988, the NRC issued Bulletin 88-07 Supplement 1 which required BWRs to implement Interim Corrective Actions" (ICA's). The ICA's are procedural controls to prevent operation in certain areas of the reactor power-flow curve where instability may occur. In June 1991 the BWR Owners Group issued NEDO 31960 which provided possible long term solution (LTS) options. The LTS option 1A was the simplest design. It involved modifying the flow biased scram to enforce the ICAs. The NRC had concerns with this design. They also wanted an on-line stability monitor.

In August 1992 WNP-2 experienced an instability event during startup. In July 1994 the NRC issued Generic Letter 94-02 which required improved ICAs, an LTS option selection and implementation schedule. In GNRO-94/0111 dated September 1994 GGNS committed to implementing the enhanced LTS option 1A. This option was presented in NEDO 32339P-A Supplement 2 which has been reviewed and approved by the NRC. It addressed the NRC concerns with LTS option 1A.

SAFETY EVALUATION: The new FCTR cards will be supplied as Class 1E and seismically qualified components since enforcement of the exclusion region is now considered to be a safety function. Software V&V was done by GE per NEDC 32339P-A Supplement 2, which has been reviewed and approved by the NRC. The new FCTR setpoint curves will be determined per NEDO 32339 Rev. 1. The new setpoint curves are slightly less conservative than the existing curves at some points. The new microprocessor controlled FCTR cards also introduce a 250 millisecond delay in the flow biased trip circuit. This delay is predominantly due to the 10 hz digital filter. These differences are acceptable because the current transient analysis does not take credit for the flow biased scram (Reference NEDO 32339 Rev. 1 Page 6-2) and the new setpoints are within the current TRM allowable values. The APRM clamped high flux scram which is credited for accident mitigation is not changed. A Technical Specifications change is not required prior to installation because no credit will be taken for the hardware fix to the stability problem. The ICA's will remain in effect until NRC approval of our Technical Specification change is given. An interim TRM/FSAR change will be issued to document that either the old or the new setpoints may be employed. After the SER is issued the flow biased scram setpoints will be moved from the TRM to the COLR and the requirements for the new PBDS cards will be added to the Technical Specification per NEDO 32339 Supplement 4. A phased in implementation of this design is necessary to provide an opportunity for engineering to collect on-line data and to observe the operation of the new FCTR and PBDS cards. This partial implementation will not change channel and divisional redundancy of the current NMS design. The new FCTR cards were designed in accordance with NEDC 32339P-A Supplement 2 which has been reviewed and approved by the NRC. Per NEDC 32339P-A the new FCTR and PBDS cards will not adversely affect the operation of other panel components. The power supplies have sufficient margin to handle the increased load of the new cards. The stability monitor cards serve as a backup (stability defense-in-depth feature) for the new flow control trip reference cards. They are therefore not considered to perform a safety function. These cards will be supplied Class 1E and seismically qualified, however. The PBDS alarm setpoints will be determined per NEDO 32339 Rev. 1 Reg. Guide 1.75 isolation/separation requirements will be maintained with the new annunciator and computer interfaces. The design was done in accordance with all applicable codes and standards. These changes will not adversely effect the function or operation of the Neutron Monitoring (C51) System. The changes will not compromise any safety related system, structure or component per NEDC 32339P-A, the new FCTR and PBDS card components will be very reliable with a mean time between failure of 270,000 hours. Both cards were designed and tested for the worst case environmental conditions that they could be exposed to in the APRM cabinets. They were designed and tested to ensue that they would not be susceptible to externally generated EMI and that the operation of other nearby equipment will not be affected by EMI generated by the new cards. The automatic self test feature of the microprocessor controlled cards should prevent unidentified component failures. Implementation of this design will reduce the overall probability of instability events by preventing inadvertent entry into the exclusion/restricted regions and by early detection and operator intervention. This design will also reduce the consequences of events that result in core instability.

Serial Number: 98-019-NPE Document Evaluated: TSTI 1N35-98-001-0-N, Procedure IOI 03-1-01-2, SOI 04-1-01-N11-1 and SOI 04-1-01-N23-1

DESCRIPTION OF CHANGE: This Safety Evaluation addresses increasing the GGNS reactor core thermal power from 95% MWT to 100% MWT without the Moisture Separator Reheaters' (MSR-1N35B001A/B) 2nd Stage Reheaters in service.

REASON FOR CHANGE: This safety evaluation supports increasing reactor core thermal power up to 100%. A higher than normal steam flow will be through the main turbine; however, steam line flow will be ~100%.

SAFETY EVALUATION: This evaluation concludes that increasing core thermal power above 95% without the MSR's 2nd Stage Reheaters is acceptable and does not constitute an unreviewed safety question. The proposed operating limits remain within the limits bounded by the FSAR Chapter 15 analysis.

Operating at 100% core thermal power will increase steam and feedwater flow rates. Increasing these flows with the MSR's 2nd Stage Reheaters isolated affects steam dome pressure, main steam line pressure, the high pressure turbine inlet flow, turbine control valve position, and the drain valve controls.

Increasing core thermal power increases reactor vessel steam flow. Increased steam flow increases the reactor vessel steam dome pressure by ~5 psi above the current operating pressure. Likewise, the steam line pressure sensed by the Initial Pressure Controller (IPC) will anticipate increased steam flow (i.e., due to increased steam line pressure) allowing the Turbine Control Valves to open more. The IPC pressure setpoint will remain unchanged. With the 2nd Stage MSR's Reheaters isolated, the amount of steam flow to the High Pressure Turbine increases. The Turbine Control Valves open more to accommodate the steam flow increase. With the MSR's reheaters unisolated, an increase in turbine inlet pressure is a HP flow increase. With the MSR's reheaters isolated, the turbine inlet pressure is expected not to exceed 40 PSI (i.e., 54 psia) and mass flow is ~5.5% increase to the HP turbine (Ref. 4). With the isolation of the MSR's 2nd Stage Reheaters, the #6 Highe Pressure Feedwater Heaters has decreased shell side heating while the remaining feedwater heaters experience increased shell side flows and heating.

<u>RPS Reactor Pressure Setpoint</u>

The reactor pressure increase due to 2nd Stage MSR reheat isolated and the resulting effect on the RPS reactor pressure setpoints were evaluated. The reactor pressure increase for this operating condition is expected to be about 5 psi, which will closely match 100% power.

This increased pressure will still remain below the RPS reactor vessel steam dome pressure - High Limit of 1064.7 psig. All Technical Specifications margins and safety analysis fuel operating limits are being maintained.

Feedwater and Steamline Flow Rates

Increasing to 100% core thermal power increases steam and feedwater flows. Currently these flows are not at 100% rated conditions. Increasing to rated steam and feedwater flow would not present adverse conditions such as unacceptable steam carryover to the steam line flows or feedwater flows or the feedwater control system. In fact, the feedwater temperature is expected to increase by ~4°F (Ref. 2) bringing the temperatures closer to 100% rated system conditions.

Feedwater Control System

With the change in turbine inlet pressure due to the isolation of the 2nd stage MSR, the turbine 1st stage pressure increases and the Feedwater Control System INFI 90 sensing of turbine 1st stage pressure may need calibrating for this turbine 1st pressure increase.

Turbine Control Valve Position

The Turbine Control Valve Position will open more with the increase in steam line pressure and increased steam flow. For a given flow increase, the control valve open more means it takes longer to close for a given a turbine control valve closure signal which is bounded by the current analysis. The current analysis assumes that the control valves are partially open and which corresponds to more severe transients. The reactor pressurization signature is delayed with a longer closure time of the turbine control valves. Also, longer closure of a turbine control valve event is well bounded by the limiting pressurization event (limiting load reject with no bypass). Core operating limits have been developed to ensure that moderate frequency events do not violate the MCPR safety limit.

Event though there are no significant flows left that could affect turbine 1st stage pressurization, a turbine control valve position limit will be utilized and adjusted as needed. This is because there are still steam flows off the main steam line that could be redirected back to turbine 1st stage pressure (i.e., SJAE, drains and etc). These steam flows are not significant effects on turbine 1st pressure; however, it is prudent until a new turbine control valve limit can be established. For TSIT 1N35-98-001-0-N, the turbine control valve position is limited and evaluated as this infrequent performance evolution needs.

ATT Control Valve Position and Reactor Power

Due to the increase in first stage pressure for the given thermal power, the Automatic turbine tester (ATT) will need the appropriate reactor power and control valve position for the turbine overspeed protection. Overspeed protection is addressed in TRM Section 6.3.8. The proposed change does not affect any present requirements in the Technical Specifications. The proposed change will not add any new requirements to the Technical Specifications. However, ATT will have to be performed at a lower thermal power to obtain the same Main Control Valve (MCV) margin (~40% on #1 MCV) than was previously used with 2nd stage MSR in service.

Moisture Separator, 1st Stage Reheater and Feedwater Heater Shellside Levels

With the isolation of the MSR's 2nd Stage Reheaters, the #6 High Pressure Feedwater Heaters has decreased shell side heating while the remaining feedwater heaters, moisture separators, and 1st Stage Reheaters experience increased shell side flows and heating. The increased shell side feedwater heater, MSR, and 1st stage reheater flows will be minimal and should not pose any significant changes to system processes.

Erosion/Corrosion

Erosion/Corrosion has added an item to the Flow Accelerated Corrosion (FAC) program for RFO9 inspection due to plant operation without the MSR's 2nd stage reheaters in service. The added inspection item located at the steam line piping inlet to the turbine discharging to the low pressure condenser is identified as the most likely location for flow accelerated corrosion. During the outage inspection, in the event of piping degradation at this location, other steam line inlets to turbines will be added to the inspection. (Ref. 3)

First Stage Turbine Pressure

Due to the increase in first stage inlet turbine pressure for the increased thermal power, from a turbine integrity standpoint, operation of the GGNS reactor core at 100% thermal power with the 2nd Stage Reheater out of service is permissible until RFO9. The High Pressure turbine inlet pressure is expected to rise 54 psi if all thermal power is regained. (Ref. 2)

High Pressure Turbine Inlet Flows

With the increase to 100% core thermal power at 100% rated steam line flow, the first stage inlet turbine flow rate will be increased ~5.5%. This means the High Pressure turbine inlet pressure is expected to rise 54 psi if all thermal power is regained. (Ref. 2)

With the increase in high pressure turbine inlet flows, there are no other flows that are expected to cause any significant pressurization of 1st stage pressure. The only loads (i.e., flows) that could create a pressure transient due to loss of flows are the SJAE and drains. These flows are not significant and currently would not create more noticeable disturbances if either of these were to become inoperable.

Main Turbine Lube Oil Temperature

With the increase to 100% core thermal power at 100% rated steam line flow, the first stage inlet turbine flow rate will be increased ~5.5%. This means the High Pressure Turbine will provide more electrical generation output. The turbine lube oil system specifically to the HP bearings may experience a lube oil temperature increase which is controlled by the TBCW system. Any turbine lube oil temperature increase should be controlled by the TBCW temperature control valve at the Main Turbine Lube Oil coolers heat exchanger exit.

In addition to the Balance of Plant (BOP) side, it is concluded that the Cycle 9 reload analysis bounds the Technical Specification requirements for the proposed plant operational configuration and that no changes to these requirements are needed. The plant protection systems will still meet all the requirements in the GGNS Technical Specifications and GGNS UFSAR.

Serial Number: 98-020 Document Evaluated: ER 97-0645-00-00

DESCRIPTION OF CHANGE: The Division I and II Standby Diesel Generators have had a reoccurring leakage of the Dresser couplings on the Jacket Water piping. ER 97-0645-00-00 is replacing the Dresser couplings on the supply piping to the Cylinder Head Water Jacket piping for the Division I and Division II Diesel Generators with a flexible hose configuration. The new configuration has been evaluated for stresses due to thermal movement, vibratory and seismic loads. The piping and supports for this modification have been design per ASME Section III, Class 3. The hose being installed by this ER is a metal reinforced polymer and is qualified for this application.

Additional combustibles are being installed in Emergency Diesel Generator (EDG) Division 1 and 2 Rooms 1D310 and 1D308 respectively.

REASON FOR CHANGE: The Dresser couplings being replaced on both Division I and Division II Diesel Generators have had a reoccurring history of leaking (GGCR 1997-0727-00) having a potential effect on the reliability of the Standby Diesel Generators. By replacing the leaking couplings with a flexible hose configuration, this change provides increased integrity and reliability of the Standby Diesel Generator Cooling Water System. The flexible hose configuration will absorb engine vibration and thermal movements and will not be susceptible to the leakage problem experienced by the Dresser Couplings and will therefore, provide enhanced engine performance.

SAFETY EVALUATION: The original engine mounted piping of the jacket water system was designed and fabricated to the guidelines of the Diesel Engine Manufacturer's Association (DEMA) standards and the manufacturer's own standard procedure. The applicable Code for the jacket water piping is ANSI B31.1 and Quality Classification is Quality Group D (aa), Seismic Category I and Safety Class 3 (Ref. Table 3.2-1).

ASME Section III is an approved Code for use in GGNS Safety Related piping system design and installation. The use of this Code assures that a component will meet its design requirements for design basis loading, like pressure, dead weight, thermal and seismic. The use of the ASME Section III Code allows the use of existing approved Safety Related programs for design and installation at GGNS. The use of the ASME Section III Code in lieu of ANSI B31.1 Code in the design and installation of the piping does not affect safety margins. The use of the ASME Code Section III assures the integrity of the piping system.

The seismic qualification of these changes was by analysis. The fundamental frequency of the piping was above 33 Hz and the hose lengths were designed so that their natural frequencies did not coincide with those of the engine, so that no resonant condition would occur.

This changed portion of the jacket water system will be subject to a pressure test at operating pressure and temperature. A hydrotest was not required. This is in keeping with the philosophy of the later ASME Section XI Code and which has been approved by the NRC (Ref. GGNS-M-489.0). The rationale is that at hydrotest pressure, the resulting stress in the piping is low as compared to material yield strength. It is therefore evident that the pressure test is not an integrity test, but only a "leakage test" and would not provide any increase in safety. Section XI Code replaced the hydrostatic test with system leakage test performed at nominal operating pressure based on reviews that leakage caused by hydrostatic test was not substantially different from leakage created at nominal operating pressure.

An evaluation of the Jacket Water piping configuration has shown that the total head loss in the Jacket Water system with the new design installed will change by less than 0.15 psig. Based on a review of the configuration and operating characteristics of the Jacket Water system, NPE has concluded that installation of the flexible hoses, as specified in ER 97-0645-00-00 will not degrade or adversely impact the ability of the Jacket Water system to maintain the operating temperatures of the Diesel Generators within design limits during all modes of operations.

The new piping and fittings will be purchased Safety Related to either the ASME Boiler and Pressure Vessel Code, Section III, Class 3 or the ANSI Piping Code, B31.1, meeting 10CFR50 Appendix B requirements. The flexible hoses and the flexible hose fittings were purchased safety related meeting the requirements of 10CFR50, Appendix B. The flex hoses and flex fittings meet the Society of Automotive Engineers standards SAE-J516 and SAE-J517, respectively, the hose fittings meet SAE-J514 and the testing of these parts meet the standards of SAE-J343. This assures that the materials are suitable for use in a Safety Related application for the Diesel Generator jacket water system. The new flexible hose configurations have been designed and stressed analyzed to the requirements of the ASME Section III, Class 3 Code.

The fire protection features (suppression, detection, and separation) as currently described in the FHA for EDG Division I Room 1D310 (Fire Area 61) are more than adequate to contain a fire with a dire duration of <90 minutes.

Therefore, the addition of these combustibles has no adverse effect on the ability to achieve and maintain safe shutdown in the event of a fire in this area as presently analyzed in the SAR.

The operation of the Standby Diesel Generators will not be changed by this modification. System interface with other plant systems will not be directly or indirectly affected. This change will protect against leakage of this piping by providing a more suitable piping arrangement to withstand system temperature changes, thermal growth, seismic and engine vibration. The new flexible hoses are metal reinforced polyethylene-like material to accommodate system temperature expansions and contractions. There will be on increase in the probability of occurrence or the consequences of an accident or malfunction of equipment important to safety previously evaluated in the safety analysis report. This modification will not create the possibility of an accident or malfunction of a different type than any evaluated previously in the safety analysis report. The new configuration of the piping and hose is qualified by analysis and there will be no reduction in the margin of safety as defined in the bases for any technical specifications. The new flexible hose configurations help to ensure the stated function of the jacket water portion of the Standby Diesel Generator is not jeopardized.

Serial Number: 98-021-NPE Document Evaluated: ER 97/0122-00-00

DESCRIPTION OF CHANGE: Changes authorized by ER 97/0122-00-00 are limited to the installation of a surfactant tank for use with the Radial Well (P47) System Chemical Feed Pumps. The surfactant tank will be provided and maintained by a chemical vendor, and will be designated as "Chemistry Department Controlled/Vendor Supplied" equipment. The tank will be connected to a permanently installed piping header that allows the contents of the surfactant tank to be injected (metered) into the P47 System piping. The surfactant will be injected into the P47 System using Chemical Feed Pumps SP47C006A/B which were previously installed in the P47. These feed pumps will be used to maintain the desired amount of surfactant in the P47 System. The surfactant will be used concurrently with other chemical additives to improve the overall effectiveness of the water treatment control program in use at GGNS to minimize the potential for corrosion and fouling of various cooling water systems piping and components.

REASON FOR CHANGE: The GGNS Chemistry Department maintains a water treatment program for the purpose of minimizing the corrosion and fouling of cooling water system piping and components. The effectiveness of this program will be improved by injecting a controlled amount of surfactant into the P47 System. From the P47 System injection point, the surfactant will progress through the P47 System, and interconnected systems, providing additional corrosion and fouling protection to the affected piping and components.

SAFETY EVALUATION: This Safety Evaluation has concluded that the changes proposed by ER 97/0122-00-00 do not represent an Unreviewed Safety Question. The installation of a surfactant storage tank, and the subsequent injection of the surfactant into the P47 System, will minimize the potential for corrosion and fouling of the affected cooling water system piping and components. The surfactant will be injected into the P47 System at a rate which produces the desired surfactant concentration in the P47 System. Storage and use of the bulk surfactant has been reviewed and there are no unresolved material compatibility concerns or environmental issues associated with the injection of this chemical into the P47 system. The on-site storage of bulk surfactant has been reviewed and there are no Main Control Room habitability concerns resulting from the proposed action. The storage and use of the surfactant will be controlled in a manner that does not represent a conflict with requirements contained in the GGNS National Pollutant Discharge Elimination System (NPDES) permit. The use of a surfactant in the P47 System has been previously reviewed and approved as evidenced by Safety Evaluation No. 85/158-R00 and 97/001-R00. The use of a surfactant is also discussed in Section 9.2.8.2 and Section 10.4.5.2 of the UFSAR. The proposed change will not impact the contents of the GGNS Technical Specifications (TS) or the Technical Requirements Manual (TRM). Based on review of this subject, it has been concluded that the proposed change does not represent an Unreviewed Safety Question.

Serial Number: 98-022-NPE   Document Evaluated: 98-011

DESCRIPTION OF CHANGE: A License Document Change to update the FSAR Table 5.2-8 "WATER SAMPLE LOCATIONS" to reflect instrument changes made by DCP 97/0010 and to correct and clarify alarm high values in the same FSAR Table 5.2-8 and Section 5.4.8.2 System Description of Reactor Water Cleanup System. The update to Table 5.2-8 caused by DCP 87/0010 will indicate that the conductivity instrument types are multi-range and not nonlinear. The correction between Table 5.2-8 and Section 5.4.8.2 will have the same alarm high conductivity values as listed in Section 5.4.8.2. The clarification in Table 5.2-8 and Section 5.4.8.2 will explain that alarm high conductivity values are maximum values that are dependent on operating modes of the plant. Typical alarms values are conservative, that is less than maximum values.

REASON FOR CHANGE: The License Document Change will update the information in the FSAR Table 5.2-8 to reflect the current plant configuration and to resolve differences between FSAR Table 5.2-8 and Section 5.4.8.2 in regard to alarm high values.

SAFETY EVALUATION: The License Document Change evaluated by this safety evaluation concludes that the change does not involve an unreviewed safety question. FSAR Changes to reflect conductivity instrumentation type and to correct and clarify alarm high conductivity values does not adversely impact the mitigation of an accident or safe shutdown of the reactor. The License Document Change does not affect Technical Specifications or Technical Requirement Manual.

Serial Number: 98-023-NPE Document Evaluated: ER 97/0324-00-0, ER 97/0324-01-

DESCRIPTION OF CHANGE: The generator gas system upgrade will replace the existing gas system components with a single skid which will include a dual tower gas dryer, a gas instrumentation rack with control panel and primary water tank gas supply components. The new skid will be relocated to provide additional space for future generator seal oil system upgrade.

The change will require rerouting of N44 system carbon dioxide piping. The change will not affect fire protection system carbon dioxide piping or operation. The change will require rerouting of connecting piping for hydrogen to he new skid. The nitrogen supply bottles will be relocated outside the building to the yard. The existing methane supply piping will be rerouted and supply nitrogen. Also, the existing carbon dioxide flash evaporator and high pressure storage bottles will be removed.

The change will require rerouting of connecting tubing from instrument air system (P53). The change will require cooling water source for dryer cooler. Turbine Building Cooling Water (TBCW) system (P43) will supply approximately 1 gpm flow to the dryer cooler. The change will have negligible affect on P43 system heat load. The change will require rerouting of connecting piping for existing carbon dioxide vent and purity meter vent. The change will require rerouting of connecting piping of existing primary water vent to misc. vents & drains for A-CPSI turbine generator equipment (N31 system). The change will require routing of a new drain to the existing chemical radwaste (CHRW) system. The change will delete existing vent from dryer to the generator bearing exhaust fan of lube oil system (N34). The change will require rerouting of the existing mechanical vacuum pump cooler outlet piping of the plant service water (PSW) system (P44) to remove interference with new skid. The removable barrier between elevations 129' and 133' near the new gas rack location will require removal and modification. The vertical portion of the existing removable barrier will be reused. Existing piping support N1N19G002H09 of condensate system (N19) will be modified to remove interference with new skid.

Appropriate signs will be made and located around the generator gas rack indicating the requirement that there is a 15 foot zone of influence round the hydrogen gas dryer and associated components on skids in which no ignition sources are permitted (i.e. open flames, use of spark producing tools, use of electrical equipment which is not acceptable for Class I, Group B, Division II locations as identified by Article 500 in the National Electric Code 70). A baffle will be constructed at the ceiling above the gas rack to minimize the possibility of hydrogen leakage infiltrating the generator exciter housing.

New conduit will be installed from the replacement gas dryer skid to the existing tray sections above the GAC panel. New signal cable will be installed from the skid to the GAC panel for annunciator/indicator circuits. New signal cable will also be installed from the replacement skid to a local Plant Data System mux cabinet to support computer point additions. New poser cable will be installed from a BOP MCC to the new skid. Lighting in the area of the new skid will be replaced with fixtures suitable for Class 1 and Division 11 Group B hazardous area service.

REASON FOR CHANGE: The generator gas system upgrade project will replace the existing gas dryer to further decrease the hydrogen moisture content (i.e., dew point). However, detailed design reviews have revealed that almost all replacement parts on the gas racks are obsolete. The replacement parts will require modification to the existing piping. The new skid will be relocated to provide necessary space for future generator seal oil system upgrade. The existing methane supply is no longer required. Therefore, methane supply piping will be used for nitrogen supply. The p43 system will supply cooling water to the new dryer. In order to remove moisture, drain will be routed to CHRW. New dryer does not require vent. Therefore, vent to the N34 system ahs been deleted. Also, P44 system piping, N19 system pipe support and removable barrier interferes with the new skid location. Therefore, these components will be modified to remove interference with new skid.

SAFETY EVALUATION: The generator gas rack (N44D001) is a part of the hydrogen & carbon dioxide (N44) system. The design change will affect the N43, N31, N34, N19, P43, P44, C91, and P53 systems. The proposed change will enhance the main generator gas system without affecting the operation of interfacing systems. The proposed change will update information provided in UFSAR Section 10.2.5 and figures in UFSAR Sections 9.2.9, 9.3.1 and 9.5.1. The change will not affect the design information provided in the UFSAR Sections for N19, N31, N34, N43 and P44 systems. UFSAR Section 3.2 classifies systems N19, N31, N34, N43, P43, P44, and P53 as "Other" which means that their failure will not compromise any safety related functions or prevent safe shutdown. UFSAR Table 3.2-1 classifies these systems and their associated components as non-safety related, non-seismic, quality group D and ANSI B31.1. The modifications made by this design change is in compliance with the criteria listed in UFSAR Table 3.2-1.

The replacement of the gas dryer and associated components are not safety related and will not affect any safety related equipment or systems. The proposed change does not affect any parameters specified in the GGNS Technical Specifications (TS) or GGNS Technical Requirements Manual (TRM). The changes will not affect any equipment important to safety.

The main generator gas system is a potential fire and explosion hazard, but changes to N44 system components are not anticipated to increase the probability of operational occurrences or accidents. A baffle will be constructed at the ceiling above the gas rack to minimize the possibility of hydrogen leakage infiltrating the generator exciter housing. All electrical components on the new skid meet the NEC requirements for Class 1 Division 11 Group B hazardous area service, or are installed in an air purged enclosure per NFPA 496. Additional modifications are included to ensure that the area lighting meets the NEC criteria above. Piping and tubing for the proposed change will be designed and installed to meet or exceed the original design requirements for minimizing potential hydrogen leaks. Hydrogen gas system components will be relocated to an open area near the existing location so that potential small hydrogen leaks will not accumulate to combustible concentrations. The design has been evaluated against the applicable design criteria, installation and operational requirements. All necessary requirements and commitments are met. A loss of hydrogen gas supply would continue to result in insufficient cooling of the main generator and ultimately require shutdown of the main generator.

The N19, N31, N34, N43, N44, P43, P44, C91 and P53 systems are not part of the reactor coolant pressure boundary nor are they required for safe shutdown of the plant. The design change will not alter the design, function or operation of any equipment important to safety as evaluated in the UFSAR. The gas system components change will not affect the reliability of equipment important to safety since it has been designed in accordance with all necessary design criteria, commitments and requirements. The proposed change will enhance operation of the main generator gas system without affecting any safety related systems. The capability to safely shutdown the reactor will not be impacted by these changes. Therefore, implementation of ERs 97/0324-00 and 97/0324-01 will not adversely impact plant operation or any system important to safety.

Serial Number: 98-024-NPE   Document Evaluated: ER 97/0487

DESCRIPTION OF CHANGE: This safety evaluation addresses a change in the amount of water coverage necessary for control blade movements using the auxiliary hoists on the refueling and fuel handling platforms. The actual reduction of water coverage and normal-up interlock setpoint changes will be determined by P&SE, but the total water coverage reduction may not exceed 10 inches (6'2" water coverage). This change will apply only for RFO9 while in modes 4 or 5 (therefore will not include a permanent change to the FSAR) and does not approve operation of any equipment in a manner inconsistent with its original design.

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

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

There are 34 control blades scheduled to be exchanged in RFO9. By reducing the water coverage limit for the blades, the overall blade moving process will be facilitated and made more efficient. Also, compliance with the reduced water coverage limit will be controlled by the redundant normal up interlocks.

SAFETY EVALUATION: A 10 inch reduction in the water coverage (shielding) above the control rod blade will produce exposure rates approximately 5 times greater than normal. These higher rates will only be experienced during the brief time when the blades are being moved through the cattle chute and will not present a conflict with regulatory limits.

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

Serial Number: 98-025-PSE Document Evaluated: TA 970012

DESCRIPTION OF CHANGE: The manway cover on the Condensate Phase Separator tank will be removed to allow the use of a submersible pump to transfer water from the tank to the floor drain collection subsystem.

REASON FOR CHANGE: To implement a temporary alteration which will mitigate problems with clogging the screens during decanting operations.

SAFETY EVALUATION: The temporary alteration (TA) is for the removal of the manway cover to allow decanting (Dewatering) of the Condensate Phase Separator (CPS) tank using a submersible pump via a fire hose to a floor drain. There is no Technical Specification for the method of or rate for decanting the CPS Tank. Removal of the manway cover has no effect on the operation of the CPS Tanks or any other equipment in the liquid radwaste system. Therefore, this modification does not reduce the margin of safety as defined in the BASES for any Technical Specification.

The fire hose used to transfer water from the CPS tank to the floor drain has been satisfactorily inservice tested using the submersible pump in accordance with ANSI B31.1. Section II of the ASME Boiler and Pressure Vessel Code does not specifically state the use of flexible hose as an acceptable material. However, because the hose is chemically, thermally, and hydrostaticly compatible with the system, its use meets the intent of this code to safeguard against rupture or other failure. By meeting this intent, its use will preclude any hazard to the health and safety of GGNS personnel and the public. Thus preventing any increase in the probability of occurrence of an accident previously evaluated in the SAR.

There is a small probability for a spill accident due to the open manway. However, the level indication in the tank is unaffected thereby allowing operators to monitor tank level. There is an open, installed 8 inch overflow line 3 feet below the level of the manway which goes to an open floor drain. Each tank is located in a separate room with a floor drain in each room. Each room has a manway access with a 6 to 8 inch berm. Any spill that may occur will be contained by the berm and flow down the floor drain. Any water that flows over the berm will flow to floor drains in the pump areas adjacent to the tank rooms. The radwaste building ventilation system maintains a negative pressure on the building thus preventing the release of airborne radioactivity to the environment. This accident is bounded by the spill accidents evaluated in Section 15.7.2 (rupture of evaporator bottoms tank) and 15.7.3 (rupture of RWCU Phase Separator Tank) of the UFSAR. Therefore, there are no unreviewed safety questions created by this modification.

Serial Number: 98-027-NPE   Document Evaluated: ER 96-0403-00-00

DESCRIPTION OF CHANGE: Replace the existing GE supplied EPA breaker units with new breaker units that do not use GE logic cards. The new units will use discrete trip relays to sense any abnormal power quality condition in order to protect the associated RPS bus. The use of the discrete trip relays will improve the overall reliability of the EPA breaker units. GE's logic cards have a documented history of failure while the discrete relays being used are solid state and have tighter trip point tolerance. This difference will increase EPA reliability and reduce the potential for unnecessary challenges to safety systems.

REASON FOR CHANGE: Multiple half scrams have occurred due to RPS EPA breaker (1C71S003A-H) trips. In each case an EPA logic card malfunction was found to be the cause of the breaker trip. Maintenance history on the EPAs indicates a generic reliability concern due to premature failures of various GE logic card components.

SAFETY EVALUATION: ER 96/0403-00-00 will completely replace the existing EPAs that utilize GE logic cards to monitor the Reactor Protection System power supplies for undervoltage, overvoltage and underfrequency conditions. The replacement EPAs will utilize solid state relays instead of the logic cards to sense the undervoltage, overvoltage, and underfrequency conditions. The trip devices will actuate and deenergize the undervoltage coil of a molded case circuit breaker housed in the EPA enclosure; with the undervoltage coil deenergized, the molded case breaker will trip open. This feature is consistent with the current design. The setpoint allowable limits for the EPAs are listed in the Technical Specifications and these limits are not being change per ER 96/0403-00-00. Since Technical Specifications allowable values will not be changed, voltage at the scram solenoids will not be affected. The new EPAs are designed as Class 1E, Seismic Category 1 components to ensure that they will perform as required under the required design basis conditions. The replacement of the EPAs by ER 96/0403 will not alter the ability of the Reactor Protection System to perform its required functions due to the fact that if the RPS power supply system fails, that portion of the distribution system will deenergize and a half scram signal will be created. This is considered a fail-safe design and is not impacted by the replacement of the EPAs. The new EPAs are functionally equivalent to the existing EPAs and are designed to meet the performance requirements of the existing EPAs. The new Electrical Protection Assemblies do not create an Unreviewed Safety Question. Changes to UFSAR Section 8.3.1.1.5.2 and Technical Specifications Bases Section 3.3.8.2 are required in order to update the description of the EPAs. These changes will be incorporated per Licensing Document Change Request 97-048. The installation of ER 96/0403-00-00 will enhance the reliability of the EPAs. This modification does not introduce any activity that will adversely impact the safe operation of the plant.

Serial Number: 98-028-NPE Document Evaluated: ER 96/0577-00-00

DESCRIPTION OF CHANGE: ER 96/0577-00-00 will reassign the upstream isolation for BOP transformer 13 and 23 from the 34.6 kV breakers to the upstream switchers. This will be accomplished by enabling the automatic opening function of switchers 589-1103C and 589-2103C and removing the trip permissive to upstream 34.6 kV breakers 552-1103 and 552-2103 from the transformers' protective relaying. The associated telemetrics will be disabled and abandoned. Originally, these switchers were automatically tripped open upon transformer fault detection, but were disabled via MCP 92/1041 due to high maintenance on a leaking pressurized gas canister. As a result of this modification, the upstream 34.6 kV breakers were the only isolation devices for the transformers.

REASON FOR CHANGE: The telemetrics utilized for the remote trip function from the transformers to the upstream 34.6 kV breakers has been determined to be unreliable and a source for sporadic feeder breaker tripping due to noise induced onto the telemetrics. Should the upstream 34.6 kV breakers sporadically trip open, all power to the radial well switchgear house will be lost. This modification does not affect any text or analysis in the UFSAR. This 10CFR50.59 Safety Evaluation is performed to support of the required UFSAR figure change (Figure 8.3-10b). This figure change consist of deleting the loads from breaker 100 of 1DD6 and breaker 35 of 1DE1. These DC sources/breakers provided power to the telemetric equipment which is being abandoned.

SAFETY EVALUATION: Re-assigning the isolation function by activation of the switchers and de-activation of the trip permissive to the upstream breaker from the associated transformer protective relaying is the most viable option for providing required protection for the transformers while eliminating a source for sporadic breaker tripping. Limiting conditions of operations as defined in the GGNS Technical Specifications will not be affected. The basis for evaluation for any accident as defined in the UFSAR will not be affected. No new conditions are created which may affect any system or equipment important to safety as previously evaluated in the UFSAR. This modification consist of wiring/jumper changes and no new equipment will be installed. All wiring changes will maintain divisional separation per Regulatory Guide 1.75. Consequently, Figure 8.3-10b of the UFSAR requires revision to reflect the removed loads form the respective DC buses. The switchers will not be subjected to operate under conditions/fault currents to which they are not rated. Class 1E power or its sources will not be adversely affected by this modification.

Serial Number: 98-029-NPE   Document Evaluated: ER 96/0559-00-0

DESCRIPTION OF CHANGE: The proposed change will replace the existing ten-disc (five discs per flow, 44-inch last-stage blades) LP No. 2 rotor with an eight-disc (four discs per flow, 46-inch last-stage blades) advanced design rotor to provide efficiency improvements and reduced maintenance requirements which result in cost savings. Also, the first two discs per flow of the existing LP No. 2 turbine rotor are combined to a single disc for the new LP No. 2 turbine rotor. The new rotor discs are made from 3.5% NiCrMoV steel material. This design improvement is expected to increase disc inspection interval up to 100,000 equivalent operating hours. The change is limited to the replacement of the turbine rotor, the inner & inner casing, the stationary blade rings, the diffusers and associated components. Also, the proposed change will increase the number of bolts used in LP shaft seal compensator joints from 20 to 38 and replace the existing gasket with a thicker gasket. The additional bolts are required to achieve the required gasket compression.

Minor modification of the coupling on LP No. 3 (turbine end) will also be conducted to ensure adequate clamping force is provided for the new turbine configuration.

REASON FOR CHANGE: The design of the 1970 vintage nuclear LP turbine was based on extensive experience gained with disc-type rotors of fossil turbines built in the 1950's. In the meantime, Siemens and Kraftwerk Union (KWU) began manufacturing turbines that improved thermal performance, while maintaining and enhancing the already high degree of reliability and availability of their turbines. Siemens Power Corporation (SPC) will supply and install the LP No. 2 turbine replacement components to increase the efficiency of the turbine and to increase the interval between inspections. The proposed change will replace the existing ten-disc LP No. 2 rotor with an eight-disc advanced design rotor to provide additional electrical megawatts for the same reactor thermal output (increase efficiency of the turbine-generator).

Also, the original LP No. 3 coupling (Turbine End) bolt holes were bored before shrinking the coupling on the rotor. Subsequent to the shrinking process, the shape of the pre-bored coupling bolt holes changed. In order to resolve this discrepancy, the existing coupling bolt holes were bored to a larger size to correct the hole shape. By boring the holes to suit the new LP No. 2 turbine, then again when the new LP No. 3 is installed, bolt and nut pressure contact surfaces could be reduced to a point material overload would occur. Modifications to the LP No. 3 coupling will be made to ensure that proper bolt and nut pressure contact surfaces are maintained during the future installation of the upgraded LP No. 3 rotor.

SAFETY EVALUATION: The LP No. 2 turbine (N31D002B) is a part of the turbine-generator (N31 system). The proposed change will affect turbine-generator design parameters listed in UFSAR Sections 1.3, 10.1, 10.2 and 10.4.7. The proposed change will also update information provided in UFSAR Section 3.5.1.3. The change will affect the design information provided for turbine cycle heat balances by increasing generator output as shown on UFSAR Figures 10.1-1 and 10.1-2. However, the change will have no significant affect on interfacing UFSAR Sections for main and reheat steam (N11), heater vents and drains (N23), main and R.F.P turbine seal steam and rains (N33), moistui separator-reheater vents and drains (N35), extraction steam (N36) and turbine bypass (N37) systems. UFSAR Section 3.2 classifies the affected systems (N11, N19, N21, N23, N33, N35, N36 and N37) as "Other" which means that a loss of system function would not affect the safe shutdown of the plant. UFSAR Table 3.2-1 classifies these systems and their associated components as non-safety related, non-seismic, quality group D and ANSI B31.1. No other changes to the existing LP No. 2 turbine configuration will be required. The replacement components supplied by SPC have been designed in accordance with the original standards (German standards) used to construct the existing turbines.

SPC has submitted a missile analysis report for the new LP rotors (FS 4/1018/1995). The report compares disc 1 and disc 4 (end stage disc) of the new LP No. 2 rotor has additional mass, lower average temperature and an additional row of blading. Also, this report shows that the fragment with the maximum translation energy is considered to be the most dangerous, since it is subject to the minimum loss due to friction, and hence the translation energy is the deciding criterion for the penetration of safety barriers. As stated in the report, the translation energy for disc 1 of the new LP No. 2 rotor is 5.7×10^6 Joules, which is lower than the translation energy (10.8×10^6 Joules) of disc 5 of the existing LP No. 2 rotor. Therefore, the LP turbine missile analysis addressed in UFSAR Section 3.5.1.3 is not affected by the LP No. 2 turbine upgrade. The turbine stop and control valve parameters and overspeed protection function are not affected by this modification and therefore do not represent a change to the Technical Specifications or LP turbine missiles analysis (UFSAR Section 3.5.1.3).

Also, the function of the LP No. 3 coupling (Turbine End) to transfer torque from the HP turbine, LP No. 1 and LP No. 2 turbine rotors to the LP No. 3 turbine rotor and to withstand maximum short-circuit torque (without major coupling bolt damage) is not affected by this design modification. SPC has submitted a design report (DG 96/006) which evaluates the coupling of the existing LP No. 3 rotor and the new LP No. 1 and LP No. 2 rotors. The results of the evaluation show that the coupled Turbine Generator rotor system torsional natural frequencies are free from excitable torsional frequencies in the range of 57 HZ to 63 HZ and/or 114 HZ to 126 HZ. Also, the proposed LP No. 2 turbine upgrade components have excellent erosion corrosion (EC) resistance material properties. The design has been evaluated against the applicable design criteria, installation and operational requirements, and all necessary requirements and commitments are met. The change will not affect any equipment important to safety. The modifications made by this design change will not impose a change to the criteria listed in UFSAR Table 3.2-1. The LP No. 1 turbine (FRR No. 11322) modifications information provided by SPC indicates that this ER will not affect any parameters specified in the cycle 10 reload safety analysis (i.e., HP turbine first stage pressure, HP control valve positions, etc.). Other HP and LP turbine parameters, such as extraction steam pressures, will remain within the ranges specified for the existing turbine design. The modification will enhance the turbine efficiency without affecting the operation of the reactor pressure control system.

Serial Number: 98-030-NPE Document Evaluated: ER GGNS-97-0051-00

DESCRIPTION OF CHANGE: This safety evaluation addresses the issues concerning the RFO9 fuel shuffle. These issues include the movement of fresh and irradiated fuel in the containment and spent fuel pool and the storage of fresh and irradiated fuel in the containment and spent fuel pool. This evaluation does not address other core maintenance and inspection activities conducted in conjunction with the fuel shuffle, such as ISI and control blade replacement. This evaluation assumes that such activities are conducted in accordance with applicable Technical Specifications and appropriate work instructions.

REASON FOR CHANGE: Reload fuel is necessary for Cycle 10 operation.

SAFETY EVALUATION: This evaluation confirmed that the current Technical Specification on shutdown margin would be satisfied for all RFO9 interim core configurations. The Technical Specification fuel pool criticality requirements would also be satisfied in both the upper containment pool and the spent fuel pool (including the blackness test area).

The fuel and core parameters assumed in the current fuel handling accident analysis were confirmed to be bounding for Cycle 9 and RFO9 such that the existing fuel handling accident remains applicable to the RFO9 core shuffle. The light load curves, which are based on the fuel handling accident, also remain applicable for the RFO9 core shuffle. The Cycle 10 reload fuel assemblies are of the same mechanical design as the Cycle 9 reload bundles. The existing seismic and structural analyses for the core and the racks remaining bounding for the new fuel design.

Serial Number: 98-031-PSE   Document Evaluated: TSTI 1C11-96-001-0-S

DESCRIPTION OF CHANGE: UFSAR Section 4.6.1.1.2.4.1b describes current stabilizer operation as 4 GPM to insert and 2 GPM to withdraw a control rod. UFSAR Section 4.6.1.1.2.4.2.3 describes the total flow rate through the stabilizer valves as 16 GPM (sum of flows needed to insert and withdraw 4 control rods). This safety evaluation is necessary to evaluate changing these UFSAR values via TSTI to determine feasibility of a design modification to restore CRD system operation in the gang drive mode and evaluate the effects of the change on system operation during normal plant operation. This TSTI will increase stabilizer flows to approximate the average CRD driven flows. This will reduce the Drive Water Pressure drop when a CRD is moved in gang drive which should reduce CRD movement problems in gang drive and restore the gang drive mode of the CRD system. The purpose and function of the stabilizer valves will not change. However, the stabilizer flows will be readjusted for the purpose of the TSTI to determine if increased stabilizer flows should become a permanent plant design change. Insert total flow will be increased from 4.0 to 5.2 GPM per drive. Withdraw flow will be increased from 2.0 to 4.2 GPM per drive. The total Stabilizer flow will change from 16.0 to 20.8 GPM. Increasing CRD Stabilizer Flows will slightly increase CRD stroke speeds in both the insert and withdraw mode. To ensure there will be no adverse effects, TSTI will verify proper single and continuous CRD movements in individual and gang drive modes at established test conditions that ensure no core operating limits will be violated. The TSTI has also added specific instructions to Operations to cover the time period during and after the changes to the stabilizers valves. After the changes are complete, the ability to insert rods will not be significantly affected, only the speeds will be increased. This change has absolutely no negative effects on the ability to insert rods should plant conditions required such actions. During the time that the changes are being made, specific precautions and instructions to the Operators have been included in the TSTI to increase drive pressure per existing plant procedures to insert control rods if needed. The ability to shutdown the plant via scram has not been affected by this change. The stabilizer flow settings will be returned to their nominal pre-TSTI values at the completion of the TSTI. The effect of the change will be evaluated to determine if the changes should become part of the permanent plant design.

REASON FOR CHANGE: Current control drives have seal leakage of 4 to 5 GPM on the average with some being even higher. The stabilizer valves were set for new drives with low leakage, on the order of 2 to 4 GPM. The current stabilizer settings are 4 GPM to insert each rod and 2 GPM to withdraw which results in a total flow setting of 16 GPM. With existing drive leakage, the net effect of trying to move rods in a gang is that the stabilizer valves do not pass enough flow to keep the drive pressure constant. This results in the pressure through the pressure control valve dropping when the rods are given a movement signal. The drive pressure drop is large enough that the entire gang of up to 4 rods will not withdraw or notch out in unison. Often none or may be one of the rods actually withdraw or notch out when commanded to do so in gang mode. The operators then have to select and move each rod individually to complete the step. Most times, operators will not even attempt to use gang mode since they know it almost never works. This change will attempt to restore the gang mode of the CRD system by increasing the stabilizer flows to maintain drive pressure closer to initial system setting.

SAFETY EVALUATION: No change to Technical Specifications nor the TRM is necessary as stabilizer flows are not specifically addressed in these documents. The expected drive pressure increase and resulting increased stroke speeds are not addressed in these documents either. The expected pressure increase (322 psid) is bounded by currently approved maximum allowable drive pressure of 475 psid (see SE 95-0072-R00). The increased change of inadvertent over-notching a rod past its indented position during withdrawals is bounded by the Rod Withdrawal Error (RWE) analysis (UFSAR Section 15.4.2). Engineering experience and estimates based on the proposed change indicate the worst case over-notching will be below the Rod Withdrawal Limiter (RWL) allowed range of 4 notches below the HPSP which is based on the RWE analysis (UFSAR Section 15.4.2).

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

Serial Number: 98-032-NSR Document Evaluated: LDC 98-004

DESCRIPTION OF CHANGE: This change will eliminate the TRM requirement to submit special reports for valid/invalid EDG failures. It will also withdraw GGNS commitment to Reg. Guide 1.108, which contained the special report requirements, and add Reg. Guide 1.9, Rev. 3. Appendix 3A of the FSAR will reflect the Reg. Guide changes. Sections of the FSAR, TRM, and Technical Specifications bases that reference Reg. Guide 1.108 will be changed.

REASON FOR CHANGE: Under the Maintenance Rule, EDG reporting requirements fall under 10 CFR Part 21 and 10 CFR 50.72 and 73. Special Reports are an unnecessary duplication. The special reporting requirements were contained in Reg. Guide 1.108 Reg. Guide 1.9, Rev. 3 encompasses all other pertinent requirements of Reg. Guide 1.108.

SAFETY EVALUATION: Approval to remove the special report requirements for the EDG is contained in Generic Letter 94-01. The one condition stated in the generic letter that must be meet in order to delete the special report requirement was implementation of the Maintenance Rule for the EDGs. GGNS has meet this stipulation. Therefore, the special report requirement for EDGs is being deleted.

Serial Number: 98-033-NSR   Document Evaluated: LDC 98-025

DESCRIPTION OF CHANGE: Change the note in TRM SR TR3.8.1.2 from "This surveillance shall not be performed in MODE 1, or 2" to Inspections that call for significant engine internals disassembly or require a retest that cannot be performed on-line, shall not be performed in MODE 1, or 2.

REASON FOR CHANGE: As applied to the existing maintenance programs, the TRM SR TR3.8.1.2 wording is confusing and needs to be clarified. No distinction is made between inspections that can be performed on-line and inspections that can only be performed during an outage (Refuel). The proposed change will assure the vendor's recommended maintenance programs are implemented in a manner that is in accordance with the vendor's intent.

SAFETY EVALUATION: TRM SR TR3.8.1.2 requires implementation of the vendor's recommended maintenance program for the diesel generators. The proposed change does not alter the effectiveness of the TRM SR TR3.8.1.2 requirement. The vendor recommended maintenance program will still be implemented as specified by the vendor. As the proposed change specifies that scheduled preventative maintenance activities requiring significant engine internals disassembly cannot be performed in Mode 1 or 2, the original intent of the note is retained. The specified frequency of TRM SR TR3.8.1.2 is every 18 months. The current vendor program provides greater flexibility for performance of engine maintenance/inspections. The current program emphasizes that activities be performed within the specified frequency and does not restrict EDG maintenance/inspections to the scope of a refueling outage. The proposed change is considered a clarification only. Any maintenance inspections will be bounded by Technical Specification LCO times.

The diesel generators are not initiators for any accident analyzed in the SAR. However, the consequences of a diesel generator failure are addressed in the SAR. Maintenance is required to be performed throughout the cycle to properly maintain diesel generator reliability. Existing controls assure that the conduct of maintenance is properly assessed and reliability is balanced against unavailability. Thus the proposed change does not introduce any new failure mechanisms not previously considered by the SAR. As no new failure mechanisms exist, the probability and possibility of accidents or malfunctions are unaffected. This change is not a test or experiment. No new SSCs are involved. No impact on the operation of any SSC will occur, and neither the quantity or quality of any effluents are impacted. No increase in dose will occur. Accordingly, the proposed change does not constitute an Unreviewed Safety Question or change to the Technical Specifications, and does not impact the Environmental Protection Plan.

Serial Number: 98-034-NPE Document Evaluated: ER 96/0711-00-00

DESCRIPTION OF CHANGE: ER 96/0711-00-00 provides the instructions and details to remove and/or abandon the differential temperature switches and associated thermocouples utilized for the logics of RHR, RWCU, RCIC and Main Steam Line tunnel isolation circuits. These devices and equipment have been removed from the GGNS Technical Specifications and added to the TRM. These inputs have been evaluated and are not required for system performance and the current isolation functions where used. This ER affects only the differential temperature isolation functions and related annunciator and recorder inputs. Revisions to the UFSAR to reflect the removal or spare status of related equipment include numerous Section and Table revisions, as well as revisions to System Figures 7.6-2a, 7.6-2b, 7.6-4, and Figure 7.6-017 (P&ID M-1090B), Figure 9.4-003 (P&ID M-1104B), Figure 9.4-010 (P&ID M-1103A) and Figure 9.4-12 (P&ID M-1100B). TRM changes are also required to delete the differential temperature switch information from Section TR3.3.6.1 and Tables TR3.3.6.1-2 and TR3.3.6.1-3. Technical Specification Bases Section B3.3.6.1 will also be revised to clarify discussion of RWCU area flow/temperature instrumentation correlation.

REASON FOR CHANGE: Failures of the Panalarm (Riley) temperature switches have repeatedly caused RHR, RWCU, RCIC and half MSIV system isolations. Despite installation of upgraded switch models, failures have continued to occur in these applications. Removal of the differential temperature switches from the isolation logics will improve plant reliability by reducing system isolations and the possibilities of MSIV isolation. The isolation functions and testing requirements for these switches have been deleted from the GGNS Technical Specifications by Amendment 120 as a result of analysis performed by Engineering Report GGNS-90-0024.

SAFETY EVALUATION: Safety Evaluation Number 95-0014-R00 was previously issued and approved for Improved Technical Specification TRM changes related to the deletion of Main Steam Line, RCIC, RWCU and RHR area differential temperature monitoring surveillances and testing. NRC approval of T/S Change Request 90/0003 and incorporation into the GGNS Technical Specifications was completed by Technical Specification Amendment 120. Evaluation of the Leak Detection and isolation features of the switches has concluded that there will be no impact on any design basis analysis performed for GGNS. Analyses for the applicable areas do not take credit for the differential temperature instruments for detection or isolation of affected area pipe ruptures. None of their functions are assumed or credited for mitigation of consequences of any UFSAR analyzed event. Expected radiological consequences following a pipe rupture are bounded by existing design basis analysis. There are no pressure/temperature analyses, radiation dose calculations or equipment qualification parameters that take credit for the operation or performance of the differential temperature switches. These modifications will not result in any unreviewed safety question.

Serial Number: 98-035-NPE   Document Evaluated: ER 97/0089-00-00

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

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

Analysis of change: This Safety Evaluation evaluates the effects of the new strainer on the ability of the ECCS to satisfy the requirements of 10 CFR 50.46, and the analytical methods used to evaluate hydrodynamic loads on the strainer. Additionally, this evaluation covers the full extent of physical installation issues including UFSAR changes and Technical Specification Bases changes. See Appendix A (starting on Page 22) for a detailed discussion of issues related to the installation of the new ECCS/RCIC suction strainer. Appendix B (starting on Page 71) provides a detailed discussion of the affect of this design change on the original evaluation of Humphrey concerns affect on the design of the new ECCS/RCIC strainer. Appendix C (starting on Page 143) provides a detailed evaluation of the change with respect to requirements presented in RG 1.82, Rev. 2 "Water Sources for Long Term Recirculation Cooling Following a Loss of Coolant Accident". See LDCR 97-074 for a discussion of UFSAR and Technical Specification Bases changes.

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

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

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

- Two Low Pressure Coolant Injection Subsystems (Division II)

- High Pressure Core Spray (Division III)

Equipment in each group is independent from the other groups. In addition, the HPCS and RCIC systems are independent from each other to provide additional diversity for high pressure water sources, and to provide single failure protection for the control rod drop accident (Refs. 6.3.1.32, 6.3.1.34, 6.3.1.81).

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

This change maintains/improves the current design basis performance of a safety system assumed to function in the accident analysis and does not decrease the reliability of safety systems assumed to function in the accident analysis.

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

This change does not adversely affect the overall ECCS or RCIC systems performance or reliability in a manner that could lead to an accident occurring. This change does not cause the ECCS or RCIC systems to be operated outside their design basis limits, i.e., the environmental conditions, seismic, hydrodynamic and other applicable loads, and system NPSH requirements have been considered in the new strainer design. The new strainer cannot affect any system interface in a way that could lead to an accident. The new strainer will not result in degradation of safety systems. To the contrary, it is intended to improve the availability of the ECCS and RCIC systems by providing a mechanism to reduce the possibility of system unavailability. Because the new strainer is passive, no operator error has been introduced. Additionally, the margin of safety as defined in the bases for the Technical Specifications has not been reduced.

Serial Number: 98-036-NPE Document Evaluated: Engineering Report GGNS-94-0039-R2

DESCRIPTION OF CHANGE: This safety evaluation assesses a comprehensive study of the safety functions for the feedwater check valves (FWCVs B21F010A/B and B21F032A/B) as determined by a thorough characterization of the operational conditions for these functions (Engineering Report GGNS-94-0039, Rev. 2). Testing of the B21F010A/B and B21F032A/B is performed per the surveillance requirements of Technical Specification 3.6.1.3, Primary Containment Isolation Valves (PCIVs). The valves are leak tested to the criteria of Appendix J to 10 CFR 50 for containment isolation valves, which requires a Type C Local Leak Rate Test (LLRT) at a differential test pressure of ≥ 11.5 psig. An LLRT must be performed every outage. However, a thorough engineering evaluation of the functional requirements for the FWCVs has concluded that the inboard FWCVs do not perform any functions for which Appendix J leak testing need be imposed. Therefore, this safety evaluation provides the basis for eliminating leak testing requirements for the inboard FWCVs (B21F0101A/B). All other testing requirements for these valves (i.e., functional testing per ASME Section XI) are not affected by this change.

This change modifies the UFSAR and TRM to delete the requirement to perform 10 CFR 50, Appendix J, Type C liquid leakage testing for the inboard feedwater check valves. It has been determined that the feedwater penetrations are sealed with a qualified seal system for 30 days and the inboard containment isolation valves (CIV's) do not constitute credible primary containment atmospheric leakage pathways during and following a Design Basis Accident (DBA). Therefore, Type C liquid leakage testing of these valves is not required by Appendix J to ensure that post-accident radiological releases from the containment are consistent with the accident analysis and remain bounded by the applicable licensing acceptance limits. Thus, there are no analytical leakage limits associated with the inboard feedwater check valves that warrant leak rate testing. Functional testing will continue to be performed to ensure valve operability and position indication in accordance with ASME Section XI (i.e., valve exercising open and close).

REASON FOR CHANGE: A thorough engineering evaluation has determined that the post-accident performance requirements for the inboard FWCVs do not include any containment isolation functions for which 10CFR50 Appendix J leakage criteria need be applied. Therefore, the Appendix J leakage testing requirements will no longer be imposed on the inboard FWCVs so as to:

1. establish testing requirements commensurate with the actual component functions

2. test the valves in a manner consistent with that of other isolation valves with similar functional performance requirements

3. reduce personnel radiation exposures associated with unnecessary valve leakage rate testing.

These valves will continue to be functionally tested in accordance with ASME Section XI requirements.

SAFETY EVALUATION: The Code of Federal Regulations, 10 CFR 50, Appendix J establishes requirements for containment leakage testing for all operating licensees of water cooled power reactors. Three tests are specified in the regulation: Type A (integrated leakage), Type B (penetration leakage), and Type C (containment isolation valve local leakage). A containment isolation valve (CIV) is defined in Appendix J as "any valve which is relied upon to perform a containment isolation function." Containment is defined as "...an essentially leak tight barrier against the uncontrolled release of radioactivity to the environment." Therefore, for the purposes of Appendix J leakage rate testing, an Appendix J CIV is a valve which must isolate a potential fission product release pathway to the environment following a postulated accident. Consequently, the CIV's leakage must be maintained within the allowable limits established by the applicable accident analyses.

As part of the Appendix J, Option B rulemaking, the NRC has endorsed the NEI industry guidance (NEI-94-01, which was Regulatory Guide 1.163, September 1995). This guidance states that a Type C leakage rate test is not required for "Primary containment boundaries that do not constitute potential primary containment atmospheric pathways during and following a Design Basis Accident (DBA); {or} Boundaries sealed with a qualified seal system..." The guidance specifies ANSI/ANS 56.8-1994, "Containment System Leakage Testing Requirements", as well as the current licensing basis for each plant, to define specific testing requirements. The standard also specifically exempts "primary containment boundaries that do not constitute potential primary containment atmospheric pathways during and following a DBA" from Type B or Type C testing.

ANSI/ANS 56.8-1994 paragraph, 3.4, "Qualified Seal System Testing Requirements" provides the guidelines for testing a qualified seal system which states:

> "Primary containment barriers sealed with a qualified seal system are not required to be local leak rate tested. If a seal system is used as primary containment barrier, it shall be periodically tested to prove its functionality. This functional test shall demonstrate that the seal system is capable of sealing the primary containment barrier(s) with the sealing liquid at a differential pressure of not less than $1.1P_a$ for at least 30 days following a DBA. Qualified seal system testing is specified in the plant's licensing basis."

The feedwater leakage control system (FWLCS) is functionally tested by local leak rate testing the 1B21F032A/B and 1B21F065A/B (to ensure that FWLCS leakage is less than the 1 gpm analytical limit) and by the ASME Section XI Pump and Valve testing program.

Therefore, the establishment of Appendix J leak testing criteria for the FWCVs is based on the functional performance requirements of the FWCVs as determined by the applicable accident analyses. The FWCVs perform active safety-related design functions to:

- prevent the loss of reactor coolant inventory during certain feedwater line breaks (i.e., containment isolation function via reactor isolation)

- prevent the transport of radioactive material through the feedwater leakage pathway (i.e., containment isolation function) during the short-term period following an accident involving core damage (e.g., DBA-LOCA) and during operation of the Feedwater Leakage Control System (FWLCS).

The analyses summarized in this engineering report evaluated all postulated accident conditions associated with the containment isolation functions in which the FWCV allowable leakage must mitigate the consequences of an accident to within the limits of 10CFR100. The primary analyses performed in support of this engineering study were:

- a comprehensive transient thermal-hydraulic analysis of the DBA-LOCA with feedwater line blowdown (Ref. 1)

- a post accident suppression pool inventory analysis (Ref. 2)

- a radiological dose analysis for a worst case (i.e., inside the drywell) feedwater line break (Ref. 3)

- a radiological dose analysis of the post-LOCA operation of the FWLCS (Ref. 4)

- a comprehensive evaluation of potential containment leak paths (Ref. 5)

- a seismic capabilities evaluation of the feedwater system (Ref. Engineering Report GGNS-94-0039, Rev. 2, Appendices B, C, and D).

The analysis results show that for the nonlimiting feedwater line break accidents, complete failure of the containment isolation function of the FWCVs (i.e., gross leakage through all feedwater leak pathway isolation valves) will not result in offsite and control room dose consequences exceeding the applicable regulatory limits. For accidents such as the limiting DBA-LOCA, no leak path exists through the feedwater lines during the reactor blowdown phase since the direction of flow for the steam/liquid mixture is only from the feedwater lines into the reactor. Following the blowdown phase and prior to the initiation of the FWLCS (i.e., the leak phase), sufficient subcooled water remains in various portions of the feedwater piping to form liquid water loop seals that effectively isolate this leak path. The feedwater leak paths remain isolated by these loop seals until the FWLCS has been initiated and refloods the containment portions of the feedwater lines with suppression pool water. The reliance on non-safety related FW piping and components following DBA events meet the applicable NRC requirements within the current license commitments and regulations.

The inboard FWCVs are located inside the containment (specifically the drywell) and are in a system protected against missiles and pipe whip, designed to seismic Category I requirements, and classified as Quality Group A (per Regulatory Guide 1.26). The system and piping will not be adversely affected by single active failures. Under the existing licensing basis, a pipe rupture of the seismically qualified feedwater piping does not have to be assumed concurrent with a LOCA, except as direct consequence of the LOCA. Consideration of consequential failures of the feedwater system from LOCAs outside of containment are beyond the Appendix J design considerations. A single active failure of the CIV, under LOCA conditions, will be accommodated during the feedwater piping depressurization with the FWLCS providing a leakage barrier for the containment atmosphere.

These valves are in a system that is monitored for leakage as part of the ASME Section XI Pressure Testing program. Program elements for feedwater piping in the drywell steam tunnel include visual examinations during outages, periodic leakage tests, a corrective action program to correct leakage problems, and preventative maintenance activities. The purpose of the program is to detect and correct degradation of the pressure boundaries of the systems, thereby reducing potential post-accident releases and the resultant dose consequences.

The FWLCS is designed to prevent the release of radioactivity through the feedwater line isolation valves by providing a continuous flow of water through the feedwater lines following a loss of all offsite power coincident with the postulated design basis loss-of-coolant accident. The FWLCS consists of two independent subsystems designed to eliminate through-line leakage in the feedwater piping by providing a positive seal between the containment isolation check valves and the outboard isolation valve. The outboard subsystem uses residual heat removal (RHR) jockey pump "A" and the inboard subsystem uses RHR jockey "B" to supply sealing water on the upstream and downstream sides of the outboard containment isolation check valve, respectively. Following a LOCA, the FWLCS is manually initiated from the control room. The sealing water from each jockey pump is routed to both feedwater lines. The sealing fluid from the RHR jockey pump "B" discharge line fills the feedwater line between the containment isolation check valves. The sealing water through the valve eventually fills the feedwater line up to the reactor vessel and finally the water returns to the suppression pool through the LOCA break. Since the source of sealing water is the suppression pool, a 30 day water supply is assured.

The feedwater line break inside containment is assumed to occur as described in the GGNS initial cycle analysis presented in the UFSAR. The break size is based on the inside area of the feedwater sparger piping which provides the limiting flow area for the break. The event assumes a loss of offsite power combined with the worst single equipment failure, which for this event is a failure of the HPCS system. The vessel depressurization and inventory loss results in partial core uncovery with peak cladding temperature remaining below fuel melt limits throughout the event and no significant fuel damage occurs. For conservatism, the dose analysis for this postulated event uses license basis requirements in which all of the exposed cladding is assumed to fail. This event is categorized as a limiting fault.

The results of this evaluation concluded that inboard FWCVs perform reactor vessel pressure isolation functions between the containment and the feedwater system which is outside the containment boundary but are not considered fission product release path barriers. The accident analysis does not explicitly credit the isolation of these valves, but rather considers potential liquid leakage (suppression pool water containing accident source terms) from this system through pathways such as pump seals and valve packing. The feedwater system has been evaluated as a physical barrier that is credited with limiting leakage outside primary containment per the applicable regulatory requirements. The feedwater system from the RPV to the outboard motor operated isolation valves (1B21F065A/B) are visually inspected for pressure boundary leaks per the ASME Section XI pressure testing program. The accident analysis treatment of these potential leak paths and the associated testing methods are appropriate for this system. The affected valves perform no other safety function except for reactor vessel pressure isolation (i.e., containment isolation) during certain feedwater line break events. However, the radiological consequences for these events are such that Type C testing or ASME leakage testing are not required since the applicable licensing acceptance limits are met without any reliance on the FWCV isolation function.

Therefore, this safety evaluation concludes that the proposed change to the leak testing criteria for the inboard FWCVs is acceptable and does not involve any unreviewed safety question.

Serial Number: 98-037-NPE Document Evaluated: ER 97/0352-00-00

DESCRIPTION OF CHANGE: This design change will increase the air flow of the Q1T51B002 room cooler fan from a design value of 8000 CFM to a new design value of 9100 CFM by increasing the angle of attack on the fan blades.

REASON FOR CHANGE: Based on the increased cooling duty load requirements for the Q1T51B002 LPCS pump room cooler from MNCR 94-0028, there is a 1% margin between the cooling duty load and the design cooling capability of the room cooler. The increased air flow will result in an increased design cooling capability for the LPCS pump room cooler and increased margin.

SAFETY EVALUATION: Increasing the air flow of the LPCS pump room cooler fan Q1T51B002 will increase the energy being removed from this room by the cooler and rejected to the SSW system (Ultimate Heat Sink). The rejection of additional heat from the LPCS pump room will result in a lower temperature not only in the LPCS pump room but also in the surrounding compartments due to increased heat rejection to the LPCS pump room or decreased heat rejection from the LPCS pump room. This condition is bounded by the existing analysis for the Auxiliary Building temperatures which uses the lower existing heat rejection rate for the LPCS pump room cooler.

The diesel generator loading calculations will not be affected by this modification as the existing calculations use the motor nameplate data. The thermal overloads for the motor are also set based on nameplate data and the current drawn is verified to be within these bounds.

NPE has evaluated the additional heat rejection to the SSW system (Ultimate Heat Sink) and concluded that it is bounded by the existing Ultimate Heat Sink Analysis. This analysis assumes that the design heat rejection capability of the serviced heat exchangers is the cooling duty load of the Ultimate Heat Sink.

Serial Number: 98-038-NPE Document Evaluated: ER 97/0052-00-00

DESCRIPTION OF CHANGE: This ER (ER 97/0052-00-00) extends the disc stops on Feedwater Check Valves 1B21F010A and 1B21F010B. The valves, which are located in the Drywell, are Powell 24"X 20" X 24", 900lb, lift check valves. These valves serve as primary containment inboard isolation valves for the Feedwater System. The new disc stops will be constructed of SA-516 Gr. 70 carbon steel, which is the same type and grade steel as the valve cap.

REASON FOR CHANGE: Feedwater Check Valves 1B21F010A and 1B21F010B have experienced wear to the internal disc guides. The current valve design includes 2" long disc stops on the backside of the valve disc. These stops contact the valve cap when the disc is pushed to the full travel position by the feedwater flow. It was suspected, and later confirmed by analysis, that the flow through the valve did not produce enough lift to maintain the disc stops in contact with the valve cap with sufficient force to prevent disc fluctuations which results in excessive guide wear. The analysis performed by KALSI Engineering, M-97/0052-Q1B21F010-8.0-1-0 (Analysis to Improve Performance of Feedwater Lift Check Valves at GGNS), indicates that an extension of the disc stops (from 2" to 8½) will cause the disc to hit the stops during normal operation which will prevent disc fluctuations and thus prevent excessive guide wear.

SAFETY EVALUATION: The intended purpose for the installation of the extended disc stops is to increase the reliability of Feedwater Check Valves 1B21F010A&B. This change does not degrade below the current design basis the performance of a safety system assumed to function in the accident analysis and does not decrease the reliability of safety systems assumed to function in the accident analysis.

The change does not cause a greater reliance to be placed on any specific system, structure or component to perform a safety function. A failure of one of the new disc stops will have the same result as the failure of the existing disc stops. The installation or failure of the new disc stops will not increase the probability or consequences of analyzed failures. No increase of either the offsite or the onsite radiation does would result because of a failure of a new disc stop.

The installation of the new disc stops has been analyzed for its impact on the ability of Feedwater Check Valves 1B21F010A&B to perform their safety related function (primary containment isolation valves), and the modification has been found not to adversely impact this safety related function. The extended stops are qualified for the design loads.

This change does not adversely affect the overall performance or reliability of a safety system in a manner that could lead to an accident. This change does not cause a safety system to be operated outside of its design basis limits. The new disc stops cannot affect any system interface in a way that could lead to an accident. The new Feedwater Check Valve 1B21F010A&B disc stops will not result in degradation of safety systems. The change is intended to improve the reliability of these primary containment isolation valves by providing a mechanism to reduce the possibility of component unavailability. Additionally, the margin of safety as defined in the bases for the Technical Specifications has not been reduced.

Serial Number: 98-039-NPE Document Evaluated: LDC 98-028

DESCRIPTION OF CHANGE: This evaluation addressed changes to the UFSAR to incorporate the findings in Engineering Report #98-0029 which justifies reduced separation for certain cable and raceway configurations. The new reduced separation requirements are being included in the SAR in accordance with Regulatory Guide 1.75 Revision 1.

The specific changes to the GGNS UFSAR are as follows:

The wording,

"Reduced spatial separation distances less than those specified above are contained in Table 8.3-13 for Grand Gulf specific configurations, "

will be added to the end of Sections 8.3.1.4.1.c and 8.3.1.4.1.d of the UFSAR.

Additionally, a new table, Table 8.3-13 will be added to Section 8.3 of the UFSAR. This table is contained in Attachment 2 to this Safety Evaluation.

REASON FOR CHANGE: GGNS is incorporating the findings in Engineering Report #98-0029 which justifies reduced separation for certain cable and raceway configurations. As per Regulatory Guide 1.75 Revision 1 which allows a reduction in separation based upon test analyses, any such changes are required to be submitted with the UFSAR.

SAFETY EVALUATION: These changes to the GGNS UFSAR are being made to comply with the requirements of Regulatory Guide 1.75 Revision 1 to which the plant is committed through Appendix 3A in the UFSAR. Section C.6 of this regulatory guide states:

"Analyses performed in accordance with (Section 5.1.1.2 of IEEE 384-1974) should be submitted as part of the Safety Analysis Report and should identify those circuits installed in accordance with these sections."

Section 5.1.1.2 of IEEE 384-1974 allows the minimum electrical separation distance to be established by analysis based upon tests of cable installations.

Through these changes, GGNS is incorporating the findings in Engineering Report #98-0029 which justifies reduced separation for certain cable and raceway configurations.

These changes to the GGNS UFSAR will allow for reduced separation between cables for the configurations found to be acceptable in Engineering Report #98-0029.

These changes involve neither modification to plant hardware nor change in plant operations. The capability of safety-related systems required to respond to transient or accident conditions are not adversely impacted by these changes. These changes have no impact on the basis for any Technical Specification at GGNS. Therefore, these changes do not involve an unreviewed safety question.

Serial Number: 98-040-NSR Document Evaluated: LDC 98-029

DESCRIPTION OF CHANGE: The TRM/UFSAR Appendix 16B Surveillance SR 6.9.5.1.b is being deleted. The current surveillance requires an every 12 hour verification of the monorail auxiliary hoist override switch to be in the 1000-pound or 500-pound position during the hoist operation for handling of new fuel assemblies. Also, The Note for surveillance SR 6.9.5.6 is changed to read "Not required to be performed when the load override switch on the monorail auxiliary hoist is in the 500-pounds positions".

REASON FOR CHANGE: The monorail auxiliary hoist override switch has only two settings. It can either be set at 1000-pound position or 500-pound position. SR 6.9.5.1.a requires the switch to be in the 500-pound position when handling irradiated fuel assemblies or control rods and verified every 12 hours. This requirement is not being changed. Since the switch has only two settings, the surveillance SR 6.9.5.1.b to verify the switch in 1000-pound position or 500-pound position can never fail. Therefore, the surveillance to verify either position is considered unnecessary. The Note for SR 6.9.5.6 1000-pound functional test is being revised to state that the surveillance is not required when the override switch is in the 500-pounds position. This also clarifies that the 1000-pound function is still required for compliance with LCO operability requirements.

Furthermore, there is no documented technical basis for the above surveillance requirements.

SAFETY EVALUATION: The interlock affected by this change primarily protects refueling equipment, fuel storage racks or fuel assemblies during refueling operations. It also protects the monorail auxiliary hoist by preventing lifting of loads above its design capacity.

The failure of the interlock function caused by the monorail auxiliary override switch settings is evaluated for the Fuel Handling Accidents (FHA) in the auxiliary building. The interlock function is not an event initiator or mitigator in the postulated FHAs discussed in UFSAR Sections 15.7.4 and 15.7.6 or in any of the reactivity events discussed in UFSAR Section 15.4.1.1. This interlock is credited in the analysis presented in UFSAR Appendix 9D "GGNS Compliance with NUREG-0612, Control of Heavy Loads at Nuclear Power Plants". This analyses does credit the existence of this interlock to prevent the monorail hoist from being of concern for drop of heavy load. But this analysis does not credit any specific surveillance. The change eliminates every 12 hours surveillance requirement to verify the monorail auxiliary hoist override switch to be in the 1000-pound or 500-pound position, during hoist operation, while handling the new fuel assemblies. Since the override switch has only two settings, the verification of the either positions can never fail. Also, the Note for SR 6.9.5.6 is being clarified to read "Not required to be performed when the load override switch on the monorail auxiliary hoist is in the 500-pounds position".

Since this surveillance is only a verification of switch position and not a function of the subject interlock, there is no affect on the frequency of the failure of this equipment.

Since no change is made in the design or operation of the hoist, the interlock will continue to provide the intended equipment protection after the change.

The surveillance affected by this change were removed from the Technical Specifications via Amendment #120. Therefore, this change does not require a change in the Technical Specifications.

The proposed change does not result in an unreviewed safety question.

Serial Number: 98-041-NSR Document Evaluated: LDC 98-009

DESCRIPTION OF CHANGE: Revise [illegible] commitment to perform valve line-up checks on all safety related systems during each refueling outage. That commitment was made in response to I&E Bulletin 79-08 and is repeated in Section 18.1.29 of the UFSAR. The new commitment as described in the proposed change to the UFSAR in LDC 98-009 will be to perform such checks on the ESF systems during the refueling outage and allows the checks for the other safety related systems to be completed during the operating cycle.

REASON FOR CHANGE: The existing commitment had been made in response to IEB 79-08, which was intended to assure the functionality of ESF systems. The original commitment went beyond this intent; this change is being made to reduce the level of effort required during a refueling outage. The change is still considered to exceed the scope and intent of the original issue described in the IEB.

SAFETY EVALUATION: The change has been evaluated. The change has been determined to not involve an unreviewed safety question and does not involve any changes to the Technical Specifications.

Serial Number: 98-042-NPE   Document Evaluated: ER 97/0693-00 R2

DESCRIPTION OF CHANGE: In July 1997, GGNS issued a Condition Report (CR) 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 lead to localized heating of the containment wall surrounding the penetrations. This evaluation provided minimum PSW flow s and piping line-ups to ensure containment wall temperatures were maintained below existing FSAR limits. As part of the rework procedures, Plant Staff was to measure PSW flow to the coolers to verify compliance with design. These flows could not be accurately measured during RFO9. In light of this, cooling water supply to Penetration 83 coolers will be changed to Plant Chilled Water. Penetration 88 cooling water supply will not be changed but has been evaluated for zero flow condition at maximum temperatures during RWCU Pre-pump operation. It has been earlier determined that cooling is only required to Penetration 88 during RWCU Pre-pump operation. Penetration 87 jacket cooler was earlier determined not to require cooling water flow and it will not be considered as part of this evaluation.

Chilled water will be supplied by connecting a flexible metal braided hose to existing PCW piping on the discharge piping of the Auxiliary Building Steam Tunnel Cooler, N1T41B011. The hose will be routed to the jacket cooler, N1G33B003C, and returned to the N1T41B011 cooler discharge line by an identical hose. A control valve will be installed between the supply and discharge connection to divert PCW to the jacket cooler. Adequate isolation valves will be installed to facilitate system operation. All components, except the braided hose, installed in the supply and return line will conform to JBD piping class (ANSI B31.1, Primary pressure 125psi @ 350°F) as specified in MS-03. The flexible hose does not conform to the JBD pipe class requirements, but has a working pressure and temperature well in excess of those of the piping and is considered acceptable. All new piping, components, and hoses are considered non-safety related and non-seismic. The routing of the piping and hoses is such that they will have no adverse interaction with safety related system, structures, or components.

The CTMT wall concrete was evaluated for potential degradation due to operation of Penetration 88 jacket cooler, N1G33B003B, with no flow during RWCU Pre-pump mode. NPE determined in Engineering Report 98-0035, Rev. 0 that any further degradation to the concrete compressive strength would be small and would remain within acceptable limits.

REASON FOR CHANGE: A condition report documents PSW system flow rates below design limits because of pipe fouling. Attempts to clean the piping during RFO9 have not been successful. To provide cooling to Penetration 83, the cooling water supply to the penetration cooler will be changed to Plant Chilled Water. Since cooling to Penetration 88 is only needed in RWCU Pre-pump mode, no change will be made to enhance flow to the cooler until RFO10 and NPE has evaluated effects to the concrete for the short period of time Containment concrete temperatures will exceed the FSAR limit of 200°F.

SAFETY EVALUATION: Safety Evaluation 98-0034-00 provided a history of the operation of the jacket coolers for Containment Penetrations 83, 87, and 88. The previous calculations, required flow rates, and degradation of Containment wall concrete compressive strength were evaluated based on a design that supplied PSW as a source of cooling water to jacket coolers. PSW piping is plugged and adequate flows can not be established. To ensure adequate flow, Plant chilled Water will be used as a supply of cooling water to Penetration 83. PCW will be able to provide required heat transfer requirements and flow to ensure concrete temperatures are below the FSAR limit of 200°F. To accomplish this, the existing PSW piping to Penetration 83 will be removed and isolated on either side of the jacket cooler, N1G33B003C, at existing isolation valve. Plant Chilled Water will be supplied by attaching a flexible metal braided hose to existing PCW piping downstream of the Auxiliary Building Steam Tunnel Cooler, N1T41B011. The hose will be routed to the cooler and returned to the N1T41B011 cooler discharge line by an identical hose. A control valve will be installed between the supply and discharge connection to divert PCW to the jacket cooler. The new piping and hose will be routed to prevent any adverse interaction with any safety related systems, structures, or components. Included in the design of the PCW to the jacket cooler will be a temperature element, N1P71N014, to allow monitoring of the PCW supply temperatures. This element will be used to verify that PCW supplied to the jacket cooler is not below the Lowest Service Metal Temperature (LSMT). Operator action will preclude operation of RWCU should the penetration fall below the LSMT.

No change to cooling water flow to Penetration 88 cooler will be considered and the flow rate is conservatively assumed to be zero. Due to the short period of time the surrounding concrete will be above the FSAR limit, NPE determined in Engineering Report 98-0035, Rev. 0 that any further degradation to the concrete compressive strength would be small and would remain within acceptable limits. It is concluded that this change does not increase the probability or consequences or introduce the possibility of a different type accident than previously evaluated in the UFSAR.

Serial Number: 98-043-NPE Document Evaluated: MCP 94/1001-00

DESCRIPTION OF CHANGE: This Safety Evaluation assesses the operation of Containment Hatchway crane (Jib crane) during operational modes 1, 2, and 3. This crane is a pedestal mounted hydraulic telescope crane (National Crane Model 647A), and is classified as Non-Q, Seismic Category II/I, safety class others. Evaluation No.: CFR 85/4503R10 has evaluated the use of the Jib crane for handling of the miscellaneous loads at elevation 208'-10" of the Containment (refueling floor) during modes 4 and 5 and periodic cycling of the crane hydraulic system during normal plant operation. This crane has been installed to prevent the Containment Polar crane from becoming a critical path item during refueling outages. However, many routine small loads are required to be handled, e.g. SLC barrels, from Elevation 117' to 208' over the Containment hatchway area, Region 5, shown in Attachment 1, during normal plant operation. For these circumstances, the Containment Polar crane can not be used safely due to its travel path limitation to reach this region. Since the use of Containment Hatchway crane is limited to modes 4 and 5, it can not be used in lieu of the Polar crane for handling of loads during normal plant operation. EER 92/6007 was initiated to emphasize the potential personnel and equipment safety hazards which exist with the current method of handling of loads over this area. Therefore, it is prudent to use the Containment Hatchway crane for handling loads during normal plant operation. This MCP allows the limited 1000 pounds or less lifting load at each crane boom position and requires the additional safety measures for the operation of crane boom to ensure safe handling of loads using the Containment Hatchway crane during normal plant operation.

REASON FOR CHANGE: The use of Containment Hatchway crane is needed to transport small routine loads in the Containment, Region 5 (Attached Figure 1) during normal plant operation. This will eliminate the potential personnel and equipment safety hazards which exist with the current method of load handling due to unavailability of the Containment Polar crane. Therefore, the use of Containment Hatchway crane during normal plant operation ensures compliance with the safety measures imposed in the GGNS plant procedure for control and use of cranes and hoists.

SAFETY EVALUATION: The use of Containment Hatchway crane during modes 4 and 5 and periodic cycling of the crane hydraulic system during normal plant operation has been assessed in Evaluation No.: CFR 85/4503R10. However, the use of this crane is being changed by MCP 94/1001 Rev. 0 to permit its operation during the plant operational modes 1, 2, and 3. This MCP allows the use of crane during these modes based on the limited 1000 pounds or less lifting load at each crane boom position. Additional crane boom operational requirements for the proximity of its positions to the existing safety related equipment and structures, and the maximum load/height restrictions are imposed to ensure the handling of loads complies with the applicable plant rigging requirements, the operating license, and the load limit of Technical Specification 3/4 9.7. Since the maximum 1000 pounds lifting load at each crane boom position is less than 1140 pounds Technical Specifications limit, the load drop criteria in NUREG-0612, UFSAR Appendix 9D and Technical Specification 3/4 9.7 is not required to be postulated. The criteria for movement of loads less than 1140 pounds over the fuel assemblies in the spent fuel pool or the upper Containment fuel storage rack when secondary containment is in effect, is not also applicable since the crane travel path is limited to Regions 5 and 6 (Attached Figure 1). Accordingly, the evaluations of the crane assembly and its supporting structures for normal, seismic and abnormal loadings during modes 1, 2, and 3 have concluded that they remain within design code allowables. Therefore, the change described in this MCP does not increase the probability or the consequences of any accident previously evaluated in the SAR, does not create the possibility of a new accident or malfunction, and does not reduce any margin of safety defined in Technical Specification.

Serial Number: 98-044-NPE Document Evaluated: ER 97/0458-00-R00

DESCRIPTION OF CHANGE: The analyzer for RWCU Inlet (1P33AITN081C) will be replaced with a dual channel (Oxygen/Hydrogen) Orbisphere analyzer Model 3623. The analyzers for RX Recirc (1P33AITN081A) and CRD (1P33AITN081B) will be replaced with a single channel (oxygen only) Orbisphere analyzer Model 3600. The oxygen signals from these analyzers will be connected similar to the existing one. The hydrogen signal output from the RWCU Inlet analyzer is not connected to any remote device.

At the Turbine Building Sample Panel 1H22-P121, the existing single channel oxygen analyzer, L&N Model 7691, will be replaced. The analyzer for RX Feed Pump Discharge (1P33AITN012) will be replaced with a dual channel (oxygen/hydrogen) Orbishphere analyzer Model 3623 and the analyzer for CNDS Demin Comb. Eff (1P33AITN009) will be replaced with a single channel (oxygen only) Orbisphere Model 3600. The hydrogen signal from the the RX Feed Pump Discharge Inlet will be connected to the PDS. The oxygen signals from these analyzers will be connected similar to the existing connections.

The existing recorder in the main control room, G33R611, will be revised to monitor the additional dissolved oxygen signals from the RX Recirc and RWCU Inlet Oxygen analyzers. At present only the CRD Water Dissolved Oxygen level is monitored in this recorder.

REASON FOR CHANGE: Per Vendor (Leeds & Northrup) letter, with the implementation of Hydrogen Water Chemistry during RFO9, the existing analyzers for RX Recirc Oxygen, RWCU Inlet Oxygen and RX Feed Pump discharge (1P33AITN081A,C and 1P33AITN012 respectively) will result in erroneous reading due to the presence of hydrogen in the process water. Also, these instruments are obsolete and no longer supported by the vendor. A dual channel analyzer will be utilized to replace the RWCU Inlet Oxygen and RX Feed Pump Discharge monitors, since Hydrogen water chemistry requires the measurement of oxygen and hydrogen at the RWCU Inlet and at RX Feed Pump Discharge. L&N does not manufacture a dual channel (oxygen/hydrogen) analyzer but one is available from Orbisphere Laboratories. A single channel analyzer will replace the Recirc Oxygen Analyzer. Analyzers 1P33AITN081B (CRD Water Diss. Oxygen) and 1P33AITN009 (CNDS DEMIN COMB EFF Diss. Oxygen), that are scoped under ER 97/0457, will require replacement due to obsolescence. Since these instruments are also located in the same panel as those listed in this ER, it was decided that the replacement will be performed under one ER (ER 97/0458) package.

SAFETY EVALUATION: The reactor sample panel and the turbine building sample panel are non safety related, non seismic category I. All associated instruments are non safety related, their only function is to provide non safety related local/control room indication. None of the instruments is required to initiate or control any engineered or safety systems. The tubing inside the panels is per ANSI, not ASME. The sample lines are connected to ASME piping but the panels are adequately isolated by isolation valves and flow restrictors (reference FSAR Section 9.3.2.3). This isolation capability is not being affected. The changes that are made will be done in accordance with applicable codes and standards including ANSI B31.1 and J-621.0. No new interfaces to equipment important to safety will be created. This design change should improve the reliability of the dissolved oxygen monitors in the panel.

UFSAR Figures 9.3-5 and 9.3-7 will be revised to show the new components added as a result of the monitored sampling point parameters. There is presently no Technical Specification or TRM requirement for the affected oxygen analyzers. The changes of this ER will not require that they be added to the Technical Specifications. The basic function of the analyzers (provide nonsafety related indication) is not changed. The ability to monitor dissolved hydrogen is being added but this capability will not be a Technical Specification or TRM requirement.

The 4-20 ma output of the new analyzers is current limited. The containment electrical penetrations are therefore protected per Reg. Guide 1.63 and separate fuses and/or breakers are not required for the output circuits.

Serial Number: 98-045-NPE Document Evaluated: ER 98/0312-00

DESCRIPTION OF CHANGE: During RFO9, a core shroud inspection tool ring became dislodged from its strongback, at 2 of 4 suspension joints, while being lifted from the core. The two disconnected suspension points were adjacent and 90 degrees apart. This allowed the ring to shift to the point that it came to bear against the top of the drywell flange, the drywell manway covers, and the drywell head studs. The load did not strike any vessel internals as it shifted. The core shroud inspection tool ring weighs approximately 850 pounds. The strongback assembly used to secure the tool weighs approximately 640 pounds. The tool was secured with nine ropes to the polar crane, the refueling platform, and the auxiliary platform while recovery plans were developed.

This Safety Evaluation provides an assessment of the method for lifting the tool and strongback. This lift is a one-time recovery effort needed to remove the core shroud inspection tool ring and strongback.

Four slings will be used to rig the 850 pound core shroud inspection tool ring to the polar crane main hoist at each of four suspension points. Each sling will have a minimum rated capacity of 1500 pounds, and may have intermediate rigging components (e.g., chain fall hoists, etc.) provided their rated capacity equals or exceeds that of the sling and at least two redundant load paths are provided. The 3 ton hoist evaluated in ER 98/0209-00 will be used as the load path for the 640 pound strongback. Consequently, this evolution entails the simultaneous lifting of two light loads. A light load is defined in UFSAR Appendix 9D as less than 1140 pounds. However, as an added precaution, since the loads are still partially attached, the lifts will be made per the requirements for lifting heavy loads as given in NUREG 0612. Ropes currently used to secure the load will be untied from the bridges prior to the lift.

The entire operation will be treated as an infrequently performed evolution under the requirements of Procedure 01-S-06-2, "Conduct of Operations". Secondary Containment and the Standby Gas Treatment System will be required operable during this evolution.

REASON FOR CHANGE: The core shroud inspection tool ring is currently supported by the two remaining lift points and the top edge of the drywell flange, the drywell manway covers, and the drywell head studs. It is positioned above the reactor core. It is therefore necessary to move the core shroud inspection tool ring and strongback away from the reactor core. While nine nylon ropes, each having a working strength of 576 pounds have been rigged to temporarily secure the load, they are not suitable for making the lift.

SAFETY EVALUATION: The core shroud inspection tool ring and strongback will be moved using rigging components with sufficient margins of safety to ensure that a credible failure mechanism capable of causing a load drop does not exist. Four slings will be used to rig the 850 pound core shroud inspection tool ring to the polar crane hoist at each of four suspension points. Each sling will have a minimum rated capacity of 1500 pounds. Slings may be grouped on intermediate lift devices of sufficient capacity, but at least two redundant load paths will be provided. The 3 ton hoist evaluated in ER 98/0209-00 will be used as the load path for the 640 pound strongback. While this evolution entails the simultaneous lifting of two light loads, it will conservatively be conducted per the NUREG 0612 requirements for heavy loads since the total weight is greater than the 1140 pound light load limit. Considering the 1490 pound combined weight of the core shroud inspection tool ring and strongback, the use of four slings with a minimum rating of 1500 pounds effectively provides at least two redundant load paths to the polar crane main hoist. Ropes currently used to secure the load will be untied from the bridges (following rigging with the slings) in order to make the lift. These measures provide a defense-in-depth approach for controlling the handling of loads so that load handling accidents have a very low probability of occurrence. To maintain conservative design features, secondary containment and the SGTS will also be maintained operable during this evolution. Since the safe load handling methods will ensure that the tool and strongback can be removed from the pool without creating a credible drop mechanism, no equipment important to safety will be affected. Therefore, the probability or consequences of any accidents previously evaluated in the UFSAR will not be increased and the probability or consequences of any malfunction of equipment important to safety will not be increased. The movement of the tool and strongback complies with existing commitments for load handling and will not reduce the margin of safety of any basis to the Technical Specifications.

Serial Number: 98-046-NPE Document Evaluated: MS 48.0, Rev. 6

DESCRIPTION OF CHANGE: This safety evaluation assesses the reload-related changes associated with Cycle 10 operation as presented in Revision 6 to the Core Operating Limits Report, Mechanical Standard 48.0. 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 10 reload analysis and the issues considered in this evaluation. Other related safety evaluations associated with the Cycle 10 reload include SE # 98-0010-R00 addressing the Cycle 10 fuel receipt and SE #98-0028-R00 addressing the RFO9 core shuffle.

REASON FOR CHANGE: Cycle 10 operation will require new core operating limits and the Core Operating Limits Report has been revised to include these new limits.

SAFETY EVALUATION: This evaluation concludes that the reload-related changes associated with Cycle 10 operation (i) will require no additional changes to the current GGNS Technical Specifications, and (ii) will not constitute an unreviewed safety question.

Serial Number: 98-047-NPE Document Evaluated: ER 97/0278-00-00,

DESCRIPTION OF CHANGE: This modification requires the removal of the access port inner gasket and it requires welding of the access port inner cover to the guard pipe for the following access ports: Q1B21N420A,Q1B21N420B, Q1B21N420C, Q1B21N420D, Q1B21N421A, Q1B21N421B, Q1B21N421C, Q1B21N421D, Q1B21N422A, Q1B21N422B, Q1B21N423A, Q1B21N423B, Q1B21N426, Q1B21N427, Q1E12N406, Q1E12N407, Q1E51N411, Q1E51N412, Q1E51N413, Q1E51N414, Q1G33N406A, and Q1G33N406B. These welds provide a leak-tight seal around the access ports and form part of the Containment boundary along with the inner cover and the guard pipe. The welds were analyzed in Calculation MC-Q1111-98012 in accordance with ASME Section III, Subsection NC. These new welds are considered to be special seal welds. LDC 98-023 has been initiated to change UFSAR Section 3.6a.2.4.3 and Table 6.2-49, which describe the access ports and how they function.

REASON FOR CHANGE: GGCR1996-0238-00 was written to document that guard pipe ISI access ports 1B21N420A, 1B21N420C, 1B21N420D, 1B21N421C, 1B21N421D, and 1B21N423B failed local leak rate tests. The leakage was above the assigned allowable rate but was not above the primary containment leakage rate per Technical Specification Surveillance Requirement 3.6.1.1.1. A weld will be utilized to prevent this problem from recurring.

SAFETY EVALUATION: ER 97/0278-00-00 allows the removal of the inner gasket on the guard pipe access ports and allows the installation of a weld between the inner cover and the guard pipe. These modifications are applicable to all 22 access ports listed in this Executive Summary. These modifications do not in any way adversely affect the process piping. The postulated failure of the process piping as described in UFSAR Section 3.6A.2.4.1 is unaffected as well. The modifications insure that the Containment boundary function of the access ports is maintained. The Containment boundary now consists of the guard pipe, the inner covers of the access ports, and the new weld. Based on these facts, the modifications will not increase the probability of occurrence or the consequences of an accident previously evaluated in the UFSAR. Additionally, no new mechanisms are introduced that would create the possibility for a malfunction of equipment important to safety.

These modifications enhance the leak-tightness of the access ports and the guard pipe will continue to divert flow to the Drywell as designed, and therefore, will in no way increase the probability of occurrence or increase the consequences of a malfunction of the access port portion of the Containment boundary or ASME pressure boundary. The Containment Boundary now consists of guard pipe, the inner covers of the access ports, and the new weld. The ASME pressure boundary now consists of the inner cover, in conjunction with the guard pipe. The outer gasket, outer cover, pressure bolts, and nuts are no longer considered part of the Containment boundary or the ASME pressure boundary.

The ER does not introduce any new failure mechanism in the associated process piping nor does it create any potential leakage paths. The ER insures that similar leakage around the access ports to that documented for the past LLRTs is eradicated. No new accidents of a different type than any previously analyzed in the UFSAR were identified.

Technical Specification 3.6.1.1 provides the requirements for the operability of Primary Containment boundary. Specifically, Surveillance Requirement 3.6.1.1.1 requires leakage rate testing in accordance with 10CFR50, Appendix J and provides the acceptance criteria for these tests. The installation of the new weld will insure test results on the guard pipe access ports will meet the acceptance criteria. Type B leakage testing of the access ports will no longer be required because of the installation of the welds. No changes are required to the Technical Specifications or their bases nor is the margin of safety of any Technical Specification or its bases reduced.

Serial Number: 98-048-NPE   Document Evaluated: MCP 96/1002

DESCRIPTION OF CHANGE: This MCP will provide pressure relieving/equalization capabilities for valves 1E12-F064A/B and 1E12-F004C. This change is in response to concerns addressed in MNCRs 0270-95 and 0287-95 and CR's 1997-0379-00 and 1997-0381-00.

REASON FOR CHANGE: MNCR 0270-95 identified a concern with the potential for 1E12-F004C to fail to open on demand. NPE evaluated the situation and concluded that a pressure locking phenomena was possible for the valve. On 1E12-F004C, to alleviate the potential for pressure locking, a 1-1/2" line will be routed from the area communicating between the valve seats, to the upstream side (Suppression Pool side) of the valve. The equalization line will contain a ¾" manual isolation valve to be used during ADHRS operation. The equalization line will contain two branch connections with ¾" manual isolation valves to be used during construction and penetration testing. These valves (1E12-F439C, 1E12-F440C, 1E12-F441C and 1E12-F442C) are to be considered containment isolation test connection valves. MNCR 0287-95 identified a concern with the potential for 1E12-F064A/B to fail to open on demand. NPE evaluated the situation and concluded that a pressure locking phenomena was possible for each valve. On 1E12-F064A/B, to alleviate the potential for pressure locking, a 1-1/2" line will be routed from the area communicating between the valve seats and from the downstream side (Suppression Pool side) of each RHR minimum flow line. The test connection lines will contain branch connections with ¾" manual isolation valves to be used during construction and penetration testing. These valves (1E12-F434A/B, 1E12-F435A/B, 1E12-F436A/B and 1E12-F437A/B) are to be considered containment isolation test connection valves. These lines will be connected by an equalization line which will contain a ¾" manual isolation valve to be used during Shutdown Cooling operation.

SAFETY EVALUATION: The purpose of the installed pressure equalization lines is to perform the passive function of providing pressure equalization for the valves (1E12-F064A/B and 1E12-F004C) internals of prevent pressure locking. The primary containment integrity and its aspects with respect to 1E12-F064A/B and 1E12-F004C remains unchanged by this modification. The outboard valve disc continues to perform the sealing function to isolate the RHR system in the case of excessive leakage. Each valve will be tested for leakage in manner consistent with FSAR Section 6.2.6, FSAR Table 6.2-49, and Technical Specification 3.6.1.3 requirements. A leakage rate for the valve will than be determined and compared to an acceptance limit. The combined leakage rate for all penetrations and all valves subject to Type B and C tests shall be less than or equal to 0.60 L_a. Furthermore, this modification does not result in testing which differs from that currently described in the FSAR. Thus, bypassing of a single disc (inboard) will have no effect on containment integrity and should not be construed as such. Additionally, the new containment isolation test connection valves associated with 1E12-F064A/B and 1E12-F004C are designed, installed and tested to ensure the integrity of the primary containment. This design change will have no effect on the operability of the valves or the associated E12 system. The design has been evaluated against the applicable design criteria, installation, and operational requirements, and all necessary requirements and commitments are met. Although these valves do not receive automatic containment isolation signals, the ability of each to isolate as required by Technical Specification 3.6.1.3 is unaffected by this modification. Furthermore, 1E12-F064A/B are required to open for Minimum Flow operation as addressed in 3.3.5.1. This modification is intended to provide added assurance that each valve will open if required. Finally, for 1E12-F004C, its integrity for ADHRS boundary isolation remains unaffected with the addition of the ¾" manual isolation valve. Disc flexing is predicted to only occur during ADHRS operation, however, with the ¾" manual isolation valve closed there should be no inadvertent flow. Likewise, for 1E12-F064A/B, their integrity for Shutdown Cooling boundary isolation remain unaffected with the addition of the ¾" manual isolation valves. Disc flexing is predicted to only occur during Shutdown Cooling operation, however, with the appropriate ¾" manual isolation valve closed there should be no inadvertent flow. Since all functions discussed in the Technical Specifications remain unchanged by this design, all margins of safety remain unaffected. The updating of MS-02 to reflect the addition of the test connection in no way affects the Technical Specifications or the UFSAR. Similarly, the update to M-189.3 in no way affects the Technical Specifications. As stated above the LPCI Mode of operation will not be adversely impacted. RHR system pressures during a Design Basis Accident when the RHR trains would be injecting are well below the pressure required to flex the disc. With no disc flexing there would be no bypass leakage across the min flow valves.

Serial Number: 98-049-NPE Document Evaluated: ER 97/0288-00-00

DESCRIPTION OF CHANGE: The Nuclear Change Response to ER 97/0288-00-00 enhances the operating torque margin (i.e., the difference between the torque required to operate the valve and the torque limit of the actuator) for RWCU motor operated valves (MOVs) Q1G33F001, Q1G33F004, Q1G33F053, Q1G33F250, Q1G33F251, and Q1G33F253 by replacing the motor pinion gear and worm shaft gear sets in each actuator to increase the overall actuator ratio (OAR), and by replacing the actuator motor with an actuator motor capable of increased torque output. All six valves listed above will have their yoke legs stiffened, and an increased Kalsi thrust rating will be applied to each valve actuator. Also, the instantaneous breaker settings for the valve motors are being increased due to the larger horsepower motors.

REASON FOR CHANGE: The objective of this modification is to increase the output torque and thrust capability of the actuators installed on RWCU MOVs Q1G33F001, Q1G33F004, Q1G33F053, Q1G33F250, Q1G33F0251, and Q1G33F0253 by changing the actuator gearing and installing a more powerful actuator motor on each valve. In addition a 162% Kalsi thrust rating will be invoked.

SAFETY EVALUATION: The changes made by ER 97/0288-00-00 will increase the degraded voltage actuator capability (DVAC) torque for the Limitorque motor operators installed on valves Q1G33F001, Q1G33F004, Q1G33F053, Q1G33F250, Q1G33F0251, and Q1G33F0253. This will be accomplished by replacing the existing motor pinion gear and worm shaft gear set in the actuators on the valves to increase the actuator overall ratio (OAR), and the existing actuator motors will be replaced with actuator motors capable of increased torque output. The gear and motor changes made by ER 97/0288-00-00 will increase the operating margin for each of the six RWCU motor operated valves listed above, while addressing industry concerns related to gearbox "run" efficiency.

The UFSAR does not address the operating torque/thrust requirements or capabilities of motor operated valves Q1G33F001, Q1G33F004, Q1G33F053, Q1G33F250, Q1G33F0251, and Q1G33F0253. The motor and gear changes do not alter the function or operation of these MOVs. The slower stem speed resulting from the OAR change on these valves causes the nominal stroke time for valves Q1G33F001, Q1G33F004, Q1G33F053, Q1G33F250, Q1G33F0251, and Q1G33F0253 to increase. Based on the fact that the post-modification calculated stroke times for each of these six valves are bounded by the maximum isolation time contained in TRM Table TR3.6.1.3-1 and the analytical stroke times contained in UFSAR Table 5.2-5 and UFSAR Table 6.2-44, the new stroke times will not affect the safety analysis, function or operation of the valves.

ER 97/0288-00-00 does not change out the electrical penetration assembly, and reactor containment electrical penetrations are not described in the Technical Specification; however, they are described in the Technical Requirements Manual (TRM) Section 6.8.1. TRM Section 6.8.1 states that each of the primary and backup overcurrent protective devices associated with each primary containment electrical penetration circuit, as shown in Table 6.8.1-1, shall be operable. Table 6.8.1-1 lists the breaker settings for 52-1511-07, 52-1511-24, 52-1631-51, 52-1631-53, and 52-1641-07 and is being updated to reflect the breaker setting changes due to the ER. The protective settings for the breakers are set such that the breakers will trip before the penetration can be damaged by fault currents. Fault protection is still provided using the new settings by breakers and fuses which protect them as demonstrated in calculation EC-Q1111-98001. The ER does not change the requirements to test these breakers.

The increased Kalsi thrust rating will be applied by increasing the bolt torque on the connections between the actuator and the valve yoke legs. These changes will allow the valve to withstand the higher actuator torque and thrust capability initiated by the motor and gear changes.

During the design review process, it was determined that the piping stress and nozzle load calculations associated with this portion of RWCU were in a non-conforming condition as documented in GGCR1998-0231-00 and GGCR1998-0231-01. The valve yoke stiffeners have been designed in accordance with the plant design basis criteria as delineated in UFSAR Section 3.9.3. The increases in valve weights and changes in valve CGs due to the installation of the yoke stiffeners and larger motors as specified in this ER will be evaluated in accordance with the plant design basis criteria as delineated in UFSAR Sections 3.7.1.3, 3.7.3, 3.9.1.1, 3.9.2.2, 3.9.3, and Appendix 3.9.3A as part of the resolution of GGCR1998-0231-00 and GGCR2998-0231-01. Based on the preliminary results of the analysis performed to address the modifications made to the valve assemblies by this ER and the non-conforming conditions documented in CR 1998-0231-00 and CR1998-0231-01, it has been concluded that the stresses in the associated RWCU piping are within ASME Code Allowable Limits. The functional capability and structural integrity of the RWCU piping impacted by the MOV modifications specified in this ER have been addressed in Condition Reports CR 1998-0231-00 and CR1998-0231-01. This ER should not be closed out until final resolution of Condition Reports CR1998-0231-00 and CR1998-0231-01 has been reached.

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

Serial Number: 98-050-NSR Document Evaluated: LDCR 98-038

DESCRIPTION OF CHANGE: The change 1) retitles the current Manager, Plant Maintenance position 2) creates a new Manager, Plant Maintenance position 3) changes the reporting alignment for the Manager, Plant Modification and Construction and changes the title to Superintendent, Plant Modification and Construction.

REASON FOR CHANGE: Changes are to reflect the organizational arrangement for the management team of Grand Gulf.

SAFETY EVALUATION: This change does not diminish the qualification level of the staff or affect the capability for the staff to meet regulatory requirements. Therefore, this change does not create an unreviewed safety question. This change is mostly administrative in nature and does not directly effect any plant equipment or procedures. Plant procedures will require revising to reflect the organization after approval and acceptance by the General Manager, Plant Operations.

Serial Number: 98-052-NPE Document Evaluated: ER 97/0031-00-00

DESCRIPTION OF CHANGE: ER 97/0031 provides the design and details to change the control circuit power supply source of the Recirculation Flow Control Valves B33F060A and B33F060B from the existing source to a more stable power provided by each valve's respective Quantum Modicon controller. Additionally, the ER will replace, in both A&B FCV loops, one channel of a dual channel scaler that, although in the circuit, performs no control function with a signal limiter card to limit the position demand error signal that is sent to the position controller.

REASON FOR CHANGE: CR 19960109 identifies the anomaly of Recirculation Flow Control Valve (F33F060B) movement without operator action. The causal factors which resulted in this event are: (1) inherent drift of the FCV control circuit design (2) normal operation during plant outages when the Flow Control Valves (FCVs) are operated at maximum or near-maximum open condition (3) a high limit setpoint on the A/M Balance Card B33K642B-2. These three factors combined create the opportunity for the FCV to slowly drift open until the FCV approaches near the high limit setpoint of the A/M Balance Card. Aggravating the drift problem is the design of the control circuit which results in a small error signal (created by the drift and subsequent pull of the signal by the high limit of the A/M Balance Card) being amplified by a signal to the Position Controller. ER 97/0031-00-00 will eliminate the drift problem by replacing the analog voltage signal from the A/M Balance Card with a steady non-drifting signal from the digital logic of each Recirculation Loop's FCV Modicon controller. Further the high limit of the A/M Balance Card will be removed and effectively relocated downstream of the signal characterizer, such that regardless of error signal, the position demand signal is limited to 100% valve position.

SAFETY EVALUATION: The Recirculation system FCVs operate to change Recirculation flow to change reactor power output by altering core void sweep characteristics which changes neutron flux. The safety functions related to the FCVs are: (1) the FCVs act as part of the reactor coolant pressure boundary (2) the rate of change of the FCV stroke cannot exceed 11% per second in either direction to meet the assumptions of design basis transient and accident analysis (3) the FCVs are designed to lock in place (HPU trip and lockout) during LOCA, as sensed by high drywell pressure. Additionally the FCVs are designed to close in response to a Reactor Feedpump trip with a reactor water level decrease (provided Recirc pumps are in Fast speed). This runback feature, however, is not a safety function.

The proposed changes in ER 97/0031-00-00, providing a more stable position demand signal and adding limiter downstream of the signal characterizer, do not affect any safety functions. Further, the change of signal source from the A/M Balance card to a Modicon analog voltage output module does not create any new concerns regarding types or frequency of failure of the position demand signal. The state of the art Modicon controller is considered at least as reliable as the Foxboro Spec 200 system and loss of the signal from Modicon produces FCV lockup, which is the same results as loss of power to the Foxboro cards. Additionally, the Foxboro cards are powered from the same source as the Modicon controllers, so no new or different failure is created by utilizing Modicon as a signal source. The signal limiter to be installed downstream of the signal characterizer is replacing one channel of a dual channel scaler that performs no circuit function. The limiter is a commonly used card in the Foxboro panels that does not have a failure history and is therefore considered as reliable as the scaler that it replaces. The limiter will provide added assurance that the FCV will not receive a position demand signal of >100% valve position. The conclusion of this safety evaluation is that no change to Technical Specifications is required nor are any unreviewed safety questions created as a result of ER 97/0031-00-00.

Serial Number: 98-053-NPE Document Evaluated: ER 97/0561-00-00

DESCRIPTION OF CHANGE: Changes proposed by ER 97/0561-00-00 involve the installation of a ¾" diameter pressure equalization line on 1N21F009A and 1N21F009B, the outlet isolation valves for High Pressure Feedwater Heaters, 1N21B006A and 1N21B006B. The pressure equalization line will be installed on the 1N21F009A/B valves and will provide an equalization pathway between the valve internals and the downstream piping. The equalization lines will have a normally open, manually operated isolation valve. The equalization line will provide a passive, non-safety related function. The purpose of the equalization line is to eliminate the potential for pressure locking of the 1N21F009A/B valves. Normal Feedwater (N21) System operation will not be affected by the proposed change, however, reliability and availability of the remote operation (opening) feature of 1N21F009A/B will be enhanced by this change.

REASON FOR CHANGE: The proposed change is being conducted to improve Operations Department personnel's ability to remotely open 1N21F009A and 1N21F009B. The design of these valves introduces the potential for pressure locking to occur within the valve internals. If pressure locking occurs, the remote valve operating device(s) may not be capable of OPENING the valve, however, pressure locking isn't a concern with respect to valve CLOSURE. The pressure locking phenomena is an ongoing industry issue and respect to reliability and availability of safety related valves, however, 1N21F009A and 1N21F009B are not safety related. The reason for this change is to improve the Operator's ability to remotely operate (open) the subject valves.

SAFETY EVALUATION: The change authorized by ER 97/0561-00 is confined to the N21 System 1N21F009A/B valves. These valves are located in Area 6 of the Turbine Building, 133'Elevation, and are the outlet isolation valves for High Pressure Feedwater Heaters 1N21B006A and 1N21B006B. The proposed change is limited to the installation of a passive pressure equalization line on the 1N21F009A/B valves and is being installed to improve the reliability of the associated remote valve operating devices to open the affect valves. This change will not alter the design operation of the N21 system nor will this change adversely impact safety related systems, structures, or components. There are no adverse impacts on Seismic Category I equipment or structures, nor are any Seismic II/I concerns introduced by the proposed change. However, piping installed by this ER is seismically evaluated based on Uniform Building Code seismic loading criteria. While the 1N21F009A/B valves are used in the delivery of Feedwater to the Reactor Vessel, these valves serve no safety related functions. The proposed change will not affect the feedwater flow rates, temperatures, or pressures. Implementation of the proposed change will not affect the plant's ability to respond to, eliminate, or mitigate radiological consequences resulting from abnormal or emergency operating conditions. The proposed change will have no impact on the GGNS Technical Specifications or the Technical Requirements Manual. Thus, the proposed change does not represent an Unreviewed Safety Question (USQ) or an Unreviewed Environmental Question (UEQ).

Serial Number: 98-054-HP   Document Evaluated: LDC 98-008

DESCRIPTION OF CHANGE: These proposed changes to Chapter 12 of FSAR, will allow better use of resources, take advantage of advancements in new technologies and methodologies, and remove outdated information that is no longer applicable. Changes are as follows:

1. Updates the words "man-rem" to "person-rem".

2. Changes "(See subsection 12.1.3.1)" IN 12.1.3.1.a, 12.1.3.1.d, AND 12.1.3.j, to "(See subsection 12.5.3.2)".

3. Modifies sentence 11 in 12.5.1.1 to make this statement accurate.

4. Delete Neutron in Chapter 12.5.2.2.4 sentence 12.

5. Reduce quantities of direct reading ion chambers survey meters in Table 12.5-1 from 6 to 2.

6. Reduce quantities of direct reading ion chambers (200mR) in Table 12.5-1 from 700 to 200.

7. Reduce quantities of respirators (air purifying) in Table 12.5-1 from 200 to 50.

8. Delete reference to vendor "teletector" type in Table 12.5-1. Clarify the remarks and correct typographical errors in Table 12.5-1.

    a. Replace ("Teletector" with 13 feet extend. Probe.) with extendible.
    b. Delete (with adjust. Alarm)
    c. Replace (0-25) with 0 - 2.5
    d. Replace (0-3.5) with 0 - 2.5)

REASON FOR CHANGE: 10 CFR 20 and the Regulatory guides 1.7, 8.13, 8.15 and 8.8, provide guidance to management on accomplishing certain aspects of the radiation protection program. These changes will correct editorial discrepancies and update Chapter 12 of the FSAR.

SAFETY EVALUATION: These changes and updates are consistent with 10CFR20, and the Regulatory Guides 1.7, 8.13, 8.15 and 8.8. These references govern the content and intent of Chapter 12. Reducing instrument and respirator quantities in Table 12.5-1, deleting neutron calibration sources and the editorial updates will have no effect on the radiological safety at GGNS. These changes will enhance the operation of Health Physics in its capacity as described in Chapter 12 of the FSAR.

Serial Number: 98-055-NPE Document Evaluated: LDC 1998-020

DESCRIPTION OF CHANGE: Section 5.2.2.10 of the UFSAR (Inspection and Testing), is being changed to:

Clarify initial testing requirements (prior to startup) and testing requirements presently met (post startup).

Clarify the "as found" set pressure and "recertification" set pressure testing requirements for the Main Steam Safety Relief Valves (S/RV's).

Increase the service life inspection frequency from every five years to every six years.

REASON FOR CHANGE: The clarifications made separating the initial testing requirements from the present testing requirements were made as an overall improvement to allow for better understanding of the UFSAR commitments.

The clarifications made on the "as found" testing requirements and the "recertification" testing requirements were made to correct past typographical errors.

The reason for the change from a five year service life to a six year service life is to allow the twenty S/RV's that are presently installed in the plant to remain installed in the plant until they are required to be removed and tested in accordance with the GGNS Pump and Valve Inservice Testing (IST) Program (Program Plan GGNS M-189.1). If the service life remains at five years, certain S/RV's will have to be removed and replaced prior to their selected time which is based on ASME code requirements.

DISCUSSION:

During refueling outage eight, all twenty of the S/RV's were installed with less than 5 years of service life remaining (seventeen installed had 42 months, three had 24 months). This action was considered normal based on past replacement of all twenty S/RV's every refueling outage. Prior to RFO8, the set pressure tolerance acceptance criteria changed from ±1% to ±3% design set pressure which allowed a sample of S/RV's to be tested as representation of the total population of installed S/RV's.

During RFO9, a minimum of six S/RV's will be removed, the present selection of the valves to be removed include two of the valves that have a 24 month service life. However, one of the valves with a 24 month service life will remain installed until RFO10. Leaving this valve installed will result in exceeding the present 5 year service life by 12 months. Extending the service life to 6 years would account for the additional 12 months the valve will remain in service. A one time deviation to the FSAR is not warranted because the same time extension will be needed on six additional S/RV's at the start of cycle 11.

The commitment in the FSAR to perform a five year service life inspection on the Main Steam S/RV's was based on the response to NRC question 211.49 which addressed concerns on safety/relief valve performance on operating BWR's. The concern regarded the performance of S/RV's installed in operating plants during the time of construction and licensing of GGNS, and that new plants should have significantly better performing S/RV's. Many of the older plants were experiencing failures with multiple stage pilot operated S/RV's. The safety function of this type of S/RV requires operation of pilot valve that is susceptible to excessive leakage and corrosive bonding to cylinder walls; thereby preventing proper safety function operation.

At GGNS the S/RV's are Dikkers 8 x 10 direct acting, spring loaded, safety valves with attached Sempress pneumatic actuators and Eugen Seitz control valves for the relief mode of operation, and do not require the operation of a pilot valve for the safety function. Prior to valve being installed in the plant, each S/RV is tested and inspected to ensure the valve will perform its safety function when needed. The actuator and control valve are tested for leakage using a very stringent leakage criteria.

To date, the five year service life internal inspection/rebuild has been implemented twice for 43 S/RV's. This amounts to a total of 86 disassembly/inspections/rebuilds being performed. Out of 86 inspections, none of the inspections have identified seal degradation or internal wear/erosion that would be considered service induced. All 43 valves have been rotated in an out of the plant since power operation; this amounts to 152 "as found" tests and 160 "recertification" tests being performed.

During the establishment of the Electrical Environmental Qualification program (ES-19) to comply with NUREG-0588 requirements, extensive testing was performed on the elastomers installed in the Eugen Seitz control valves. The testing results determined that the limiting component in the Eugen Seitz control valve is the Dow Corning lubricant that is applied to the elastomers which has a lifetime expectancy of 19.9 years.

SAFETY EVALUATION: The safety mode of operation, which consists of direct action of the steam pressure against a single-loaded disk that will open when the valve inlet pressure force exceeds the spring force, is not being affected by this change.

The relief mode of operation, which consists of using an auxiliary actuating device consisting of a pneumatic piston/cylinder and mechanical linkage assembly which opens the valve by overcoming the spring force, is the only mode that could be affected by the recommended change. However, based on the total numbers of past acceptable inspections and tests that have been performed, changing the inspection frequency from 5 years to 6 years should not be considered and extension of time that would have an effect on the valve performance or decrease the margin of safety of the valves.

Although leaving these valves installed in the plant for an additional 12 months (until they are required to be removed and tested in accordance with the ASME code) would exceed the five year service life, the time between the last acceptable test performed and the next time they will be tested will not exceeded five years. This change does not change or degrade the tests and inspections performed on the S/RV's which provides assurance of operation when the valves are needed.

The elastomer materials for the Sempress actuator are manufactured with the same materials (Viton) that were extensively tested to establish service life requirements for the control valve. This would suggest that the service life could be increase from five years to six years without allowing degradation that could cause failure.

Only one S/RV will need service life extension during cycle 10. An additional six S/RV's will need the service life extension during cycle 11. The current ASME code requirement which requires all valves to be removed from the plant and tested every five years will basically mandate that all valves will be disassembled/inspected/rebuilt after five years of service after cycle 11. This is based on the S/RV's needing at least 4.5 years of service life prior to being reinstalled in the plant.

Serial Number: 98-056-NPE   Document Evaluated: ER 97/0201-00-00

DESCRIPTION OF CHANGE: Modify various sections of steel grating installed as Vital Area Security Barriers, at various locations, to support performance of routine preventive maintenance activities on plant ventilation system equipment. The ventilation systems affected are the Emergency Switchgear and Battery Room Ventilation (Z77) System in the Control Building (4 locations) and the Standby Service Water (SSW) Pumphouse Ventilation (Y47) System located at the "A" and "B" SSW Pumphouses (8 locations). The steel grating to be modified is installed on the external surfaces (walls) of the affected buildings and acts to prevent unauthorized personnel entry into the associated areas via the ventilation system ducting. The existing sections of grating are welded in place and must be removed to perform the preventive maintenance. The existing grating will be modified to install a section of grating that can be removed to facilitate required maintenance and replaced upon completion of the maintenance tasks. The modified grating(s) will be fabricated using materials similar to the existing grating, and will continue to be welded in place. Thus, physical security barrier performance will be maintained and the required maintenance activities can be accomplished in a more efficient manner.

REASON FOR CHANGE: The existing gratings are welded in place and must be removed to perform the required preventive maintenance and then reinstalled upon completion of the maintenance item(s). The current design increases the resources needed to complete the periodic maintenance tasks. The existing grating will be modified in a manner that continues to provide the necessary security barrier functions while reducing the resources required to perform the periodic maintenance tasks.

SAFETY EVALUATION: The proposed activity, modifying steel grating installed as physical security barrier(s), will not directly impact operation of any plant system. The sections of grating proposed for modification are installed as security barriers, covering the intake and exhaust ports for the Y47 System and Z77 System. The proposed change uses grating material similar to the existing grating, thus the flow of air through the modified sections of grating will not be altered and performance of the associated ventilation system(s) will not be diminished. UFSAR Figure 3.8-95 will require an update as a result of the proposed activity. As depicted in the referenced UFSAR Figures, the affected grating is installed on the exterior walls of the associated structures, thus in-plant equipment and systems are not affected by implementation of the proposed changed. The affected sections of grating are being modified in a manner that will not represent or introduce any Seismic II/I concerns. The use of similar grating materials and welding the grating in place maintains the integrity and effectiveness of the vital area physical security barriers. Thus, the proposed activity does not represent an Unreviewed Safety Question.

Serial Number: 98-057-NPE   Document Evaluated: LDC 98-010

DESCRIPTION OF CHANGE: The Technical Requirements Manual (TRM) Section 7.6.3.10.3(e) provides the functional testing requirements for snubbers, "During the first refueling shutdown and at least once per 18 months thereafter during shutdown or within 60 days of a planned shutdown, a representative sample of snubbers shall be tested...". The sampling method described in the TRM for snubber testing was based on maintaining a 95% confidence level that 90% of the snubber population will remain operable at any given time. The change being submitted for the TRM allows for the functional testing frequency for selected sizes of mechanical snubbers (PSA 3's, 10's, and 35's) to be extended from every cycle to every other cycle. After the change, the original basis of maintaining a 95% confidence level that 90% of the snubbers are operable will still be met. The remaining sizes (PSA ¼'s, ½'s, 1's and 100's) of mechanical snubbers will be randomly sampled and functional testing performed every cycle.

REASON FOR CHANGE: TRM Section 7.6.3.10.3(e) requires functionally testing a random sample of snubbers every outage. However, the functional test results for the PSA 3's, 10's, and 35's have a near perfect test history. For the eight test periods, 323 PSA 3's, 10's, and 35's have been tested at GGNS with one PSA 10 and one PSA 35 failing functional testing. Therefore, on the basis of these test results and a Weibull analysis of the cumulative test data (Ref. Engineering Report GGNS-98-0001), it is concluded that snubber testing of the PSA 3's, 10's, and 35's every cycle is conservative and a frequency that is Weibull based will provide a 95% confidence that 90% of the population of PSA 3's, 10's, and 35's will remain operable in accordance with the TRM Basis.

SAFETY EVALUATION: The surveillance frequency for functionally testing PSA 3's, 10's, 35's mechanical snubbers will be changed to a Weibull based criteria providing a testing interval of twice the previous interval. The testing frequency for the remaining mechanical snubber sizes, PSA ¼'s, ½'s, 1's and 100's will continue to be every cycle using a 10% sample. During the cycles when the total population of mechanical snubber sizes are to be tested the sample plan may be either the 10% plan or the 37 plan. The testing results of the previous eight outages at GGNS which show nearly perfect test results and the Weibull analysis of the test data shows that the proposed functional testing frequency will be adequate to maintain a 95% confidence level that 90% of the snubbers will remain operable. The functional testing of the snubbers is not limited to the random sample. The snubbers removed from the plant and replaced with regreased/refurbished snubbers are functionally tested. Snubbers (PSA 10's and smaller) removed for other purposes (i.e., ISI inspections, temporary lead shielding, etc) are at the least handstroked. This additional testing has not resulted in any failures of the PSA 3's, 10's and 35's. Engineering Report GGNS-98-0001 documents the results of a Weibull statistical analysis performed for the subject snubber sizes. The Weibull analysis uses the historical Grand Gulf test results and statistically justifies the proposed decreased frequency for functionally testing the PSA 3's, 10's, and 35's only. The Weibull analysis shows that the 95/90 confidence level will be maintained by going to a 36 month testing cycle. This is the original basis for the statistical sampling methodology describe in the TRM. The snubber performance will not be adversely affected by this change since functional tests will continue to be performed for the mechanical snubbers. The larger size mechanical snubbers will continue to perform in their intended manner, and no new failure modes are created by this change, nor are any margins of safety reduced as a result of this change.

Serial Number: 98-058-NPE Document Evaluated: ER 96/0936-04

DESCRIPTION OF CHANGE: This engineering request provides final installation activities to support operation of a Hydrogen Water Chemistry (HWC) System at Grand Gulf Nuclear Station (GGNS). This system consists of bulk hydrogen and oxygen storage sites, a hydrogen injection subsystem, an oxygen injection subsystem, and a control logic subsystem.

The bulk hydrogen and oxygen storage facilities are located on the plant north end of the Unit 2 cooling tower basin area. Report SL-5104, "Site Report, Hydrogen Water Chemistry System", dated January 16, 1997 provides an in depth discussion of the acceptability of this site with respect to NFPA, CGA, OSHA and EPRI requirements and recommendations. A detailed evaluation of the siting and operation of the hydrogen and oxygen storage facilities was provided in ER 96/0936-00 (97-0039-R00) and will not be repeated here.

Hydrogen and oxygen supply piping is routed underground in a single trench and enters the Turbine Building west wall column 3 at elevation 138' as detailed in ER 96/0936-02. The respective gas supply pipes are connected to the hydrogen flow control rack (1H22-P731) installed near Column-Row A-2 at floor elevation 113'-0" and oxygen flow control rack (1H22-P730) installed near Column-Row B-4 at floor elevation 113'-0". Remote operated isolation valves will be installed on both of the flow control racks and at the hydrogen pressure regulating/isolation station (1H22-P732) attached to the Turbine Building exterior wall near column-row 5-A at grade elevation to allow automatic isolation of the hydrogen and oxygen gas supply. Four (4) combustible gas monitors are installed in areas above hydrogen piping containing valves or instrumentation to monitor for hydrogen leakage. Two of these monitors are mounted to the hydrogen flow control rack (1H22-P731) and two (1P73XI-N010A/B) on the Turbine Building ceiling above floor elevation 113'-0", near the injection point (between Column-Row A-3 and A-5).

Design of the hydrogen piping system and racks minimizes potential leakage points via the use of welded joints and fire-rated ball valves. While there is no specific NFPA separation requirements between indoor hydrogen transmission systems (piping runs) and electrical equipment, there is guidance found in NFPA 70 and 497A for electrical separation requirements for hydrogen process areas. The hydrogen isolation skid and hydrogen flow control skids can be categorized as hydrogen process area. For these areas, these two NFPA codes would require electrical equipment within a 15 foot zone of influence around these skids to have electrical equipment classified for (as a minimum) Class 1, Group B, Division 2 hazardous locations. The hydrogen isolation skid area complies with this requirement. The hydrogen flow control area has been evaluated and determined not to require the 15 foot separation zone since electrical equipment on the skid is classified for Class 1, Group B, Division 1 locations and adequate combustible gas detection is provided and interlocked to isolate the hydrogen feed upon detection. Location of the pressure regulation/isolation station (1H22-P732) with respect to the isolated phase bus ducts was evaluated as part of ER 96/0936-01.

Gaseous hydrogen will be injected into a new recirculation loop between the condensate booster pump discharge and suction piping. Gaseous oxygen will be injected into the offgas system between the offgas preheater and recombiner for each recombiner train. Each of the two existing offgas hydrogen analyzers is being replaced by a combination hydrogen/oxygen monitoring panel to ensure that oxygen and hydrogen flows are properly balanced, and that satisfactory recombination has taken place. The HWC system control logic, including all related system interfaces, is detailed and evaluated as part of ER 96/0936-05. In general, the control logic uses the existing Bailey Infi-90 system (1H22-P172) and a new HWC system operator interface panel (1H22-P734).

REASON FOR CHANGE: Operation of the HWC system is intended to reduce rates of intergranular stress corrosion cracking (IGSCC) in recirculation piping and various other stress corrosion cracking (SCC) processes in reactor vessel internal materials. Reduction of SCC is achieved by injecting hydrogen into the reactor coolant thereby suppressing the formation of radiolytic oxygen, which reduces the electrochemical corrosion potential (ECP). An ECP level below -230 mV (SHE) has been demonstrated to arrest SCC growth, however, any reduction in the ECP level below the normal water chemistry (NWC) levels of above 150 mv (SHE) has been shown to reduce SCC growth rates. Addition of hydrogen to the feedwater system reduces oxygen production in the reactor core resulting in an offgas stream that does not contain sufficient oxygen to recombine all of the hydrogen. To balance the gas mixture, the HWC system adds oxygen upstream of the recombiner. After the addition of the oxygen, the offgas system recombiner operates as essentially the same gas loading as initially designed in normal water chemistry. An offgas oxygen monitor is added as a part of the HWC system to confirm that the oxygen addition flow is correct.

SAFETY EVALUATION: New UFSAR sections provide discussion of the HWC system components and operation. Revised UFSAR sections are provided to describe the revised operational configuration of the offgas combustible gas monitors, presence of hydrogen and oxygen piping in the plant, and operational impacts of the HWC system. No changes to Technical Specifications or Technical Requirements Manual (TRM) are required or recommended as part of this ER (Changes may be required to the Main Steam Line Radiation Monitor (MSLRM) setpoints to compensate for the increased "normal" full power background levels, which are referenced in the TRM. However, since the TRM refers to a level above full power background versus a specific discrete value, there is no HWC system based impact.) Review of the normal operational and failure conditions associated with the HWC system equipment did not identify any unreviewed safety questions associated with the proposed design or impacts to other station systems. Note that the detailed evaluations for the bulk gas storage facilities and control system operation/interfaces are detailed in 97-0039-R00 and ER 96/0936-05, and thus are not specifically included as part of this evaluation's scope. The system as proposed meets the intent of the EPRI Guidelines for BWR HWC Systems, which has been generically evaluated by the NRC for all operating BWRs. Therefore, this ER meets the requirements of 10 CFR 50.59 and may proceed in accordance with applicable station procedures. The attached document (P73-00045-01) provides additional details on the station system interface issues addressed in the body of this evaluation, and will be referenced when applicable.

Serial Number: 98-059-PSE Document Evaluated: TA 98-0011

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

REASON FOR CHANGE: BOP Bus 14AE, System R21, provides power to non-safety related components and instrumentation. Required maintenance and cleaning of the 14AE BOP Bus requires that it be deenergized for approximately 24-48 hours. As a matter of convenience, BOP power is being provided to select BOP loads which constitutes a change to the facility since BOP LCC/MCC are identified on plant drawings. This work will be conducted while the reactor is in Mode 4 or 5.

SAFETY EVALUATION: Temporary power from BOP LCC or MCC breakers is required to supply additional power needs during refueling outages or system outages, and to provide power to necessary BOP loads during bus maintenance activities. During the upcoming BOP Bus 14AE outages, additional power requirements will be supplied temporarily from BOP Buses 11HD, 12HE and 13ad. This safety evaluation only addresses BOP power and does not address power supplied from or to Class 1E circuits.

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

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

Serial Number: 98-061-PSE Document Evaluated: TA 98-0007

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

REASON FOR CHANGE: ESF Bus 16AB, System R12, provides power to safety and non-safety related components and instrumentation. Required maintenance and cleaning of the 16AB ESF bus requires that it be deenergized for approximately 24-48 hours. This work will be conducted while the reactor is in Mode 4.

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

Serial Number: 98-062-NPE Document Evaluated: Engineering Calculation XC-Q1E30-95001, Rev. 1

DESCRIPTION OF CHANGE: Revision 1 to Engineering Calculation XC-Q1E30-95001 was prepared to: 1) establish a total leak rate limit for all water tested containment isolation valves taking into consideration changes in the assumptions used to determine the suppression pool inventory and; 2) to address a concern regarding suppression pool water coverage of the RHR A and B test return lines as described in GGCR 1997-0201-00. The maximum allowable leakage limit from the suppression pool is determined to assure that the requirements of UFSAR Chapter 6.2 are consistently maintained. This limit will be used to complement the Technical Specification limit of 1 gpm leakage for each hydrostatically tested valve. The total leak rate limit determined in Calculation XC-Q1E30-95001, Rev. 1 is 56 gpm and is based on no credit for the existing procedurally required operator actions to control the suppression pool level for the initial 12 hours following a DBA.

REASON FOR CHANGE: Changes in the assumptions (e.g., post-accident reflood volumes) used to determine the suppression pool inventory have been identified. Engineering Calculation XC-Q1E30-95-001 was revised to incorporate these changes and to provide guidelines for acceptable water tested containment isolation valve leakage limits. This calculation was originally prepared for input to Engineering Report GGNS-94-0039, "Evaluation of Changes To GGNS Feedwater Isolation Valve Leak Testing Methodology".

SAFETY EVALUATION: Currently, GE document 22A7411 and SDC-E30 establish a design drawdown volume for the suppression pool. These criteria were established for determining the thermal and hydraulic containment design parameters. However, for establishing the most appropriate containment leak rate testing requirements, criteria based on the mechanistic plant response to the postulated DBA LOP/LOCA should be used to more accurately account for the actual distribution of suppression pool inventory and leakage. This distribution is determined in calculation XC-Q1E30-95001 and is based on the conservative application of license basis assumptions and input parameters such as credit for operator actions, leakage characterizations, reflood volumes, and suppression pool inventory makeup. Using these criteria, a hydrostatic testing leakage limit of 56 gpm for the suppression pool is established. This leakage limit maintains an adequate suppression pool volume for heat sink capability and minimum vent submergence without any makeup for 12 hours following a DBA. This limit will be used to complement the current Technical Specification limit of 1 gpm leakage for each hydrostatically tested valve. The 56 gpm limit assures that total leakage from all hydrostatically tested valves will not result in an excessive loss of suppression pool inventory. Revision 1 to Calculation XC-Q1E30-95001 results in a reduction of the primary containment isolation valve combined leak rate limit by only 3 gpm. The time period assumed for operator action was reduced from 24 hours to 12 hours. This reduction is still very conservative based on the requirements specified in ANSI/ANS-58.8 ("Time Response Design Criteria for Safety-Related Operator Actions") and a review of design accident analyses that demonstrated that no critical containment accident parameter is reached beyond 12 hours. In actuality, since the emergency procedures require close monitoring of the suppression pool level, operator action is expected to occur much earlier and could be credited as early as 30 minutes (GGNS license basis commitments to ANSI/ANS-58.8).

To address the concern regarding suppression pool coverage of the RHR A and B test return lines (documented in GGCR 1997-0201-00), Calculation XC-Q1E30-95001 was revised to determine the actual post-accident water coverage for the open ends of the piping. The results of the calculation demonstrated that, although these two lines terminate in the suppression pool slightly (~5") above the design minimum drawdown level (EL. 107'6"), the piping remains covered with water (by ~3") during the entire post-accident period. These results are due to the different criteria that are established for containment design purposes (i.e., to assure minimum drywell vent submergence, ECCS pump suction, and containment heat sink capabilities) versus the more mechanic analysis applied for determining leak rate testing requirements for submerged piping. Both of these results were developed for different design purposes using the license basis requirements applicable to each application.

This changes does not affect the redundancy, diversity, or separation of any safety related structure, system or component (SSC). It also does not impact any non-safety SSC which could inturn affect a safety-related SSC.

The function of Primary Containment Isolation Valves is to limit fission product releases during and following postulated Design Basis Accidents (DBAs) to within limits. The operability requirements for PCIVs help ensure that an adequate primary containment boundary is maintained during and after an accident by minimizing potential leakpaths to the environment. They provide assurance that the primary containment function assumed in the safety analysis is maintained. Surveillance of hydrostatically tested lines provides assurance that the requirements of UFSAR Section 6.2 are met. Using conservative assumptions, design parameters, and considerations of operator actions, leakage, and suppression pool inventory, Engineering Calculation XC-Q130-95001, Rev. 1 demonstrates that the PCIV function and operability are unchanged and that there is assurance that the primary containment function assumed in the safety analysis is maintained. This calculation revision establishes a more restrictive, combined leakage limit for the hydrostatically tested containment isolation valves based on the elevation of the RHR A and B test return lines. The probability or consequences of a loss of any non-safety or safety-related system, structure, or component is unaltered by this change. In addition, there is no change to the current seismic qualification, quality classification, missile or flooding protections, environmental qualification, high energy line break, or masonry walls.

The change to the leakage limit for hydrostatically tested containment isolation valves, from a total of 59 gpm to 56 gpm, is acceptable if the change still provides system functionality, does not adversely affect the public health and safety, and does not significantly increase offsite or onsite exposures following a design basis accident. Engineering Calculation XC-Q1E30-95001, Rev. 1 includes a review of applicable design documents, regulatory requirements, technical specifications, and the plant safety analyses. The changes were found to be acceptable.

Based on the response to this safety evaluation, Engineering Calculation XC-Q1E30-95001, Rev. 1 and the conclusions which it makes do not increase the probability of occurrence or consequences of an accident or malfunction of equipment important to safety previously evaluated in the SAR. The Primary Containment and Primary Containment Isolation system design, design requirements, and system interfaces are unchanged. The change does not create the possibility of an accident or equipment malfunction of a different type than any previously evaluated in the SAR.

Since the primary containment isolation valves remain operable, the hydrostatic testing limit as given in UFSAR Section 6.3.3.2 is actually slightly more restrictive, and the time period before operator action is required is adequate and conservative, the ability to maintain the containment barrier to fission product releases is ensured in accordance with the technical specification bases. Therefore, the change does not reduce the margin of safety as defined in the basis for any Technical Specification.

Engineering Calculation XC-Q1E30-95001 does not supersede the engineering design results or criteria of SDC-E30, GE 22A7411, or calculations XC-Q1B13-92008 and MC-Q1E30-90112. Additionally, Engineering Calculation XC-Q1E30-95001 is prepared for input to Engineering Report GGNS-94-0039.

Serial Number: 98-063-NPE   Document Evaluated: ER 97-0607-03-00

DESCRIPTION OF CHANGE: This SE addresses the startup and operation of GGNS with 114 tubes plugged in "A" Moisture Separator Reheater (MSR) and 56 tubes plugged in "B" MSR (tag nos. 1N35B001A and B). The tube plugging was performed as a REPAIR in accordance with ER 97-0607. As tube plugging is permanent, the heat transfer capabilities of the MSRs are now changed as compared to the original design and this evaluation covers the impacts of this change on the operation of the plant.

The impact on licensing basis documents is the changes to the UFSAR sections, tables and figures listed in this SE. These changes reflect corrected values for temperatures and flows and new vendor heat balances.

REASON FOR CHANGE: The 2nd Stage Reheater tube bundle in MSR 1N35B001A had known failures allowing a substantial amount of main steam to leak to the shell side. This condition was unacceptable from a thermal efficiency and equipment longevity point of view. The MSRs were inspected during RFO9 and tubes were plugged.

SAFETY EVALUATION: With less than 10% reduction in the heat transfer area of the MSRs, the primary impact will be a lower LP turbine inlet temperature. The evaluation shows that the other effects are expected to be minor.

To restore the "A" MSR to operation, the leaking tubes, the tubes that failed the hydrostatic testing, and a 1 or 2 tube deep barrier of tubes were plugged. Also, the 2 columns of ¾ inch tubes with the smallest radius were plugged. An extensive sampling of each bundle was pneumatically tested with station service air pressure and eddy current technology with 100% coverage of the small radius tubes where the damage was located. Hydrostatic testing was used to further define the degree of damage to tubes near to the failed tubes. The results are that MSR "A" has 73 one inch and 41 ¾ inch tubes plugged (including the seven 1st stage tubes plugged because of eddy current indications and proximity to the failure area). In order to minimize the temperature outlet mismatch between "A" and "B" MSRs and to guard against tube failures, the 2 inner columns of tubes in the 2nd stage of "B" MSR were plugged. This resulted in the plugging of 22 one inch and 34 ¾ inch tubes in the "B" MSR. In total, the MSRs have lost 170 out of the 3840 original tubes. Except for the seven ¾ inch tubes plugged in the 1st stage of "A", all of the plugged tubes are in the 2nd stage bundles of the MSRs so an impact on the outlet temperature from the MSRs is expected (See Ref. 1).

According to the MSR manufacturer's data, the temperature rise across the MSR 1st stages was bout 65°F and about 96°F across the 2nd stages. The plugging of tubes during RFO9 leaves the following heat transfer area percentages:

	MSR A	MSR B
1st Stage	99.3	100
2nd Stage	87.9	94.2

Therefore, the temperature rise across each tube bundle will be reduced according to the reduction in heat transfer area. Accordingly, after RFO9 the temperature rise across MSR "A" will be reduced about 13°F and the temperature rise across "B" MSR will be reduced about 6°F. Thus, there will be approximately a 7°F mismatch between the outlet temperatures of "A" and "B" MSRs.

This is well within the allowable maximum long term mismatch value of 10°F (Ref. 2). A small decrease in the MSR outlet temperature with the associated minor increase in moisture will not be a concern for future turbine operation (Ref. 3). Additionally, there are no restrictions on LP3 with respect to operational safety or life expectancy since it will be replaced in RF10.

In this evaluation the plugging of a portion of the MSR tubes is shown not to have an adverse effect on any systems or components that are important to safety. The loss of part of the heat transfer area between main steam and the high pressure turbine exhaust flow to the LP turbines will not prevent the safety related systems that must function to safely shut down the reactor from performing their functions. The proper functioning of the MSRs is more directly related to energy conversion for electrical output than to reactor safety. MSR tube plugging will have an effect on the efficiency of the main turbine because of the amount of LP turbine inlet superheat. Although feedwater temperature could decrease a few degrees (2-3°F), this is well bounded in the accident analyses of UFSAR Chapter 15.

This evaluation concludes that the operation of GGNS with a portion of the MSR tubes plugged is acceptable and does not constitute an unreviewed safety question.

Serial Number: 98-064-NPE Document Evaluated: ER 96/0984-01-00

DESCRIPTION OF CHANGE: Install non-safety related, non seismic, flow transmitter (1N21-FT-N061) to Zinc Injection Skid (1N21D018) in order to provide a remote signal to 1C91P005-1 (computer point 1N211N061). ER 96/0984-01-00 only covers the installation of the Zinc Skid flow transmitter. This design change does not address the operational impact on the heat balance equation. This will be addressed by Plant Engineering when utilizing this input for the core thermal power heat balance equation.

REASON FOR CHANGE: Remote indication of Zinc Injection Skid (1N21D018) flow does not currently exist. Installation of flow transmitter for this skid will provide 4 to 20 mA flow signal for use by Plant Staff.

SAFETY EVALUATION: The addition of a flow transmitter to the zinc injection subsystem will not create any interface with any safety-related system or component. The design change proposed by ER 96/0984-01-00 will not degrade or prevent any actions required to mitigate the consequences resulting from a malfunction of equipment important to safety as previously evaluated in the UFSAR. The flow transmitter along with its associated power supply, conduit and tubing will not alter the design, function or operation of any equipment important to safety. This change will not compromise any safety-related system or prevent safe shutdown since no new interface with equipment important to safety is created nor is such equipment prevented from operating as designed. The addition of the flow transmitter will not change the function of the Feedwater System or the zinc injection subsystem. This design change does not create any Unreviewed Safety Question. UFSAR Section 10.4.7.2 will be revised per LDCR 98-027 in order to accurately describe the Zinc Injection Skid instrumentation after implementation of ER 96/0984-01-00. This modification does not introduce any activity that will adversely impact the safe operation of the plant.

Serial Number: 98-065-PSE Document Evaluated: CEWO 98-0001

DESCRIPTION OF CHANGE: This evaluation addresses use of the Caldon, Inc. ultrasonic External Leading Edge Flow Meter 9LEFM0 which provides improved accuracy in the feedwater flow values input to the core thermal power calculation. Reference 1 covered the initial use of the instrumentation, however operation with the LEFM in service was restricted to above 95% power due to limitations on the uncertainty analysis at that time. These limitations primarily impacted the validity of the MCPR Safety Limit when at lower power levels. Reference 2 has now been completed, providing the necessary analyses to confirm that operation of the LEFM at power levels as low as 55% of rated is acceptable. This safety evaluation is therefore intended to provide the ability to extend the valid range of LEFM operation to as low as 55% of rated power. In particular, the LEFM input will be activated once 58% of rated power is reached on increases, and deactivated at 55% power on decreases. This will be done by making the appropriate changes to the PDS computer points which select the source of the feedwater flow measurement for input to the core thermal power (CTP) calculation.

In addition, two new permissives will be added. One will prevent use of the LEFM values in the heat balance unless the temperature of each feedwater loop is ≥ 220°F. This is needed to comply with assumptions o the Ref. 2 analysis. Another permissive will require that both feedwater pumps be in operation by ensuring that each loop flow is greater than an appropriate minimum amount. Although not strictly require, this permissive avoids the added complexity of additional PDS logic needed to deal with one feed pump operation scenarios.

REASON FOR CHANGE: The current limitations on use of the LEFM have proven to be cumbersome for Operations and Reactor Engineering. During power changes, the LEFM input is automatically "turned on" or "turned off" with regard to its use in calculating the CTP at prescribed power levels of 98% and 95%, respectively. This results in extra monitoring and administrative requirements during routine power changes for turbine testing, sequences exchanges, or control rod adjustments. In some cases, one hour holds in power increase are also required to allow the program which calculates the fuel preconditioning ramp rate to adjust itself for the drop in indicated power level when the LEFM is turned on. Thus, extending the LEFM operating range to a lower power level will eliminate this extra burden and provide a seamless power calculation during normal power change evolutions. The minimum LEFM power level must still be supported by uncertainty analyses since the LEFM is potentially less accurate at lower feedwater flows due to additional uncertainties associated with the effect of feedwater temperature on the acoustics involved. It must also be at a point which is convenient for plant operations. 55% of rated power was chosen to accomplish both of these. It is well below the normal power operating range but is at a point consistent with fast recirculation pump speed operation so as to avoid any overlap with the recirculation pump upshift maneuver during startup. Also, significant plant data are available for operation at that point for the necessary input into the analyses.

SAFETY EVALUATION: No Technical Specification (TS) changes are required since feedwater flow measurement related to the plant heat balance is not directly discussed in the TS, nor need it be. No TRM changes are required. The feedwater flow instrumentation is non-safety related and is not needed for mitigation or prevention of any transient or accident. No modifications to the permanently installed feedwater flow instruments, panel indications, circuitry, or vessel level control system are being made. Changes are in computer software only. A UFSAR change is needed to modify previous changes made by ER 96/0885 which provided for permanent installation of the LEFM. This new change will clarify that the LEFM is not restricted to use only near 100% rated power as was previously stated.

Feedwater flow measurement uncertainty is considered in the determination of the MCPR Safety Limit (SL), however analyses and test results confirm that the overall LEFM uncertainty is much less than that assumed in the SL determination even at 555 of rated power. Software checks on feedwater temperature are being added to ensure that the validity range of the analysis is not exceeded in the event of a loss of feedwater heating. The accuracy of the plant heat balance used to calculate core thermal power also depends significantly on the accuracy of the feedwater flow values. Transient and accident analyses allow for some uncertainty in the initial power level for postulated events, and the use of the LEFM down to 55% of rated power remains bounded by this assumed uncertainty. The proposed change does not introduce new types of events or make the likelihood or consequences of any analyzed event or equipment failure worse. No margin of safety is reduced. Thus, no unreviewed safety question is created as a result of this change.

Serial Number: 98-066-NPE Document Evaluated: ER 98/0358-00-00

DESCRIPTION OF CHANGE: The proposed change will remove an instrument tap root isolation valve (1N11FX301) from the Main Steam (N11) system. A section of stainless steel tubing will be installed to replace the root valve and reestablish continuity of the instrument line. The individual instruments served by 1N11FX301 have individual isolation valves located near the instrument, thus removal of the instrument tap root valve will not prevent isolation of the affected instruments. The overall routing and function of the affected instrument tubing, or the associated instruments, will not be altered by the proposed change.

REASON FOR CHANGE: The instrument root valve (1N11FX301) being removed by the proposed change is connected to Main Steam Line "A" and is used as an isolation valve for an instrument sensing line. The sensing line is used for several instruments including 1N11N052B and N052D, which provide Main Steam Line pressure signals for the Reactor Protection System. Root valve 1N11FX301 has demonstrated a low level of reliability due to thermal growth and movement of the associated steam piping. To eliminate the potential source of steam leak and minimize the associated operational challenges, the proposed change removes this root valve from the N11 System. There are no tests or experiments associated with the proposed change.

SAFETY EVALUATION: The removal of root valve 1N11FX301 from the associated instrument sensing line will eliminate the possibility of using this valve to isolate the associated instrument line in the event of a tubing leak or rupture. Thus, should a steam leak occur due to a leak or rupture of the associated instrument tubing, other methods will be required to isolate the steam leak. In extreme conditions, the Main Steam Line Isolation Valves could be closed to depressurize and isolate the leaking area. Section 15.6.4.2.2 of the UFSAR discusses the potential for, and consequences of, a steam leak in the Turbine Building (outside containment). The event analyzed and presented in this section of the UFSAR adequately bounds the potential scenarios that may result from the proposed change. There are no new systems or components added by the proposed change, thus the existing accident scenarios and analyses presented in the UFSAR will not be adversely impacted by the proposed change. The proposed change will affect UFSAR Figure No. 10-3-001-1 since 1N11FX301 is currently depicted on thus UFSAR Figure. However, removal of instrument root valve 1N11FX301 will not result in the operation of any plant system or component in a manner that is inconsistent with information contained in the UFSAR. The proposed change is entirely contained within the confines of the power block and will not affect or impact the plant's radiological or non-radiological effluents. Thus, the proposed change will have no adverse environmental impacts. After a review of the proposed changed, it has been concluded that removal of root valve 1N11FX301 does not represent an Unreviewed Safety Question and will have no adverse affects on the environment.

Serial Number: 98-067-NPE Document Evaluated: ER 97/0443-00-02

DESCRIPTION OF CHANGE: This modification proves for the removal of a Unit II Cardox Fire Suppression Panel, associated ETL panel and other peripheral equipment. The removal of this panel is required for its use as a test subject for Seismic qualification. The panel selected for removal is N2P64D209, originally installed for protection of Unit II Division III switchgear room (OC213). This panel was installed during Unit II construction, per A-0630, this panel has not been activated. This ER & Safety Evaluation has been revised due to a scope change required for disposition of GGCR1997-1055-00. This scope change is required to remove the non-active Unit II CO_2 valve panels from the daisychain logic of the main CO_2 tank valve. Presently, as stated on M-0035F, the Unit II panels are not required for Unit I function, but their physical presence is required since they complete the CO_2 valve logic circuitry associated with the Unit I panels. This Safety Evaluation is only required to support the FSAR figure change to remove panel N2P64D209 and valve open inputs from other Unit II Cardox panels from figure 9.5-6 (P&ID M-0035F).

REASON FOR CHANGE: GGCR1997-0284-00 documents a potential for inadvertent discharge of carbon dioxide (the fire suppression agent) and damper closure during a Seismic event as a result of chatter by the Cardox panels' internal initiation relays. Currently, all areas provided protection by this panel type are at risk of this inadvertent discharge and HVAC damper closure until is has been shown that this chatter, if any, of the initiation relays is acceptable. The removal of the Unit II CO_2 panels from the CO_2 valve logic will eliminate a possible failure mechanism that would cause an inadvertent pressurization of the CO_2 header.

SAFETY EVALUATION: The CO_2 panel to be removed has never been declared operational/activated and no credit is taken for its installation. Due to its status, it is not credited for fire suppression in any room. Currently, the permissive to open the main CO_2 tank valve from the Unit II CO_2 panels is defeated since they have not been activated and are not credited for any fire suppression function. Their physical removal from the valve's control circuit will not change the required valve logic or function of the required/operable CO_2 panels. No new or additional equipment is being added via this modification, nor is the function of any equipment being changed. Currently, the GGNS Fire Hazards Analysis Report (M-500) does not take credit for the presence of these panels or the suppression they were to provide. Section 6.2.4 of the GGNS TRM lists the CO_2 panels required to be operational when the equipment protected is required to be operational. These panels are not listed. Neither the GGNS TRM, Technical Specifications, nor its bases reference, credit or determine any margin of safety on the presence of these panels. Therefore, no change to the GGNS Technical Specifications or bases is required. The removal of these panels will not increase the probability of occurrence or the consequences of an accident previously evaluated in the SAR. Nor will it increase the probability of occurrence or the consequences of a malfunction of equipment important to safety previously evaluated in the SAR. It will not create the possibility for an accident of a different type than any previously evaluated in the SAR or a malfunction of equipment important to safety of a different type than any previously evaluated in the SAR. The revision to the figure provides for correct reflection of plant configuration as described in the SAR.

Serial Number: 98-068-NPE   Document Evaluated: ER 97/0377-00-01

DESCRIPTION OF CHANGE: ER 97/0377-00 changes the range of the Standby Service Water (SSW) pump discharge pressure transmitter and recorder and changes the low discharge pressure alarm setpoint.

REASON FOR CHANGE: CR 1997-0297-00 was initiated to document that the SSW pump discharge pressure low alarm setpoint was not evaluated when the SSW pump was changed by DCP 82/5020 (Ref. 2.2). Although not formally documented in the CR, Plant Staff has informed NPE that the discharge pressure transmitter is over ranged during pump starts.

SAFETY EVALUATION: The SSW system including the UHS is a safety related system that provides cooling water for removing the heat from the containment after an accident. The pump discharge pressure instrumentation provides indication, alarm and recording of the pump discharge pressure. The proposed changes to the pump discharge pressure indication maintain the design function of the instrumentation to indicate the pump discharge pressure over the normal range of system pressures and to provide an alarm indicating gross system leakage or a gross pump failure. No changes to the Technical Specifications are required, and the proposed changes do not create an Unreviewed Safety Question.

Serial Number: 98-069-NPE   Document Evaluated: Temp. Alt. 97-0013

DESCRIPTION OF CHANGE: Installation of a temporary discharge line from the Main Condenser Water Box Drain and Recirculation Pump 1N71-C002B to facilitate pumping down the water boxes. The change will allow pumping of circulating water to either of the circulating water pump pits and/or to storm drains to permit complete condenser water box drainage and condenser tube repairs during reduced power operations (i.e. capable of operating at reduced reactor power with one circulating water pump and one train of condenser water boxes.

The temporary line will be routed from minimum flow line 8"-JBD-235, at El. 87' of the Turbine Building to El. 133', through the yard to the Circulation Water Pump House and into one of the pump pits. An alternate flow path will be proved to discharge to nearby storm drains. The Water Box Drain and Recirculation Pump logic will be jumpered to allow operation of either pump with discharge through the temporary discharge piping.

REASON FOR CHANGE: The current configuration of the piping discharge from the condenser water box drainage and recirculation pumps, 1N71C002A and 1N71C002B does not permit complete condenser water box drainage during power operations without opening the cooling tower bypass valves due to the low discharge head of the pump(s) and the high static head required to discharge through the natural draft cooling tower nozzles. Opening of the bypass valve on one circulating water train causes the other train's hot side inventory to reverse flow through the open bypass valve and eliminates any significant cooling of the circulating water. The proposed temporary alteration will significantly reduce the static head required to pump the water box inventory by discharging to the circulating water pump pits, while allowing the remaining circulating water train to provide cooling.

SAFETY EVALUATION: The circulating water system, including the condenser water box drainage and recirculation subsystem, serves no safety functions as discussed in UFSAR Section 10.4.5. Loss of condenser vacuum and flooding events, analyzed in the UFSAR, envelope the occurrence of postulated events due to implementation of this temporary alteration. The location and routing for the temporary piping in the turbine building present no seismic II/I hazards. The temporary alteration does not alter or affect the condenser vacuum low setpoints.

Implementation of the temporary alteration will not create any new interface, or new failure mode which would affect any equipment, components or systems which are safety related or important to safety. Implementation of the temporary alteration will not create any new or affect existing functions which mitigate the consequences of a malfunction of equipment important to safety. Implementation of the temporary alteration will not change any function, parameter or operating characteristic which would create an interface with or affect any safety related components, equipment or systems. The location of the temporary piping outside of the turbine building does not pose any new failure modes for the generator isophase bus ducts in the general area.

Serial Number: 98-070-NPE Document Evaluated: TSR 98-008

DESCRIPTION OF CHANGE: TSR 98-008 is generated to perform work on RWCU Heat Exchanger Shell C replacement and Movats on valve G33F251 during RFO9. Temporary lead shielding is required to be installed per previously approved TSR 92-20. TSR 98-008 is requested to installed one additional piece of lead shielding on 6"-DBZ-22 vertical run (near valve G33F251). The lead shielding will be installed during Operating Modes 4 and 5 only and must be removed prior to restart.

REASON FOR CHANGE: The temporary lead shielding will be attached to certain portions of the RWCU system in order to reduce radiation exposure to personnel performing work in this area.

SAFETY EVALUATION: The code qualification of the affected piping with temporary lead shielding has been documented in NPE Mechanical Calculation MC-Q1G33-92026, Rev. 0A. The subject piping was conservatively analyzed for SRV load.

Review of the Mechanical calculation MC-Q1G33-92026 indicated that the increase in pipe weight due to the addition of the temporary lead shielding will have no impact on the safety or the ability of the piping system to perform its safety function.

This conclusion is based on the fact that the effect of the additional temporary lead shielding weight is negligible relative to the contribution of the hydrodynamic loads to the piping total combined stress of piping system. Hydrodynamic loads (i.e. SRV) are not expected events during Modes 4 and 5.

All applicable ASME Code stress allowable met. Therefore, the system operability in Operating Modes 4 and 5 is not affected by the addition of the temporary lead shielding. All the lead shielding must be installed during Operating Modes 4 and 5 only, and must be removed prior to the plant restart. The lead blankets may be wrapped around the piping at the locations and in the amounts shown in TSR 98-008. Also, no other lead shielding or any other additional weight can be attached to the piping while this shielding is attached.

Temporary addition of lead shielding does not result in any permanent changes to location, routing, or types of supports, nor does it alter any component performance characteristics, design parameters, or operational parameters of the affected system after the temporary lead shielding is removed.

Serial Number: 98-071-NPE   Document Evaluated: ER 98/0229-00-00

DESCRIPTION OF CHANGE: ER 98/0229-00-00 will remove the screen element from strainer N1E31D002 upstream of flow sensor N1E31N021 in line JBD-312. This ER will also install two ¾" valves, one upstream and one downstream of the loop seal in which the flow sensor is located to permit flushing and filling the loop. Line JBD-312 is the condensate drain line from several Drywell coolers to the Drywell floor drain. The referenced flowsensor (N1E31N021) is required per GGNS Technical Specifications Section 3.4.7 to monitor unidentified leakage.

REASON FOR CHANGE: The strainer was installed to filter flow for the original flow element installed as 1E31N021 which was a turbine type flowmeter that had extremely tight internal tolerances which allowed the flowmeter to readily clog up. DCP 91/0113 replaced this flowmeter with a meter not prone to clogging and installed a loop seal to keep its electrodes submersed as required, but the DCP did not remove the unnecessary upstream strainer, N1E31D002. During the last operating cycle, this strainer clogged and prevented effective use of the flowmeter and caused back flow out of the associated Drywell cooler drip pans. The installation of the two ¾" valves will permit flushing the line during shutdown to remove deposits and permit accurate filling of the line to meet instrument requirements of keeping its electrodes submersed.

SAFETY EVALUATION: This modification will not delete, add or change any permissive, interlock or interface that currently exists, but rather will enhance the performance of current equipment to be more reliable and accurate to increase system availability by reducing the possibility for clogging the drain line. The existing strainer is not required for service with the newer flowmeter. The strainer housing will remain as part of line JBD-312 to facilitate work reduction and maintain all current piping stress analysis. The installation of the two small valves are at existing plugged NPT "Tee" fittings utilized as 90° elbows in the existing non-nuclear B31.1 piping. The existing function and pipe class is not being changed by this modification. Failure of the new valves is not likely since this line is not under pressure. Should the valve leak, it would not adversely affect safety or the Leak Detection system. All associated piping supports calculations have been reviewed and do not require revision due to this modification. The design change will not create the possibility of an new accident not previously evaluated in the FSAR and will not increase the consequences or possibility of occurrence of an accident currently evaluated in the FSAR. No equipment important to safety will be compromised by the removal of the strainer screen element or the installation of the two new valves. No changes are required to the GGNS Technical Specifications and no margin of safety will be reduced or encroached upon. This modification will make detection of leakage within the drywell more accurate and reliable by removing a known location for clogging and providing for proper filling of the loop seal.

Serial Number: 98-072-NPE Document Evaluated: ER 96/0778-00-00

DESCRIPTION OF CHANGE: This ER evaluates and provides details for tapping existing Fuel Pool Drain Tank Level signals to create inputs to PDS for this condition. Signals at Control Room Panels H13P632 and H13P642 will be utilized to provide these inputs to the C88 Transient Test System. Using these inputs to C88 will allow creation of associated PDS points which can be utilized for continuous monitoring of tank level.

REASON FOR CHANGE: During normal plant operations, the Fuel Pool Cooling and Cleanup System is in service with flow to the Upper Containment Pools and the Spent Fuel Pool. While starting and stopping the system or when the flow path is changed, an operator is stationed in the Control Room back panel are (H13P632/H13P642) to monitor the Drain Tank's Level Indication. This is to ensure that the pumps have sufficient suction and to prevent overflowing of the pools. During refueling outages, when the Drywell and the Reactor Vessel Heads are removed and the vessel is flooded, the Fuel Pool Drain Tank Level Indication becomes the only remote means of Vessel Level monitoring. Since the Operator that is assigned to monitor vessel level is usually the Operator that is responsible for monitoring Nuclear Instrumentation while core alterations progress, the individual must walk back and forth between the two locations. Creating these PDS computer points will provide another means to continuously monitor Drain/Tank Reactor Vessel Levels, in addition to the existing system indication.

SAFETY EVALUATION: Utilization of existing signals from FPCCU instrumentation to provide PDS computer points for continuous monitoring of Drain Tank or Vessel Level does not adversely impact any existing system configuration. The safety related, divisionalized instrumentation circuits are not degraded by this modification. The new ties to the C88 Transient Test System will likewise be divisionalized (associated) to provide the necessary separation and protection. Existing FPCUU tank level instrumentation shall remain continuously indicated in the Control Room. This modification only provides an additional source of information for the Operator. UFSAR Figure 9.1-026-2 will be revised to reflect addition of the computer points to the FPCCU (G41) System. No other UFSAR revisions are required.

Serial Number: 98-075-NPE   Document Evaluated: ER 97/0324-00

DESCRIPTION OF CHANGE: The generator gas system upgrade will replace the existing gas system components with a single skid which will include a dual tower gas dryer, a gas instrumentation rack with control panel and primary water tank gas supply components. The new skid will be relocated to provide additional space for future generator seal oil system upgrade. The existing gas dryer, the hydrogen supply regulator and back pressure regulator for the primary water tank, the hydrogen supply regulator to the leakage water collection tank (primary water supply unit), and all components of the existing gas rack will be replaced by the new skid.

The change will require rerouting of N44 system carbon dioxide piping. The change will not affect fire protection system carbon dioxide piping or operation. The change will require rerouting of connecting piping for hydrogen to the new skid. The nitrogen supply bottles will be relocated outside the building to the yard. The existing methane supply piping will be rerouted and supply nitrogen. Also, the existing carbon dioxide flash evaporator and high pressure storage bottles will be removed.

The change will require rerouting of connecting tubing from instrument air system (P53). The change will require rerouting of connecting piping for existing carbon dioxide vent and purity meter vent. The change will require rerouting of connecting piping of existing primary water vent to misc. vents and drains for A-CPSI turbine generator equipment (N31 system). The change will require routing of a new drain to the existing chemical radwaste (CHRW) system. The change will delete existing vent from dryer to the generator bearing exhaust fan of lube oil system (N34). The change will require rerouting of the existing mechanical vacuum pump cooler outlet piping of the plant service water (PSW) system (P44) to remove interference with new skid. The removable barrier between elevations 129' and 133' near the new gas rack location will require removal and modification. The vertical portion of the existing removable barrier will be reused. Existing piping support N1N19G002H09 of condensate system (N19) will be modified to remove interference with new skid.

Appropriate signs will be made and located around the generator gas rack indicating the requirement that there is a 15 foot zone of influence around the hydrogen gas dryer and associated components on skids in which no ignition sources are permitted (i.e. open flames, use of spark producing tools, use of electrical equipment which is not acceptable for Class I, Group B, Division II locations as identified by Article 500 in the National Electric Code 70). A baffle will be constructed at the ceiling above the gas rack to minimize the possibility of hydrogen leakage infiltrating the generator exciter housing.

New conduit will be installed from the replacement gas dryer skid to the existing tray sections above the GAC panel. New signal cable will be installed from the skid to the GAC panel for annunciator/indicator circuits. New signal cable will also be installed from the replacement skid to a local Plant Data System mux cabinet to support computer point additions. New power cable will be installed from a BOP MCC to the new skid. Lighting in the area of the new skid will be replaced with fixtures suitable for Class 1 and Division II Group B hazardous area service.

The current configuration of the TBCW return line of the dryer cooler ties into the return line of the Battery Room A/C unit (U41b032-N). Plant Staff has reported that A/C unit is not functioning correctly due to inadequate TBCW flow. TBCW return line needs to be relocated (tied into system at different location) to allow flow during outages and low power operations. Plant staff has also identified this item as a battery life concern. The Change Notice (CN) will involve relocation of the TBCW return line tie-in to downstream of the P43F503 valve. This proposed change will provide adequate flow to the gas dryer cooler and A/C unit, thus restoring reliable operation. The change will have negligible affect on P43 system heat load.

REASON FOR CHANGE: The generator gas system upgrade project will replace the existing gas dryer to further decrease the hydrogen moisture content (i.e., dew point). However, detailed design reviews have revealed that almost all replacement parts on the gas racks are obsolete. The hydrogen supply pressure regulator and back pressure regulator for the primary water tank also are obsolete. The replacement parts will require modification to the existing piping. The new skid will be relocated to provide necessary space for future generator seal oil system upgrade. The existing methane supply is no longer required. Therefore, methane supply piping will be used for nitrogen supply. The P43 system will supply cooling water to the new dryer. In order to remove moisture, drain will be routed to CHRW. New dryer does not require vent. Therefore, vent to the N34 system has been deleted. Also, P44 system piping, N19 system pipe support and removable barrier interferes with the new skid location. Therefore, these components will be modified to remove interference with new skid. The P43 system will supply cooling water to the new dryer cooler. The current configuration of the TBCW return line of the dryer cooler tie into the return line of the Battery Room A/C unit. Plant Staff has reported that A/C unit is not functioning correctly due to inadequate TBCW flow. There is a flow restriction due to hydrogen temperature control valve (P43F503). Also, this restriction will affect inadequate flow for the dryer cooler.

Based on engineering evaluation (CI065676) performed for A/C unit, TBCW return line needs to be relocated (tied into system at different location) to allow flow during outages and low power operations.

SAFETY EVALUATION: The generator gas rack (N44D001) is part of the hydrogen and carbon dioxide (N44) system. The design change will affect the N43, N31, N34, N19, P43, P44, C91, and P53 systems. The proposed change will enhance the main generator gas system without affecting the operation of interfacing systems. The proposed change will update information provided in UFSAR Section 10.2.5 and Figures in UFSAR

Sections 9.2.9, 9.3.1 and 9.5.1. The change will not affect the design information provided in the UFSAR Sections for N19, N31, N34, N43 and P44 systems. UFSAR Section 3.2 classifies systems N19, N31, N34, N43, N44, P43, P44, and P53 as "Other" which means that their failure will not compromise any safety related functions or prevent safe shutdown. UFSAR Table 3.2-1 classifies these systems and their associated components as non-safety related, non-seismic, quality Group D and ANSI B31.1. The modifications made by this design change is in compliance with the criteria listed in UFSAR Table 3.2-1.

The replacement of the gas dryer and associated components are not safety related and will not affect any safety related equipment or systems. The proposed change does not affect any parameters specified in the GGNS Technical Specifications (TS) or GGNS Technical Requirements Manual (TRM). The changes will not affect any equipment important to safety.

The main generator gas system is a potential fire and explosion hazard, but changes to N44 system components are not anticipated to increase the probability of operational occurrences or accidents. A baffle will be constructed at the ceiling above the gas rack to minimize the possibility of hydrogen leakage infiltrating the generator exciter housing. All electrical components on the new skid meet the NEC criteria above. Piping and tubing for the proposed change will be designed and installed to meet or exceed the original design requirements for minimizing potential hydrogen leaks. Hydrogen gas system components will be relocated to an open area near the existing location so that potential small hydrogen leaks will not accumulate to combustible concentrations. The design has been evaluated against the applicable design criteria, installation and operational requirements. All necessary requirements and commitments are met. A loss of hydrogen gas supply would continue to result in insufficient cooling of the main generator and ultimately require shutdown of the main generator.

The N19, N31, N34, N43, N44, P43, C91 and P53 systems are not part of the reactor coolant pressure boundary nor are they required for safe shutdown of the plant. The design change will not alter the design, function or operation of any equipment important to safety as evaluated in the UFSAR. The gas system components change will not affect the reliability of equipment important to safety since it has been designed in accordance with all necessary design criteria, commitments and requirements. The proposed change will enhance operation of the main generator gas system without affecting any safety related systems. The capability to safely shut down the reactor will not be impacted by these changes. Similarly, proposed change to relocate TBCW tie-in for return line of dryer cooler and A/C unit will not alter the design, function or operation of any equipment important to safety as evaluated in the UFSAR. Therefore, implementation of CN, ERs 97/0324-00 and 97/0324-01 will not adversely impact plant operation or any system important to safety.

Serial Number: 98-077-NPE   Document Evaluated: ER 97/0693-00-02

DESCRIPTION OF CHANGE: In July 1997, GGNS issued a Condition Report (CR) 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 lead to localized heating of the containment wall surrounding the penetrations. As part of the evaluation of this CR, NPR evaluated the minimum flow rates required for current RWCU operating modes and supplied them to Plant Staff in Engineering Request 97-0693-00, Rev. 1. This evaluation provided minimum PSW flow s and piping line-ups to ensure containment wall temperatures were maintained below existing FSAR limits. As part of the rework procedures, Plant Staff was to measure PSW flow to the coolers to verify compliance with design. These flows could not be accurately measured during RFO9. In light of this, cooling water supply to Penetration 83 coolers will be changed to Plant Chilled Water. Penetration 88 cooling water supply will not be changed but has been evaluated for zero flow condition at maximum temperatures during RWCU Pre-pump operation. It has been earlier determined that cooling is only required to Penetration 88 during RWCU Pre-pump operation. Penetration 87 jacket cooler was earlier determined not to require cooling water flow and it will not be considered as part of this evaluation.

Chilled water will be supplied by connecting a flexible metal braided hose to existing PCW piping on the discharge piping of the Auxiliary Building Steam Tunnel Cooler, N1T41B011. The hose will be routed to the jacket cooler, N1G33B003C, and returned to the N1T41B011 cooler discharge line by an identical hose. A control valve will be installed between the supply and discharge connection to divert PCW to the jacket cooler. Adequate isolation valves will be installed to facilitate system operation. All components, except the braided hose, installed in the supply and return line will conform to JBD piping class (ANSI B31.1, Primary pressure 125psi @ 350°F) as specified in MS-03. The flexible hose does not conform to the JBD pipe class requirements, but has a working pressure and temperature well in excess of those of the piping and is considered acceptable. All new piping, components, and hoses are considered non-safety related and non-seismic. The routing of the piping and hoses is such that they will have no adverse interaction with safety related system, structures, or components.

The CTMT wall concrete was evaluated for potential degradation due to operation of Penetration 88 jacket cooler, N1G33B003B, with no flow during RWCU Pre-pump mode. NPE determined in Engineering Report 98-0035, Rev. 0 that any further degradation to the concrete compressive strength would be small and would remain within acceptable limits.

REASON FOR CHANGE: A condition report documents PSW system flow rates below design limits because of pipe fouling. Attempts to clean the piping during RFO9 have not been successful. To provide cooling to Penetration 83, the cooling water supply to the penetration cooler will be changed to Plant Chilled Water. Since cooling to Penetration 88 is only needed in RWCU Pre-pump mode, no change will be made to enhance flow to the cooler until RFO10 and NPE has evaluated effects to the concrete for the short period of time Containment concrete temperatures will exceed the FSAR limit of 200°F.

SAFETY EVALUATION: Safety Evaluation 98-0034-00 provided a history of the operation of the jacket coolers for Containment Penetrations 83, 87, and 88. The previous calculations, required flow rates, and degradation of Containment wall concrete compressive strength were evaluated based on a designed that supplied PSW as a source of cooling water to jacket coolers. PSW piping is plugged and adequate flows can not be established. To ensure adequate flow, Plant chilled Water will be used as a supply of cooling water to Penetration 83. PCW will be able to provide required heat transfer requirements and flow to ensure concrete temperatures are below the FSAR limit of 200°F. To accomplish this, the existing PSW piping to Penetration 83 will be removed and isolated on either side of the jacket cooler, N1G33B003C, at existing isolation valve. Plant Chilled Water will be supplied by attaching a flexible metal braided hose to existing PCW piping downstream of the Auxiliary Building Steam Tunnel Cooler, N1T41B011. The hose will be routed to the cooler and returned to the N1T41B011 cooler discharge line by an identical hose. A control valve will be installed between the supply and discharge connection to divert PCW to the jacket cooler. The new piping and hose will be routed to prevent any adverse interaction with any safety related systems, structures, or components. Included in the design of the PCW to the jacket cooler will be a temperature element, N1P71N014, to allow monitoring of the PCW supply temperatures. This element will be used to verify that PCW supplied to the jacket cooler is maintained such that the Lowest Service Metal Temperature (LSMT) of the penetration is not exceeded. Operator action will ensure operation of RWCU above the LSMT of the penetration.

No change to cooling water flow to Penetration 88 cooler will be considered and the flow rate is conservatively assumed to be zero. Due to the short period of time the surrounding concrete will be above the FSAR limit, NPE determined in Engineering Report 98-0035, Rev. 0 that any further degradation to the concrete compressive strength would be small and would remain within acceptable limits. It is concluded that this change does not increase the probability or consequences or introduce the possibility of a different type accident than previously evaluated in the UFSAR.

CCE-97-0001

Commitment Number: AECM-83/0353 Source Documen. Number: NOV416/83-18-1

Commitment Change Title: Visitor/Escort Violation ?)-416/83-18-1

COMMITMENT DESCRIPTION: "A visitor receipt system has been implemented which requires escort to acknowledge their responsibilities and escortees to maintain the accountability receipt in their possession while in the protected area. Also, close proximity controls of being within 25 feet of a visitor and in constant visual contact have been established". The receipting system was implemented by the fabrication and production of Visitor/Escort Control Form to be issued each time a visitor is processed.

JUSTIFICATION FOR CHANGE OR DELETION: 10 CFR 73.55(d)(6) states "Individuals not authorized by the licensee to enter protected areas without escort shall be escorted by a watchman or other individual designated by the licensee while in a protected area and shall be badged to indicate that an escort is required. In addition, the licensee shall require that each individual register his or her name, date, time, purpose of visit, employment affiliation, citizenship, and name of the individual to be visited. The licensee shall retain the register of information for three years after the last entry in the register". The Visitor/Escort Control Form does not contain this information nor is it retained as an official record.

The official register is completed during visitor processing and the Visitor/Escort Control Form, which serves no useful purpose, is completed in addition to the register. Completion of this form is a time consuming process that could be eliminated by deletion of the form. Additionally, the Visitor/Escort Control Form is a multi-page form which has to be purchased from outside sources as it cannot be locally produced. Finally, the commitment is more than 10 years old and no additional violations have occurred.

This change deletes the requirement to complete a Visitor/Escort Control Form only, and <u>does not</u> alter the requirement to maintain close proximity controls of being within 25 feet of, and in visual contact with, a visitor. The commitment to maintain a Visitor/Escort Control From, being within 25 feet of, and in visual contact with, a visitor was never incorporated into the licensee Security Plan; however, the commitment to maintain a distance of 25 feet or less, and be in visual observation of a visitor will continue to be required by station procedures.

CCE-97-0002

Commitment Number: 16510 Source Document Number: GNRI-92/0241

Commitment Change Title: Amendment 102 allowed removal of lists of components from TS

COMMITMENT DESCRIPTION: GNRI-92/0241 was Amendment 102 to the Facility Operating License which allowed removal of lists of components from the TS as long as the lists were maintained in a controlled plant procedure and that the UFSAR be updated with these lists.

JUSTIFICATION FOR CHANGE OR DELETION: Safety Evaluation (SE) 93-092 R00 allowed removal of the TRM component listings from 01-S-15-9 since these listings are maintained in the OLM and the UFSAR. Based on the fact that the commitment was actually changed by issuance and approval or SE 93-092 R00, the commitment in this procedure to GNRI-92/0241 is being deleted.

CCE-97-0003

Commitment Number: 24232&24233 Source Document Number: AECM-87/0052

Commitment Change Title: Condition Reporting System Not Used To Identify Receipt Inspection Deficiencies

COMMITMENT DESCRIPTION: The original commitment was made to Unresolved Item 416/86-22-01 documented in MAEC 86/0290. The response documented in AECM 87/0052 reported a new tracking system developed to separately process nonconforming items not installed in the plant. The new system eliminated non-installed equipment nonconformances from the MNCR process and thus allowed the proper focus of resources on nonconformances that bear directly on installed equipment in the operating plant. A change to this commitment took credit for the CR process being able to differentiate between Installed and Non-installed equipment (warehouse items).

JUSTIFICATION FOR CHANGE OR DELETION: The original commitment required a separate process for installed material and non-installed material type deficiencies. The majority of non-installed deficiencies were receipt inspection failures. As GGNS moves its CR process to a similar process with the other EOI sites, the CR will not be used to document material receipt inspection failures. A different process developed by MP&C is utilized for documenting receipt inspection failures. Additionally, the responsibility for receipt inspection performance has been shifted from the QP Group to the MP&C Group. These two groups work under different procedures and processes; and the CR is not the mechanism that is used by MP&C to document receipt inspection failures. Non-Material problems will be processed slightly different than Material related problems. Non-installed Material problems will also be processed slightly different than installed material problems.

Non-installed material problems will go normally be processed directly to the Materials Purchasing and Contracts group for disposition and installed material problems will go to another Engineering Group for disposition.

Another concern in this unresolved item was the apparent lack of focus by the station to material issues. This comment was because there was only one system for processing receipt inspection failures and other material nonconformances, hence the development of two systems. Even though CRs will still be used to document non-installed material nonconformances, the numbers of these deficiencies and the differences in the tracking and processing of CRs will still allow the proper focus of resources on nonconformances that bear directly on installed equipment in the operating plant.

This commitment is no longer needed as the CR will not be used for documenting receipt inspection failures as the MNCR was at the time of the initial violation and processing will still allow proper resource focus on installed plant material issues.

CCE-97-0004

Commitment Number: 32832 Source Document Number: GNRO-97/00011

Commitment Change Title: GL 96-06 Change to Corrective Action Schedule

COMMITMENT DESCRIPTION: Resolve nonconformance of 18 penetrations prior to startup from RFO9.

JUSTIFICATION FOR CHANGE OR DELETION: The resolution of this issue was interpreted as a startup restraint for RFO9 based on interpretations in GL 91-18 Rev. 1, issue October 8, 1997 has revised that interpretation to remove this issue as a startup restraint. Due to emerging new information relative to this issue, resolution is being delayed until startup from RF10.

Restore the 18 penetrations to conformance with the ASME Code.

CCE-98-0001

Commitment Number: 32154 Source Document Number: NEI 91-04

Commitment Change Title: Severe Accident Management Implementation Schedule Change

COMMITMENT DESCRIPTION: Implement BWROG Accident Management Guidelines using Section 5 of NEI 91-04, Rev. 1 no later than 12/31/98.

JUSTIFICATION FOR CHANGE OR DELETION: Implement BWROG Accident Management Guidelines using Section of NEI 91-04, Rev. 1 within 2 years following receipt of the BWROG Accident Management Guidelines. Based on receipt of the BWROG Accident Guidelines in 8/96 implementation date is currently being tracked as 8/98.

CCE-98-0002

Commitment Number: 16847/25058 Source Document Number: GNRO-94/00059

Commitment Change Title: Deletion of SP suction strainer backflushing guidance

COMMITMENT DESCRIPTION: Proceduralize operator guidance on available mean for suppression pool strainer backflush.

JUSTIFICATION FOR CHANGE OR DELETION: Installation of new large passive strainer per ER 97/0089. This new strainer has been designed to maintain approach velocity very low and has significantly more strainer surface area than existing strainers. The new strainer has been specifically designed to satisfactory function under both LOCA-induced and non-LOCA induced debris loading. Backflushing is no longer needed and probably would not be effective due to increased strainer surface area.

CCE-98-0003

Commitment Number: 33021 Source Document Number: GNRO-97/00071

Commitment Change Title: Verifying both SRV logic channels are not defeated simultaneously

COMMITMENT DESCRIPTION: Operations will revise 06-OP-1000-D-0001 to include a sign-off that the operator has verified the opposite division SRV Group Test Switch is not in "Test" prior to placing the other SRV Group Test Switch to Test.

JUSTIFICATION FOR CHANGE OR DELETION: Installation of capacitors in affected panels per ER 97/0313 will prevent spurious initiation of SRV logic. Based on evaluation by System Engineering and experience with trip units in panel since installation of capacitors, the spurious isolation problem is solved. It is no longer required to defeat the SRV logic when performing channel checks per 06-OP-1000-D-0001.

CCE-98-0004

Commitment Number: 32778 Source Document Number: GNRO-96/00086

Commitment Change Title: Core Stability Long-Term Solution Implementation Schedule

COMMITMENT DESCRIPTION: The core stability long-term solution will be implemented and operational at GGNS 6 months following startup from the refueling outage (4th Quarter 1998).

JUSTIFICATION FOR CHANGE OR DELETION: The core stability license amendment request has been submitted to the NRC by letter dated July 20, 1998 (GNRO-98/00053). The core stability license amendment submittal requests a 120 day window from receipt of the NRC's SER to implement the core stability long-term solution at GGNS.

Generic Letter 92-04 required licensees to ensure that their facilities satisfy the requirements of GDC 10 - detect and/or suppress thermal-hydraulic oscillations and GDC 12 - ensure that the MCPR safety limit is not violated. Installation of stability long-term solution hardware was denoted as an acceptable method to satisfy these requirements. The original commitment was made as GGNS' compliance with Generic Letter 92-04 requested actions. The ICAs are acceptable for protecting plants against uncontrolled power oscillations in the interim. The ICAs are currently in place at GGNS.

CCE-98-0006

Commitment Number: 23947&23948 Source Document Number: AECM-83/0152

Commitment Change Title: No work order existed for a specific problem being worked.

COMMITMENT DESCRIPTION: The original commitment was in response to NRC Violation 416/82-65-02 documented by MAEC-82/225, 9/29/82. Administrative Procedure 01-S-07-1, "Control of Work on Plant Equipment and Facilities", was revised to eliminate the use of manpower support work orders. On Green and White tagged systems, SRI,TIPE and Manpower MWO's will not be honored.

JUSTIFICATION FOR CHANGE: The original commitment was identified during start-up testing and system turnover to Plant Staff from Contractor. All Plant Systems have been turned over to Plant Staff for operation. Personnel working on permanent plant components or systems must use a work authorizing document as described in 01-S-07-1, documentation of work performed is required. Manpower work document is irrelevant, as documentation is required for work activity performed.