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December 4, 1995

U.S. Nuclear Regulatory Commission
Mail Station P1-37
Washington, D.C. 20555

Attention: Document Control Desk

Subject: Grand Gulf Nuclear Station
Docket No. 50-416
License No. NPF-29
Report of 10CFR50.59 Safety Evaluations -
January 1, 1994 through June 30, 1995

GNRO-95/ 00132

Gentlemen:

In accordance with the requirements of 10CFR50.59(b), Entergy Operations, Inc. is reporting those changes, tests, and experiments under the requirements of 10CFR50.59 for the period of January 1, 1994 through June 30, 1995. A summary of these changes, tests, and experiments is contained in the attachment. If further information is required, please contact this office.

Yours truly,

MJM/ACG/mtc
attachment: Table of Contents of 10CFR50.59 Safety Evaluations
cc: (See Next Page)

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Serial Number: 93-072-NPE Document Evaluated: DCP 88/0172-01

DESCRIPTION OF CHANGE: This change will install a new remote multiplexer loop to enhance the operation of the plant computer systems. Installation of new fiber optic cable will require opening and closing a 3 psi pressure boundary, opening and closing the secondary containment boundary and opening and closing of penetrations through fire rated barriers. As a result of this change, it was found that the cable flame retardance testing requirements described in the UFSAR required clarification. An UFSAR change request is to be initiated to provide this clarification.

REASON FOR CHANGE: The present plant computer systems have reached an age where they require ever increasing maintenance and improvements to support the needs of Plant Staff.

SAFETY EVALUATION: The plant computer system is not addressed in the technical specifications, and installation of the new computer equipment will not require that a change be made to the technical specifications.

The computer equipment to be installed by this design is not required to mitigate the consequences of any accident or transient, and its installation will not compromise the operation of any safety related system, structure or component. No new interfaces with safety related equipment will be created by the installation of the new computer system components. The opening and closing of penetrations for the installation of the new cable will not affect the penetrations' ability to perform from that previously evaluated. Appropriate penetration seal design requirements, including the functional requirements of UFSAR Section 6.2.3 for secondary containment and fire barrier requirements described in UFSAR Section 9.5.1.2.2.9 have been maintained and the penetration seal details for the affected control and auxiliary building penetrations ensure the integrity of the 3 psi pressure boundary. Operational considerations have also been provided for penetrations affected by this change to comply with the technical specifications and the UFSAR. No portion of this change will alter the radionuclide release rate, duration, create new release mechanisms or impact radiation release barriers.

The UFSAR change associated with this design is to allow the use of cables which have been tested to demonstrate "as a minimum" compliance with IEEE 383 fire test. UFSAR Section 8.3.3.1 and Table 9.5-11, Section D.3.f state that cables used at GGNS have been tested to certify compliance with IEEE No. 383 or ICEA S-19-81 flame retardant tests or are specifically listed as exceptions. The bases for this UFSAR requirement is found in SRP 9.5.1 (NUREG-0800), BTP CMEB 9.5-1, Section C.5.e.3. This requirement states: "Electrical cable construction should, as a minimum, pass the flame test in the current IEEE Std. 383". Therefore, this change is in conformance with the requirements of SRP 9.5.1.

The fiber optic cable to be installed by this design change has not been tested to either IEEE 383 or ICEA S-19-81 and is not listed in the UFSAR as an exception. Although the Siecor cable has not been specifically tested to IEEE 383, its flame retardant characteristics are considered superior to cable which passes IEEE 383.

Serial Number: 93-129-NPE Document Evaluated: DCP 91/0088-9, Rev. 1

DESCRIPTION OF CHANGE: This design change installed a modified plug and stem assembly into the condensate booster pump recirculation valve and installed a completely modified valve design for the feedwater pump recirculation valves.

The modified plug assembly for the booster pump recirculation valve incorporates a dished surface on the downstream face of the valve plug as compared to the originally designed flat surface. The dished surface prevents pressure pulses from impacting rapidly on the plug's lower surface causing vibration. The modification also increases the size of the pressure balancing ports across the plug to allow faster pressure equalization between the bottom of the plug and the valve bonnet area. The plug assembly modification also incorporates a vendor product upgrade for the stem to plug connection. The original stem was threaded into the plug. The upgrade eliminates the stem threads and reduces the potential for stem failure due to stress concentrations in the stem thread.

The modified valve design for the feedwater pump recirculation includes chrome-moly valve bodies, chrome-moly outlet reducers, piloted port plug design, and increasing area flow passage through the drag valve cage.

Finally, the modification includes a review of the current replacement parts and materials provided by the manufacturer. The DCP supplement documents the review and approval of the original equipment manufacturer's (OEM's) current replacement part bill of materials. The review disapproves the use of Stellite #6 for the seat ring overlay and provides for alternative materials. The review also approves other packing material in addition to the vendor supplied Grafoil packing, but prohibits the use of teflon packing normally supplied by the manufacturer.

REASON FOR CHANGE: These valves have exhibited excessive vibration. The modifications will reduce the vibration caused by downstream pressure pulsations. The vendor has changed materials for several replacement parts since the valves were originally purchased. The modification provides approvals and limitations for the current replacement parts, including prohibiting Stellite, which has been incorporated by the OEM in the upgraded product line since original GGNS valve purchases.

Both feedwater pump recirculation valves experienced steam erosion of the valve body, while one valve's steam leakage resulted in through-wall erosion at the outlet reducer. The modifications to the valves are designed to improve the ability of the valve plug to maintain a tight seal when closed, while also providing for erosion resistant materials downstream of the valve plug in the event that seat leakage does occur.

SAFETY EVALUATION: The plant equipment subject to modifications under this DCP supplement serves no safety function, and is non-safety related. All design changes associated with this DCP supplement are non-safety related. All modifications are designed in accordance with the original specifications for system reliability. The changes under consideration do not affect any equipment required to perform safety functions in response to plant accidents and/or transients previously evaluated. No new accident or transient initiating event contributors are introduced by the subject changes. Therefore, the subject changes do not introduce any unreviewed safety questions, or represent any changes to the GGNS Technical Specifications.

Serial Number: 93-131-NPE Document Evaluated: DCP 91/0201-01, Rev. 0

DESCRIPTION OF CHANGE: This change provides the design and installation of electrical power cable in a section of ductbank that is normally energized and will require the referenced ductbank to be completely de-energized. Therefore the work will be performed during refuel outage six. The area of construction is the site area yard and the 500 kV switchyard. The work to the 13.8 kV site power loop under this supplement will not impact any safety system in the plant, as this supplement does not energize the installed cables. This change also will evaluate the electrical impact to plant systems related to the overall site power loop project.

REASON FOR CHANGE: The installation of electrical power cable in a section of ductbank that is normally energized would constitute a personnel hazard. Therefore, the referenced ductbank should be completely de-energized prior to any work in the referenced area.

SAFETY EVALUATION: The 13.8 kV site power loop is a non-safety related system. The function of the site power loop is to provide AC power to most of the facilities external to the power block. The implementation of the design will not impact plant safety because it will be installed when all of the circuits in the referenced ductbank have been de-energized and the electrical power cables will not be terminated. DCP 91/0201-00, Revision 0 will perform remaining design and construction. After construction of the switchgear pads and conduit, the finish grading of the site yard area will be returned to their pre-construction condition and will not impede the natural flow of water to storm drains and probable maximum precipitation (PMP) drainage structures. Although the system is non-safety related, the design of the system will conform to the IEEE, i.e., references listed above so the design will be consistent with engineering and industry practices.

Serial Number: 93-140-NPE Document Evaluated: MCP 93/1071

DESCRIPTION OF CHANGE: This change provides design documentation for the installation of a connection to the Circulating Water System for the following:

a. injection of water treatment chemicals;
b. utilization of monitoring equipment;
c. withdrawing a sample of circulating water;
d. other uses as determined by Plant Staff.

REASON FOR CHANGE: The fill in the GGNS natural draft cooling was replaced during RF05 due to a loss of cooling tower efficiency resulting from excessive fouling. Based on ongoing evaluations of water treatment options, Plant Chemistry has determined that injection of water treatment chemicals immediately upstream of the cooling tower is desirable. This change installs connections in the piping at the inlet to the cooling tower for these purposes. The materials installed per MCP 93/1071 need not be compatible with the water treatment chemicals since the chemicals will be injected through a quill that will protect the connections.

SAFETY EVALUATION: The Circulating Water System is not addressed in the GGNS Technical Specifications; however, the condenser vacuum setpoint is addressed. None of the evaluated changes alters or affects the condenser vacuum low setpoint as addressed in Technical Specification 3/4.3.2 for main steam line isolation.

Loss of the Circulating Water System may result in a turbine trip through the loss of condenser vacuum. Implementation of the evaluated changes will not adversely affect the circulating water supply to the condenser and will therefore not adversely affect condenser vacuum. All of the described changes will maintain all controls normally depended upon to respond to changes in Circulating Water System operation.

The changes will not alter or affect the operability of existing safety related equipment. In addition, a Circulating Water System analysis has shown that failure of the Circulating Water System will not compromise any safety related systems or prevent safe shutdown.

The design change will not alter the design, function, or operation of any equipment important to safety as evaluated in the UFSAR. The Circulating Water System serves no safety related function. The changes 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.

No additional modes of failure are created by implementation of the described changes. Therefore, the existing evaluations are considered bounding for the system.

The technical specifications do not contain any margins of safety for the operation or design of the Circulating Water System. Implementation of the described changes will not affect or prevent safe shutdown of the reactor vessel.

Serial Number: 93-147-NPE Document Evaluated: DCP 91/0088-2

DESCRIPTION OF CHANGE: This change replaces the existing mechanical-hydraulic control (MHC) system on both reactor feed pump turbines (RFPT) with a digital electro-hydraulic control (DEHC) system. The modification will include removing the MHC components on the RFPT front standard (from the electric-automatic positioner (EAP) to the secondary operating cylinder connected to the RFPT cam segmented drive gear). The existing RFPT low pressure secondary operating cylinder will be replaced with a high pressure cylinder to provide RFPT governor valve positioner cam rotation. The EAP and the balance of the MHC system will be replaced with digital processors for receiving the flow demand signal and directing the position of the RFPT governor valves.

The modification involves the installation of new hydraulic power units (HPU) for supplying clean, high pressure control fluid to the new high pressure actuator, as well as the installation of new electrical panels for housing the digital processors. The modification also includes a new RFPT control panel, a modification to the RFPT speed indicators, and a modification to the annunciators on the Control Room H13P680 panel. The change also provides software emulation of previous mechanical features in order to retain the control characteristics which conform to UFSAR assumptions and established pump performance criteria. This includes setting the high speed stop (HSS) to provide: (a) $\leq$ 130% nuclear boiler rated (NBR) feed flow on maximum demand; (b) $\geq$115.5% NBR feed flow with two pumps operating for transients; and (c) $\geq$80% NBR feed flow for single pump operation. Also, the low speed stop (LSS) is set low enough to cease all feed flow to the vessel against a vessel dome pressure of 865 psia (850 psig) as required by Paragraph 4.4.3.3 of Reference 12.

A sound power telephone station will be installed at the location of panels 1N21-P001A and 1N21-P001B (Turbine Building, Elev. 113, Area 1).

Implementation of this change requires opening and closing of safety related penetrations as well as opening and closing of penetrations through fire rated barriers. It also includes the opening and closing of the Control Room envelope airtight boundary for installation of new cable and conduit.

REASON FOR CHANGE: The existing electric-automatic positioner (EAP) on the RFPT was manufactured by DAHL and was supplied by GE. The DAHL EAP is no longer available and GGNS has no spare DAHL EAPs. Other original equipment manufacturer (OEM) supplied parts for the RFPT are also becoming obsolete and are more difficult to obtain.

The modification also provides for improved feedwater system reliability by adding redundancy to the RFPT governor valve positioning mechanisms and by installing a clean control fluid system.

SAFETY EVALUATION: This change does not affect the technical specifications, the Bases for any technical specifications, or the operating license. The values and bases for the Minimum Critical Power Ratio (MCPR) operating limits are unchanged. Opening of penetrations through fire rated barriers is addressed under Operating License Amendment 82. The amendment relocated the Fire Protection Technical Specifications to UFSAR Appendix 16A. However, operational requirements have been provided to ensure compliance with Appendix 16A, Section 3/4.7.7. Operational considerations have also been included to ensure that the control room leak rate requirements of License Condition 2.C.38 will continue to be met.

This change does not increase the probability for occurrence of accidents or malfunction of equipment important to safety previously evaluated in the UFSAR because design requirements are provided to ensure compliance with the technical specifications, and equipment quality, reliability, and operating characteristics meet established design and licensing requirements. This change does not increase the consequences of an accident or malfunction of equipment important to safety from that previously evaluated in the UFSAR because appropriate design requirements and operational considerations have been provided to ensure that equipment performance remains within the limits currently assumed in existing accident and transient analyses.

This change also does not create the possibility for an accident or malfunction of equipment important to safety of a different type than any previously evaluated in the UFSAR because limiting system failure modes are unchanged. This change does not reduce the margin of safety as defined in the basis for technical specifications since credible limiting and non-limiting events which may affect fission product barriers remain clearly bounded by existing analyses.

Serial Number: 93-174-NPE   Document Evaluated: MCP 93/1079

DESCRIPTION OF CHANGE: The closure of the bypass stop and control valves (trip function) upon failure of water injection as described in Section 10.4.4 is deleted. Also, the existing water injection to the control fluid pressure switch is deleted. In addition, a bypass is added to the water injection supply to each bypass stop and control valve. The change will affect condensate (N19), control fluid (N32), turbine bypass system (N37) and main and reheat steam (N11) systems.

REASON FOR CHANGE: The turbine bypass system (N37) is designed to control reactor pressure whenever reactor steam generation exceeds turbine steam system (N11) requirements. However, trip function system (N32) of the bypass valves is deleted to minimize inadvertent turbine trip, if water injection is not established within 10 seconds to pressure breakdown assembly of the respective bypass valve. Also, a bypass to the water injection is added to minimize hydraulic transients, when the water injection valve is actuated rapidly to supply water to the respective pressure breakdown assembly of the bypass valve. The change does not affect the water supply from water injection valve to the pressure breakdown assembly of the respective bypass valve.

The condensate (N19), control fluid (N32), turbine bypass system (N37) and main and reheat steam (N11) systems serve no safety function. Systems analysis have shown that failure of these systems will not compromise any safety-related systems or prevent safe shutdown. In addition, the effects of the turbine bypass system malfunctions on the reactor operation are bounded by events presented in UFSAR.

SAFETY EVALUATION: UFSAR Section 3.2 classifies piping of the affected systems (N11, N19 and N37) as "other", meaning loss of system function would not affect safe shutdown of the plant. UFSAR Section 3.2 does not address the N32 system. UFSAR Table 3.2-1 classifies this same piping and associated valves as non-safety related, non-seismic, Quality Group D, and ANSI B31.1. This design change deletes the trip function of the turbine bypass stop and control valves upon failure of water injection, and deletes water injection to control fluid pressure switch. In addition, bypass is added to water injection supply to the pressure breakdown assembly of each bypass stop and control valve. This change is designed in accordance with ANSI B31.1, Power Piping Code and meets all applicable design and installation requirements. The change will not affect any equipment important to safety. The modifications made by this design change will not impose a change to these criteria listed in Table 3.2-1. Furthermore, the postulated bypass system malfunction and the effects of such failures on other systems and components are clearly bounded by UFSAR, Accident Analysis.

Serial Number: 94-001-NPE   Document Evaluated: CN 92/00349

DESCRIPTION OF CHANGE: This change will provide a design to install one each "enable/disable" handswitch for radial pumps SP47C001J-N and SP47C001K-N in panel SP47-P003. This "enable/disable" handswitch will isolate the manual start/stop permissive from the control room to the Network 90 digital control system for these pumps, in support of programming and troubleshooting of the Network 90 system. In addition to installation of these switches, this design will provide a setpoint for the lube oil level switches associated with the above referenced radial pumps. All work per this change is non-safety related and non-seismic.

Per Section 9.2.10 of the UFSAR, "The radial well system has no safety-related function as defined in Section 3.2 of the UFSAR. Failure of the system will not compromise any safety-related system or component and will not prevent safe reactor shutdown. The radial system provides make-up to the Standby Service Water System cooling tower basins through the PSW system, but this makeup capability is not required to safely shut down the reactor following a LOCA".

REASON FOR CHANGE: No convenient method presently exists for isolation of the Network 90 system controls for Well Number 4 from manual pump start/stop inputs from the control room. This change will allow local enabling or disabling of the Network 90 system from manual start/stop inputs from the control room for each pump of Well Number 4.

SAFETY EVALUATION: The consequences of an accident previously evaluated in the UFSAR will not be affected by implementation of this change as no accident parameters are being modified and no existing safety function are being modified. This change will be performed in accordance with the electrical separation/isolation requirements of Regulatory Guide 1.75 which will insure the ability of safety related systems to perform their intended functions and be maintained. The UFSAR change is a figure change only to Figure 9.2-027-02.

Serial Number: 94-002-PSE   Document Evaluated: WO #113753

DESCRIPTION OF CHANGE: The Heater Drain Tank Level Control Valves 1N23F518A,B,C,D&E are air operated valves. These valves share a common air supply to the operators. Valve 1N23F518E will not control in the auto mode due to air leakage in the supply line. The air supply to the operators closes the valves to maintain level in the heater drain tank. If the air supply is removed the valves will open allowing level to decrease in the tank. The valves will be gagged closed and the air supply line leaks will be repaired.

REASON FOR CHANGE: Since the valves share a common supply line with no isolation valves all the level control valves will need to be gagged closed to prevent loss of level in the heater drain tank when the air leaks are repaired.

SAFETY EVALUATION: The heater drain tank level will need to be monitored during the performance of the work. If the tank levels should increase to an undesirable level then the gags should be removed from the 1N23F518A,B,C&D valves. This would allow the control valves to maintain the proper drain tank level. The level should be maintained with the four control valves.

Serial Number: 94-003-NPE   Document Evaluated: FSAR CR 93/0027

DESCRIPTION OF CHANGE: UFSAR Table 9.4-7 "MISCELLANEOUS SAFETY-RELATED VENTILATION AND COOLING SYSTEM COMPONENT DESCRIPTION" presents information concerning the performance of the various safety related air handling units. This table is being revised to delete the current listed cooling capacity and replace that value with the cooling capacity required to maintain the space served at its design conditions.

REASON FOR CHANGE: The heat transfer values currently listed in UFSAR Table 9.4-7 reflect cooler capacities other than the actual load seen in the space served by the cooler. These values are being revised to reflect the current required cooling capacities of the units.

SAFETY EVALUATION: The heat transfer values currently listed in UFSAR Table 9.4-7 are being replaced with the actual load seen in the space served by the cooler. These values are all within the capabilities of the coolers. The cooler performance is not affected. The change makes the UFSAR consistent with the HVAC calculations.

Serial Number: 94-004-NPE Document Evaluated: MCP 93/1020

DESCRIPTION OF CHANGE: This change will replace the existing instrument air dryer mufflers with mufflers of a design that is not as susceptible to moisture and desiccant fines.

REASON FOR CHANGE: ERR 93/6042 requested a change in the instrument air dryer muffler. The present muffler design appears to be susceptible to a buildup of moisture and desiccant fines which can result in a restriction of air flow during the blowdown and purge phases of the drying cycle. If the air flow through the muffler is unnecessarily restricted, a high back pressure results in inefficient dryer operation and moisture can remain in the desiccant towers which may not be removed during the purge cycle. This results in a higher instrument air dew point.

SAFETY EVALUATION: The change will not affect any present GGNS Technical Specification nor does it impose any new requirements in the technical specification. The installation of instrument air dryer mufflers of a different type will not adversely affect the operation of the air dryers or the instrument air system. In addition, no safety related or non-safety related component will be adversely affected by the change. All required instrument air quality standards will be retained and air quality will not be degraded by the change. The installation of a dryer muffler of a different type will not compromise any safety related system or component, prevent a safe shutdown or degrade any component's ability to mitigate the consequences of an accident.

Serial Number: 94-005-PSE Document Evaluated: Temp Alt 94-0002

DESCRIPTION OF CHANGE: A temporary filter will be installed in parallel with the normal flow path from the equipment drain demineralizer (SG17D002) to the equipment drain sample tanks (SG17A003A&B). This filter will remove resin fines, filter media particles, iron particles, and other impurities from water being transferred into the equipment drain sample tanks for makeup to condensate storage tank (CST).

This change will require removal of the actuator and bonnet from valve SG17F063A and the bonnets from Check Valves SG17F068A&B. The inlet connection for the parallel flow path to the temporary filter will be via a modified bonnet through the SG17F063A valve body with discharge into SG17F068A and/or B, also via modified bonnets. Control of flow discharge path into Equipment Drain Sample Tank A or B will be by manual valves installed with 3" carbon steel pipes by the temporary alteration. Valves SG17F067A&B will be disabled closed to divert flow from the normal flow path through the temporary filter.

Valve position interlocks between SG17F063A and B will be defeated. The off normal recirculation path will be available through the SG17F063B. The conductivity interlock between SG17F063A and conductivity cells (upstream and downstream of the equipment drain demineralizer tanks), will be defeated.

REASON FOR CHANGE: The filter (supplied by MEMCOR) will be installed on the outlet side of the equipment drain demineralizer tank to eliminate "resin fines". "Resin fines" have been found by Chemistry, through a series of tests, to cause conductivity spikes in the reactor. The equipment drain demineralizer is introducing "resin fines" into the reactor by the way of equipment drain sample tank, CST and the hot well. The automatic controls to Valves SG17F067A&B will be defeated so that these valves will be the boundary between equipment drain demineralizer (SG17D002) and the equipment drain sample tanks (SG17A003A&B), respectively.

SAFETY EVALUATION: The routing of 3" carbon steel pipes and location of the new filter and valves will be contained within the same area/rooms as existing piping that has been analyzed by the UFSAR (all spill/release mechanisms, resulting from this temporary alteration, are bounded by existing analysis). The piping, filter and valves that will be installed per this temporary alteration will comply with system design UFSAR 11.2.2.1 and are compatible with Plant Chemistry requirements. System interlocks associated with the disabled valves are bounded by chemistry sampling of water before discharge to CST. The installation of the temporary alteration does not change the technical specifications nor reduce the margin of safety.

Serial Number: 94-006-PSE Document Evaluated: TSTI 1N21-93-009-0-S

DESCRIPTION OF CHANGE: Perform a chemical tracer test for determination of feedwater flow rate. This test will be performed by injecting $_3Li^7$, in the form of lithium nitrate solution (38,000 ppm Li) and diluted with demineralized water (approximately 2,000 :1), with temporary injection equipment into each train of the feedwater system at a point just upstream of the #5 feedwater heater. A total of 2.7 grams of Li will be injected over a period of 3 minutes and 50 seconds. Multiple samples will be extracted at specific intervals from a point downstream of each feedwater flow element. (The samples are subsequently analyzed for chemical tracer concentration and the mass flowrate of the feedwater system is then calculated based on the sample results.) This test will be simultaneously performed on both trains of the feedwater system. This testing will be performed in OPERATIONAL CONDITION 1. Temporary injection and sampling equipment will be installed as required to accomplish this task. The ABB-CE CHEMTRAC™ System to be installed has been designed for independent measurement of feedwater flow and in-service testing of Safety Class 1, 2, and 3 centrifugal and displacement type pumps as required by Section XI of the ASME boiler and pressure vessel code per Reference 2.

REASON FOR CHANGE: In order to verify the accuracy of the installed feedwater flow rate measurement loops.

SAFETY EVALUATION: A safety evaluation concludes that the proposed testing:

(a) Will not exceed the technical specifications limit for reactor water conductivity,

(b) Meets the chemistry limits,

(c) Is bounded by the loss of feedwater accident analysis in terms of consequences,

(d) Cannot affect the conclusions of the Increase in Reactor Coolant Inventory accident analysis,

(e) Will not result in unduly high lithium concentration that could cause accelerated Zircaloy corrosion, and

(f) No margin of safety will be reduced by the proposed testing.

Serial Number: 94-007-NPE Document Evaluated: MCP 91/1122, Rev. 0

DESCRIPTION OF CHANGE: This change is issued to delete the reactor feed pump discharge pH monitor for the Process Sampling System and remove or spare all equipment used for remote monitoring. This change does not include the removal or sparing of the annunciator and computer points. Annunciators and computer points are under the scope of MCP 91/1145.

Components that will be removed under MCP 91/1122 are the following:

Analyzer Element - 1P33-AE-N003
Transmitter - 1P33-AIT-N004
Recorder - 1P33-AR-R011
Flow Indicator/Controller - 1P33-FIC-R044B

REASON FOR CHANGE: The reactor feed pump discharge pH monitor is no longer used. The feedwater chemistry is controlled by operations based on grab samples analyzed in the chemistry laboratory. Therefore, the monitor components should be removed as appropriate for the complete demolition of this monitoring instrumentation.

The original design intent of this monitor was to be able to measure the pH of the reactor feed pump discharge for the purpose of automatic pH checking of the solution. Based on the results of this sample, injection of acid is performed in order to control pH in the reactor feed water. However, this monitor is not up to current technology standards for automatic analysis and recording of system pH. The RFP pH is determined by analyzing grab samples in the chemistry laboratory. Injection of acid is based on these results.

SAFETY EVALUATION: All of the equipment, instrumentation and tubing associated with the reactor feed pump discharge monitor of the Process Sampling System are non-safety related. In addition, the analyzer element does not provide a safety-related pressure boundary of the discharge piping. The analyzer element which is downstream of the F973 valve shall be disconnected and removed. The isolation valve, F973 shall be closed and capped. There are currently no Class 1E electrical interfaces associated with the monitoring equipment. No electrical or mechanical safety-related interfaces shall be either created or deleted by implementation of this change. The affected system provides indication and annunciation only; it performs no actuation function.

This safety evaluation is issued primarily to determine the impact of changing Figure 9.3-6 to reflect the removed pH monitoring equipment. There is no challenge to plant safety by performing this demolition and updating the Piping & Instrument Drawings.

Serial Number: 94-008-PLS Document Evaluated: 08-S-04-498

DESCRIPTION OF CHANGE: Chemistry Instruction 08-S-04-498, Rev. 0, is a new directive written to provide instructions for adding Sodium Hypochlorite to the N71 Circulating Water System (CWS) through the N71-FX409A and N71-FX409B valves.

In the proposed procedure, HYPO is to be shipped in from offsite. Injection frequency is set for three times per week at 1 to 2 hours for each injection. The point of injection is to be the 120" hot water return lines to the cooling tower. In this way, HYPO is applied directly to the area of concern (cooling tower fill) first. A free chlorine residual of 1.0 to 1.5 ppm is to be maintained for each injection period. The point of control for chlorine residual is to be the bottom of the cooling tower fill. CWS blowdown will be secured during HYPO injection to prevent chlorine residuals in excess of NPDES limits from reaching the 001 outfall. Chlorine residuals are to be monitored after injection and de-chlorination used as necessary to meet NPDES limits.

REASON FOR CHANGE: The main cooling tower uses a high efficiency fill that is subject to fouling from microbiological activity. Plant experience has shown that fill fouling can measurably reduce overall plant efficiency in as little as two fuel cycles. Chemistry Control has evaluated ways to control this fouling. The most successful approach evaluated is the use of Sodium Hypochlorite (HYPO) in conjunction with a surfactant-type biodispersant. The effectiveness of HYPO was demonstrated from both:

1. in-plant experience with daily HYPO injections to P44 Plant Service Water, and

2. the results of pilot-scale testing on a model cooling tower using the same type and depth of fill found in the main cooling tower.

Zinc was also evaluated for fill fouling control.

SAFETY EVALUATION: Onsite testing has demonstrated the effectiveness of this program for microbiological control in the cooling tower fill. Additional onsite testing has demonstrated that the program will not cause increased corrosion rates, or increased scale formation on affected surfaces. The only metal not directly tested for corrosion rates was the bronze alloy from which the circulating water pump impellers are constructed. Based upon results of daily HYPO feed to PSW over a 2-year period, corrosion rates on similar metallurgies (90/10 cupronickel and 122 copper) were not adversely affected.

To ensure plant NPDES limits are met, the procedure controlling HYPO feed to CWS requires monitoring of the plant outfall for residuals. The N71-F513A and N71-F513B valves will be manually closed during HYPO feed to ensure no chlorinated water is sent to the 001 outfall. Immediately following HYPO feed to CWS, cooling tower blowdown will be monitored for chlorine residuals in excess of NPDES limits.

Serial Number: 94-009-PLS Document Evaluated: 08-S-03-14, Rev. 11

DESCRIPTION OF CHANGE: This change adds a surfactant-based dispersant and a microbicide to the list of chemicals used in P42 Component Cooling Water, P43 Turbine Building Cooling Water, P71 Plant Chilled Water, and P72 Drywell Chilled Water Systems. This change also adds sodium hypochlorite to the list of chemicals used in N71 Circulating Water System. NOTE: This safety evaluation addresses only the changes affecting the closed loop systems, P42, P43, P71, and P72. The change affecting N71 Circulating Water System is addressed in Safety Evaluation SE-94-006-R00.

REASON FOR CHANGE: The failure of the evaporator tubes in Drywell Chiller 1B on January 2, 1994, resulted in ingress of freon and oil to the closed loop chilled water system. Most of the freon flashed to the atmosphere by way of the P72 makeup tank on 185' Auxiliary Area 10. However, the oil remained in the system causing fouling of piping and heat transfer surfaces. Organic fouling was confirmed by inspection of corrosion coupons in the system.

Left untreated, sessile bacterial masses will grow in the oily foulant layer. Bacteria can cause (1) loss of heat transfer in the drywell chillers and drywell coolers, (2) degradation of the nitrite-based corrosion inhibitor, and (3) microbiologically influenced corrosion.

A dispersant is needed for flushing foulants from the system. The product selected for this application is CALGON CL-362. However, any similar cationically-charged surfactant-based dispersant would be suitable for flushing organic foulants from the system.

A microbicide is needed to inhibit the growth of microbial colonies. The product selected for this application is CALGON H-300. However, any commercially available 45% glutaraldehyde solution is suitable for this application. Glutaraldehyde is particularly suited to this application because its biocidal properties are quickly activated in alkaline pH environments such as the closed loop systems. Glutaraldehyde also breaks down (time rate varies depending on specific application) to non-toxic components, making it relatively safe for discharge.

This evaluation addresses not only the use of these chemicals in P72, but also addresses their use in P42, P43, and P71 for future reference should application be deemed necessary by Chemistry Control.

SAFETY EVALUATION: The application of dispersant and microbicide will return fouled systems to a cleaner and therefore more efficient and reliable operating condition. The use of these chemicals reduces the possibility of corrosion inhibitor degradation due to microbiological attack. They also prevent the onset of microbiologically influenced corrosion. The potential corrosivity of these chemicals has been evaluated and is insignificant at use dosages. The surfactant has been evaluated for foaming and pump cavitation potential and found to be satisfactory.

Serial Number: 94-010-NPE   Document Evaluated: MNCR 0119/91 Final Disposition Third Sub.

DESCRIPTION OF CHANGE: Incorporates software changes documented by as-built walkdown into plant communications drawings. Adds a note to these drawings to allow type HAJ paging stations to be substituted for Type HA paging stations. These drawings are included in UFSAR Chapter 9 as Figures 9.5-9b thru 9.5-9i.

REASON FOR CHANGE: To resolve non-conformances between plant communications drawings and actual as-built field conditions. The replacement of Type HA stations with Type HAJ stations is required due to the manufacturer providing Type HAJ as replacement part.

SAFETY EVALUATION: MNCR 0119/91 was written to identify discrepancies between communications system drawings and actual field conditions. This disposition will provide as-builts for the drawings and review the installation of one additional communication station in the control room. This disposition will also allow the substitution of Type HAJ paging stations for Type HA paging stations. Installation of this station, along with the necessary cabling, or the substitution of HAJ for HA paging stations will not increase accident or malfunction probabilities or consequences. It will not create the probability of a different type of accident or malfunction of equipment and does not reduce any margin of safety described in the technical specification bases.

Serial Number: 94-011-NPE Document Evaluated: MCP 92/1047

DESCRIPTION OF CHANGE: The PASS in-line atmospheric oxygen analyzer system will be replaced by an Orbisphere dual oxygen/hydrogen analyzer system that is pressure and temperature compensating. The new system consists of an oxygen sensor, a hydrogen sensor, a pressure sensor and the indicating instrument. The tubing will be rerouted so that the flow through the sensors will be discharged into the drywell instead of the turbine building vent system. The hydrogen calibration gas and instrument purge air for the hydrogen sensor will be discharged at the intake of the post accident sample room filter train.

REASON FOR CHANGE: The Beckman atmospheric oxygen analyzer cell P33-AE-N136 is not working. This is because the cell is not pressure compensating as advertised by the vendor.

The Chemistry department would also prefer to have a separate P33 system PASS hydrogen analyzer. Presently, the E61-J001A, 1B, 2A, 2B hydrogen analyzers are utilized.

SAFETY EVALUATION: The changes are non-seismic Category I, non-seismic Category II/I and non-safety related. These instruments are for operator convenience only. They can be used to determine the concentration of oxygen and hydrogen gas in the containment/drywell atmosphere following an accident but are not required for Regulatory Guide 1.97. Per UFSAR Table 7.5-2, GGNS uses grab samples to determine this information. The changes of this MCP will not prevent the grab samples from being taken.

These changes 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 reliability and monitoring capability. The failure of PASS will not initiate any evaluated transient or accident. The in-line instrumentation is not required to mitigate the consequences of any evaluated transient or accident. During LOCA conditions, the drywell/containment atmospheric samples will be highly contaminated. Failure of the hydrogen sensor membrane is not a concern because the instrument purge air exhaust is discharged at the intake of the post accident sample room filter train. The instrument air line is protected from backflow by two check valves. Discharging the hydrogen calibration gas into the intake of the post accident sample room filter train is not a concern because the amount of hydrogen introduced is insignificant (.11 scfh vs. 500 scfh). Discharging the flow through the sensors to the drywell instead of the turbine building vent system (via the post accident sample room filter train) may reduce radiological release. No interfaces with safety related or important to safety systems are created. This change will therefore not introduce an unreviewed safety question. The in-line PASS instrumentation is not addressed in the technical specification (ref. Technical Specification Sections 6.8.3.a.7 and 6.8.3.c).

Serial Number: 94-012-NSRA Document Evaluated: TSPS 128 Rev. 1

DESCRIPTION OF CHANGE: Change the Revision 0 Technical Specification Position Statement (TSPS) 128 required actions in the event of inoperable ESF switchgear room coolers from an allowed out of service time of 72 hours for a single ESF switchgear room cooler and a management escalation process to ensure that timely corrective action is being commensurate with the safety significance of the out of service equipment if not restored in 72 hours. Also, the requirement to enter Technical Specification ACTION 3.8.3.a.1 if more than one ESF switchgear room cooler is inoperable. These actions are changed to requiring equivalent actions to the actions required by the technical specifications for the associated standby service water being inoperable unless it has been shown that the associated room will stay within its evaluated temperature during a DBA.

REASON FOR CHANGE: MNCR 94-0028 identified that certain ESF switchgear rooms would not stay within their evaluated temperatures during a DBA if the associated ESF switchgear room cooler was inoperable. As a result, more restrictive actions are needed for the evaluated ESF switchgear room cooler inoperabilities.

SAFETY EVALUATION: The actions required by TSPS Revision 1 are equivalent to the actions required by the technical specifications for associated inoperable SSW subsystems for any ESF switchgear room cooler failures which may result in the evaluated temperatures for the room being exceeded. The affects of the failures evaluated are bounded by the condition of an associated inoperable SSW subsystem including the effects being limited to the associated division and the TSPS required actions are equivalent to the technical specifications required ACTIONS for SSW inoperabilities. Therefore, the evaluated inoperabilities and the associated TSPS required actions are bounded by the SSW subsystem failures evaluated in the UFSAR and form the bases for the associated technical specifications.

Serial Number: 94-014-NPE   Document Evaluated: MNCR 93/0366

DESCRIPTION OF CHANGE: The damaged sections of Culvert 16 are being replaced, but without the security grating indicated in UFSAR Table 2.4-28. Concrete pads blocking complete closure of Turbine Building Door 1T301 are being removed to allow the door to close completely. The floor slab under the affected section is being refinished. UFSAR Table 2.4-28 is being revised to delete the reference to security grating on Culvert 16.

REASON FOR CHANGE: Culvert 16 has been partially crushed due to the passage of heavy vehicular traffic. UFSAR Table 2.4-28 currently shows that security grating is required to block passage through Culvert 16. However, Culvert 16 is located in the Owner Controlled Area but lies entirely outside the Protected Area. Access to the Protected Area cannot be gained by crawling through the culvert. Therefore, the grating is not required. Door 1T301 cannot effectively function as a fire door with the concrete pads blocking closure. Additional leakage into the power block can result during heavy precipitation as a result of its current condition.

SAFETY EVALUATION: The deletion of the security grating for Culvert 16 will add conservatism to the probable maximum precipitation (PMP) drainage calculations contained in Calculation CC-Q1Y23-91050, Revision 0, Pages 29-32. As a result, the water levels recorded for Area J in UFSAR Section 2.4.3.5.3 will be somewhat overstated (conservative). Therefore UFSAR Table 2.4-28 will be revised to delete the requirements for a security grating for Culvert 16. Door 1T301 is being returned to a condition which will allow it to serve its intended function. Door 1T301 is not among the doors governed by the Technical Requirements Manual (fire protection requirements are no longer contained in the technical specifications.)

Serial Number: 94-015-NPE Document Evaluated: DCP 88/0040 Rev. 0

DESCRIPTION OF CHANGE: This change installs two current operated transistor switches in each of the 193 hydraulic control unit (HCU) terminal boxes. In each terminal box, the power lead for both the "A" and "B" scram pilot solenoid valve will be replaced with a #16 AWG jumper of sufficient length to be routed through an integral current transformer (CT) window of one of the two switches and re-terminated to the original locations. The transistor switches for selected HCUs will be wired in series with existing non-safety related low accumulator pressure and high accumulator level alarm contacts within the HCU. The transponder cards for the RC&IS circuits associated with these particular HCUs is modified by replacing a 47 Kohm resistor (R16) with a 2 Kohm resistor. This replacement prevents an excessive voltage drop across R16 due to leakage current when the transistor switch is in the "off" state.

A stick-on symbol of the appropriate color is installed on the RC&IS control rod display screen to designate which HCU locations are monitored. In addition, the mylar insert for the "SCRAM ACCUM TROUBLE" (1H13-P680-4A2-D4) is replaced with a new insert which reads "HCU TROUBLE" and the labels for the Jay-El pushbuttons located on 1H13-P680-6C which presently read "ACCUM FAULT" and "ACKN ACCUM FAULT", a portion of 1C11-HSS-M610 and 1C11-HSS-M611, respectively, is modified to read "HCU FAULT" and "ACKN HCU FAULT".

REASON FOR CHANGE: At present, there is no indication or alarm function to detect the loss of power to the scram pilot solenoids. In March, 1988, a loose connection in a daisy-chain of power to scram pilot solenoids in RPS Loop "A" resulted in a scram when coupled with a surveillance test on RPS Loop "B". This change re-terminated field cables and installed jumpers within each of the eight pilot scram solenoid terminal boxes to prevent a single loose connection, with the exception of the field cable feeds, from de-energizing more than one solenoid. This change provides the ability to detect a loss of power to the daisy-chains.

SAFETY EVALUATION: The physical installation of the current operated transistor switches has been evaluated for Seismic II/I concerns and been determined to be acceptable due to the low mass of the switches. The wire use for the jumper replacements required for the scram pilot solenoids has adequate current carrying capacity and is Class 1E qualified. The replacement resistors are of similar composition and wattage rating as the original resistors and will be installed on the transponder cards in the same fashion as the existing resistors.

Since there is no interface between the scram pilot solenoid power circuits and the current-operated transistor switches and there are no Seismic II/I concerns, there will be no adverse impact on this safety system.

Serial Number: 94-016-NSRA Document Evaluated: Bi-Annual 10CFR50.59 Screening Effectiveness Audit

DESCRIPTION OF CHANGE: As a result of problem areas identified during a GGNS 10CFR50.59 audit by the NRC in April 1988, we committed to perform periodic audits of 50.59 screening effectiveness. After discussions involving our Licensing and Quality Groups, responsibility for the audits was given to Quality Programs, and the frequency established was every two years. The proposed change will give the responsibility for evaluating 50.59 screening effectiveness to the Nuclear Safety and Regulatory Affairs group, and will increase the evaluation frequency to quarterly, or as decided based on results, not to exceed two years.

REASON FOR CHANGE: NS&RA is the process owner for the 50.59 program. The most effective means to improve the quality of 50.59 screenings and evaluations is for the process owner to frequently perform internal assessments and transmit lessons learned and constructive comments to qualified reviewers on a timely basis. Any adjustments to the process which may be deemed necessary can then promptly be implemented. In addition, information concerning adverse trends or weaknesses can be provided to the PSRC and SRC to satisfy their oversight requirements.

The criteria for evaluating Screening and Evaluation Effectiveness is one part of a comprehensive 50.59 self-evaluation process that NS&RA is constructing. We are establishing a set of indicators which will allow us to perform frequent assessments of the health of the 50.59 program, and we will provide the results of those assessments to both the quality organizations and the PSRC and SRC. In summary, we will focus our improvement efforts on areas that favorably impact the program from the perspective of both end-users and evaluators.

SAFETY EVALUATION: Grand Gulf originally committed to perform these audits to cause an improvement in the effectiveness of 50.59 screenings, to ensure that the higher-quality of screenings would be maintained, and to direct management attention towards these areas through oversight activities. Transferring the responsibility for these evaluations to the process owner will not cause a reduction in the level of safety called for by 10CFR50.59, and will, in fact, lead to higher quality screenings and evaluations through timely feedback to both evaluators and review committees. NS&RA will provide the quality organization with the review results, which they can review and input into their overall performance monitoring effort.

Serial Number: 94-017-PSE Document Evaluated: WO #118447 and Temp Alt #094-0005

DESCRIPTION OF CHANGE: This change will lock open one of the four banks of louvers associated with the steam tunnel cooler outside containment 1T41B011. This alteration will be performed in accordance with the control of Temporary Alterations Administrative Procedure 01-S-06-3.

REASON FOR CHANGE: Installation of the temporary alteration will be necessitated by:

-- the need to stabilize and/or lower the main steam tunnel temperature. Its temperature has been trending upward and may soon encroach upon Technical Specification 3/4.7.8 temperature limit of 125°F due to steam leaks within the steam tunnel, the unavailability of Plant Chiller "A", and the broken closed louver on 1T41B011.

-- high radiation levels within the main steam tunnel require a quick fix if the linkage mechanism for the damper can not be reworked to design configuration from outside the cooler housing.

SAFETY EVALUATION: The safety evaluation has concluded that there are no unreviewed safety questions associated with the installation of the temporary alteration which locks open one of the four sets of louvers for Steam Tunnel Cooler 1T41B011.

Installation of the temporary alteration will not impact the Seismic Category I status of 1T41B011. Furthermore, its installation will not create any Seismic II/I concerns nor missile hazards for the safety related components housed within the steam tunnel outside containment.

No new failure modes of safety related components within the steam tunnel are created by installation of the temporary alteration.

No new radionuclide release mechanisms are created nor will there be an impact on radiation release barriers with the installation of the temporary alteration.

The worst case outcome that may result from installation of the temporary alteration would culminate with the entrance into the Action Statement associated with Technical Specification 3/4.7.8. Accordingly, the margin of safety as defined in the basis for any technical specification will not be reduced by installation of the temporary alteration.

Serial Number: 94-018-PSE   Document Evaluated: DCP 82/3590 Rev. 1

DESCRIPTION OF CHANGE: This change replaces cable 1ED2681 with a larger size cable. Revision 1 of DCP 82/3590 provides hangers to support conduit 1BERN686 (which is part of the routing of the cable replaced in Revision 0) and provides for sparing the 'old' smaller cable. The conduit support work was completed by another design package (DCP 83/0065-01 and CN 86/0040) and has been deleted from DCP 82/3590 Rev. 1 by CN 93/0050. The remaining work to complete and close DCP 82/3590 Rev. 1 is the sparing and labeling of the old cable. The spare cable is to be coiled at the "From" end in tray 1TETNE82 and the "To" end is to be coiled in the pull box at the end of conduit ERNS34. The cable is to be labeled SPARE0332.

REASON FOR CHANGE: The reason for the remaining work is to adequately identify and spare the unused cable. Sparing the cable for possible future use is the most economic decision. Labeling as spare provides for ease in locating the cable for troubleshooting or later use.

SAFETY EVALUATION: This safety evaluation is being accomplished because the original 1982 safety evaluation does not meet 1994 safety evaluation standards. No unreviewed safety question or change to technical specification exist as a result of the change identified in any portion, completed or incomplete of DCP 82/3590 Revision 1.

Serial Number: 94-019-NPE Document Evaluated: MCP 90/1050, CN 93/0106 and SCN 94/0027A

DESCRIPTION OF CHANGE: This design modification adds a flow-indicating transmitter to measure flow in the containment ventilation duct during the low-volume-purge mode of operation. The flow-indicating transmitter will provide local indication and a signal to Process Radiation Monitoring Panel D17-P018. Flow indication from the D17-FIT-R300 transmitter will be available both locally and via the D17-J600 control terminal in the Control Room and the D17-J100 control terminal in the TSC.

REASON FOR CHANGE: The purpose of this change is to provide flow indication of the containment ventilation duct. This indication is to be used to manually match the velocity in the Containment Ventilation Radiation Monitoring Sample Panel (N1D17-P002) during the low-volume-purge mode of operation.

SAFETY EVALUATION: The changes made by this modification do not create any unreviewed safety questions. The instrument added does not perform any safety function and does not affect the qualifications of any other instrumentation. This new instrument provides indication only and does not affect any safety-related parameter in the plant. The instrument tubing that is modified for this change is non-seismic Category I, non-seismic Category II/I, and non-safety related. The power supplied to the instrument is non-1E power. The conduit supports from the instrument to panels 1D17-P002 and 1D17-P018 are non-safety related but are designed to meet the requirements of Seismic Category II/I. The instrument tubing modified by this design package is connected to Regulatory Guide 1.97 post accident monitoring instrumentation, 1D17-N200B&H. 1D17-N200B&H are Regulatory Guide 1.97 Type E, Category 3 variables and have no safety related function. There are no Appendix R concerns created by the implementation of this design and divisional separation, per Regulatory Guide 1.75, of electrical components added or modified by this MCP are not compromised by the implementation of this design change.

Serial Number: 94-020-NPE Document Evaluated: SCN 93/0003A for SERI-MS-37 Rev. 2

DESCRIPTION OF CHANGE: Master parts list (MPL) numbers have been assigned to specific components of the Division I and II diesel generator (D/G) pneumatic control systems in accordance with EER 91/6002, MCP 93/1006 Rev. 0, and DCP 86/4014 Rev. 0. These components are routinely maintained by Plant Staff as part of the D/G Design Review/Quality Revalidation (DR/QR) Program. Corresponding MPL numbers have been added per Standard Change Notice (SCN) 93/0003A to the Transamerica Delaval, Inc. (TDI) components listed in Section 3, Page 37 (SERI Maintenance and Surveillance Program) of SERI-MS-37, Rev. 2.

REASON FOR CHANGE: The pneumatic control systems on the Division I and II D/G contain numerous components such as shuttle valves, regulators, etc., that are not uniquely identified because they are part of the D/G package. Some of the components routinely maintained by Plant Staff have been identified in Table 1 of EER 91/6002. Several of the components identified in EER 91/6002 are also listed in SERI-MS-37, Revision 2.

The D/G vendor (TDI) has assigned item numbers in addition to descriptions and part numbers for these components. However, in some instances these numbers are either illegible on vendor documents, or identically numbered components are installed in separate areas in the field. These conditions impede maintenance activities, and create the potential for error. Therefore, MPL numbers have been assigned per MCP 93/1006 Revision 0 and DCP 86/4014 Revision 0 to the D/G pneumatic controls systems components identified in EER 91/6002. Labeling the identified valves, regulators, pressure gauges, etc. with MPL numbers simplifies the process of locating the components in the field.

SAFETY EVALUATION: The implementation of SCN 93/0003A will not adversely impact any of the controls, maintenance and surveillance requirements, or the D/G DR/QR baseline inspections contained in SERI-MS-37 Revision 2 which ensure the operability and reliability of the Division I and II D/G. Furthermore, the proposed changes do not affect or change any licensing conditions of the technical specifications, nor result in an unreviewed safety question or reduce the margin of safety as defined in the basis for any technical specification.

Serial Number: 94-021-NPE Document Evaluated: MNCR 0111-93

DESCRIPTION OF CHANGE: Gate valves with blind flanges are installed in branch lines off of the inlet and outlet piping of pump 1P71C001A. These valves do not appear on the P&ID or piping drawings M-1353J&K. The valves are not installed in the main headers but are installed in branch lines off of the main headers (10"JBD-643 and 10"JBD-644).

REASON FOR CHANGE: To document the existent of the branch lines with isolation valves in the Plant Chilled Water System (P71) on the system P&ID and piping drawings. Also to assign valve numbers to these existing valves.

SAFETY EVALUATION: In the subsequent paragraphs, the changes to be made are reviewed to determine if an unreviewed safety question, as defined by 10CFR50.59 will exist.

The two 6" gate valves with blind flanges added to this system will not adversely affect the Plant Chilled Water System (P71) function, operation, or performance in any way. The isolatable branch lines could be used in various situations by plant operations. The valves were installed per ANSI B31.1 code requirements and have been analyzed for seismic loads and are acceptable. Therefore, the P71 system will function in its intended manner. Based on these conclusions MNCR 0111-93 and its disposition ("Accept-As-Is") will not create an unreviewed safety question.

Serial Number: 94-022-NPE Document Evaluated: MCP 91/1142 Rev. 0

DESCRIPTION OF CHANGE: This change is issued to delete the turbidity monitors for the Process Sampling System and remove or spare all equipment used for remote monitoring. This change does not include the removal or sparing of the annunciator and computer points. Annunciators and computer points are under the scope of this change.

Components that will be removed under this change are the following:

Analyzer Elements 1P33-AE-N037 & -AE-N007
Transmitter Switches 1P33-ATS-K005 & -ATS-K002
Transmitters 1P33-AIT-N038 & -AIT-N006

REASON FOR CHANGE: The condensate demineralizer combined effluent turbidity monitor (1P33-AIT-N038) and the forward pumped heater drain turbidity monitor (1P33-AIT-N006) are no longer used. The detector windows are prone to fouling and are not appropriate for the intended function of monitoring turbidity. Therefore, the monitor components should be removed or abandon in place as appropriate for the complete demolition of these monitors.

The original design intent of these monitors was to be able to measure the suspended concentration of corrosion and iron utilizing a conductivity monitor and correlate this to turbidity. However, the actual function is to measure the concentration for conductivity based on the size and shape of the suspended particles, which in practice does not correspond with percentage of iron/corrosion. Also, there is no effective correlation between turbidity and iron or corrosion concentration.

SAFETY EVALUATION: All of the equipment, instrumentation and tubing associated with the turbidity monitors of the Process Sampling System are non-safety related. The analyzer elements do not form a safety related pressure boundary of the discharge piping. These analyzer elements shall be disconnected and removed. There are currently no Class 1E electrical interfaces associated with the monitoring equipment. No electrical or mechanical safety related interfaces shall be either created or deleted by implementation of this change.

Serial Number: 94-023-NPE Document Evaluated: MNCR 0129-93

DESCRIPTION OF CHANGE: UFSAR Figure 11.5-1 (P&ID M-1107G) is being changed to reflect the as-built configuration of the offgas sample piping.

REASON FOR CHANGE: MNCR 0129-93 was written to address a discrepancy between the P&ID (UFSAR Figure 11.5-1), the piping FSK and the as-built plant configuration. In addition, the MNCR requested an evaluation of the material used for the valve internals. It was determined the valve material was acceptable for its intended usage. However, the P&ID incorrectly indicated that a quick disconnect fitting was utilized for the purge air inlet filter D17J010. This filter is hard piped to sample valve 1D17F002 and does not use a quick disconnect type fitting as indicated on the UFSAR figure.

SAFETY EVALUATION: The change in the piping configuration, as depicted in the UFSAR, will not affect any present GGNS Technical Specifications nor does it impose any new requirements in the technical specifications. The air purge filter, sample valve and associated piping are not safety related. Their operation or failure would not compromise any safety related system or prevent a safe shutdown. The change to the UFSAR figure represents the plant configuration and will not result in the failure of any system or component required to safely shut down and cool down the plant or to mitigate the consequences of an accident.

Serial Number: 94-024-NPE Document Evaluated: MCP 94/1005 Rev. 0

DESCRIPTION OF CHANGE: This change provides design documentation for the installation of chemical injection quills for injection of water treatment chemicals to the Circulating Water System.

REASON FOR CHANGE: The fill in the GGNS natural draft cooling was replaced during RF05 due to a loss of cooling tower efficiency resulting from excessive fouling. Based on ongoing evaluations of water treatment options, Plant Chemistry has determined that injection of water treatment chemicals immediately upstream of the cooling tower is desirable. Connections in the piping at the inlet to the cooling tower were installed for these purposes. Suitable injection quills are not available as off-the-shelf items. Therefore, MCP 94/1005 provides the design documentation for fabrication and installation of appropriate injection quills. The parts of the quill assembly that will be exposed to the water treatment chemicals are titanium.

SAFETY EVALUATION: None of the evaluated changes alters or affects the condenser vacuum low setpoint as addressed in Technical Specification 3/4.3.2 for main steam line isolation.

An increase in reactor pressure as a result of a turbine trip is evaluated by UFSAR Chapter 15. Loss of the Circulating Water System may result in a turbine trip through the loss of condenser vacuum. Implementation of the evaluated changes will not adversely affect the circulating water supply to the condenser and will therefore not adversely affect condenser vacuum.

The changes does not alter or affect the operability of existing safety related equipment. In addition, a Circulating Water System analysis has shown that failure of the Circulating Water System will not compromise any safety related systems or prevent safe shutdown.

The design change package does not alter the design, function, or operation of any equipment important to safety as evaluated in the UFSAR. The Circulating Water System serves no safety related function. The changes 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.

Turbine Building flooding by a gross failure of the circulating water piping has been evaluated in the UFSAR. No additional modes of failure are created by implementation of the described changes. Therefore, the existing evaluations are considered bounding for the system.

The technical specifications do not contain any margins of safety for the operation or design of the Circulating Water System. Implementation of the described changes will not affect or prevent safe shutdown of the reactor vessel.

Serial Number: 94-025-NPE Document Evaluated: MNCR 0071-93

DESCRIPTION OF CHANGE: MNCR 93/0071 documents that smoke detector SZ17XSN103 is installed in a section of HVAC air duct which has been disconnected and capped off per FCNs 1334 and 1339. This change will provide administrative controls for manual shutdown of the CB fan coil unit upon detection of a fire by the area smoke detectors. This safety evaluation only applies to those areas serviced by fan coil unit NSZ17B002.

REASON FOR CHANGE: With smoke detector SZ17XSN103 being located in a closed off section of HVAC duct, it was incapable of detecting smoke in the return air duct. Therefore, the existing condition did not allow a means to control the spread of smoke. P65 area smoke detectors will be used instead of smoke detector SZ17XSN103 for the detection of smoke and the CB fan coil unit will be shutdown manually instead of automatically.

SAFETY EVALUATION: Section D.4.a of Appendix A to Branch Technical Position APCSB 9.5-1, dated August 23, 1976, states that the products of combustion which need to be removed from a specific fire area should be evaluated to determine how they will be controlled. To satisfy this guideline, administrative controls and area smoke detectors will be used to provide manual shutdown of the CB fan coil unit.

Manual fan shutdown by operations personnel upon the detection of smoke by area smoke detectors is considered a satisfactory method to control smoke for this system because of the following:

1. The areas served by this system are normally unoccupied.

2. Access to the areas served by this system is not required for manual actions associated with plant shutdown in the event of a fire in the area containing the CB fan coil unit or any area served by this unit.

3. All of the areas served by this system are protected with an ionization type smoke detection system. This type of system is an early warning system.

4. The area smoke detection system provides visual and audible annunciation in the Control Room so that the fire brigade can respond quickly to shut down the fan before a significant amount of smoke is spread to other areas.

5. The administrative controls instructs the fire brigade of the potential for false alarms from areas where smoke has been introduced by the CB fan coil unit.

6. In the event that smoke is introduced to other areas served by the CB fan coil unit, it can be exhausted after the fire by the control building purge system.

Conclusion -- Regulatory guidelines do not dictate how smoke should be controlled and based on the conditions outlined above, manual shutdown of the CB fan coil unit, instead of automatic shutdown, is considered a satisfactory means of controlling smoke for this system.

Serial Number: 94-026-NPE   Document Evaluated: TSR 94-02

DESCRIPTION OF CHANGE: The change consists of the temporary addition of lead shielding blankets to certain portions of the fuel pool cooling and cleanup (FPCCU) piping for the purpose of reducing personnel exposure during fire proofing removal to support tubing and hanger replacement per MNCR 90-0007 activities. The lines are to be shielded from the inside of the 90° elbow and extend up to the 185' penetrations AP-14E & AP-15E (Reference Isolation M-1351D). The added weight to each loop will be 45 lb/ft. The temporary lead shielding will remain in place only until July 30, 1994.

REASON FOR CHANGE: Temporary Shielding Request 94-02 estimated personnel exposure savings of 0.480 personrem by the use of temporary lead blankets.

SAFETY EVALUATION: Calculations MC-Q1G41-94018, Rev. 0 and CC-Q1G41-94011, Rev. 0 qualified the structural integrity of the piping and supports for the subject piping for Operating Modes 1 through 5. The structural integrity of the pipe and supports are confirmed by the results of these calculations being within code allowables.

No permanent plant changes will result due to the introduction of the shielding.

References

1. Response to Temporary Shielding Request 94-02.

2. Calculation No. MC-Q1G41-94018, Rev. 0, "Temporary Lead Shielding for TSR 94-02".

3. Calculation No. NPE-PDS-166, Rev. 7.

4. ASME Boiler and Pressure Vessel Code, Section III, "Nuclear Components," 1974 Edition with addenda through the Summer 75 Addenda.

5. TSR 94-02.

6. Calculation No. CC-Q1G41-94011, Rev. 0 "Support Load Evaluation for TSR 94-02.

Serial Number: 94-027-NPE Document Evaluated: DCP 91/0201R00, SCN 93/0059B

DESCRIPTION OF CHANGE: This change provides the design and installation enhancement of the 13.8 kV site power loop to include the capability of using the 34.5 kV/13.8 kV switchgear for a redundant AC power source. Bus 22R/29UD is the preferred voltage source for the site power loop. The design of the Site Power Loop System will conform to the IEEE, ANSI standards and codes referenced above, so that the design will be consistent with industry and engineering practices. Administrative procedures will be established so that the MP&L 115/13.8 kV source and Bus 29UD source will not energize the 13.8 kV site power loop at the same time.

The safety evaluation has been revised to reflect the removal of the explicit revision levels of referenced calculations.

REASON FOR CHANGE: The 13.8 kV site power loop was installed during the construction phase of the plant and has evolved as a power source for most structures external to the power block. The changes to the 13.8 kV power loop will help to make the existing power loop a more reliable power source and to provide a better means of switching substations on the AC power loop and connect the AC power loop to a redundant AC power source. The current 13.8 kV AC supply is provided from the 115 kV MP&L substation and is vulnerable to system disturbances and other external phenomena, hence the need for additional reliability.

SAFETY EVALUATION: The Electrical Calculation EC-N1R23-91042 "Power Systems Analysis for 13.8 kV site power loop" performed specific short circuit and voltage regulation studies to support modifications to the 13.8 kV site power loop. The calculation demonstrated that the addition of the 13.8 kV site power loop to the main power block electrical system at Bus 29UD will not adversely effect the existing electrical (ESF or BOP) systems while providing reliable service to existing 13.8 kV electrical loads with regard to voltage requirements and short circuit protection. Part of the methodology of the calculation demonstrated that the ESF divisional loads will not be adversely effected during a worst case scenario, LOCA condition with degraded 500kV grid voltage. Therefore there will be no impact to the safety related or associated systems at GGNS. This change will upgrade the site power loop to provide redundancy by establishing a new source of AC power by connecting to Bus 29UD, a more reliable power supply for most structures that are external to the power block. Administrative procedures will prevent the 115kV and 500kV AC sources from energizing the site power loop at the same time.

Serial Number: 94-028-PLS Document Evaluated: 04-1-01-E12-1 TCN 108
04-1-01-E21-1 TCN 16

DESCRIPTION OF CHANGE: This procedure change allows the use of the P11 system to backflush the RHR and LPCS suppression pool suction strainers in the event of clogging. The expected conditions precipitating the need for using this change is post-LOCA. "Normal" strainer clogging will be handled in accordance with the corrective maintenance procedures at GGNS. This procedure shall only be used during emergency conditions in conjunction with the emergency operating procedures (EOPs).

REASON FOR CHANGE: As part of GGNS's response to NRC Bulletin 93-02, Supplement 1 GGNS committed to evaluate and proceduralize a method to use existing system interconnections to facilitate backflushing of the ECCS suction strainers.

SAFETY EVALUATION: The purpose of the procedure change is to specify a method to backflush the RHR "A", "B", "C" and LPCS suppression pool suction strainers during emergency conditions. The main focus of this safety evaluation addresses defeating the interlock between the E12-F004A/B and the E12-F006A/B. This interlock, which must be defeated in order to successfully backflush the RHR "A" and "B" strainers, was designed to prevent inadvertent draining of the reactor during routine plant operations. The intent of this procedure change is to provide a means during an unlikely emergency to clean the strainers. Therefore, the need to maintain the interlock is not necessary as these subsequent actions would be considered outside of the design basis. This change is not to be construed as a permanent solution to the ECCS suction strainer clogging issue, rather it is an interim alternative means to mitigate the consequences of strainer blockage. All mechanical design aspects with regards to piping and component integrity and capability have been evaluated as being acceptable. The various cautions noted in the procedure change ensures no different accident conditions are created or worsened.

Serial Number: 94-029-NPE Document Evaluated: FSAR CR No. 94/0013

DESCRIPTION OF CHANGE: Underwriters Laboratories (UL) performs testing of many configurations of structural steel and sprayed on fireproofing to determine thicknesses of fireproofing required to provide certain fire ratings to the steel. UL assigns design numbers to each tested configuration. Currently, the UFSAR references certain UL design numbers that provide the design details to protect structural. The UL design numbers referenced in the UFSAR, N711, N712, and D717 are for spray on fireproofing material(s) which are no longer manufactured. Due to manufacturing of new material, new design numbers are also utilized that are not included in the UFSAR. Therefore, to alleviate changing the UFSAR for each time a new fireproofing material is changed, the requested change to the UFSAR will remove any specific reference to UL design numbers.

REASON FOR CHANGE: Due to specific UL design numbers being listed in Section 9.5.1.2.2.9 of the UFSAR, a change is required anytime a fireproofing material is replaced. Removal of the specific UL design numbers will alleviate the change to the UFSAR when new design numbers are added.

SAFETY EVALUATION: The requested change to the UFSAR will remove the specific UL design numbers presently in Section 9.5.1.2.2.9 of the UFSAR and allow for any configuration to be used which has been tested by a recognized independent testing laboratory. The qualification of tested configurations of spray on fireproofing is required to be performed by a recognized independent testing laboratory, such as Underwriters Laboratories. Additionally, the tested configuration shall ensure structural steel forming a part of or supporting a fire barrier (i.e., wall, floor, etc.) will provide the required fire resistance. Implementing this change will allow use of replacement materials for materials no longer manufactured without requiring a change to the UFSAR. The fire rating of the barrier and supporting structural steel will be maintained per the requirements of Appendix R to 10CFR50 and our commitment to Appendix A of APSCB-BTP-9.5-1.

Serial Number: 94-030-NPE   Document Evaluated: MCP 91/1142, Rev. 0 and CN 94/0020

DESCRIPTION OF CHANGE: This change is issued to delete the turbidity monitors for the Process Sampling System and remove or spare all equipment used for remote monitoring.

Components that will be removed under MCP 91/1142 are the following: Analyzer Elements; 1P33-AE-N037 & -AE-N007, Transmitter Switches; 1P33-ATS-K005 & -ATS-K002, Transmitters; 1P33-AIT-N038 & AIT-N006.

REASON FOR CHANGE: The condensate demineralizer combined effluent turbidity monitor (1P33-AIT-N038) and the forward pumped heater drain turbidity monitor (1P33-AIT-N006) are no longer used. The detector windows are prone to fouling and are not appropriate for the intended function of monitoring turbidity. Therefore, the monitor components should be removed or abandon in place as appropriate for the complete demolition of these monitors.

The original design intent of these monitors was to be able to measure the suspended concentration of corrosion and iron utilizing a conductivity monitor and correlate this to turbidity. However, the actual function is to measure the concentration for conductivity based on the size and shape of the suspended particles, which in practice does not correspond with percentage of iron/corrosion. Also, there is no effective correlation between turbidity and iron or corrosion concentration.

SAFETY EVALUATION: All of the equipment, instrumentation and tubing associated with the turbidity monitors of the Process Sampling System are non-safety related. The analyzer elements do not form a safety related pressure boundary of the discharge piping. These analyzer elements shall be disconnected and removed. There are currently no Class 1E electrical interfaces associated with the monitoring equipment. No electrical or mechanical safety-related interfaces shall be either created or deleted by implementation of this change.

Serial Number: 94-031-NSRA Document Evaluated: Interim Qualifications for 50.59 Evaluators

DESCRIPTION OF CHANGE: Site Directive G4.110, Safety and Environmental Review and Evaluation, is being revised to allow Grand Gulf personnel and contractors to be qualified on an interim basis to originate 50.59 screenings, reviews and evaluations, until the next 50.59 training class. This qualification will be based on the applicant's experience, training at other sites, completion of required reading, the recommendation of the applicant's supervisor, and successful completion of a 50.59 practical examination.

REASON FOR CHANGE: The demand for 50.59 qualification at Grand Gulf only requires that one training session per year be offered. Since personnel are hired throughout the year at GGNS, technical personnel are now forced to wait several months for training. This interim qualification will allow technical personnel to originate 50.59 documents until the next training session.

SAFETY EVALUATION: This interim qualification will only be granted to personnel who already have a firm understanding of the principles of 10CFR50.59 evaluations. It would constitute an unnecessary waste of recourses to prevent them from originating 50.59s for several months. The primary concern is that the evaluator be familiar with any site-specific equipment, conditions, procedures or policies. This will be ensured by allowing the person to originate the documents only, requiring someone with site-specific insight to act as the independent reviewer. This change therefore does not constitute an unreviewed safety question.

Serial Number: 94-032-NSRA Document Evaluated: AECM-85/0164

DESCRIPTION OF CHANGE: Rescind the commitment made to the NRC via AECM-85/0164 to submit updates to the Fire Hazards Analysis (FHA) consistent with the update requirements of 10CFR50.71(e).

REASON FOR CHANGE: Submittal of updates of the FHA to the NRC on a periodic bases results in an unnecessary burden on the Plant Staff.

SAFETY EVALUATION: The proposed change does not result in any changes to plant equipment, methods of using plant equipment, or affect the amount of combustibles in the plant. The revision of the commitment does not result in any change to the approved fire protection program which could adversely affect the ability to achieve and maintain safe shutdown in the event of a fire.

The submittal of updates to the FHA only provided a mechanism for NRC to review the appropriateness of licensee activities after-the-fact, but provided no regulatory authority once the report was submitted (i.e., no requirement for Commission approval). In addition to the submittal of updates to the FHA, a summary of changes made to the FHA is required to be provided to the NRC via the 10CFR50.59 summary report and changes in the UFSAR resulting from changes in the FHA are required to be submitted in accordance with 10CFR50.71(e). Given that the update to the FHA was allowed to be provided to the Commission up to 2 years after the change was made, it was clearly not necessary to assure operation of the facility in a safe manner for the interval between the revision of the FHA and submittal of the update. Additionally, given there is no requirement for the Commission to approve the change identified in the update, this update is not necessary to assure operation of the facility in a safe manner following its submittal to the Commission.

Serial Number: 94-033-NSRA   Document Evaluated: MAEC-90/0079

DESCRIPTION OF CHANGE: The proposed change will delete the verbal commitment to revise procedure 01-S-06-3 to require the General Manager's review and authorization for temporary alterations that are greater than one year old and will not be closed during the next refueling outage. The change was made to close an inspector follow-up item.

REASON FOR CHANGE: This proposal is being made to remove the responsibility of review and authorization of the above described temporary alterations from the General Manager and assign it to the temporary alteration review committee.

SAFETY EVALUATION: The deletion of this commitment does not constitute an unreviewed safety question. The change does not affect the operation of safety related system nor does it increase the probability or consequences of an accident. This change is administrative in nature and would not affect the health and safety of the public.

Serial Number: 94-034-NPE Document Evaluated: MNCR 308-92

DESCRIPTION OF CHANGE: The setpoint of E12-N652A, B, C is being changed from 1000 gpm to 1154 gpm.

REASON FOR CHANGE: The existing setpoint for E12-N652A, B, C does not account for instrument uncertainty as required in the current revision of the GE design specification data sheet.

SAFETY EVALUATION: The trip units E12-N652A, B, C monitor RHR flow via transmitters E12-N052A, B, C and flow elements E12-N014A, B, C. These trip units control the RHR minimum flow bypass valves E12-F064A, B, C. For RHR pump protection, the minimum flow valves will automatically open if the pumps are running and the main line flow is less than the low flow trip setpoint. The minimum flow valves will close at flows above the low flow trip reset. Flow modeling analysis based on MC-Q1E12-93008 has shown that the minimum flow restricting orifices limit bypass flow such that required injection flow will be available even if the valves fail to close. Raising the setpoint of the trip units to account for instrument uncertainty will therefore not adversely effect RHR system performance. Raising the setpoint will prevent RHR pump damage that may occur at discharge flows below the 1000 gpm process limit. If ADHRS is operating and the RHR C pump is started contrary to the requirements of 04-1-01-E12-1, the minimum flow valve E12-F064C may not open. This is of no consequence because the only purpose in opening is to protect the RHR C pump from damage due to low flow, and in this case a flow path for the C pump already exists (e.g., valve E12-F042C is open to the vessel for ADHRS flow). Also, the whole purpose of raising the setpoint is to increase the level of confidence that the pump is being protected from low flow conditions.

No interfaces are created by this design change. No new failure modes are introduced. The changes will not compromise any safety related system, structure or component. The failure of the trip units will not initiate any evaluated transient or accident. The E12 (RHR) system operation and function will not change. The trip units are not required to mitigate the consequences of any evaluated transient or accident. No new interfaces are created. This change will therefore not introduce an unreviewed safety question. The trip units are not currently addressed in the technical specification and this change will not require that they be added to the technical specification. The stroke time required to manually close the valves will not be affected. Therefore, the margin of safety as defined in the basis for any technical specification is not reduced.

Procedural controls are in place to limit vessel drain down to the suppression pool through the E12-F064A, B, C valves. Changing the minimum flow setpoint will have an insignificantly conservative effect on the vessel drain down event should it occur. This is because the amount of time required for operator intervention will be increased by an infinitesimally small amount during pump startup.

Serial Number: 94-036-NPE Document Evaluated: MNCR 94-0034, UFSAR CR NL 94-0008 and FHARR 94/0001

DESCRIPTION OF CHANGE: There are five columns within the concrete block wall between Division II and Division III switchgear rooms. MNCR 0034-94 identifies deficiencies with the fireproofing on the flanges of these columns. The MNCR "Accept-As-Is" these deficiencies.

REASON FOR CHANGE: The repair of these column flanges would be difficult and impracticable because safety related equipment would have to be temporarily relocated. To disconnect, move and then reconnect safety related equipment should only be done if absolutely necessary. FPE 94-0002 evaluated the deficient fireproofing and determined that the fireproofing on the column flanges is not required. Therefore, the FHA and UFSAR must be updated to reflect the acceptance of this deviation.

SAFETY EVALUATION: The concrete block wall between Division II and Division III switchgear rooms is listed as a 3-hour barrier on the fire protection plan drawings. FPE 94-0002 Rev. 0 was prepared to evaluate what affect the deficient fireproofing might have on maintaining acceptable separation between Fire Areas 36 and 38. It was determined that the wall between the Division II and Division III switchgear rooms does not separate redundant safe shutdown components and a fire in one room will not propagate to the other. Therefore, the need to provide fireproofing on these columns will not adversely affect the ability to achieve and maintain safe shutdown in the event of a fire.

Operating License Amendment No. 82 relocated the Fire Protection Technical Specifications to UFSAR Appendix 16A and Technical Requirements Manual (TRM). This change will not affect technical specifications or the bases for any technical specifications, UFSAR Appendix 16A or Technical Requirements Manual (TRM), because these documents do not contain specific design details for fire barriers. This change will not increase the probability for occurrence of accidents or a malfunction of equipment important to safety previously evaluated in the UFSAR because this change does not add ignition sources or make other changes which would increase the probability of occurrence of an accident (i.e., a fire) previously evaluated in the UFSAR. This change will not increase the consequences of an accident or malfunction of equipment important to safety from that previously evaluated in the UFSAR because the postulated fire will not spread or adversely affect equipment or components beyond that previously evaluated in the UFSAR. 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 UFSAR because a fire on either side of the barrier will be contained within that area and will not propagate to affect equipment outside the fire area of origin.

Serial Number: 94-037-NPE Document Evaluated: MC-Q1F15-89010, Rev. 3

DESCRIPTION OF CHANGE: Calculation MC-Q1F15-89010, Rev. 3 updates the GGNS Fuel Handling Accident analysis to consider a leakage path bypassing the SGTS filter trains consistent with the GGNS LOCA dose analysis in Calculation XC-Q1111-92010, Rev. 0. In addition, the calculation methodology is considerably simplified from that presented in Revision 2. This calculation confirms the continued applicability of the impact energy restrictions transmitted to the plant in GIN-94/00401.

REASON FOR CHANGE: The original GGNS Fuel Handling Accident analysis in UFSAR Sections 15.7.4 and 15.7.6 does not consider the effect of a direct leakage path from the Auxiliary Building to the environment due to bypass leakage around the SGTS filter trains. This revised calculation re-evaluates the fuel handling accident considering this additional release.

SAFETY EVALUATION: The consequences of the worst case fuel handling accident were calculated to be significantly less than the NRC acceptance criteria reported in SRP Section 15.7.4 and General Design Criterion 19.

Serial Number: 94-039-PLS Document Evaluated: 01-S-05-6

DESCRIPTION OF CHANGE: Commitment described in AECM-88/0024 was removed from RPTS in Revision 24 of 01-S-05-6 without explanation. In this AECM dated February 3, 1988, GGNS said it would improve legibility of drawings for NPE by revising 01-S-05-6 to require an aperture card file and reader with the Control Room stick file. On March 6, 1989, IPC-89/0634 advised Administrative Services to remove the aperture card file and reader because of the improved legibility of drawings. "They [drawings] are all now legible," states the IPC. "Maintaining the aperture card file in the Control Room now that the stick file drawings are legible is clearly unneeded."

REASON FOR CHANGE: The commitment in the AECM was taken out of the RPTS listing in Revision 24 of 01-S-05-6 because of the removal of the aperture card file based on the recommendation in IPC 89/0634. However, the technical reviewer did not review the commitment and the 50.59 applicability review for the procedure indicated that no safety evaluation was necessary.

SAFETY EVALUATION: The removal of the commitment in AECM-88/0024 from Revision 24 of 01-S-05-6 does not pose an unreviewed safety question, nor does it impact the safe operation of Grand Gulf Nuclear Station. The commitment was made to help improve legibility of Control Room stick file drawings, but within a year, NPE had solved its drawing legibility problems and the aperture card file was no longer necessary, as explained in IPC-89/0634. In addition, 01-S-05-6 contains a requirement that Document Control will annually audit all stick files to ensure legibility of drawings.

Serial Number: 94-040-NPE Document Evaluated: DCP 88/0052

DESCRIPTION OF CHANGE: This change renovates the 93' elevation of the Control Building (Fire Area 26) to provide office space for Health Physics personnel. This change supplement also provides the HVAC, seismic recorders, and fire protection modifications required to support this renovation.

Air handling unit NSZ17B003-N serves the area being reworked. Its supply and return ductwork are rerouted to support the new floor arrangement. The unit's airflow is slightly reduced, from 14,670 cfm to 12,305 cfm, to support the new heat loads. The makeup air supply to the Hot Machine Shop is unaffected. Due to the floor layout and its ventilation requirements, both makeup and exhaust airflows are reduced. Makeup airflow is being reduced from 11,360 cfm to 9,795 cfm. Exhaust airflow is changing form 11,430 cfm to 4,830 cfm. HVAC modifications are non-safety related changes to a non-safety related system.

The modifications to system NSP64D140 shall comply with NFPA 13, Standard for the Installation of Sprinkler Systems.

Peak Recording Accelerograph SC85-XR-R008 will be relocated by this supplement to preclude a trip hazard for personnel exiting the HP checkout in its present location.

REASON FOR CHANGE: This change modifies the 93' elevation of the Control Building to provide additional office space. This supplement to the change provides the HVAC modifications required, sprinkler systems modifications, seismic recording device relocation, furniture limitations, and penetration details.

SAFETY EVALUATION: The changes implemented in this DCP do not impact any technical specification controlled parameters, or reduce the margin of safety as defined in the basis of any technical specification. This design change does not increase the probability of occurrence of an accident or malfunction of any equipment important to safety previously evaluated in the UFSAR.

Moreover, this change will not result in increased consequences of an accident previously evaluated in the UFSAR, or increase the consequences of a malfunction of any equipment previously evaluated in the UFSAR that is important to safety.

Serial Number: 94-041-PLS   Document Evaluated: 08-S-03-14 Rev. 13

DESCRIPTION OF CHANGE: Revision 13 to Chemistry Procedure 08-S-03-14 replaces CALGON LCS-60 with BETZ Powerline 3203 on the list of chemicals used in P42 Component Cooling Water, P43 Turbine Building Cooling Water, P71 Plant Chilled Water, and P72 Drywell Chilled Water Systems.

REASON FCR CHANGE: The current closed loop corrosion inhibitor (CALGON LCS-60) performs exceptionally well in our closed loop systems, but contains boron in the form of borax. Borax functions as an excellent pH buffer in closed loop inhibitors. However, its use is undesirable in BWR plants because of the potential for boron intrusion to the reactor coolant via radwaste recovery and recycle.

Chemistry control has evaluated alternative inhibitors that do not contain boron, but still satisfactorily control pH and provide the same degree of inhibition we have enjoyed with CALGON LCS-60. The result of these evaluations demonstrates that the organic neutralizing amine Morpholine is a satisfactory choice for use in our closed loop corrosion inhibition program.

This safety evaluation discusses the use of Morpholine as a component of BETZ Powerline 3203.

SAFETY EVALUATION: The replacement of Borax with Morpholine is satisfactory for our closed loop applications. Onsite testing demonstrated that corrosion rates are at or below minimum detection limits for all base metallurgies in these systems. The incorporation of polymeric dispersant provides benefits not currently enjoyed with the LCS-60 product. Reference documentation shows that Morpholine will have no detrimental effects on the radwaste demineralizers or their ability to remove impurities from water.

Serial Number: 94-042-NPE   Document Evaluated: MCP 93/1004

DESCRIPTION OF CHANGE: Deletion of 115 kV `Source Lost' annunciator and computer point. The input from the offsite power system (via Mississippi Power & Light) is not available to the switchyard. Annunciator window in Control Room to be left blank and computer point spared.

REASON FOR CHANGE: The instrumentation required to provide a signal that the incoming lines are available was not installed in the switchyard. Indication is available in the Control Room via status lights for sources energized and meters for voltage readings.

SAFETY EVALUATION: The alarm status provided by an annunciator and computer point has no affect on the availability of 115 kV incoming power line. Indication exists in the Control Room for operators to determine the status of incoming lines for operation of equipment important to safety. UFSAR Section 8.2.4, Operating Limits, basis item 3 describes the monitoring of the 115 kV line integrity by MP&L dispatchers. Therefore, because the alarm is not required or assumed by any analysis, safety is not adversely affected by its deletion.

Serial Number: 94-043-NPE Document Evaluated: DCP 88/0172-02

DESCRIPTION OF CHANGE: This change addresses the installation of a new non-safety related computer system communications panel and four new non-safety related computer system data server panels, all in the Control Building computer room (OC403). This change also provides for deletion of existing non-safety related Computer Panels 1C91-P844, 1C91-P856 and SC91-P890 currently located in OC403; as well as the relocation of several computer system subcomponents presently installed in panel 1C91-P856.

REASON FOR CHANGE: The present plant computer systems have reached an age where they require ever increasing maintenance and improvements to support the needs of Plant Staff. The existing systems are overwhelmingly diverse, and vendor support is becoming less than adequate to maintain these systems.

SAFETY EVALUATION: Neither the existing computer equipment affected by this design, nor the new computer equipment to be installed by this design is required to mitigate the consequences of any accident or transient. The computer system components affected by this design are non-safety related, and implementation of this design will not compromise the operation of any existing safety related system, structure or component. No new interfaces with safety related equipment will be created by the proposed computer system modifications.

Based on the discussion, no portion of this change will increase the probability or consequences of accidents or malfunctions of equipment important to safety previously evaluated in the UFSAR. Further, 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 UFSAR.

Since neither cable combustibility nor the computer system is addressed by technical specifications or the Technical Requirements Manual, and since the operation of existing safety related systems, structures and components is unaffected by this design; this change will not reduce the margin of safety as defined in the basis for technical specifications.

Serial Number: 94-044-NPE Document Evaluated: MCP 93/1012 Rev. 0

DESCRIPTION OF CHANGE: This change provides for existing temperature monitors to be read from remote locations in order to reduce exposure and time required to record data. Remote temperature indicators and selector switches will be mounted for the following motors: RHR Pumps A, B, C, LPCS and HPCS.

REASON FOR CHANGE: To provide a better means for recording and trending of bearing and stator temperature, thereby, increasing reliability and motor service.

SAFETY EVALUATION: Review of this change has determined that the addition of the remote temperature monitoring devices for the HPCS, LPCS, and RHR-A, B, and C motors is an extension of existing equipment and will not alter the probability of occurrence or the consequences of occurrence of any malfunction of equipment important to safety. Nor will it create any new hazard which would be detrimental to the function of any equipment which has or has not been previously evaluated in the UFSAR. The function of the added monitors has been reviewed as indicated in the attached Failure Mode Effects Analysis (FMEA) and determination has been made that "The design of the digital thermometer system for the ECCS pump motors will not degrade the ability of the motors to perform under any conditions since all failure modes and their effects have been determined to be acceptable." The temperature monitoring equipment is not required to be operable for any safety function. It is an aid to monitor motor condition for preventative maintenance activities.

Serial Number: 94-045-PSE   Document Evaluated: Temp Alt 94-0008

DESCRIPTION OF CHANGE: Temporary Alteration 94-0008 disables the high generator bearing temperature alarm function on the HPCS diesel generator provided by temperature switch 1P81N052B. This switch provides a local alarm to alert operators that a high generator bearing temperature condition exists during diesel generator operation. This switch does not provide a protective trip function. The temperature switch is intended to provide the operator with sufficient information to take the necessary corrective action if a high temperature condition occurs. This switch also provides a trouble alarm in the Control Room to alert Control Room operators to the presence of a local alarm.

REASON FOR CHANGE: This temporary alteration is being installed to relieve operator burden. Due to recent failure of generator bearing temperature switch 1P81-N052B, the local engine control panel alarm and Control Room panel diesel trouble alarm have been sealed in the alarm state. The Control Room diesel trouble alarm annunciates when various local diesel alarms annunciate. With the Control Room trouble alarm sealed in, operators must tour the HPCS diesel room periodically to verify that no additional local alarms exist. Presently, the lead time for replacement parts is several weeks. This temporary alteration will silence the Control Room trouble alarm by bypassing the erroneous high generator bearing temperature alarm, and eliminate the need for periodic touring of the HPCS diesel room.

SAFETY EVALUATION: The safety evaluation concluded that disabling the high generator bearing temperature alarm is not an unreviewed safety question, does not reduce the margin of safety and does not require a change to the technical specifications. Disabling this alarm does not prevent the HPCS diesel generator from performing its intended design safety function.

Serial Number: 94-046-PSE Document Evaluated: WO #125781

DESCRIPTION OF CHANGE: Siemens Power Corporation (SPC) will be performing fuel bundle inspections on two SPC 9x9-5 lead fuel assemblies (LTA-901 and LTA-902) and five SPC 8x8 fuel assemblies (XNC-665, XNB-265, XNB-369, XNB-469, and XNC-605). Also they will examine the suspect failed fuel assembly (AND-043 and/or XNC-703) discharged after Cycle 6 operation. These inspections will be performed IAW MWO 125781. The SPC equipment will be mounted on the refueling floor curb alongside the fuel preparation machines. The SPC equipment includes a periscope, a fuel rod elevator, a rod cleaning system, a gamma scanner and an instrument test fixture to hold the necessary measuring devices. A camera will be placed on the spent fuel racks to provide a stable view of the bundle during visual inspection while lowering the bundle into the rack.

The failure analysis for the suspect failed fuel assembly will:

- find failed rods and inspect
- remove failed rods and store in spent fuel pool rack H1 if severely damaged
- attempt to determine the cause of the failure

After the inspections, the bundles will be placed in storage in the spent fuel pool. They will not be reloaded into the reactor core.

REASON FOR CHANGE: The 9x9-5 and 8x8 fuel assemblies are being inspected to gather data about the fuel's physical characteristics following operation.

The failure analysis is being performed to positively identify which bundle had failed fuel and to attempt to determine the cause of the failure.

SAFETY EVALUATION: The fuel examinations should provide useful information to SPC to be used in future designs of fuel bundles.

Most aspects of fuel inspection were evaluated in previous safety evaluations. Safety Evaluation 011-92 covered the temporary installation of the fuel inspection equipment, the reconstitution of failed bundles, and the removal of the fuel inspection equipment. Safety Evaluation 047-92 examined installation of the fuel inspection equipment:

The fuel bundle inspection will not increase accident or malfunction probabilities or consequences. It will not create any risk of a different type accident or malfunction and does not reduce the margin of safety in the technical specifications bases. There are no unreviewed safety questions.

Serial Number: 94-047-NPE Document Evaluated: (Cycle 7 Reload Analysis)

DESCRIPTION OF CHANGE: This safety evaluation assesses operation with the Cycle 7 core configuration. Attachment 1 provides a detailed description of the Cycle 7 core and the issues considered in this evaluation. Revision 1 to this safety evaluation was developed in response to QDR 0159-94 and includes the increased exposure limits required for operation to the end of Cycle 7.

REASON FOR CHANGE: This evaluation addresses the core changes associated with the Cycle 7 reload and operation.

SAFETY EVALUATION: This evaluation concludes that the core changes associated with the Cycle 7 reload and operation (i) will require no changes to the current GGNS Technical Specifications, and (ii) will not constitute an unresolved safety question.

Serial Number: 94-048-NPE   Document Evaluated: MNCR 93/0366 & Supp 1

DESCRIPTION OF CHANGE: The damaged sections of Culvert 16 are being replaced, but without the security grating indicated in UFSAR Table 2.4-28. Concrete pads blocking the closure of Turbine Building Door 1T301 are being removed. The floor slab under the affected section is being refinished. UFSAR Table 2.4-28 is being revised to delete the reference to security grating on Culvert 16.

REASON FOR CHANGE: Culvert 16 has been partially crushed due to the passage of heavy vehicular traffic. UFSAR Table 2.4-28 currently shows that security grating is required to block passage through Culvert 16. However, Culvert 16 is located in the Owner Controlled Area but lies entirely outside the Protected Area. Access to the Protected Area cannot be gained by crawling through the culvert. Therefore the grating is not required. Door 1T301 cannot effectively function as a fire door with concrete pads blocking closure. Additional leakage into the power block can result during heavy precipitation as a result of its current condition.

SAFETY EVALUATION: The deletion of the security grating for Culvert 16 will add conservatism to the probable maximum precipitation (PMP) drainage calculations contained in Calculation CC-Q1Y23-91050, Revision 0, Pages 29-32. As a result, the water levels recorded for Area J in UFSAR Section 2.4.3.5.3 will be somewhat overstated (conservative). Therefore, UFSAR Table 2.4-28 will be revised to delete the requirements for a security grating for Culvert 16. Door 1T301 is being returned to a condition which will allow it to serve its intended function. Door 1T301 is not among the doors governed by the Technical Requirements Manual (fire protection requirements are no longer contained in the technical specifications).

Serial Number: 94-049-PSE   Document Evaluated: MNCR 0117/94

DESCRIPTION OF CHANGE: The interim disposition of this change will install an in-line pipe cap downstream of valve B21-F137B to eliminate the seat leakage through feedwater piping drain valves B21-F136B and B21-F137B. The final disposition of this change will require a rework of the valve seats or valve replacement to restore the integrity of the valve seats.

REASON FOR CHANGE: Feedwater piping redundant drain valves B21-F136B and B21-F137B have developed seat leakage during routine plant operation.

SAFETY EVALUATION: Temporarily capping the drain line from the nuclear boiler system feedwater supply line will not alter the reactor coolant leakage requirements specified per GGNS Technical Specifications.

The design requirements for temporarily capping the drain line has been evaluated to be equivalent to the material specification, design, and fabrication of ASME Class 1 piping, and as such maintains the pressure boundary integrity of the feedwater piping. The final disposition of this change will restore the drain line to its original design configuration.

Serial Number: 94-050-NPE Document Evaluated: MNCR 0116-94

DESCRIPTION OF CHANGE: The outlet of the Condensate Cleanup System (N22) common vent post strainer ties into 3" HBD-761 and 3" HBD-1058 (Reference M-1064F). This change will provide a blind flange on 3" HBD-1058 which is a source of contamination due to the fact that it is open ended rather than tied to the Turbine Building Ventilation System. Any ventilation function performed by 3" HBD-1058 will now be performed solely by 3" HBD-761 which is routed to the Condensate Clean Waste Tank (CCWT). In turn, the CCWT is equipped with an 18" line to the Turbine Building Ventilation System.

Additional editorial changes will be made to UFSAR Figures 9.3-9 (Sheet 2) and 9.4-6. These changes are editorial only and do not reflect any physical modification to plant design.

REASON FOR CHANGE: The common vent post strainer outlet line denoted as 3" HBD-1058 is resulting in unnecessary contamination due to the fact that it is open ended rather than tied into the Turbine Building Ventilation System.

SAFETY EVALUATION: UFSAR Section 10.4.6.3 states that the Condensate Cleanup System (N22) provides no safety function. The piping to be modified by this change is located in the Turbine Building and is non-seismic, non-safety related. The N22 system analysis has shown that a failure of the system will not compromise any safety-related systems or prevent safe shutdown. According to UFSAR Section 10.4.6.2, the Condensate Cleanup System is provided with an automatic bypass to maintain condensate flow in the event a high differential pressure is realized across the Condensate Cleanup System. This design change does not affect any technical specification and does not involve an unreviewed safety question. Furthermore, installation of a blind flange on outlet piping of the common vent post strainers will not increase the consequences or the probability of occurrence of the loss of feedwater transient analyzed in UFSAR Section 15.2.7, or any other accident or transient analyzed in Chapter 15 of the UFSAR.

Serial Number: 94-051-NPE Document Evaluated: DCP 88/0040 Rev. 0

DESCRIPTION OF CHANGE: This change installs two current operated transistor switches in each of the 193 Hydraulic Control Unit (HCU) terminal boxes. In each terminal box, the power lead for both the "A" and "B" scram pilot solenoid valve will be replaced with a #16 AWG jumper of sufficient length to be routed through an integral current transformer (CT) window of one of the two switches [one current operated transistor switch will be utilized for the "A" coil and the other will be utilized for the "B" coil] and re-terminated to the original locations. The transistor switches will detect low or loss of current flow, via the current transformer, in the conductor monitored.

A stick-on symbol of the appropriate color will be installed on the RC&IS control rod display to designate which HCU locations are monitored. In addition, the mylar insert for the "SCRAM ACCUM TROUBLE" (1H13-P680-4A2-D4) will be replaced with a new insert which reads "HCU TROUBLE" and the labels for the Jay-El pushbuttons located on 1H13-P680-6C which presently read "ACCUM FAULT" and "ACKN ACCUM FAULT", a portion of 1C11-HSS-M610 and 1C11-HSS-M611, respectively, will be modified to read "HCU FAULT" and "ACKN HCU FAULT".

REASON FOR CHANGE: At present, there is no indication or alarm function to detect the loss of power to the scram pilot solenoids. In March, 1988, a loose connection in a daisy-chain of power to scram pilot solenoids in RPS Loop "A" resulted in a scram when coupled with a surveillance test on RPS Loop "B". This change will provide the ability to detect a loss of power to the daisy-chains. The eight HCUs selected and identified earlier will provide detection for each of the eight power feeds for the scram pilot solenoids.

SAFETY EVALUATION: The physical installation of the current operated transistor switches has been evaluated for Seismic II/I concerns and been determined to be acceptable due to the low mass of the switches. No electrical interface is required between the conductors providing power to the scram pilot solenoids and the new switches. The wire used for the jumper replacements required for the scram pilot solenoids has adequate current carrying capacity and is Class 1E qualified. The RC&IS circuitry and transponder cards, along with the pushbuttons and annunciator windows modified, are non-safety related. The replacement resistors are of similar composition and wattage rating as the original resistors and will be installed on the transponder cards in the same fashion as the existing resistors.

Since there is no interface between the scram pilot solenoid power circuits and the current-operated transistor switches and there are no Seismic II/I concerns, there will be no adverse impact on this safety system.

Serial Number: 94-053-NPE Document Evaluated: CN 93/0388

DESCRIPTION OF CHANGE: Pressure indicating switches SG17-N230, N233 and N255 provide pump shutoff on high pump discharge pressure for the waste surge, equipment drain collection, and floor drain collector pumps. CN 93/0388 to DCP 84/4065 will electrically disconnect these switches. Therefore, these instruments will no longer provide pump protection functions. These instruments however, will provide local indication of pump discharge pressure.

REASON FOR CHANGE: Normal operating pressure for the systems is close to the dead head pressure of the pumps. Discharge pressure switches cannot be set to provide pump protection without causing unnecessary pump trips. Therefore, the pump protection function of these switches will be removed in lieu of redesigning the system process in order to lower pump discharge pressure under normal operation.

SAFETY EVALUATION: This change affects the Radwaste System only. The Radwaste System and the affected pumps and instrumentation are not safety related. Table 3.2-1 specifies the pumps, piping and valves as safety class "Other", Q-List "N" and Seismic Category "N/A". The instrumentation affected by this change are not safety related, not connected to Class 1E power and are listed as "N" in JS-08, Revision 2. Removal of this pump protection function will allow the pumps to operate at elevated pressures without shutting off the pump, however, this will not create any new hazards. The design rating for the piping system is 240 psig per MS-02. The pumps affected by this change are centrifugal pumps which take suction from a vented tank, therefore the maximum discharge pressure is limited to the dead head pressure of the pumps plus the height of the storage tank overflow above the pump. Based on the pump curves and Level Setting Diagram J-0612, the maximum discharge pressure of the pumps is approximately 205 psig at the pump discharge. This is below the design rating for the system. Therefore the removal of this function will not create any new hazards due to elevated pump discharge pressures.

A failure of the pumps due to this protection function being eliminated will not impair the system or plant operation. "The Liquid Radwaste System is designed so that a failure or maintenance of any frequently used system or component will not impair the system or plant operation" (Reference UFSAR 11.2.1.1). The tanks from which these pumps take suction are cross-connected with the other pumps in the system such that a failed pump can be isolated and the waste processed using an alternate pump. Therefore removal of this pump protection function will not impair the system or plant operation.

Serial Number: 94-054-NPE Document Evaluated: MNCR 0127/94

DESCRIPTION OF CHANGE: This change documents generator rotor cooling water leakage that has caused a mass imbalance in the generator. The mass imbalance will be temporarily corrected by drilling a hole in the cooling water piping in the rotor casing 180° from the leaking pipe. The leaking pipe will also be drilled to ensure equal water transfer from both pipes. An equal mass of leaking water will be captured in the voids created by the annulus around both pipes. The two radially opposed and equal water masses trapped inside the rotor casing will restore the rotor balance. The final resolution of the leaking pipe will be to replace the damaged components by reworking the machine to the existing design requirements during the next refueling outage.

REASON FOR CHANGE: To allow the generator to be dynamically balanced for operation at rated speed. Generator imbalance was caused by cooling water leakage trapped inside the rotor casing.

SAFETY EVALUATION: The modifications to the generator do not alter the results of the missile analysis originally completed for the GGNS turbine/generator since the generator retaining ring burst were determined to be contained by the generator casing. The modification does not require a change to the technical specifications or to the UFSAR. The modification will not increase the probability of occurrence or increase the consequences of an accident previously evaluated in the UFSAR since the modification introduces no new failure mechanisms for the generator. The generator is not safety related and is not seismically qualified.

Serial Number: 94-055-NPE Document Evaluated: NPE Calc XC-Q1J11-94003 Revision 0

DESCRIPTION OF CHANGE: Calculation XC-Q1J11-94003, Rev. 0 (Reference 1) was performed to update the radiological dose analysis for a postulated Control Rod Drop Accident (CRDA) at GGNS. This calculation supersedes Bechtel Calculation No. 5.3.32-N, Rev. 1 (Reference 3) and provides bounding dose consequences for a worst case CRDA event. The results of separate, cycle-specific CRDA fuel failure analyses performed for each reload may then be compared to the inputs to Calculation XC-Q1J11-94003, Rev. 0 to determine if predicted doses remain bounding.

REASON FOR CHANGE: The calculation was performed in order to ensure that the latest methods and assumptions applicable for GGNS radiological dose analyses are applied to the CRDA event. It was also done so that the bases for all calculational inputs to the CRDA dose analysis will be clearly documented and verified to be adequately conservative. The calculation may also be used in partial support of a planned removal from the technical specifications of the main steam isolation valve closure function of the main steam line radiation monitors.

SAFETY EVALUATION: The results of Calculation XC-Q1J11-94003, Rev. 0 change the doses predicted for a CRDA event at GGNS. These results are shown in Table 1 below:

Table 1
CRDA Dose Analyses

Case	Site Boundary (rem) 2 Hours		Low Population Zone (LPZ) (rem) 24 Hours		Control Room (rem) 24 Hours	
	Whole Body	Thyroid	Whole Body	Thyroid	Whole Body	Thyroid
Limits[1]	6	75	6	75	5	30
New Results	0.42	3.52	0.18	4.16	0.007	0.906
Previous Results	0.14	1.32	0.09	1.53	NC[2]	NC[2]

[1] Limits are "well within" (25%) of 10CFR100 limits as defined in SRP 15.4.9 for Site Boundary and LPZ.
Limits for Control Room are GDC 19 values.
[2] NC = Not calculated previously.

The predicted doses remain well under the regulatory limits and do not necessitate any modifications to plant systems, components, or procedures. Only the calculation of the dose is changed. No change to the technical specifications is required, and the systems and procedures used to prevent and mitigate a CRDA are not affected. The results of this calculation thus have no effect on the probability of accidents or equipment malfunctions previously examined in the UFSAR, and they also do not create the possibility of a new type of accident or malfunction not previously considered. While the doses from a postulated CRDA event are slightly higher than previously calculated, they remain well below 10CFR100 and GDC 19 requirements so that the consequences of the CRDA are not increased. The margin of safety is also unaffected since the acceptance limits established by the NRC have not been changed and doses remain well below these limits. Thus, there is no unreviewed safety question associated with Calculation XC-Q1J11-94003, Rev. 0.

As stated in Calculation XC-Q1J11-94003, Rev. 0, the analysis takes no credit for closure of the main steam isolation valves following a CRDA. It should also be noted that, while not explicitly stated in the calculation, credit is likewise not taken for closure of the valves which isolate the main steam line drains from the condenser. These valves (B21-F067A-D, B21-F016, and B21-F019) are not assumed to provide any fission product hold up after the CRDA event. The amount of noble gas and iodine reaching the condenser in the analysis is the same regardless of the path taken.

Serial Number: 94-056-NPE   Document Evaluated: CN 94/0032 to DCP 89/0034-01 Rev. 0

DESCRIPTION OF CHANGE: The Security System Boundary Upgrade project was initiated as part of a multi-phase project to improve the existing Security System capabilities and provide for an improved protected area boundary configuration to improve surveillance and assessment.

This change provides the design and installation of additional intrusion detection monitoring equipment in CAS, SAS, TAS (central alarm, secondary alarm and temporary alarm stations respectively) and the Control Room. Additionally, a new panel was installed in the Control Building, Area 25A, Elevation 148'-0" to route video information to all alarm stations.

REASON FOR CHANGE: These changes were initiated as part of the multi-phase project to improve the existing Security System capabilities. The changes provide improved protected area boundary monitoring for optimum surveillance and assessment capabilities.

SAFETY EVALUATION: The changes in this evaluation do not compromise any existing safety related system, structure or component, nor will they prevent safe reactor shutdown. The security equipment is part of the Security Monitoring System (C83) which is non-safety related and whose function will not change due to the equipment and monitoring changes made by this evaluation.

All cable changes to be performed will be in accordance with the separation requirements of Regulatory Guide 1.75. No seismic changes or penetration changes are addressed in this evaluation.

No evaluated accident is affected by any change to the Security Monitoring System. The components of the Security System to be changed by this evaluation 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 analysis are introduced. Therefore, these changes will not introduce an unreviewed safety question. The Security Monitoring System (C83) is not presently addressed in the technical specifications and the changes to this system per this evaluation will not require any changes or additions to the Grand Gulf Technical Specifications.

Serial Number: 94-057-NPE   Document Evaluated: MNCR 0144-93

DESCRIPTION OF CHANGE: UFSAR Appendix 9D provides compliance with NUREG-0612 "Control of Heavy Loads at Nuclear Power Plants" by satisfying the guidelines of ANSI N14.6-1978. The acceptance criteria for periodic nondestructive testing of special lifting devices as specified in ANSI N14.6-1978 contain fabrication and material anomalies which do not occur in service. Revising the acceptance criteria provides more appropriate criteria for detection of service-induced damage or deformation. The edition of ASME V specified in Appendix 9D does not address the performance of magnetic particle testing through coatings. The revision will allow the use of a later edition of ASME V which permits this improved technique.

REASON FOR CHANGE: These changes in the UFSAR will clarify the acceptance criteria for magnetic particle testing to more appropriately reflect the defects which occur due to inservice degradation and to implement the advances made in NDE technology so that magnetic particle testing may be more cost effective and consistent with ALARA considerations.

SAFETY EVALUATION: The changes being made to UFSAR Appendix 9D by UFSAR Change Request No. 94-009 still implement the intent of NUREG-0612. The clarification of acceptance criteria for magnetic particle testing and the implementation of improved NDE techniques provides full compliance with this commitment.

Serial Number: 94-058-PSE Document Evaluated: WO #128774

DESCRIPTION OF CHANGE: The primary water pump, high vibration generator trip will be bypassed and temporary non-intrusive recording instrumentation will be connected to monitor input points to this trip.

REASON FOR CHANGE: A signal that provides an input to a generator trip on primary water pump high vibration has become sporadic and is not representative of the vibration. This has been confirmed by monitoring the other channel and the actual vibration signals. In order to troubleshoot the problem it will require that this trip be disabled and a temporary recorder to monitor the signal be installed at cabinet JC16.

SAFETY EVALUATION: A failure in the primary water system section of the EGP has made it necessary to bypass the generator trip associated with high vibration in the primary water pump. The Electronic Generator Protection (EGP) System is used to protect the generator. When a condition exists that threatens the safe operation of the generator or its support systems the EGP takes the generator off line which will trip the turbine. The wiring modification to be performed will disable the trip associated with high vibration in the primary water pump. The alarm function and indication from the same instrumentation however will not be disabled. The generator will be protected by operations personnel continuously monitoring the primary water pump vibration parameters while the primary water pump trip is disabled. The alarm is 1N43L616 and the associated computer point N43K239. The alarm setpoint is 10 mils. If the alarm setpoint is reached the local primary water pump vibration will be monitored on the Siemens vibration recorder 1N32-R651 on 1H13-P822 and actions required by the ARI should be taken. If the primary water vibration monitored on 1N32R651 exceeds the trip setpoirt specified in the ARI for 1N43L616 the unit must be tripped manually.

Serial Number: 94-059-NPE Document Evaluated: MCP 94/1026

DESCRIPTION OF CHANGE: The addition of quick disconnect sample connections and process isolation valves will be installed to the existing tubing, located at the rear of the Turbine Sample Analyzing Panel 1H22-P121. The connections are installed up stream of the installed metal filters for the demineralizer effluent and condensate pump discharge systems.

REASON FOR CHANGE: To provide accurate process sampling results for demineralizer effluent and condensate pump discharge for the insolubles (iron/resin) concentrations. Grab samples are currently being worked under a work order that facilitates sampling by breaking into the existing tubing connections to take sample collections. The effort is time consuming and requires a separate work order for each sample line.

SAFETY EVALUATION: Revising the existing tubing by adding quick disconnect sample connections does not result in any operational or functional change to the P33 Process Sampling System. This change is required to allow for improved ability to evaluate demineralizer performance through improved insolubles sampling (iron and resin fines). No additional tubing supports will be required. The process isolation valves and sample connections have been designed in accordance with ANSI B31.1 code requirements. Therefore, this change will not require a change to the technical specifications and will not create an unreviewed safety question.

Serial Number: 94-060-NPE Document Evaluated: MCP 91/1144 Rev. 0

DESCRIPTION OF CHANGE: This change is issued to delete the circulating water and circulating water makeup conductivity monitors (1N71-CITS-150A/B and 1N71-CITS-N140, respectively). In addition to the monitors, this change will remove or spare in place associated conductivity elements, computer points, instrumentation, controllers, recorders, tubing, cabling and raceway.

REASON FOR CHANGE: The circulating water and circulating water makeup conductivity monitors are no longer used for chemical control and do not provide any useful data on circulating water chemistry. Blowdown and cycles of concentration are determined and controlled by lab analyses of calcium concentrations. These monitors and associated conductivity cells are prone to malfunction and are obsolete.

SAFETY EVALUATION: All of the equipment, instrumentation and tubing associated with the circulating water and circulating water makeup systems are non-safety related. In addition, the conductivity elements (CE-N050A/B and CE-N040) do not provide a safety related pressure boundary for the circulating water piping. The conductivity elements shall be removed and their associated instrument tubing removed and capped. The only Class 1E interface is with safety related panel 1H13-P870-5B/5C. This change will delete a recorder and Bailey controller in this panel. No new electrical or mechanical safety related interfaces shall be either created or deleted by implementation of this change.

Bailey controllers N71-FK-R604A and R604B are no longer used in the AUTOMATIC mode for control of blowdown discharge valves LV-F513A and F513B. These valves are controlled in MANUAL only. Therefore, the controller outputs from the conductivity monitors to these controllers are no longer required and may also be removed per this change.

Revision of UFSAR Section 10.4.5.2 will indicate the MANUAL control of circulating water blowdown flow and Figure 10.4-5 will be revised to reflect deletion of instruments/interfaces from the system.

Serial Number: 94-061-NPE   Document Evaluated: MCP 93/1052, Rev.0
SCN 94/0031 to JS-08

DESCRIPTION OF CHANGE: MNCR 93/0104 documented a nonconformance between design documents and the as-built field condition of the nitrogen pressure supply for the R60 containment penetrations. This change corrects the drawings (including UFSAR figure) and removes abandoned equipment from the field.

REASON FOR CHANGE: MNCR 93/0104 documented a nonconformance between design documents and the as-built field condition of the nitrogen pressure supply for the R60 containment penetrations.

SAFETY EVALUATION: The nitrogen pressure supply for the containment electrical penetrations is not safety related, and is only used for leak testing for the penetrations. The changes being made to document the field condition on the P&ID and UFSAR figure do not represent a change to the function or system operation of R60. The nitrogen supply stations are located in the turbine building and can not interfere with the safe shutdown of the plant in any way.

Serial Number: 94-062-NPE Document Evaluated: DCP 88/0172-04, R0, JS-08 SCN 94/0024 Rev. A

DESCRIPTION OF CHANGE: DCP 88/0172, Supplement 4, is one of a series of modifications that upgrade the non-safety related Plant Data System (PDS) in order to provide a uniform plant computer system that has sufficient capacity, capabilities, and features to support Operations, Engineering, and Maintenance users.

The scope of this supplement includes removing computer equipment, replacing existing computer equipment, installing new computer equipment, relocating the Control Room security console, and modifying the associated cabling and conduits. Much of the equipment installed outside panel 1H13-P680 will be installed as "office equipment" without plant equipment ID numbers. All of the computer equipment is in Systems C91 and C93.

Three (3) Control Room envelope penetrations will be temporarily breached to allow removal and replacement of various cables.

REASON FOR CHANGE: The present plant computer systems have reached an age where they require ever increasing maintenance and improvements to support the needs of the Plant Staff. The existing systems are overwhelmingly diverse, and vendor support is becoming less than adequate to maintain these systems.

SAFETY EVALUATION: The functions performed by both the existing and new computer equipment affected by this design change are not required to mitigate the consequences of any accident or transient. Therefore, the components associated with this design change are classified as non-safety related. The BOP Computer System is not addressed in the technical specifications and implementation of this design change will not necessitate any technical specification changes.

Although none of the components are required to perform any safety related functions, some of the replacement components are physically located in the vicinity of existing safety related components. In each of these situations the new equipment has been designed and analyzed to satisfy Seismic 2-over-1 concerns and installed in accordance with Regulatory Guide 1.75 separation requirements. No new interfaces with existing safety related systems, structures, or components will be created by this modification.

No portion of this design change will increase the probability or consequences of accidents or malfunctions of equipment important to safety as previously analyzed in the UFSAR. Furthermore, this change will not create the possibility for an accident or malfunction of a different type than any previously evaluated in the UFSAR.

Serial Number: 94-063-NPE   Document Evaluated: EERR 92/6211
(SCN 94/0045A to JS-08)

DESCRIPTION OF CHANGE: This change requests that the CO2 Storage Unit N1P64A003 setpoint for NSP64N002A be at 74%. This switch isolates valve NSP64F452 to assure the technical specification requirement that 60% remains in the vessel for fire fighting when purging the generator. NPE's response assures that the 74% setpoint is conservative. Field walk downs, with vendor concurrence, found that the level indication gauge N1P64R017 is part of the assembly switch NSP64N002A. Therefore, the indicator is removed from the P&ID as a separate component. The P&ID change requires a UFSAR figure change. No procedures, tests or experiments are involved. The UFSAR descriptions are also not affected.

REASON FOR CHANGE: To as-built Figure 9.5-5 to match the DRN to P&ID 9645-M-0035E.

SAFETY EVALUATION: Removing the level indicator as a separate component from the switch does not affect safety or environmental evaluations. Therefore, the change does not involve any unreviewed safety questions.

Serial Number: 94-064-NPE Document Evaluated: EER 94-6182

DESCRIPTION OF CHANGE: The RCIC steam supply is delivered via a 10" steam line with two normally-open ac-motor operated containment isolation valves, E51-F063 (inboard, Division B) and E51-F064 (outboard, Division A). A 1-inch bypass line around the F063 is sealed with a normally closed ac-motor operated valve, E51-F076 (Division B). This safety evaluation assesses the consequences of temporarily backseating the E51-F076 valve during Cycle 7 power operation. Although the F076 valve receives the same Group 4 isolation signal as the F063 and F064 valves, the F076 stroke time will be longer than that of the other valves, thereby serving to increase the isolation time of this penetration. Since the longest possible duration of this change is for the remainder of Cycle 7, the necessary UFSAR, engineering standard, SOI, and TRM changes have not been implemented due to the temporary nature of this change. It is anticipated that the F076 valve will be repaired and operated in the normal (closed) position by the startup of Cycle 8.

REASON FOR CHANGE: Drywell instrumentation indicates that the E51-F076 valve is leaking into the drywell at an unacceptably high rate. Backseating this valve is expected to result in a reduced stem leakage rate.

SAFETY EVALUATION: This evaluation concludes that, in the event of a RCIC steam line break outside containment, the increased closure time for this penetration will result in a more severe temperature transient in the RCIC pump room (1A104) and the adjoining piping penetration room (1A204). NPE has reviewed the equipment in the affected rooms and determined that all necessary equipment will remain operable. The containment isolation function of this penetration is still maintained within the appropriate regulatory limits and the offsite dose and reactor inventory losses remain bounded by those calculated for the main steam line break. The new position of this valve will have no effect on the operation of RCIC.

Serial Number: 94-065-NPE   Document Evaluated: MCP 92/1070 Rev. 1

DESCRIPTION OF CHANGE: Main Steam Isolation Valve Leakage Control System (MSIV-LCS) motor operated valves (MOV) Q1E32F007 and Q1E32F009 are being modified by changing the existing Limitorque actuator overall gear ratio from 11.11:1 to 22.04:1; replacing the spring packs; and replacing the motor pinion keys. This change supersedes MCP 92/1070, Revision 0, which was issued to replace the 1800 rpm motors on the actuators for the subject valves with 900 rpm motors.

REASON FOR CHANGE: MOV Q1E32F007 is included in the scope of the valve testing and torque switch setting program established in response to NRC Generic Letter 89-10. In the process of setting the torque switches for MOV Q1E32F007, the measured total stem thrust exceeded the maximum allowable stem thrust (MAST) for the valve limiting component at the setting required to achieve the minimum required stem thrust. The excessive thrust was determined to be due to overtravel of the actuator after torque switch trip. This change is being issued to reduce the actuator output speed and thus the valve stem speed, in order to reduce the overtravel after torque switch trip.

SAFETY EVALUATION: Implementation of this change does not adversely affect the function, operation or operability of the outboard subsystem bleed and depressurization MOVs and, therefore, do not increase the probability of occurrence or increase the consequences of an accident previously evaluated in the UFSAR. The modification increases the reliability of MOVs Q1E32F007 and Q1E32F009 by permitting them to operate with stem thrust values that are within the maximum allowables for the valve components as defined in Standard GGNS-MS-25.0, Revision 8, and within the limits of the degraded voltage actuator capability torque, therefore the modification does not increase the probability of occurrence or the consequences of a malfunction of equipment important to safety previously evaluated in the UFSAR. The installation of new motor gear sets, replacement of the spring packs and replacement of the motor pinion keys on the two MSIV-LCS outboard subsystem valves was evaluated and it was determined that the increase in valve operating time was within the requirements indicated in UFSAR 6.7.1.3.1(b), that the actuators continue to meet all of the environmental qualification requirements for use outside of containment, that the seismic qualification of the valves is not affected, that the newly configured actuators have sufficient capability to operate the valves under normal as well as design basis degraded voltage conditions and that no new failure modes are introduced. Therefore, the modification 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 UFSAR. Implementation of this change will not require a change to the GGNS Unit 1 Technical Specifications or reduce the margin of safety as defined in the basis for any technical specification.

Serial Number: 94-066-NPE Document Evaluated: MCP 92/1009 Rev. 1

DESCRIPTION OF CHANGE: This change replaces the Stellite hardfacing on the reactor recirculation discharge gate valve disks with an alternate hardfacing material - NOREM Type 1. Revision 0 of this change modified the valve internals by: 1) Addition of disc retainers on the upper wedge and machining of corresponding slots in the backside of the valve discs, 2) Addition of Belleville washers between the disc trunnions (inside the hole in the upper wedge) to maintain disc separation (to prevent banging of the discs), and 3) To add a spring between the upper and lower wedge to prevent similar banging of the wedges. The Revision 0 changes do not affect the safety functions of the valves and the specific design of the recirculation block valves is not discussed in either the technical specifications or the UFSAR. The general aspects of the design of these valves which are discussed in the technical specifications and UFSAR (i.e., ASME code requirements) were not impacted by the changes made in Revision 0 of this change. However, in the case of Revision 1 the Stellite hardfacing is specifically discussed in the UFSAR.

REASON FOR CHANGE: The present design of the recirculation discharge gate valves allows flow induced turning of the valve disks which causes trunnion wear and resultant valve failure due to the disks falling off of the valve wedges. Also, the cobalt content and associated high radiation dose rate of Stellite makes it an undesirable hardfacing material. Revision 0 of this change modified the reactor recirculation gate valve internals but did not change the hardfacing material. Revision 1 changes the hardfacing material on the valve disks and wedges to NOREM Type 1 which is a non-Cobalt based hardfacing material.

SAFETY EVALUATION: Replacement of the original recirculation discharge gate valve internals having Stellite hardfacing with modified disks and wedges having NOREM hardfacing will have no impact on plant safety. The new parts will continue to perform the same safety functions as the original parts but will have greater reliability because a possible failure mode will have been eliminated. The alternate hardfacing material (NOREM Type 1) has been evaluated for adhesive wear and galling resistance, erosion/corrosion and corrosion resistance, coefficient of friction, impact resistance, weldability, hardness, and chemical compatibility with reactor components, and was found to be suitable as a replacement hardfacing material in the recirculation discharge gate valves. Use of NOREM Type 1 hardfacing will in fact be an improvement due to resultant dose reduction.

Serial Number: 94-067-NPE Document Evaluated: DCP 93/0026-2

DESCRIPTION OF CHANGE: This change removes the unused resin regeneration equipment in the Condensate Cleanup System (N22) located on the 93'-0" elevation in the Turbine Building to make room for the future installation of an Advanced Resin Cleaning System. Existing interfaces to the Liquid Radwaste (G17), Condensate and Refueling Water Storage and Transfer System (P11), Makeup Water Treatment (P21), Floor and Equipment Drains (P45), Instrument Air (P53), and the 480 Vac Electrical System (R20) will be eliminated and two H22 panels will be removed.

REASON FOR CHANGE: To provide space for the Advanced Resin Cleaning System (ARCS) installed by Supplement 3 to this change.

SAFETY EVALUATION: The Condensate System and affected portions of interfacing systems provide no safety function. The only portions of interfacing systems considered safety related are those portions forming part of containment boundary which are unaffected by this change. UFSAR Section 3.2 classifies the equipment affected by this modification as non-Q and non-seismic. Eliminating the capability to regenerate condensate demineralizer resin will not increase the consequences or the probability of occurrence of any accident or transient analyzed in Chapter 15 of the UFSAR, 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 removal of this unused equipment from the plant does not affect the operation of any safety system and does not constitute an unreviewed safety question or reduction in the margin of safety.

Serial Number: 94-068-NPE Document Evaluated: MNCR 0166-94

DESCRIPTION OF CHANGE: This change was written as a result of the scram that occurred on 11/01/94.

This safety evaluation is performed in support of the first interim repair disposition of MNCR 0116-94 which re-evaluated the ground detection circuit provided by MCP 89/1108 for Bus 11DC.

This first interim disposition will eliminate the 1KΩ low impedance ground detection circuit (alarm function) from Bus 11DC. The 40KΩ high impedance ground detection circuit will not be eliminated.

REASON FOR CHANGE: Review of Bus 11DC loads identified that a postulated bolted ground fault (single active failure) on the HPCS initiation logic relays will cause an inadvertent HPCS initiation. The 1KΩ ground detection circuit installed per MCP 89/1108 will not limit a bolted ground fault current preventing this initiation. However, it should be noted that the original ground detection circuit provided by GE would also support an inadvertent HPCS initiation with a postulated bolted fault ground fault present.

The 40KΩ ground detection circuit will not conduct sufficient current to initiate an inadvertent HPCS initiation with a postulated bolted ground fault. Therefore, this circuit will not be eliminated.

SAFETY EVALUATION: Elimination of Bus 11DC's ground detection circuit (1KΩ circuit) that provides control room annunciation will:

1. Reduce the probability of occurrence of an inadvertent HPCS initiation with a postulated bolted ground fault on Bus 11DC. The 40KΩ ground detection circuit will continue to insure that reliability of Bus 11DC is not degraded by providing a method of identifying developing ground faults.

2. Not increase the consequences of an accident previously evaluated in the UFSAR because no analyzed safety function will be affected by the changes contained in this interim disposition of MNCR 0166-94.

3. Not increase the probability of occurrence of a malfunction of equipment important to safety because the existing 40KΩ ground detection circuit will continue to insure that reliability of Bus 11DC is not degraded by providing a method of identifying developing ground faults.

4. Not increase the consequences of a malfunction of equipment important to safety previously evaluated in the UFSAR because existing analyzed equipment failures remain bounding. Interim disposition of MNCR 0166-94 will not introduce any new equipment failure modes that affect equipment safety function performance.

5. Not create the possibility of an accident of a different type than any previously evaluated in the UFSAR because detection of ground faults on Bus 11DC will be maintained and the circuit to be eliminated by this interim disposition only performed an alarming function in the event of a low impedance ground fault on Bus 11DC.

6. Not increase the possibility for a malfunction of equipment important to safety of a different type than any previously evaluated in the UFSAR because interim one disposition of MNCR 0166-94 eliminates the control room annunciation of a ground fault on Bus 11DC but does not eliminate the capability of identifying ground faults on the bus. Therefore, the reliability of Bus 11DC will not be degraded by performance of this MNCR disposition.

Serial Number: 94-069-NPE   Document Evaluated: MNCR 94/0060

DESCRIPTION OF CHANGE: Change system drawings to delete the Condensate Cleanup System Precoat Tank (1N22A007) Low Level Alarm (1N22-LAL-L063).

REASON FOR CHANGE: Plant Staff initiated this change because the system drawings show the alarm as active and the annunciator is no longer in service. The precoat material has been changed from Socafloc to Ecodex and the resin feed tank is currently used for mixing instead of the precoat tank. The precoat tank is no longer used and is permanently valved out of the system. The operating procedure SOI-04-1-01-N22-1 doesn't refer to it and it remains empty.

SAFETY EVALUATION: This change is a figure change only to the system P&ID to reflect actual plant configuration. The precoat tank does not serve any purpose necessary to plant operations, and would not be required to perform any safety related function if it were in service. There are no physical changes from current plant configuration created by this change. There are no technical specification changes required and no surveillances or LCOs are affected. Implementation of this change will not increase the probability of occurrence or consequences of any accident previously evaluated in the UFSAR, or assumptions previously made regarding system performance during normal or accident conditions. There are no new failure modes introduced and no unresolved safety questions resulting from this design change.

Serial Number: 94-070-NPE Document Evaluated: MNCR 94-0167

DESCRIPTION OF CHANGE: The objective of this change is to eliminate steam leakage at the T-junction of 1-1/2" HBD-90 and 1" HBD-1479. The use of Furmanite in the Offgas System is acceptable to accomplish this objective.

The subject change allows the potential to completely eliminate the flowpath of the "B" train offgas preheater (1N64B001B) steam drain line by injecting Furmanite as required to prevent steam leakage past closed valves 1N64F011B and 1N64F576B. This change is effectively the same as capping the "B" preheater, "B" side drain line.

If the above repair method does not eliminate steam leakage at the T-junction of 1-1/2" HBD-90 and 1" HBD-1479, line 1-1/2" HBD-1479 will be crimped in two locations downstream of the tee with Furmanite injected between the crimps. This will eliminate the potential for steam leakage due to backpressure and is effectively the same as capping the "B" preheater, "B" side drain line.

There is a potential that neither of the above repairs will be required if valve manipulation alone eliminates the steam leakage. If so, the only repair will be installation of a suitable pipe clamp to prevent inleakage to the system.

REASON FOR CHANGE: The reason for this change is the T-junction of 1-1/2" HBD-90 and 1" HBD-1479 is eroded and leaking steam.

SAFETY EVALUATION: UFSAR Section 3.2 classifies the Offgas System (N64) and all of its components as "Other", meaning that loss of system function would not affect safe shutdown of the plant. Per UFSAR Table 3.2-1, the Offgas System is non-Q, non-safety related, non-seismic, and NRC Quality Group D. The modifications made by this change will not impose a change to these criteria listed in Table 3.2-1 for the Offgas System. Furthermore, the postulated worst case failure of the Offgas System analyzed in UFSAR Section 15.7.1 (Offgas System leak or failure) envelopes the occurrence and consequences of postulated accidents due to any failures associated with this change repair. The technical specifications are not affected and the margin of safety remains unchanged.

Serial Number: 94-071-NPE   Document Evaluated: MCP 91/1031, Rev. 0

DESCRIPTION OF CHANGE: This change will replace the stem and disc of check valve Q1E51-F021 with the stem and disc of a compatible globe valve.

REASON FOR CHANGE: The intent of the change is to remove the valve from the pump and valve program thereby alleviating the requirement of Specification SERI-M-189.1 that the valve be disassembled every outage to demonstrate its operability (verify that the disc will stroke fully open). This requires draining the RCIC pump discharge header between valves Q1E51-F200 and Q1E51-F013.

Valve Q1E51-F021 has no safety related function in the closed position since RCIC pump minimum flow protection is provided a dedicated containment penetration. However, it has an open safety function of providing minimum flow for RCIC pump protection. By converting the existing stop check valve to a globe valve, minimum flow will be achieved since the valve will be operated in the "locked open" position during normal power operations.

SAFETY EVALUATION: Replacing the internals (stem and disc) of the existing check valve with the stem and disc of a globe valve will not adversely affect the structural integrity of the E51 system. The piping and valve have been designed to ASME Section III code allowables and the system is properly supported for the appropriate loads.

The operability of the valve is not affected by this change, therefore the valve's ability to function in the mitigation of accidents remains unchanged (i.e., valve providing minimum flow for RCIC pump protection). Thus, this modification will not increase the consequences of any accident previously evaluated in the UFSAR.

This modification will not increase the probability of occurrence of a malfunction of equipment important to safety previously evaluated in the UFSAR. Reliability of the valve to provide minimum flow for RCIC pump protection by having it function in a "locked open" position will not be affected by this change. Thus, all intended functions of valve Q1E51-F021 will continue to be performed as designed and there is no reduction in the margin of safety as defined in the basis for any technical specification.

The change will not affect the function or operation of system E51 or any other systems and will not create an unreviewed safety question. The affected valve is not addressed in the technical specifications, therefore, no change to the technical specification is required.

Serial Number: 94-072-NPE Document Evaluated: DCP 91/0125, Rev. 0

DESCRIPTION OF CHANGE: Abandon in-place RHR B pump room sump level switch 1P45N042B and replace level switch 1P45N044B with a switch equipped with contacts for dual pump control.

REASON FOR CHANGE: A pipe hanger is installed directly on top of and touching level switch 1P45N042B which prevents access to, or removal of the switch for maintenance or calibration.

SAFETY EVALUATION: The Floor and Equipment Drain System (P45) has no safety related function. Implementation of this change only involves removing an existing flange mounted float type level instrument and replacing it with a flange mounted displacer switch equipped with tandem contacts for dual pump control and routing cables from the existing switch which is to be abandoned to the new switch. The switches are located in close proximity in the same room so no additional cables or penetrations will be required for the change. The pump control logic remains as before. Displacer switches tend to be more reliable than float switches so the performance of the switch will not be diminished. While a possible single failure mode is introduced in this particular application, of one switch for all alarming and control of the sump pumps, implementation of this change will not increase the probability of occurrence, or consequences of any accident previously evaluated in the UFSAR, or assumptions previously made regarding system performance during normal or accident conditions and no unresolved safety questions resulting from this design change.

Serial Number: 94-073-NPE Document Evaluated: UFSAR CR 94-039

DESCRIPTION OF CHANGE: This change adds the SRV bonnet vent line vacuum breakers to Table 3.9-3c (B21F100A-H, J-N, P, and R-W).

REASON FOR CHANGE: UFSAR Table 3.9-3c contains safety related active ASME Code Class 1, 2, and 3 valves for non-NSSS systems. The subject valves are active safety related ASME code valves and as such should be added to the table. The valves serve to relieve the vacuum that is created in the bonnet vent line (subexhaust line) caused by condensing the steam that leaks by the SRV piston/liner when the relief valve lifts. A negative pressure in the vent line may result in suppression pool water being drawn up in the line which could result in an unacceptable hydraulic transient in the vent line or affect the subexhaust line backpressure and tend to close the SRV prematurely. The subject valves are shown on UFSAR Figure 5.2-8 (F-6, B-6, Table I).

SAFETY EVALUATION: This change adds information to the UFSAR to provide a more accurate reflection of plant design. The change does not alter the design or operation of any system or component or affect the analysis of any event described in the UFSAR. The change will not increase accident or malfunction probabilities or consequences. The change will not create the possibility of a different type of accident or malfunction and does not reduce the margin of safety as defined in the bases for any technical specification.

Serial Number: 94-074-NPE Document Evaluated: DCP 88/0245 Rev. 0

DESCRIPTION OF CHANGE: This change redefines the security boundaries for GGNS and provides guidance in securing these new boundaries. Woven wire door 1A610 (shown on UFSAR Figures 12.3-17 and 12.6-6) is being deleted. Door OCT3 is being made an emergency exit. Door OC301 will no longer be required to be an emergency exit (Figure 12.3-14). Key card readers are being added and deleted; and the type of card reader is being changed. This change also installs new hand geometry readers at Security Island. Most of the physical changes to plant doors consist of replacement of the lock cylinders or the disabling and abandonment of the existing lock cylinder such that it cannot interfere with the operation of the lockset. Six safety related pressure doors (OC101, OC102, OC103, OC104, OC801, and OCT3) and one safety related secondary containment door (1A308) are further modified by mounting an electric deadbolt and associated electrical conduit/boxes to the face of the door. The lock cylinder and the thumbturn on secondary containment door 1A401 will be replaced with new lock cylinders.

REASON FOR CHANGE: Security deficiencies have been identified by several documents at GGNS. Also, existing key card readers are obsolete and spare parts are running low. There are also cost savings to be gained by redefining the security boundaries to reduce security postings and key card reader maintenance and repairs.

SAFETY EVALUATION: The C83 system (Security System) is not governed by technical specifications. The changes to this system required for this evaluation will enhance the system's performance and reliability. The components of the C83 system to be affected by this change 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 analysis are introduced. These changes will therefore not introduce an unreviewed safety question. Per review of Document Number 9645-A-022.4-OS-7.0-1-3 and Standard SERI-AS-02, the addition of an electric strike and a lock cylinder to a door (Q1T48Y705) lying within the secondary containment boundary will not have an adverse affect on this door's ability to function as designed. The replacement of lock cylinders in various plant doors which lie within secondary containment or the Control Room envelope with a cylinder of the same design, but keyed differently will not affect these doors' ability to function as designed. This change also required various Control Room envelope doors to have their lock cylinder cams modified to prevent them from engaging the lockset. Modification of these cams does not change the airflow currently seen by these lock cylinders, nor will it affect the doors' ability to remain latched closed. This change also provides adequate instructions and operational considerations to ensure work performed on penetrations, as well as doors, which lie within plant boundaries (i.e., Control Room envelope, secondary containment, pressure, and fire) complies with the technical specifications. For these reasons stated above, the changes required by this evaluation do not require a change to the technical specification, nor will they reduce the margin for safety as defined in the basis for any technical specification.

The C83 system (Security System) is not described in the UFSAR. The changes to this system required for this evaluation will enhance the system's performance and reliability.

The swapping of the emergency exit requirement from Door OC301 to OCT3 and the deletion of woven wire door 1A610 represent figure only changes to the UFSAR. Door 1A610 will not be required after implementation of this change. These doors are not governed by the technical specifications, and are not described in the UFSAR apart from where it is shown on floor plan figures. Therefore, its deletion does not affect the technical specifications or their margin of safety; increase the probability or consequence of an accident or malfunction evaluated in the UFSAR; and does not create the possibility of an accident or malfunction which has not already been considered in the UFSAR.

Serial Number: 94-075-NPE Document Evaluated: MCP 94/1001 Rev. 0

DESCRIPTION OF CHANGE: This change assesses the operation of containment hatchway crane (Jib crane) during Operational Modes 1, 2, and 3. Evaluation No. CFR 85/4503 R10 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 from Elevation 117' to 208' over the containment hatchway area during normal plant operation. For these circumstances, the containment polar crane cannot 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 cannot be used in lieu of the polar crane for handling of loads during normal plant operation. This change 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: Since the maximum 1000 pounds lifting load at each crane boom position is less than 1140 pounds technical specification 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 evaluation does not increase the probability or the consequences of any accident previously evaluated in the UFSAR, 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: 94-076-NPE   Document Evaluated: DCP 88/0172-03

DESCRIPTION OF CHANGE: This change addresses the following modifications:

1. installation of two new non-safety related Plant Data System (PDS) computer data acquisition panels in Control Building Computer Room OC403.

2. installation of panel mounted air cooling units on non-safety related PDS computer panels 1C91-P860, 1C91-P861, 1C91-P862, 1C91-P863, 1C91-P864, 1C91-P865, and 1C91-P866 (all located in the Control Building Computer Room OC403).

3. installation of a new transformer on Elevation 133' of the Turbine Building (1T327) and installation of a new power panel in the lower cable spreading room of the Control Building to feed the PDS panel air cooling units (both the transformer and power panel are non-safety related, non-Class 1E).

4. installation of new non-safety related PDS computer terminal workstations in the TSC, upper I&C shop, Control Room kitchen area, and the Control Room viewing gallery area on Elevation 177'.

5. deletion of various existing non-safety related C91 computer system components in Control Building Computer Room OC403 and in the main Control Room.

6. installation of power cable, specialty computer cable, and panel mounted computer system sub-components to support operation of the new PDS computer system.

REASON FOR CHANGE: The present plant computer systems have reached an age where they require ever increasing maintenance and improvements to support the needs of Plant Staff. The existing systems are overwhelmingly diverse and vendor support is becoming less than adequate to maintain these systems.

SAFETY EVALUATION: Neither the existing computer equipment affected by this design, nor the new computer equipment to be installed by this change is required to mitigate the consequences of any accident or transient. Implementation of the modifications proposed by this change will not compromise the operation of any existing safety related system, structure or component. No new interfaces with safety related equipment will be created by the proposed modifications and all cabling/equipment installations will be made in accordance with the separation requirements of Regulatory Guide 1.75.

Based on the safety evaluation discussion, no portion of this change will increase the probability or consequences of accidents or malfunctions of equipment important to safety previously evaluated in the UFSAR. Further, 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 UFSAR.

Since the computer system is not addressed by the technical specifications or the Technical Requirements Manual and since the operation of existing safety related systems, structures and components is unaffected by this design; this change will not reduce the margin of safety as defined in the basis for technical specifications.

Serial Number: 94-077-NPE Document Evaluated: Calc XC-Q1111-92010 Rev. 2

DESCRIPTION OF CHANGE: QDR 0189-94 (Reference 1) identified a deficiency in the GGNS LOCA dose analysis, Revision 1 to Calc XC-Q1111-92010. This QDR identified an incorrect number in the LOCA dose input file which resulted in incorrect modeling of SGTS releases starting at 13 minutes into the analysis. As part of the corrective actions, a revision to the calculation and an UFSAR change request have been prepared. This safety evaluation supports these documents.

REASON FOR CHANGE: This safety evaluation supports Revision 2 to Calculation XC-Q1111-92010 and UFSAR Change Request 94-034.

SAFETY EVALUATION: This evaluation confirms that, although some of the doses were calculated to increase, the offsite and control room doses still remain below the allowable regulatory limits reported in 10CFR100 and GDC 19.

Serial Number: 94-078-NPE Document Evaluated: Specification GGNS-M-183.3, Rev. 1

DESCRIPTION OF CHANGE: This specification provides the boundaries, requirements and criteria for the chemical cleaning and decontamination for portions of the Recirculation System (B33), the Reactor Water Cleanup System (G33 and G36) and the Fuel Pool Cooling and Cleanup System (G41). This change incorporates the established boundaries defined which expands the Recirculation System boundaries including portions of the reactor vessel drain piping. Also incorporated in this revision is the established boundaries defined per EER 94/6031 which adds the major portions of the Residual Heat Removal System (E12) for the decontamination.

REASON FOR CHANGE: The chemical cleaning and decontamination of the systems identified, including the expanded scope and boundaries, is being undertaken to reduce the dose rate for personnel servicing or in the vicinity of these systems.

SAFETY EVALUATION: Based on the criteria included in this change, any chemical cleaning and decontamination process meeting such criteria and selected for implementation at Grand Gulf will have no impact on the design bases, functional capabilities, and required design margins of the Reactor Recirculation System, Reactor Water Cleanup System, Residual Heat Removal System, and the Fuel Pool Cooling and Cleanup System. Therefore, it is concluded that this change does not increase the probability or the consequences of any accident evaluated in the UFSAR, does not create the possibility of a new accident or malfunction, and does not reduce any margin of safety defined in any technical specifications.

Serial Number: 94-079-NPE Document Evaluated: DCP 93/0026-1

DESCRIPTION OF CHANGE: This change will allow the installation of samplers in the Condensate Cleanup System (N22). The samplers will be located in the inlet and outlet piping of the ultrasonic resin cleaner (URC). Operation of the samplers will require an air supply from the Instrument Air System (P53), low voltage 120 Vac non-Class 1E power (R28) and demineralizer effluent from the N22 system. To regulate and control sampler operation, each sampler will be equipped with a controller.

REASON FOR CHANGE: This change will allow sampling of N22 demineralizer resin upstream and downstream of the URC operation. The resin samples may then be analyzed to determine cleaning efficiency of the URC.

SAFETY EVALUATION: UFSAR Section 10.4.6.3 states that the Condensate 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. According to UFSAR Section 10.4.6.2, the Condensate Cleanup System is provided with an automatic bypass to maintain condensate flow in the event a high differential pressure is realized across the Condensate Cleanup System. This change does not affect any technical specification and does not involve an unreviewed safety question. Furthermore, installation of samplers in the inlet and outlet lines of the ultrasonic resin cleaner will not increase the consequences or the probability of occurrence of the loss of feedwater transient analyzed in UFSAR Section 15.2.7, or any other accident or transient analyzed in Chapter 15 of the UFSAR.

With exception to piping and valves which form part of the containment boundary, UFSAR Section 3.2 classifies the Instrument Air System (P53) and all of its remaining components as "Other", meaning that loss of their function would not affect safe shutdown of the plant. Per UFSAR Table 3.2-1, the portions of P53 which do not form part of the containment boundary are NRC Quality Group D, and the Q-list and seismic category are not applicable. The modifications made by this change will not impose a change to these criteria listed in Table 3.2-1 for the Instrument Air System. Furthermore, the postulated failures of the Instrument Air System analyzed in UFSAR Section 15.2.10 (Loss of Instrument Air System) envelopes the occurrence and consequences of postulated accidents due to any failures associated with this design modification. The technical specifications are not affected and the margin of safety remains unchanged.

These samplers and controllers are not relied upon to support a safe shutdown of GGNS nor do they perform any safety related function. Therefore, this equipment is powered from Non-Class 1E low voltage 120 Vac power which is properly separated per Regulatory Guide 1.75 from Class 1E equipment/power sources.

Serial Number: 94-080-NPE Document Evaluated: MCP 93/1075 Rev. 0

DESCRIPTION OF CHANGE: This change replaces existing starting air condensate drain traps N1P75D015A, B, C, and D which are rated for 250 psig air service with traps that are rated for 300 psig air service so that the pressure rating of these components is in excess of the 285 psig setpoint for overpressure protection devices N1P75F068A, B, C, and D. The system affected is P75, Standby Diesel Generator System. This change is non-seismic Category I, non-seismic Category II/I and non-safety related.

REASON FOR CHANGE: MNCR 0113-92 identified that, in the Division I and II diesel generator starting air systems, various components rated for 275 psig or less were protected against overpressurization by relief valves N1P75F068A, B, C, and D which had a set pressure of 285 psig. Condensate drain traps N1P75D015A, B, C, and D were rated for 250 psig, the other components were rated for 275 psig. The objective of this change is to replace the existing condensate drain traps (N1P75D015A, B, C, and D) with traps that are rated for 300 psig so that these components pressure rating is in excess of the 285 psig setpoint for overpressure protection devices N1P75F068A, B, C, and D, Design Change Packages 87/0078, Revision 0 and 87/0078-1, Revision 0 will replace the remaining components.

SAFETY EVALUATION: The portion of the P75 system modified by this change is outside the safety related boundary of the Standby Diesel Generator Starting Air System. The affected portion of the system including the condensate traps are not required to function during an event requiring the standby diesel generators for accident mitigation. The replacement of the condensate traps with traps of a higher pressure rating will not prevent the diesel generators from performing their design function since the operation of the traps are identical to the traps being replaced. The new condensate traps are of a design to meet the actual system requirements of flow, temperature and pressure as well as improved maintainability and reliability.

This change does not delete any UFSAR, technical specification or quality commitments. There is a licensing commitment, by reference to the Transamerica Delaval Inc. (TDI) Diesel Generator Owners Group Maintenance Matrix in Attachment 2 to the operating license related to maintenance of the condensate traps. This commitment is contained in Standard MS-37 Revision 2 (Mechanical Standard for the Division I and II Standby Diesel Generator Maintenance and Surveillance) for component GG-115. Because the operation and maintenance requirements of the new trap is identical to that of the original trap, no change to this commitment is required by this design.

Serial Number: 94-081-NPE   Document Evaluated: MCP 91/1127, Rev. 0

DESCRIPTION OF CHANGE: The changes addressed in this evaluation are to the Makeup Water Treatment System (P21). They involve repair/replacing the acid off-load concrete pad and concrete pump pedestal, adding a drip pan (with a drain line to the acid dike sump) under the pump and adding isolation valves and drains in the suction and pump discharge lines with drains routed to the acid dike sump.

REASON FOR CHANGE: If the erosion of the concrete pedestal under the transfer pump (NSP21C019) continues unchecked, then the possibility of the pedestal collapsing and causing pump failure or line breakage while acid is being transferred is greatly increased. There is also no permanent means of draining the pump suction line after the acid off loading process is complete. The draining of the suction line is a personnel safety concern in that it would serve as a backup means of insuring the line between the pump and the acid truck would remain depressurized in the case of valve failure.

SAFETY EVALUATION: The Makeup Water System has no safety-related function as defined in Section 3.2 of the UFSAR. The modifications made by this change will in no way impact any of the accident analyses presented in the UFSAR. 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: 94-082-NSRA   Document Evaluated: Online Testing of ADHRS During Modes 1, 2, & 3

DESCRIPTION OF CHANGE: The purpose of this change is to allow, during Operational Conditions 1, 2, or 3, motor operated valve (MOV) testing for the E12-F066A&B valves and to provide the ability to perform surveillance testing of ADHRS valves, and to provide for the filling, venting, and flushing of the ADHRS 30 days prior to placing the system into operation for removal of decay heat from the reactor.

REASON FOR CHANGE: The reason for the change is to alleviate the additional non-availability of shutdown cooling methods during the early part of a refueling outage. Also, the allowance of MOV testing during normal operation will provide an added margin of safety during outages due to the availability of the ADHRS.

SAFETY EVALUATION: The performance of MOV and surveillance testing during Modes 1, 2, or 3 on the ADHRS and the filling, venting and flushing of both primary and PSW sides of ADHRS will save a projected 24 to 48 hours of outage time. The ADHRS will not be operated in any mode other than the suppression pool to suppression pool flush mode and the fill and vent mode while the plant is in Operational Conditions 1, 2, or 3. The surveillance stroke testing of the E12-F066A&B valves and the leak check surveillance test for the E12-F416 check valve will not be accomplished earlier than 30 days prior to placing the ADHRS into operation for the purpose of meeting the IST program requirements. Also, the ADHRS filling, venting, and flushing will not be accomplished any earlier than 30 days prior to placing the ADHRS into operation for the removal of decay heat. The ADHRS will be placed in a modified isolation mode following the fill, vent and flush. The PSW side will remain filled and in a wet lay up condition in accordance with the alternate lay up method specified in Admin Procedure 01-S-17-16. The PSW radiation monitor will also be placed into service during the PSW side fill, vent and flush as a precautionary measure. Current plant procedures adequately address the controls for isolation, testing and fill, vent and flushing evolutions. These specified evolutions for the ADHRS will not increase accident or malfunction probabilities or consequences. They will not create any risk of a different type of accident or malfunction and do not reduce the margin of safety as described in the technical specification bases. These proposed actions will not involve an unreviewed safety question.

Serial Number: 94-083-PSE Document Evaluated: CR PLS 93-011 Rev. 1

DESCRIPTION OF CHANGE: This change deletes the requirements to perform Type C leak rate testing on thirty-nine (39) containment isolation valves in twenty-six (26) primary containment penetrations listed in UFSAR Tables 6.2-44 and 6.2-49 and in Technical Requirements Manual (TRM) Table 3.6.4-1 (formerly Technical Specification Table 3.6.4-1). Most of these valves are isolation valves for various instrumentation and containment air sampling lines associated with the post-accident containment environmental assessment, although some of the instruments are also used during plant operation. The other valves are in post-accident cooling water lines for the drywell purge compressors. Various directives associated with the local leak rate testing program will also be revised as a result of this change.

REASON FOR CHANGE: These containment isolation valves are not required to be Type C local leak rate tested because they do not conform to the qualifiers for valves that require Type C testing under the definition of "Type C Test" as defined in 10CFR50, Appendix J, Definition II.H. They are small (two inches or less nominal pipe size), are designed to remain open in all modes of plant operation, including post-accident, and their instrument, sampling and cooling water lines inside and outside containment are designed as closed systems or loops. These valves are closed only for inservice testing in accordance with ASME Code Section XI and for maintenance on their respective closed loops. The closed loops inside and outside containment are seismic, missile protected and will be leak rate tested periodically as part of the boundary of the Type A testing required by Appendix J.

Eliminating Type C tests of these containment isolation valves will save significant outage time, man-hours and man-rem exposure.

SAFETY EVALUATION: The safety evaluation concludes that neither the probability nor the consequences of an accident or malfunction of equipment will be increased by exempting the local leak rate testing. The instrument and sampling loops outside containment and the cooling water loops inside containment are designed and constructed for operation under post-accident conditions. The consequences of an accident or malfunction of equipment are minimized by the valves' construction and periodic leak rate testing of the closed loop or system.

Serial Number: 94-084-PSE   Document Evaluated: Temp Alt 94-0009

DESCRIPTION OF CHANGE: Figure 9.2-027 shows that the discharge stop/check valves of all radial well pumps have a motor operated actuator. The motor operated actuators of these valves may be used to automatically adjust pump flow or pressure if the pump's controller is placed in the flow auto or pressure auto mode. However, Operations currently is not using the automatic function of any of the radial well discharge stop/check valves due to the unreliability of automatic operation; instead Operations selects the manual mode of the valves and places each to the desired position based on pump flow, motor amps, and well level. The change will remove the motor operated actuator from valve SP47F002B and install a manual operator in its place. The manual operator will function in the same manner as the motor operated actuator by allowing Operations to throttle the valve to the desired open position and by still performing its design function as a check valve upon flow reversal.

REASON FOR CHANGE: The gears in the motor operated actuator of valve SP47F002B have been found damaged and replacement parts for the valve are not readily available. A contractor is currently working on fabricating replacement gears for valve SP47F002B and is expected to supply the gears in a couple months time. This change will also allow valve SP47F002B to be installed so that Operations can place radial well pump SP47C001B in service if additional service water flow is required to the plant.

SAFETY EVALUATION: The Radial Well System is a non-safety related system. The failure of the Radial Well System will not affect the operation of any safety-related component and will not prevent safe reactor shutdown. This change will install a manual operator on valve SP47F002B, but the valve will still maintain its design function to operate as a check valve upon flow reversal. The change will not cause the Radial Well System to be operated in an unanalyzed manner and will not affect the operation of any safety-related component. The changes made by the temporary alteration 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 the margin of safety as described in the technical specification bases. This change does not represent an unreviewed safety question.

Serial Number: 94-085-NPE Document Evaluated: MCP 94/1050, Rev.0

DESCRIPTION OF CHANGE: Per this change, the following annunciator alarm points are deleted by removal of the annunciator alarm logic card that provides the input for the annunciator alarm windows and the blanking of the annunciator alarm window:

P870-6A-E7/DIRTY LUBE OIL STOR TK LVL HI/LO, Alarm Device SN34-LAHL-L607.

P870-3A-A3/CTMT-DRWL VENT ISOL DIV 1 OPER, Alarm Device 1M71-XA-L603A.

P870-9A-A3/CTMT-/DRWL VENT ISOL DIV 2 OPER, Alarm Device 1M71-XA-L603B.

P870-3A-H3/OTBD FW LCS PERM, Alarm Device 1E38-XA-L600A.

P870-6A-E6/CLEAN LUBE OIL STOR TK LVL HI/LO, Alarm Device SN34-LAHL-L608.

The following annunciator alarm point had the low level alarm point deleted by removal of the associated relay contacts from the circuit. The annunciator alarm window and computer point was changed to reflect the deletion of the low level alarm:

P870-5A-D3/RWST LVL HI/LO, Alarm Device 1P11-LAHL-L602.

REASON FOR CHANGE: The dirty oil storage and clean oil storage tanks' annunciator alarms (DIRTY LUBE OIL STOR TK LVL HI/LO and CLEAN LUBE OIL STOR TK LVL HI/LO) do not provide any operational value to the plant. Normally these tanks are isolated and out of service. Operations also checks the level in these tanks on a daily basis via the outside rounds. The function of the system is non-safety related.

The containment-drywell ventilation isolation Division 1 and 2 oper. (CTMT-DRWL VENT ISOL DIV 1 AND 2 OPER) annunciator alarms do not provide any operational value due to the fact the normal CTMT-DRWL ventilation lineup is in the isolated mode and the alarms are always activated.

The Outboard Feedwater Leakage Control System Permissive (OTBD FW LCS PERM) annunciator alarm is in during normal plant operation due to the system check valves holding extremely well. The SOI (04-1-01-E38-1) states in the prerequisites that the Feedwater Leakage Control System is only required in case of a LOCA, at which time it should be manually initiated once permission is obtained.

The Refueling Water Storage Tank (RWST LVL HI/LO) System is normally only used during refueling outages to store excess water. The HI level alarm will remain in service to provide a secondary means of preventing overflowing of the tank to radwaste. Normally during plant operation the tank is drained and isolated.

SAFETY EVALUATION: By deleting/modifying the following annunciator alarms, a blackboard condition for these referenced Control Room annunciators will be achieved: DIRTY LUBE OIL STOR TK LVL HI/LO Alarm (N34), CTMT-DRWL VENT ISOL DIV 1 & 2 OPER-Alarms (M71), OTBD FW LCS PERM-Alarm (E38), CLEAN LUBE OIL STOR TK LVL HI/LO Alarm (N34), and RWST LVL HI/LO Alarm (P11). These annunciators, which are not nuisance annunciators, are working as designed and they are giving accurate information. The reason for the change is that some annunciator alarm points do not provide any operational value to the plant in that they either remain energized because of the operating mode of plant operations (normal operating mode) or the referenced systems do not need annunciation because it is monitored by Operations when in use. However, in order to achieve a blackboard condition for the Control Room annunciators, the annunciators that remain activated during normal operations have been deleted/modified.

The deletion/modification of the referenced annunciator alarms do not change the design function of the referenced systems. Each referenced system will remain in compliance of IEEE 279-1971 and Regulatory Guide 1.47, as a result of the design change.

Serial Number: 94-086-PSE   Document Evaluated: Temp Alt - Defeat Drywell Chiller Low Flow Trip/ Start Permissive

DESCRIPTION OF CHANGE: The proposed temporary alteration (TA) will install jumpers to bypass the contact which provides the start permissive/shutdown function to the drywell chillers. The contacts which provide the standby function to the chilled water pumps, alarm annunciation, and computer point will remain operational.

REASON FOR CHANGE: The plant recently experienced an intermittent failure of Drywell Chilled Water Flow Switch 1P72N001B. This intermittent failure resulted in an automatic trip of the then running Drywell Chillers 1P72B001B and 1P72B002B.

The cause of the failure is erosion of the actuating paddle. Because the flow switches are mounted in an un-isolable section of the chilled water piping that is common to both pumps and all four chillers, it is not possible to replace or repair the failed part without shutting down the Chilled Water System. The likelihood of continued intermittent failure or complete failure of 1P72N001B, or similar failure of 1P72N001A is high. Failure of these switches could prevent operation of the drywell chillers allowing drywell temperature to exceed technical specification limits.

SAFETY EVALUATION: The primary components affected by this temporary alteration are Drywell Chillers 1P72B001A, 1P72B002A, 1P72B001B, and 1P72B002B. The temporary alteration provides the jumpers necessary to defeat the "low chilled water flow" trip signal to each of the four drywell chillers. The four chillers are each skid mounted and part of the (P72) Drywell Chilled Water System. The supply and return chilled water lines for the drywell coolers penetrate the containment and drywell. Isolation valves are provided in these lines which can be remotely actuated from the Control Room or closed automatically upon receiving an isolation signal. Other than this, the Drywell Chilled Water System serves no safety function as defined in Section 3.2 of the UFSAR. Failure of the system will not compromise any safety-related systems or components and will not prevent safe reactor shutdown. The temporary alteration will not alter or affect overall system performance or reliability.

Serial Number: 94-087-PSE Document Evaluated: WO #130443

DESCRIPTION OF CHANGE: The output of the Electronic Generator Protection (EGP) System for the Primary Water System to the generator trip circuits will be disabled.

REASON FOR CHANGE: A failure in a card in the primary water pump vibration portion of the EGP requires that the generator trip output from the affected portion be disabled to make repairs. There is no way to remove the generator trip output of the affected portion by itself without cutting wires, so the generator trip output of the entire Primary Water System section of the EGP must be disabled.

SAFETY EVALUATION: A failure in the Primary Water System section of the EGP has made it necessary to bypass all generator trips associated with the Primary Water System. The EGP System is used to protect the generator. When a condition exists that threatens the safe operation of the generator or its support systems the EGP takes the generator off line which will trip the turbine. The wiring modification to be performed will disable all trips associated with the Primary Water System. The alarm function and indication from all instrumentation however will not be disabled except when a card is pulled. The generator will be protected by operations personnel continuously monitoring the Primary Water System parameters while the trips are disabled. The Primary Water System flows, temperature and vibration should be monitored on the computer and appropriate actions taken if alarm limits are reached. The continued operation during trouble shooting with no primary water generator trips is within the bounds of all previous analysis. It is not the intent of this evaluation to operate the generator for an extended period of time without protective trips.

Serial Number: 94-088-NPE   Document Evaluated: UFSAR CR 94-025

DESCRIPTION OF CHANGE: Regulatory Guide 1.45 describes methods of implementing the requirements of General Design Criterion 30, "Quality of Reactor Coolant Pressure Boundary," of Appendix A to 10CFR Part 50. Types of leak detection equipment, such as sump level, airborne particulate radioactivity and temperature change as well as instrument measurement accuracy are also discussed. In particular, it is stated as a requirement that Control Room annunciation be provided to indicate the detection of a leak rate change of 1 gpm in one hour based on drywell floor drain sump level and flow monitoring. The UFSAR change clarifies Grand Gulf's exceptions to this requirement.

REASON FOR CHANGE: The regulatory guide requires a minimum of three detection methods and requires resolution and indication of 1 gpm increase on each system. GGNS has many different methods of detecting increases in unidentified leakage, including drywell temperature and pressure monitoring, drywell cooler flow and temperature monitoring, drywell airborne particulate radioactivity monitoring, drywell gaseous radioactivity monitoring and drywell sump level and fill rate/pump down timers (Reference 11). Operator rounds are required by Procedure 06-OP-1000-D-0001 (Reference 2) to calculate unidentified leakage rate every 12 hours. Review of the instrument loops for sump level indicate that Control Room alarm of a 1 gpm increase in 1 hour is not possible with existing equipment (References 1, 4 and 6). The change in the UFSAR is being made to preclude conflicts with commitment to regulatory guide requirements or having to implement a design change.

SAFETY EVALUATION: GGNS has several different methods of leak detection (References 6 and 11), and through the use of all of them they can inform operators of an increase in drywell unidentified leakage (References 7 and 11). Operator rounds measure the leakage rate every 12 hours (Reference 2) based on sump level and flow rates. UFSAR analysis is based on detection of 5 gpm total unidentified leakage (Reference 10). Because the analysis is based on 5 gpm, there is no safety consequence by not having the sump level and flow circuits or any other subsystem modified to detect the flow change stated in the regulatory guide. Taking exception to this requirement will not increase the probability of any accident analyzed in the UFSAR nor will it increase the consequences of any accident analyzed in the UFSAR. The safety analysis is not affected by this change because it is based on detection of 5 gpm.

Serial Number: 94-089-NPE Document Evaluated: UFSAR CR 94-043

DESCRIPTION OF CHANGE: This change request involves primarily a restructuring and simplification of the Nuclear Plant Engineering organizational structure and realignment of some reporting relationships. The NPE organizational description in Section 13.1 of the UFSAR is also simplified. Also contained in this change request is the deletion of Nuclear Plant Engineering staffing information from Table 13.1-1.

REASON FOR CHANGE: The reason for the change is to reflect the simplified, more efficient design engineering organizational structure to be implemented in 1995 and to simplify the Nuclear Plant Engineering organizational description contained in the UFSAR.

SAFETY EVALUATION: Because of the evaluated change being an administrative and organizational change only, there is no physical effect on the plant and also no changes in the operational methods or practices of the plant. Consequently, there is no possibility for an unreviewed safety question.

The proposed changes to Table 13.1-1 involve the deletion of NPE internal group names and staffing levels. The group names are also described in greater detail in the remainder of Section 13.1 and are thus redundant. The staffing levels are identified as "Budgeted" staffing levels; consequently, these stated staffing levels have no real bearing on the level of resources actually applied. The inclusion of these levels in the UFSAR contribute nothing to the safe and efficient operation of GGNS nor anything to the protection of the health and safety of the general public.

Serial Number: 94-090-NPE   Document Evaluated: MCP 93/1064

DESCRIPTION OF CHANGE: This change removes the out of service section UHS basin transfer piping between the SSW "B" pump and the SSW "B" basin wall penetration.

REASON FOR CHANGE: The transfer piping below the normal SSW basin water level was found to be extremely corroded. The transfer lines were originally designed to permit Unit 2 to Unit 1 transfer (or Unit 1 to Unit 2 transfer) of basin inventory following a LOP LOCA DBA initiation by using the non-LOCA unit's SSW pump. This original design feature was eliminated with the cancellation of Unit 2. Following cancellation of Unit 2 construction, a siphon line was installed between the "A" and "B" basins for a passive transfer system to perform the required basin inventory transfer from the basin divisionally associated with the postulated failed diesel generator. Since the original transfer line from SSW "B" to SSW "A" no longer serves any safety function, the corroded line will be removed rather than replaced.

SAFETY EVALUATION: Removal of the remaining corroded piping from the active transfer system does not prevent safe operation of the SSW and UHS systems.

The UHS minimum water level requirements and transfer valve surveillance requirements described in the GGNS Technical Specifications are not altered by the design change.

The modification will not increase the probability of occurrence or the consequences of an accident previously evaluated in the UFSAR, since the modification does not alter any safety related functions of equipment.

The modification will not create the possibility for an accident of a different type than any previously evaluated in the UFSAR due to maintaining existing design requirements for the modification of a piping system with no safety function.

The modification will not create the possibility for a malfunction of equipment important to safety of a different type than any previously evaluated in the UFSAR due to these modifications being made to ensure the long term integrity of the UHS fluid boundary ("A" basin transfer line penetrations) against failure from corrosion.

The modification will not reduce the margin of safety as defined in the basis for any technical specification since the implied margins of safety for the 30 day cooling water inventory are not reduced.

Serial Number: 94-091-NPE Document Evaluated: UFSAR CR 94-038

DESCRIPTION OF CHANGE: UFSAR Section 5.4.6.2.2.2.K will be changed to correct the instrument name and required accuracy for N010 and N037, and to correct the instrument name, add the instrument number and add a clarifying note for the CST low level instrument loops N635A, E. Table 7.4-1 will be revised to list individual instruments for the RCIC pump low/high suction pressure loops along with the individual ranges. Table 7.4-1 will also be changed to add a clarifying note for the RCIC steam supply low pressure loop. All the changes included in the UFSAR change request are editorial changes and do not change the facility, procedures or tests described in the Safety Analysis Report and therefore cannot introduce an unreviewed safety question.

REASON FOR CHANGE: The LPCS/RCIC Safety System Functional Assessment QSA-93/0012 that was performed during the 2nd quarter of 1993 listed 5 I&C comments that possibly could require a change to the UFSAR. Change Request 94-038 was written to change the UFSAR sections where a revision was found necessary.

SAFETY EVALUATION: The change corrects and clarifies the UFSAR. The safety functions of the RCIC system are not affected or changed in any way. The change does not alter the design or operation of any system or component or affect any analysis described in the UFSAR. The change does not introduce new malfunctions in any system or component. Because the RCIC system has not been changed and will continue to function as designed and previously evaluated, no safety margin will be reduced. The editorial changes being made by this UFSAR change request do not change the facility, procedures, or tests described in the safety analysis report.

Serial Number: 94-092-NPE   Document Evaluated: MCP 89/1109 Rev. 0 and CN 94/0063

DESCRIPTION OF CHANGE: CFRMCP89/1109R00 is not superseded by this change. This change (CFRMCP89/1109R01) will only evaluate the impact on the safe operation of GGNS for those changes to MCP 89/1109 per CN 94/0063.

CN 94/0063 will eliminate the 1KΩ ground detection circuit from Bus 11DA and 11DB. MCP 89/1109 replaced the original ground detection circuit on these buses due to obsolescence of the original ground detection meter-relay.

REASON FOR CHANGE: Interim 1 to MNCR 0166-94 discusses the scram that occurred on 11/01/94, due to an unexpected inadvertent actuation of backup scram valve 1C11-F110A. The 1KΩ ground detection circuit provided by MCP 89/1109 allowed sufficient fault current to conduct, as a result of an existing bolted fault on solenoid valve 1C11-F110A, which was a contributing cause to the scram. However, it should be noted that the maximum voltage drop across solenoid valve 1C11-F110A that occurred due to the 1KΩ ground detection circuit was limited to less than 1/2 of the vendor published minimum operating voltage for this solenoid valve. As such it is credible to assume that no inadvertent actuation should have occurred.

An operational reliability review of all loads on Bus 11DA and 11DB was conducted to identify if the 40KΩ ground detection circuit designed by MCP 89/1109 could support an inadvertent actuation upon a postulated bolted ground fault. The review concluded that it is not credible to assume an inadvertent actuation upon a postulated bolted ground fault due to the 40KΩ current limiting resistors.

SAFETY EVALUATION: The Division 1 and 2 new ground detection system consisted of two circuits. The first circuit is composed of a current sensitive relay connected between ground and each pole of the battery through 1KΩ resistors. The circuit effectively provides a 1KΩ resistance return path for DC current from ground to the battery. The 1KΩ resistance was not large enough to sufficiently limit the DC current to ensure that Division 1 load would not inadvertently energize in the event of the bolted ground fault on Bus 11DA. This circuit also provides a contact input to the Control Room computer that provides an alarm. This circuit will be removed.

The second circuit is a ammeter connected between ground and each pole of the battery through 40KΩ current limiting resistors. This circuit allows operations to monitor the DC buses during rounds for detection of a developing ground fault. The ground detection ammeter will not provide minimum design required pickup values in the event of a fault and thus reduces the possibility of inadvertent actuations. This circuit will insure that the design of Buses 11DA and 11DB meets the requirement of NRC GDC 18 - Inspection and Testing of Electrical Power Systems by providing a method of identifying grounds on these buses.

Serial Number: 94-093-PSE Document Evaluated: CN 94/0063 to MCP 89/1109

DESCRIPTION OF CHANGE: The 11DA and 11DB 125 Vdc buses are designed to be ungrounded buses; therefore, a single ground on either bus should not affect the bus because there should be no return path for current from ground to the battery. However, the ground detection circuits for both buses utilize current sensing relays to continuously monitor current from ground to the bus. This circuit effectively provides a 1KΩ resistance return path for current from ground to the battery. The 1KΩ resistance is not large enough to sufficiently limit current to ensure that no plant loads will inadvertently energize in the event of a single ground on either DC bus.

The manual function of both ground detection circuits will be unchanged. Using the provided pushbutton and ammeter, the current from ground to each bus can be monitored through a 40KΩ resistor.

This change removes the continuous monitoring portion of the ground detection circuits for Buses 11DA and 11DB. The continuous monitoring portion of both circuits provides inputs to BOP computer points which alarm in the event of a low resistance ground. UFSAR Section 3.1.2.2.9 states, "The DC system has detectors to indicate and alarm when there is a ground existing on any part of the system."

REASON FOR CHANGE: A reactor scram occurred due to a single ground on a Division 2 load which caused a low resistance current path through the continuous monitoring portion of the ground detection circuit.

SAFETY EVALUATION: Although removing the continuous monitoring portion of the ground detection circuits also disables the alarming function of the circuit, this change actually increases the reliability of the Divisions 1 and 2 DC systems by making both systems true ungrounded DC systems. Manual ground detection circuits will continue to be provided for both buses which are capable of detecting higher resistance grounds than the continuous monitoring circuit could detect.

The time delay in discovering a ground due to not having continuous monitoring does not increase the probability of an accident because no single ground can inadvertently energize a load on a true ungrounded DC system. Multiple grounds occurring between manual tests are not expected. If multiple grounds were to occur with a continuous monitoring circuit installed, the difficulty in finding and correcting a DC ground makes it unlikely that one ground would be found and corrected before the second ground occurred.

Serial Number: 94-094-NPE Document Evaluated: MNCR 94/0036

DESCRIPTION OF CHANGE: The final disposition of this change is to repair the SSW pump shaft coupling fasteners by using monel cap screws instead of carbon steel cap screws. The design change will also eliminate the lock washers from the coupling's cap screws. The final disposition also requires that the SSW pump's carbon steel suction bell and impeller housings be replaced on a six year interval. The design changes and rework intervals apply to both the "A" and "B" SSW divisions. The disposition further places requirements to inspect the HPCS-SSW pump and one Byron Jackson ECCS deep draft pump for corrosion.

REASON FOR CHANGE: MNCRs 0030/94 and 0036/94 documented SSW pump impeller wear due to corroded cap screws and lock washers on the pump shaft couplings. The design change of the cap screw material and lock washer elimination is being made per the final disposition of this change to eliminate the potential for corrosion on the shaft coupling fasteners. The design change was recommended by the original equipment manufacturer, Goulds Pumps, Inc.

The final disposition of this change further requires inspections of additional pumps as part of the preventative maintenance program for safety related deep draft pumps at GGNS.

SAFETY EVALUATION: The design change will maintain the SSW pump reliability by eliminating the potential for corrosion on the pump shaft coupling fasteners. The replacement of the carbon steel suction bell and pump bowls on a periodic basis will maintain the reliability of the SSW pumps by maintaining the pumps in the state to which they were originally designed. The inspections required by the disposition support the existing plant preventative maintenance program for safety related deep draft pumps. The design changes, periodic rework, and inspection program do not require a change to the GGNS Technical Specifications, will not increase the consequences nor increase the probability of occurrence of an accident previously evaluated in the UFSAR, will not increase the consequences nor the probability of occurrence of a malfunction of equipment important to safety previously evaluated in the UFSAR, will not create the possibility for an accident of a different type than any previously evaluated in the UFSAR, will not create the possibility for a malfunction of equipment important to safety of a different type than any previously evaluated in the UFSAR, and will not reduce the margin of safety as defined in the basis for any technical specification.

Serial Number: 94-096-NPE   Document Evaluated: CN 94/0064

DESCRIPTION OF CHANGE: Safety Evaluation Nos. CFRMCP89/1080R00 and CFRMCP89/1080R01 are not superseded by this safety evaluation. This safety evaluation (CFRMCP89/1080R02) will only evaluate the impact on the safe operation of GGNS for those changes to MCP 89/1080 and existing approved CNs (CN 90/0105 & CN 91/0058) issued against MCP 89/1080 as changed by CN 94/0064.

CN 94/0064 eliminates the 1KΩ ground detection circuit from bus 11DK and 11DL and the 200Ω ground detection circuit from bus 11DJ.

REASON FOR CHANGE: Review of bus 11DK loads, during evaluation of MNCR 0166-94, identified that the 1KΩ ground detection circuit installed per MCP 89/1080 in conjunction with a postulated bolted ground fault on the ARI/RPT initiation trip logic may cause an inadvertent recirculation pump trip. Review of bus 11DL loads did not identify any credible inadvertent actuation upon a postulated ground fault on this bus. Bus 11DJ provides +/-24Vdc instrumentation power for the main turbine generator system. As such, there is no known acceptable minimum ground fault current that could be assumed to cause no inadvertent actuation upon a postulated ground fault.

This CN eliminates the low impedance ground detection circuit installed by MCP 89/1080 on buses 11DJ, 11DK and 11DL. The high impedance ground detection circuits installed by MCP 89/1080 on these buses shall not be eliminated. These circuits do not provide a credible return path to support inadvertent actuation with a postulated ground fault. However, as stated above, there is no known acceptable minimum ground fault current for bus 11DJ; and as such to insure its high impedance ground detection system does not support an inadvertent actuation of any instrument loop associated with this bus, the pushbutton for enabling the high impedance ground detection circuit should not be depressed during normal plant operations.

CN 94/0064 addresses only battery buses 11DJ, 11DK and 11DL because its associated MCP, (MCP 89/1080) only addressed these battery buses. However, since bus 11DH is also a +/-24 Vdc instrumentation supply for the main turbine generator system and interfaces with battery 11DJ through a common selector switch, the following recommendations should be considered. Upon implementation of this CN to bus 11DJ, selector switch "CS" on distribution panel 1DH3 should be located to Position 3. This will eliminate the designed continuous ground from bus 11DH. As identified above for bus 11DJ there is also no known acceptable minimum ground fault current for bus 11DH and as such to insure its high impedance ground detection system does not support an inadvertent actuation of any instrument loop associated with this bus, the pushbutton for enabling the high impedance ground detection circuits should not be depressed during normal plant operations.

SAFETY EVALUATION: Implementation of CN 94/0064 will not have any adverse affects on the safe operation of GGNS. The requirements of Regulatory Guide 1.75, Physical Independence of Electrical Systems, shall be maintained to ensure that this change does not affect any Class 1E equipment. These buses (11DJ, 11DK and 11DL) are not utilized in the analysis for the mitigation of any accident in the UFSAR. The performance capabilities for the DC systems are not adversely impacted by this change. This change will not increase the consequences of a malfunction of equipment important to safety. These buses are not utilized for forming a basis of any technical specification.

Serial Number: 95-001-NPE Document Evaluated: MC-Q1F15-89010, Rev. 4

DESCRIPTION OF CHANGE: Calculation MC-Q1F15-89010, Revision 4 updates the GGNS Fuel Handling Accident Analysis to consider (i) increased rod average burnup limits for the 9x9-5 fuel, (ii) the increased reactor coolant temperatures permitted with the Improved Technical Specifications (ITS), and (iii) increased scrubbing efficiency for drops in which greater than 23 feet of water coverage is available. An additional case was developed reporting the results of the limiting drop without secondary containment after 12 days of decay in support of the ongoing licensing burden reduction effort. This calculation generates revised impact energy limits for the handling of loads over irradiated fuel. These new restrictions have been transmitted to the plant in GIN-94/03427.

REASON FOR CHANGE: This calculation revision was generated primarily in response to the ongoing licensing burden reduction effort to eliminate secondary containment requirements after 12 days into a refueling outage. In addition to the design basis drops, an additional case was developed to confirm the consequences of this new event are within those permitted by SRP Section 15.7.4.

SAFETY EVALUATION: The consequences of the worst case fuel handling accident were calculated to be significantly less than the NRC acceptance criteria reported in SRP Section 15.7.4 and General Design Criterion 19.

Serial Number: 95-002-NPE Document Evaluated: DCP 88/0280-1 Rev. 0

DESCRIPTION OF CHANGE: This change deletes the main steam line high-high radiation input to the RPS channel trips, the inboard and outboard MSIV trips, the main steam line drain valve trip, and the input to annunciator 1B21-XA-L710. MSL radiation high trip annunciator 1C71-RAH-L602 is also deleted. This change also standardizes the setpoints for the main steam line radiation high alarm and the offgas pre-treatment radiation high (upscale) alarms.

REASON FOR CHANGE: NEDO 31400A presents a safety analysis performed by General Electric for the participating BWROG members to support a request for removing the main steam line isolation valve (MSIV) closure function and the automatic reactor shutdown function of the main steam line radiation monitors (MSLRMs). The MSLRMs detect moderate-to-large fuel failures and close the MSIVs to stop the release of radioactivity into the steam lines. Presently, at GGNS, an alarm sounds when the radiation level exceeds 1.5x the full-power background level. At 3 times the full-power background level, trip circuits automatically shut down the reactor, close the MSIVs, trip the main steam line (MSL) drain valves, close reactor water sample valves, trip the mechanical vacuum pump, and isolate N62 offgas valves. Although no spurious trips or MSIV closures have occurred at GGNS, the proposed changes removes the potential for such trips, as well as reducing required surveillance and maintenance activities.

SAFETY EVALUATION: The MSLRM will continue to have both safety and non-safety related functions. The safety related function will be isolation of reactor water sample valves 1B33-F020-A and F019-B (Group 10 isolation). The non-safety related functions will be annunciation, trip N62 vacuum pumps N62C001A-N, B-N, C-N, and closure of offgas isolation valves N62-F007A-N/B-N.

At present, the MSIVs and the MSL drain valves receive the same isolations signals (Group 1 isolation). The MSL high radiation trip for the MSIV drain valves will not be required after the removal of the trip from the MSIVs.

Each RPS scram input also provides input to an annunciator. For the MSL high-high radiation input, this annunciator is 1C71-RAH-L602. Due to the removal of the input to the RPS system and the fact that D17 system annunciators are available for high-high radiation annunciation, 1C71-RAH-L602 will no longer be required. Annunciator 1B21-XA-L710 is for "MSIV/drain valve trip initiated". The MSL high-high radiation input to this annunciator will be removed to coincide with the logic change to the MSIV and MSL drain valve control circuits.

To minimize nuisance alarms, GGNS has chosen to use an offgas pre-treatment high radiation alarm setpoint of either 2x the normal background radiation at the detector location or 20 mR/hr, whichever is greater. The offgas pre-treatment radiation high and high-high trips have an alarm only function with no automatic actions. Technical Specification Section 3.11.2.7 requires, "The gross radioactivity (gamma) rate of the noble gases measured at the offgas recombiner effluent shall be limited to less than or equal to 380 millicuries/second, after 30 minutes decay". If this specification is exceeded, the specification requires the release rate to be restored to its limit within 72 hours or be in at least HOT SHUTDOWN within the next 12 hours and in COLD SHUTDOWN within the following 24 hours. The offgas pre-treatment radiation high-high setpoint is calculated using a conservative equation to help ensure this technical specification limit is not exceeded.

Serial Number: 95-003-PSE   Document Evaluated: TSTI 1B33-95-001-0-S
Recirc A Purge Flow Test

DESCRIPTION OF CHANGE: TSTI 1B33-95-001-0-S reduces purge flow to zero flow and then restore normal purge flow to Recirculation `A' pump seals as part of a troubleshooting effort to determine the effects of purge flow on the present anomaly of vibration and seal temperature oscillations. The test allows the option of leaving purge flow secured to the Recirculation `A' pump seals, if it is determined to be the best solution to the oscillation problem. In this case, this safety evaluation will also serve to evaluate implementation of a temporary alteration to allow purge flow to remain secured.

REASON FOR CHANGE: This test is needed as part of a troubleshooting effort to determine and eliminate the vibration and seal temperature oscillations that are currently present on Recirculation Pump `A'.

SAFETY EVALUATION: The Reactor Recirculation System safety design bases are: 1) assure adequate fuel barrier thermal margin, 2) piping integrity failure shall not compromise the ability of the reactor vessel internal to provide a refloodable volume, and 3) maintain pressure integrity during adverse combinations of loadings and forces occurring during abnormal, accident and special event conditions. The purpose of seal purge flow, as discussed in UFSAR Section 18.1.30.10 is to provide reactor grade water to minimize seal wear and prolong seal life and therefore play no role in or have an effect on the system safety design functions listed above.

Operation with no purge flow has already been approved and documented on Safety Evaluation CFMRISC0118R00. The conclusion of that evaluation was that no technical specification changes were required and that no unreviewed safety question arises as a result of operation without purge flow.

The test (TSTI 1B33-95-001-0-S) reduces purge flow in 0.1 gpm increments to ensure that seal temperatures are maintained less than 180°F and to ensure that no new temperature or vibration transients are created as a result of the change in purge flow. The controlled and slow reduction in purge flow, along with the fact that the seals are designed by the vendor (AECL) to operate without purge flow, will ensure that no abnormal conditions or system transients are created. The worst event postulated because of the changing purge flow is a complete loss of both #1 and #2 Recirculation Pump `A' seals. This event, loss of both #1 and #2 seals on one recirculation pump, is well within the bounds of UFSAR Chapter 15 analysis of a DBA LOCA.

Restoration of purge flow in 0.1 gpm increases with pauses to observe system affects, is also not expected to result in any system transients or abnormal conditions. The slow restoration will ensure no thermal transients are produced in the pump shaft or heat exchanger region, and ensure no oscillations are initiated.

Chapter 15 accident analyses involving the recirculation pump are pump trip, pump seizure, and pump shaft break. The probability of occurrence of any of these events is no more likely by restoration of purge flow because of: 1) the modification implemented in RF06 which installed a sacrificial shaft sleeve, 2) the controlled method of re-introduction of seal flow in 0.1 gpm steps, 3) seal flow maximum is a reduced flow of 1.7 to 2.0 gpm as recommended by GE in SIL 511.

The conclusion reached by this safety evaluation is that reducing purge flow to zero, restoring to normal flow or operating without purge flow does not impact technical specifications or result in any unreviewed safety question.

Serial Number: 95-004-NPE Document Evaluated: MNCR 0016-95
Interim Disposition #1

DESCRIPTION OF CHANGE: Perform an on-line leak repair of a thermal relief valve on the feedwater side of a low pressure feedwater heater. Also install a bypass line around the affected low pressure heater string isolation valve to minimize the possibility of over-pressure on the feedwater side of the low pressure heater. The on-line leak repair as well as the bypass line are temporary repairs to be reworked to the original design requirements not later than the Spring, 1995 refueling outage.

REASON FOR CHANGE: A thermal relief valve has developed a leak across the valve seat and is eroding the piping downstream of the valve. The downstream pipe is also leaking due to an apparent fatigue crack at a welded joint. Isolating the heater string and replacing the relief valve would involve a special evolution for operations, while the temporary on-line repair and bypass line will maintain a stable plant.

SAFETY EVALUATION: The Condensate System (N19) serves no safety function and failure of the system will not compromise any safety related systems or prevent safe shutdown of the plant. The Condensate System is not required to effect or support the safe shutdown of the reactor or perform in the operation of reactor safety features. The existing accident analyses related to the temporary repair are the loss of feedwater heating and the loss of feedwater. The potential safety impact from implementing the repair is bounded by the existing safety analysis in the UFSAR.

Serial Number: 95-005-NPE Document Evaluated: DCP 94/0004

DESCRIPTION OF CHANGE: Initiated 2 SCNs to Standards MS-02 and MS-03 to allow for the replacement of the existing uncoated carbon steel SSW Basin "A", Cells C&D and Basin "B", Cells C&D spray sparger piping with galvanized carbon steel piping with a special coating applied. Replacement and repair of some piping system supports. Also basin "A", Cells A&B HPCS spray sparger piping will be special coated to reduce or eliminate any future corrosion problems to this piping system.

REASON FOR CHANGE: Numerous MNCRs have been written against the SSW basin "A", Cells A,B,C&D and Basin "B", Cells C&D spray sparger piping and piping support systems for pipe wall failures and pipe and support corrosion. Replacing the existing uncoated carbon steel piping with coated, galvanized carbon steel piping for Basin "A", Cells C&D and Basin "B", Cells C&D and special coating the SSW Basin "A", Cells A&B piping will protect the piping system from the constant wetting and drying environment. This type of environment has proven to cause adverse effects to uncoated carbon steel piping systems. This change will eliminate or greatly reduce corrosion problems the SSW spray sparger has experienced in the past.

SAFETY EVALUATION: Replacing and repairing some existing pipe supports and initiating 2 SCNs to MS-02 and MS-03 to allow for the replacement of the existing uncoated carbon steel with a special coating applied to the galvanized carbon steel piping system and adding a special coating to the HPCS spray sparger piping will not in any way reduce the safety function of the spray sparger piping contained in the SSW Basin "A", Cells A,B,C&D and Basin "B", Cells C&D. These changes will only increase the life span of the existing carbon steel piping and the new galvanized carbon steel piping spray sparger piping used in these basins. These changes will also assure that the spray sparger piping system, while performing the SSW design safety function will not fail due to pipe and pipe support corrosion.

Serial Number: 95-006-NPE Document Evaluated: UFSAR CR 94/014

DESCRIPTION OF CHANGE: This change will revise the UFSAR to reflect that the maximum opening/closing stroke time requirements for the motor operated valves in the MSIV-LCS are 15 seconds for the inboard subsystem valves and 30 seconds for the outboard subsystem valves. This will reflect the intent of the original design requirements and the MSIV-LCS system design criteria. The UFSAR presently indicates a maximum opening time of 15 seconds and a maximum closing time of 30 seconds for all the motor operated valves in the MSIV-LCS.

REASON FOR CHANGE: The maximum stroke time requirements for the motor operated valves in the MSIV-LCS as stated in UFSAR Section 6.7.1.3.1b are incorrect.

SAFETY EVALUATION: This change will not affect any present GGNS Technical Specifications nor does it impose any new requirements in the specifications. The change does not alter any of the original design requirements, operating parameters, or actual operating times of any motor operated valve in the MSIV-LCS. The changes to the maximum valve operating times presently depicted in the UFSAR will not adversely affect the consequences of, or increase the probability of any accident described in the UFSAR. In addition, the change will not adversely affect any other system, equipment or event described in the UFSAR. This change will not affect the single failure analysis of the MSIV-LCS. This change will also bring the UFSAR into conformance the intent of the system design criteria, SERI-SDC-E32, Revision 0 and the original design documentation.

Serial Number: 95-007-NPE Document Evaluated: MCP 93/1055 Rev. 0

DESCRIPTION OF CHANGE: A voltage regulation transformer is being added to two circuits fed by panel 1Y77 serving security MUX panels SC83-P004 and SC83-P011. The purpose of this addition is to correct low voltage conditions at the panels. The circuits are shown on UFSAR Figure 8.3-007A. Circuit loading does not change.

REASON FOR CHANGE: MNCR 93/0137 identified low voltage conditions at panel SC83-P004. Tests showed that the voltage would drop on startup of the MUX equipment to a low of 90 volts. This is below the acceptable voltage level of the associated equipment. This change will install voltage regulating equipment to correct this condition.

SAFETY EVALUATION: The changes in this evaluation will not compromise any existing safety related system, structure or component, nor will they prevent safe reactor shutdown. The equipment affected is part of the Security Monitoring System (C83) which is non-safety related and whose function will not change due to the equipment added by this change. The cable and raceway modifications will be in accordance with the requirements of Regulatory Guide 1.75 and seismic concerns have been addressed.

Serial Number: 95-008-NPE   Document Evaluated: MNCR 0134-94

DESCRIPTION OF CHANGE: MNCR 0134-94 is dispositioned to "accept-as-is" the piping configuration of supply and drain piping in Room OM123 of the Water Treatment Building that is currently plugged at the bathroom wall penetrations. The "accept-as-is" disposition also requires that the heat trace (NSR63T471) on the 4-inch drain line associated with the bathroom drain line be maintained "open" and abandoned since the bathroom supply/drain lines are plugged and no longer used. The MNCR disposition "repair" requires that vendor supplied valve on supply piping to WC-18A be repaired by plugging as needed to prevent leakage into the bathroom facility.

REASON FOR CHANGE: MNCR 0134-94 identified a nonconformance in which Design Drawing M-0720, Revision 9 depicted bathroom fixtures that do not exist. These changes are necessary to reflect existing plant configuration on the applicable design drawings.

SAFETY EVALUATION: The piping and equipment in their existing state will not create or affect any safety limits, safety system settings or limiting conditions. Therefore, no technical specification changes will be required.

Implementation of the changes as described in the disposition of MNCR-0134-94 will not increase the probability of occurrence or consequences of an accident previously evaluated in the UFSAR. The affected components are non-safety related and their failure will not compromise any safety related system or components or prevent a safe shutdown.

Since the piping and equipment involved is non-safety related and is currently serving no function, their operation is not required to mitigate the occurrence or consequences of a malfunction equipment important to safety. Therefore, the probability of occurrence or consequences of a malfunction of equipment important to safety is not increased by disposition of MNCR 0134-94 to accept the subject piping and equipment in its existing state and consider it abandoned.

No new failure modes are created and the possibility for an accident or malfunction of equipment of a different type than any previously evaluated in the UFSAR is not created. The margin of safety as defined for any technical specifications is not changed and no technical specifications are affected.

Serial Number: 95-009-NPE Document Evaluated: DCP 91/0088-1, Rev. 1
SCN 93/0022C to JS-08

DESCRIPTION OF CHANGE: This change replaces panels 1H22-P171 and P172 with a microprocessor based distributed control system. This system is an exact replica of the existing control strategies with two exceptions. The condensate and condensate booster pump min-flow logic is being modified to include time delays and multiple trip setpoints. Also, one pump (instead of all three) will be tripped at a time. The feed pump trip logic will be modified to have multiple setpoints and delays. Some instrumentation changes will be made. The feed pump suction pressure low switches will be changed to a new model that will be able to handle the pressures at the feed pump suction (reference MNCR 94/0012). The MSR drain tank level transmitters level taps will be moved to provide more reliable system operation.

REASON FOR CHANGE: The change consists of modifying the condensate and condensate booster pump min-flow trip logic to allow operator intervention by adding a time delay in the trip circuit. Present design results in immediate trip of all pumps during a low flow condition. The new system allows the operators the opportunity to correct a situation which caused a min-flow condition by providing Control Room indication on the 1H13-P680 panel that the trip timer has started, and then by tripping the pumps in a staggered fashion. Changes to the RFP recirculation control valve controller and feed pump trips will also help Operations during these situations.

SAFETY EVALUATION: This change will not affect the technical specifications, the bases for any technical specifications, or the operating license. The values and bases for the minimum critical power ratio (MCPR) operating limits are not affected and the feedwater and condensate system equipment and instrumentation involved are not explicitly covered by the technical specifications.

This change does not increase the probability for occurrence of accidents or malfunctions of equipment important to safety previously evaluated in the UFSAR because equipment operating characteristics meet established design requirements and reliability is enhanced by the proposed changes. Failures of affected penetrations are also no more likely. To maintain integrity during tornado depressurization, penetration seal details provided for these penetrations also ensure the integrity of the 3 psi pressure boundary. Operational considerations have also been provided to ensure compliance while performing this work. Additionally, operational considerations have been provided to ensure that the integrity of the Control Room envelope as defined in UFSAR 6.4.2 is not jeopardized and to ensure that the Control Room leak rate contained in Operating License Condition 2.C.38 is not exceeded.

This change will not increase the consequences of an accident or malfunction of equipment important to safety from that previously evaluated in the UFSAR because appropriate design requirements and operational considerations have been provided to ensure that equipment performance remains within the limits currently assumed in existing accident analyses. This change will also not create the possibility for an accident or malfunction of equipment important to safety of a different type than any previously evaluated in the UFSAR because limiting system failure modes are unchanged and additional challenges to safety features or equipment important to safety will not occur. This change will not reduce the margin of safety as defined in the bases for technical specifications since limiting and non-limiting events which may affect fission product barriers remain clearly bounded by existing analyses, and penetration operability and integrity remain consistent with that assumed by the technical specifications.

Serial Number: 95-010-NPE Document Evaluated: MNCR 0174-94

DESCRIPTION OF CHANGE: The affected FSKs and P&IDs did not agree with as-built conditions. The FSKs were updated to show the tagged valve number on the affected valves. Supplement 1 attached to MNCR 0174-94 states that System Engineering review has determined that the P&ID designations for these valves are correct and do not need to be changed. Therefore, as a result of the FSK changes P&ID M-0039X was updated to show the branch line between Valves FA75 and FA76. P&ID M-0040A was updated to show the appropriate line class and coordinate in their designated area.

REASON FOR CHANGE: The affected FSKs and P&IDs did not agree with as-built plant conditions.

SAFETY EVALUATION: This change will not result in any operational or functional changes to the G17 Liquid Radwaste System or impact the operation of any other system. No technical specification requirements are changed or added as a result of this design change.

Consequently, no changes to the technical specifications are required.

The described changes will not alter or affect the operability of any safety-related equipment or systems. Failure of the G17 system will not compromise any safety related systems or prevent safe shutdown, and the evaluated design changes will not increase the probability or consequences of an accident previously evaluated.

Revising design drawings will not alter the design, function, or operation of any equipment important to safety as evaluated in the UFSAR. The Liquid Radwaste System serves no safety-related function and the modifications will not compromise any safety-related systems since no new interfaces with equipment important to safety is created nor is such equipment prevented from operating as designed.

This drawing change involves editorial changes and modification of P&IDs to show plant conditions and will not create the possibility of an accident of a different type than previously evaluated in the UFSAR. This equipment serves no safety function, and its use will not create any new failure modes.

No change to the overall radwaste process is introduced, and the modification does not reduce any margin of safety as defined in the basis for any technical specification.

Serial Number: 95-011-NPE   Document Evaluated: DCP 91/0042, Rev. 0

DESCRIPTION OF CHANGE: Replacement of the small Lonergan relief valves in the SSW system with a valve design which incorporates the following features: 1) Valve design resistance to corrosion. The new valve design uses stainless steel for the valve body and internal metal parts, 2) The SSW system is typically operated in a manner such that system operating pressures are near relief valve setpoints during system startup/shutdown. The new design provides for a soft seat which should reduce problems that result from system operation such as valve leakage and debris (silt/sand) on the valve seats, and 3) Ability of the valve design to handle the kind of debris found in the GGNS SSW system (notably silt/sand). The spring chamber of the new valve design is effectively sealed off from the discharge flow so as to prevent debris from accumulating around the valve spring and disk guide.

The replacement SSW relief valve installations are also to be configured so as to meet the ASME code, Section III, Paragraph ND-7155 requirement that liquid pressure relief valves on isolable components not be less than 3/4" nominal pipe size.

REASON FOR CHANGE: The presently installed small thermal (Lonergan) relief valves in the Standby Service Water System are susceptible to a number of problems related to poor valve design, insufficient product support by the vendor, and lack of suitability for the conditions found in the SSW system. Relief valves of a different design are required to help solve these problems.

SAFETY EVALUATION: The installation of replacement valves in place of the small Lonergan thermal relief valves and changes in the configuration of the piping installations for these valves will not impact plant safety. The replacement relief valves are to be designed and built to the requirements of the ASME code, Section III, 1974 Edition as were the original valves. The reconfigured piping installations will meet all of the original design requirements for the system.

Serial Number: 95-012-NPE Document Evaluated: UFSAR CR 94-016

DESCRIPTION OF CHANGE: UFSAR Sections 6.2.5.2.a and 6.2.5.2.b are to be revised to indicate that drywell vacuum relief is provided in either a small or large break LOCA.

REASON FOR CHANGE: The subject UFSAR sections presently imply that the drywell vacuum relief is provided by the two drywell post-LOCA and the two drywell purge vacuum relief subsystems only for a large break LOCA. It is also implied that the normal drywell vacuum breaker is used for a small break LOCA. These statements are misleading and are in conflict with UFSAR Sections 6.2.5.1.1.r and 6.2.5.2.1 that provide the correct information concerning the design functions of the drywell vacuum breakers.

SAFETY EVALUATION: The drywell vacuum relief system is designed to prevent weir wall overflow following a small break LOCA. The system is also required to control rapid weir wall overflow following a large break LOCA to limit drag and impact loads on drywell equipment and structures. The change to the UFSAR corrects inaccuracies that imply that the system functions only during a large break LOCA. The change does not alter the design or operation of any system or component or affect the analysis of any event described in the UFSAR. The change does alter or introduce new malfunctions in any system or component. The drywell vacuum relief system will continue to function as designed and evaluated and will not result any new events or malfunctions that could result in plant condition that are not analyzed. Because the system will continue to function as designed, evaluated and correctly described elsewhere in the UFSAR no safety margin will be reduced.

Serial Number: 95-013-NPE   Document Evaluated: UFSAR CR 94-019

DESCRIPTION OF CHANGE: UFSAR Table 5.4-3 is to be revised to add valves E12-F082A/B and E12-F290A/B and their associated function, normal position, and logic and/or permissive.

REASON FOR CHANGE: The justification for this change is to maintain an accurate reflection of the plant design in the UFSAR. UFSAR Section 5.4.7.2.1 states that "A tabular description of RHR system logic (i.e., interlocks, permissives) is given in Table 5.4-3". There are interlocks and permissives associated with E12-F[illegible]A/B and E12-F290A/B; therefore, they should be included in Table 5.4-3.

SAFETY EVALUATION: This change adds information to the UFSAR to provide a more accurate reflection of current plant design. The change does not alter the design or operation of any system or component or affect the analysis of any event described in the UFSAR. The change will not increase accident or malfunction probabilities or consequences. The change will not create any risk of a different type of accident or malfunction and does not reduce a margin of safety as described in the technical specification bases.

Serial Number: 95-014-NPE Document Evaluated: DCP 91/0039 Rev. 0 and DCP 91/0039 S1 Rev. 0

DESCRIPTION OF CHANGE: The safety related pneumatic system on the airlocks is redesigned to eliminate as many potential leakage sources as possible. New instruments and mechanical components or are designed to replace the corresponding obsolete components. The new air system is designed to eliminate unnecessary, redundant instruments, tubing and fittings by utilizing one air accumulator in each containment airlock per door and two air accumulators in parallel in the drywell airlock per door. Both door seals and both door latch cylinders on each containment door will be operated on one air accumulator in lieu of the current configuration of one door seal and one door latch cylinder on each door utilizing one air accumulator.

The configuration of the pneumatic system is modified by changing the door latch cylinders current configuration of "air-to close/spring-to-open" to "spring-to-close/air-to-open". The existing door latch cylinders is modified with conversions kits.

The new components in the safety related pneumatic system, including the air regulating valves, check valves, flow control valves, seal inflation valves (4-way clevis operated valves), equalizing valves (3-way clevis operated valves) and pressure switches have been dedicated to have less than 0.4 standard cubic centimeters per minute (sccm) leakage.

REASON FOR CHANGE: These design changes provide the requirements for the modification of the upper and lower containment airlocks and drywell airlock safety related pneumatic system. During refueling outages, the safety related pneumatic system which operates the airlock door seals and door latch air cylinders requires a tubing drop test to ensure the integrity of the system. This tubing drop test, which is required by the GGNS Technical Specification, has been getting more difficult to pass each outage due to excessive leakage at the tubing fittings and components. Also, the instruments and mechanical equipment in the system are obsolete and can not be replaced with "like-for-like" components.

SAFETY EVALUATION: The modification of the pneumatic system will meet the original design requirements. The tubing modification reduces the overall length of the tubing and reduce the number of fittings in the tubing, thus reducing the potential leakage sources in the pneumatic system. The two door air seals of at least one door will be supplied with air maintained at a minimum pressure of 60 psig for a minimum of 30 days even with the single failure of one of the redundant air accumulators. This new configuration will still maintain two seals per airlock with the potential of the air accumulator or tubing failure. The ultimate pressure capacity of the airlocks will not be affected by the modifications; therefore the pressure integrity of the containment or drywell will not be degraded. No surveillance or testing requirements in the Improved Technical Specifications for the airlocks will be changed based on the implementation of the airlock modifications nor will any basis for the Improved Technical Specifications be affected. The probability or consequences of an accident or malfunction of equipment previously evaluated in the UFSAR has not increased and the possibility of an accident or malfunction of equipment important to safety different from any previously evaluated in the UFSAR will not occur with the implementation of these DCPs.

Serial Number: 95-015-NPE   Document Evaluated: N/A (Cycle 8 Fuel Receipt)

DESCRIPTION OF CHANGE: This safety evaluation addresses the issues concerning receipt of the Cycle 8 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). Since this fuel contains trace isotopic impurities, the impact of these isotopes on handling and operation of this fuel is also addressed. Attachment 1 addresses each of these issues in more detail.

REASON FOR CHANGE: Reload fuel is necessary for Cycle 8 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, (iii) the reload batch can be safely moved to and stored in either the new fuel vault or spent fuel pool, and (iv) no adverse consequences arise due to isotopic impurities in the fuel.

Serial Number: 95-016-NEAD   Document Evaluated: Letter, GE to EOI, "Contract No. G-1160-C001", 4/7/94, EDC File QR-021-13

DESCRIPTION OF CHANGE: All the GGNS fuel channels supplied in the past have used Zircaloy-4 material with its usage stated in the UFSAR. However, the Cycle 8 reload (and probably near-term future cycles) will use channels with Zircaloy 2. While the Cycle 8 reload channels have been reviewed by the vendor and determined to be the equivalent of the original core channels (Reference 1), the material change requires a UFSAR change and hence a safety evaluation. More information on the Cycle 8 and earlier fuel channels is provided in Section III.A.

REASON FOR CHANGE: The Cycle 8 reload channels will be supplied by GE which several years ago changed their standard channel from Zircaloy 4 to Zircaloy 2. The use of Zircaloy 2 channels is expected to yield greater corrosion resistance and reduced hydrogen pickup (and hence less embrittlement at high burnup). More information on the relative performance of these two materials is provided in Section III.B.

SAFETY EVALUATION: The use of Zircaloy 2 will not affect the channel's mechanical, neutronic, or thermal-hydraulic performance nor will it affect the performance of the fuel bundles, control blades or incore instrumentation. Slightly less channel corrosion and hydrogen embrittlement are expected using Zircaloy 2 relative to Zircaloy 4. The use of Zircaloy 2 channels will not result in an unreviewed safety question.

Serial Number: 95-017-HP Document Evaluated: UFSAR Section 11.2.1.1, Tables 1.3-5, 11.2-10 & 11.3-9

DESCRIPTION OF CHANGE: A temporary onsite laundry facility has been found to be cost beneficial for RF07. However, the UFSAR currently limits GGNS use to an offsite commercial laundry facility. This evaluation addresses the use of temporary onsite laundry services. These facilities are described in detail in Attachment 1.

REASON FOR CHANGE: This change will allow GGNS to utilize a licensed commercial laundry to handle contaminated protective clothing onsite during outage periods. GGNS does not intend to provide a fixed laundry facility that would require discharge of liquids to the environment. The utilization of a temporary onsite laundry facility will allow greater flexibility with protective clothing inventory control and decrease labor requirements for several disciplines at GGNS.

SAFETY EVALUATION: This safety evaluation concludes that this change does not involve an unreviewed safety question. The change will not increase the potential of an accident nor will it result in the malfunction of plant equipment. GGNS has evaluated the placement of this temporary facility and certain limitations will be imposed to assure that the issue of safety, effluent and site radiological controls are maintained at the current high standards.

Serial Number: 95-018-NPE Document Evaluated: DCP 88/0019 Rev. 0

DESCRIPTION OF CHANGE: The tank mixing spargers of the SRT and CPSTs will be modified to allow for improved mixing of slurries in these tanks. In addition, condensate flush connections will be provided on the slurry sample stations of the SRT, CPSTs, and RWCU PSDTs.

REASON FOR CHANGE: The purpose of this change is to alleviate the clogging of recirculation loop piping on the following Liquid Radioactive Waste (G17) System waste collector tanks:

- Spent Resin Tank (SRT) NSG17A007,
- RWCU Phase Separator/Decay Tanks (PSDT) NSG17A010A/B, and
- Condensate Phase Separator Tanks (CPST) NSG17A016A/B.

SAFETY EVALUATION: Since this modification does not render any change to the process by which liquid radwaste is treated before offsite release, the requirements of Technical Specification 6.15, "Major Changes to Radioactive Waste Treatment Systems" are not applicable.

This change improves the slurry mixing and recirculation loop flushing capabilities of the Liquid Radwaste System waste collector tanks and slurry design requirements are maintained by this modification. Therefore, there is no increase in the probability of a previously evaluated equipment failure, no new effluent release scenarios are postulated.

This change improves the capabilities of the Liquid Radwaste System to perform required functions. The affected components do not perform any functions related to safety. The possibility of a malfunction of equipment important to safety is thus not created. No change to the overall radwaste process is introduced, thus the commitments to ALARA and GDC 60 of 10CFR50 Appendix A continue to be met.

Serial Number: 95-020-NSRA Document Evaluated: ITS TRM Change

DESCRIPTION OF CHANGE: This change will remove the requirements for operability of the high pressure core spray (HPCS) line break instrumentatio: The instrumentation is designed to provide an indication in the control room of a break in the HPCS line between the inside of the reactor vessel and the core shroud. The instrumentation is for indication only and does not provide any advance indication of a pending HPCS line break, does not initiate any plant equipment, and the receipt of the alarm in the control room requires no compensatory operator action.

REASON FOR CHANGE: This alarm must be calibrated at normal operating temperature and pressure. The alarm, therefore, is annunciated during the entire startup process. This alarm, though minor, is contrary to the "blackboard" concept of startup procedures. A more important reason for the change is to reduce personnel resources required to manually collect the calibration data during startup. Due to the "alarm-only" function of the instrument, and the plant resources expended on demonstrating and maintaining operability of the instrument, continued required operability of this instrument is unwarranted.

SAFETY EVALUATION: The proposed change is to an instrument which provides an alarm only in the control room. That this alarm is not part of the operability requirements for the HPCS system was identified in submittals supporting the Improved Technical Specifications program which has received NRC approval (TS Amendment 120). The alarm does not provide any advance indication of an HPCS line break, does not initiate any plant equipment, and does not require any compensatory operator action. The monitored section of pipe is a no-break zone, therefore failures of this pipe are not credible events. There has never been any GE or NRC requirement for any direct or automatic action safety function to be initiated by this break detection feature nor was any such direct safety action intended by its design. The ECCS are initiated and controlled by separately monitored parameters, therefore ECCS operation would not be affected by this change. For these reasons, this change does not represent an unreviewed safety question.

Serial Number: 95-021-NPE Document Evaluated: MCP 92/1049 Rev. 1 and CN 95/0001

DESCRIPTION OF CHANGE: A time delay relay set for one second will replace the existing E12A-K19A relay in panel 1H13-P629 at the output of Optical Isolator E12A-AT7. The time delay relay will be set to provide sufficient delay to allow electrical transients to decay without inadvertently energizing relay E12A-K19A.

REASON FOR CHANGE: MNCR 0001-92 documented that the performance of a monthly I&C procedure which initiates the RCIC trip/isolation relay has caused RHR "A" pump E12-C002-A to trip on occasion. The cause of this condition has been identified by the detection of noise pulses (electrical transients) which are generated by the changing state of the RCIC trip/throttle valve dual coil solenoid and are propagated into Optical Isolator E12-AT7 via relay E51A-K8. MNCR 0001-92 was dispositioned "Repair per MCP 92/1049." This change provides the design to replace the existing E12A-K19A relay with a time delay relay set for one second. The addition of the intentional one-second time delay, via a Class 1E time delay relay, at the output of Optical Isolator E12-AT7 (RHR "A" pump suction loss trip function) will ensure that energization of the E12A-K19A relay is delayed sufficiently to allow electrical transients to decay and thereby prevent inadvertent/spurious tripping of RHR "A" pump E12-C002-A due to electrical transients.

SAFETY EVALUATION: The changes addressed by this evaluation will not compromise any existing safety related system, structure, or component nor will they prevent safe reactor shutdown. No evaluated accident is initiated by a failure of the affected relay (E12-K19A) or operation of the RHR "A" shutdown cooling pump suction protection. The addition of a one-second time delay in the RHR "A" logic will not initiate any evaluated transient or accident. Operation of function of the E12 (RHR) system will not be changed other than as previously discussed due to the addition of the Class 1E time delay relay.

The E12A-K19A relay is not required to mitigate the consequences of any evaluated transient or accident, other than not compromising those modes of RHR that provide such functions. Use of a Class 1E relay for the subject replacement ensures that failures remain bounded by the single failure criteria. No new unbounded interfaces are created and no unacceptable failure modes are introduced. These changes therefore do not introduce any unreviewed safety question and the existing technical specifications for the RHR are not impacted.

Serial Number: 95-022-OPS   Document Evaluated: Offsite Dose Calculation Manual Revision 17

DESCRIPTION OF CHANGE: This change incorporates editorial changes consistent with Grand Gulf Nuclear Station's implementation of Improved Technical Specifications (ITS). The revision also contains technical changes updating parameters (atmospheric dispersion/deposition, critical offsite receptors) used in the calculation of offsite doses from routine, long term releases of gaseous radioactive materials. UFSAR CR 94-044 is an editorial change to reference the source document (EER 93/6246) for meteorological parameters used in the calculation of doses resulting from routine long-term releases. The periodic update of these parameters is a routine activity and improves the accuracy of offsite dose calculations.

REASON FOR CHANGE: To provide consistency with changes to the technical specifications resulting from ITS, ODCM radiological effluent specifications (also located in the Technical Requirements Manual) are being revised. The ITS changes deal primarily with the format of the specifications and amount of Bases information associated with the specification. Several additional changes to the ODCM are proposed which will result in more accurate gaseous effluent dose calculations. The parameters and effects include: updating atmospheric dispersion/deposition parameters, moving the 10CFR50 Appendix I controlling receptor from an onsite garden to the limiting offsite receptor, eliminating the grass/cow/milk ingestion pathway from 10CFR50 Appendix I dose calculations. Results of GGNS' 1994 land use census were considered for changes to controlling receptor location and ingestion pathways.

SAFETY EVALUATION: The editorial changes in ODCM Revision 17 will provide consistency with the Improved Technical Specification format. The technical changes (dose calculation parameters) will provide more accurate estimates of offsite gaseous doses. The net effect of the technical changes on doses resulting from routine long-term releases of gaseous radioactive materials will be an increase noble gas and tritium dose and a decrease in radioactive iodines and particulate doses. GGNS' previous releases have been well below 10CFR50 Appendix I limits. GGNS' worst calendar quarter release data was reviewed against the proposed changes and doses remain well below 10CFR50 Appendix I limits.

Serial Number: 95-023-NSRA Document Evaluated: ITS TRM Revision 6

DESCRIPTION OF CHANGE: This change deletes the commitment to perform logic system functional tests (LSFTs) on the following Reactor Protection System (RPS) and isolation actuation instrumentation functions: the Main Steam Line Radiation Monitor (MSLRM) - RPS and isolation actuation, Average Power Range Monitor (APRM) flow-biased simulated thermal power - RPS, and Main Steam Line (MSL), Reactor Core Isolation Cooling (RCIC), Reactor Water Cleanup (RWCU), and Residual Heat Removal (RHR) area differential temperature - isolation actuation. This evaluation only effects those portions of the associated systems' logic used by the evaluated function. Any LSFT requirement on portions of the logic required by the technical specifications is not affected by this change.

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

SAFETY EVALUATION: The MSLRM is not credited for a reactor scram initiation for any Updated Final Safety Analysis Report (UFSAR) analyzed event. In addition, a dose analysis for control rod drop accident has been performed without taking credit for the isolation of the main steam lines on a MSLRM high radiation signal. The resulting consequences are much lower than the regulatory limits of 10CFR100 values as applied in SRP 15.4.9 and General Design Criteria (GDC) 19.

The MSLRMs also provide signals to the Group 10 isolation valves and the mechanical vacuum pumps. The MSLRMs are not assumed to provide the signal to isolate these valves during events which include drywell pressurization. As a result, the MSLRMs are not required by the accident analyses to isolate these valves in the event of an accident. Although the tripping of the mechanical vacuum pumps is an assumed function of the MSLRMs during the design basis control rod drop event, the technical specifications never required this function to be LSFT tested. Therefore, deleting the commitments regarding LSFT requirements for the MSLRMs does not affect any testing associated with the tripping of the mechanical vacuum pumps.

The APRM-FBSTP scram is not assumed to function to provide a scram signal in any safety analysis. The only relevant event is loss of feedwater heating (LFWH). For this event the APRM-FBSTP scram can trip before the neutron flux scram because of its lower power setpoint. Since Cycle 1, this scram signal has not been used by GGNS in the analysis. The initial analysis methodology has been replaced by an approved methodology which examines steady-state conditions before and after a LFWH event. With this analysis there is no need for any scram for the event.

Differential temperature monitors are installed in certain areas containing high energy lines. None of the differential temperature instruments is assumed to initiate automatic isolation of any pipe rupture. Redundancy is provided in the various isolation parameters to satisfy single failure criteria.

The only radiological evaluation found in Chapter 15 for MSL, RCIC, RWCU, and RHR systems is that for the main steam line break (MSLB). Isolation of the MSLB is assumed at 5.5 seconds due to high steam line flow. Consequently, elimination of automatic isolation upon high differential temperature would not affect radiation dose estimates for this event.

Engineering Report GGNS-90-0024 shows that the design basis mass releases would not be increased following a pipe rupture in the RCIC, RWCU, and RHR systems if the differential temperature isolation instrumentation did not function. Therefore, no increase in the radiological consequences will occur as a result of the deletion of LSFT surveillance requirement for differential temperature monitors.

None of the functions for which the requirement to perform LSFT testing is being deleted is assumed of function to mitigate the consequences of any UFSAR analyzed event. Deleting this requirement does not increase the probability of spurious actuation of the evaluated functions or introduce any new system operating modes. In addition, none of the evaluated functions are contained in the technical specifications. Therefore, deleting this surveillance requirement does not require a change to the technical specifications or result in an unreviewed safety question.

Serial Number: 95-024-PSE Document Evaluated: TSTI 1B33-95-002-0-S Recirc "A" Purge Flow Test

DESCRIPTION OF CHANGE: This change is generated to reduce purge flow to zero flow and then restore to normal purge flow to Recirculation "A" pump seals as part of a troubleshooting effort to determine the effects of purge flow on the present anomaly of vibration and seal temperature oscillations. That test was conducted with inconclusive results because adjustment of regulator 1C11D012A resulted in a reduction of purge flow to only 1.6 gpm instead of zero or near zero as expected (the regulator adjusting knob bottomed out with no further reduction in flow). Because the test did not yield the expected results, the test was stopped until the results could be evaluated and an action plan developed.

This evaluation reviews a second and subsequent test, this change which is a continuation of the first test, to reduce purge flow to zero after the regulator minimum flow is reached, using manual isolation valves B33F026A and B33F024A. The first test (95-001) assumed a slow stepped reduction in purge flow in 0.1 gpm increments. With the knowledge gained from the first test, this test evaluates the potential for a less than slow reduction in purge flow, once the lower limit of the regulator is reached.

REASON FOR CHANGE: This test is needed as part of a troubleshooting effort to determine and eliminate the vibration and seal temperature oscillations that are currently present on Recirculation Pump "A".

SAFETY EVALUATION: Operation with no purge flow has already been approved and documented on safety evaluation CFMRISC0118R00. The conclusion of that evaluation was that no technical specification changes were required and that no unreviewed safety question arises as a result of operation without purge flow. This change is necessary because CFMRISC0118R00 evaluated only for operation without purge flow. This additional evaluation is necessary to address changing purge flow and then restoring purge flow (if required) with a recirculation pump in operation.

The test will reduce purge flow in a controlled manner to ensure that seal temperatures are maintained less than 180°F, and to ensure that no new temperature or vibration transients are created as a result of securing or restoring purge flow. The controlled reduction in purge flow, while monitoring pump and seal temperature and vibration parameters, along with the fact that the seals are designed by the vendor (AECL) to operate without purge flow, will ensure that no abnormal conditions or system transients are created. The worst event postulated because of the changing purge flow is a complete loss of both #1 and #2 Recirculation Pump "A" seals. This event, loss of both #1 and #2 seals on one recirculation pump, is well within the bounds of UFSAR Chapter 15 analysis of a DBA LOCA.

Restoration of purge flow, in a controlled manner with pauses to observe system affects, is also not expected to result in any system transients or abnormal conditions. The slow restoration will ensure no thermal transients are produced in the pump shaft or heat exchanger region, and ensure no oscillations are initiated.

Chapter 15 accident analyses involving the recirculation pump are pump trip, pump seizure, and pump shaft break. The probability of occurrence of any of these events is no more likely by restoration of purge flow because of: 1) The modification implemented in RFO6 which installed a sacrificial shaft sleeve, 2) The controlled method of re-introduction of seal flow while monitoring pump vibration and seal temperatures, and 3) Seal flow maximum is a reduced flow of 1.7 to 2.0 gpm as recommended by GE in SIL 511. Note: Our operating procedure maintains 1.5 to 2.0 gpm which is more conservative.

By reducing purge flow to zero, restoring to normal flow or operating without purge flow does not impact technical specifications or result in any unreviewed safety question.

Serial Number: 95-025-NPE Document Evaluated: GGNS-MS-25.0, Rev. 9

DESCRIPTION OF CHANGE: Revision 9 to Standard GGNS-MS-25.0, Mechanical Standard for Motor Operated Valve Torque and Limit Switches, contains the changes summarized in Items 1 through 8 below:

1. Incorporates Standard Change Notices 93/0001, 93/0002, 94/0001, 94/0002, and 94/0007.

2. Removes valves Q1P41F007A/B, Q1P42F028A, and Q1P42F032A from the scope of the Generic Letter (GL) 89-10 program.

3. Adds Paragraph 5.6 to the standard to provide the minimum torque for actuator-to-valve mounting fasteners for valves which are justified to operate at thrust values in excess of current actuator design thrust limits.

4. Revises the maximum allowable stem thrust (MAST) and/or the limiting component stress allowable thrust (LCSAT) for valves Q1B21F016, Q1B21F019, Q1E12F024A/B, Q1E12F028A/B, Q1E51F022, Q1G33F001, Q1G33F004, Q1G33F028, Q1G33F034, Q1G33F039, Q1G33F040, Q1G33F250, Q1G33F251, Q1G33F252, Q1G33F253, Q1G41F029, Q1G41F044, Q1P42F114, Q1P42F116, and Q1P42F117.

5. Revises the degraded voltage actuator capability (DVAC) torque limits for valves with AC motors using new methodology to account for the effects of elevated ambient temperature on motor torque in response to Limitorque Corporation's 10CFR21 Notification and Technical Update 93-03.

6. Revises the minimum required stem thrust (MRST) in Appendix J for valves Q1E12F004A-C, Q1E12F006A/B, Q1E12F024A/B, Q1E12F066A/B, Q1E12F094, and Q1E12F096 to incorporate the 0.64 valve factor determined to be bounding by valve grouping analysis for valve family POW 300GA2.

7. Revises the maximum expected differential pressure (MEDP) and the minimum required stem thrust or torque (MRST) for valves in the following systems: B21, B33, C11, C41, E21, E22, E51, G33, P41, and P42.

8. Revises the MEDP and MRST for valve Q1E22F004.

REASON FOR CHANGE: The changes represented by Item 1 have previously been reviewed and will not be discussed in this safety evaluation. The remaining changes summarized above (Items 2-8) are a result of the continuing evolution of the GGNS motor operated valve program which was established in response to Generic Letter (GL) 89-10, Safety Related Motor Operated Valve Testing and Surveillance, and its supplements. NPE has concluded that select valves originally in the GL 89-10 program can be removed from the scope (Reference Engineering Report 91-0052, Rev. 2). Industry sponsored testing has shown that certain sizes of Limitorque actuators can be operated under higher than rated thrust conditions. This permits an increase in the maximum allowable stem thrust and provides a larger thrust "window" for establishing torque switch settings (Reference Calculation MC-Q1111-91123, Rev. 11). Limitorque Corporation's 10CFR21 Notification and Technical Update 93-03, concerning the effect of elevated temperature on motor torque output for AC motor actuators, resulted in a reevaluation of degraded voltage actuator capability for all AC actuators and a recalculation of DVAC torque limits (Reference Calculation MC-Q1111-93035, Rev. 4). Grand Gulf's valve grouping methodology resulted in the conclusion that select valves would require greater stem thrust than that predicted using the GGNS standard valve factor of 0.5 in order to perform their design basis functions. This required a recalculation of the MRST values for these valves (Reference MC-Q1111-91132, Rev. 7). GL 89-10 Supplement 4 permitted licensees of BWR plants to remove the consideration of valve mispositioning as a criterion in design basis reviews of GL 89-10 MOVs. This resulted in new MEDP values for some of the program MOVs and subsequent recalculation of corresponding MRST values. Of all the valves included in the scope of MS-25, only one (Q1E22F004) has a MEDP that is specifically cited in the UFSAR. An increase in the calculated MEDP for valve Q1E22F004 (Reference NPE Calculation MC-Q1E22-93043, Rev. 0, resulted in a change to the UFSAR.

SAFETY EVALUATION: The use of the MOV MEDP, MRST, MAST and DVACT parameters included in Revision 9 to GGNS-MS-25.0, will ensure that each valve within the scope of the GL 89-10 MOV program will perform its design basis function(s). The change to the MEDP value and the valve component stresses for valve Q1E22F004, which are summarized in UFSAR Table 3.9-2ac, have been evaluated and determined to be acceptable. Change Request 94-045 has been initiated to change the UFSAR. The changes included in Revision 9 to GGNS-MS-25.0 do not modify plant components, systems or structures. There are no unreviewed safety questions, technical specification changes or reductions in the margins of safety as defined by the bases for any technical specification.

Serial Number: 95-026-NSRA Document Evaluated: TRM Section 7.6.3.10.3.e

DESCRIPTION OF CHANGE: Safety Evaluation 94-0106-R00 will allow functional testing of accessible snubbers to begin no sooner than 72 hours prior to shutdown for refueling. Previously, the Technical Requirements Manual, Section 7.6.3.10, "Snubber Program", required that functional testing of snubbers occur "...at least once per 18 months during shutdown".

REASON FOR CHANGE: This change will reduce outage burden and could result in dose reduction for those systems where dose increases when they are open for outage activities and for systems used during shutdown (such as SDC).

SAFETY EVALUATION: Research of the UFSAR and technical specifications indicates that there is no basis to not allow functional testing of accessible snubbers to begin no sooner than 72 hours prior to shutdown for refueling. PRA group compared the snubber testing scheme planned for this 72 hour period with PRA guidance for removing equipment from service and provided the following guidance: 1) When a snubber associated with a system in the left-hand-side column ("System/Train Important to PRA") in the guidance table is removed, snubbers associated with any of the systems/trains in the corresponding right-hand-side column ("Systems to Avoid Placing Out of Service Concurrently") **will not be removed for testing**, or, 2) When a system or train is already out of service (due to reasons other than snubber testing), then snubbers associated with the corresponding right-hand-side column **will not be removed for testing.** No restrictions are placed, from the PRA insights, on snubbers associated with systems which do not show up on the above guidance table.

This guidance will be followed during the subject 72 hour period at power operation; therefore, the risk profiles during this 72 hour period will be consistent with those during normal operation, and are therefore acceptable. It should be noted that removal of a snubber for testing does not mean that the system will not function during an accident; it means that the response of that system during a seismic event will be degraded. The likelihood of a seismic event of a severe enough magnitude to disable the system is negligible.

This change in timing to begin functional testing will not revise the actual testing method, nor will it change the number and type of snubbers to be tested. As long as functional testing is conducted in accordance with Administrative and Surveillance procedures, system integrity will not be compromised and no unreviewed safety questions will exist. Compliance to procedural guides and specifications governing the testing of snubbers will ensure that system integrity is not compromised.

Serial Number: 95-027-NSRA   Document Evaluated: IRM Control Rod Block Input

DESCRIPTION OF CHANGE: This change allows the temporary bypassing of more than one IRM input per trip system into the control rod block logic while in Mode 2. The evaluated change has no effect on the allowable number of IRM inputs into the RPS logic.

REASON FOR CHANGE: Allow the temporary jumpering of an IRM input in addition to an input already bypassed in the trip system to permit plant startup under the enforcement of LCO 3.3.1.

SAFETY EVALUATION: As discussed in the NRC's SER for Technical Specification Amendment 109, the control rod blocks provided by the IRMs in Mode 2 are provided to actuate to prevent IRM or APRM scrams by the RPS. Accordingly, these control rod blocks function to act as backups to the IRM and APRM scrams and are not credited in any design basis transient analyses for Mode 2. Defense-in-depth in this mode is also provided by SRM, APRM and rod pattern control system rod blocks, procedural controls on rod withdrawal sequences, core reload analyses, and the RPS scrams.

Adequate neutron monitoring of core for the transient analyses is still provided by the RPS trip system with only one bypassed IRM input since the RPS trip system with more than one inoperable IRM will be in trip. The Technical Specification Amendment 120 bases for the required action for inoperable RPS inputs discusses plant operation following placing a trip system in trip and states that placing the trip system or channel in trip conservatively compensates for the inoperability and restores capability of the system to accommodate a single failure. Additionally, the operable IRMs will continue to have the ability to initiate control rod blocks while an additional IRM input is bypassed.

The IRM control rod block in Mode 2 is not credited in any design basis transient analyses and the number of IRM inputs which can be bypassed is not contained in the technical specifications.

Therefore, the evaluated condition does not require a technical specification change or result in an unreviewed safety question.

This change does not relax the TRM requirements to restore the number of IRM inputs to the minimum number required within 7 days or put in place a control rod block in effect.

Serial Number: 95-028-NPE Document Evaluated: N/A (RF07 Core Shuffle)

DESCRIPTION OF CHANGE: During RF07, 288 depleted Siemens 9x9-5 fuel assemblies will be replaced with fresh 9x9-5 fuel assemblies. These new assemblies have been designed and built specifically for the Grand Gulf Cycle 8 core and are currently stored in the Auxiliary Building. This change addresses the RF07 core shuffle and considers the following items:

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

b. storage of fresh and irradiated fuel in the containment pool during RF07;

c. storage of irradiated discharge fuel in the spent fuel pool during RF07 and Cycle 8; and

d. shutdown margin for all interim RF07 core configurations and the final Cycle 8 core loading in Modes 4 and 5.

REASON FOR CHANGE: Cycle 8 operation requires the addition of fresh fuel assemblies and the removal of depleted assemblies from the reactor vessel. Actual Cycle 8 operation with these fuel assemblies will be assessed in an upcoming safety evaluation.

SAFETY EVALUATION: This change concludes that (i) a fuel handling accident during RF07 will not result in doses above the allowable limits, (ii) the criticality acceptance criteria for fuel pools are satisfied through Cycle 8, and (iii) the acceptance criterion for shutdown margin is satisfied for interim RF07 and final Cycle 8 core configurations during Modes 4 and 5.

Serial Number: 95-029-NPE   Document Evaluated: Engr Rpt GGNS-95-0001 Rev. 0

DESCRIPTION OF CHANGE: This change addresses natural circulation core cooling during Mode 5 high water level conditions. This mode of core cooling has been evaluated in the reference and the results are reported in the evaluated Engineering Report GGNS-95-0001. The requirements and system lineups necessary for natural circulation are also described in this report.

REASON FOR CHANGE: Natural circulation core cooling will provide an alternate method of decay heat removal and coolant circulation in the event the normal systems (RHR shutdown cooling and ADHRS) are unavailable.

SAFETY EVALUATION: This change concludes that core cooling via natural circulation does not involve an unreviewed safety question. This mode will not increase the probability of any accident nor will it increase the consequences of any accident. Analyses have been performed to confirm that natural circulation core cooling during Mode 5 high water level conditions provides adequate core decay heat removal and coolant circulation and therefore may be designated as an alternate method for both decay heat removal and coolant circulation.

Serial Number: 95-030-NPE Document Evaluated: MNCR 0082-94

DESCRIPTION OF CHANGE: This change disposition repairs spalled concrete exposing the spiral reinforcement steel on Column D-5 of the standby service water (SSW), Basin B cooling tower. Current conditions indicate that: (a) ACI minimum concrete cover is not provided, and (b) the centerline of the reinforcement pattern within the column is potentially located approximately 1-3/4 inches off the centerline of the column. The impact on the structural adequacy of the column due to eccentricity of the reinforcement steel has been evaluated using PCACOL™V2.20, a computer program developed by the Portland Cement Association. PCACOL™ is based upon the ACI 318-83 code rather than the ACI 318-71 code described in UFSAR Section 3.8. Further, PCACOL™ is not among the programs listed for structural analysis in UFSAR Appendix 3D.

REASON FOR CHANGE: The concrete is spalled to a depth of approximately 1/4 inch and covers an area of approximately 90 square inches (15"x6") on the south face of the column. The spalled concrete exposes several of the column's spiral shear reinforcement bars and shows them to be placed only 1/4 inch from the surface of the concrete. The 1/4 inch concrete cover provided is much less than the 2 inches required by the drawings and ACI 318-71, even allowing for ACI placement tolerances. In addition, it is likely that the entire reinforcement pattern is located off-center of the column, resulting in an eccentricity of approximately 1-3/4 inches between the centerline of the column and the centerline of the reinforcement pattern. Such an eccentricity could conceivably affect the column's structural capacity and was therefore evaluated. Analysis of the potential impact of this condition is most appropriately performed using computer solution due to the complexity of the calculations involved. Although PCACOL™ is based upon the ACI 318-83 code rather than ACI 318-71, it is especially suited to this type of analysis.

The change disposition and safety evaluation is also extended to include generic implications associated with remaining columns.

SAFETY EVALUATION: Limiting Conditions of Operation and Surveillance Requirements related to the Standby Service Water Systems are contained in Technical Specification Sections 3/4.7.1 but do not include specific requirements for the concrete structure and columns. Since structural adequacy of the SSW structures is maintained, the technical specifications and the margins of safety associated with their bases are not affected.

Inspection of the exposed reinforcement steel does not indicate that corrosion has progressed to a significant extent, and spalling/cracking in the adjacent concrete was not observed. Therefore, it is likely that corrosion is confined to the area identified. This condition will be verified when the reinforcement steel is cleaned and inspected during repair. Additionally, no evidence of spalling or cracking was present in the remaining columns during inspection. In the future, the columns will be inspected at least semi-annually to insure that any evidence of reinforcement steel corrosion is promptly detected and appropriate remedial measures are taken. For the reasons outlined above, structural integrity of the columns is assured. Therefore, the probability and consequence of an accident evaluated in the UFSAR resulting in radionuclide release and associated with the SSW systems, have not changed. Similarly, the probability and consequences of a malfunction of equipment important to safety evaluated in the UFSAR, resulting in radionuclide release and associated with the SSW systems, have not changed.

The condition of the column reinforcement steel, the methods of analysis, and the repair details do not directly affect any plant system or containment boundary; nor do they affect the SSW systems or structures such that interfacing plant systems are affected. Therefore, the possibility of a new type of accident or equipment malfunction has not been introduced.

Serial Number: 95-031-NPE Document Evaluated: MCP 89/1109 Rev. 0 and CN 94/0061

DESCRIPTION OF CHANGE: Safety Evaluation Nos. CFRMCP 89/1109R00 and CFMCP89/1109R01 are not superseded by this change.

CN 94/0061 will eliminate design information from MCP 89/1109 to replace the original ground detection system with a 1KΩ ground detection circuit on Buses 11DD, 11DE, 11DG and 21DG and a 200Ω ground detection circuit on Bus 11DH. MCP 89/1109 issued design information that was to replace the original ground detection circuit on these buses due to obsolescence of the original ground detection meter-relay.

REASON FOR CHANGE: This CN will eliminate the design information for the ground detection circuit issued by MCP 89/1109 for Buses 11DD, 11DE, 11DH, 11DG, and 21DG. The existing ground detection circuits for these referenced buses shall remain in service to monitor and detect grounds.

SAFETY EVALUATION: The ground detection circuit design information per MCP 89/1109, for Buses 11DD, 11DE, 11DH, 11DG, and 21DG will be eliminated. The 1KΩ and the 200Ω ground detection schemes for the referenced buses were never modified per MCP 89/1109R00, therefore the referenced buses remain bounded by the existing plant design and in conformance with all applicable regulatory and safety requirements. Therefore, implementation of CN 94/0061 will not have any adverse impact on the safe operation of GGNS. The performance capabilities for the DC systems are not adversely impacted by this change. The existing ground detection circuits, Buses 11DD, 11DE, 11DH, 11DG, and 21DG, will provide a method of identifying grounds on these buses.

Serial Number: 95-032-NSRA Document Evaluated: Bases for Tech Spec Surv Reqmt SR 3.5.1.7 and 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.

Not performing specific response time testing of this actuation instrumentation is consistent with Generic Letter 93-05, "Line-Item Technical Specifications Improvements to Reduce Surveillance Requirements for Testing During Power Operation," which identified that the response time testing of isolation actuation instrumentation need not be performed if the actuation instrumentation acted concurrently with the diesel generator start and had a response time acceptance criteria the same as the diesel generator start time. Deletion of the requirement to perform response time testing of isolation actuation instrumentation whose required response time is the same as the diesel generator was requested via GNRO-94/00130, dated 11/10/93 and granted by the NRC in Technical Specification Amendment 120, dated 02/21/94 (GNRI-95/00044). In addition, 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 10/06/86 (MAEC-86/0330).

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

SAFETY EVALUATION: During the design basis event the HPCS actuation instrumentation functions concurrently with the starting of the associated diesel generator. The accident analyses assumes a 10 second time delay in the starting of the HPCS system to allow the associated diesel generator to start and supply AC power prior to the pump starting or system valve movement. The response time of the actuation instrumentation is very small when compared to the 10 second diesel generator 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.

The most detailed analysis supporting elimination of these 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. Also, discussed in the NRC's SER was the NRC's review of the deletion of response time testing requirements for instrumentation with much shorter response time requirements (e.g., 33 seconds) and agreed with the BWROG that it was reasonable to assume that technicians would identify any instrumentation response times delays greater than 5 seconds.

Since the response time of these instruments is masked by the diesel generator start time (10 seconds), the remaining required surveillances for this instrumentation (e.g., CHANNEL CHECKS, CHANNEL FUNCTIONAL TESTs, and CHANNEL CALIBRATIONS) provides 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 remaining required surveillance 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: 95-033-NPE Document Evaluated: MCP 93/1041

DESCRIPTION OF CHANGE: This change will provide the design documentation for installation of a suitable connection and quill assembly to the Radial Well System. This change will also install a new chlorine analyzer in place of the existing chlorine analyzer P44-AITS-N035.

REASON FOR CHANGE: EER 91/6434 identified valve P47-F034 on the plant service water (PSW) test connection line as being cracked. No freeze protection is currently used on the line and it appears that the valve froze and cracked. Therefore, it is necessary that it be replaced or repaired. However, in addition to valve P47-F034 being cracked, the existing line and valve are too small to be utilized for chemical injection in the event of feed system failure or maintenance.

MNCR 896-83 identified chlorine analyzer P44-AITS-N035 as not being installed per installation detail J-0145S (drain not provided). As a result, the chlorine analyzer was abandoned in place per MCP 89/1022. It was not being used by the plant and was serving no function.

Chemistry has requested that the subject chlorine analyzer for the PSW system be put in service to monitor free or total residual chlorine. This will alleviate the necessity of having Chemistry personnel physically monitor the system 3 hours per day, 7 days per week.

SAFETY EVALUATION: The Plant Service Water (P44) and Radial Well (P47) Systems are not addressed in the GGNS Technical Specifications. This change will not result in any operational or functional changes to these systems or impact the operation of any other system. No technical specification requirements are changed or added as a result of this design change.

The design changes to be implemented per this change do not affect any existing system cross-ties nor do the changes create any new cross-ties. In addition, non-safety related piping for the PSW system is routed so that a pipe break will not flood or damage any safety related equipment. The described design changes will not alter or affect the operability of safety related equipment or systems. Failure of the affected systems will not compromise any safety related systems or prevent safe shutdown, and the evaluated design changes will not increase the probability or consequences of an accident previously evaluated in the UFSAR.

The changes will not compromise any safety related systems or prevent safe shutdown since no new interface with equipment important to safety is created nor is such equipment prevented from operating as designed.

Installation of injection quill P47D010 along with replacement and utilization of previously abandoned chlorine analyzer P44-AITS-N035 will not create the possibility of an accident of a different type than previously evaluated in the UFSAR. This equipment serves no safety function and its use will not create any new failure modes.

The technical specifications do not contain any margins of safety for the operation or design of the Plant Service Water or Radial Well Systems. The changes do not affect any systems in the technical specification or add additional requirements that need to be included in the technical specifications.

Serial Number: 95-034-NSRA Document Evaluated: 10-S-02-2

DESCRIPTION OF CHANGE: Updates DCC telephone numbers, changes the method for quarterly updating ERO phone numbers, incorporates activation of ENMC and EIC at the Alert classification.

REASON FOR CHANGE: Changes procedure to reflect current practices.

SAFETY EVALUATION: This procedure revision does not require a change to the technical specifications and does not involve an unreviewed safety question.

Serial Number: 95-035-NPE Document Evaluated: MCP 94/1024 and SCN 94/0010A to MS-02

DESCRIPTION OF CHANGE: This change will provide pressure relieving/equalization capabilities for valves 1E22-F004 and 1E22-F015. This change is in response to concerns addressed in MNCR 0007-94.

REASON FOR CHANGE: MNCR 0007-94 identified a concern with the potential for 1E22-F004 and 1E22-F015 to fail to open on demand. NPE evaluated the situation and concluded that a pressure locking phenomena was possible for each valve. On 1E22-F004, to alleviate the potential for pressure locking, a 3/4" line will be routed from the area communicating between the valve seats, to the downstream side (reactor side) of the high pressure injection line. The equalization line will contain a 3/4" manual isolation valve to be used during HPCS pump testing. On 1E22-F015, to alleviate the potential for pressure locking, a 3/4" line will be routed from the area communicating between the valve seats, to the upstream side (suppression pool side) of the suction header line. The equalization line will contain a branch connection with 3/4" manual isolation valves to be used during construction and penetration testing.

SAFETY EVALUATION: 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. Testing 1E22-F004's upstream (outboard) disc in the forward direction is consistent with testing in the accident direction. Consequently, this modification does not result in testing which differs from that currently described in the UFSAR. Thus, bypassing of a single disc (inboard) will have no effect on containment integrity and should not be construed as such. Additionally, as described in 3/4.4.3.2, 1E22-F004 serves as a reactor coolant pressure boundary valve. Test methodology remains unchanged and the ability of this valve to isolate the HPCS piping is unaffected. Additionally, the new containment isolation test connection valves associated with 1E22-F015 are designed, installed and tested to ensure the integrity of the primary containment. This change will have no effect on the operability of the valves or the associated E22 system. Although these valves do not receive a containment isolation signal, the ability of each to isolate as required by Technical Specification 3/4.6.4 is unaffected by this modification. Each valve does receive ECCS signals to open as addressed in 3/4.5.1 and 3/4.5.2. This modification is intended to provide added assurance that each valve will open if required. Finally, for 1E22-F004, its integrity for HPCS pump testing boundary isolation remains unaffected with the addition of the 3/4" manual isolation valve. Disc flexing is predicted to occur during HPCS pump testing, however, with the 3/4" manual isolation valve closed there should be no inadvertent injection flow. Since all functions discussed in the technical specifications remain unchanged by this change, all margins of safety remain unaffected. The updating of MS-02 to reflect the addition of the equalizing line in no way affects the technical specifications or the UFSAR.

Serial Number: 95-036-NPE Document Evaluated: MCP 94/1048 Rev. 0

DESCRIPTION OF CHANGE: Existing orifice plates N1N37-D008, D009, and D010, along with the existing 3/4"-FBD-46 line that supports the three orifice plates will be removed. The 3/4"-FBD-46 branch connection to the applicable 4"-FBD-46 and 3"-FBD-46 lines will be plugged to maintain the pressure boundaries for the applicable 4"-FBD-46 and 3"-FBD-46 piping.

Three new orifice plates will be installed in a 2" schedule 160 alloy steel (chromium molybdenum) jumper line from the 6"-FBD-46 condensate supply header to piping downstream of N1N37-F100A/B/C and downstream of N1N37-F107A/B/C. Each jumper line will have three orifice plates in series to gently break down the condensate pressure to near the saturation pressure of the condenser. The total number of orifice plates for all three bypass lines is therefore increased from three to nine.

REASON FOR CHANGE: Bypass line, 3/4" FBD-46 pipe developed leaks due to steam erosion (reference MNCR 0126-94). The function of the bypass piping is to provide a keep-full function (to prevent water hammer) for the condensate supply line to the bypass control valve pressure breakdown assembly. Evaluation of the probable cause was determined to be pipe erosion due to flashing.

SCN 95/0002A against MS-02 will add the service conditions for the new line Class FAD-1 and SCN 95/0001A against MS-03 will add the standard material requirements for line Class FAD.

SAFETY EVALUATION: Replacing carbon steel piping and associated fittings with chromium-molybdenum piping will not affect the Turbine Bypass System (N37) function, operation, or performance in any way. The effect of the new piping on the system will have a significant improvement in the form of system reliability. The piping has been designed to ANSI B31.1 code requirements and supported for the appropriate deadweight and thermal loads only. Therefore, the piping and their supports will function in their intended manner. The system affected by this change has no safety related functions. Failure of this system will not compromise any safety related system or component and will not prevent safe reactor shutdown. This system is not addressed in the technical specifications and no new requirements are being added. Therefore, this change will not require a change to the technical specifications and will not create an unreviewed safety question.

Serial Number: 95-037-NPE Document Evaluated: MCP 94/1017

DESCRIPTION OF CHANGE: Modify the valve logic of D23F591, D23F592, D23F593, D23F594, M71F594, and M71F595 so that the valves have auto isolation with a manual override feature. The control circuits shall be redesigned such that the individual valves can be manually reopened 30 seconds after the isolation signal is received (even if the isolation signal is still present) by the corresponding control room handswitches, but the valves shall automatically reclose if the isolation signal clears and then reappears.

REASON FOR CHANGE: The Drywell Fission Product Sample Rack H22P140 and the PASS Containment/Drywell Atmospheric Sample Rack P33P008 are non-safety related but essentially form a part of the containment pressure boundary with the current design. This is because the remotely operated manual isolation valves D23F591, D23F592, D23F593, D23F594, M71F594, and M71F595 associated with the sample lines for these panels are normally open. If these valves are not closed within 20 minutes after a LOCA the 10CFR100 limits could be exceeded.

SAFETY EVALUATION: The changes of this evaluation will not compromise any safety related system structure or component. The failure of the affected equipment will not result in any evaluated transient or accident. The purpose of these valves is to ensure that any fission product release is contained in the event of an accident. Since the "single active failure criteria" of the M71 system is maintained, there is no increase in the probability of containment malfunction. This change actually improves the ability of the affected isolation valves to meet the assumptions in the LOCA dose analysis. This change will ensure that the affected valves are closed much faster than the 20 minutes assumed for manual valves in the event of a LOCA which will reduce the fission product release from these sample lines to essentially zero. The seismic qualification of the affected panels will be maintained.

Since this design changes the valve logic to automatic, the probability of spurious valve closures of D23F591, D23F592, D23F593, D23F594, M71F594, and M71F595 is increased. The 1D23C001 drywell sample pump is not safety related but it is required to be operational by technical specification and is continuously running during normal plant conditions for Regulatory Guide 1.45 leak detection. If the isolation valves were to close because of a spurious isolation signal, the pump will be damaged if the valves are not quickly reopened. This is not a safety concern. A low flow condition will activate an alarm to alert the operators to take appropriate actions and the D23C001 sample pump operability requirements still apply. It is possible that the Drywell Monitoring System may be exposed to a maximum of 30 psig before the isolation valves can close during a LOCA. This is higher than the original 2 psig design rating but based on testing and vendor information the system can withstand a pressure of greater than 33 psig. The changes to the control circuits of the affected valves will not prevent the valves from being closed manually. The stroke time required to manually close the valves is not affected.

Pump damage should not be a concern for the PASS atmospheric sample pump 1P33C009 which is also required by technical specification. This is because PASS samples are taken only periodically and it would be extremely unlikely to have a spurious isolation while a sample is being taken. Also the operator taking the sample is at the panel where the sample is taken and would be monitoring flow conditions.

Serial Number: 95-038-NPE Document Evaluated: Engr Rpt GGNS-94-0039 Rev. 0

DESCRIPTION OF CHANGE: This change assesses a comprehensive study of the safety functions for the feedwater check valves (FWCVs B21F010A/B and B21F032A/B) and a thorough characterization of the operational conditions for these functions.

REASON FOR CHANGE: 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 change 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 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. 0, evaluate 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 analysis results show that for the non-limiting 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.

Serial Number: 95-039-NPE Document Evaluated: DCP 94-003 and SCN 94-002 to Spec 9645-M-242.2

DESCRIPTION OF CHANGE: This change modifies feedwater check valves Q1B21-F032A&B from a one piece disc and arm (Q1B21-F032A has a carbon steel disc/arm and Q1B21-F032B has a stainless steel disc/arm) to a two piece disc and arm assembly of stainless steel. UFSAR Section 5.4.9.3 description of the lowest service metal temperature for the disc of these check valves will be revised to reflect the use of a stainless steel disc in Q1B21-F032A. In addition, the lower body guide rib of Q1B21-F010A&B will be widened. This change will also remove the resilient seats and return valves Q1B21-F010A&B and Q1B21-F032A&B to hard seats only.

REASON FOR CHANGE: The above modifications will enhance the reliability of alignment and leak tightness of feedwater check valves Q1B21-F010A&B and Q1B21-F032A&B. Q1B21-F032B has a requirement for a stainless steel disc because of low temperature brittle fracture concerns. A stainless steel disc/arm assembly will be installed in Q1B21-F032A for the ease of spare parts.

SAFETY EVALUATION: With respect to the lowest service metal temperature description of Q1B21-F032A&B, the use of a stainless steel disc and arm assembly in Q1B21-F032A is conservative, as it eliminates possible brittle fracture concerns. The above modifications to Q1B21-F010A&B and Q1B21-F032A&B will not change the performance or function of any plant component or system important to safety. There is no increase in the probability of occurrence or the consequences of an accident or malfunction of equipment important to safety. There are no new malfunctions of components or systems resulting from these modifications. These modifications will enhance the ability of these feedwater check valves to meet the containment leakage requirements of the technical specifications.

Serial Number: 95-040-NPE   Document Evaluated: Engr. Rpt GGNS-94-0008 Rev. 0

DESCRIPTION OF CHANGE: Technical Requirements Manual (TRM), Section 7.6.3.10.3.e requires the functional testing of a representative sample of each type of snubber. For the RFO7 test period a one time deviation will be made to the functional testing of snubbers. For this test period a "freedom of motion" test will be performed on 100% of the PSA 1/4 and 1/2 snubbers in lieu of the functional test. The remaining categories of mechanical snubbers and the hydraulic snubbers will be functionally tested in accordance with the Technical Requirements Manual using the 10% sample plan.

REASON FOR CHANGE: The functional failure in the snubber population has been concentrated in the smaller sizes, PSA 1/4 and 1/2 sizes. The failure mechanism has been primarily due to high drag forces (i.e., corrosion, bent guide rods/screw shaft, worn parts, loose capstan spring, etc.). Testing 100% of the smaller size snubbers by the "freedom of motion" test will provide a high confidence level that the population of small size snubbers installed are all operable and will function as designed. The other sizes will be functionally tested in accordance with TRM using a 10% random sample. The PSA 1/4 and 1/2 snubbers tested by the "freedom of motion" and are suspect, will be put on the test bench and functionally tested. All functional test failures will be reported and evaluated by engineering in accordance with the requirements of TRM Section 7.6.3.10.3.g.

SAFETY EVALUATION: Performance of the "freedom of motion" test during RFO7 on 100% of the population of PSA 1/4 and 1/2 snubbers will provide a high level of confidence in these snubber sizes. The functional test requirements in the TRM calls for both a drag test and activation test be performed on the snubbers to qualify the snubbers as operable. The "freedom of motion test" can be used to determine whether a snubber has high drag whereas the acceleration causing activation to occur cannot. The experience at Grand Gulf to date has shown that there have not been any failures due solely to activation (Reference GIN 95/00190). In addition all PSA 1/4 and 1/2 snubbers which are suspect or fail the "freedom of motion" test will be functionally tested for both drag and activation on the test bench. The remaining snubber population, i.e., PSA 1, 3, 10, 35, 100 and hydraulic will be functionally tested per the 10% plan as defined in the TRM. This one time deviation will provide a high confidence that the total snubber population installed in the plant will meet the design requirements.

Serial Number: 95-041-NSRA   Document Evaluated: Safety Evaluation Report Section 3.9.2

DESCRIPTION OF CHANGE: Performance of inspections and examinations of vessel internals during RF06 revealed that the mixer section of Jet Pump 10 had become displaced from its mounting assembly and also identified recordable indications on the hold down beam of Jet Pump 21. Subsequent analysis of the failure mechanism for Jet Pump Beam 10 as well as indications discovered on Beam 21 were confirmed as being IGSCC induced. As part of the corrective actions for this occurrence, all 24 jet pump hold down beams were replaced with like components based on the approval of 10CFR50.59 Safety Evaluation (SE) 93-0120-R00, where GGNS took exception to the commitment for replacement of jet pump beams. The like-for-like replacement was based on an exception to a SER commitment, which required replacement of the originally installed BWR 4-6 jet pump hold down (Group 1) beams where excessive cracking is identified, with improved heat treated hold down (Group 2) beams (Reference SER 3.9.2 and AECM-80/0268). Based on the failure data available at the time for Jet Pump Beams 10, and 21, SE 93-0120-R00 only provided justification for operation during one cycle (Cycle 7) with the like replacement beams installed.

This change revises the original SE 93-0120-R00 such that the limit on in-vessel operating time with the jet pump beams installed during RFO6 is specified by Calculation QR-479-17.35 and MNCR 0213-93. This eliminates the generic life restriction and provides specific restrictions for the individual beams based on their analyzed configurations. These restrictions, coupled with the examinations required by NUREG-CR-03052 provides assurance that jet pump beams are replaced before their integrity is reduced to an unacceptable level.

REASON FOR CHANGE: This change is being written to eliminate the limitation placed on in-vessel operating time for the jet pump beams installed during RFO6. This revision is based on industry experience, NUREG-CR-03052, Calculation QR-479-17.35, and MNCR 0213-93.

SAFETY EVALUATION: The NRC issued IE Bulletin 80-07 to address BWR jet pump assembly failures that had occurred in earlier vintage BWRs utilizing the BWR-3 jet pump beam design. Subsequent analysis indicated that the BWR 4-6 (Group 1) jet pump beams may be susceptible to the same failure mechanism. Extensive studies concluded that jet pump beam failure is caused by slowly progressing stress corrosion cracking. As a long term resolution to this failure mechanism, plants were to reduce the preloading on installed Group 1 beams from 30 kips to 25 kips and conduct periodic examinations of the beams for cracking. Another long term solution was to make a commitment to the NRC that required replacement of cracked BWR 4-6 beams with new heat treated (Group 2) beams. This was the commitment in SER 3.9.2 to which GGNS took exception, based on approval of SE 93-0120-R00.

The same requirements imposed on the original jet pump beams (reduced preload and periodic examination) are also applicable to the replacement beams installed during RFO6. Industry experience has not indicated any failures of Group 1 jet pump hold down beams for the first five years in service. The jet pump beams that were replaced during RFO6 at GGNS performed reliably for 8 years of commercial operation without any failures. Based on past performance history, reduced beam preloading, periodic examinations, and the analysis provided in Calculation QR-479-17.35, jet pump beam service life is controlled sufficiently to ensure undetected degradation does not occur. The service life is unique to each individual beam based on the as-built thickness of the beam ears. These dimensions with specified service life is contained both in the calculation and Interim Number 3 disposition to MNCR 0213-93. As a minimum, Jet Pump Beam 24 will require replacement before returning to operation during RFO8. Replacement of remaining beams is as specified in MNCR 0213-93 with the life span beginning at RFO6 (when the beams were installed).

Serial Number: 95-042-NPE   Document Evaluated: DCP 91/0088-1 Rev. 1

DESCRIPTION OF CHANGE: This change will replace panels 1H22-P171 and P172 with a microprocessor based distributed control system. This system will be an exact replica of the existing control strategies with two exceptions. The condensate and condensate booster pump min-flow logic is being modified to include time delays and multiple trip set points. Also, one pump (instead of all three) will be tripped at a time. The feed pump trip logic will be modified to have multiple setpoints and delays.

The feedwater heater dump valve controller logic is being modified to help prevent heater isolation during rapid reductions in power by providing gain scheduling and a reduced setpoint during low power conditions.

REASON FOR CHANGE: The change to a microprocessor based system designed for fault tolerance will accept redundant power supply feeds, and allow for easy changes to configuration logic in the future. One such change will be implemented with installation of the new cabinets. This change will consist of modifying the condensate and condensate booster pump min-flow trip logic to allow operator intervention by adding a time delay in the trip circuit. Present design results in immediate trip of all pumps during a low flow condition. The new system will allow the operators the opportunity to correct a situation which caused a min-flow condition by providing Control Room indication on the 1H13-P680 panel that the trip timer has started, and then by tripping the pumps in a staggered fashion. Changes to the RFP recirculation control valve controller and feed pump trips will also help operations during these situations. Changes to support improved feedwater availability by preventing heater isolation and the use of transmitter "auctioneering" will enhance system availability.

SAFETY EVALUATION: This change will not affect the technical specifications, the bases for any technical specifications, or the operating license. The values and bases for the Minimum Critical Power Ratio (MCPR) operating limits are not affected, and the feedwater and condensate system equipment and instrumentation involved are not explicitly covered by the technical specifications.

This change will not increase the probability for occurrence of accidents or malfunctions of equipment important to safety previously evaluated in the UFSAR because equipment operating characteristics meet established design requirements and reliability is enhanced by the proposed changes. Failures of affected penetrations are also no more likely.

To maintain integrity during tornado depressurization, penetration seal details provided for these penetrations also ensure the integrity of the 3 psi pressure boundary. Operational considerations have also been provided to ensure compliance while performing this work. Additionally, operational considerations have been provided to ensure that the integrity of the Control Room envelope as defined in UFSAR 6.4.2 is not jeopardized and to ensure that the Control Room leak rate contained in Operating License Condition 2.C.38 is not exceeded.

This change will not increase the consequences of an accident or malfunction of equipment important to safety from that previously evaluated in the UFSAR because appropriate design requirements and operational considerations have been provided to ensure that equipment performance remains within the limits currently assumed in existing accident analyses. This change will also not create the possibility for an accident or malfunction of equipment important to safety of a different type than any previously evaluated in the UFSAR because limiting system failure modes are unchanged, and additional challenges to safety features or equipment important to safety will not occur. This change will not reduce the margin of safety as defined in the bases for technical specifications since limiting and non-limiting events which may affect fission product barriers remain clearly bounded by existing analyses, and penetration operability and integrity remain consistent with that assumed by the technical specifications.

Serial Number: 95-043-NPE Document Evaluated: MCP 94/1018

DESCRIPTION OF CHANGE: This change replaces four of the six remote operated target rock electric solenoid valves (1N64-F010A/B & 1N64-F576A/B) with manual valves. Two of the remote operated electric solenoid valves (1N64-F012A/B) will be deleted since manual valves exist in the associated lines to meet operational requirements. Control Room handswitches and position indication for all six valves is to be removed.

REASON FOR CHANGE: EER 92/6249 requested target rock valves 1N64-F010A/B and F576A/B be replaced with manual valves because the existing electric solenoid valves had proven to be unreliable and high maintenance items. The EER also requested that target rock valves 1N64-F012A/B be removed since manual valves exist in the associated lines to meet operational requirements.

Cr-Mo pipe and manual valves will be utilized to replace existing carbon steel piping and manual valves for the offgas preheater drain lines. This change is needed to limit the greater than expected affects of flow accelerated corrosion (FAC), and Cr-Mo is more resistant to FAC than carbon steel.

SAFETY EVALUATION: The modifications made by this change will not impose a change to the criteria listed in Table 3.2-1 for the Offgas System. Furthermore, the postulated worst case failure of the Offgas System analyzed in UFSAR Sections 15.2.5 (Loss of Condenser Vacuum) and 15.7.1 (Offgas System Leak or Failure) envelopes the occurrence and consequences of postulated accidents due to any failures associated with this design change. In addition, the use of Radflex seal material will be adequate for use as a radiation seal and will meet the intent of UFSAR Section 12.3.1.1.2.c. The changes do not increase the probability of increase in consequence or occurrence 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 and valves installed by this design change meet ANSI B31.1 code requirements and is supported for the appropriate dead weight and thermal loads.

The handswitches for 1N64-F010A/B are the only solenoid valve controls located in the Control Room. Since Control Room handswitches and indication are to be removed for valves 1N64-F010A/B as a result of this design change, Operations will no longer have remote capability to align offgas preheaters to the opposite train steam trap or bypass a flooded steam trap. These are considered to be abnormal events requiring an operator to enter the offgas preheater room to operate the required valves. The estimated dosage for an operator to enter preheater rooms to operate condensate drain line valves during abnormal conditions is about 20 mrem/entry. This is considerably less than dose received during maintenance activities on the solenoid valves and is an acceptable dose limit.

Serial Number: 95-044-NPE Document Evaluated: TSR 94-09

DESCRIPTION OF CHANGE: The change consists of supporting the temporary addition of lead shielding blankets from the CRD housing restraint beam at Azimuth 180°. The temporary shielding is being added for the purpose of reducing personnel exposure during refueling maintenance activities underneath the reactor vessel. This change allows a maximum of 900 pounds of temporary lead shielding be supported from the CRD housing restraint beam during Modes 4 and 5.

REASON FOR CHANGE: The temporary lead shielding is being added for the purpose of reducing personnel exposure during refueling maintenance activities underneath the reactor vessel. This change estimates that a 1.343 net personrem savings (per outage) will be obtained from the installation of this temporary shielding.

SAFETY EVALUATION: The existing CRD housing restraint beam was evaluated for the temporary addition of lead shielding. The beam was evaluated in accordance with the existing design codes and for the applicable loading conditions which require evaluation during Modes 4 and 5. Also the change was reviewed to ensure no new hazards were created.

Based on the review described above, the CRD housing restraint beam was found to be structurally adequate for the temporary addition of lead shielding provided the criteria given in the change disposition is followed.

Serial Number: 95-045-NPE Document Evaluated: TRM Section 7.6.3.10.3.e

DESCRIPTION OF CHANGE: The Technical Requirements Manual (TRM), Section 7.6.3.10.3.e requires the Nuclear Regulatory Commission (NRC) to be notified prior to changing the snubber functional test plan. This proposed change to the TRM will delete this requirement and allow usage of either acceptable test plan (37-plan or the 10%-plan) without submitting a notification to the NRC.

REASON FOR CHANGE: Section 7.6.3.10.3.e of the TRM reads, "The sample plan shall be selected prior to the test period and cannot be changed during the test period. A written report of the sample plan selected shall be submitted to the NRC pursuant to Section 50.4 of 10CFR Part 50 prior to the test period or the sample plan used in the prior test period shall be implemented". The section(s) discussing snubber inspection and testing has been relocated from the technical specifications to the Technical Requirements Manual (reference GNRO-93-00109). The TRM provides increased flexibility over the technical specifications by not requiring prior approval of the NRC for changes. Therefore deleting the requirement to notify the NRC will facilitate the choice of an acceptable snubber test plan with the flexibility to change, up to the testing period if warranted (i.e., take advantage of recent changes in industry philosophy, new issues, etc.).

SAFETY EVALUATION: This evaluation is for a change to the TRM deleting the reporting function of notifying the NRC for changing from one test plan to another test plan. The 37-plan and the 10%-plan are acceptable functional test plans in the TRM and yield a similar confidence level. These test plans will not be affected by this change without performing an evaluation that will yield an equivalent or higher confidence level test plan. Therefore, there will not be a reduction in the number of snubbers functionally tested and the requirements for test period and evaluation of failures will not be changed. Overall, due to increases in new information and changes in the industry, the added flexibility provided by deleting this requirement will be a proactive step toward improved performance. The requirement in the TRM is strictly a reporting function and does not affect snubber functional testing.

Serial Number: 95-049-NPE   Document Evaluated: MCP 94/1064

DESCRIPTION OF CHANGE: This change removes the fiberglass insulation from the Drywell Cooling System supply ductwork, fans, plenums and coolers and, because of the insulation removal, rebalances appropriate sections of the system to maintain drywell maximum air temperatures below the Technical Requirements Manual limit of 150°F.

REASON FOR CHANGE: The insulation is being removed to eliminate a potential source of emergency core cooling pump strainer clogging.

SAFETY EVALUATION: Implementation of the proposed change will result in no significant changes in the drywell environment and initial conditions and assumptions of the accident analyses in the UFSAR are preserved. The ability to quantify leakage within the drywell is maintained unaffected, and diverse means for determining changes in leakage are preserved. Therefore, this change does not constitute an unreviewed safety question and the margins of safety in the bases to the technical specifications are not reduced.

Serial Number: 95-046-NPE   Document Evaluated: MCP 95/1010, Rev. 0

DESCRIPTION OF CHANGE: This change provides details for the replacement of non-class 1E Battery 1K3. The replacement cells are C&D Power Systems Type LCR-33.

REASON FOR CHANGE: Battery 1K3 requires replacement because it is approaching the end of its service life due to elevated temperatures within this room. DCP 88/0103, Revision 0 has added cooling to this room which will preclude accelerated aging of the replacement battery added by this change.

SAFETY EVALUATION: Electrical Calculation EC-N1L11-95002, Revision 0 verifies the acceptance of the capacity and voltage ratings of the new battery. Battery 1K3 is primarily used to supply DC power to BOP inverters 1Y80, 1Y98 and 1Y99. This new battery is acceptable because it will meet existing requirements for design and operation of the "K" 125 Vdc system as evaluated by battery sizing calculation EC-N1L11-95002. The only association of Battery 1K3 to Class 1E equipment is the source for the battery's chargers. Battery 1K3's chargers, 1DK4 and 1DK5, are fed by Class 1E Load Centers 15BA1 and 15BA2, respectively. The feeder breakers for these chargers are tripped upon an accident signal per the requirements of Regulatory Guide 1.75. Replacement of the battery will not have any impact on the isolation of Battery 1K3 from the Class 1E Bus 15AA per the requirements of Regulatory Guide 1.75. The UFSAR change listed in Section A would move the description of sizing the 1L battery from Section 8.3.2.1.1 to Section 8.3.2.1.6, which is a more appropriate location for the sizing methodology. In the process of this transfer the 1K battery will be added to this description to reflect that it also has been sized per design intent of industry standards. Figure 8.3.2.1.6 requires change to reflect the amp-hour discharge capacity change from 2550 to 2320.

Serial Number: 95-047-NPE   Document Evaluated: DCP 92/0046 Rev. 1

DESCRIPTION OF CHANGE: The proposed change will improve the efficiency of the high pressure (HP) turbine by replacing the HP rotor, inner casings and associated components. The turbine operates under speed control system during initial start-up. During plant start up, the turbine control system is switched from speed control to reactor pressure control. The initial design of the HP turbine resulted in a certain pressure drop across the HP control valves to control the reactor pressure, thus limiting the turbine inlet pressure. The proposed change will increase first stage operating pressure of the HP turbine by shifting some pressure drop away from the HP control valves to the HP turbine blade path to improve the efficiency of the turbine. The HP turbine first stage pressure will be slightly above the calibrated span of some transmitters (C11 and C71 systems). However, trips associated with these transmitters occurs well within the calibrated span of the transmitters. The proposed change is expected to provide a gain of approximately 20-26 Mwe.

REASON FOR CHANGE: The proposed change will provide additional electrical megawatts for the same reactor thermal output (increase efficiency of the turbine-generator). Siemens Power Corporation (OEM) has proposed to improve the HP turbine efficiency by replacing rotor and inner casing with new design T4 blade profiles. Also, one stage will be added and the last two stages will have twisted blades to improve efficiency.

SAFETY EVALUATION: UFSAR Section 3.2 classifies affected systems as "Other", meaning loss of system function would not affect safe shutdown of the plant. UFSAR Table 3.2-1 classifies these systems components as non-safety related, non-seismic, Quality Group D, and ANSI B31.1. The proposed change will replace the existing HP rotor and inner casing with new design parts that will restore the efficiency of the HP turbine. No other changes to the existing HP turbine configuration will be required. The existing components and replacement components supplied by Siemens have been designed and manufactured in accordance with original standards (German standards). The components shall meet or exceed design and installation requirements of ANSI B31.1, Power Piping Code. The change will not affect any equipment important to safety. The modifications made by this change will not impose a change to the criteria listed in Table 3.2-1. The change will enhance the turbine efficiency without affecting operation of reactor pressure control system. Furthermore, all HP turbine parameter changes (i.e., HP control valve positions, etc.) have been incorporated into the Cycle 8 reload safety analysis.

Serial Number: 95-048-NPE   Document Evaluated: MCP 94/1049

DESCRIPTION OF CHANGE: To reduce erosion and corrosion, alloy steel piping (2-1/4% Cr. 1% Mo.) is utilized to replace the existing carbon steel piping in the condensate drain pot drain line routed to the condenser. Long radius butt weld elbows and bends are used rather than socket weld elbows to reduce the erosion rate. Existing carbon steel manual valves are replaced with Chromium-Molybdenum body valves.

REASON FOR CHANGE: MNCR 0067-94 documents a pin-hole leak through a fillet weld above a 90° elbow located on condensate drain line 1 1/2"-HBD-125; (reference: P&ID M-1083B C-2). The piping in question has a flow of wet steam and/or condensate in a steel pipe and has wall thinning normally associated with flow accelerated corrosion (FAC). The current condition of a significant portion of the piping requires replacement due to wall thinning or undesired maintenance will be required. Therefore, the piping will be replaced with a material known to be more resistant to FAC than the existing carbon steel.

SAFETY EVALUATION: Replacing carbon steel piping, fittings and valves with an erosion resistant alloy steel (Chromium-Molybdenum) will not affect the Reactor Core Isolation Cooling System (RCIC) or the N19 Condensate System function, operation, or performance in any way. The effect of the replaced piping on the system will have a significant improvement in its ability to remain in service without leaks due to erosion problems.

The 3/4" test connection was evaluated for postulated missile consequences in the initial installation of the piping system. Unrestrained sections of piping such as vents, drains, and test connections are evaluated as potential missiles if the failure of a single circumferential weld could cause their ejection.

The replacement of carbon steel pipe with Chromium-Molybdenum pipe and modification of pipe supports does not affect the integrity of operability of the system or any other safety system. The reanalysis of the piping system has shown that all ASME Section III code allowables have been met; therefore, the probability of piping failure has not increased. The modification will not introduce any new postulated piping failures and the existing hazards evaluations are not affected. Therefore, the probability of occurrence of a malfunction of equipment important to safety previously evaluated in the UFSAR, either as a consequence of a piping system failure (such as caused by jet impingement or pipe whip due to pipe break) or in response to any such postulated event (where a single failure is considered) is not increased.

Portions of this system (RCIC) are addressed in the technical specifications. This system function with ECCS and the limiting condition for operation are not affected by this modification. Using an erosion resistant alloy steel (Chromium-Molybdenum) will have no effect on RCIC performing its intended function and will not require a change to the technical specifications.

This change will not increase the consequences of an accident or malfunction of equipment important to safety from that previously evaluated in the UFSAR because appropriate design requirement and operational considerations have been provided to comply with the technical specifications and the UFSAR. 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 UFSAR because the opening and closing of the penetrations for the installation of new pipe will not affect the penetration's ability to perform from that previously evaluated. This change will not reduce the margin of safety as defined in the basis of any technical specification because Operation License Amendment Number 82 relocated the Fire Protection Technical Specifications to UFSAR Appendix 16A. Additionally, requirements of Technical Specification 3/4.6.6 have been included as operational considerations to ensure that secondary containment boundary is maintained.

Serial Number: 95-050-NPE   Document Evaluated: MCP 95/1002 Rev. 0

DESCRIPTION OF CHANGE: This change will disable the Main Generator Tritium Monitoring System. The design purpose of the tritium system was to rapidly detect minor amounts of moisture in the generator in order to protect the generator retaining rings. The original generator retaining rings were made of 18-5 stainless steel which is highly susceptible to inner-granular stress corrosion cracking (IGSCC). Any significant moisture in the generator could have introduced a high potential for IGSCC resulting in a failure of the retaining rings with the generator on line. MCP 94/1055 replaces the 18-5 retaining rings with 18-18 retaining rings. These rings are not susceptible to IGSCC. This replacement will eliminate the need for the Tritium Monitoring System. In support of this, two main control room and four generator auxiliaries cabinet (GAC) annunciators will be deleted, three power supply breakers will be spared, nine BOP computer points will be deleted, eight instruments will be tagged as "abandoned-in-place," and the cards associated with tritium monitoring devices located in the electrical generator protection (EGP) cabinet will be removed.

REASON FOR CHANGE: Due to the use of type 18-18 retainer rings to be installed during RF07, existing leak detection and the addition of hydrogen dewpoint monitoring to be installed in RF07, the use of a tritium measuring device (TMD) for leakage detection of primary water in the generator is no longer required. Disabling the tritium system will eliminate near critical path outage time required to calibrate the Tritium Monitoring System. The RF07 work scope includes a high pressure turbine upgrade as well as replacing the generator and exciter. The extensive scope of turbine work will approach or exceed outage critical path if the Tritium Monitoring System calibration and maintenance are also included. Disabling the Tritium Monitoring System will reduce outage scope and will minimize plant total outage time.

SAFETY EVALUATION: The disabling of the Tritium Monitoring System does not alter the results of the missile analysis originally completed for the GGNS turbine/generator since a generator retaining ring burst was determined to be contained by the generator casing. The modification does not require a change to the technical specifications or to the text of the UFSAR. Figure 10.2.1, a replication of P&ID M-1044, will be modified to indicate that certain instruments associated with the Tritium Monitoring System have been abandoned-in-place and that annunciators and computer points associated with this modification have been deleted. The modification will not increase the probability of occurrence or increase the consequences of an accident previously evaluated in the UFSAR since the modification introduces no new failure mechanisms for the generator. The generator is not safety related and is not seismically qualified. No penetrations for the control building or Control Room envelope will be opened due to the deletion of the Control Room annunciators associated with this modification.

Serial Number: 95-051-NSRA Document Evaluated: TR 3.3.2.1, Table TR 3.3.2.1-2

DESCRIPTION OF CHANGE: The TRM is being revised to make applicability and surveillance requirements for APRM, IRM, and SRM rod blocks consistent with those for the neutron monitoring indication and RPS functions. The applicability requirement for APRM rod blocks in Mode 5 is being deleted from the TRM to be consistent with the APRM RPS technical specification requirements. The TRM applicability requirements for IRM rod blocks are being changed to require the rod blocks in Mode 5 only if a control rod is withdrawn from a core cell containing one or more fuel assemblies. This change is consistent with the IRM RPS Technical Specification requirements. The surveillance interval for SRM and IRM rod block channel functional tests in the TRM are being revised from once per "7 days and prior to each reactor startup" to once per "31 days and prior to reactor startup". The SRM Technical Specification required channel functional test for monitoring and indication is currently once per 31 days. The IRM Technical Specification required channel functional test for the RPS function is once per 7 days and is not affected by this change. Although the IRM rod block functional test frequency will now be less than the RPS requirements, the change is justified based on the function not being required to support accident analyses or a technical specification margin of safety. The neutron monitoring and RPS functions of these instruments are not being modified in any way.

REASON FOR CHANGE: The TRM presently has overly restrictive operability and surveillance requirements for neutron monitoring rod blocks during Mode 5 and Mode 2. Modifying these requirements as described above will allow the establishment of appropriate controls commensurate with the safety significance of the function.

SAFETY EVALUATION: As discussed in the NRC's SERs for Technical Specification Amendments 109 and 120 (GNRI-93/00221 & GNRI-95/00044), the APRM, IRM, and SRM control rod blocks act as backups to Rod Withdraw Limiter (RWL), Rod Pattern Controller (RPC) or refueling interlock rod blocks or in some cases as backup to IRM or APRM scrams. In all cases, no credit is taken in any design basis transient analyses for the functioning of APRM, IRM or SRM control rod blocks.

Adequate defense-in-depth is provided to prevent or mitigate any postulated inadvertent criticality or reactivity anomalies by RWL, RPC, refueling interlocks, mode switch interlocks, procedural controls on rod withdrawal sequences, core reload analyses performed each cycle and/or RPS scrams. Adequate neutron monitoring of the core for transient analyses is still provided by the RPS trip system.

Technical Specification Amendments 109 and 120 removed all APRM, IRM and SRM control rod blocks from the technical specification based on the conclusion that their function provided only redundant capability to the TS required RPC, RWL, and reactor mode switch functions for preventing control rod withdrawal errors while in various modes of operation. In addition, they perform no safety function in the mitigation of design basis transients.

Therefore, the evaluated change does not require a technical specification change or result in an unreviewed safety question.

Serial Number: 95-052-PSE Document Evaluated: TSTI 1G33-95-001-0-S

DESCRIPTION OF CHANGE: This change evaluates the chemical decontamination of the Reactor Water Cleanup System (RWCU) using Westinghouse's CANDEREM-AP-CANDEREM process delineated in Technical Special Test Instruction (TSTI) 1G33-95-001-0-S.

This change reviews the heavy load evaluation (EER 95/6057) and chemical and radioactive liquid spills.

REASON FOR CHANGE: To reduce radiation fields and thus the exposure experience by personnel working on or around the RWCU system or its components.

SAFETY EVALUATION: The conclusion of this safety evaluation is that the performance of TSTI 1G33-95-001-0-S does not constitute or raise any unreviewed safety questions nor does it require a change to the GGNS Unit 1 Technical Specifications.

The possibility of a chemical/radiological spill will be minimized by the following:

- A leak check of the chemical decontamination skids, hoses, system connections and the RWCU system will be performed prior to any chemical injections. This will prove the integrity of the system thus reducing the potential for chemical or radioactive liquid leaks during performance of the chemical decontamination.
- Boundary valves for the decontamination will be controlled by issuance of red tags per Administrative Procedure 01-S-06-1.
- The chemical decontamination equipment and hoses will be continuously manned and monitored during performance of the decontamination process.
- A berm will be constructed around the Westinghouse decontamination skids to contain any leaks/spills.
- Isolation valves provided by Westinghouse between the decontamination skids and the RWCU system will be controlled from the chemical decontamination control panel. These valves are in addition to skid isolation valves available to isolate the decontamination skids from each other under abnormal conditions.

The chemicals to be used in the decontamination have been evaluated to ensure there will be no detrimental affects on system integrity. In addition, system piping will have both pre and post decontamination UT examinations made at several locations throughout the system to assure no significant pipe wall loss has occurred.

Radioactive isotopes removed from the RWCU system will be contained in Westinghouse's ion exchange media (resins). Disposal of this solid radwaste will be handled by means similar to standard plant procedures for disposal of solid radwaste. The resins will be transported to high integrity containers (HICs) inside a transportation cask. This transportation cask will be secured to a trailer in the Auxiliary Building train bay surrounded by a berm. As the radioactive materials are fixed to the resins there will be no airborne concerns with the transfer.

Serial Number: 95-053-PSE Document Evaluated: CI 44365 and CI 33341
MCP 92/1009 Rev. 1

DESCRIPTION OF CHANGE: This change documents the safety review of implementing MCP 92/1009 Revision 1 while relying on jet pump plugs as the barrier to prevent a loss of reactor water inventory - an event with potential to drain the vessel.

During RFO7, with the reactor in Operating Mode 5, fuel loaded in the core, reactor water level at reactor cavity high level (≈ Elevation 207.6') and vented jet pump plugs installed in both loops, MCP 92/1009 Revision 1 will be implemented, one loop at a time. The design package replaces the disc pack on Recirculation Discharge Gate Valves (RDGVs) B33F067A and B33F067B.

REASON FOR CHANGE: GE SIL 528 identified a potential failure of the Recirculation System RDGVs, due to a flow induced rotation of the valve discs. A permanent modification to prevent all of the valve identified failure modes has been issued. After further analysis, implementation of MCP 92/1009 Revision 1 will be accomplished with the jet pump plugs relied upon as the reactor coolant boundary.

SAFETY EVALUATION: Implementation of MCP 92/1009 Revision 1 with fuel in the reactor core creates a potential to drain the vessel through the open recirculation valve body. This evaluation documents the safety review of the planned work with reliance on the jet pump plugs as the reactor coolant boundary.

Installation of the jet pump plugs will allow recirculation loop discharge piping drain down, while maintaining reactor water level, so that the RDGVs, one loop at a time can be modified. The suction side of the recirculation piping will be isolated using the maintenance isolation valves B33F023A and B33F023B. Note - The PRA reviewed use of the suction plugs, however the unavailability of suction plugs for use in RFO7 dictates using the suction isolation valves, which are considered more reliable than the suction plugs. On the discharge side, the jet pump plugs will be relied upon as the sole reactor coolant barrier, for a limited time during valve bonnet removal, seating measurement, ISI visual inspection, and valve bonnet replacement. At all other times a blank flange will be installed in place of the valve bonnet, to minimize the time of reliance on the jet pump plugs as a reactor coolant boundary.

Serial Number: 95-054-PSE   Document Evaluated: TSTI 1B33-95-004-0-S

DESCRIPTION OF CHANGE: During RF07 the Recirculation System will be chemically decontaminated in accordance with the instructions in Technical Special Test Instruction (TSTI) 1B33-95-004-0-S. This safety evaluation documents the safety review of this change. The boundaries, requirements and criteria for the chemical cleaning and decontamination are separately reviewed by Safety Evaluation SE-94-026-R00.

REASON FOR CHANGE: The chemical cleaning and decontamination is being undertaken to resolve the dose rate for personnel servicing or in the vicinity of the systems.

SAFETY EVALUATION: The conclusion of this safety evaluation is that the performance of this change does not require a change to the GGNS Unit 1 Technical Specifications nor does performance of the change involve or raise any unreviewed safety questions.

The chemical decontamination efforts addressed in this change are being accomplished only in Operational Modes 4 or 5. The reactor vessel head will be removed and reactor pressure at atmospheric. No safety limits are affected or reduced as a result of performing this change to chemically decontaminate the recirculation loops.

Serial Number: 95-055-PSE Document Evaluated: Temp Alt 95-0005

DESCRIPTION OF CHANGE: This change will install temporary spool pieces on suction side of TURB BLDG CHLD WTR PMP A (1P71-C004A), AUX BLDG CHLD WTR PMP A (1P71-C003A) AND D/W CHLD WTR PMP A (1P72-C001A), to allow for cross-tie of plant chilled water and drywell chilled water to temporary chillers located in the yard, north of the Water Treatment Building during the PSW outage in RF07.

REASON FOR CHANGE: Because of the extreme heat stress that can be experienced during a plant service water outage due to losing both P71 and P72, a decision was made to provide temporary cooling to the Control Building, containment, drywell and Auxiliary Building during RF07.

SAFETY EVALUATION: The changes described herein 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 9.2.7 and 9.2.11 of the UFSAR other than the containment, Auxiliary Building, and drywell isolation valves neither P71 nor P72 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. Therefore, cross-tying the two systems while still maintaining their individual isolation capabilities will not affect any safety related equipment or system.

Three temporary chillers of Freon R22 (460# per chiller) will be located outside on the yard (Elevation 133'). The amount of freon in the temporary chillers and the location of the temporary chillers relative to the Control Room intake structure will meet the requirements of EERR 93/6145. In case of accidental release of one chiller, there is no direct pathway for the release of gas to diffuse to the Control Room air intake located in the Control Building roof (Elevation 206'). Therefore, the gas concentration at the air intake will be minimal.

Fill and makeup water to the chilled water loop will be provided by connection to the P21 system via 1P21F137 and 1P21F026. A check valve will be installed to prevent any possible back flow. Connection to the P21 system will therefore not introduce any unreviewed safety question.

Section 10.4.5 of the UFSAR discusses the effects of flooding of the Turbine Building due to a pipe break in the Circulating Water System. The volume of 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.

Serial Number: 95-056-NPE Document Evaluated: MNCR 93-0084

DESCRIPTION OF CHANGE: An underground fire water main supplying water to fire hydrants located in the northwest lay down area and fire training facility was not previously identified on fire protection P&IDs (Drawings M-0035 series) or fire water loop drawings (Drawing C-0035 series). This underground connection (including post indicator isolation valve, underground fire main, and fire hydrants) is connected to a dead end underground fire main which is connected to the plant's P64 Fire Protection Water System. This connection is being accepted-as-is and necessary drawing changes made. Fire protection P&IDs are included in the UFSAR in Figures 9.5-001 through 9.5-008E and Figure 9.5-008 (Drawing M-0035H) will be revised to show this connection and isolation valve.

REASON FOR CHANGE: The above described underground connection has been installed and in service since initial construction of the plant. The connection and isolation valve are being added to drawings to give plant operators necessary information for isolating this line when required.

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

This change will not increase the probability or consequences of accidents or malfunction of equipment important to safety previously evaluated in the UFSAR. 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 UFSAR. "Fire suppression systems" and "yard fire hydrants and hydrant hose houses" are not addressed by technical specifications. However, both are addressed in the Technical Requirements Manual (Sections 3/4.7.6.2 and 3/4.7.6.6 respectively). The connection and related equipment discussed above do not provide protection for equipment required by the Fire Protection Program or TRM; therefore, no change to the TRM is required.

Serial Number: 95-057-NPE Document Evaluated: EER 95/6028

DESCRIPTION OF CHANGE: This change response will provide the design requirements for temporary removal of six snubbers and one spring hanger to allow performance of ISI weld inspection on the "A" Recirculation System piping. Additionally, this response considered the removal of the temporary lead shielding which was installed on the discharge and suction piping per TSR 93-03, and the addition of temporary lead shielding on the discharge header and a maximum of 90 pounds of lead shielding on recirculation RPV nozzles.

REASON FOR CHANGE: The performance of an ISI weld inspection on the Recirculation System piping, Loop A, requires that six snubbers and one spring hanger be temporarily disassembled to allow inspection access to pipe welds and the addition of temporary lead shielding on the discharge header because of ALARA concerns during RF07.

SAFETY EVALUATION: The modifications made by this EER will not affect the system function, operation, or performance in any way. The capability of the system to perform its safety function is not affected. Temporary removal of six snubbers, one spring hanger and addition of temporary lead shielding will not adversely affect the structural integrity of the associated piping. The piping and pipe support designs meet ASME Section III requirements and are qualified as Seismic Category I (References 1 & 7). Therefore, the piping and pipe supports will function in their intended manner.

The modification will not introduce any new postulated piping failures and the existing hazards evaluations are not affected. No new system interfaces with any equipment have been created and no existing interfaces have been adversely affected. No new failure modes for the system or any equipment have been created. By remaining within the same allowables specified by the applicable codes as stipulated for piping, supports, and supporting structures, the margins of safety provided by these allowables are not affected.

Serial Number: 95-058-NPE Document Evaluated: MNCR 93-106

DESCRIPTION OF CHANGE: This change documents that piping assumed to be insulated in HVAC calculations is uninsulated. The change was dispositioned to "ACCEPT-AS-IS" a portion of the uninsulated piping and to "REPAIR" by insulating the remainder of the piping. The HVAC calculation has been revised to reflect these conditions. These results in the heat loads for the RHR A, B, C, and LPCS room coolers increasing above the values currently stated in UFSAR Table 9.4-7.

REASON FOR CHANGE: This change documents that piping assumed to be insulated in HVAC calculations is uninsulated. During the resolution of this change, the HVAC calculation was revised to reflect the conditions in the rooms after the repairs to the insulation are made. As a result of this, the loads for the RHR A, B, C, and LPCS room coolers increased to values above those currently shown in Table 9.4-7.

SAFETY EVALUATION: This change documented that piping in the RHR A, B and C pump and heat exchanger rooms, the RCIC room, and the LPCS pump room that was assumed to be insulated in the HVAC calculations was not insulated. Portions of the piping were insulated and portions were accepted without insulation. The HVAC calculation was revised to reflect the final configuration. This resulted in heat loads for the RHR A, B, C, and LPCS rooms that are in excess of the loads currently reflected in UFSAR Table 9.4-7. However, this load is within the capacity of the RHR A, B, C, and LPCS room coolers. The loads for the remaining cooler, RCIC, is unchanged (by insulating the pipe originally assumed to be insulated) from the values currently shown in Table 9.4-7.

Serial Number: 95-059-PSE   Document Evaluated: CI 47371 & MWP 19940004

DESCRIPTION OF CHANGE: During RFO7 work is scheduled to begin in the Division III (SSW C) fan tower cells while Division I (SSW A) is declared operable. Because the Division III fan tower cells are adjacent to the Division I cells and are over the SSW A basin, an evaluation must be performed to determine if the work activities will adversely affect operation and the operability of the SSW A system.

REASON FOR CHANGE: MNCR 0181-93 and DCP 94/0004 require replacement of various hangers and painting in the HPCS SSW fan tower cells during RFO7. Due to the start of the HPCS outage in RFO7, the Division III fan tower work must begin while Division I is declared operable and SSW A is required to be ready to maintain safe reactor shutdown.

SAFETY EVALUATION: Work activities in the Division III fan tower cells will not affect any technical specification parameters, limited condition of operations, or any safety limits of Technical Specifications 3.5.2 (ECCS-Shutdown), 3.8.2 (AC Sources-Shutdown), and 3.9.8.8 (RHR-High Water Level), TRM TR 3.7.1 (SSW & UHS-Shutdown), or any other technical specification.

Due to the redundant design of the SSW system and to the separation of the Division I and Division III fan tower cells, the work activities will affect only the components of the Division III fan tower and in no way will affect operation of the Division I fans.

Polyethylene lining will be placed on top of the Division III fan tower fill (ceramic brick) and then plank board will be placed over it. If the protective lining was to fail, the size of items that could fall into the basin would be limited to the size of the holes in the ceramic brick (approximately 3" x 1/2"). Due to the location of the Division III fan tower cells (north & southeast quadrants of SSW A basin), the probability of these items being pulled into the SSW A pump (southwest quadrant) is extremely low due to the low fluid velocities that would be present in those quadrants while the pump was operating and due to basin layout. Additionally, the pump's suction screen would block out any items that could be potentially damaging to the pump or system components. Therefore, working in the Division III fan towers will not adversely affect the operation of the SSW A pump.

The work in the Division III fan tower cells will not affect the operation of the SSW radiation monitors and will not prevent the operator from performing applicable corrective actions in case of an alarm. Additionally, the work activities will not adversely affect the capacity of the SSW A basin to maintain a barrier from the environment and will not increase the possibility of an unmonitored release to the environment.

Therefore, working in the Division III fan towers while Division I is declared operable will not increase accident or malfunction probabilities or consequences, will not create any risk of a different type of accident or malfunction, and will not reduce the margin of safety as described in the technical specification bases. Therefore, the work activities will not create an unreviewed safety question.

Serial Number: 95-060-NPE Document Evaluated: MCP 94/1055

DESCRIPTION OF CHANGE: MNCR 0127/94 documented generator rotor cooling water leakage that caused a mass imbalance in the generator in the Fall of 1994. The mass imbalance was temporarily corrected by drilling a hole in the cooling water piping inside the rotor 180 degrees from the leaking pipe in order to re-establish a mass balance on the rotor. This change provides a permanent design change to the rotor water pipes by replacing the pipe design with a rotor bore liner design. The liner is installed by rolling a section of pipe inside the rotor bore similar to rolling a tube to tubesheet connection in a heat exchanger. This change will also provide the design documentation for replacing the originally designed 18%Mn/5%Cr (18/5) rotor coil and waterbox retaining rings with new stress corrosion crack insensitive 18%Mn/18%Cr/(18/18) material. The original 18/5 retaining ring material has been found by the utility industry to be highly susceptible to IGSCC. This change also provides the design documentation to install teflon hoses for the rotor cooling water supply and return lines. All the rotor upgrade work described in this change has been performed by the OEM (Siemens Power Corp.) at the SPC repair facility in West Allis, Wisconsin on a spare rotor. The upgraded, spare rotor is to be installed during RF07, at which time the rotor with the failed water pipes will be removed from service.

REASON FOR CHANGE: The reason for changing the design of the main generator rotor water passageway pipes is to prevent future primary water leakage into the rotor bore and potentially into the generator. One of the original rotor water passageway pipes developed a pressure boundary leak, allowing cooling water leakage into the rotor bore and imbalancing the generator during the Fall of 1994. The generator imbalance resulted in a turbine trip. The rotor coil and waterbox retaining ring material change from 18%Mn-5%Cr (18/5) to 18%Mn-18%Cr (18/18) will prevent IGSCC of the retaining rings. The new teflon hoses between the cooling waterbox and rotor coils will extend the time required between generator inspections due to the improved material.

SAFETY EVALUATION: The generator modifications do not alter the existing turbine/generator missile analysis since the LP turbine blades remain as the limiting failure components in a turbine overspeed condition. The existing generator retaining ring burst analysis indicates that a ring burst would be contained by the massive generator casing. The existing generator rotor failure analysis indicates that a failed rotor would be contained by the stator core. The modification does not require a change to the technical specifications or to the UFSAR. The modification will not increase the probability of occurrence or increase the consequences of an accident previously evaluated in the UFSAR. The generator is not safety related and is not seismically qualified.

Serial Number: 95-061-NPE   Document Evaluated: MNCR 0036/94

DESCRIPTION OF CHANGE: The final disposition of MNCR 0036/94 is to repair the SSW pump shaft coupling fasteners by using monel cap screws instead of carbon steel cap screws. The design change will require the use of stainless steel lock washers on the coupling's cap screws. The design change will also prohibit the use of carbon steel lock washers on the coupling's cap screws. The final MNCR disposition also requires that the SSW pump's carbon steel suction bell and impeller housings be replaced on a six year interval. The design changes and rework intervals apply to both the "A" and "B" SSW divisions. The disposition further places requirements to inspect the HPCS-SSW pump, and one Byron Jackson ECCS deep draft pump for corrosion.

REASON FOR CHANGE: MNCRs 0030/94 and 0036/94 documented SSW pump impeller wear due to corroded cap screws and lock washers on the pump shaft couplings. The design change of the cap screw and lock washer material is being made per the final disposition of MNCR 0036/94 to eliminate the potential for corrosion on the shaft coupling fasteners. The design change was recommended by the original equipment manufacturer, Goulds Pumps, Inc. The final disposition of this change further requires inspections of additional pumps as part of the preventative maintenance program for safety related deep draft pumps at GGNS.

SAFETY EVALUATION: The design changes will maintain the SSW pump reliability by eliminating the potential for corrosion on the pump shaft coupling fasteners. The replacement of the carbon steel suction bell and pump bowls on a periodic basis will maintain the reliability of the SSW pumps by maintaining the pumps in the state to which they were originally designed. The inspections required by the disposition support the existing plant preventative maintenance program for safety related deep draft pumps. The design changes, periodic rework, and inspection program do not require a change to the GGNS Technical Specifications, will not increase the consequences nor increase the probability of occurrence of an accident previously evaluated in the UFSAR, will not increase the consequences nor the probability of occurrence of a malfunction of equipment important to safety previously evaluated in the UFSAR, will not create the possibility for an accident of a different type than any previously evaluated in the UFSAR, will not create the possibility for a malfunction of equipment important to safety of a different type than any previously evaluated in the SAR, and will not reduce the margin of safety as defined in the basis for any technical specification.

Serial Number: 95-062-NPE   Document Evaluated: DCP 93/0028-1

DESCRIPTION OF CHANGE: This change will install recirculation piping for the zinc injection system to be installed at a later date. The supply line consists of a 2" line routed from 18" DBD-26 to a normally accessible area in Area 6 of the Turbine Building at floor Elevation 133'-0". The return line will be routed from 30" FBD-23 to a normal accessible area in Area 6 of the Turbine Building at floor Elevation 113'-0". These lines will have an isolation valve at the large pipe header and there will be double isolation valves and pipe caps at the termination points.

SCN 95/0007A to MS-02 adds service numbers 233 and 234 for the DBD piping and service number 59 for the FBD piping to be installed. The service number will provide the design pressure rating, service conditions, seismic category, insulation class, special consideration and a description of the pipe service.

REASON FOR CHANGE: The risk of Inter-Granular Stress Corrosion Cracking (IGSCC) on reactor vessel nozzles and internals increases annually from the date of initial vessel service. Hydrogen Water Chemistry (HWC) has been proven to reduce the risk of reactor vessel IGSCC, but causes the drywell dose rate to increase. If HWC is implemented at GGNS, some method of dose rate control is needed to limit the drywell dose rates.

The zinc injection system (new GGNS System Number P85) will be installed at a later date. However, the interface piping is located in areas of the Turbine Building that are inaccessible during normal plant operation. Therefore, this change supplement is being issued to install, during RFO7, the portion of the recirculation piping that will be located in the inaccessible areas.

SAFETY EVALUATION: The addition of the zinc injection recirculation piping will not affect the integrity, function, operation, or performance of the Feedwater System (N21) in any way or any safety system. The piping and pipe support designs meet the ANSI B31.1 code requirements and the piping is supported for the appropriate loads.

Analysis of the piping system has shown that all code allowables have been met; therefore, the probability of piping failure has not increased. The modification will not introduce any new postulated piping failures and the hazards evaluations are not affected. Therefore, the probability of occurrence of a malfunction of equipment important to safety previously evaluated in the UFSAR, either as a consequence of a piping system failure (such as caused by jet impingement or pipe whip due to pipe break) or in response to any such postulated event (where a single failure is considered), is not increased.

The aforementioned piping changes will not create an unreviewed safety question. The technical specifications are not affected, and the margin of safety remains unchanged.

The penetration changes will not affect the technical specifications or the bases for any technical specifications because Turbine Building radiation boundaries are not controlled by the technical specifications. The penetration seal material meets radiation boundary requirements given in UFSAR Sections 12.3. This change will not increase the probability for occurrence of accidents or malfunction of equipment important to safety previously evaluated in the UFSAR because the penetration seals provided are designed to meet the functional requirements of UFSAR Section 12.3.1.1.2.c for radiation boundaries. This change will not increase the consequences of an accident or malfunction of equipment important to safety from that previously evaluated in the UFSAR because appropriate design requirements have been provided to comply with UFSAR Section 12.3. 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 UFSAR because the penetration seal detail provided has been previously evaluated and approved as a radiation seal.

Serial Number: 95-063-NPE Document Evaluated: N/A (Cycle 8 Reload Analysis)

DESCRIPTION OF CHANGE: This change assesses the changes associated with the Cycle 8 reload fuel and operation with the Cycle 8 core configuration. 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. Like Cycle 7, the Cycle 8 core will consist of a full core of Siemens Power Corporation (SPC) 9x9-5 fuel assemblies. Attachment 1 provides a detailed description of the Cycle 8 core and the issues considered in this evaluation.

REASON FOR CHANGE: The Cycle 8 reload fuel consists of 288 SPC 9x9-5 fuel bundles necessary for Cycle 8 operation.

SAFETY EVALUATION: This change concludes that the core changes associated with the Cycle 8 reload and operation: (i) will require no changes to the current GGNS Technical Specifications, and (ii) will not constitute an unresolved safety question. The Cycle 8 core has been shown to meet all requirements in the GGNS Technical Specifications, GGNS UFSAR, 10CFR, and the Standard Review Plan.

Serial Number: 95-064-NSRA Document Evaluated: Drywell Personnel Airlock Door Seal Air System Leak Testing Criteria

DESCRIPTION OF CHANGE: This change is being prepared in order to change the acceptance criteria for the Drywell Airlock Seal Air System test. Therefore, the 2 psig decrease in a 48 hour period will be increased to a value more commensurate with the actual safety function of the drywell airlock. The proposed leakage rate is not to exceed 30 psig in a 24 hour period.

REASON FOR CHANGE: The current test requires the airlock to be subjected to stringent acceptance criteria that exceeds its actual safety function. The current acceptance criteria for the Drywell Airlock Seal Air System leak rate were arbitrarily chosen due to the similarities of the containment airlocks to the drywell airlock. The function of the containment system is different from the drywell; therefore, the testing criteria for the associated airlocks should be different. The safety function of containment is to remain essentially leak-tight for 30 days. However, the safety function of the drywell will be completed well within this period of time.

SAFETY EVALUATION: This evaluation concluded that the change did not involve an unreviewed safety question. The change will not increase the potential for an accident nor will it result in the malfunction of plant equipment. Even though this change increases the acceptable leakage rate for drywell airlock seal testing, all safety functions will be performed as required. The function of the drywell is completed within twenty-four hours and any increase in drywell bypass from the drywell airlock will not impact the progression or consequences of an accident with regard to containment peak pressure, hydrogen control, equipment qualifications or accident radiological consequences. Therefore, this change would not reduce the margin of safety provided by the airlock system.

The proposed change to the TRM will change the acceptance criteria for the Drywell Airlock Door Seal Air System test from a test that ensures the system will be pressurized for 30 days to one that ensures the system will be pressurized for 24 hours.

Serial Number: 95-065-NPE Document Evaluated: MCP 95/1027

DESCRIPTION OF CHANGE: This change removes valve QSZ51F025B from the Control Room HVAC unit QSZ51B002B-B and replaces it with carbon steel pipe. The piping is fabricated from socket welded fittings and schedule 80 pipe.

REASON FOR CHANGE: As documented in MNCR 95-0133, a freon leak developed at a screwed connection, copper to carbon steel, downstream of valve QSZ51F025B. A replacement valve and associated fitting could not be acquired. The maintenance valve was replaced with carbon steel pipe.

SAFETY EVALUATION: The valve that is being replaced, QSZ51F025B, is installed to facilitate maintenance of the Control Room HVAC unit. The valve serves as an isolation valve between the condenser and the compressor. Removal of the valve will not degrade system performance and the ability to isolate the condenser from the compressor is maintained by a valve integral to the compressor. The valve is being replaced with piping fabricated to the equivalent of GBC piping per MS-03. The materials used to fabricate the replacement piping will neither adversely affect the refrigerant nor be adversely affected by the refrigerant. The materials used to fabricate the replacement piping are of greater strength than the copper tubing originally installed. The piping will be designed to the requirements of ASME Section III 1974 Edition through Summer 1975 Addenda. The seismic qualification of the Control Room HVAC unit will be maintained by replacing the valve with carbon steel piping.

Serial Number: 95-066-PSE   Document Evaluated: Change Request 95-036

DESCRIPTION OF CHANGE: UFSAR Section 18.1.34 lists requirements for the integrity of systems outside containment likely to contain radioactive material for pressurized water reactors and boiling water reactors. The paragraph in this section dealing with water leakage requires observable leakage past vent and drain valves be eliminated. The paragraph dealing with gas leakage requires any detected leakage be eliminated. In each instance, the word `eliminated' will be replaced with `reduced to as-low-as-practical levels'.

REASON FOR CHANGE: Technical specifications and the UFSAR establish release limits for water and gas leakage to the environment. The requirement to totally eliminate specific leakage paths can cause significant system intrusions with little appreciable affect to total leakage rates.

SAFETY EVALUATION: This change corrects an overly-burdensome requirement and is not contrary to the letter or spirit of technical specifications or applicable NUREGs. Other testing already in place determines collective leakage for water and gas systems, with corrective action required for leakage above an acceptable amount. No unresolved safety questions are introduced, and this change would not affect the impact of postulated accidents.

Serial Number: 95-067-NPE Document Evaluated: CN 95-0061

DESCRIPTION OF CHANGE: This change clarifies test requirements for valve 1E51F021 and resizes orifice 1E51D005 to ensure proper flow rates are maintained in the RCIC minimum flow line.

REASON FOR CHANGE: The original purpose of this change was to convert valve 1E51F021 from a check valve to a globe valve and maintaining it in a locked open. This was an effort to delete the valve from the pump and valve program, thereby eliminating valve disassembly and demonstration of operational requirements every outage.

Based on information available at the time of IFC of the design change NPE required, in Sections 7.4 and 8.9 of MCP 91/1031, that the valve be tested to ensure that flow obtained before modification of the valve was maintained after the change (90 gpm + 10% with the RCIC pump at maximum speed). Current requirements for the RCIC minimum flow line, based on vendor recommendations, indicate that flow should be maintained at 95 gpm or higher for all pump operating conditions. Recent discussions with the system engineer have indicated that the minimum flow line, as it currently exists, does not meet the current vendor recommended flow requirements and is not expected to pass the designated flow test at less than maximum pump speed. Therefore, a design change to resize orifice 1E51D005 is required to obtain the required flow in the minimum flow line.

SAFETY EVALUATION: The affected system is E51, Reactor Core Isolation Cooling System (RCIC). The piping modified is located in the Auxiliary Building, Elevation 93'-0".

Increasing the bore size of orifice will not adversely affect the structural integrity of the E51 system. The piping and associated components have been designed to ASME Section III code allowables and the system is properly supported for the appropriate loads.

System operability is not affected by this change and its ability to function in the mitigation of accidents remains unchanged. Thus, this modification will not increase the consequences of any accident previously evaluated in the UFSAR.

This modification will not increase the probability of occurrence of a malfunction of equipment important to safety previously evaluated in the UFSAR. Thus, all intended functions of the E51 system will continue to be performed as designed, and there is no reduction in the margin of safety as defined in the basis for any technical specification.

The change will not affect the function or operation of system E51 or any other systems and will not create an unreviewed safety question. The affected orifice is not addressed in the technical specifications, therefore, no change to the technical specification is required.

Serial Number: 95-068-NSRA Document Evaluated: GGNS Procedure G2.501, Rev. 5

DESCRIPTION OF CHANGE: Staffing consolidation and reduction efforts have led to the elimination of the corporate environmentalist position. This position was included in the subject procedure as a member of the SRC. To reflect this change, a change to the procedure is required and to preclude having to change the procedure to reflect future consolidations and to better reflect the current UFSAR the generic title used in 13.1.1.3.6 will be utilized in the procedure.

REASON FOR CHANGE: This revision brings the procedure into compliance with the new TRM and revised TS after Amendment 120. Additionally, this revision will remove position specific titles for certain corporate positions on the SRC and make the procedure reflect the UFSAR. Currently, UFSAR Section 13.4.2 list the corporate environmentalist as an SRC member. This position no longer exist.

SAFETY EVALUATION: The effects of this change will not change the effectiveness of the SRC or reduce the level of qualifications of the members making up the committee. This change will make it less cumbersome to make organizational changes in the future by elimination of specific titles for certain corporate positions on the committee. The qualification requirements and expertise requirements currently specified by the TRM are not effected by this change. There will be no unreviewed safety question created by this change.

Serial Number: 95-069-NPE Document Evaluated: UFSAR CR 95-017

DESCRIPTION OF CHANGE: Plant inspections for the IPEEE have identified several safety related components and unshielded openings (in addition to those already described in UFSAR Table 3.5-8, Section 3.8.4.1.1.4.d, and Section 9.5.8.3) which are at least partially exposed and vulnerable to tornado generated missiles. These items will be added to the items described in the UFSAR Table 3.5-8.

QDR 0032-95 identified deficiencies in the description of the tornado missile evaluation/basis in the UFSAR. UFSAR Section 3.5.2.5 will be revised to provide the basis for the acceptability of components and openings which are potentially vulnerable to tornado generated missiles. UFSAR Section 3.8.4.4.5, Appendix 3A, and Appendix 3G will also be revised to provide appropriate references.

REASON FOR CHANGE: This revision of the UFSAR is being made to more accurately reflect current as-built conditions, delete the reference to the enclosure building from Table 3.5-8, and clearly state the basis for accepting unshielded components and openings which are potentially vulnerable to tornado missile hazards.

SAFETY EVALUATION: The Unit 2 enclosure building is not addressed in the GGNS Unit 1 technical specifications. Further, the Technical Specifications do not contain requirements related to plant design against postulated tornado effects or otherwise give details related to missile shielding. Therefore, the GGNS Unit 1 Technical Specifications need not be changed and margins of safety associated with their bases are not affected.

Calculation CC-Q1111-94004 demonstrates that the cumulative annual probability of a tornado missile strike and its effects on the vulnerable safety related targets, is 0.77×10^{-8}. Therefore, the annual probability of tornado missile damage to any of these targets is smaller than the screening value of 1×10^{-7} cited in NUREG 75/087, Section 3.5.1.4; NUREG 0800, Section 3.5.1.4; Regulatory Guide 1.117; NUREG 0831, Section 3.5.2; and EPRI Report NP-2005. Hence, the presence of the potentially vulnerable targets, and the absence of the Unit 2 enclosure building do not represent a deviation from existing regulatory requirements. Therefore, UFSAR revision to more accurately describe potentially vulnerable tornado missile targets, and the regulatory basis for their acceptance, does not increase the probability or consequence of UFSAR evaluated accidents resulting in radionuclide release. Similarly, it does not increase the probability or consequence of UFSAR evaluated malfunctions of equipment important to safety.

Serial Number: 95-070-PSE Document Evaluated: 03-1-01-6

DESCRIPTION OF CHANGE: To allow performing the reactor vessel in-service leak test with the disc removed from 1E12-F050B. The disc will be removed and the retaining plate/cap assembly will be installed as normal. Valve 1E12-F053B, the other RCS PIV in series with 1E12-F050B in the 12"-DBB-68 line, will become a test boundary valve. The 3/4"-DBB-66 test connection line from DBB-68 will also be pressurized and valve 1E12-F058B will become a test boundary valve. Reference P&ID M-1085A.

REASON FOR CHANGE: This will allow the reactor vessel in-service leak test to be performed while rework is being performed on the 1E12-F050B disc per MNCR 0184-95. Reference WO 144875.

SAFETY EVALUATION: Per operability review for MNCR 0184-95, 1E12-F050B is declared inoperable until repaired and retested satisfactorily. This safety evaluation assumes 1E12-F050B is inoperable and reviews only performing the reactor vessel in-service leak test with disc removed. With the disc removed but the retainer plate/cap assembly installed as normal, the pressure integrity of 1E12-F050B is still intact. However, without the disc, the normal test boundary must be moved upstream to the 1E12-F058B valves. This change in test boundary does not create an unreviewed safety question as summarized in the safety evaluation review.

Serial Number: 95-071-NSRA Document Evaluated: TRM Sections 7.4.1.2, 7.4.1.5, and UFSAR Section 13.4.1

DESCRIPTION OF CHANGE: Currently, language in Section 7.4.1.2 of the TRM states that the PSRC shall be composed of nine members of GGNS onsite operating organization management at the superintendent level or above. The GGNS UFSAR in Section 13.4.1 lists the nine individuals by position, that make up the PSRC. This proposed change would allow the addition of personnel as permanent/voting members of the PSRC.

This proposed change would modify the wording in TRM Section 7.4.1.2 to read, "...PSRC shall be composed of members of GGNS operating/engineering organization management ..."

The UFSAR Section 13.4.1 would be modified to add the position of Manager, Design Engineering to the list of PSRC members. The heading for UFSAR Section 13.4.1 will be changed to read, "...shall be composed of at a minimum, the following individuals."

Additionally, Plant Procedure 01-S-01-3 will be changed to reflect the TRM and UFSAR changes.

REASON FOR CHANGE: The current requirements for the composition of the PSRC do not contain provisions for adding members. The current language does not allow GGNS Design Engineering management personnel with expertise that may be of operational value to the plant to have a vote on the PSRC. Therefore, this change is proposed to allow the addition of GGNS operating/engineering management personnel, such as Design Engineering, that could have significant input to the safe operation of GGNS as voting members of the PSRC.

SAFETY EVALUATION: The addition of voting members to the PSRC is administrative in nature and will not increase the probability of occurrence or the consequences of any accident or equipment malfunction previously evaluated in the UFSAR. Neither would this change create the possibility of an accident or equipment malfunction of a different type than any previously evaluated in the UFSAR. This change would not affect the minimum requirements for membership on the PSRC as stated in the UFSAR or TRM. Therefore, no unresolved safety questions would result from this change.

Serial Number: 95-072-NPE   Document Evaluated: MCP 95/1019, SCN 95/0008A (MS-02)

DESCRIPTION OF CHANGE: This change will provide the work instructions to cut, cap and abandon in place the small bore drain lines off the primary lines from the 1st stage and 2nd stage reheater drain tanks. The lines were originally installed to provide drainage for an Auxiliary Steam System blanket for MSR lay-up during extended outages. Also the drain lines that are currently connected to the associated steam traps for these lines are also being cut, capped and abandoned in place. The affected lines are installed in the Moisture Separator-Reheater Vents and Drains System (N35). Reference: 1"-DBD-125, 1"-DBD-126, 1/2"-DBD-127, 1"-GBD-158, 1"-GBD-159, and 1/2"-GBD-160.

REASON FOR CHANGE: The referenced lines above are not used during normal plant operations and were intended to be used as drain lines when an auxiliary steam blanket was required for the MSRs during extended outages. This change documented three leaks in line 1"-GBD-158. This line was found to be experiencing severe pipe wall thinning in certain areas. An on-line leak repair was required to stop the worst leak located on the eroded 90° elbow. At that time it was decided that all lines would be cut, capped and abandoned in-place along with the drain lines connected to the steam traps located in these lines. The reason these lines experienced pipe wall thinning was seat leakage through the isolation valves which in turn introduced flow into a normally isolated line. By capping these lines at their main header connection points, future steam leaks have been eliminated.

SAFETY EVALUATION: Since these lines were only required to provide a drainage path for an auxiliary steam blanket for MSR lay-up during long outage periods and are not required during normal plant operation, the cutting, capping and abandoning in-place of these lines will not have any adverse affects on any safety related system or subsystem required for GGNS safe shutdown. These changes will not have any affect on current plant operation or change the current operation of the N35 system or change current operation of any other plant system during normal plant operation. Therefore, these changes will not reduce any margins of safety discussed in the GGNS Technical Specifications or the latest revision of the GGNS UFSAR.

Serial Number: 95-073-NPE Document Evaluated: DRN 4916

DESCRIPTION OF CHANGE: This change corrects the Siemens tag numbers associated with valves 1N11F304A and 1N11F304C as shown on P&ID M-1051C, Revision 14 (UFSAR Figure 10.3-3). This change does not require any physical work or change to the facility.

REASON FOR CHANGE: The present association between Siemens valve tags and Bechtel valve numbers for valves 1N11F304A and 1N11F304C, as depicted on P&ID M-1051C, Revision 14, is not consistent with other plant configuration information as well as with existing actual plant configuration. Based on evaluation performed during the disposition of MNCR 95-0105, Supplement 1, it was confirmed that Siemens tag SA11S008 should be associated with Bechtel valve number 1N11F304A and tag SA11S006 with 1N11F304C. This association is not properly reflected on the P&ID (UFSAR Figure 10.3-3), thus requiring revision per DRN 4916.

SAFETY EVALUATION: It has been determined, via proper valve operation and MNCR 0105-95, that the valve association as listed per the Siemens documents is correct and the association as shown on P&ID M-1051C (UFSAR Figure 10.3-3) is incorrect. There is no change to the facility as a result of this change. All work has been evaluated and performed by MNCR 0105-95. This change will reflect the proper valve tag association. Since this change does not require a facility change, no increase in the probability or occurrence of an accident will occur, no new or unevaluated accident will be created, nor will any margin of safety be affected.

Serial Number: 95-074-PSE Document Evaluated: Change to TRM Tables TR3.6.1.3.1-1 and TR3.6.4.2-1

DESCRIPTION OF CHANGE: This change revises the maximum allowable stroke times for various primary and secondary containment isolation valves listed in Technical Requirements Manual (TRM) Tables TR3.6.1.3.1-1 (Primary Containment Isolation Valves) and TR3.6.4.2-1 (Secondary Containment Isolation Valves). It does not revise times for drywell isolation valves or any other valves.

REASON FOR CHANGE: Engineering Evaluation Report (EER) No. 94/6182 was initiated because Specification GGNS-M-189.1, GGNS Unit 1 Pump and Valve Inservice Testing Program, Appendix A, "Bases for Maximum Stroke Times of Power Actuated Valves," implies that numerous valves in the Inservice Testing (IST) Program have analytically-based maximum stroke time limits. Although some of the valves listed in Appendix A have paragraphs in the Safety Analysis Report (SAR) listed, for which explicit time limits are given, many of the valves are identified only as having limits in either GGNS Technical Specifications Table 3.6.4-1 for primary containment isolation valves or Table 3.6.6.2-1 for secondary containment isolation valves. Tables 3.6.4-1 and 3.6.6.2-1 have been relocated to the Technical Requirements Manual (TRM) as Tables TR3.6.1.3.1-1 and TR3.6.4.2-1, respectively.

In "Detailed Description of Problem," the EER noted that many of the valves listed in TRM Tables TR3.6.1.3.1-1 and TR3.6.4.2-1 do not have analytical safety bases for the listed maximum allowable stroke times. In many cases, the listed maximum allowable stroke times for these valves had been arbitrarily determined based on either previous performance data or on information supplied by the manufacturer based on performance of typical valves of the same model/size.

Due to maintenance, design changes, and/or normal wear, performance of some valves has approached or exceeded limits in the TRM tables, resulting in additional man-hours and man-rem exposure to adjust their performance to again be within TRM limits.

NPE's response identified numerous valves with primary containment and secondary containment isolation functions for which the maximum stroke time limits in the TRM were shorter than could be justified by their primary or secondary containment isolation function. The valves considered in the TRM changes for which this safety evaluation is written are primary containment isolation valves with indirect pathways for leakage, for which the NPE response identified a 60-second closing time limit, and secondary containment isolation valves, for which the NPE response identified a 120-second closing time limit.

SAFETY EVALUATION: The safety evaluation concludes that increasing the maximum allowable isolation times for the primary and secondary containment isolation valves identified in these TRM changes to the isolation times supported by the UFSAR safety analyses does not increase the probability or consequences of an accident or malfunction of equipment important to safety. The times currently listed for these valves in the TRM tables identified above do not have any safety significance to the plant.

Performance of inservice stroke testing of these valves will continue in accordance with the GGNS Unit 1 Pump & Valve Inservice Testing Program (Specification GGNS-M-189.1), as required by ASME Boiler and Pressure Vessel Code (ASME Code), Section XI, "Inservice Inspection," which is referenced in and required by GGNS Technical Specification 5.5.6 and TRM 7.6.3.3. This practice will minimize the possibility of operating with degraded components, which will prevent an increase in the probability of occurrence and the consequences of malfunctioning equipment.

Serial Number: 95-075-PSE Document Evaluated: TRM 7.7.1.1

DESCRIPTION OF CHANGE: This change removes the commitment to submit a summary startup report to the NRC under certain conditions. This requirement is contained in Regulatory Guide 1.16, C.1.a, and appears in the Technical Requirements Manual 7.7.1.1 as well as in the UFSAR regulatory guide commitments listing. This report has typically been submitted following each reload and this would no longer be required.

REASON FOR CHANGE: The report contains only basic information and references other programs and reports available on site. NRC approval of startup tests is not required, and complete information is readily available in plant records of the various tests performed after each reload. Adequate programs are in place to ensure that testing is done and that results are analyzed and screened for non-conformances or other problems. Summarizing this information for review and filing by the NRC does not enhance plant safety. It does, however, require use of valuable plant time and resources.

SAFETY EVALUATION: This proposed change serves only to remove an administrative requirement associated with commitment to Regulatory Guide 1.16, C.1.a. Elimination of the summary startup test report currently sent to the NRC after each reload will in no way increase the probability or consequences of accidents or malfunctions previously evaluated. NRC approval of startup test results is not required by Regulatory Guide 1.16, and the report is submitted for information purposes only. No evidence could be found that the Commission in any way based the GGNS Safety Evaluation Report on a requirement to submit startup test results following each reload. No modifications to the facility or to the conduct or review of startup testing will be done under this change. No new types of events could be created by elimination of this report, nor is any margin of safety affected. Thus elimination of the TRM requirement for the summary startup test report and removal of the UFSAR commitment to comply with this aspect of Regulatory Guide 1.16 do not present an unreviewed safety question.

Serial Number: 95-076-NPE Document Evaluated: MNCR 0206-95

DESCRIPTION OF CHANGE: During RF07, this change documented that the seals in the Auxiliary Building railroad bay drain lines were not installed. Therefore, the change made by the second interim disposition of MNCR 0206-95 was to install seven short hollow stem test plugs in the floor drains located in the railroad bay in Area 10 Elevation 133' of the Auxiliary Building. The railroad bay is outside the boundary of the secondary containment and the seven floor drains in the railroad bay are connected to the drainage piping inside the secondary containment, therefore, the installation of the plugs will establish the leak tight seal for the secondary containment boundary. UFSAR Section 6.2.3.1.1 states "To preclude air inleakage into the SGTS region, the six-inch floor drains provided for the railroad bay in the Auxiliary Building have been permanently sealed. Therefore, these lines are not furnished with isolation valves." The installation of the test plugs will prevent inleakage into the SGTS region.

REASON FOR CHANGE: The reason for the installation of the temporary plugs is to establish secondary containment boundary and allow the plant to obtain modes 1, 2, and 3.

SAFETY EVALUATION: The change made will install test plugs rated at 42 psi in the drain piping located in the railroad bay of the Auxiliary Building. The plugs were evaluated for the existing environmental conditions and based upon engineering judgment, the structural integrity of the plugs will last for 45 days. The maximum differential pressure the plugs will have to overcome will be 3 psi due to tornado depressurization loads. Adding the plugs to the drain lines maintains the secondary containment boundary and will not impact the technical specification. There are three principal accidents for which credit is taken for secondary containment operability. These are a LOCA, a fuel handling accident inside the primary containment and a fuel handling accident in the Auxiliary Building. The secondary containment performs no active function in response to each of these limiting events; however its leak tightness is required to ensure that the release of radioactive materials from the primary containment is restricted to those leakage paths and associated leakage rates assumed in the accident analysis, and that fission products entrapped within the secondary containment structure will be treated by the SGT system prior to discharge to the environment. Therefore, for this time frame, the plugs do not increase the probability of the occurrence or consequences of an accident or malfunction of equipment important to safety. Blocking the drain lines does not create the possibility of an accident or malfunction of a different type than evaluated previously in the UFSAR because UFSAR Section 6.2.3.1.1 states that the lines are to be sealed. This change will not require any system to be operated abnormally therefore, adding the plugs will not reduce the margin of safety as defined in the basis for any technical specification.

Serial Number: 95-077-PSE Document Evaluated: UFSAR Tables 6.2-44, 6.2-49 and 16B-3.6.4-1

DESCRIPTION OF CHANGE: This change deletes the requirements to perform Type C local leak rate testing on nine test connection valves listed in UFSAR Tables 6.2-44 and 6.2-49 and in Technical Requirements Manual (TRM) Table TR3.6.1.3-1 (formerly Technical Specification Table 3.6.4-1). The nine valves are the following:

Valve No.	Penetration No.	Penetration Service
1B21F025A	5	Main Steam A
1B21F025B	6	Main Steam B
1B21F025C	7	Main Steam C
1B21F025D	8	Main Steam D
1B21F030A and F063A	9	Feedwater A
1B21F030B and F063B	10	Feedwater B
1E51F072	17	RCIC Steam Supply

Various directives associated with the local leak rate testing program will also be revised as a result of this change.

REASON FOR CHANGE: The test connection valves are not required to be Type C local leak rate tested because they do not conform to the characteristics of valves that are required to be Type C tested under the definition of "Type C Test" as defined in 10CFR50, Appendix J, Definition II.H. They are small manual valves, are locked in closed position during power operation, are operated infrequently, and are not capable of remote or automatic operation. In addition, because these test connection pipes attach to their process pipes between inboard and outboard main isolation valves and have additional valves and pipe caps in series, these penetrations present multiple independent barriers to leakage through the penetration.

Eliminating the Type C tests of these test connection valves will save significant outage time, man-hours and man-rem exposure.

SAFETY EVALUATION: The change concludes that neither the probability nor the consequences of an accident or malfunction of equipment will be increased by exempting the local leak rate testing. Local leak rate testing is not effective in detecting mispositioned valves, which is the only likely reason that the probability of an accident or malfunction of equipment would be increased. The consequences of an accident or malfunction of equipment are minimized by the valves' construction, their infrequent operation, and administrative controls on the valves' disk positions.

Although these valves seem to be specifically identified in Appendix J, Definition II.H, as requiring Type C testing, this evaluation justifies why they should be exempted. In addition, GIN-95/01815 documents Nuclear Safety & Regulatory Affairs (NS&RA) position that these changes may be made under the provisions of the 10CFR50.59 program.

Serial Number: 95-078-NPE Document Evaluated: UFSAR CR NPEFSAR-92/0023

DESCRIPTION OF CHANGE: This change addresses the UFSAR Change Request to add the latest edition of Uniform Building Code (UBC) for design of the modified and new non-Category 1 structures whose failures do not impact any Category 1 structures.

REASON FOR CHANGE: Currently design of non-Category 1 structures at GGNS is performed in accordance with the UBC, 1970 Edition, as required by UFSAR Section 3.7. Since some of the code seismic requirements have been revised from the 1970 Edition, the modified and new non-Category 1 structures may be designed to the latest edition of code to improve seismic design evaluations.

SAFETY EVALUATION: The proposed change has been initiated to add the latest edition of UBC for the design of modified and new non-Category 1 structures. The latest revision of UBC contains the revised seismic requirements for the seismic design evaluation. Prior to initiating the subject UFSAR change, a review was done to ensure the ability of the existing seismic design of non-Category 1 structures to meet the requirements contain in the latest edition of UBC. The conclusion of review is that the existing design is unaffected by the code latest edition. On the basis of this review and the fact that the failure of non-Category 1 structures does not affect Category 1 structures, it is concluded that the change described in this UFSAR change request does not increase the probability or the consequences of any accident evaluated in the UFSAR, do not create the possibility of a new accident or malfunction, and do not reduce any margin of safety defined in any technical specifications.

Serial Number: 95-079-NPE   Document Evaluated: Calc XC-Q1111-92010, Rev. 3

DESCRIPTION OF CHANGE: QDR 0049-95 identified a deficiency in the GGNS LOCA dose analysis, Calculation XC-Q1111-92010, Revision 2. This change identified the drywell bypass flowrate in the Bechtel calculation supporting the LOCA model was significantly less than that 35,000 scfm predicted for a 3 psi drywell-containment pressure differential. This 35,000 scfm parameter was also reported in the bases to the technical specifications. This revision of the LOCA dose analysis also considered the following improvements to the analysis:

- changes to the iodine spray removal coefficients based on the guidance in SRP 6.5.2 and NUREG/CR-5966,
- revised containment mixing rates based on the guidance in NUREG/CR-0304,
- revised MSIV leakage delay time,
- consideration of leakage from ESF systems outside containment that were not previously addressed,
- evaluation of 1 gpm HFTS leakage, and
- increased drywell personnel airlock door seal leakage after 24 hours.

As part of the corrective actions to QDR 0049-95, a revision to the calculation and an UFSAR change request have been prepared. This change supports these documents.

REASON FOR CHANGE: This change supports Revision 3 to Calculation XC-Q1111-92010 and UFSAR Change Request 95-053.

SAFETY EVALUATION: This change confirms that, although some of the doses were calculated to increase, the offsite and control room doses still remain below the allowable regulatory limits reported in 10CFR100 and GDC 19.

Serial Number: 95-080-NPE Document Evaluated: DCP 91/0098

DESCRIPTION OF CHANGE: DCP 91/0098 provides design documentation for the following:

a. Installation of a PSW supply line for the primary source of circulating water lube water (CWLW).

b. Installation of a CW supply line for the backup source of CWLW.

c. Installation of two 100% capacity basket strainers piped in parallel in the common portion of the CW supply to remove large solids of a low specific gravity that could degrade the performance of the centrifugal separators discussed herein.

d. Installation of a PCW supply and return line for thrust bearing cooling (TBC).

e. Installation of three new 7.5 hp lube water pumps to replace the two existing 2 hp lube water pumps.

f. Installation of a separator to remove solids of a high specific gravity and a basket strainer to remove solids which could pass through the separator.

g. Abandonment of the Domestic Water and Makeup Water Treatment System supplies for lube water and thrust bearing cooling.

h. Addition of a five minute time delay between the lube water flow low-low signal and a CW pump trip due to lube water lube water flow low-low.

i. Deletion of the existing auto start feature for the lube water pumps and the associated relabling of the control room handswitches.

j. Replacement of the low lube water flow alarm with a circulating water lube/cooling trouble alarm.

REASON FOR CHANGE: The circulating water pumps require external sources of clear cool water for cooling the Kingsbury thrust bearing and for lube water to the two cutless rubber lower bearings. The current source of water is the Domestic Water (DW) (P66) System with a backup supply from the Makeup Water Treatment (MWT) (P21) System. The MWT supply is routed from downstream of the charcoal filters but upstream of the Ionics trailer. These water supplies are inadequate.

SAFETY EVALUATION: The Circulating Water System is not addressed in the GGNS Technical Specifications; however, the condenser vacuum setpoint is addressed. None of the evaluated changes alters or affects the condenser vacuum low setpoint as addressed in Technical Specification 3/4.3.2 for main steam line isolation. The design change does not affect that part of the MWT system, the PSW system, the DW system, or the PCW system which is addressed in the technical specifications, specifically, valves forming a part of containment and/or secondary containment boundary.