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June 30, 1994

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, 1993 through December 31, 1993

GNRO-94/00094

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, 1993 through December 31, 1993. A summary of these changes, tests, and experiments is contained in the attachment. If further information is required, please contact this office.

Yours truly,

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

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10CFR50.59 SAFETY EVALUATIONS FOR THE PERIOD

JANUARY 1, 1993 THROUGH DECEMBER 31, 1993

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10CFR50.59 SAFETY EVALUATIONS FOR THE PERIOD

JANUARY 1, 1993 THROUGH DECEMBER 31, 1993

Serial Number: 90-001 Document Evaluated: MCP 89-1085-S00-R00

DESCRIPTION OF CHANGE: MNCR 241-89 documented the fact that 37 scheme cables were terminated in Computer Cabinet SC91-P008-2 but there was no corresponding computer wiring (i.e., the cables dead-ended in the computer cabinet).

A review revealed that each of the terminal locations listed had an assigned Liquid Radwaste System computer point number. These points, with a few exceptions, were intended to provide a digital high/low input from various level switches for which an analog level input with high/low computer setpoints already exists. The exceptions were the three instances which are all fed from conductivity analyzers, two instances in which the points are fed from pressure differential switches, and two instances in which the points are fed from flow switches.

For the cases in which the computer points are fed from conductivity analyzers and pressure differential switches, no existing computer point provides comparable information. The three points fed from the conductivity analyzers appear on the system P&IDs but the ones fed from the pressure differential switches do not.

For the instances in which the points are fed from a flow switch, there are comparable analog computer points in existence.

Section 11.2.2.5 of the UFSAR covers the instrumentation application for the Liquid Radwaste System. This section of the UFSAR states the equipment collector tanks, waste surge tanks, equipment drain sample tanks, floor drain collector tank, floor drain sample tanks, condensate demineralizer regeneration solution receiving tanks, miscellaneous chemical waste receiver tank, distillate sample tanks, and evaporator bottoms tanks are each provided with alarm points and computer logging for each excessively high or low tank level. This section also states that the flow through each filter shall be monitored by the plant computer and an excessively low flow will be alarmed and logged by the plant computer. This section further states that the radwaste demineralizers have an excessively high differential pressure alarmed and logged by the plant computer.

For all scheme cables in which the computer points have a comparable analog point in existence, the cables will be spared and all affected drawings will be revised to indicate this change. The analog points will have a high/low level alarms or a low flow alarm, as applicable.

For the scheme cables in which there is no comparable computer point existing, instructions will be provided to make these points active.

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REASON FOR CHANGE: The UFSAR must be revised to reflect changes made as a result of the disposition of the MNCR concerning the radwaste computer points. This is an evaluation of the necessary changes.

SAFETY EVALUATION: The computer points which do not alrea. ave comparable analog inputs will be made operable, and the ones with existing analog inputs will be designated as spare. These actions will ensure that the probability of an accident or malfunction will not be increased. These points have no active function in mitigating the consequences of an accident, therefore the consequences of an accident or malfunction are not affected. All monitoring originally intended to be performed will be accomplished with the revised configuration, so no margins of safety will be decreased.

Serial Number: 91-001-NPE   Document Evaluated: DCP 90-0313-S00-R00

DESCRIPTION OF CHANGE: The modifications include the following:

1) Tubular Grapple Mast (NF500)

The current refueling platform grapple mast is a four section, telescoping, triangular, truss-stiffened component. It will be replaced by a new tubular mast assembly which has an increased section modulus for rigidity and is a direct replacement for the existing triangular mast assembly on the refueling platform. The mast assembly is comprised of four telescoping tubular sections very much like the existing triangular mast assemblies. The cables and electrical lines are routed inside the mast assembly and out through the lower end to attach to the grapple head assembly. The outer tube assembly is suspended from the platform at its upper end by means of a pin and hanger joint. The upper end of the inner tube assembly is suspended from the dual cable of the platform's main hoist. As the inner tube assembly and the grapple are raised and lowered, the telescopic mast is retracted and extended, respectively. The change in configuration (truss-stiffened triangular to closed cylindrical tube) has increased the weight of the new NF500 mast relative to the current mast design. The NF500 mast weighs approximately 420 pounds more than the current NF400 mast. The additional weight necessitates changes in the technical specification requirements which demonstrate operability of the refueling platform. The three surveillances which must be changed are load limit interlocks for jam cutoff, grapple engaged loaded interlock, primary and redundant load interlocks.

2) Test Weights

The new test weights are required to perform Technical Specification 3/4.9.6.1 surveillances for the new NF500 mast, because of the increased weight of the mast.

3) Camera System

During the refueling outage and at other times when fuel handling is taking place, several activities are performed which require close observation of the fuel bundle.

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Examples of these activities include verification of the fuel bundle identification on the bail, verification of high and low fuel bundles after completion of core reconfiguration, grappling the fuel assembly, and visual inspection of selected vessel internals. Currently, this is performed by using binoculars to view the bail handle from the refueling platform or viewing through a TV camera attached to either the exterior of the mast or dangling on the end of the TV camera cable. This new camera will be mounted internal to the grapple head and will give close-up pictures of the bail to be grappled without use of a wet, contaminated camera cable external to the mast which must be handled manually and disturbs the water surface distorting the view. The internal camera will not interfere with the vessel shroud when loading fuel at any fuel cell location. The TV camera system meets the following objectives:

- Long life, greater than 1xE8 R (accumulated dose). This is equivalent to over 1500 fuel moves based upon an average dose rate of 3x5E5 R/hr and 8 minutes in contact with an irradiated fuel assembly per move. At this rate, the camera lens will last a minimum of one refueling outage.

- Compatible with either the new NF500 or the old NF400 triangular masts with minimal changes.

- Continuous TV picture during mast movement.

- Viewing from a horizontal position to vertical position (through grapple mouth).

- Remote camera controls for operators on the trolley or personnel on the bridge.

4) Cable Reel Alarm

The new NF500 mast is composed of concentric telescoping stainless steel tubes which will prevent visual detection of slack hose/cables. There is a possibility that if the slack electrical cables/hose are allowed to accumulate in the new tubular mast they could be damaged. In the existing design configuration, drop of a grapple bundle upon loss of air is precluded by the grapple head design and this design is unchanged in the new configuration. However, damage to the air-hose could render the grapple head inoperative to grapple a bundle. Therefore, no enhance uninterrupted refueling activities, a device for detecting slack in cables/hose should be installed to preclude damage. This device is the cable reel alarm which uses three microprocessor-based ratemeters to compare the relative payout and take-up speeds of the hoses and cables attached to the refueling mast. should the speed of the air hose, electrical cable, or camera cable differ from the speed of the hoist cable, hoist motion will be stopped. The interlock prevents hoist movement in the up direction until the alarm has been reset, but will allow movement in the down direction with the use of the travel override pushbutton. This system will be activated when visual detection of the slack hoses and/or cables in the mast is impossible because of configuration and has not been redressed by other actions.

5) Load-Limit Relay

A load-limit protective relay was inadvertently incorporated in the past, to obtain redundant rod block interlocks. The redundant rod block interlock function was already accomplished by other load-limit protective relays. The inadvertently incorporated spare will be deleted.

REASON FOR CHANGE: The purpose of DCP 90/0313 is to enhance the operational reliability of the refueling platforms in order to support a minimum length refueling outage schedule.

SAFETY EVALUATION: Though there are many design/operational issues which were addressed in the design of these operational enhancements, the following is a list of the major considerations and a brief description of the outcome.

- Camera must not interfere with vessel or other underwater components/structures when the mast is in use - camera was put inside mast and grapple head to eliminate interferences.

- Camera will work in highly radioactive area - camera chosen has high tolerance to radiation exposure and will last at least one refueling outage before replacement is needed.

- Effects of the heavier mast on refueling bridge - seismic requalification and supporting analysis show new mast and refuel bridge within design allowables.

- Effects the new components (TV system and slack cable system) have on seismic qualification of bridge and themselves - components were designed not to affect seismic qualification of bridge and are designed for II/I considerations.

- In the existing design configuration, the drop of a grappled bundle upon loss of air is precluded by the grapple head design and this design is unchanged in the new configuration.

- Effects of heavier mast on the Fuel Handling Accident (FHA).

    a) FHA as described in UFSAR 15.7.6 - this FHA assumes that an irradiated fuel assembly is dropped into the reactor core when the reactor vessel head is off. A failure of the grapple-cable or the fuel handling platform's hoist attachment to the platform is not considered credible because multiple failures would be required to drop the lifted fuel assembly and mast (SEGE-87/019); therefore, the weight of the current mast is not considered in this FHA analysis. The new mast has the same multiple failure features and its failure is also not considered credible. Therefore, the FHA as described in UFSAR 15.7.6 is not affected by the installation of the heavier mast.

b) FHA as described in UFSAR 15.7.4 - this FHA assumes a non-fuel assembly weight of 1140 pounds or less is dropped on spent fuel assemblies. The 1140 pounds is based on the NUREG-0612 definition of heavy loads which is the combined weight of a single spent fuel assembly and its associated handling tool. It is determined by adding the weights of the movable mast sections and the grapple head to the weight of a fuel assembly. The inputs for determining the 1140 pound limit were the weights of the movable sections of the original mast, the weight of the grapple head, and the weight of GE fuel. Using the weights of ANF fuel and the movable sections of the new mast in conjunction with the grapple head yields a new heavy load limit of 1405 pounds. While this new limit would relieve some operational restrictions on lifted loads, it would necessitate numerous programmatic and procedural changes relative to the control of heavy loads as well as reanalysis of the non-fuel drop detailed in FSAR 15.7.4. Although the combined weight of fuel assembly and tool is being increased by this DCP (and thus the defined NUREG-0612 heavy load value will be increased), the current 1140 pound technical specification limit will not be increased. Increasing the value would move Technical Specification 3/4.9.7 in a non-conservative direction as larger weights would be allowed over spent fuel assemblies. Therefore, the existing Technical Specification 3/4.9.7 will be retained without change.

Serial Number: 91-003-NPE Document Evaluated: DCP 89-0128-S00-R00

DESCRIPTION OF CHANGE: There is presently no way to monitor the mobil make-up water trailer (MMWT) effluent conductivity. The MMWT presently provides the total input to the demineralized water storage tank. BYPL 89/128 was generated to provide the plant with a means of monitoring the conductivity of the MMWT effluent. This DCP will provide the required design and instructions to install a conductivity monitoring system for the MMWT effluent. This will be accomplished as follows:

Installed System

Per telecon report GTC 91/00330 the IWT system is no longer in use and it is not anticipated that this system will ever be used again. There is a conductivity monitoring system on the effluent of each the anion resin beds. This system consists of conductivity elements in the water treatment building, conductivity monitor and recorder in the radwaste control room on the H22P088 panel. The required conductivity monitoring system will be provided by using portions of the existing system, particularly the cable for the signal from the conductivity elements located in the water treatment building to the radwaste control room, the panel mounting and some of the interconnecting wiring in panel H22P088.

General Design Requirements

Based on a review of chemistry logs the effluent of the MMWT is between .05 and .06 μmhos therefore the new conductivity system shall have a range of 0 to 0.2 μ mhos. It will provide an alarm on the H22P088 panel.

Conductivity Elements

Replace the installed L&N Model 7073 analogue monitor, located in H22P088 of the radwaste control room, with a L&N Model 7082 digital monitor. The new monitor is a state of the art monitor which will provide the accuracy and range required.

Alarm

The new conductivity monitor has the ability to provide an alarm output. This alarm output will be routed to the installed alarm window for the "A" anion effluent in the H22P088 panel and the window re-engraved.

Recorder

The output from the conductivity monitor will be routed to pen 1 of the installed recorder. Pen 2 will be disconnected.

REASON FOR CHANGE: There is presently no way to monitor the mobil make-up water trailer (MMWT) effluent conductivity. The MMWT presently provides the total input to the demineralized water storage tank.

SAFETY EVALUATION: The mobil make-up water trailer is the source makeup to the demineralized water storage tank, which is not addressed in the FSAR as a precursor to an accident. Make-up water to the demineralizer storage tank does not provide a mitigating function for a previously evaluated accident. This sourceof water is not required for the operation of any equipment which would be an accident precursor or mitigate the consequences of a previously evaluated accident. If this source of water were lost it could not affect any equipment important to safety therefore it could not cause an accident or malfunction of equipment not previously evaluated. Therefore, the installation of a conductivity monitoring system on the effluent of the MMWT and the modification of the installed conductivity monitoring system on the effluent of the IWT system will not increase existing safety concerns or add new ones.

Serial Number: 91-004-NPE   Document Evaluated: DCP 91-0021-S00-R00

DESCRIPTION OF CHANGE: The outdated Westronics M11E analog recorder E31R608 will be replaced with a state of the art Westronics series 3200 digital recorder and the setpoint of E31R608 point 21 (E31N015D) will be raised. The outdated Westronics M5E analog recorder E31R611 will be replaced with a state of the art Westronics series 2400 digital recorder. The 10 amp fuse in the power circuit of each recorder will be replaced with two 3 amp fuses in series. This will adequately isolate the non-Q recorders from Class 1E power.

REASON FOR CHANGE: Annunciator E31-TAH-L619B (drywell ambient temp high) stays in alarm because thermocouple E31-TE-N015D is operating at the 145F setpoint. This constant alarm condition is masking the status of seven other drywell thermocopules which are associated with the same contact output of the recorder E31-TJRS-R608. The setpoint of E31-TE-N015D could be raised if the recorder had a spare contact output, E31R608, however,is a Westornics M11E recorder which has only 6 contact outputs and none are spare (reference EER 89/6058).

Annunciator E31-TAH-L620B (RWCU equipment area differential temperature high) is being activated when a high delta T condition (31F difference between E31N035A and E31N036A) does not exist. Raising the setpoint would prevent this, but the problem is caused by the large deadband of the obsolete and unreliable Westronics M5E recorder E31R611 (reference EER 91/6088).

SAFETY EVALUATION: Temperature recorder E31R608 and delta T recorder E31R611 monitor various areas of the plant for leak detection purposes. Each recorder has six contact outputs which power Control Room annunciators when a setpoint is exceeded. This alerts the operators to potential steam leaks. These recorders are non-safety related, however, and perform no active safety related function. They are also not required for Regulatory Guide 1.97 indication and no credit is taken in the UFSAR for operator actions based on information taken from the recorders or their annunciators. The existing recorders are connected to Class 1E power. The new recorders will be adequately isolated from the Class 1E bus per Engineering Report GGNS-91-0014. Both recorders are fed from the same Class 1E bus. The load on this bus is not increased because the combined load of the new recorders (155 VA) is less than the combined load of the old recorders (200 VA). The seismic qualification of the safety related panel in which the recorders are mounted will be maintained. No panel modification will be required.

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The changes of this DCP 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 recorders. This design change will be an improvement in terms of reliability and monitoring capability. The changes of this DCP will not compromise any existing safety related system, structure or component. The failure of the recorders will not initiate any evaluated transient or accident. The E31 (Leak Detection) System operation and function will not change. The recorders are not required to mitigate the consequences of any evaluated transient or accident. No new interfaces are created and no new failure modes are introduced. This change will therefore not introduce an unreviewed safety question. The recorders are not currently addressed in the technical specification and this change will not require that they be added to the technical specification.

Serial Number: 92-005-NPE Document Evaluated: MCP 92/1050 R1

DESCRIPTION OF CHANGE: This change is being issued to replace the SMB-000-5 actuators on the Suppression Pool Makeup (SPMU) System dump valves with larger SMB-00-10 actuators.

REASON FOR CHANGE: The NRC issued Generic Letter 89-10 which requires that licensees establish: 1) the torque/thrust required to open/close all safety related motor operated valves against the maximum expected design differential pressure and flow rate, 2) a program to set/verify the torque switch settings based on the calculated torque/thrust requirements, and 3) a testing program under which the safety related MOVs are actually stroked against the design flow rate and differential pressure. The torque required to stroke a butterfly valve is dependent on both the flow rate and differential pressure across the valve. An evaluation for the SPMU System dump valves was performed to establish the maximum expected differential pressure and flow rate (i.e., design conditions) which the valve would be expected to stroke against (i.e., open/close) to perform their design safety function. The manufacturer of the valves, Henry Pratt Company, was contracted to evaluate the valves at the maximum expected differential pressure and flow rate and provide the torque required to stroke the valve. The valves were then evaluated to determine: 1) if the required torque exceeded the maximum torque which could be applied to the valve without overstressing the weak link and 2) if the actuator was capable, considering torque switch settings and spring pack ratings, of delivering the torque required to stroke the valve. This evaluation revealed that the actuators on the Suppression Pool Makeup (SPMU) System dump valves were undersized and would be required to operate at torque values which were two (2) times the manufacturer's published torque rating. Limitorque states that their SMB actuators are capable of surviving a one time overtorque of two (2) times the published torque rating of 90 ft-lbs without sacrifice to the actuator qualification, however, Limitorque does require that the gearing in the actuators which have been overtorqued be visually inspected for cracks and/or excessive wear. Therefore, since the actuators on the suppression pool dump valves had been overtorqued a number of times during startup testing and no inspections, as required by Limitorque, had been performed, MNCR 0048-92 was generated. As an interim solution the existing SMB-000-5 actuators, which were undersized, were replaced with refurbished SMB-000-5 actuators obtained from Unit 2 stock. This interim fix was justified for operation until RF05 based on: 1) the fact that Limitorque states the SMB-000 actuators can survive a one time overtorque of 200% without sacrifice of the actuator qualification, 2) the fact that the refurbished Unit 2 actuators have never been overtorqued, and 3) the fact that the valves are only required to stroke open one time to perform their safety function.

MCP 92/1050 is being issued to replace the existing SMB-000-5 actuators, which are undersized, with SMB-00-10 actuators which are properly sized for the application.

SAFETY EVALUATION: Implementation of MCP 92/1050, Revision 1, will not increase the probability of occurrence or increase the consequences of an accident previously evaluated in the FSAR. The modifications will increase the reliability of the SPMU System by providing actuators which are properly sized for the application, therefore the modifications will not increase the probability of occurrence or the consequences of a malfunction of equipment important to safety previously evaluated in the FSAR. The installation of the SMB-00-10 actuators on the SPMU System dump valves was evaluated and it was determined that the seismic loading imposed by the valves was within code allowables, the hydrogen which could be generated from the aluminum limit switch covers was negligible, the valve stroke times were more conservative than the existing stroke times and the actuators meet all of the environmental qualification requirements for use in the containment. Also, the overcurrent protective devices are properly coordinated for operation of the equipment and protection of the electrical penetration in accordance with Regulatory Guide 1.63 and the thermal overload relays are set appropriately. 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 FSAR.

Implementation of MCP 92/1050, Revision 1, will require a change to the GGNS Unit 1 Technical Specifications Table 3.8.4.1-1 to revise the overcurrent protective device setpoint but will not reduce the margin of safety as defined in the basis for any technical specification. Note: The technical specification change was approved in Amendment 100 on 06/01/92 under exigent circumstances. The conclusions of the safety evaluation remained valid.

Serial Number: 92-006-NPE Document Evaluated: DCP 90/0009 R0

DESCRIPTION OF CHANGE: This change will change the setpoint of the ATWS Alternate Rod Insertion (ARI)/Recirculation Pump Trip (RPT) units, in accordance with the appropriate calculation.

REASON FOR CHANGE: The setpoint for the ATWS ARI/RPT high reactor dome pressure is being changed to increase the spurious trip avoidance probability of this trip function. This change will decrease the probability of spurious high dome pressure ARI/RPT trips on an MSIV closure.

SAFETY EVALUATION: This change changes the setpoint for the high reactor dome pressure trip of the ATWS ARI/RPT system. Increasing the high dome pressure setpoint for ARI/RPT as described above will not prevent the system from functioning as per the original design intent as described in UFSAR Section 5.4.1.7.10. This change does not affect Grand Gulf's compliance with 10CFR50.62, the ATWS rule, as described in Section 15.8.1 of the UFSAR. Therefore, this setpoint change will not increase the probability of occurrence of an accident. This change is still bounded by the existing accident analysis in Chapter 15 of the UFSAR. The change does not prevent any equipment relied upon to mitigate the consequences of a malfunction of equipment important to safety or any evaluated transient or accident from performing its safety function. Therefore, the consequences of a malfunction of equipment important to safety or an accident previously evaluated in the UFSAR is not increased. No new interface is created which would affect components, equipment or systems which perform safety functions. This change will create no new failure modes not already enveloped by present UFSAR analyses. No new accident precursors are being introduced by this change. The Technical Specification Bases 3/4.3.4 states that the anticipated transient without scram recirculation pump trip (ATWS-RPT) system provides a means of limiting the consequences of the unlikely occurrence of a failure to scram during an anticipated transient. This change in setpoint does not prevent the system from performing its trip function as per the original design. The trip function of this setpoint to initiate the ATWS-ARI/RPT system will still meet with the requirements of the NRC ATWS Rule 10CFR50.62, as described in Section 3/4.3.4 of the technical specification.

Serial Number: 92-007-NPE Document Evaluated: DCP 89/0187

DESCRIPTION OF CHANGE: The Reactor Water Cleanup (RWCU) Filter Demineralizer Discharge Pressure Hi alarm logic will be modified by installing a Bailey 745 alarm unit (to be designated as 1G33-KS-601) in series with the Hi pressure switch (PSH-N014) in the alarm circuit and using the 1-5 Vdc output of the RWCU blowdown valve controller (1G33-HC-R606) to drive the alarm unit. The alarm unit will be set to trip at a setpoint corresponding to an approximate 5% open signal (CR) from the controller, thereby enabling the Hi alarm only when the system is manually placed in the blowdown mode. The present function of closing the control valve (F033) on Hi or Lo pressure in the blowdown line will be unchanged.

REASON FOR CHANGE: The RWCU Filter Demineralizer Discharge Pressure Hi/Lo Annunciator (1G33-PAHL-L608) is in constant alarm.

This alarm is generated by a pressure switch (1G33-PSH-N014) located downstream of the RWCU blowdown line flow control valve (1G33-F033). During normal operation of the RWCU system, this section of the blowdown line is pressurized by leakage past the flow control valve to pressures exceeding the pressure switch setpoint.

SAFETY EVALUATION: The blowdown portion of the RWCU System (G33) is non-safety related and none of the components perform a function important to safety. Implementation of this design change will not affect the operability, LCO or surveillance requirements of any technical specification therefore no changes to the technical specifications will be required. Since no modifications are being made which will affect the overall system performance in a manner which could lead to an accident, 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 RWCU 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: 92-022-NPE Document Evaluated: MCP 92/1093 R0

DESCRIPTION OF CHANGE: Each blind flange that is located at the chemical injection points of the Standby Service Water (SSW) Basins A & B will be replaced with regular flanges, pipe, plug valve, and a hose connection.

REASON FOR CHANGE: This modification will delete the need of a mechanic to remove the blind flange before each chemical injection and replace the blind flange after each chemical injection.

SAFETY EVALUATION: Replacing the blind flange with regular flanges, pipe, plug valve, and hose connection will not affect the Standby Service Water System function, operation, or performance in any way.

No technical specification change and no unreviewed safety question result from this change.

Serial Number: 92-031-NPE Document Evaluated: MCP 91/1087 R0

DESCRIPTION OF CHANGE: This change provides for replacing the lube oil reservoir drain plug with a threaded valve arrangement for instrument air compressors 1P53-C001-B and 2P53-C001-N; and service air compressors SP52-C001A-N and SP52-C001B-N. The valve arrangement consists of a threaded nipple, gate valve and drain plug.

REASON FOR CHANGE: Due to the physical location of the original drain plugs, a design change is needed to facilitate draining of the lube oil reservoir during compressor maintenance. Using the original drain plug arrangement would not provide controlled reservoir drainage and the oil could flow along the reservoir frame, down the compressor's concrete base and on the floor. The air compressor vendor anticipated this enhancement and allowed for such an improvement in the air compressors' vendor manual.

SAFETY EVALUATION: The modifications will not affect the systems function, operation, or performance in any way. The Service Air and Instrument Air Systems are non-seismic Category I and non-safety related. These systems have no safety related functions and failure of these systems will not compromise any safety related system or component and will not prevent safe reactor shutdown. Providing no supports for the short drain piping will not adversely affect the structural integrity of the associated piping nor the compressor. The piping design meets ANSI B31.1 and the vendor requirements. Therefore, the piping will function in its intended manner. No existing system interfaces are affected and no new system interactions are created.

The support review of the short unsupported drain piping has shown that the ANSI B31.1 code allowables have been met. Therefore, the probability of a piping failure has not increased. The modification will not introduce any new postulated piping failures and the existing hazards evaluations are not affected. No new system interfaces have been adversely affected. No new failure modes for the system or any equipment have been created.

By remaining within the code allowables and vendor requirements, the margins of safety provided by these allowables are not affected. For some of the air system isolation valves, response times are provided in Technical Specifications 3/4.6 and 3/4.8. However, no technical specifications exist for the Instrument Air or Service Air Systems. The modifications will not create an unreviewed safety question.

Serial Number: 92-035-NPE Document Evaluated: DCP 88/0053 R0

DESCRIPTION OF CHANGE: Dean/Dale & Dean, an architectural firm in Jackson, was retained for the design and construction of a chemical storage facility south by southeast of Unit 1 inside the protected area. The facility is designed to be partially sprinkled. Automatic sprinklers will be supplied from the 12" fire protection underground water main located south of the Radwaste/Water Treatment Buildings by an 8" underground feeder main to be installed by the Chemical Storage Facility Architect/Engineer (A/E). Various chemicals were evaluated for storage in the new facility and in the existing warehouse.

REASON FOR CHANGE: This evaluation was performed to accept the chemical storage facility for use at GGNS.

SAFETY EVALUATION: The proposed addition of chemicals in the new locations will not result in unacceptable concentrations of toxic chemicals in the Control Room in the event of a spill and/or fire involving these chemicals nor will these chemicals pose an explosion hazard to any safety related structure, system, or component. Fire protection piping will comply with NFPA 13 and NFPA 24 and is compatible with the piping utilized in the Grand Gulf Fire Protection System. Therefore, the proposed change does not increase the consequences of an accident already evaluated in the FSAR since the Control Room operators' abilities to function are not degraded by the postulated spill and/or fire and there is no explosion hazard postulated from the storage of these chemicals. The storage facilities and containers are non-safety grade. They are not relied upon to function after an accident and there is no accident postulated as the result of a fire main failure. The change does not increase the probability of an accident due to the malfunction evaluated (spill and/or fire) since the Control Room operators' abilities are not degraded. Also, the connection of the chemical storage facility to the Unit 1 fire protection water supply will not adversely affect the ability of the Unit 1 fire water supply to meet anticipated fire protection water demands. The change does not create the possibility of an accident or an equipment malfunction different from those already evaluated in the FSAR since adequate section control is provided to minimize the affects of fire protection underground pipe leaks or failures, Control Room habitability will be maintained in the event of a chemical spill and/or fire and there is no explosion hazard postulated from the storage of these chemicals. The margin of safety is not reduced since the habitability of the Control Room and the operability of the fire protection system is not impaired.

Serial Number: 92-036-NPE Document Evaluated: MCP 91/1143 R0

DESCRIPTION OF CHANGE: This change is to delete the conductivity monitors for the Standby Service Water (SSW) System and remove or spare all equipment and tubing used for remote monitoring of the monitors. Components that will be removed or abandoned in place are Conductivity Elements and Transmitter Switches.

REASON FOR CHANGE: The sample elements for the SSW Basin A, B conductivity analyzers are located after the check valves in the discharge piping of the SSW pumps. This location allows the conductivity analyzers to sample stagnant water when the pumps are not running resulting in nuisance high conductivity alarms due to fouling of the analyzers by scale and deposits. The alarms are no longer used by Operations as indication of when to initiate blowdown of the system in order to remove high concentrations of iron and corrosion suspended in the SSW System.

The original design of the SSW System was to provide operator alarms when blowdown of the piping and basins was needed to discharge high concentrations of iron and corrosive particles. These metallic particles can collect and form scale on the pipe and basin walls.

Presently, the SSW System is placed in service weekly for 24 hours and sampled after four hours of operation with the grab sample solution analyzed in the laboratory. Chemical additives and blowdown needed to control particle buildup are determined on the basis of this analysis. Therefore, the aforementioned equipment can be disconnected and either removed or abandoned in place.

SAFETY EVALUATION: All of the equipment, instrumentation and tubing associated with the conductivity monitoring system are non-safety related, although the sample elements do form a safety-related pressure boundary of the SSW System piping. All work will be performed downstream of the 3/8" globe valve that constitutes the safety/non-safety interface between the SSW piping and the conductivity sampling tubing. Therefore, this change will have no impact on safety related piping or tubing. There are currently no Class 1E electrical interfaces associated with the conductivity monitoring equipment. No electrical or mechanical safety-related interfaces shall be either created or deleted by implementation other than those addressed within this evaluation.

Serial Number: 92-038-NPE Document Evaluated: MCP 92/1097

DESCRIPTION OF CHANGE: This change will change the component MPL number for a restricting orifice located in the seal flushing water supply to the Reactor Water Cleanup (RWCU) precoat pump from N1G36D051 to N1G36D057.

REASON FOR CHANGE: A previous change installed a moisture separator in the backwash receiving tank vent line. This moisture separator was designated N1G36D051. This MPL number had been previously assigned to an orifice located in the seal flush water supply to the RWCU precoat pump. This change will assign a new MPL number to the orifice plate to eliminate the duplication of numbers.

SAFETY EVALUATION: This change will not affect any present technical specification requirements. Changing the MPL number will not affect the operation of the RWCU system or its associated components. The orifice plate which is installed in ANSI B31.1 piping will not be physically changed except the orifice plate marking. This marking will be changed to indicate the new MPL number of the orifice plate is N1G36D057.

Serial Number: 92-041-NPE Document Evaluated: MCP 91/1145 R0

DESCRIPTION OF CHANGE: This change deletes the annunciation and computer points for reactor feed pump discharge pH monitor, the condensate demineralizer combined effluent and feedwater heater drain turbidity monitors, and the standby service water conductivity monitors.

REASON FOR CHANGE: Removal of the instruments and components with which these annunciator and computer points are connected is addressed within the scope of other change packages. This change is organized so that these nuisance alarms and indicators can be disconnected prior to the completion of removing all of the associated equipment for each monitoring system that is being deleted. Therefore, the annunciator and computer points should be removed as appropriate prior to the complete demolition of this monitoring instrumentation.

SAFETY EVALUATION: There are currently no Class 1E electrical interfaces associated with the subject monitoring equipment and annunciation or computer points. No electrical or mechanical safety-related interfaces shall be either created or deleted by implementation. Each point provides for indication and annunciation only; they perform no actuation function.

All of the annunciator points to be disconnected are non-safety related. In addition, all of the computer points are connected to the Balance of Plant (BOP) Computer System only. Each annunciator and computer point may be disconnected without impacting any safety-related system, structure or component.

This safety evaluation is issued primarily to determine the impact of changing UFSAR Figures 9.2-1, 9.3-6 and 9.3-7 to reflect the removed annunciator and computer points for the monitoring equipment. There is no challenge to plant safety by performing this demolition and updating the piping & instrument drawings.

Serial Number: 92-043-NPE Document Evaluated. DCP 83/0480 R1

DESCRIPTION OF CHANGE: This change provides details and instructions for permanent connection of the air conditioning unit in Balance of Plant (BOP) Panel M72-P800 to 120 Vac BOP power.

REASON FOR CHANGE: The air conditioning unit was installed earlier but no provisions were made for permanent connection to 120 Vac power.

SAFETY EVALUATION: This modification is an enhancement to the M72 system in that a temporary power connection is being discarded and the air conditioning unit in BOP Panel M72-P800 will be permanently connected to 120 Vac BOP power. Permanent connection of air conditioning unit to 120 Vac BOP power will not compromise any safety related system or component and will not prevent safe reactor shutdown. All cabling and raceway modifications to be performed will be in accordance with the separation requirements of Regulatory Guide 1.75 and where required, seismic supports have been provided to preclude the creation of any Seismic II/I concerns.

Serial Number: 92-044-NPE Document Evaluated: DCP 89/0072 R0

DESCRIPTION OF CHANGE: This change will install six 2" flush connection with blind flange on lines in the Condensate/Refueling Water (P11) and Suppression Pool Cleanup (P60) Systems. Also, this change will increase the size of three drain lines to allow easier removal of "hot" sludge from the system.

REASON FOR CHANGE: High traffic areas of the Auxiliary Building in which general area radiation dose rates are excessive have been identified. The source of the radiation doses is the P11 and P60 piping which is routed through these areas. Also one inch drains are not large enough to accommodate the removal of crud deposits from the system adequately during hydrolasing. Radiation levels in this piping are increasing with system operation, and thus maintaining plant ALARA goals is becoming more difficult.

SAFETY EVALUATION: Adding flush connections and increasing the size of drain lines will not adversely affect the structural integrity of the associated piping. The piping has been designed to ANSI B31.1 code requirements and supported for the appropriate deadweight, thermal and seismic loads. The modification will not affect the function, operation or performance of P11, P60 or any other system. The affected systems serve no safety function. Systems analysis has shown that failure of either system will not compromise any safety-related systems or prevent safe shutdown. A portion of the P60 which interfaces with the residual heat removal system is designed to Seismic Category I and ASME Section III, Class 2 requirements. These requirements are maintained by this design change.

Use of hydrolase equipment to remove crud deposits from the pipe could cause a reduction in pipe wall thickness over time. The inspection criteria specified will ensure that adequate pipe wall thicknesses are maintained in these lines .

In addition, this system is not addressed in the technical specifications and this design change will not affect the function or operation of this or any other system. Therefore, no changes to the technical specifications are required.

Serial Number: 92-051-NPE Document Evaluated: MCP 92/109, R0

DESCRIPTION OF CHANGE: An underground piping connection which will tie new Unit 2 Auxiliary Building fire protection systems to the existing Grand Gulf Fire Water System (P64) is provided. The scope of this work is limited to the installation of underground piping and a valve between the new contractor-installed 10" underground pipe and the existing fire water main. The connection is required to supply fire water for an NFPA 14 Class II fire protection standpipe system and an NFPA 13 wet-pipe sprinkler system to be installed in the Unit 2 Auxiliary Building.

An aboveground piping connection which will tie a new Unit 2 Turbine Building fire protection system to the existing fire protection water main which is located inside the north wall of the Unit 2 Turbine Building at the 113'-0" elevation is provided. The scope of this work is limited to the installation of connecting aboveground piping and valves. The connection is required to supply fire water for an NFPA 14 Class II fire protection standpipe system to be installed in the Unit 2 Turbine Building.

REASON FOR CHANGE: To provide a controlled design for the connection of Unit 2 Auxiliary Building and Unit 2 Auxiliary Building and Unit 2 Turbine Building fire suppression systems to the Unit 1 Fire Water Supply System, with provisions for isolating the Unit 1 Fire Water System from the Unit 2 Auxiliary Building and Unit 2 Turbine Building systems.

SAFETY EVALUATION: This change does not increase the probability of occurrence or the consequences of an accident or malfunction of equipment important to safety previously evaluated in the Safety Analysis Report because: 1) No accidents are postulated in the FSAR as a result of the failure of a fire protection water main; 2) The analysis of safe shutdown in the event of a major fire, as described in UFSAR Appendix 9C, is not affected by this change; 3) The ability of the fire water distribution system to meet anticipated fire protection water demands will not be adversely affected.

This change does not create the possibility of an accident or malfunction of equipment important to safety of a different type than any evaluated previously in the Safety Analysis Report because: 1) This change does not introduce fire, seismic, missile, flooding, structural or other hazards to equipment required to achieve and maintain safe shutdown of the reactor; 2) This change does not introduce new or different equipment failure modes.

This change does not reduce the margin of safety as defined in the basis for any technical specification because: 1) Installation of a connection to the 12" diameter underground fire water supply loop main and a connection to the 10" diameter aboveground fire water supply main will not adversely affect the ability of the fire water distribution system to meet anticipated fire protection water demands; 2) Fire protection system redundancies have not been reduced and the effects of single failure have not been increased.

Serial Number: 92-063-NPE Document Evaluated: DCP 89/0159-S00-R00

DESCRIPTION OF CHANGE: This change provides the mechanical/HVAC, piping, civil, and electrical/I&C design requirements necessary to resolve heat related problems generated by excessively high temperatures inside feedwater heater rooms.

REASON FOR CHANGE: The purpose of this change is to reduce high temperatures inside feedwater heater rooms by increasing exhaust airflow from each feedwater heater room in order to lower room temperatures to approximately 150°F, thereby, allowing equipment in each room to operate within design ratings.

SAFETY EVALUATION: Various components such as cabling, internal jumper wiring, phenolic switch components, terminal blocks, motor lead insulation, and grease inside the MOV main gear boxes, were damaged by high temperatures inside feedwater heater rooms.

This station modification provides the mechanical/HVAC, piping, civil, and electrical/I&C design requirements necessary to lower and maintain temperatures in the affected areas at approximately 150°F to preserve equipment integrity.

The primary system affected is the Turbine Building Ventilation System. This system has no safety related function as described in UFSAR Section 3.2. Failure of the system will not compromise any safety related system or component, and will not prevent safe shutdown of the plant. This design resolves heat related problems associated with each room by increasing exhaust airflow, which will not adversely affect system operation or function.

The design meets the applicable codes, material requirements, standards, and quality assurance requirements, therefore, implementation of the design will not result in increased dose rates, accidents, or equipment failures. Adequate fire detection equipment (smoke/fire detectors) are installed to provide early detection, and prevent fire related damage to surrounding equipment. Electrical cabling and wiring for the solenoid valves will be processed in accordance with Regulatory Guide 1.75, therefore, overall system performance and reliability is maintained.

Serial Number: 92-065-NPE Document Evaluated: DCP 91/0010-S00-R00

DESCRIPTION OF CHANGE: This change provides the design details for rerouting the condensate drain piping from twenty-one Turbine Building Ventilation (U41) System room cooler fan coil units. They are currently routed to dirty radwaste (DRW) drains for processing by the floor drain subsystem of the Liquid Radwaste (G17) System. The new routing will drain the condensate to the CRW to be processed through the G17 equipment drain subsystem. Additionally, sight glasses will be installed for two unit drain lines outside the high radiation area.

REASON FOR CHANGE: This change reduces the amount of relatively clean water which is unnecessarily being introduced into the DRW drains. The result of routing fan coil unit drains to CRW will be a decrease in the volume of liquid radwaste which must be processed in the floor drain subsystem and normally discharged to the environment. The condensate water will be processed through the G17 equipment drain subsystem and reused, as necessary, in the Condensate (P11) System. This will increase condensate inventory and reduce the makeup requirements from the Makeup Water (P21) System.

Plant personnel occasionally need to estimate the amount of water coming from a fan coil unit drain. This can normally be done by inspecting the drip pan. However, some units are located in high radiation areas which are inaccessible during normal power operation. A sight glass in the drain line, located in an accessible area, enables personnel to check the flow through the line without having to inspect the drip pan.

SAFETY EVALUATION: The rerouting of cooler unit condensate drains from DRW and CRW will not affect the Turbine Building Ventilation System function, operation, or performance in any way. The penetrations utilized are non-safety related and all concerns for structural integrity, ALARA, and safety have been addressed in the design change package. The capability of the equipment drain and suspended drain systems to direct the additional 3 gpm of flow to the Liquid Radwaste System were acceptable as designed. The effect of decreased DRW and increased CRW liquid volumes by 3 gpm is insignificant compared to the current amount of liquid radwaste processing of approximately 13-15 gpm and the maximum design process flow of 159 gpm. The piping has been designed to ANSI B31.1 code requirements and supported for the appropriate deadweight loads. Therefore, the piping and their supports will function in their intended manner. None of the systems have any safety-related functions. Failure of these systems will not compromise any safety-related system or component and will not prevent safe reactor shutdown. Therefore this change will not create an unreviewed safety question.

Serial Number: 92-066-NPE   Document Evaluated: DCP 91/0032-S00-R00

DESCRIPTION OF CHANGE: This change provides for installation of an additional seal steam generator shell side level switch. This switch provides a sufficient number of level switches to permit use of two out of three isolation logic for the seal steam generator (SSG) isolation valves. Manual intervention is, and will continue to be, required to reopen all these valves except for the seal steam generator outlet valve. One valve will also require that two out of three of the level switches are in the reset position for the valve to open automatically.

REASON FOR CHANGE: Decommission of the Auxiliary Steam System has eliminated the only backup for main and Reactor Feed Pump Turbine (RFPT) turbine seal steam source, resulting in complete dependence on the N33 System.

The present instrumentation may cause a spurious isolation of the seal steam generator with failure of a single seal steam generator level switch in the high level position, causing loss of the seal steam. Restoration of seal steam requires that both the present level switches be reset from their high level setpoints and that the seal steam generator pressure switch to be in the "normal pressure" position.

SAFETY EVALUATION: The addition of a third high water level switch and use of two out of three isolation logic will eliminate spurious seal steam generator isolation caused a single water level switch failure.

The high water level isolation function (before and after the modification) prevents water carryover from the seal steam generator to the seal glands for the main and RFP turbines.

No technical specification change or unreviewed safety question results from this change.

Serial Number: 92-067-NPE   Document Evaluated: MCP 88/1023-S00-R00

DESCRIPTION OF CHANGE: This change replaces the existing trim in the flow control valves for the Control Room air conditioning unit with trim that allows the valve to operate near mid-range during normal plant operation and ensures the units ability to perform its design function in the event of an accident. The existing valve trim is sized to provide 151 gpm of 90°F water to the condenser during both normal and accident conditions. With the new trim the valves are sized for 45 gpm of 74°F water during normal operation and 85.5 gpm of 90°F water in the failed open position.

REASON FOR CHANGE: The flow control valves for the Control Room air conditioning unit are sized such that during normal plant operation the valve operates near the full closed position. In addition, in the event of an accident involving a loss of offsite power, instrument air will be lost causing these valves to travel to the full open position. This will result in excess cooling water flow to the condenser, possibly resulting in a unit trip due to low compressor suction pressure.

SAFETY EVALUATION: The revised flows to the Control Room air conditioning units do not adversely affect the ability of the unit to perform its safety function. The new valve internals do not affect the structural integrity of the valve.

This modification will ensure an adequate, regulated supply of water to the Control Room air conditioning unit.

No technical specification change or unreviewed safety question results from this change.

Serial Number: 92-068-NPE Document Evaluated: MCP 92/1048-S00-R00

DESCRIPTION OF CHANGE: Seal water piping for Control Rod Drive (CRD) pumps is carbon steel with threaded connections. This piping will be replaced with stainless steel and socket welded fittings will be used in lieu of threaded. Threaded connections on components which cannot be socket welded will be seal welded. Also, flanges will be added in the cross-tie piping between the two pumps to facilitate removal of the piping for maintenance.

REASON FOR CHANGE: The existing carbon steel piping has experienced recurring leaks at the threaded connections. The cause of the leaks is thinning of the pipe wall by erosion/corrosion, which has resulted in through-wall leaks at threaded connections. The changes will reduce susceptibility of the piping to erosion/corrosion and will eliminate or reduce any leaks which might occur at threaded connections.

SAFETY EVALUATION: The function and operation of the C11 system is unaffected by the changes in pipe material, type of connections, or addition of flanges. The affected piping supplies flushing water to clean and cool the mechanical seals on the CRD pumps. Per FSAR Table 3.2-1, the piping is safety class "Other", Quality Group D and non-Q. The piping serves no safety function and failure of this piping will not compromise any safety related system or prevent reactor shutdown.

The modifications will not adversely affect the structural integrity of the associated piping. The piping has been designed per ANSI B31.1 code requirements and supported for the appropriate seismic, deadweight, and thermal loads. Therefore, this change will not create an unreviewed safety question. In addition, the affected piping is not addressed in the technical specifications and no new requirements are being added. Therefore, no changes to the technical specifications are required.

Serial Number: 92-069-NPE Document Evaluated: DCP 91/0050-S01-R00

DESCRIPTION OF CHANGE: The change provides for the modification of the existing power cable configuration from the crane bridge to the existing 5-ton hoist on the drywell valve handling crane. The modification will result in the addition of conduit, cable and a festoon system to re-wire the 480 Vac power cable to the 5-ton hoist. The modification will not significantly effect the existing power and control configuration of the 5-ton hoist.

REASON FOR CHANGE: The modification to the drywell valve handling crane will relocate a hanging power cable to the 5-ton hoist that is presenting a personnel hazard. This modification will enhance the crane's reliability, safety, and ease of use.

SAFETY EVALUATION: Apart from seismic integrity, the crane itself is not important to the safe operation of the plant or to mitigate accidents. All new equipment added to the crane (e.g., raceway, cabling and festoon system) will be installed Seismic Category II/I to ensure that it will not collapse on adjacent safety-related equipment during a seismic event.

During normal plant operation, the crane is de-energized and electrically disconnected from its source via an upstream disconnect switch. As a result, there will be no chance for an electrical malfunction to occur on the crane during plant operation.

At worst, an electrical malfunction during crane operation (during Modes 4 and 5 only) would result in: 1) inadvertent travel of the 5-ton hoist; 2) over-travel of the 5-ton hoist; 3) loss of power to the 5-ton hoist; or 4) snagging of the festoon cable on unforeseen obstructions. The impact of these malfunctions are minimized by the following considerations: 1) the selected hoist speed is very slow (10.5 fpm) and would result in minimal damage on an inadvertent travel situation; 2) over-travel of the hoist (when lifting) is prevented by internal hoist mounted limit switches; 3) loss of hoist power would result in the hoist stopping, however, mechanical brakes internal to the hoist would prevent it from dropping its load; and 4) the festoon cabling is adequately protected with fuses to provide equipment and personnel protection if it should be damaged.

The new cabling added for the 5-ton hoist is adequately sized and protected in accordance with plant design standards and the National Electric Code. The electrical cable added to the crane for this modification (enclosed in raceway and in free-air) has been reviewed to ensure that it is suitable for high radiation environments and does not adversely impact the station's existing Appendix R analysis. No technical specification change or unreviewed safety question results from this change.

Serial Number: 92-070-NPE Document Evaluated: DCP 91/0112-S00-R00

DESCRIPTION OF CHANGE: This change provides replacement battery cells for Division I (1A3) and II (1B3). The replacement battery cells shall be LCR-33 (larger cell size), manufactured by C&D - Charter Power Systems, Inc. The replacement batteries shall consist of 60 lead acid cells electrically connected in series to establish a nominal 125 Vdc battery power supply. The replacement batteries shall use existing intercell and interior connectors, terminal lugs, racks and all other equipment necessary for installation and operations.

REASON FOR CHANGE: Division I (1A3) and II (1B3) 125 Vdc batteries require replacement because they are approaching the end of their service life. However, to provide more capacity margin to serve the Division I and II loads, a new battery cell type will be utilized.

SAFETY EVALUATION: Electrical calculations have been reviewed and the capacity and voltage available with the new cells is greater than the existing installation. Thus, from a voltage and capacity perspective, the new battery cells meet or exceed these requirements. The function of the Division I (1A3) and II (1B3) 125 Vdc batteries is to supply DC power to Division I and II loads. Since the battery cell size will be increased, the capacity for the batteries to furnish current for a specified time (ampere-hour) will be increased. The new ampere-hour rate will be 2320, at 8 hours. Based on design evaluations, the divisional batteries can be fully recharged with one of the existing 400 ampere chargers and largest nontransient loading at a time less than the design limit, 12 hours. The design of the Divisional Batteries, 1A3 and 1B3, shall be maintained in accordance with Regulatory Guide 1.75 and IEEE 308, in that no single failure in any 125 Vdc system will result in conditions that prevent safe shutdown of the plant and that each system will remain located in an area separated physically from other ESF systems . Also, since the batteries (1A3 and 1B3) will be replaced, they will remain in the existing battery rooms 0C211 and 0C207 that is located in Seismic Category I structures. These rooms are ventilated and are apart from the battery chargers and distribution centers.

No technical specification change or unreviewed safety question results from this change.

Serial Number: 92-071-NPE Document Evaluated: DCP-88/0284-S00-R00

DESCRIPTION OF CHANGE: This change will perform the following modifications to the reactor recirculation pumps: 1) install thermal sleeves on the pump shaft and heat exchanger in the tapered region of the pump shaft; 2) install an impeller bore cap in the hollow portion of the pump shaft; 3) install a throttle ring on the inside bottom of the rotating baffle inner cylinder; 4) remove the inner cylinder of the heat exchanger; and 5) replace the bottom of the heat exchanger with a new (shorter vertical length) design.

REASON FOR CHANGE: On three separate occasions GGNS was required to shut down due to shaft cracking of the reactor recirculation pumps. During inspections of the removed shafts and heat exchangers similar cracking was noted on the inside of the heat exchangers (in the same general area as the shaft cracking) and chevron shaped wear indications were found on the inside of the journal bearing (attributed to flow turbulence from the hydrostatic bearing impinging on the bottom of the heat exchanger). The above described modifications are intended to prevent shaft/heat exchanger cracking by having the location of highest thermally induced stress (and resultant cracking) occur on the thermal sleeves and to prevent the journal bearing wear by moving the bottom of the heat exchanger up out of the way of flow form the hydrostatic bearing.

SAFETY EVALUATION: The modifications to the reactor recirculation pump shafts and heat exchangers do not require a change to the GGNS Technical Specifications and do not represent an unreviewed safety question. The modifications are being performed per the requirements of the ASME Code Section XI. The increase in the pump mass and moment of inertia are only .081% and .037%, respectively, and therefore have an insignificant impact on the missile and seismic analyses related to the pump. Design, materials, and construction of the modified reactor recirculation pump shafts and heat exchangers meet or exceed that of the original design and therefore the probability and consequences of analyzed accidents and transients, including shaft failure, are not increased. The specified performance of the pump with regard to flow, discharge head, critical speed, flow coastdown, etc. remains unchanged from the original design.

No technical specification change or unreviewed safety question results from this change.

Serial Number: 93-001-PSE Document Evaluated: FSAR CR PLS-92-011

DESCRIPTION OF CHANGE: This change deletes the requirement for a sprinkler system for the hydraulic baler in Room OR106 for fire suppression. A design change provided the design for a new sprinkler system; however, the baler is no longer a fire hazard since it has been drained of hydraulic oil and will not be used in the future. Administrative controls have been established by Health Physics to prevent filling the baler's hydraulic oil reservoir with flammable fluid. Specifically, Health Physics has placed a sign on the baler prohibiting filling of the hydraulic oil system. Consequently, the need for a sprinkler system for fire suppression is no longer required.

REASON FOR CHANGE: The hydraulic baler in Room OR106 will not be used by Plant Staff.

SAFETY EVALUATION: The baler's hydraulic oil reservoir has been drained and administrative controls implemented to prevent refilling. Since the baler is not a fire hazard without hydraulic oil, fire suppression is not required for Room OR106 because there are no other fire loads in that room that will require a sprinkler system for fire suppression. Installing a new sprinkler system involves significant expense for installation, testing and maintenance. Since the baler will not be used, and its hydraulic oil reservoir has been emptied to eliminate the potential for a fire from the baler, there is no potential fire hazard for the sprinkler system to suppress.

The safety evaluation concluded that the proposed change to the UFSAR does not represent an unreviewed safety question or require a change to the GGNS Technical Specifications.

Serial Number: 93-002-NSRA Document Evaluated: FSAR Chapter 15

DESCRIPTION OF CHANGE: Changes several position titles and several reporting assignments.

REASON FOR CHANGE: This change has the net effect of flattening the organizational structure of the departments and, therefore, will result in better communications, more consistent implementation of requirements and enhanced performance of the affected sections.

SAFETY EVALUATION: This change only affects the organizational structure of GGNS. The changes do not relax any requirements associated with the performance of duties at Grand Gulf. These changes do not result in any reduction in the duties, responsibilities or authority of the subject positions and, therefore, will not have any effect on current practices. These changes will not result in any reduction of the qualification requirements (ANSI 18.1 and Regulatory Guide 1.8) for the positions and will, therefore, not result in any reduction of the capabilities of the departments.

These changes are a betterment and, therefore, cannot result in any adverse consequences to the plant.

Serial Number: 93-003-PSE Document Evaluated: WO# 88825

DESCRIPTION OF CHANGE: The change involves temporarily defeating the Low Flow Trip Function and the automatic minimum flow control function of Reactor Feedpump 'A'. The low flow trip will be defeated by pulling the alarm card 1N21K005A.

REASON FOR CHANGE: A body-to-bonnet leak has developed on valve 1N21FX023 which is the root valve for flow transmitter 1N21K005A.

SAFETY EVALUATION: This change defeats equipment features of Reactor Feedpump 'A'. These features are in standby mode in the plant condition at which the work is to be performed (100% power). The operator will be briefed with raised level of attention while these functions are defeated. If an accident were to occur during this time, the feedwater system would still be able to perform its function described in the accident analyses. No unreviewed safety question is involved.

Serial Number: 93-004-NPE Document Evaluated: Disposition to QDR 0261-92

DESCRIPTION OF CHANGE: GGNS is committed to meet the requirements of Regulatory Guide 1.109 for all normal releases and non-limiting accidents (e.g., steam line breaks in FSAR 15.6.4.5) and Regulatory Guide 1.3 for the DBA-LOCA (i.e., FSAR 15.6.5.5). Regulatory Guide 1.3 (Page 1.3-2, Section 2d) requires the use of the more conservative iodine dose conversion factors given in ICRP Publication 2 while Regulatory Guide 1.109 (Table E-7) allows the use of more realistic values. QDR 0261-92 documented a discrepancy in that the original GGNS DBA-LOCA dose analysis used the Regulatory Guide 1.109 dose conversion factors instead of those dictated by ICRP-2 as required by Regulatory Guide 1.3 and GGNS FSAR Appendix 3A and Section 15.6.5.5. Therefore, the radiological consequences resulting from a postulated DBA-LOCA as described in the GGNS FSAR are not conservative relative to the FSAR commitments and regulatory requirements. As per the QDR disposition, the DBA-LOCA dose analysis and the FSAR will be revised based on new dose conversion factors.

REASON FOR CHANGE: To reflect the new analysis and changes in commitments, methodologies, input parameters, and assumptions.

SAFETY EVALUATION: The impact of this condition was assessed using the new TACT5-based program. This evaluation showed that the use of ICRP-2 dose conversion factors for the original LOCA dose analysis would have resulted in about 20% higher offsite and Control Room thyroid doses. There was no significant change in the calculated whole body radiation doses. Although the offsite limits would not have been exceeded by the use of the more conservative factors, the calculated Control Room thyroid doses would have been higher than that prescribed by regulatory limits (10CFR50 Appendix A, GDC 19). However, the dose limits would not have been exceeded using the actual Control Room inleakage value controlled via the GGNS Operating License as opposed to the higher two-unit value used in the original analysis. The use of Regulatory Guide 1.109 dose conversion factors for accident analyses was allowed by the NRC for other plants which specifically described this practice in their individual FSARs to use the Regulatory Guide 1.109 thyroid dose conversion factors for general accident analysis work. In accepting this practice for those plants, the NRC has recognized the unnecessary conservatism in the ICRP-2 values. Therefore, the nature of the GGNS deficiency is limited to the FSAR description of the regulatory guide commitments and does not introduce any unreviewed safety question nor does it impact any technical specification requirements or bases.

Serial Number: 93-005-NPE Document Evaluated: FSAR CR NL-92/004

DESCRIPTION OF CHANGE: The change updates the Division I and Division II battery requirements specified in the FSAR. As a result of efforts to review the design basis of the electrical systems, the Divisions I and II battery load profiles were revisited and installed as-built loads were calculated. The resulting minimum required test profiles, with margin built in, were calculated.

Calculations were also performed to ensure battery capacity for the required duty cycle. These calculations use the methodology presented in IEEE 485-1978 and demonstrate that the existing batteries are sized for the design conditions. Also, calculations were performed which demonstrate that the emergency loads which are required to operate during the battery design basis event receive adequate voltage.

REASON FOR CHANGE: Modify the FSAR to more accurately reflect the installed condition of the plant and meet commitment to the NRC contained in GNRO-91/00171 that the battery service test loads and minimum allowable battery terminals voltages be supported by the battery sizing and voltage drop calculations.

SAFETY EVALUATION: The determination of the battery load profile and terminal voltages and the testing of the system with the calculated profile ensures the operability of the system. The load profile was developed from the postulated design basis accident scenario for the battery and the capability of the battery bank was determined using IEEE 485-1978 battery sizing methodology. Design margin is included to ensure that calculated loads will envelope actual emergency load and to account for ambient temperature conditions and battery age. Also, voltage drop calculations have been performed which verify that essential components receive adequate voltage to perform their safety functions. The revised profile is bounded by the original profile which was based on assumptions of load. The load profile and voltage requirements are appropriate for service testing which demonstrates battery capability.

Serial Number: 93-008-PSE Document Evaluated: UFSAR CR PLS-93-006

DESCRIPTION OF CHANGE: Changes Surveillance Requirement 4.7.6.1.2c in UFSAR Appendix 16A to eliminate the restriction of inspecting the diesel driven fire suppression pump only in plant shutdown.

REASON FOR CHANGE: This scheduling change will reduce outage-related activities for Mechanical Maintenance.

SAFETY EVALUATION: The requirement to perform this surveillance requirement during shutdown has no substantive safety related basis. The firewater pump diesels are required to be operable at all times, regardless of plant condition, to ensure that sufficient water capacity is available for fire suppression. Impairment of fire protection system components occurs when their ability to provide suitable quantities of water for fire suppression is diminished, regardless of plant condition. The requirement specified in Technical Requirements Manual Paragraph 3.7.6.1 for loss of fire suppression water system components is irrespective of plant condition, and specifies "At all times" for applicability. Accordingly, the requirement to perform the diesel engine inspections during shutdown is inconsistent with the applicability and should be changed to reflect actual requirements and limitations.

This change will not affect the fire water pump diesels or their ability to perform their intended design function. There will be no design changes required to support this change. Operation of the fire water pump and diesel engine will not be changed in any way by the change. System performanc[illegible]ll not be affected by the change. Maintenance and surveillance procedures an[illegible]ivities will be unaffected by the change. No unreviewed safety question[illegible]hange to the technical specifications result from this change.

Serial Number: 93-009-NSRA Document Evaluated: Commit. Nos. 15797, 16274, 16275

DESCRIPTION OF CHANGE: Currently, if the Post Accident Sampling System (PASS) panel or a specific PASS sample pathway is found inoperable, operability will be restored within 24 hours or Region II NRC will be notified by telephone to describe the event and the assessment of actions planned to restore operability. The reporting requirement for the PASS system is being deleted. The reporting requirement set forth in 10CFR50.72 which requires one-hour notification in the event of a major loss of emergency assessment capability is not affected by this change.

REASON FOR CHANGE: Initially, the reporting requirement was to heighten the awareness of plant personnel of the importance of PASS operability and to ensure the NRC was made aware of PASS inoperability. However, based on current practices and verbal discussion with Region II management, it has been determined that the reporting requirement is not beneficial in assuring PASS operability. Therefore, there is no benefit by reporting the inoperability of a specific pathway of the PASS system.

SAFETY EVALUATION: This safety evaluation concluded that the change does not involve an unreviewed safety question. There is no accident evaluation in the UFSAR for the PASS. PASS is used after an accident as a means to estimate the extent of core damage but has no role in the mitigation of an accident or safe shutdown of the reactor. The deletion of the commitment to report PASS inoperabilities would not affect compliance with regulations or compliance with the plant's operating license.

Serial Number: 93-010-NPE Document Evaluated: DRNs 3831 & 3832

DESCRIPTION OF CHANGE: Revise UFSAR Figures 8.3-007 and 8.3-007A to reflect the actual fuse part numbers including vendor identification, for the non-Class 1E UPS power panel fuses.

REASON FOR CHANGE: This change to the UFSAR is required to provide accurate information for the fuses utilized in the Non-Class IE UPS power panels depicted on Figures 8.3-007 and 8.3-007A and is made to clarify existing as-built system information.

SAFETY EVALUATION: This change is a software design change required to provide accurate UFSAR detail and equipment information. This change does not affect the analysis or description of the UPS system defined by related UFSAR Section 8.3.1.1.4.1.5 nor does it change any technical specification details or requirements for plant power systems. No plant modifications result from this change. The change does not represent an unreviewed safety question or require a change to the GGNS Technical Specifications.

Serial Number: 93-011-NPE Document Evaluated: UFSAR CR NPE-93-005

DESCRIPTION OF CHANGE: This change eliminates the position of Manager, Engineering Support from the Nuclear Plant Engineering organization, combines Drafting into Configuration Management, transfers Configuration Management to Civil Engineering and Procurement Engineering to Electrical Engineering Discipline. Engineering Planning and Control Group reports to the Director, Design Engineering. There is also a new position created for Technical Assistant reporting to the Director, Design Engineering.

REASON FOR CHANGE: This change is made to increase the efficiency and effectiveness of the Nuclear Plant Engineering organization by eliminating an unnecessary level of management and consolidating the groups.

SAFETY EVALUATION: The change is administrative in nature and has no adverse affects on the functioning of the design engineering organization. This change will improve effectiveness by allowing discipline principal engineers involved in the engineering support functions to have a better control on cost and scheduling. The Director, Design Engineering also has an additional position of Technical Assistant assisting him on the technical issues.

No unreviewed safety question or technical specification change result from this change.

Serial Number: 93-012-PLS Document Evaluated: 04-1-01-P11-1 TCN 12

DESCRIPTION OF CHANGE: This change will allow the discharge of potentially radioactively contaminated liquids from the Condensate Storage Tank (CST) dike sump into the Liquid Radwaste System.

REASON FOR CHANGE: This change will allow the pumpdown of the CST dike to the Radwaste G17 system waste surge collection tank without requiring Plant Chemistry to sample dike for discharge into the normal plant liquid radwaste collection systems.

SAFETY EVALUATION: Such waste as is collected by the Liquid Radwaste System is analyzed for batch type releases by Plant Chemistry prior to discharge to the environs. All such releases are continuously monitored by the Liquid Radwaste System with automatic isolation capability upon receipt of an Effluent Hi Rad signal.

Therefore, allowing the discharge of potentially radioactive liquids into the liquid radwaste collection system will not result in an unmonitored release to the environs.

This change does not represent an unreviewed safety question or require a change to the GGNS Technical Specifications.

Serial Number: 93-013-NPE Document Evaluated: MCP 92/1006 R00

DESCRIPTION OF CHANGE: This change will install a 45 second time delay relay in both the A & B circuits of the Fuel Pool Cooling and Cleanup (FPCC) System such that differential flow transients due to system startup or operator manipulation of the system would be dampened. A Control Room annunciator will also be installed to immediately alert operations personnel if a high differential flow condition develops and either of the time delay relays are energized.

REASON FOR CHANGE: At present the (HIGH FILTER/DEMINERALIZER DIFFERENTIAL FLOW) circuit is such that upon receipt of a high differential flow signal from the filter demineralizer leak detection system the associated isolation valves will isolate and the associated FPCC pump will receive a stop signal. This causes problems during any FPCC system manipulation by the operators. The operator must put this portion of the system in bypass to prevent any spurious pump trips or system isolation.

SAFETY EVALUATION: The changes will not compromise any existing safety related system, structure or component nor will they prevent safe reactor shutdown. The addition of this time delay has been reviewed to ensure that the maximum system water loss would not exceed 3000 gallons.

The relays are not required to mitigate the consequences of any evaluated transient or accident. No new unbounded interfaces are created and no new failure modes are introduced. These changes will therefore not introduce an unreviewed safety question. This portion of the circuit is not currently addressed in the technical specifications and these changes will not require that they be added to the Grand Gulf Technical Specifications.

Serial Number: 93-016-NPE Document Evaluated: Calc MC-Q1Z51-91146 R00

DESCRIPTION OF CHANGE: This change supports the addition of above ground gasoline and diesel storage tanks for use by site maintenance vehicles and other equipment. They will replace existing underground tanks.

REASON FOR CHANGE: To evaluate the new fuel tanks at the locations specified.

SAFETY EVALUATION: The proposed addition of the fuel storage tanks in the locations evaluated will not result in unacceptable concentrations of toxic chemicals in the Control Room in the event of a spill and/or fire involving these tanks. Since the concentrations of toxic chemicals in the Control Room are a function of only the distances between the Control Building and the locations of the fuel tanks, these fuel tanks can be installed at any location as long as their distances from the Control Building are at least as great as those used in this calculation. The proposed change does not increase the consequences of an accident already evaluated in the UFSAR since the Control Room operators' abilities to function are not degraded by the postulated spill and/or fire. The tanks are non-safety grade. They are not relied upon to function after an accident. The change does not increase the probability of an accident due to the malfunction evaluated (spill and/or fire) since the Control Room operators abilities are not degraded. The change does not create the possibility of an accident or an equipment malfunction different from those already evaluated in the SAR. The margin of safety is not reduced since the habitability of the Control Room is not impaired.

Serial Number: 93-017-NPE Document Evaluated: Calc MC-Q1G41-92015

DESCRIPTION OF CHANGE: The fuel cycle loading and decay heat loads are updated based on existing historical information and Calculation MC-Q1G41-92015. This information includes the fuel cycle duration, the refueling outage length, the number of fuel bundles moved for each previous outage as well as RF05, and projected bundle moves for future outages.

REASON FOR CHANGE: Updating of the UFSAR Tables will provide a realistic fuel cycle loading and decay heat loads for Grand Gulf Nuclear Station based on results provided in Calculation MC-Q1G41-92015.

SAFETY EVALUATION: Updating the UFSAR Tables will provide realistic decay heat load and fuel cycle loading data and decay heat loads for Grand Gulf Nuclear Station, based on the results from Calculation MC-Q1G41-92015. Using the new calculated values, this change updates the decay heat load results of Tables 9.1-10, 9.1-12, 9.1-13 and Pages 9-ix, 9.1-15 and 9.1-19. The update of Table 9.1-10 addresses information reflecting the fuel moves for past as well as future refueling outages. The updated information in Table 9.1-12 provides the Grand Gulf maximum spent fuel pool heat loads. This heat load is calculated for successive fuel cycles for which the cumulative spent fuel may be accommodated in the pool. Table 9.1-12 provides a decay heat load at the eve of RF012 which is within the cooling capacity of one fuel pool cooling and cleanup heat exchanger. The updated Table 9.1-13 provides the maximum core reload spent fuel pool heat load based on a core reload 284 fuel bundles. Page 9-ix is updated to describe Tables 9.1-12 and 9.1-13. Page 9.1-15 is updated to describe the operating time for the offloaded reactor fuel for the outage; which is used as input to Table 9.1-12. Page 9.1-19 is updated to describe the loading cycle scenario in Table 9.1-12.

No technical specification change and no unreviewed safety question result from this change.

Serial Number: 93-018-NSRA Document Evaluated: Refueling Temporary Hatch Cover

DESCRIPTION OF CHANGE: The change deletes the requirement to stage a temporary hatch cover for the containment equipment hatch and substitutes appropriate controls in the refueling Integrated Operating Instruction for initiating immediate actions to re-establish secondary containment as a more appropriate use of resources. This change also deletes the need for installing temporary covers on open (bonnet removed) primary containment isolation valves.

REASON FOR CHANGE: The temporary hatch cover or other temporary valve covers were not designed or built to meet any safety system criteria; therefore, it is unlikely that they would provide any real mitigative function, such as reducing release rate. On the other hand, secondary containment is a qualified known barrier and applying available resources to re-establish this barrier would be more reasonable.

SAFETY EVALUATION: Use of the temporary covers is not credited as a mitigative measure in any safety analyses or other design or license basis documents for events of concern (e.g., fuel handling accident and vessel draindown). Secondary containment is required to be operable before conditions can exist for other events of concern (e.g., loss of shutdown cooling and other load drops). Initiating action to establish secondary containment has a bigger potential for minimizing releases. Therefore, no unreviewed safety question exists for this change.

Serial Number: 93-019-NPE Document Evaluated: MNCR 92/0028
3rd Submittal

DESCRIPTION OF CHANGE: Post-LOCA ambient temperatures, within certain Control Building rooms, that are in excess of values specified in the UFSAR have been identified. As a result of this, the values in the UFSAR and other documents need to be updated.

REASON FOR CHANGE: Electrical heat loads used as input into Calculation 3.8.31 were reviewed and found to be in error. These inputs were revised to reflect the correct electrical loads. A calculation was performed to reflect the correct loads. This resulted in post design basis accident temperatures in excess of the 104°F stated in the FSAR.

SAFETY EVALUATION: All affected safety related electrical equipment has been evaluated for operation under the elevated ambient temperatures expected in the design basis accident environment following the event. All evaluated equipment has been determined to be able to perform its safety function under the postulated conditions, or else not required for safe shutdown. Safety related equipment determined not necessary for safe shutdown was further evaluated to verify that it would not degrade the Class 1E electrical source.

The temperature of Elevation 133 affects the temperature of Elevation 148 which in turn affects the cooling load seen by the Control Room HVAC system. The impact of this temperature increase on the cooling load for the Control Room HVAC System was evaluated and it was determined that this cooling load is within the capabilities of the Control Room HVAC unit. It should be noted that, even if flows described above are not established technical specification temperature limits of 90°F for the Control Room are not exceeded.

The impact of the elevated temperatures on the equipment located in the areas served by the Safeguard Switchgear and Battery Room Ventilation System and the impact of these temperatures on adjacent areas has been evaluated and determined to be acceptable. The conditions these areas will experience are well within limits and will not adversely affect any equipment in the rooms.

At winter design conditions, the design minimum indoor room temperature at 65°F could not be maintained with both fans running on low speed. The calculation conservatively determined that the bulk air temperatures in the safeguard battery and switchgear rooms could be as low as 58°F at winter design outdoor

temperatures. The electrical equipment was reviewed and it was concluded that no adverse effects would result. No technical specification change and no unreviewed safety question result from this change.

Serial Number: 93-020-NPE Document Evaluated: UFSAR CR NPE-92-041

DESCRIPTION OF CHANGE: This change provides the justification necessary to implement the upgraded LOCA dose models.

REASON FOR CHANGE: Certain unfiltered bypass leakage pathways were discovered and repaired during RF05. In the course of investigating this event it was determined that LOCA dose calculations did not include an allowance for this bypass leakage. It was also noted that Grand Gulf dose models contained unnecessary conservatism and had not been updated for some time.

In order to provide a more realistic (yet still conservative) safety analysis tool while implementing necessary corrective action an extensive effort was launched to upgrade the LOCA dose calculation models, assumptions and inputs.

The purposes of these revisions are to update the offsite accident χ/Q values to incorporate more current meteorological data and to update the design basis LOCA offsite dose analysis to reflect changes in the design basis assumption and regulatory guidance.

SAFETY EVALUATION: The proposed UFSAR changes do not require a change to the plant technical specifications nor do they involve a change to any safety related system or component. These analyses do not affect the probability of occurrence of any accidents or equipment malfunction. The consequences for any previously analyzed accident or equipment malfunction are not increased, nor is safety margin decreased, since no physical or operational changes to the plant are implemented. The post-LOCA offsite (exclusion area boundary and low population zone) doses remain well within the limits of 10CFR100 and the Control Room doses are within the limits provided in 10CFR50, General Design Criteria 19.

Serial Number: 93-021-PSE Document Evaluated: WO #96028

DESCRIPTION OF CHANGE: To provide instructions for inspecting the Emergency Core Cooling System (ECCS) Pump Suction Strainers for Residual Heat Removal (RHR)-A, RHR B, RHR C, Low Pressure Core Spray (LPCS), and High Pressure Core Spray (HPCS). It also provides instructions for inspecting the Reactor Core Isolation Cooling (RCIC) Suction Strainer and the Suppression Pool Suction Strainers.

The inspection will be done during power operations and will use the remote controlled Minirover (robotic, photographic submarine).

REASON FOR CHANGE: There currently exists an industry wide and NRC concern about ECCS pump suction strainers. The purpose of the inspection is to verify the structural integrity of the strainers.

SAFETY EVALUATION: The safety evaluation considered the effects of ECCS suction strainer damage due to impaction from the Minirover; the potential of the 75-pound Minirover becoming a missile during its time in the containment; and the potential effects of the Minirover emitting radio frequency (RF) waves or induced voltages into underwater suppression pool cables.

The evaluation concluded that the loss of one ECCS loop was adequately addressed by technical specifications; that work order controls would prevent the Minirover from becoming a missile due to the dynamic effects of postulated accidents; and that the Minirover's emitted induced voltage and RF waves are inconsequential.

This change does not result in an unreviewed safety question or a change to the technical specifications.

Serial Number: 93-022-NPE Document Evaluated: MNCR 0018-93

DESCRIPTION OF CHANGE: This change revises the drawings and UFSAR figures to reflect the actual configuration of spectacle blind flanges on the Residual Heat Removal (RHR) Pumps A, B & C and the Low Pressure Core Spray (LPCS) Pump. This will allow the spectacle blind flanges which were left installed at the discharge flanges of the pumps to remain, even though the drawings stated they were to be replaced with ring spacers after hydrotesting and prior to system operation.

REASON FOR CHANGE: The continued presence of the flanges constitute a nonconformance. Drawings and UFSAR figures must be revised to nullify the nonconformance.

SAFETY EVALUATION: The presence of the flanges does not adversely affect the structural integrity of the affected systems. The pump nozzles and piping meet ASME Section III (ASME III) code allowables and the system is adequately supported for the appropriate loads.

The operability of the pumps is not affected by this change; therefore, the ability of the RHR and LPCS systems to mitigate accidents remains unchanged. Thus, this modification will not increase the consequences of any accident previously evaluated in the UFSAR.

The presence of the flanges does not increase the probability of occurrence of a malfunction of equipment important to safety previously evaluated in the UFSAR. The operational reliability of the RHR and LPCS systems is not affected. Thus, all intended functions of the RHR and LPCS systems 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 continued presence of the flanges will not affect the function or operation of the affected systems or any other systems and will not create an unreviewed safety question. The spectacle blind flanges are not addressed in the technical specifications; therefore, no change to the technical specifications is required.

Serial Number: 93-023-PSE Document Evaluated: WO #96028

DESCRIPTION OF CHANGE: This change provides instructions for inspecting and video taping the suppression pool floor/walls:

-- to establish overall cleanliness level

-- to determine scope of cleaning requirements for RF06

-- to provide input for the GGNS suppression pool/Emergency Core Cooling System (ECCS) strainers task force

The inspection will be done during power operations and will use the remote controlled MINIROVER (robotic, photographic submarine).

REASON FOR CHANGE: This change aids in ensuring suppression pool cleanliness. There currently exists an industry wide and NRC concern about ECCS pump suction strainer integrity and overall suppression pool cleanliness.

SAFETY EVALUATION: The safety evaluation considered the effects of ECCS suction strainer damage due to impaction from the Minirover; the potential of the 75 pound Minirover becoming a missile during its time in the containment; and the potential effects of the Minirover emitting VHF waves or induced voltages into underwater suppression pool cables.

The evaluation concluded that the loss of one ECCS loop was adequately addressed by technical specifications; that work order controls would prevent the Minirover from becoming a missile due to the dynamic effects of postulated accidents; and that the Minirover's emitted induced voltage and VHF waves are inconsequential.

Serial Number: 93-024-PLS   Document Evaluated: MAEC 82/0093

DESCRIPTION OF CHANGE: Remove the Health Physics (HP) commitments in MAEC-82/00093, Section 8. These commitments required that surveys be performed by individuals who had been specifically qualified by direct observation of the HP Supervisor or alternate and that radiation work permit authorizations were to be by the HP Supervisor, alternate or Senior ANSI-qualified HP.

REASON FOR CHANGE: Staffing and experience levels of GGNS HPs no longer warrant this level of supervisory oversight.

SAFETY EVALUATION: The commitments are above and beyond the requirements of 10CFR20 as required by Section 6.11 of GGNS Technical Specifications. Deleting these commitments will not require a technical specification change, will not adversely affect the Health Physics department's ability to protect plant personnel and the public from exposure to radiation and will not create any significant radiological hazard to plant staff or the public. The commitments were a result of an inspection of the qualifications of the utility Health Physics personnel. At that time, six (6) ANSI-qualified HPs, with limited commercial experience, were employed by the utility. The additional controls were deemed necessary to ensure proper actions by a very inexperienced staff. Of the more than 50 GGNS ANSI-qualified HPs currently on site, over 30 have five or more years of HP experience. Training and qualification to perform HP functions are controlled by an INPO accredited program. Health Physics commitments to technical specifications, FSAR, regulatory guides and the training program provide assurance of acceptable quality in the administration and oversight of the HP program. The requirements listed in Technical Specification Sections 6.2, 6.3, 6.4, 6.8, and 6.11; UFSAR Sections 12.1, 12.5 and 13.1 and the applicable regulatory guides will not be changed by rescinding these commitments. This change is administrative and will have no adverse affect on the safe operation of the plant.

This change does not result in an unreviewed safety question or a change to the technical specifications.

Serial Number: 93-025-PLS Document Evaluated: EER 93/6121

DESCRIPTION OF CHANGE: Increase the Control Rod Drive Hydraulic Unit Area Radiation Monitor (ARM) setpoint from 15 mr/hr to 30 mr/hr.

REASON FOR CHANGE: The proposed change restores the ARM function, which is to warn personnel of increasing or abnormally high radiation levels.

SAFETY EVALUATION: This ARM is in continuous alarm. The general area readings are 8-30 mr/hr. There is an overhead posted high radiation area. The area radiation levels have increased over time as a result of plant operation.

Area surveys by station Health Physics personnel show that the monitor readings indicate actual radiation levels. Instrument and Calibration personnel have confirmed proper monitor operation.

This change does not result in an unreviewed safety question or a change to the technical specifications.

Serial Number: 93-026-NPE Document Evaluated: CN 93-0024

DESCRIPTION OF CHANGE: This change removes the limit switch from the Control Room Heating, Ventilation and Air Conditioning (HVAC) unit condenser water flow control valves.

REASON FOR CHANGE: The change is to prevent the tripping of the Control Room HVAC unit due to loss of condenser water flow, caused by the flow control valves operating very close to their seats.

SAFETY EVALUATION: The limit switches were installed to trip the unit's compressor in the event condenser water flow was lost. With the limit switches removed, the units are still protected from a loss of condenser water flow. The units will trip on either low suction pressure or high discharge pressure. Operating the valves so close to their seats could create the possibility of erosion/corrosion. To preclude this, the valves will be included in the GGNS erosion/corrosion monitoring program.

This change does not result in an unreviewed safety question or a change to the technical specifications.

Serial Number: 93-027-NSRA Document Evaluated: 01-S-01-4

DESCRIPTION OF CHANGE: Changes Assistant Manager, Plant Maintenance to Technical Coordinator, Plant Maintenance.

REASON FOR CHANGE: This change has the net effect of flattening the organizational structure of the departments and, therefore, will result in better communication, more consistent implementation of requirements and enhanced performance of the affected sections.

SAFETY EVALUATION: This change only affects the organizational structure of GGNS. The change does not relax any requirements associated with the performance of duties at Grand Gulf. This change does not result in any reduction in the duties, responsibilities or authority of the subject position and, therefore, will not have any effect on current practices. This change will not result in any reduction of the qualification requirements (ANSI 18.1 and Regulatory Guide 1.8) for the position and will, therefore, not result in any reduction of the capabilities of the department.

This change is a betterment and, therefore, cannot result in any adverse consequences to the plant.

Serial Number: 93-030-NPE Document Evaluated: DCP 90/0005-2 R00

DESCRIPTION OF CHANGE: This change will provide the design requirements for all the necessary air accumulator supports, piping modifications, valve replacements, removal of stop check valves and removal of one relief valve on each set of Automatic Depressurization System (ADS) accumulators associated with the replacement of existing carbon steel ADS and non-ADS accumulators with stainless steel accumulators.

REASON FOR CHANGE: Air accumulators and receivers used to supply the main steam isolation valves and the main steam safety/relief valves (SRV) were fabricated out of carbon steel with a protective coating on the interior surface to prevent corrosion. In 1983, failure of the protective coating on the accumulators was reported. To correct this problem, a design change was issued to repair the damaged coating. The repair consisted of the addition of a handhole to allow the interior surface to be sandblasted and application of a new protective coating to the interior of the accumulators. The repair to the accumulators was considered temporary due to the finite life of the coating. This design change will provide a permanent solution to the degradation of coatings inside the ADS air accumulators by replacing the carbon steel accumulators with stainless steel accumulators.

A consistent design replacement accumulator check valve shall be installed for the applicable accumulators to replace the valves installed during RF04.

Also, valves and piping are being removed that are not required to maintain system function.

SAFETY EVALUATION: The change does not result in any operational or functional change to the affected system. The size of the ADS and non-ADS accumulators will not change. The material for accumulators, associated piping and drain valves will be changed from carbon steel to stainless steel in order to provide improved corrosion resistance. Replacement of check valves on the accumulator inlet piping will not affect the design function of these valves. Removal of stop check valves which are currently locked closed will not affect the function of the system. Calculations have been performed to show that removal of one out of two relief valves associated with each pair of accumulators will not adversely affect the pressure relief capacity. All piping and support modifications have been designed in accordance with ASME Section III or ANSI B31.1 code requirements as applicable. Therefore, this change will not require a change to the technical specifications and will not create an unreviewed safety question.

Serial Number: 93-033-PSE Document Evaluated: 07-S-14-187, R7

DESCRIPTION OF CHANGE: Revise the UFSAR concerning installation of the portable radiation shield (cattle chute). Current revision of the UFSAR Section 9.1.4.2.10.2.3.1 provides information that six studs are removed in line with the fuel transfer canal to provide a path for fuel movement. Seven studs must be removed in this path to facilitate installation of the cattle chute which is the path for fuel movement.

REASON FOR CHANGE: To correct the current revision of UFSAR 9.1.4.2.10.2.3.1 which provides information incorrectly as to the number of vessel studs that must be removed to facilitate cattle chute installation for fuel movement during refueling activities.

SAFETY EVALUATION: This information provided in the UFSAR is descriptive in nature and does not have any basis for the exact number of studs necessary to be removed. The number of studs removed is inherently required due to cattle chute installation which provides shielding on the drywell bulkhead during fuel movement. Procedure requirements continue to adhere to handling requirements for each stud singly and grouped together in the vessel head stud rack. All requirements under the heavy loads program at GGNS are met and will continue to satisfy the applicable requirements of NUREG-0612.

This change does not result in an unreviewed safety question or a change to the technical specifications.

Serial Number: 93-037-NPE Document Evaluated: CMU Wall Modifications in the Control Building

DESCRIPTION OF CHANGE: An NRC onsite inspection of Concrete Masonry Unit (CMU) modifications produced NRC Violation 416/83-12-01, Failure to Complete Masonry Wall Modifications in accordance with drawing requirements. In response to the subject NRC violation, a field walkdown of the CMU walls in the Control Building and Auxiliary Building, that have safety related equipment attached to or located in proximity to, was performed to verify the "as-built" condition of these walls. During this walkdown several of the CMU wall structural supports in the Control Building were found that should be modified to enhance their structural integrity. Calculation analysis of the subject supports showed that the actual stresses were less than, but close to, the design allowable stresses. Although structurally adequate now, future attachments to the subject walls could jeopardize the wall's structural integrity. Therefore, modifications should be implemented to enhance the supports' structural integrity, thereby strengthening the CMU walls for future attachments.

REASON FOR CHANGE: These changes are necessary to ensure that future addition of loads to the CMU walls do not exceed the design allowable stresses for the walls. The walls are structurally adequate at the present time, but the calculated loads are close to the allowable stresses.

SAFETY EVALUATION: The walls are considered structurally adequate now. The addition of structural supports to the CMU walls are an enhancement to the existing facility, and will not increase the probability or consequences of an accident or malfunction. In addition, no margins of safety will be reduced by this enhancement.

Serial Number: 93-038-NPE Document Evaluated: MCP 92/1059 R00

DESCRIPTION OF CHANGE: This change replaces the existing actuator assemblies on two Standby Service Water (SSW) System pump discharge valves and two cooling tower return valves with larger actuator assemblies.

REASON FOR CHANGE: It was determined that the actuators installed on the SSW pump discharge valves and cooling tower return valves had been operated in excess of their design torque rating. Justification was provided for continued operation of the valves during operating cycle 6 until hardware upgrades could be made in RF06. This change implements the upgrade to a larger actuator assembly for the affected valves.

SAFETY EVALUATION: Implementation will not adversely affect the function, operation or operability of the SSW pump discharge and cooling tower return motor operated butterfly valves, and will therefore not increase the probability of occurrence or increase the consequences of an accident previously evaluated in the SAR. The modification will increase the reliability of the valves by permitting them to operate with shaft torque values that are within the manufacturer's design torque ratings for the actuator assemblies and the maximum torque allowables for the valve components while not adversely affecting valve operating time. Therefore, the modification will not increase the probability of occurrence or the consequences of a malfunction of equipment important to safety previously evaluated in the SAR. The replacement of the actuator assemblies with larger actuator assemblies on the four valves was evaluated and it was determined that: a) the increase in actuator assembly torque rating and output would permit operation of the valve under all conditions without exceeding actuator design ratings or calculated valve stress allowables, b) the slight decrease in valve operating time will have no detrimental impact on valve or system operation, c) the new actuator assemblies are suitable for safety related use outside of containment, d) there is no adverse impact to the seismic qualification of the valves or to the piping stresses by the installation of heavier actuator assemblies, e) there is no adverse impact to the electrical distribution system created by the increased loads presented by the new 10 ft-lb motors, and f) no new failure modes are introduced by the modification. 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 SAR. 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 since it will not adversely affect valve or system function or operation.

Serial Number: 93-039-NPE Document Evaluated: DCP 92/0028-01 R0

DESCRIPTION OF CHANGE: This change replaces/reroutes heat damaged cables in the Feedwater Heater Room 'B'. These cables will be replaced by splicing and rerouting them around the feedwater heater room. Specifically, this change evaluates the necessary work to splice/reroute the various cables and completes the physical work to install new splice boxes, conduits and trays in the condenser bay.

REASON FOR CHANGE: Many cables have been identified to have brittle and/or cracked cable jackets caused by excessive heat in the immediate area of the cable trays which they are routed in. Therefore, a new routing path for these cables in Feedwater Heater Room 'B' must be made to ensure no recurrence of this heat-related problem occurs to ensure reliability of the affected systems.

SAFETY EVALUATION: The changes will not compromise any existing safety related system, structure or component, nor will they prevent safe reactor shutdown. The cables for the various systems involved have non-safety related equipment functions.

The new splice boxes, conduits and trays provide new and alternate routing paths for the involved cables to be rerouted around Feedwater Heater Room 'B'. Each cable will be spliced individually to ensure positive identification, therefore, introduce no new modes for failure that could be propagated into any existing system.

All cabling and raceway modifications to be performed will be in accordance with the separation requirements of Regulatory Guide 1.75.

No evaluated accident is affected by any addition of these splice boxes and conduits. These components will not be required to mitigate the consequences of any evaluated accident. No components of any present system will be changed. No new interfaces are created and no new failure modes are introduced. These changes will, therefore, not introduce an unreviewed safety question.

Serial Number: 93-040-NPE Document Evaluated: DCP 89/0033-3

DESCRIPTION OF CHANGE: For the Residual Heat Removal (RHR) System (E12), the analog Westronics M5E recorder E12R601 will be replaced with a Westronics model 2400 digital recorder.

REASON FOR CHANGE: The Westronics M5E recorder is obsolete and spare parts are difficult to obtain.

SAFETY EVALUATION: In the event of an accident, the recorder will be used by the operators to monitor the exhaust temperature of the RHR heat exchangers. The recorder is required for Regulatory Guide 1.97, Type D, Category 2 indication and is classified as safety related display instrumentation. The recorder is non-safety related, however, and is supplied non 1E UPS power.

The 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 recorder. This change will be an improvement in terms of recorder reliability and monitoring capability. The non-safety related recorder has been installed so that the seismic qualification of the safety related panel in which it is mounted will be maintained. Regulatory Guide 1.75 separation is also maintained. The changes will not compromise any existing safety related system, structure or component. The failure of the recorder and the circuits to which it is connected will not initiate any evaluated transient or accident. The E12 system operation and function will not change. No new interfaces are created. This change will therefore not introduce an unreviewed safety question. The recorder is not currently addressed in the technical specification and this change will not require that it be added to the technical specification.

Serial Number: 93-041-NPE   Document Evaluated: MCP 91/1068 R1

DESCRIPTION OF CHANGE: The method of connecting fuel servicing tools to the Fuel Handling Platform (FHP) Monorail Auxiliary Hoist is changed. The hoist's current configuration uses twin cables to provide a redundant load path during the movement of new fuel with the Auxiliary Grapple. Movement of irradiated fuel with this hoist is not permitted. When moving non-fuel loads, a dual cable adapter (DCA) is used. This change specifies different, dual cables for use with the auxiliary grapple and provides a single cable connection detail for non-fuel loads, which will replace the existing DCA.

REASON FOR CHANGE: The existing cables have threaded connections which are too short, and lack cotter pin holes, thereby preventing connection of the auxiliary grapple and DCA per existing drawings. Additionally, GGNS has observed a history of bending in the DCA's bottom connection bolt. Excessive bending stresses appear to result during load maneuvers, which are unavoidable. Since the bottom connection bolt cannot be easily replaced, the entire DCA must be replaced at significant cost whenever the bolt is bent. Therefore, a different cable is required to allow connection of the auxiliary grapple per the existing detail. Also, a connection detail which is more tolerant of incidental loading and more cost effective is desired to replace the DCA.

SAFETY EVALUATION: Technical Specification 3/4.9.6 specifies interlock settings and other requirements which have been imposed to insure that irradiated fuel is moved only with the main hoist, to insure that hoists have sufficient capacity, and to protect fuel bundles from excessive lifting force in the event they become stuck. Additionally, Technical Specification 3/4.9.12 imposes limits on movement of the FHP to protect personnel from excessive radiation in the vicinity of the Horizontal Fuel Transfer System (HFTS). These changes affect equipment used to move new fuel and non-fuel loads and will not require irradiated fuel to be moved with this hoist. No additional loads are added. Additionally, these changes do not increase or decrease the lifting force generated by the hoist or affect any parameter which could increase personnel radiation exposures due to the HFTS. Therefore, the requirements of Technical Specifications 3/4.9.6 and 3/4.9.12 do not require change, and the margin of safety associated with these technical specifications is not reduced.

UFSAR Sections 9.1.4.1 and 9.1.4.2.7.3 require that a redundant load path exists for hoists lifting fuel "so that no single component failure will result in a fuel bundle drop". The auxiliary grapple will connect directly to the two new cables, per an existing connection detail, thereby providing a redundant load path. The single cable connection detail, which replaces the DCA, will not be used to lift new or irradiated fuel, and therefore does not require a redundant load path.

Therefore, the probability of an accident evaluated in the UFSAR, or a malfunction which could result in an accident evaluated in the UFSAR has not increased. Similarly, no parameter has changed which could alter radionuclide population, release rate or duration; create new release mechanisms; or impact radiation release barriers.

Accidents resulting from dropped fuel and non-fuel loads, during fuel handling operations in the Auxiliary Building, have already been addressed in Section 15.7.4 of the UFSAR. The cables and other hardware used to attach fuel servicing equipment to the Monorail Auxiliary Hoist (MAH) do not interface with any other system, structure or component. Therefore, the changes to these components will not create the possibility of an accident or a malfunction which has not already been considered in the load drop analyses in the UFSAR.

Serial Number: 93-042-NPE Document Evaluated: DCP 89/0033-4

DESCRIPTION OF CHANGE: For the Turbine Control System (N32), the Leeds & Northrup (L&N) recorders N32R615, 619 and 620 will be replaced with a single Westronics model 3000 digital recorder (N32R650). The L&N recorders N32R637, 638 and 639 will be replaced with a single Westronics model 3000 recorder (N32R651). N32R640 will be replaced with a Westronics model 3000 recorder.

REASON FOR CHANGE: The L&N model 512 recorder is obsolete and spare parts are difficult to obtain.

SAFETY EVALUATION: The affected recorders are non-safety related and are not required to perform any active or passive safety related functions. They are not required for Regulatory Guide 1.97 indication. The 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 recorders. This change will be an improvement in terms of recorder reliability and monitoring capability. The non-safety related recorder N32R640 has been installed so that the seismic qualification of the safety related panel in which it is mounted will be maintained. Regulatory Guide 1.75 separation is also maintained. The other recorders are mounted in a non-safety related panel.

The changes will not compromise any existing safety related system, structure or component. The failure of the recorders and the circuits to which they are connected will not initiate any evaluated transient or accident. The N32 system operation and function will not change. No interfaces with safety related or important to safety systems are created. This change will therefore not introduce an unreviewed safety question. The recorders are not required to mitigate the consequences of any evaluated transient or accident. The recorders are not currently addressed in the technical specification and this change will not require that they be added to the technical specification.

Serial Number: 93-043-NPE Document Evaluated: DCP 89/0033-2

DESCRIPTION OF CHANGE: For the Area Radiation Monitoring System (D21), the analog Leeds & Northrup (L&N) model 524 recorder D21R600B will be replaced with a Westronics 3000 digital recorder and reidentified as D21R600. The recorders D21R600A,C will be deleted and their inputs will be added to D21R600.

REASON FOR CHANGE: The recorders D21R600A,B,(C) are obsolete L&N model 524 (512) recorders and spare parts are difficult to obtain.

SAFETY EVALUATION: In the event of an accident, the affected recorders can be used by the operators to determine the magnitude of the release of radioactive materials and for continuously assessing such releases. They are only required for Regulatory Guide 1.97, Type E, Category 3 indication. These recorders are non-safety related, non-seismic, non-seismic Category II/I and are all supplied non-1E UPS power.

The 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 recorders. This change will be an improvement in terms of reliability and monitoring capability. The changes will not compromise any existing safety related system, structure or component. The failure of the recorders and the circuits to which they are connected will not initiate any evaluated transient or accident. The D21 system operation and function will not change. No new interfaces are created. This change will therefore not introduce an unreviewed safety question. The recorders are not currently addressed in the technical specification and this change will not require that they be added to the technical specification.

Serial Number: 93-048-NPE Document Evaluated: DCP 89/0033-5

DESCRIPTION OF CHANGE: For the Offgas System (N64), the recorder N64R613 will be replaced with a Westronics model 3200 digital recorder. The recorder N64R602 will be deleted and the inputs will be added to N64R613. The high temperature setpoint of the former N64R602 inputs will be changed from 875°F to 830°F.

REASON FOR CHANGE: The recorders N64R602 and N64R613 are obsolete Westronics M11E analog models and spare parts are difficult to obtain. The high temperature setpoint of N64R602 is 875°F instead of 830°F as specified in the GE design specification data sheet.

SAFETY EVALUATION: The affected recorders are non-seismic, non-seismic category II/I and non-safety related. They are not required to perform any active or passive safety related functions. They are not connected to class IE power. They are not required for Regulatory Guide 1.97 indication. The 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 recorders. This change will be an improvement in terms of recorder reliability and monitoring capability.

The changes will not compromise any existing safety related system, structure or component. The failure of the recorders and the circuits to which they are connected will not initiate any evaluated transient or accident. The N64 system operation and function will not change. No interfaces with safety related or important to safety systems are created. This change will therefore not introduce an unreviewed safety question. The recorders are not required to mitigate the consequences of any evaluated transient or accident. The recorders are not currently addressed in the technical specification and this change will not require that they be added to the technical specification.

Serial Number: 93-049-NPE Document Evaluated: DCP 89/0033-1

DESCRIPTION OF CHANGE: For the Nuclear Boiler System (B21), the analog Westronics M11E recorder will be replaced with a Westronics Model 3200 digital recorder. An analog Westronics M5E recorder will be replaced with a Johnson-Yokogawa digital recorder. Another analog Westronics M5E recorder will be deleted and the inputs will be added to the Johnson-Yokogawa digital recorder.

REASON FOR CHANGE: The replaced recorders are obsolete Westronics analog models and spare parts are difficult to obtain. This change will be an improvement in terms of reliability and monitoring capability.

SAFETY EVALUATION: The affected recorders could alert the operators to a malfunction of safety related equipment. However, they are not required for Regulatory Guide 1.97 indication and no credit is taken in the UFSAR accident analysis for operator actions based on information taken from the recorders or their annunciators. These recorders are non-safety related, non-seismic and non-seismic Category II/I. They perform no active or passive safety related function.

The 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 recorders. The changes will not compromise any existing safety related system, structure or component. The failure of the recorders and the circuits to which they are associated will not initiate any evaluated transient or accident. The system operation and function will not change. The recorders are not required to mitigate the consequences of any evaluated transient or accident. No new interfaces (other than the addition of the non-Q Reactor Recirculation System thermocouples to the non-Q Johnson-Yokogawa recorder) are created. This change will therefore not introduce an unreviewed safety question. The recorders are not currently addressed in the technical specification and this change will not require that they be added to the technical specification.

Serial Number: 93-055-NPE Document Evaluated: DCP 92/0028-00 R00

DESCRIPTION OF CHANGE: This change replaces/reroutes heat damaged cables in the Feedwater Heater Room 'B'. These cables will be replaced by splicing and rerouting them around the Feedwater Heater Room.

REASON FOR CHANGE: Many cables have been identified to have brittle and/or cracked cable jackets caused by excessive heat in the immediate area of the cable trays which they are routed in. Therefore, a new routing path for these cables in Feedwater Heater Room 'B' must be made to ensure no recurrence of this heat-related problem occurs to ensure reliability of the affected systems.

SAFETY EVALUATION: The changes will not compromise any existing safety related system, structure or component, nor will they prevent safe reactor shutdown. The cables for the various systems involved have non-safety related equipment functions.

The new splice boxes, conduits and trays provide new and alternate routing paths for the involved cables to be rerouted around Feedwater Heater Room 'B'. Each cable will be spliced individually to ensure positive identification. Therefore, no new modes for failure that could be propagated into any existing system are introduced.

All cabling and raceway modifications to be performed will be in accordance with the separation requirements of Regulatory Guide 1.75.

No evaluated accident is affected by any addition of these splice boxes, conduits and trays. These components will not be required to mitigate the consequences of any evaluated accident. No components of any present system will be changed. No new interfaces are created and no new failure modes are introduced. These changes will, therefore, not introduce an unreviewed safety question.

Serial Number: 93-056-CORP Document Evaluated: NEAD-SE/001

DESCRIPTION OF CHANGE: Extend the applicability of a previous safety evaluation (NSSE-92/003) from a fuel assembly burnup of 38 GWD/MTU to 40 GWD/MTU. Safety Evaluation NSSE-92/003 was related to the abnormal growth characteristics identified at EOC-5 for the final batch of SNP 8x8 fuel where near-closure of the fuel rod to upper tie plate gap was noted.

REASON FOR CHANGE: It is projected that the peak assembly exposure for the SNP 8x8 fuel will be 39.6 GWD/MTU at EOC-6, at which time this fuel will be discharged from the core. Safety Evaluation NSSE-92/003 addressed burnups only to 38 GWD/MTU which was at the time of the evaluation the maximum predicted EOC-6 burnup for this fuel.

SAFETY EVALUATION: Safety Evaluation NSSE-92/003 conservatively assumed, based on the limited available data, that the abnormal growth would not be self-limiting but would continue at a constant rate throughout the life of the fuel. That evaluation concluded that all fuel design criteria would be met except for the rod-to-rod spacing (due to rod bow resulting from the closure of the gap with the upper tie plate) but that the decreased rod-to-rod spacing would not pose any safety issues given the low power the 8x8 fuel would operate in Cycle 6 for the then expected 38 GWD/MTU (after which these bundles would be discharged). Based on the recent fuel inspections at GGNS and other plants, it is now known that the cause of this condition is the binding between the fuel channel and the lower tie plate and that this binding will slowly be relieved by the thermal creep of the channel. Based on fuel rod growth measurements, axial rod loading tests, and a channel creep analysis, it was determined that all fuel design criteria will be met throughout the designed 40 GWD/MTU burnup.

No technical specification changes and no unreviewed safety question result from this change.

Serial Number: 93-057-PSE Document Evaluated: WO #99737

DESCRIPTION OF CHANGE: For the Seal Steam System, the change allows the lifting of leads to remove Level Switch 1N33N200 from operation.

REASON FOR CHANGE: The level switch cannot be isolated for repair or replacement due to leaking isolation valves.

SAFETY EVALUATION: Level Switch 1N33N200 is non-safety related, non-seismic Category I, non-ASME, and non-Class IE. Removal of the switch from the system does not inhibit the isolation function of the remaining level switches. Therefore, this work does not represent an unreviewed safety question.

Serial Number: 93-058-PSE Document Evaluated: Temp Alt 93/0C7

DESCRIPTION OF CHANGE: For the Seal Steam System, the change removes the control signal to Valve 1N33F505A by removing Card 316 of Controller 1N33R604, causing the valve to go full closed.

REASON FOR CHANGE: The change allows maintenance on Valve 1N33F505A to be performed. The hydraulic actuator of Valve 1N33F505A has developed internal control problems causing the valve to oscillate on a constant input signal for Controller 1N33R604. The oscillation of the valve is causing perturbations in the seal steam supply header.

SAFETY EVALUATION: The seal steam system is a non-safety related system. The failure of the seal steam system will not affect the operation of any safety related component and will not prevent safe reactor shutdown. With 1N33F505A out of service, Valve 1N33F505B can sufficiently control seal steam pressure to the limits as specified in Section 10.4.3.2 of the UFSAR. The changes made 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. The temporary alteration does not represent an unreviewed safety question.

Serial Number: 93-059-PSE   Document Evaluated: Temp Alt 93-009

DESCRIPTION OF CHANGE: The change removes the filter at the purge port of 1D17F002 of the Process Radiation Monitoring System. A tubing cap is installed in its place. The purge port of 1D17F002 is electrically opened to purge water and contamination from the radiation monitor and offgas pre-treatment sample panel.

REASON FOR CHANGE: Solenoid Valve 1D17-SV-F002 has failed. During normal operation of the Offgas Pre-Treatment Radiation Monitor, contaminated offgas air leaks from the valve's purge port. The valve is welded in place and it is in close proximity to feedwater controls. Replacement would require welding and, during plant operation, is not prudent.

SAFETY EVALUATION: The placement of the pipe cap on the purge port will not effect the operation of the radiation monitor because this path is only used to purge the radiation monitor sample chamber and sample extraction panel. During purge the radiation monitor is declared inoperable. The radiation monitor and sample panel can be purged in the normal manner after removal of the temporary tubing cap.

This change does not result in an unreviewed safety question or a change to the technical specifications.

Serial Number: 93-060-QUAL Document Evaluated: AECM 83/0431
Action Item 4

DESCRIPTION OF CHANGE: Delete the requirement to perform an annual audit regarding incorporation of technical specification amendments into surveillance procedures.

In August of 1983, Mississippi Power & Light (MP&L) responded to an NRC letter via AECM-83/0431 concerning inadequacies found in the Technical Specification Surveillance Program. Several action items were identified by the NRC. MP&L provided response of corrective actions for each item. One of those tasks was to perform a semi-annual audit of technical specification changes to assure that the changes are being incorporated into surveillance procedures. This frequency was later changed from semi-annual to annual.

REASON FOR CHANGE: A detailed review of the audit history revealed that no significant deficiencies have been identified in the last seven audits. The administrative program in place adequately controls the incorporation of changes.

SAFETY EVALUATION: There have not been any deficiencies identified in these audits where the procedures were not revised to incorporate the amendments. The program in place has been verified to be adequate to control this process. Deleting the requirement for an annual audit of this process will have no adverse impact on its effectiveness and will not result in any adverse consequences to the plant.

As a result of our review adequate controls have been verified for implementation of the process. Therefore, the requirement to audit this program at the required frequencies is no longer necessary.

Future audits or assessments of this program may be performed at the discretion of Management.

Serial Number: 93-061-HP Document Evaluated: UFSAR CR PLS-93-009

DESCRIPTION OF CHANGE: Section 12.5.3.6, Paragraph 2 is revised to change the processing frequency for thermo-luminescent devices (TLDs) from a monthly to a quarterly basis.

REASON FOR CHANGE: This change to quarterly processing will help standardize dosimetry processing procedures at system plants.

SAFETY EVALUATION: There are no regulatory requirements defining a wear period. This change will have no effect on the personnel monitoring in the Radiologically Controlled Area (RCA). This change to the GGNS UFSAR will not increase the probability of an accident or cause any malfunction of safety related equipment. Thermoluminescent dosimetry is a monitoring process only. TLDs are used as a quantitative measurement of personnel exposure to beta, gamma, and neutron radiation.

This change does not result in an unreviewed safety question or in a change to the technical specifications.

Serial Number: 93-062-NPE Document Evaluated: UFSAR CR NPE-92-026

DESCRIPTION OF CHANGE: MCP 92/1065, Revision 0, was issued for the Standby Gas Treatment System (SGTS) to: 1) provide a shaft seal design which was inherently leak tight to minimize/eliminate the possibility of unfiltered air being drawn into the fan housing from around the fan shaft and discharged to the environment, and 2) provide the work instructions required to assure that all potential leak paths on the suction side of the fan housing were made as leak tight as possible to prevent unfiltered air being drawn into the fan housing and discharged to the environment. The Design Basis Accident (DBA) LOCA offsite dose was recalculated using revised methodology and assumptions (reference Calculation XC-Q1111-92010, Rev. 0). To account for unidentified leakage which could potentially bypass the Standby Gas Treatment System (SGTS), a bypass leakage (i.e., a leak from the secondary containment directly to the environment) of 50 scfm was assumed in the new DBA LOCA offsite dose analysis. The bypass leakage was assumed to coincide with the initiation of the SGTS and continue for the duration of the accident. The incorporation of the assumed 50 scfm bypass leakage into the GGNS Unit 1 DBA LOCA analysis in conjunction with the modifications made to the SGTS by MCP 92/1065 conservatively envelope any leakage which could exist post LOCA. Section 6.5.3 of the UFSAR will be revised to include a discussion about the 50 scfm of bypass leakage which was assumed in the new DBA LOCA offsite dose analysis.

REASON FOR CHANGE: Section 6.5.3 of the UFSAR is being revised to include a discussion of the bypass leakage which has been incorporated into the DBA LOCA offsite dose analysis in order to delineate that the assumed bypass leakage does not apply exclusively to the SGTS effluent. This change is required to prevent future misunderstandings or misinterpretations of the bypass leakage assumed in the LOCA offsite dose analysis. This revision is considered to be an enhancement of the existing UFSAR text, not a change to the facility.

SAFETY EVALUATION: This revision of the existing UFSAR text is considered to be an editorial enhancement which will not require a change to the GGNS Unit 1 Technical Specifications, reduce the margin of safety as defined in the basis for any technical specification or increase the probability of occurrence or the consequences of an accident previously evaluated in the UFSAR. Implementation of the text change to include a discussion of the bypass leakage which has been incorporated into the DBA LOCA offsite dose analysis in order to delineate that the assumed bypass leakage does not apply exclusively to the SGTS will not increase the probability of occurrence or the consequences of a malfunction of equipment important to safety previously evaluated in the SAR, create the possibility for an accident or the malfunction of equipment important to safety of a different type than any previously evaluated in the UFSAR.

Serial Number: 93-063-CORP Document Evaluated: UFSAR CR PLS-93-008

DESCRIPTION OF CHANGE: UFSAR Section 10.2.5.3 is changed to state, "A fence is erected around the hydrogen bulk storage unit to further protect the storage area and markings/signs are posted in accordance with NFPA requirements". UFSAR Section 10.2.5.3 currently includes a general description of the hydrogen bulk storage unit and stated, "A fence is erected around the hydrogen bulk storage unit to further protect the storage area. 'No Smoking' signs and 'Danger Regulating Station' signs are posted in accordance with NFPA requirements."

REASON FOR CHANGE: UFSAR Section 10.2.5.3 is changed to more clearly reflect actual plant conditions. The markings/signs provided at the hydrogen bulk storage unit are clearly equivalent to the requirements of NFPA 50A and therefore adequate.

SAFETY EVALUATION: This change will not affect technical specifications or the Bases for any technical specifications because NFPA marking requirements associated with gaseous hydrogen system are not addressed by the technical specifications. 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 adequate markings are provided at the hydrogen storage tanks to alert personnel of the potential hydrogen gas hazards. This change will not increase the consequences of an accident or malfunction of equipment important to safety previously evaluated in the UFSAR because the hydrogen bulk storage unit is placarded consistent with the requirements specified in NFPA 50A. 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 new failure mechanisms associated with the hydrogen bulk storage unit have not been introduced and exposure severity is unaffected.

Serial Number: 93-064-PSE Document Evaluated: UFSAR CR PLS-93-007

DESCRIPTION OF CHANGE: This change deletes the requirements to perform Type C leak rate testing on numerous test connection, vent and drain valves 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).

REASON FOR CHANGE: The test connection, vent and drain 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, vent and drain pipes usually attach to their process pipes between inboard and outboard main isolation valves and, for test connections, have pipe caps in series on many of the penetrations, these penetrations present multiple independent barriers to leakage through the penetration.

Eliminating the Type C tests of these test connection, vent and drain 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. Local leak rate testing is not effective in detecting mispositioned valves, which is the only likely reason for increasing the probability of an accident or malfunction of equipment. The consequences of an accident or malfunction of equipment are minimized by the valve's construction, infrequent operation, and administrative controls on disk position.

This change does not result in an unreviewed safety question or a change to the technical specifications.

Serial Number: 93-065-NPE Document Evaluated: QDR 0125-93

DESCRIPTION OF CHANGE: QDR 0125-93 documented a deficiency in the development of the Cycle 6 MCPR operating limits. For statepoints between 70 and 100% rated power, Siemens Power Corporation's (SPC) analysis indicated that the Cycle 6 $MCPR_p$ operating limit would not protect the safety limit for the limiting pressurization events (Load Reject No Bypass (LRNB) and Feedwater Controller Failure (FWCF)) at cycle exposures after EOC-12 effective full power days (EFPD) in a particular area of the power/flow map with the conservative assumptions used in the original reload analysis. These assumptions include neglecting the MCPR margin gained in coastdown operation and requiring the core to be all-rods-out (ARO) at statepoints as early as EOC-30 EFPD. SPC's re-analysis has used current core follow data to conservatively predict the EOC conditions instead of using those assumptions in the original reload analysis. The results of this recent re-analysis concluded that the safety limit is protected by the current operating limits since control rod insertion was found to be necessary to operate in this region of the power/flow map after EOC-12 EFPD and prior to coastdown. In coastdown operation, SPC's analysis determined the additional MCPR margin achieved from coastdown would protect the safety limit at all statepoints. This re-analysis demonstrates that GGNS Cycle 6 will not operate in the area of concern on the power/flow map with the conservative assumptions used in the original reload analysis and that no administrative controls are necessary to protect the safety limit. On these bases, SPC concluded that the current MCPR operating limits were acceptable for the remainder of Cycle 6.

REASON FOR CHANGE: QDR 0125-93 documented a deficiency in the development of the Cycle 6 MCPR operating limits. As a result of this deficiency, SPC has re-analyzed the limiting pressurization events at off-rated conditions for Cycle 6 and the results are evaluated here.

SAFETY EVALUATION: SPC's analyses conclude that the resulting MCPRs from all off-rated LRNB and FWCF events initiated from actual operating conditions are bounded by the Cycle 6 MCPR operating limits at all exposures. Therefore, the current core operating limits are acceptable for the remainder of Cycle 6.

This evaluation does not result in an unreviewed safety question or a change to the technical specifications.

Serial Number: 93-066-PSE   Document Evaluated: Temp Alt 93/012

DESCRIPTION OF CHANGE: This change temporarily installs a 2" pancake downstream of the normally-closed Division 2 Diesel Generator Jacket Water Standpipe Drain Valve.

REASON FOR CHANGE: Division 2 jacket water is leaking past the seat of normally closed drain valve. No replacement valves are available on site.

SAFETY EVALUATION: The Jacket Water System provides the cooling medium for the engine, lubricating oil, cooler, and intercoolers. The Jacket Water System gets its supply from the jacket water standpipe. The temporary installation of this alteration does not degrade the B31.1 pipe code classification, safety class 3, and Seismic Category I classification of the jacket water piping.

This temporary change does not result in an unreviewed safety question or a change to the technical specifications.

Serial Number: 93-067-NPE Document Evaluated: Fuel Cycle 7 Receipt

DESCRIPTION OF CHANGE: This evaluation addresses the shipping, handling and storage of the reload fuel necessary for GGNS Cycle 7. The fuel is a split batch of 276 9x9-5 fuel bundles (3.20 w/o and 3.42 nominal enrichments) designed and fabricated by Siemens Power Corporation (SPC). The Cycle 7 reload fuel is similar to fuel previously loaded into Grand Gulf with minor differences in enrichment, number of bundles, and the presence of impure uranium. The evaluation considers the environmental effects of fuel transportation as new and spent fuel, movement and storage of the new fuel in either the new fuel vault or the spent fuel pool. Because of trace amounts of isotopic impurities in the fuel, the neutronic analysis and Health Physics procedures are also evaluated.

REASON FOR CHANGE: This evaluation addresses the shipping, handling and storage of reload fuel batch ANF-1.6 necessary for GGNS Cycle 7.

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

No unreviewed safety question results from these activities. Also, no technical specification changes are required.

Serial Number: 93-068-NSRA   Document Evaluated: 06-OP-T48-R-0002 & R-0003

DESCRIPTION OF CHANGE: The containment equipment hatch has been removed routinely as part of previous refueling outages. This effectively increases the volume subjected to drawdown by Standby Gas Treatment System (SGTS) in case SGTS safety function was actuated.

This evaluation provides the justification necessary to maintain the design basis assumption for the SGTS safety function of secondary containment drawdown. This evaluation also provides justification that primary containment integrity is not a prerequisite to determine SGTS performance for the drawdown criterion.

REASON FOR CHANGE: Performance of the secondary containment drawdown surveillance via SGTS is technically achievable <u>with</u> or <u>without</u> primary containment integrity. That is, SGTS drawdown tests can be performed either for only the secondary containment volume or for the primary containment and drywell volumes in addition to the secondary containment volume. This change will allow the performance of the secondary containment drawdown test with or without containment integrity established.

SAFETY EVALUATION: This safety evaluation considers the SGTS performance criterion for secondary containment drawdown time with and without the added volumes of primary containment and drywell. The volume of free air space in the primary containment and drywell is known and was used to calculate the increased time duration of SGTS operation to reduce pressure to -0.25" wg. The additional time required to evacuate to this design basis pressure is approximately 16 seconds with one SGTS fan in service. Therefore, the primary containment equipment hatchway can be open or closed while performing the secondary containment drawdown surveillance to verify SGTS safety function performance.

This evaluation does not involve an unreviewed safety question. In addition, a no significant hazards consideration results from this evaluation. Fuel handling accidents have been evaluated in the UFSAR for two different scenarios. One fuel handling accident scenario was postulated to occur inside primary containment and a second scenario was postulated to occur in the spent fuel pool area of the Auxiliary Building. The consequences of a postulated fuel handling accident inside primary containment without containment integrity established was previously determined to be bounded by calculated consequences of a fuel handling accident within the secondary containment.

Serial Number: 93-069-PSE Document Evaluated: WO #104361 & WO #104708

DESCRIPTION OF CHANGE: Temporary chillers and coolers are being utilized to provide temporary, additional cooling capacity to the Turbine Building 166' elevation for pre-outage work. This change provides temporary power to the two chiller skids and booster pump to be located outside the Water Treatment Building and temporary fill/makeup water for the chilled water loop to the coolers which will be located on the Turbine Building 166' elevation. The change also provides necessary scaffolding and supports for the temporary power cable and the temporary chilled water loop hoses.

REASON FOR CHANGE: Due to the extreme temperature on the Turbine Building 166' elevation, supplemental cooling capacity is needed to reduce heat stress for people engaged in pre-outage activities such as fuel receipt and preparations for turbine work.

SAFETY EVALUATION: The changes described herein will not compromise any existing safety related system, structure, or component. These changes will not prevent safe reactor shutdown or affect the ability to maintain the reactor in a safe shutdown condition.

Location of the temporary chillers was evaluated for potential hazard to Control Room personnel. The amount of Freon R22 and the promixity to the Control Room intake structure is bounded by the previous NPE analysis, which was used to evaluate freon storage in the GGNS Unit 1 warehouse. Therefore, no significant hazard to Control Room personnel is presented by the location of the temporary chillers and the freon contained in the system.

Section 3.6 of the FSAR discusses the effects of high and moderate energy pipe breaks on containment, essential systems, components, and equipment; and other essential structures. Although the temporary hoses meet the criteria for moderate energy piping, they are not routed in safety related areas or structures. Therefore, a hose burst will not introduce any unreviewed safety concern.

Section 10.4.5 of the FSAR 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: 93-070-NPE Document Evaluated: MNCR 0142-93

DESCRIPTION OF CHANGE: This change provides for an online leak repair, installation of a temporary pipe support for reactor feed pump (RFP) minimum flow line, and temporary isolation of the RFP minimum flow control valve via manual isolation of a maintenance block valve.

REASON FOR CHANGE: MNCR 0142-93 documents a leak through an 8" x 12" reducer connecting the RFP minimum flow control valve to the main condenser inlet stub tube/piping. Installation of a temporary pipe support for the minimum flow line and temporary isolation of the RFP minimum flow control valve is required to maintain condenser in-leakage within the capability of the condenser air removal system.

SAFETY EVALUATION: These temporary changes 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 unchanged. These changes will not increase the probability for occurrence of accidents or malfunction of equipment important to safety previously evaluated in the UFSAR because design requirements provided to ensure compliance with the technical specifications are unchanged. Further, replacing automatic minimum flow protection with temporary manual controls, installation of the temporary pipe support, and the temporary online leak repair will ensure that the reliability and operating characteristics of the existing equipment remain unchanged. These changes 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 and transient analyses. These changes 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. Credible event initiators remain bounded by existing initiators such that existing characteristic transient signatures remain unchanged. These changes will 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: 93-074 NSRA   Document Evaluated: 01-S-06-5, Rev. 23

DESCRIPTION OF CHANGE: Revise procedure to reflect changes to regulatory reporting requirements in accordance with 10CFR50.72 and 50.73; to reflect current regulatory reporting criterion for high groundwater level in accordance with GNRI-93/00025; incorporate process enhancements.

REASON FOR CHANGE: To accurately reflect regulatory reporting criteria and provide process enhancements.

SAFETY EVALUATION: This safety evaluation concluded that no unreviewed safety question exists as a result of this revision. The changes proposed will not inhibit the safe operation of GGNS.

Serial Number: 93-081-NPE Document Evaluated: DCP 91/0050-00

DESCRIPTION OF CHANGE: This change provides for the following modifications to the Drywell Valve Handling Crane: 1) addition of a 1-ton hoist to one end of the crane bridge (with attendant festoon system and controls); 2) addition of a motorized crane bridge drive with two speed capability; 3) addition of a new multi-function pendant, festoon system, and take-up reel; 4) relocation of the crane bridge light fixtures; and 5) replacement of the existing crane bridge carriers with new carriers that have flangeless wheels and side rollers.

REASON FOR CHANGE: The modifications to the drywell valve handling crane will provide a second hoist to facilitate MSRV replacement operations, motorized travel of the crane bridge, a single control pendant, and smoother travel of the crane bridge along the rails with less potential for binding. All these modifications will enhance the crane's reliability, capability and ease of use.

SAFETY EVALUATION: Apart from seismic integrity, the crane itself is not important to the safe operation of the plant or to mitigate accidents. The crane modifications are seismically analyzed to ensure that they will not collapse on adjacent safety-related equipment during a seismic event. In addition, new electrical components will be installed Seismic Category II/I.

During plant operation, the crane will be de-energized and locked in place in a designated location. As a result, there will be no change for an electrical malfunction to occur on the crane during plant operation.

At worst, a malfunction during crane operation (during Modes 4 and 5 only) would result in: 1) over-travel of the crane bridge, 1 ton or 5 ton hoists; 2) inadvertent travel of the crane bridge, 1 ton or 5 ton hoists, 3) loss of power to the crane travel motor or hoists; or 4) snagging of the hoists' or control pendant's festoon cabling on unforeseen obstructions. The impact of these malfunctions are minimized by the following considerations: 1) over-travel of the 1 and 5 ton hoists is prevented by internal hoist mounted limited switches; 2) the selected crane and hoist speeds are very slow and would result in minimal damage on inadvertent travel situations; 3) loss of crane/hoist power would result in stopping the crane and/or hoists, however, mechanical brakes internal to the hoists would prevent them from dropping their loads; and 4) the festoon cables are adequately protected with fuses to provide equipment and personnel protection if it should be damaged during crane/hoist travel.

The change does not result in an unreviewed safety question or a change to the technical specifications.

Serial Number: 93-083-NPE   Document Evaluated: MCP 93/1010 R00

DESCRIPTION OF CHANGE: In order to determine when Air Filter/Mist Eliminator NSM61-D005 and Aftercooler NSM61-D012 should be disassembled for inspection and cleaning, it is necessary to monitor the pressures and temperatures of the air and water passing through them. These components are part of the integrated leak rate test (ILRT) skid. The ILRT skid will be modified per MCP 93/1010, Revision 0 to install thermowells on the inlet and outlet ports/pipes of the aftercooler, and to install pressure points on the inlet and outlet pipes of the air filter/mist eliminator.

This design change consists of the following modifications:

1. Install one (1) thermowell each, in the PSW system's 3"-JBD-982 (inlet) pipe, and 2"-JBD-983 (outlet) pipe of aftercooler unit NSM61-D012.

2. Install one (1) thermowell each, in the air flow side of pipe 6"-HBD-1283 (inlet), and 8"-HBD-1284 (outlet) pipe of the aftercooler.

3. Install one (1) pressure gage connection each, in pipe 8"-HBD-1284 (inlet) side, and 8"-HBD-1285 (outlet) side of air filter NSM61-D005.

4. Install two (2) 3"-150# pipe flanges in bypass line 3"-HBD-1809, between valve N1M61-F042 and pipe 6"-HBD-1283. This modification will simplify removal and reinstallation of the aftercooler by allowing the aftercooler inlet pipe (6"-HBD-1283) to be removable, thus allowing the aftercooler to be moved from under an obstructing pipe support.

REASON FOR CHANGE: The purpose of this MCP is to provide a method to determine when Air Filter/Mist Eliminator NSM61-D005 and Aftercooler NSM61-D012 should be disassembled for inspection and cleaning. This design change is intended to enhance performance, and increase the reliability of the ILRT skid to facilitate minimum length refueling outages. Additionally, the modifications provided in MCP 93/1010, Revision 0 will provide the Maintenance Superintendent with sufficient information to determine when to perform the required Preventative Maintenance Instructions on the ILRT skid components specified above.

SAFETY EVALUATION: This station modification provides the design requirements necessary to modify the ILRT pressurization skid to provide performance indicators for the aftercooler and air filter-mist eliminator.

The addition of thermowells in the aftercooler piping, and pressure gages in the air filter piping will aid maintenance personnel in determining when these components should be inspected for cleaning.

The primary systems affected by MCP 93/1010, Revision 0 are (M61) Containment Leak Rate Test System,, and (P44) Plant Service Water System. The portion of the M61 system affected by this MCP is non-safety related. The Plant Service Water System (P44) has no safety design basis as defined in UFSAR Section 3.2. Modifications made to the containment leakage rate test skid per this MCP will not compromise any safety related systems or components, and will not prevent safe shutdown of the plant. The design modifications provided by MCP 93/1010, Revision 0 meet the applicable codes, material requirements, standards, and quality assurance requirements, therefore, implementation of the design will not result in increased dose rates, accidents, or equipment failures. In addition, the modifications described in MCP 93/1010, Revision 0 are consistent with the statements and descriptions in the UFSAR such that all design commitments are met, and overall system performance and reliability is maintained.

Serial Number: 93-084-NPE Document Evaluated: MNCR 91-0119-2nd

DESCRIPTION OF CHANGE: Incorporate software changes documented by as-built walkdown into plant communications drawings. These drawings are included in UFSAR Chapter 9 as Figures 9.5-9b thru 9.5-9i.

REASON FOR CHANGE: To resolve nonconformances between plant communications drawings and actual as-built field conditions.

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. The installation of this station along with the necessary cabling 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 odes not reduce any margin of safety described in the technical specification bases.

Serial Number: 93-085-PSE Document Evaluated: TSTI-1P41-92-005-0-S

DESCRIPTION OF CHANGE: GGNS has stated that the Fuel Pool Cooling System can be run in a one pump, two heat exchanger mode, with Standby Service Water (SSW) supplying cooling water. This test will prove that the required SSW flow (2508 gpm) can be obtained, which is double the normal flow for FPCC heat exchanger supply piping. All SSW A LOCA loads will be valved in plus both fuel pool heat exchangers. Throttle valve positions will be changed to obtain the required flows and maintain minimum flow to the Division I diesel and RHR heat exchanger. All throttle valve position changes will be recorded. The minimum flows to all components supplied by SSW A except the drywell purge compressor are expected to be met during this test. Component flows will be checked after each step that will affect system flow, any component flow not meeting minimum requirements will be declared inoperable and the appropriate LCOs entered. This test will require opening the breakers to the SSW A supply and return valves to the fuel pool heat exchangers which bypasses a MOV interlock that prevents SSW A and the Component Cooling Water (CCW) System from being cross connected. The CCW supply and return valves will be tagged closed and the breakers to these valves tagged opened to ensure SSW A and CCW remain isolated from each other. The fuel pool heat exchangers will still be able to be isolated from each other and SSW A anytime during this test. All valve logic for lining up the B fuel pool heat exchanger to SSW B will be in place and unaffected during this test, this logic also assures that SSW A and SSW B do not become cross-connected.

The FPCC system will be operated normally with one pump supplying one heat exchanger.

The breaker for the SSW A RECIRC LINE MOV will be opened with the valve in the closed position. The breaker for this valve will be opened to prevent it from opening. The required position for a LOCA lineup for the 1P41F006A is closed, therefore the valve will be in a conservative position.

REASON FOR CHANGE: This test will be performed to demonstrate that the Standby Service Water System is capable of providing a supply of cooling water (2508 gpm) sufficient to operate the Fuel Pool Cooling System in a one pump, two heat exchanger mode. All SSW "A" LOCA loads will be valved in during the test.

SAFETY EVALUATION: This test will not require a change to the technical specifications. The proper LCOs will be entered for inoperable equipment. The limits for fuel pool temperature will be adhered to. The SSW system will be operated in an abnormal manner, but within the system design flows and pressures.

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The test will not degrade the SSW system nor any other systems. The barriers between the SSW system and the CCW system will be maintained by tagging the CCW supply and return valves closed, and tagging their breakers open. In addition, process monitors are in place which will detect any radioactivity introduced into the system. The SSW system is designed to have two redundant trains, each capable of cooling the reactor down in case of an accident. The "B" train of SSW will remain operable during the test to provide cooling. The "A" train will be operable, but will have the added load of fuel pool cooling. In summary, equipment in this test will either be operated in its normal manner, or precautions will be taken to ensure that no portions of the test will reduce the protections into the systems. The precautions taken, coupled with the system monitoring associated with the test itself, demonstrate that the test will not constitute an unreviewed safety question.

Serial Number: 93-086-NPE   Document Evaluated: CN 93/0064 to DCP 87/0066 R0

DESCRIPTION OF CHANGE: The change replaces a previously specified flow indicator controller (1P33R110) with a presently mounted like item (1P33R044B) which is more conveniently located on the 1H22P121 panel for operator use. The change will not functionally alter the design of the affected system P33 chromatograph installation.

REASON FOR CHANGE: The change is being made to improve operator convenience. The flow indicator controller previously specified is remotely located on the panel with respect to the other like items used for the ion chromatograph installation.

SAFETY EVALUATION: The ion chromatograph installation will not perform any safety related function nor any function important to safety. The installation is in non-seismic areas and are not required to meet Seismic II/I design.

The use of a like instrument in a more convenient location on Panel 1H22 P121 will not degrade the performance of the ion chromatograph installation in any manner.

Serial Number: 93-087-PSE Document Evaluated: UFSAR CR PLS-93-011

DESCRIPTION OF CHANGE: This change deletes the requirements to perform Type C leak rate testing on forty-three (43) containment isolation valves in twenty-eight (28) 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: 93-088-NPE Document Evaluated: DCP 91/0077-2-R0, R1

DESCRIPTION OF CHANGE: DCP 91/0-0077-2 provides the design details for installation of Division I and Division II SRV group test switches. The function of these switches is to disable the auto function of the individual SRV control circuitry for the respective division when in the TEST position. When in the NORMAL position, the SRV group test switch enables the auto portion of the SRV control circuitry. The SRV group test switches will at no time inhibit either the ADS function or the manual function of the associated divisional SRVs. Control Room annunciation is provided in accordance with Regulatory Guide 1.47 UFSAR Section 7.3.1.1.1.4.12.10 will be revised to include the SRV group test switch.

In addition to the SRV group test switch installation, UFSAR Section 7.3.1.1.1.4.4 will be revised to correctly reflect the current as-built conditions of the individual SRV control switches associated with ADS as three position OFF-AUTO-OPEN. The mechanical safety/relief function of the SRVs cannot be inhibited by positioning the SRV control to OFF. Control Room annunciation is provided for any instance of a individual SRV control switch turned to the OFF position, in accordance with Regulatory Guide 1.47.

REASON FOR CHANGE: BYPL 91-0077-92, Item 3, identified a risk of overtravel when turning the individual SRV control switches from the OFF position to the AUTO position. In the event of overtravel, an SRV control switch could be taken into the OPEN position causing inadvertent operation of its SRV.

UFSAR Section 7.3.1.1.1.4.4 incorrectly addresses the individual SRV control switches associated with ADS as being two position OPEN-AUTO. The switches are actually three position OFF-AUTO-OPEN. UFSAR Section 7.3.1.1.1.4.4 will be revised to reflect the as-built plant condition.

SAFETY EVALUATION: The subject changes do not represent any changes in operation, function or ability to perform any functions as presently described in the UFSAR. The only changes to UFSAR information will be to include a description of the SRV group test switch in Section 7.3.1.1.1.4.12.0 and to correct Section 7.3.1.1.1.4.4. to reflect the individual SRV control switches as three position "OFF-AUTO-OPEN". The subject design change will reduce the possibility of an inadvertent SRV lift during the performance of surveillance procedures, maintenance, or other activities in which the SRV group test switches are used. The changes under consideration do not adversely affect any equipment 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. The subject changes do not introduce any unreviewed safety questions, or represent any changes to the GGNS Technical Specifications.

Serial Number: 93-089-NPE Document Evaluated: CN 93/0098 to DCP 89-4-1 R00

DESCRIPTION OF CHANGE: Design Change Notice (CN) 93/0098 provides instructions for removal of the 36" x 30" x14" plenum mounted on Computer Cabinet SC83-P800 and blanking-off two eight inch diameter supply ducts connected to the plenum. In addition, this CN requires that the affected portion of the Z51 system be rebalanced so that the 290 cfm air flow originally supplied to Computer Cabinet SC83-P800 will be distributed between two 24" x 24" supply registers located inside Computer Room OC604.

REASON FOR CHANGE: DCP 89/0034-1, Revision 0 provides the design for installation of intrusion detection monitoring equipment in Central Alarm Station (CAS), Secondary Alarm Station (SAS) and Temporary Alarm Station (TAS). This document includes instructions to remove Computer Cabinet SC83-P800 from Room OC604, however, no directions are provided for removal of the plenum attached to the cabinet, or modifying the Z51 ductwork after the plenum is detached.

The computer cabinet will be completely removed from Room OC604 and replaced with new panels in accordance with DCP 89/0034-1, Revision 0 There are no HVAC requirements for the replacement panels, therefore, the Z51 ductwork originally designed to supply cooling for the computer cabinet is no longer required. Design Change Notice 93/0098 provides design criteria for required HVAC modifications not included in the original design change.

SAFETY EVALUATION: Safety Evaluation CFR 89/0034-01-R00 is applicable to design modifications implemented per DCP 90/0034-1, Revision 0. Safety Evaluation 93-0103-R00 addresses only the HVAC modifications contained in CN 93/0098.

The system affected by CN 93/0098 is the Control Room HVAC System (Z51). The function of this system is to provide a suitable environment for safety-related equipment in the Control Room envelope. This function is safety-related. This CN disposition does not compromise the capability of the Z51 system to perform this function, nor does this CN jeopardize the Entergy Operations plant security plan.

The affected Z51 ductwork has been evaluated to ensure that its failure or physical collapse will not affect essential components. This evaluation is not affected by the duct modifications described in CN 93/0098. The modifications described in

this CN are consistent with the statements and descriptions in the UFSAR such that all design commitments are met, and overall system performance and reliability is assured.

These modifications comply with applicable codes, material requirements, standards, and quality assurance requirements, therefore, implementation of the changes will not result in increased dose rates, accidents, or equipment failures. The design change will not result in increased probability of occurrence or consequences of an accident or malfunction of equipment important to safety previously evaluated in the FSAR. Furthermore, the design will not create the possibility for an accident or malfunction of equipment of a different type than any previously evaluated in the FSAR and will not result in a reduction of the margin of safety as defined in the basis for any technical specification.

Serial Number: 93-091-NPE Document Evaluated: MCP 92/1133 R00

DESCRIPTION OF CHANGE: This change will provide pressure relieving/equalization capabilities for each RHR valve (F042A, F042B, F042C, and F005) back to its respective injection line.

REASON FOR CHANGE: MNCR 0270-92 identified a concern with the potential for these valves to fail to open on demand. NPE evaluated the situation and concluded that a pressure locking phenomena was possible for each valve. 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 of the respective low pressure injection lines.

SAFETY EVALUATION: The purpose of the installed pressure equalization line is to perform passive pressure equalization for the valve internals, with respect to the reactor, to prevent pressure locking. The primary containment integrity aspects of these valves remains unchanged by this modification. The upstream valve disc continues to perform the sealing function to isolate the RHR System in the case of excessive leakage. A leakage rate for each valve will be determined and compared to an acceptance limit. The combined leakage rate for all penetrations and all valves subject to Type B and C tests shall be less than or equal to 0.60 L_a. Testing the upstream 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 FSAR. Bypassing of a single disc (downstream/inboard) will have no effect on containment integrity and should not be construed as such. These valves serve as reactor coolant pressure boundary valves. Test methodology remains unchanged and the ability of these valves to isolate the high pressure piping from the low pressure piping is unaffected. This design change will have no effect on the operability of the gate valves or associated systems. The design has been evaluated against the applicable design criteria, installation, and operational requirements, and all necessary requirements and commitments are met.

Although these valves do not receive containment isolation signal, the ability of the valves to isolate as required by technical specifications is unaffected by this modification. This modification is intended to provide added assurance that the valve will open if required.

These valves do receive ECCS initiation signals to open; this modification is intended to provide added assurance that the valves will open if required.

Finally, integrity for containment spray boundary isolation remains unaffected. Valve testing has shown disc flexing to begin at approximately 300 psid and qualitative analysis predicts maximum expected pressure on the valves during containment spray as approximately 150 psig. Therefore, disc flexing will not introduce additional bypass flow, th.. nay tend to reduce available spray flow, during a containment spray event. Since all functions discussed in the technical specifications remains unchanged by this design, all margins of safety remain unaffected.

Serial Number: 93-092-NPE Document Evaluated: MCP 92/1116 R00

DESCRIPTION OF CHANGE: Due to the inability to drain the piping down stream of the Q1P11F424 and Q1P11F075 valves, a 1" drain line with associated valves will be installed per this MCP.

The 6"-HBD-337 line where the 1" drain line will be installed, is used by operations as an alternate fire protection for the containment during outages and requests that PM&C do not drain the line during installation of the 1" drain line. Therefore, a "Hot Tap" will be required during RF06 to install the drain line. The "Hot Tap" process requires the use of a gate valve. The new required gate valve will be installed upstream of the globe valve in the 1" HBD-1139 line.

REASON FOR CHANGE: To allow the local leak rate test crew (LLRT) a more practical means of draining the piping at Penetration 56.

SAFETY EVALUATION: Adding a 1" drain line with associated fitting and valves will not affect the Condensate & Refueling Water Storage & Transfer System (P11) 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 drainability. The piping has been designed to ANSI B31.1 code requirements. A calculation was performed to document the acceptability of the drain line to remain unsupported. Therefore, the piping will function in its intended manner. The piping affected by this MCP 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.

Valves in this system that form the containment isolation (Q1P11F004 and Q1P11F075) are addressed in the technical specifications (TS). TS 3/4.6.1 addresses the primary containment integrity aspects of the valves. The primary containment integrity remains unchanged by this modification. The valves will still perform their intended function to isolate the CRWST in case of excessive leakage. Adding a 1" drain line with valves downstream of these isolation valves will have no effect on primary containment integrity and is not a change to the technical specifications. TS 3/4.6.4 addresses the containment isolation aspects of the valves. The P11F075 valve is an automatic isolating valve with a maximum isolation time of 10 seconds, the P11F095 and P11F004 are manual isolation valves. The ability of the valves to isolate as required by the TS is not affected by this modification. This modification is to provide a practical means for the LLRT crew to drain water off the valves at Penetration 56 prior to testing. Therefore, based on these conclusions changes to the technical specifications are not required and will not create an unreviewed safety question.

Serial Number: 93-093-NPE Document Evaluated: MCP 93/1017 R00

DESCRIPTION OF CHANGE: This design will provide redundant Non-1E UPS power to the IRM/APRM neutron flux monitoring recorders. This design is in response to the NRC's Safety Evaluation Report performed on the BWR Owner's Group Topical Report NEDO-31558, issued January 13, 1993.

REASON FOR CHANGE: The Safety Evaluation Report performed on the BWR Owner's Group Topical Report NEDO-31558, issued January 13, 1993, recommended that each licensee perform a plant specific evaluation of the power distribution to the neutron flux monitoring instrumentation. This review should verify that, in addition to the events identified in NEDO-31558, a single power supply failure would not cause the loss of redundant channels of neutron flux monitoring instrumentation.

GGNS has reviewed the power distribution for neutron flux monitoring instrumentation and concludes that each division is powered from separate and reliable class 1E uninterruptable power supplies (UPS). Loss of a single UPS will not cause loss of redundant neutron flux monitoring instrumentation.

However, review of the power for the neutron flux monitoring recorders (C51-R603A-D) has identified that these recorders are all powered by the same non-class 1E UPS power supply.

SAFETY EVALUATION: The design will not create any unreviewed safety questions, impact the technical specification or reduce the margin of safety as defined in the basis for any technical specification. This change will only involve the addition and deletion of jumpers local to the subject recorders. The design will be implemented in accordance with Regulatory Guide 1.75 to insure that proper separation and isolation between Class 1E equipment and circuits is maintained with these Non-Class 1E recorders and their circuits.

Serial Number: 93-094-PSE Document Evaluated: WO #105629

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

This safety evaluation addresses the operability concerns associated with supplying temporary power from BOP Bus 111HD to loads normally supplied by Bus 15AA.

The additional power requirements being placed on Bus 11HD are negligible and no loading calculations are required. No components being supplied temporary power will be considered operable. In all cases, temporary power is being supplied as a matter of convenience and not plant safety. Required LCOs will be entered when normal power is removed.

REASON FOR CHANGE: The ESF 15AA Bus will be removed from [illegible] to perform maintenance and cleaning of the bus. Loads normally supplied by [illegible] bus will be temporarily powered by the 11HD bus. The loads will be considered inoperable, and appropriate LCOs entered when the normal power is removed. The temporary power is being furnished as a matter of convenience only.

SAFETY EVALUATION: As normal power is removed from the 15AA Bus loads, the equipment will be considered inoperable, and the appropriate LCOs will be entered. The temporary loads on the 11HD Bus are negligible and will not diminish the quality of power to its permanent loads. With redundant divisions operable, the 15AA Bus loads are not required to perform a safety function, and are not required to be sequenced back on line after a loss of offsite power. This test will not constitute an unreviewed safety question.

Serial Number: 93-095-NPE Document Evaluated: MNCR 93-0106

DESCRIPTION OF CHANGE: MNCR 93-106 documents that piping assumed to be insulated in HVAC calculations is uninsulated. The MNCR was dispositioned to "ACCEPT-AS-IS" a portion of the uninsulated piping and to "REPAIR" by insulating the remainder of the piping. This results in the heat load for the LPCS room cooler to increase above the value currently stated in UFSAR Table 9.4-7.

REASON FOR CHANGE: MNCR 93-106 documents that piping assumed to be insulated in HVAC calculations is uninsulated. During the resolution of the MNCR, the HVAC calculations were revised to reflect the conditions in the room after the repairs to the insulation are made. As a result of this, the load for the LPCS room cooler increased to a value above that shown in Table 9.4-7.

SAFETY EVALUATION: MNCR 93-106 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 a heat load for the LPCS room that is in excess of the load currently reflected in UFSAR Table 9.4-7.

However, this load is within the capacity of the LPCS room cooler. The loads for the remaining coolers RHR A, B, C and RCIC were unchanged (by insulating the pipe originally assumed to insulated) or within the loads currently shown in UFSAR Table 9.4-7.

Serial Number: 93-096-NPE   Document Evaluated: GIN-93/04065

DESCRIPTION OF CHANGE: This safety evaluation addresses the results of Engineering Report GGNS-93-0023, which examines the safety significance of specifying that the primary containment isolation function is not required for Valve Groups 5, 6A, and 8 during certain refueling related operations in Operational Conditions (Modes 4 and 5.

The proposed change will remove the requirement to maintain valve operability for automatic or manual primary containment isolation for Valve Groups 5, 6A, and 8 during the following conditions:

a. When performing core alterations during Mode 5;

b. When handling irradiated fuel in the primary or secondary containment during Mode 4 or 5; or

c. When performing operations with the potential for draining the vessel during Mode 4 or 5.

REASON FOR CHANGE: During refueling outages, Plant Operations must maintain operability of many containment and drywell isolation valves even though neither drywell or containment integrity is required. The removal of primary containment isolation operability requirements for drywell and containment isolation valves in Valve Groups 5, 6A, and 8 during refueling will result in substantial plant benefit. Some of the more significant benefits are:

a. Reduced exposure. Requiring one valve to be operable in each penetration results in Operations personnel visiting each valve location several times to complete a valve tagout and return to service procedure. Some of the subject valves are located in areas which result in significant unnecessary exposure due to the operability requirements and resulting tagout procedures.

b. Reduced man-hours. With the proposed changes, maintenance could be performed on the two isolation valves in a single penetration at the same time reducing the tim ent tagging valves and tracking valve operability.

c. A more manageable outage schedule. The proposed changes in valve operability requirements will produce additional flexibility in the outage schedule due to the ability to schedule valve maintenance within a broader outage window. This in turn could result in a shorter outage schedule or allow more scheduled time for plant modifications.

SAFETY EVALUATION: During cold shutdown and refueling, drywell and containment integrity is not required to maintain the plant within the envelope of conditions considered by the plant safety analysis. Likewise, primary containment isolation operability of drywell and containment isolation valves in Valve Groups 5, 6A, and 8 is not required to maintain the plant within the envelope of conditions considered by the plant safety analyses.

The proposed changes do not require any changes to the plant technical specifications, nor do they involve a physical change to any safety related system or component. The primary containment isolation function will continue to be provided where required by analyses. The proposed changes only administratively modify the listing of components necessary to provide the isolation function. Removal of containment isolation operability requirements for Valve Groups 5, 6A, and 8 during the specified operational states does not change other safety functions, system operability requirements, or interlocks required by system design or technical specifications for these valves. Any physical modifications to the valves or their controls will be conducted under approved plant procedures.

The only previously analyzed accident with offsite dose consequences which could occur during refueling or cold shutdown is the fuel handling accident. The offsite dose consequences of this accident are unchanged and the release pathway for radionuclides released from fuel damaged during this accident is not modified by the proposed change. Likewise, the function and operation of the safety related components and equipment provided to mitigate the offsite dose consequences of this accident are unchanged by the proposed change. Therefore, the proposed change does not involve an unreviewed safety question.

Serial Number: 93-097-NPE Document Evaluated: GIN 93/04078

DESCRIPTION OF CHANGE: During the sixth refueling outage (RF06) at Grand Gulf, 276 depleted SPC 8x8 and 9x9-5 fuel assemblies will be replaced with fresh unirradiated SPC 9x9-5 fuel assemblies. These new assemblies have been designed and built specifically for the Grand Gulf Cycle 7 core and are currently stored in the Auxiliary Building. The following items are evaluated in this safety evaluation:

a. Movement of fresh and irradiated fuel in the containment and the Auxiliary Building,

b. Storage of fresh and irradiated fuel in the upper containment pool during RF06,

c. Storage of fresh and irradiated fuel in the spent fuel pool during RF06 and Cycle 7, and

d. Shutdown margin for all interim RF06 core configurations and the final Cycle 7 core loading in Modes 4 and 5.

REASON FOR CHANGE: Cycle 7 operation requires the addition of fresh fuel assemblies and the removal of depleted assemblies from the reactor vessel. This safety evaluation assesses these actions. Actual Cycle 7 operation with these fuel assemblies will be assessed in an upcoming safety evaluation.

SAFETY EVALUATION: This evaluation concludes that (1) a fuel handling accident during RF06 in either the Containment or the Auxiliary Building will not result in doses above the allowable limits, (ii) the criticality criteria for the upper containment pool and the spent fuel pool are satisfied through Cycle 7, and (iii) the criterion for shutdown margin is satisfied for interim RF06 and final Cycle 7 core configurations during Modes 4 and 5.

Serial Number: 93-098-NSRA Document Evaluated: 05-1-02-1-4 Temp 1

DESCRIPTION OF CHANGE: Provides instruction for restoring power for selected ECCS systems (refueling outages only) during a station blackout (SBO) when the Division III diesel is not available. This is done by the connection and operation of a temporary emergency power supply. The point in time that this directive will be used is beyond the design bases as described in the UFSAR.

REASON FOR CHANGE: Provide instructions to connect a temporary power supply (diesel generator or gas powered turbine generator) to ESF-12 during refueling outages to ensure power available for selected ECCS systems when a station blackout condition occurs and Division III diesel generator is not available. This provides additional assurance that the equipment will be able to fulfill intended functions.

SAFETY EVALUATION: The point in time that this directive will be used is beyond the design bases as described in the UFSAR. All actions implemented by this procedure are done only if actions assumed in the design bases have failed to mitigate the consequences of the event. This directive will not affect the GGNS ability to meet the coping requirements as required by the Station Blackout Rule.

Serial Number: 93-103-NPE Document Evaluated: MCP 92/1115 R00

DESCRIPTION OF CHANGE: Due to the inability to drain the piping down stream of the Q1G33F040 and Q1G33F039 valves, a 1" drain line with associated valves will be installed per this MCP.

REASON FOR CHANGE: To allow the local leak rate test crew (LLRT) a more practical means of draining the piping down stream of the valves at Penetration 83.

SAFETY EVALUATION: Adding a 1" drain line with associated fitting and valves will not affect the Reactor Water Cleanup System (G33) function, operation, or performance in any way. The effect of the new piping on the system will have a significant improvement in the ability to drain the line. The new piping has been designed to ASME Section III code requirements and has been analyzed for seismic loads and has been found acceptable. Therefore, the piping will function in its intended manner. The drain line was added downstream of the section of the piping that forms the containment isolation, therefore the piping was designed to ASME Section III code requirement.

This 1" drain line was evaluated for postulated missile consequences. 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 (Reference 3.5.1.1.2.c). The 1" drain line will be supported because of its potential to become a missile. Therefore, the weld and support will be considered as two failures before the 1" drain line could become a postulated missile.

Portions of this system are addressed in the Technical Specifications (TS), Q1G33F039 and Q1G33F040 are primary containment isolation valves. TS 3/4.6.4 specifically list these valves. TS 3/4.6.1 addresses the primary containment integrity aspects of the valves. The primary containment integrity remains unchanged by this modification. The valves will still perform their intended function to isolate the RWCU in case of excessive leakage. Adding a 1" drain line with valves downstream of these isolation valves will have no effect on primary containment integrity and is not a change to the technical specifications. TS 3/4.6.4 addresses the containment isolation aspects of the valves. The two valves in question are automatic isolating valves with a maximum isolation time of 35 seconds. The ability of the valves to isolate as required by the TS is not affected by this modification. This modification is to provide a practical means for the LLRT crew to drain water off the valves at Penetration 83 during testing. Therefore based on these conclusions changes to the technical specifications are not required and will not create an unreviewed safety question.

Serial Number: 93-104-NSRA Document Evaluated: GNRI-92/00241

DESCRIPTION OF CHANGE: The commitment contained in GNRO-92/00117 (PCOL 91/17) and reiterated in GNRI-92/00241 (Amendment 102 to GGNS operating license) states "lists removed from the TS will be incorporated into plant procedures subject to the administrative controls of TS 6.8 and 6.5.3". These lists are currently contained in pendin. UFSAR changes and in the Technical Requirement Manual (TRM). The TRM is Attachment 2 of Administrative Procedure 01-S-15-9 and is also distributed as part of the Operating License Manual (OLM). All other TS amendments involving relocated TS received through Amendment 108 have not included this requirement.

This change will enable removal of the TRM from 01-S-15-9. The TRM will continue to be controlled by 01-S-15-9 and issued as part of the OLM. The TRM need not be part of 01-S-15-9 to satisfy the intent of NRC commitments related to controlling those items removed from the TS provided equivalent controls are in place. The TRM information is contained in other controlled documents such as the FSAR and the controls on those documents in conjunction with the implementation process for changes to the Operating License Manual (OLM) contain the necessary procedural control elements to adequately manage the TRM's contents. 01-S-15-9 which controls contents of the TRM will be retained as an administrative procedure subject to the administrative controls of TS 6.8 and 6.5.3.

REASON FOR CHANGE: The TRM contains information extracted from various controlled documents such as the UFSAR. The TRM is maintained as part of the Operating License Manual. Controlling the TRM contents under the administrative procedure program (i.e., with the TRM part of 01-S-15-9) creates a redundant and overly burdensome control mechanism as TRM information can appear in the OLM, UFSAR and 01-s-15-9. Because all other TS amendments involving relocated TS received through Amendment 108 have not included this requirement, the proposed changes will also provide consistency with these other amendments. In addition, the current administrative procedure change process is overly cumbersome and time consuming with little or no value added for the type of information contained in the TRM (e.g., valve lists).

SAFETY EVALUATION: This evaluation addresses modifying a commitment contained in GNRO-92/00117 (PCOL 91/17) and reiterated in GNRI-92/00241 (Amendment 102 to GGNS operating license). This commitment states "lists removed from the TS will be incorporated into plant procedures subject to the administrative controls of TS 6.8 and 6.5.3". These lists are currently contained in

pending UFSAR changes and in the Technical Requirements Manual (TRM). The TRM is Attachment 2 of Administrative Procedure 01-S-15-9 and is also distributed as part of the Operating License Manual (OLM). The TRM need not be part of 01-S-15-9 to satisfy the intent of NRC commitments related to controlling those items removed from the TS provided equivalent controls are in place. The TRM information is contained in other controlled documents such as the FSAR and the controls on those documents in conjunction with the implementation process for changes to the Operating License Manual (OLM) contain the necessary procedural control elements to adequately manage the TRM's contents. 01-S-15-9 which controls contents of the TRM will be retained as an administrative procedure subject to the administrative controls of TS 6.8 and 6.5.3.

Modification of this commitment does not require any changes to the plant technical specifications, nor does it involve a physical change to any safety related system or component. The change only modifies requirements for administratively controlling lists removed from the TS. Adequate controls exist outside of the administrative controls imposed in the commitment. Specific changes to the relocated lists will be conducted under approved plant procedures. The proposed changes do not affect the probability of occurrence of any accident or equipment malfunction. The consequences of previously analyzed accidents or equipment malfunctions are not increased, nor is any safety margin decreased. Thus, the proposed change does not involve an unreviewed safety question.

Serial Number: 93-106-NSRA Document Evaluated: ONEP 05-1-02-III-1

DESCRIPTION OF CHANGE: The proposed change to 05-1-02-III-1 introduces an additional method of decay heat removal. This method employs an ECCS System (taking suction from the suppression pool (SP)) to provide cooling flow to the vessel for decay heat removal. The SP Makeup System will provide a means to provide a recirculation flow path to the suction source of the ECCS pump. This change is being made in order to increase flexibility and redundancy.

REASON FOR CHANGE: The change provides an additional method to provide adequate heat removal capability following a loss of all desired sources of decay heat removal. This method would be employed prior to use of the undesired fire water injection method.

SAFETY EVALUATION: This method does not require a change in the GGNS TS nor does it pose an unreviewed safety question. The proposed change to the Off Normal Event procedure governing inadequate decay heat removal provides an redundant emergency method of heat removal in the core in the event of the loss of SDC and other alternate methods of decay heat removal. The method would only be used if the main steam line plugs are installed and for some reason cannot be removed. This method would be used prior to injecting fire water into the vessel for cooling. The method does cause a change in radiological condition inside the Containment and Auxiliary Building. Plant personnel will be evacuated prior to the operation of this cooling method until an evaluation of the radiological condition is performed.

Serial Number: 93-107-NPE Document Evaluated: MNCR 93-0022

DESCRIPTION OF CHANGE: This evaluation addresses the disposition provided for MNCR 0022-93, generated to document an overthrust condition on a number of valves. NPE Design Engineering reviewed and addressed the overthrust condition for each valve. NPE reviewed the seismic calculation for the HPCS Injection Valve and concluded that, based on worst case design conditions, the temperature at which the material properties for the valve components were taken (i.e., 575°F) was significantly above the actual temperatures which the valve will see. NPE evaluated the temperature of the process fluid in the HPCS System as well as the temperature in the area where the valve is installed for both normal operating and post accident conditions, and concluded that the post accident conditions would be the more limiting of the two. The post-LOCA ambient temperature in the area remains below 220°F, except for one 25 second period in which the temperature ramps up to approximately 315°F and then back down to less than 220°F. This temperature excursion does not last long enough to significantly change the temperature of the valve due to its large mass. It would take much longer than 25 seconds to increase the temperature of a valve with a mass of 2,500 lbs from 220°F to 240°F with a differential temperature of only 95° F. The temperature of the fluid being processed through the valve has a post-LOCA maximum design temperature of 185°F and would therefore act as a heat sink. Based on these facts, NPE concluded that a post-LOCA design temperature of 240°F would be conservative for the worst-case accident scenario. Our calculation was revised using a temperature of 240°F to select the ASME code stress allowable limits for the yoke flange and disk. In addition, the factor-of-safety used when evaluating the stem for buckling was reduced from 4 to 2.5. Section III of the ASME code does not address stem buckling. The NSSS vendor for Grand Gulf arbitrarily chose a factor-of-safety of 4, which is extremely conservative, in the stem buckling evaluation for this valve. Using a factor-of-safety of 2 for design work is considered sound engineering practice, therefore, using a factor-of-safety of 2.5 in the stem buckling analysis for this valve is adequate and will in no way jeopardize the structural integrity or function capability of the valve. By using a temperature of 240°F to obtain the allowable stress limits for the yoke flange and disk, and by using a factor-of-safety of 2.5 in the stem buckling evaluation, NPE increased the maximum allowable stem thrust (MAST) for the valve from 43,354 lbf to 48,405 lbf in the OPEN direction and 77,258 lbf to 90,000 lbf in the CLOSE direction. Since the corrected total CLOSE thrust is less than the revised CLOSE MAST, NPE concluded that the current torque switch setting is acceptable.

REASON FOR CHANGE: MNCR 0022-93 was generated to document an overthrust condition on a number of valves as a result of a 10CFR Part 21 notification issued by Liberty Technologies on their VOTES MOV diagnostic test equipment. The VOTES diagnostic test equipment was used to set/verify the torque switch settings of the valves identified in MNCR 0022-93. NPE reviewed and addressed the overthrust condition for each valve on an individual basis. This evaluation addresses the "ACCEPT-AS-IS" disposition provided for the HPCS injection valve. NPE increased the OPEN and CLOSE MAST values for the valves by using a temperature of 240°F instead of 575°F to obtain the allowable stress limits for the yoke flange and disk and using a factor-of-safety of 2.5 instead of 4 in the stem buckling evaluation.

SAFETY EVALUATION: NPE has evaluated the reduction in the design temperature for the yoke flange and disk, the reduction of the factor-of-safety used in the stem buckling evaluation and the resulting increase in the MAST for the HPCS injection valve and determined that these changes will in no way jeopardize the structural integrity, functional capability or adversely impact the ability of the valve to perform its design safety functions. Therefore, NPE concluded that the specified changes in the seismic analysis for the valve will not create an unreviewed safety question.

Serial Number: 93-108-NPE   Document Evaluated: ER GGNS-93-0024

DESCRIPTION OF CHANGE: Engineering Report GGNS-93-0024 provides a summary of the methodology and acceptance criteria used to evaluate the temporary installation of lead shielding on safety-related piping systems during Reactor Modes 4 and 5. The methodology and acceptance criteria also addresses the temporary removal of snubbers for functional testing while the lead shielding is installed. It is provided for piping components, pipe support and building structures components, equipment nozzles, valves and containment penetrations.

The methodology and acceptance criteria provided in the engineering report may be used in response to a Temporary Shielding Request (TSR), which provides the specific change to the safety-related piping systems. As such, the engineering report does not, in and by itself, constitute a change, test or experiment. It defines alternate methodology and additional acceptance criteria for the temporary installation of the lead shielding.

REASON FOR CHANGE: Lead shielding is temporarily installed to reduce personnel exposure during the performance of various maintenance, repair or replacement activities conducted during refueling outages. The engineering report provides the methodology and acceptance criteria to be used in evaluating this temporary change.

SAFETY EVALUATION: Evaluations performed in accordance with this engineering report assure that the safety-related piping systems are able to perform their intended function, as described in UFSAR 3.9.3, for all applicable design basis events with lead shielding temporarily installed.

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

Serial Number: 93-109-NPE Document Evaluated: TSR 93-002 & 93-003

DESCRIPTION OF CHANGE: The change consists of the temporary addition of lead shielding blankets to certain portions of the Reactor Recirculation (RR) and Residual Heat Removal (RHR) piping for the purpose of reducing personnel exposure during refueling maintenance activities. Lines to be shielded in the drywell include the jet pump risers and both recirculation pump suction/discharge lines. In the Auxiliary Building, miscellaneous RHR piping to be shielded (each division) includes the pump suction from the suppression pool, shutdown cooing suction cross-tie, and pump discharge to the RHR heat exchangers/bypass line. Specific activities include the design of separate shielding support scaffolding for the vertical sections of recirculation piping, piping reanalysis, pipe support reanalysis, and associated equipment/penetration nozzle loading reanalysis.

The temporary shielding and associated supports will only be installed during reactor Modes 4 and 5 and removed prior to plant startup.

REASON FOR CHANGE: Temporary Shielding Requests 93-002 and 93-003 estimated personnel exposure savings of 24.7 and 100.7 person rem,. respectively.

SAFETY EVALUATION: Existing analyses of the subject RR and RHR piping systems (i.e., pipe and piping components) were amended in accordance with existing specifications and standards (Reference M-220.0 & MS-44) to account for the shielding. These analyses also considered the removal snubbers to permit snubber functional testing during the period when the lead shielding is to be installed. All pipe stresses were shown to be within ASME Code (Reference 15) allowable stress limits. Pipe supports, building structure components (i.e., structural steel and containment penetrations), and equipment nozzles affected by the addition of the shielding were requalified for the new reaction loads obtained from the piping analysis.

Acceptance criteria for the piping, pipe supports, building structures, and equipment were defined in Engineering Report GGNS-93-0024. The results of the amended analyses indicate that when considering any normal, upset and faulted design condition event; the RR and RHR Systems remain capable of performing their respective safety functions as required for reactor Mode 4/5 events (UFSAR 3.9.3.1.1.1.6). Since all acceptance criteria are met, RR and RHR Systems operability for reactor Modes 4/5 is not affected by the addition of the shielding.

Guidelines regarding the limitations on the quantity and combinations of snubbers that can be removed concurrently have been clearly identified. The temporary shielding (up to the amounts assumed in the analyses) and associated supports may only be installed during reactor Modes 4/5, and must be removed prior to plant startup.

The scaffolding structures were analyzed for normal and seismic loading (i.e., Seismic Category II/I). Acceptance criteria was defined in the Civil Design Criteria Manual. The results of the analysis indicate that the scaffolding structure is capable of withstanding a seismic event.

Serial Number: 93-110-NPE Document Evaluated: DCP 93/00111-01-00

DESCRIPTION OF CHANGE: UFSAR Section 7.7.1.1.3.1.2 states that the water level measurement design for the narrow-range, wide-range and fuel zone instruments utilize differential pressure measurements based on a common condensate chamber type reference leg. Each of the various differential pressure instruments connected is calibrated for a specific reactor/containment pressure and temperature condition.

This DCP adds a purge function to the reactor vessel water level reference leg sensing line. During Reactor Modes 1-3, the CRD System will provide a source of high pressure water, that is not laden with non-condensible gases, to continuously purge (≈4 lbm/hr) the Reactor Vessel Level Indication System (RVLIS) reference legs of reactor condensate.

REASON FOR CHANGE: NRC Bulletin 93-03, "Resolution of Issues Related to Reactor Vessel Water Level Instrumentation in BWRs", notes that during power operation, the condensate used to fill the reactor water level instrumentation reference legs will contain dissolved non-condensible gases. Reference leg leakage, convective mixing, and diffusion will eventually distribute these gases throughout the reference leg. When these gases come out of solution, they may affect the accuracy of connected essential level instrumentation.

The NRC Staff has concluded that additional compensatory measures are needed to assure reliable reactor vessel level indication for all conditions. Therefore, each licensee has been required to make hardware modifications necessary to ensure the level instrumentation system design is of high functional reliability for long-term operation. This design change installs an interim hardware fix known to be acceptable to the NRC, until a satisfactory permanent fix can be designed, installed, and licensed.

SAFETY EVALUATION: The purge function being added by DCP 93/0011-01 is not safety related, but it does add a new safety/non-safety interface for each of the subject reference legs. Therefore, safety/non-safety interactions were evaluated. The design addresses the impacts on the following safety features described in the UFSAR.

UFSAR Section 3.2, "Classification of Structures, Components, and Systems": The purge function and all of the components/supports used to implement it have been classified in accordance with UFSAR Section 3.2. The design is consistent with criteria associated with those classifications (e.g., missile protection, protection against the dynamic effects associated with the postulated rupture of piping, seismic, and environmental).

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UFSAR Section 3.6, "Protection Against the Dynamic Effects Associated with the Postulated Rupture of Piping": Implementation of this design will not create a new source of high energy piping.

UFSAR Section 3.11, Environmental Design of Safety Related Mechanical and Electrical Components": The addition of the reference leg purge function as designed will not impact the environmental conditions to which existing equipment is qualified.

UFSAR Section 4.6, "Functional Design of Reactivity Control Systems": The additional demand imposed on the CRD Hydraulic System by the reference leg purge function will not degrade CRD performance.

UFSAR Section 7.2, "Reactor Trip System (Reactor Protection System) - Instrumentation and Controls": Implementation of this design will not degrade existing separation and redundancy requirements.

UFSAR Section 7.3, "Engineered Safety Features": Implementation of this design will not degrade existing separation and redundancy requirements.

UFSAR Section 7.5, "Safety Related Display Information": Instrument accuracies are programmatically controlled at GGNS.

The BWROG is addressing the issue of limiting conditions of operation regarding the purge support function. The recommendations are: 1) a single purge line may be out-of-service for up to 30 days, and/or 2) multiple purge lines may be out-of-service for up to 7 days, before the operability of the associated level instruments is impacted. The basis for these limits are presented in Sections 4.1 and 4.2 of Reference 11.

In conclusion, credible failures cannot affect existing reactor and protection system functions directly or consequentially. The design of existing equipment important to safety has not been altered. (Note: The thermal analysis only addressed thermal stresses, including cyclic fatigue, for a duration of two fuel cycles.) The effects on equipment for the failure modes associated with the purge function are enveloped by those of the existing reactor water level instrumentation design. Substantial objective evidence has been identified (e.g., BWROG efforts) which indicate that existing technical specification limits regarding reactor water level trip functions will not have to be altered. However, final determination is contingent upon the preoperational test results obtained. The results must be reviewed by NPE prior to final acceptance of this modification.

Serial Number: 93-111-NPE Document Evaluated: MNCR 92-0066

DESCRIPTION OF CHANGE: The approval of MNCR 0066-92 will allow standard 1/2 inch shackles, which are rated at 2 tons and conform to the plant staff rigging procedure, to be added to the bottom of each of the eight slings which connect the extension frame to the nut tray of the head strongback/carousel.

REASON FOR CHANGE: The hole in the nut tray lugs on the head strongback/carousel are not large enough to accommodate the hook on the end of the sling which connects to the extension frame. The addition of a shackle to the bottom of each sling will allow the lug and hook to connect.

SAFETY EVALUATION: This change allows the use of shackles on the bottom of the slings which connect the extension frame on the head strongback/carousel to the nut tray. The shackles have a higher allowable rating than the maximum load applied to the slings, will be inspected periodically per plant procedures and will not reduce the redundancy of the system. Therefore, this change will not increase the probability of the occurrence or the consequences of an accident or malfunction of equipment important to safety previously evaluated in the safety analysis report. Since the structural integrity and redundancy of the system is maintained and the change will not increase the amount of the design load that will be lifted, alter the load path, create any new interferences, or change the function or operation of the head strongback/carousel, the possibility of the occurrence of malfunction of a different type than any evaluated previously in the safety analysis report is not created. This change does not reduce the margin of safety as defined in the basis for any technical specification because the structural integrity and redundancy of the system is maintained and the change will not increase the amount of the design load that will be lifted, alter the load path, create any new interferences, or change the function or operation of the head strongback/carousel.

Serial Number: 93-112-NPE Document Evaluated: DCP 93/0011-00-R01

DESCRIPTION OF CHANGE: The reactor vessel fuel zone differential pressure Transmitters C and D reference leg connections are being changed. The reference leg connections will be connected to the variable leg taps used by the narrow range transmitters. The span of the fuel zone recorder will be increased from 200 to 300 "wc.

REASON FOR CHANGE: The RPV level transmitter condensing pots are susceptible to non-condensible gasses being dissolved into the reference leg liquid. Upon rapid depressurization the non-condensible gasses may come out of solution at a rate high enough to cause incorrect level readings. The liquid level of the reactor is required to be reliably monitored after a rapid depressurization. The variable leg of the narrow range RPV level transmitter is not as susceptible to concentrated dissolved gasses to the extent that the reference leg is when utilizing the condensing chamber.

SAFETY EVALUATION: The RPV fuel zone instrumentation that is affected does not provide any automatic safety feature actuation. The instrumentation is used for post accident RPV liquid level monitoring. The new configuration for the instrumentation will permit more reliable indication during normal and emergency operation of the reactor at GGNS and will not reduce the safe operation of the plant.

The activity retubes the two fuel zone RPV level transmitter reference legs to eliminate the reference leg instrument lines that utilize condensing pots. The new reference leg is the variable leg for other level transmitters on the same instrument rack. The only mechanism by which concentrated dissolved gases could get into the new reference leg would be through the equalizing valves for these level transmitters which utilize condensing chambers. This change requires testing of the equalizing valves to ensure that this will not happen. Although the possibility of degassing in this new reference leg exists, it is much lower than for transmitters using condensing chambers because the mechanism of getting gases into the new fuel zone reference leg is reduced.

The reference legs utilizing the condensing pots are considered cold in relation to the RPV. The portion of the new fuel zone reference leg between the vessel and the drywell side of the shield wall can be influenced by the heat of the vessel. This is assumed to have an acceptable affect on the accuracy of the new fuel zone configuration.

Other instruments utilizing the tubing runs affected by this design change have automatic safety actuation features. The configuration for instrumentation with automatic safety actuation features will remain the same.

The new tubing configuration will, however, change the plant response to a single human error: the situation whereby the equalizing valve for the fuel zone transmitter is opened. The lower tap of the fuel zone transmitter is connected to the discharge nozzle of the jet pumps, therefore the pressure in the instrument line will always be higher with the jet pumps running than when not running.

For the existing design, when this valve is opened and the jet pumps are running the higher pressure is put on the reference leg side of the transmitter, causing them to indicate a lower level than actual. If the jet pumps are not running, the reference leg will be drained and level indication will go high.

With the modified design, when this equalizing valve is opened a different set of instruments is affected and the higher pressure is now put on the variable leg side of these instruments, making them read higher than actual. Since the exact pressure induced into the instrument lines due to jet pump operation is unknown, two bounding possibilities will be evaluated for the purposes of this safety evaluation. The first possibility is that jet pump pressure is sufficiently high enough to get a level 8 trip; the second is that the pressure is low enough to go undetected.

If the pressure is low enough to go undetected, there will be no adverse affects on the other instrumentation. If an accident occurs in this situation and water level drops below the upper tap of the fuel zone transmitter, this instrument line will begin to drain down through the equalizing valve out the lower tap of the fuel zone transmitter. In this situation there will be no consequences with respect to the other transmitters, because this level will be below their calibrated range. The fuel zone transmitter affected will indicate its highest level reading at all times and will not change even if water level is low. This will be the only instrument affected and will be identifiable as an inoperable instrument by comparison with the other level indications. Also, as a result of this same human error, the existing configuration, which has been analyzed, will result in the same failure for the fuel zone indication as with the new configuration.

If jet pump pressure is high enough to get level 8 trips none of the associated events alone will result in a plant transient.

Therefore, this modification will not create any new type failure mechanism or increase the probability of occurrence for any existing failure mechanism for those instruments with automatic safety actuation features or those safety related instruments required to provide post accident level indication.

The tubing change made by this modification does not cross divisional boundaries and therefore divisional separation requirements are not impacted by this design change. The changes made by this change are seismically acceptable and this design change meets the same requirements as the original design. Therefore, this change does not increase the probability of occurrence for any existing failure mechanism.

Serial Number: 93-113-NSRA Document Evaluated: Feedwater Spargers

DESCRIPTION OF CHANGE: As a result of BWR feedwater nozzle cracking during the mid to late 1970s, the NRC issued NUREG-0619, "BWR Feedwater Nozzle and Control Rod Drive Return Line Nozzle Cracking". This document provided recommendations to minimize or possibly eliminate the problems associated with feedwater nozzle cracking.

Because of the confidence in ultrasonic (UT) examination technology available at the time, NUREG-0619 addressed UT examination of feedwater nozzles to be augmented with periodic liquid dye penetrant (PT) and visual (VT) examinations at a frequency consistent with the feedwater nozzle and sparger assumed vulnerability to cracking. GGNS's commitments in response to NUREG-0619 require performance of the following actions:

- volumetrically examine each of the six feedwater nozzles inner radius (blend radii), bore and safe-end regions through the use of an external ultrasonic inspection technique every second refueling outage.

- augment the above external UT examination by removing the sparger from one feedwater nozzle and performing a PT surface examination of that nozzle's bore and inner radius (blend radii), accessible areas of the other nozzles every nine refueling outages or 135 startup/shutdown cycles, whichever occurs first, and

- perform a visual examination of the flow holes and welds in the sparger arms and sparger tees every fourth refueling outage.

This evaluation affords GGNS the option of electing a revised commitment for examining the feedwater nozzles for indications of cracking. The revised commitment would require performance of the following actions in accordance with the methods and frequency recommended by ASME Section XI:

- two of the feedwater nozzles inner radius (blend radii) and bore regions will be volumetrically examined by use of an automated examination technique capable of reliably detecting cracks of 0.25 inch or less in depth, each 40 month period, with all six nozzles being inspected during each 10 year time interval.

  no change is being made to the examination method or frequency for the safe-end welds by this evaluation.

- the commitment to perform PT surface examination of the inner radius (blend radii) and bore region of one feedwater nozzle and accessible areas of the other nozzles every ninth refueling outage would be deleted.

  the visual examination of the sparger requirement is being retained.

Ultrasonic examination of the feedwater nozzles per the revised commitment will be performed using an automated examination methodology that has been demonstrated to accurately and reliably detect and depth size feedwater nozzle cracks of 0.25 inch or less in depth.

REASON FOR CHANGE: At the time NUREG-0619 was issued, the technical basis and understanding of the issues associated with feedwater nozzle cracking and the reliability of ultrasonic examinations were uncertain, and GGNS opted for the conservative course and adopted the augmented examination plan.

In retrospect, GGNS has eliminated the mechanisms that induce feedwater nozzle cracking and is supplementing the existing commitment to perform feedwater nozzle examinations in accordance with guidance from NUREG-0619, with a revised commitment to examine feedwater nozzles for indications of cracking in accordance with guidance from ASME Section XI. Historical data has indicated that GGNS has had no significant problems with feedwater nozzle cracking.

The automated UT process proposed to be used at GGNS is capable of detecting and depth sizing feedwater nozzle fatigue cracks of 0.25 inch or less in depth. The ASME Section XI (IWB 3512) limit is 10% of wall thickness, which is approximately 0.679 inch for the feedwater nozzles. This automated methodology has been demonstrated at nuclear plants and at General Electric Offices.

SAFETY EVALUATION: Feedwater nozzles experience cracking primarily because of metal stress induced by the temperature differential between colder incoming feedwater and the higher temperature water in the reactor vessel. Turbulent mixing of the hot water returning from the steam separator, dryer and the incoming colder feedwater coupled with bypass leakage past the junction of the thermal sleeve and nozzle safe-end results in cold water impinging upon the nozzle bore region. This results in thermal stress cycling of the nozzle bore region, thus causing nozzle cracking.

Additionally, the presence of stainless steel cladding on the feedwater nozzle surfaces contributes to fatigue cracking because the stresses from thermal cycling (particularity high frequency cycling) are higher in stainless steel than in unclad metals.

The Staff's SER states that "For some reactors with high (420°F) operating feedwater temperatures, the combination of clad removal and a zero leakage triple sleeve sparger may be all that is necessary to suppress cracking within the design lifetime". GGNS's design temperature is 420°F. Feedwater nozzle cracking that was discovered in domestic and foreign BWRs during the mid to late 1970s was attributed primarily to the rapid cycling of hot and cold water and the presence of stainless steel clad on the nozzles. Because GGNS does not have stainless steel cladding on the nozzles, has a feedwater flow controller in operation to control rapid cycling of hot and cold water and incorporates the triple thermal sleeve sparger design that eliminates bypass flow past the thermal sleeve and safe-end that induces nozzle cracking, cracking is not expected to occur at the feedwater nozzle bore regions or blend radii. Additionally, improved UT techniques are now able to detect cracks reliably before they exceed ASME code allowable limits.

Eliminating the PT examination of the feedwater nozzles inner radii and bore region every ninth refueling outages (135 startup/shutdown cycles) and extending the interval for performing the UT examinations of the nozzles from the NUREG-0619 augmented examination provides an alternative to the commitment for complying with the existing commitment. Since the design conditions at other plants which prompted the NUREG-0619 concerns (e.g., sparger design, nozzle cladding) do not exist at Grand Gulf, the commitment is being supplemented with an alternative commitment to comply with the normal ASME code Section XI requirements for feedwater nozzle examinations. The GGNS feedwater nozzle design, past examination history, plant operating characteristics, and normal ASME code testing with reliable UT techniques provide adequate assurance of feedwater nozzle integrity such that no unreviewed safety question exists.

Serial Number: 93-114-PSE Document Evaluated: WO #108347

DESCRIPTION OF CHANGE: Temporarily change the refueling platform monorail and frame mounted auxiliary hoist load cell setpoints from 500 to 1000 pounds and perform functional test.

REASON FOR CHANGE: Removal of Jet Pump #10 assembly mixer will utilize these auxiliary hoists per MWO 108400. Capacity greater than 500 pounds is necessary.

SAFETY EVALUATION: The subject auxiliary hoists are designed and tested for a rating of 1000 pounds, but are limited at GGNS to 500 pounds by technical specifications applicable to handling fuel and control rods. Since the refueling platform will be prohibited from any other usage during handling of Jet Pump #10 and since an LCO will provide administrative control, Technical Specification 3/4.9.6.1 will not be applicable. Thus, no technical specification or technical specification bases are affected. Also, since the hoists are capable structurally and operationally to handle the extra load, the subject change does not increase the probability or consequences of any accident or malfunction of equipment designed in the UFSAR. Since only the Jet Pump #10 parts will be handled in the changed configuration and since only the equipment designed for jet pump handling (jet pump grapple, etc.) will be used in conjunction with the subject auxiliary hoists, no new accident or equipment malfunction scenarios are created. No change to the applicable UFSAR sections is desirable because the change is temporary (for RF06 handling of Jet Pump #10 only).

Serial Number: 93-116-NPE Document Evaluated: MCP 93/1063 R00

DESCRIPTION OF CHANGE: MCP 93/1063 will specify partial removal of UHS basin transfer piping between the SSW "A" pump and the SSW "A" basin wall penetration. The affected SSW "A" basin penetration will be sealed by grouting to maintain the integrity of the basin fluid boundary. The MCP will also specify sealing of the SSW "B" train pump transfer line penetration into the SSW "A" basin. A hole will be drilled in the SSW "B" train transfer line between the transfer valve and the SSW "B" basin penetration for thermal relief.

REASON FOR CHANGE: The transfer piping below the normal SSW basin water 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 lines no longer serve any safety function, the corroded lines will be removed rather than replaced.

SAFETY EVALUATION: Due to the cancellation of Unit 2, and the subsequent installation of the passive siphon line between SSW basins, the active transfer mechanism installed as part of the original plant design no longer serves a safety function. Partial removal of 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 FSAR since the modification does not alter any safety related functions of equipment, and since the modifications maintain original design requirements for fluid boundaries and seismic qualifications.

The modification will not increase the probability of occurrence of a malfunction or increase the consequences of a malfunction of equipment important to safety previously evaluated in the FSAR since the modifications are required to meet the original system design requirements for fluid boundaries and for structural integrity, including seismic qualifications and ASME B&PV Code Section III.

The modification will not create the possibility for an accident of a different type than any previously evaluated in the FSAR 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 SAR due to the limitation of one operating cycle prior to additional modification being made to ensure the long term integrity of the UHS fluid boundary ("B" 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 days cooling water inventory are not reduced.

Serial Number: 93-117-NPE   Document Evaluated: MNCR 93-0195-R00

DESCRIPTION OF CHANGE: Change the model number of the installed Division III diesel generator jacket water high temperature switches from Fenwall to Square D Class 9025, Type BWG22. The Morrison Knudsen Part Number 208000 applies to both types.

REASON FOR CHANGE: The Square D switches were supplied as original equipment on the diesels, subsequently the switches became obsolete and Morrison Knudsen (suppliers of the diesels) revised their drawings to show the Fenwall switches as equivalent replacements. Design documentation was based on those revised drawings and did not depict the actual field installation.

SAFETY EVALUATION: Changing the model number of the switches does not require any change to hardware which was supplied as original equipment with the diesels. The original switches were specified by the manufacturer and conforms to the same design requirements as the recommended replacements. The original switches were seismically qualified with the diesels and perform the same function as the recommended replacements. Since no modifications are being made which will affect the overall system performance in a manner which could lead to an accident, 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 unreviewed safety questions resulting from this design change.

Serial Number: 93-118-NPE Document Evaluated: SCN 93-0002

DESCRIPTION OF CHANGE: This change eliminates requirement to perform hydraulic lifter leak down test. The test has been deleted by the TDI Diesel Generator Owners Group and is no longer considered mandatory by the manufacturer (Cooper).

REASON FOR CHANGE: MS-37 is based on Revision 2 of Appendix II (Generic Maintenance Matrix) of the Design Review and Quality Revalidation (DR/QR) Report prepared for MP&L, Grand Gulf Nuclear Station by TDI Diesel Generator Owners Group. Revision 3 of Appendix II of the DR/QR (approved 05/07/91 by Owners Group) eliminated requirement to perform leak down test on hydraulic lifters. The Owners Group justification for deleting the test was that an accurate method to determine leak down did not exist. The Owners Group stated that satisfactory lifter operation is monitored by pumping up lifters during reassembly and monitoring lifter operation during engine run.

The current criteria for the test is that the lifters should bleed down in 1.5 to 3 seconds. This test is based on the lifter manufacturer's factory test requirements which is performed under laboratory like conditions. It has been found that during site testing, a majority of the hydraulic lifters fail the leak down test because they are exceeding the 3 second leak down rate portion of the criteria (the majority of the failed lifters have a leak down rate of between 3 and 4 seconds). This has been experienced on lifters removed from the engine and on new lifters from the warehouse.

The lifter is designed to have a fixed leak down rate by maintaining a set diametrical clearance between the barrel and plunger. As the lifter is compressed hydraulic fluid is forced out of the lifter through the clearance. There is no credible mechanism that would cause the lifter leak down time to increase with use except build up of varnish. Build up of varnish has not been observed and would not be possible with new lifters from the warehouse. Wear of the components would increase the diametrical clearance causing the leak down time to be shorter not longer. It is concluded that the test failures are the result of attempting to reproduce the results of a laboratory type test under field conditions.

This change incorporates the allowance not to perform the lifter leak down test. This constitutes a change to the TDI Diesel Generator Maintenance/Surveillance Program, however, changes to the program are allowed if they are performed in accordance with 10CFR50.59 (reference Facility Operating License Attachment 2, Transamerica Delaval Inc. (TDI) Diesel Generator Maintenance and Surveillance Requirements (NUREG-1216, August 1986)).

SAFETY EVALUATION: The change eliminates a commitment to perform a leak down test of hydraulic lifters for DG11 and DG12. However, elimination of this inspection has been approved by TDI Diesel Generator Owners Group which initially specified the test. The NRC allows changes to the TDI Diesel Generator Maintenance Program as long as it does not change items specifically spelled out in Attachment 2 to the operating license and is performed in accordance with 10CFR50.59.

The change does not adversely affect the operability or reliability of the diesel generators; does not involve Phase 1 surveillance requirements of Paragraph 3 of Attachment 2 of the GGNS operating license; does not change license conditions or technical specifications; does not result in an unreviewed safety question; and does not reduce the margin of safety as defined in the basis for any technical specification.

Serial Number: 93-119-NPE Document Evaluated: GGNS-DCS-01 R00

DESCRIPTION OF CHANGE: This 50.59 safety evaluation is being written to address the design change to GGNS safety related and non-safety related valves, including valves which are in contact with reactor coolant, that will be authorized by Design Change Standard GGNS-DCS-01, "Design Change Standard for Valve Packing", Revision 0. DCS-01 implements the recommendations of EPRI Report NP-5697, May 1988, "Valve Stem Packing Improvements" (with the exception of live-loaded packing systems) and General Electric Company Service Information Letter (SIL) No. 513, "Use of Graphite Packing Rings in Valve Packing Chambers". In accordance with the recommendations of these documents, valves may be repacked with graphite materials and the packing configuration may be reduced to a single set consisting of two braided outer rings and two or three die-formed tape inner rings wherever possible. The stuffing box lantern rings in valves which do not have an active leak-off connection (i.e., piped to a collection point for leakage monitoring, or used to provide a pressurized fluid between the upper and lower packing sets) will be removed and replaced with carbon bushings which will take up excess room in the stuffing box and provide stem guidance. Valves which have active leak-off connections will be repacked with upper and lower sets of braided graphite rings and die-formed tape rings to allow the leak-off function to be maintained.

REASON FOR CHANGE: The valves installed in GGNS were supplied with a variety of packing materials and packing configurations. Many valves contain braided asbestos packing reinforced with Inconel 600 wire or contain graphite-impregnated braided asbestos packing. In larger valves the packing is often contained in a deep stuffing box, often in an upper and lower packing set arrangement separated by a lantern ring. Replacement asbestos packing is no longer available. Many existing valve packings contain zinc (as a corrosion inhibitor) in quantities considered detrimental due to potential metal embrittlement. These considerations, and the results of industry testing of improved packing systems, have resulted in the decision to replace existing packing materials, as valves require repacking, with graphite packing materials. Because the new packing materials are both braided packing and die-formed tape packing, the statement in UFSAR 12.3.1.1.3e2(a) concerning the use of "braided packing without any loose filler material" for valve packing in reactor coolant and auxiliary system design must be updated. Also, graphite packing materials are not "chloride free"; consequently, the statement in UFSAR 12.3.1.1.3e2(b) must be updated to state that packings with restricted chloride and other contaminant content will be used.

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SAFETY EVALUATION: NPE has evaluated the change of valve stem packing materials and packing configuration for those valves discussed in UFSAR 12.3.1.1.3e2, and for safety related valves in general which are governed by the GGNS Unit 1 Technical Specifications, and has determined that there will be no adverse effect on plant safety resulting from the change. No change to the GGNS Unit 1 Technical Specifications will be required. The change will not increase the probability of occurrence or consequences of an accident previously evaluated in the FSAR, increase the probability or consequences of a malfunction of equipment important to safety previously evaluated in the FSAR, create the possibility for an accident or malfunction of equipment important to safety of a different type than any previously evaluated in the FSAR or reduce the margin of safety as defined in the basis for any technical specification.

Serial Number: 93-120-NPE   Document Evaluated: MNCR 93-0077

DESCRIPTION OF CHANGE: This evaluation addresses disposition for the HPCS injection valve in MNCR 0077-93, generated to document the fact that, based on actual flow test results, the torque switch on the valve may have been set too low to assure that the actuator would deliver enough thrust to close the valve against the design basis maximum expected differential pressure (i.e., MEDP). When the torque switch setting was increased to provide the thrust values specified by Design Engineering (NPE) the following problems arose:

1. Due to the extremely small window or margin between the minimum required stem thrust (MRST) and the maximum allowable stem thrust (MAST) and the inaccuracies associated with the diagnostic test equipment field personnel were unable to set the torque switch such that the MRST value specified by NPE was met without exceeding the MAST value for the valve. Therefore the valve had to be left in an overthrust condition in order to assure that the valve would OPEN and CLOSE to perform its design safety functions. NPE has revised the seismic calculation for the valve and increased the OPEN and CLOSE MAST values; with the new OPEN and CLOSE MAST values the as-left torque switch setting will not result in an overthrust condition for the valve.

2. In order to achieve the minimum required stem thrust value specified by NPE, field personnel had to exceed the degraded voltage actuator capability (DVAC) torque value. In order for the actuator installed on the valve to produce enough thrust to satisfy the MRST value without exceeding the DVAC torque, the voltage at the motor terminals must be greater than or equal to 90% of the rated voltage. The undervoltage relays on the electrical bus are currently set to trip at a value equivalent to approximately 85% of the rated bus voltage, which means that, under worst case accident load conditions, the voltage at the motor terminals may not be high enough to assure that the actuator will produce enough thrust to close the valve against the design basis maximum expected differential pressure. In order to assure that the voltage at the motor terminals for the valve will be greater than 90% of rated during worst-case accident load conditions, the primary taps on the MCC transformer must be set at -5%. This change will assure that the DVAC torque capability of the actuator installed on the valve will be greater than the torque requirements for the current torque switch setting. A calculation has been performed to insure that the proposed tap changes are acceptable as stated in this evaluation.

REASON FOR CHANGE: An MNCR was generated to document the fact that, based on actual flow test results, the torque switch on the HPCS injection valve may have been set too low to assure that the actuator would deliver enough thrust to close the valve against the design basis maximum expected differential pressure (i.e., MEDP). When the torque switch setting was increased to provide the specified thrust values, problems arose as described above.

UFSAR Table 3.9-2ac summarizes, in detail, the design operating stresses which were calculated for the valve, therefore, the table must be updated to reflect the results of the revised calculation.

SAFETY EVALUATION: NPE has evaluated the increase in the OPEN and CLOSE MAST values for the valve and the raising of the primary taps on the high pressure core spray MCC transformer and determined that these changes will in no way jeopardize the structural integrity, functional capability or adversely impact the ability of the HPCS injection valve to perform its design safety functions or adversely impact any electrical equipment powered from the Division III Bus. Therefore, based on the facts presented above NPE has concluded that the specified changes for the valve will not require a change to the GGNS Unit 1 Technical Specifications and do not constitute an unreviewed safety question.

Serial Number: 93-121-NPE Document Evaluated: MCP 92/1051 R01 CN 93/0142

DESCRIPTION OF CHANGE: MCP 92/1051, Revision 1 will provide new undervoltage relays for the existing undervoltage relays: 127S1-17AC, 127S2-17AC, 127S3-17AC, 127S4-17AC, and 127N1 and 127N2 for Division III 4160 volt circuit breakers 152-1704, 152-1705 and 152-1706. The location of the replacement relays will remain the same as the existing relays; Division III 4160 volt switchgear, E22-S004; cubicles 101, 102, 105, 107 and 108. The new relays will be solid state AC voltage sensing relays that will pickup for a decline/loss of bus voltage for Division III. Since the undervoltage relays will be solid state, a power supply to operate the relay will be required. The nominal voltage to be furnished to the power supply shall be 125 Vdc, supplied by the Division III 125 Vdc batteries, 11DC. The new undervoltage relays (replacement relays for the existing undervoltage relays) shall be certified to IEEE 323 since the components shall be utilized as safety related/Class 1E equipment, including Seismic Category I.

Protection of the power supply shall be provided by fuses which shall be certified per IEEE 323 since the components shall be safety related/Class 1E equipment. No cables will be installed in support of this design. The jumper wire to be utilized internally to the Division III cubicles listed above will be Class 1E certified. The time delay relays (General Electric, type SAM - 62S1 & S2) associated with the existing undervoltage relays (127S1, 127S2, 127S3, & 127S4) shall be removed, because the time delay feature with the new undervoltage relays shall be utilized.

REASON FOR CHANGE: During a recent performance of surveillance for the bus undervoltage relays for Division III (January 1992), one of the bus undervoltage relays was found out of GGNS Technical Specifications and manufacturer's limits (127S1-17AC). This relay has a documented history of drifting outside GGNS Technical Specification limits and the allowable manufacturer's limits. In addition, it has been documented from previous surveillances that other undervoltage relays (same type and design function) for Bus 17AC have been drifting through the entire allowable range. The referenced relays are 127S2-17AC, 127S3-17AC, & 127S4). The time delay relay (GE, type SAM - 62S1 & S2) shall be removed and the time delay feature in the new relays (127S1, 127S2, 127S3, & 127S4) shall be utilized because they are more accurate and there will be less components in the circuit to fail. The time delay function is inherent to occur with these relays.

In addition to the relays previously mentioned, there are other undervoltage relays (same manufacturer and type) that provide assurances that the Division III diesel generator will provide voltage to Bus 17AC for a loss of AC power, 127N1 and 127N2 - 4160 volt circuit breakers 152-1704, 152-1705, and 152-1706. Since these relays are the same type, similar problems have the potential to occur with these relays.

SAFETY EVALUATION: The function of the referenced undervoltage relays is to provide undervoltage protection for Bus 17AC for loss of AC voltage. The undervoltage relay replacements will not adversely impact current engineering design for the Division III protection scheme. The existing Division III circuits will be maintained in accordance with Regulatory Guide 1.75 and IEEE 308, in that no single failure in any of the undervoltage relays will result in conditions that prevent safe shutdown of the plant and that each system will remain located in an area separated physically from other ESF systems. The new undervoltage relays shall be solid state relays, however the design does not consist of programmable software. Therefore, the use of these relays will not create an unreviewed safety question. With the implementation of this design change, new loads shall be added to the 125 volt DC system, Division III batteries 11DC. However, no adverse impact will result to the Division III 125 volt DC system or the batteries. Electrical protection of the 125 Vdc power supply shall be via Class 1E fuses. With the removal of the SAM relays (62S1 & S2), the technical specification time delay requirement will not be affected because the time delay feature in the new undervoltage relays shall be utilized. All protective fuses utilized in the design change shall be safety related and shall be considered functional for Seismic I use because of the small size, low mass, compact construction and/or lack of moving parts. A seismic test report for the replacement undervoltage relays, including verification that the accuracy rating of the pickup settings is within the specified requirement; + 1.95 of setpoint shall be provided to ensure proper operation of the relays during a seismic event.

Serial Number: 93-123-NPE Document Evaluated: TSR 93-017

DESCRIPTION OF CHANGE: The change consists of the temporary addition of lead shielding blankets to certain portions of the LPCI and LPCS piping for the purpose of reducing personnel exposure during refueling maintenance activities. Lines to be shielded from RPV to F041C valve (Reference Isolation M-1348T), from RPV to F007 valve (Reference Isolation M-1350B) and from Valve F039A to 14"x12" reducer (Reference Isolation M-1348F).

The temporary shielding will only be installed during Reactor Modes 4&5 and removed prior to plant startup.

REASON FOR CHANGE: Temporary Shielding Request 93-17 estimated personnel exposure savings of 2,070 personrem.

SAFETY EVALUATION: A review of the parent calculations PDS-66, 55 and 101 indicates that the increase in pipe weight due to the addition of the temporary lead shielding will have no impact on the safety or the structural integrity of the piping system. This conclusion is based on the fact that the effect of the temporary lead shielding weight is negligible relative to the contribution of the hydrodynamic loads to the piping total combined stress of each piping system affected. Hydrodynamic loads (i.e., SRV, annulus pressurization, etc.) are not creditable events during Modes 4 & 5.

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

Serial Number: 93-124-PSE Document Evaluated: Temp Alt 93/0019

DESCRIPTION OF CHANGE: Temporary Alteration 93/019 will cross-tie Plant Chilled Water (P71) to Drywell Chilled Water (P72) as a means of supply temporary cooling to drywell and containment during RF06. This task will be accomplished in two steps:

1. Tying a temporary chiller into the P71 system through the Turbine Building heating heat exchanger (1P71B002).

2. P71 will be cross-tied to P72 in the Auxiliary Building.

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, management made a decision to provide temporary cooling to Containment, Drywell and Auxiliary Buildings during RF06.

SAFETY EVALUATION: DCP 88/0021 (base package and Supplement 1), decommissioning of the Auxiliary Steam System (N12), has safety evaluations which evaluate the removal or abandonment of the Auxiliary Steam System. The Turbine Building Heating Heat Exchanger 1P71B002, was originally designed to be used with the N12 system and is therefore being abandoned as part of this DCP. Both the base package and the supplement are scheduled to be worked during RF07. Also, Safety Evaluation 93-0087, Revision 0 evaluated the installation of temporary cooling equipment for Turbine Building 166 elevation. This temporary equipment included the chillers being used by this temporary alteration and a booster pump.

As stated in Section 9.2.7 and 9.2.11 of the FSAR, 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 system 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.

The routing for the temporary piping being used in this temporary alteration will be in close proximity to some electrical panels and other electrical panels. The routing on Turbine Building Elevation 133 will run behind Panel 1H22P176 and MCC 13B22. The P176 panel is an N62/N64 control panel and the 13B22 MCC contains electrical loads that are not safety related or required for safe shutdown; therefore the routing on Elevation 133 in the Turbine Building is not a concern.

If the cross-tie between P71 and P72 is made between Elevations 139 and 119 of the Auxiliary Building, the tempo:ary pipe routing will be in close proximity to MCC 14B11 and LCCs 14BE1 and 12BE2. All electrical equipment listed above is on Elevation 119 of the Auxiliary Building. If the cross :ie is made, all piping routed in close proximity to electrical equipment will have splash guarding installed to prevent a pipe break from spraying down the equipment.

Serial Number: 93-125-NSRA Document Evaluated: Jet Pump Beam Replacement

DESCRIPTION OF CHANGE: Performance of inspections and examinations 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 beams of Jet Pumps 8 and 21. These jet pump hold down beams are being replaced with like components.

This safety evaluation is being written to take exception to the commitment regarding replacement of jet pump beams. GGNS committed (SER 3.9.2 and AECM-80/0268)) to replace the currently installed BWR 4-6 jet pump hold down beams where excessive cracking is identified, with improved heat treated hold down beams developed by General Electric (GE), if approved by the NRC. The improved design beams have subsequently received NRC approval, however, the heat treated beams are not readily available.

REASON FOR CHANGE: This change takes exception to the existing commitment by allowing jet pump hold down beams to be replaced with like components (new BWR 4-6 beams) rather than with heat treated hold down beams as committed in the GGNS SER (Section 3.9.2).

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 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 BWR 4-6 beams from 30 kips to 25 kpis and conduct periodic examinations of the beams for cracking. Another long term solution was to replace cracked BWR 4-6 beams with new heat treated beams.

As recommended by GE in SIL 330, GGNS reduced the preload on the jet pump beams from 30 kips to 25 kips in order to increase beam operating time and incorporated a requirement to perform periodic examination of the jet pump hold down beams into the GGNS ISI program. Industry experience has not indicated any failures of jet pump hold down beams for the first five years in service. The jet pump beams installed at GGNS have performed reliably for 8 years of commercial operation without any failures. Based on past performance history and reduced beam preloading, replacement of jet pump hold down beams with like components is justified for the next operating cycle.

Serial Number: 93-126-NPE Document Evaluated: TSR 93-007

DESCRIPTION OF CHANGE: TSR 007-93 requested the addition of lead blankets on the containment floor drain piping and the containment equipment drain piping. The location of the temporary lead shielding is in containment, floor elevation 135'-4". The drainage lines to be shielded are identified on Drawing M-1280, Revision 5 as CRW 30 and DRW 21. The lead blankets will be added from the top of the sumps at Elevation 138'-9" upward for 5'. The added weight to each drainage piping system will be 60 lbs/ft or a total of 300 lb. The temporary lead shielding will remain in place until the end of 1993.

REASON FOR CHANGE: The reason for the addition of temporary lead shielding is to reduce the exposure to personnel performing: 1) I&C surveillances, PMs, Maintenance, 2) Mechanical Maintenance PMs & Maintenance, 3) Operations rounds, C11 & HPU surveillances, and 4) P&SE monitorings B33 systems. The estimated net personrem savings is calculated to be 3.560 and the cost benefit margin is estimated to be $34,600.

SAFETY EVALUATION: Calculations MC-N1P48-93023 Revision 0 and MC-N1P48-93024 Revision 0 qualified the structural integrity of the piping and supports for the subject drainage system for Operating Modes 1 through 5. The structural integrity of the pipe and supports are confirmed by the results of the calculation being within code allowables.

Serial Number: 93-127-PSE Document Evaluated: WP 19931061 00 00

DESCRIPTION OF CHANGE: The discharge isolation valve for the A Reactor Recirculation Pump requires repair on the lower stem to upper wedge connection. The repair activities include the removal of the bonnet to install a new lock bolt, which replaces the missing OEM lock pin, and then reinstalling the bonnet. The discs wiii be locked in the seats for a majority of the repair activities. The jet pump plugs will form part of the reactor coolant pressure boundary during a small portion of the repair activities, approximately one to five minutes based on mock-up simulation of the work activity, and thus represent a function "important to safety" consisting of maintaining reactor coolant inventory.

REASON FOR CHANGE: A recent fiber optic inspection performed on the discharge isolation valve showed that the lock pin for the lower stem to upper wedge connection was missing. This change provides the design for the valve internals repair. There is no isolation between this valve and the reactor vessel. The valve must be repaired to ensure that it will perform its intended design function during the upcoming operating cycle. Repairs to be performed will restore the valve to an acceptable design configuration.

SAFETY EVALUATION: The safety significance of this repair centers around the potential to drain the vessel during the repair activities. The repair of the valve is necessary to restore the valve to its intended design and ensure it will function as designed during the upcoming operating cycle. Specific actions and contingencies described below will be required to ensure that this potential is kept to an acceptably low level.

- Redundant gags to maintain the valve discs on the seats while the bonnet is removed.

- Jet pump plugs shall be installed.

- Continuous monitoring of upper pool level and valve leakage.

- All refueling pool gates shall be removed and fuel transfer canal open.

- All work above and in adjacent areas to the valve shall be stopped.

- No work in the annulus area above or near Jet Pumps 1 through 12.

- Suspend fuel movement (with the exception of placing a grappled bundle in a safe location) if a decrease in upper pool level is observed.

- The requirements of our infrequently performed tests or evolutions procedure, shall be followed.

  Special training, mockup and management/supervisory oversight shall be required for the development, review and conduct of infrequently performed evolutions that have the potential of significantly degrading the plant margin of safety.

- No other operations with a potential to drain the vessel may be conducted during valve repairs.

- A passive means of securing the valve in the closed position without the need for offsite power shall be established.

- Suppression pool level shall be maintained at a minimum of approximately 16 feet.

- Two separate suction paths for the removal of decay heat shall be available.

- One method of decay heat removal should be provided by an operable SDC system.

- Bus 12 shall be available.

  This bus will provide drywell lighting and power for the drill motor to be used to raise the valve bonnet. No ous maintenance should be in progress that has the potential for disrupting this power supply.

- Back-up power supply for the drill motor shall be available.

- Additional power requirements shall include two service transformers and two offsite power supplies.

- Repair work should not be performed during or immediately prior to severe weather.

- Division 1 diesel generator, LPCI "A" and SSW "A" shall be OPERABLE. Either LPCI "B" or LPCS shall be functional. Division 3 ECCS consisting of the Division 3 diesel generator, high pressure core spray and SSW "C" shall be functional.

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Any of the available ECCS pumps will provide sufficient coolant to the vessel to prevent fuel from being uncovered in the event of a loss of vessel inventory. In conclusion, these actions provide assurance that no unreviewed safety question exists for the performance of the repair described above.

Serial Number: 93-128-PSE Document Evaluated: TSTI 1N21-93-005-0-N

DESCRIPTION OF CHANGE: This TSTI represents the testing to be performed on the reactor feedpump turbine (RFPT) controls during Plant Operating Conditions 2 and 1 following RF06. The objective of this TSTI are:

- Demonstrate that the RFPT has proper response and stability to control inputs and small speed demand step changes.
- Demonstrate that between 2100 rpm and 3000 rpm, that the RFPTs are controllable and that speed is stable when they are being controlled by the FW master controller in MANUAL MODE.
- Demonstrate that the transient response of reactor water level control system related variables to test inputs do not diverge.
- Demonstrate that any oscillatory modes of response of reactor water level control system related variables to test inputs have a decay ratio of less than or equal to 0.25.
- Demonstrate that the open loop dynamic response of each RFPT to small (<10% of rated pump flow) and large (>20% of rated pump flow) step disturbances meets the criteria of Reference 7.

REASON FOR CHANGE: This TSTI provides partial retest for DCP 91/0088-2 which upgraded the reactor feed pump turbine mechanical hydraulic controls to digital electro-hydraulic controls.

SAFETY EVALUATION: The majority of this TSTI repeats the power ascension testing of the feedwater and Feedwater Control System described in UFSAR Section 14.2.12.3.20.2. The remainder tests the RFPT off-line using main steam and discharging to the main condenser through the RFP minimum flow lines. The test demonstrates that the new control system produces a transient response to small and large step changes that is the same or better than the original system. Therefore, the overall response of the Feedwater/Feedwater Control System is not adversely impacted by the modification. The original power ascension testing (UFSAR Section 14.2.12.3.20.2) as conducted with a relatively new reactor core with less operating history than the present core. In addition, though this TSTI will be performed at similar power levels as the original testing, there is a potential that different control rod patterns and core flow conditions may exist. However, the testing performed by this TSTI is dependent only on reactor power, not core history, control rod pattern, or core flow rate.

Serial Number: 93-130-NPE   Document Evaluated: CN 93/0143 to DCP 0088-03 R00

DESCRIPTION OF CHANGE. This change will provide the electrical power and grounding required for reliable operation of the equipment to be installed as part of the Feedwater System Reliability Upgrade Project.

To provide the power for the Digital Process Control (DCS) cabinets, a new 125 Vdc input/120 Vac output inverter, 480/120 Vac bypass transformer, and distribution panel will be installed to provide power for one of the two 120 Vac inputs required for each DCS panel. The second source of 120 Vac power will be derived from existing power panels via 208/120 Vac isolation transformers. There will be three isolation transformers installed for each panel location. The output of the new distribution panel and the new isolation transformers will be dedicated to the DCS equipment to minimize transient voltages which may be impressed on the system. All of the components installed for this portion of the DCP and the associated cabling and raceway will be installed inside the Turbine Building.

To ensure a solid signal common reference is achieved for the DCS cabinets, each cabinet location will have a direct tie to the earth ground mat. Excavation of areas outside the Turbine Building will be required to access the ground loop conductor. Some of these areas are inside the tie back wall and will be below the clay seal and in the structural backfill.

To provide the power for the hydraulic skids, 480 Vac will be provided by BOP Load Centers 12BE1 and 14BE2. The power for the AHUs will be derived from these skids. All 120 Vac required for skid controls will be derived from control transformers installed on the skids. All cabling and raceway for these circuits will be installed in the Turbine Building.

To accomplish the cable installations described above, opening and closure of several penetrations in the Turbine Building will be required. Instructions have been provided to ensure opening and closure in accordance with approved design specifications.

REASON FOR CHANGE: As part of the Feedwater System Reliability Upgrade, new equipment will be installed and certain modifications to existing equipment will be performed to enhance the Feedwater System performance.

This change will provide the electrical power and grounding required for the reliable operation of the equipment installed as part of this upgrade.

SAFETY EVALUATION: All power sources utilized for this DCP supplement will be non-safety related. The existing raceways utilized will be BOP and the new raceways installed will be routed and tagged as BOP. The cables utilized for this design are appropriately sized for their loads and are adequately protected. All cable additions, with the exception of the portion of the ground cable installed outdoor, will be installed in the Turbine Building.

The affected penetrations are through fire barriers which are not required to separate safe shutdown components or to protect safe shutdown components from fire hazards. These penetrations are not governed by the Fire Hazards Analysis or the FSAR, and are not addressed in the technical specifications. The excavation and backfilling activities required to access the ground loop conductor will be performed in accordance with accepted standards.

The design documents referenced above ensure that the Category I structural backfill and clay seal will be properly restored. Furthermore, there are no safety related components in the general area where excavation is to occur. Additionally, this change affects non-safety related penetrations and the excavation activities are limited to the western portions of the Turbine Building. Neither the affected penetrations nor the excavation activities represent any change that constitutes an unreviewed safety question.

Serial Number: 93-132-NPE Document Evaluated: TSR 93-018

DESCRIPTION OF CHANGE: The change consists of the temporary addition of lead shielding blankets to certain portions of the RHR B piping for the purpose of reducing personnel exposure during refueling outage when taking erosion/corrosion examinations. The line to be shielded is 4-GBB-85 starting at the tee west of 1E12-F086 to the branch connection with 12-GBB-86 which is approximately 7.6 feet (Reference M-1348E).

The temporary lead shielding will only be installed during Reactor Modes 4 and 5 and removed prior to the plant startup.

REASON FOR CHANGE: Temporary lead shielding 93-18 has a estimated personnel exposure savings of .344 personrem.

SAFETY EVALUATION: A review of the parent calculation PDS-365 indicates that the increase in pipe weight due to the addition of the temporary lead shielding will have no impact on the safety or the structural integrity of the piping system. This conclusion is based upon the fact that the effect of the temporary lead shielding weight is negligible relative to the contribution of the hydrodynamic loads to the piping system. Hydrodynamic loads are not creditable events during Reactor Modes 4 and 5 (Reference GGNS 93-0024). No permanent plant changes will result due to the introduction of the shielding.

Serial Number: 93-133-NPE Document Evaluated: MNCR 93-0148

DESCRIPTION OF CHANGE: MNCR 0148-93 describes a leak which was found between P47F112 valve and P47F117 valve. A blind flange will be placed downstream of the P47F112 valve and the remaining piping and valves downstream will be abandoned in place.

REASON FOR CHANGE: A leak was detected in the piping downstream of the P47F112 valve and the remaining piping and valves are not required for operation of the plant.

SAFETY EVALUATION: The purpose of this MNCR is to repair a leak in the PSW Radial Well system by installing a blind flange and abandoning the remaining piping. The PSW system is a non-safety related system whose failure will not compromise any safety-related systems or components or prevent safe shutdown of the reactor. The modification associated with MNCR 0148-93 will not involve any equipment considered in any accident or malfunction previously analyzed in the UFSAR. The modifications meet the requirements of the applicable codes and standards and the abandoned piping is located underground and will not cause any safety hazards. Therefore, implementation of MNCR 0148-93 will not increase accident, or malfunction, probabilities or consequences. Implementation will not create the probability of a different type of accident or malfunction of equipment and does not reduce a margin of safety as described in the technical specification bases. The proposed modifications will not involve an unreviewed safety question.

Serial Number: 93-134-NPE   Document Evaluated: UFSAR CR NPE-93-013

DESCRIPTION OF CHANGE: Delete Table 3.10-1, which only contains a listing of master part list (MPL) numbers of the seismically qualified components and associated references to the table in the UFSAR. Also make minor editorial changes to the test in Section 3.10.

REASON FOR CHANGE: Table 3.10-1 only provides a listing of master part list numbers of those components which are seismically qualified. Detailed information pertaining the seismic qualification of the devices listed in Table 3.10-1 is maintained in the Seismic Qualification Central File Index (SQCF index). The SQCF index and the SQCF is maintained by controlled administrative procedures, auditable under the GGNS QA Program.

The SQCF is considered an extension of the UFSAR and is controlled under the GGNS QA program and 10CFR50.59 program for any revisions. Maintaining both Table 3.10-1 and the SQCF index is a duplication of information which is not necessary and is costly to maintain. Since the SQCF contains all the information for the qualification of the qualified components in the plant and maintains the design control for these components, Table 3.10-1, which only maintains the listing of the MPL numbers of the qualified components, is not necessary.

SAFETY EVALUATION: Deletion of Table 3.10-1, which contains the listing of the MPL numbers of the seismically qualified components, will have no adverse effects on the design control of the seismic qualification of components. The probability of occurrence or the consequences of an accident previously evaluated in the FSAR will not be increased by the deletion of the table of MPL numbers. The increase in the probability of occurrence or the consequences of a malfunction of equipment important to safety previously evaluated will not be increased. No new accidents will be created with the deletion of the table. The seismic qualification of all components will still be maintained and controlled by the SQCF as is currently being done under the same administrative procedures and will still be auditable under the GGNS QA program. The deletion of the table will remove the duplication of work by not requiring revisions to both the SQCF index and the UFSAR. The revisions of the SQCF will still require the same design control and design verification as the changes to the UFSAR require.

Serial Number: 93-135-NPE Document Evaluated: UFSAR CR NPE 92-036

DESCRIPTION OF CHANGE: UFSAR Table 5.4-3 which lists the RHR pump and valve automatic logic actuations, must be revised to reflect the correct logic configuration for RHR to radwaste (Automatic Isolation Group 2) valves as follows: RHR discharge to Radwaste Inboard and Outboard Isolation Valves E12-F040A and E12-F049B, RHR A&B Isolation Valves E12-F037A&B to upper containment pool, and RHR A&B Inboard and Outboard Process Sampling Isolation Valves E12-F060A&B and E12-F075A&B. All of these valves are isolated by the same RHR system isolation signals as specified for Group 3 Containment Isolation Valves E12-F037A&B in the TRM. The Group 2 drywell pressure signal is used in place of the Group 3 reactor vessel pressure signal for Valves E12-F037A&B.

REASON FOR CHANGE: This change to the UFSAR is required to ensure appropriate interpretation of the RHR system's function and operation. This change to UFSAR Table 5.4-3 will provide actual RHR system logic information that is currently depicted inaccurately in UFSAR detail.

SAFETY EVALUATION: This revision of the UFSAR is not being done in association with any design change or modification to the plant. This change is required to bring the UFSAR into agreement with RHR system design configuration and to provide the proper logic detail required in UFSAR Table 5.4-3. This change does not affect the analysis or description of the RHR system defined by UFSAR Section 5.4.7 nor does it change any TRM or technical specification detail or requirement. The logic is consistent with RHR system isolation signals included in the TRM. The changes to Table 5.4-3 do not affect system function or operation and cannot create the possibility of accidents of a different type. Since the logic is consistent with existing system logic in the TRM, the margin of safety is not affected.

Serial Number: 93-136-NPE Document Evaluated: CN 93/0176
DCP 91/088-9

DESCRIPTION OF CHANGE: Add scope to DCP 91/0088-9, Revision 1 to change the failed position of the reactor feed pump recirculation valves from fail open to fail close.

REASON FOR CHANGE: The change will produce a higher seating force on the valve plug to prevent steam leakage across the valve seat when the valve is normally closed. The change will also improve the reliability of the Feedwater System to provide feed flow to the reactor vessel on a loss of instrument air.

SAFETY EVALUATION: The Feedwater System serves no safety function. Systems analysis has shown that failure of the Feedwater System will not compromise any safety related systems or prevent safe shutdown. The proposed design change will not alter the normal operation of th[illegible]ctor feed pump minimum flow protection and will maintain pump [illegible] against low flow even in the event that the recirculation valves ar[illegible] to loss of instrument air or due to any other failure mechanism associ[illegible] recirculation valves. The only FSAR change required is the figure chan[illegible]ing the system process diagram which shows the failed position of the recircu[illegible] valve.

Serial Number: 93-137-PSE Document Evaluated: Temp Alt 93/0021

DESCRIPTION OF CHANGE: In order to support testing of the diesel generator (D/G) while the Standby Service Water (SSW) B cooling tower spray headers are being replaced the normal water return path cannot be used. The flanged joint downstream of the 1P41F005B valve which had a startup strainer installed will have a blank installed. The 2P41F011 valve will be removed and temporary piping connected to the piping which was downstream of 2P41F011. The temporary piping will direct flow from the SSW B return header directly to the basin through a floor penetration.

The temporary alteration will not affect either basin fill capability, basin level instrumentation or cause basin contents to be discharged.

REASON FOR CHANGE: This temporary arrangement will facilitate diesel generator testing while the SSW B cooling tower spray headers are being replaced. This is an evaluation of the temporary piping arrangement.

SAFETY EVALUATION: The SSW B loop will be declared inoperable during this evaluation, with appropriate actions taken. The system will be operated within its design flow and pressure parameters, and will not affect the continued operation of the SSW A loop. The overall system integrity will not be reduced and no barriers to radiation release will be removed. This temporary arrangement will not pose an unreviewed safety question.

Serial Number: 93-138-PSE Document Evaluated: EER 93/6294

DESCRIPTION OF CHANGE: The suppression pool floor and annular area between the drywell and containment require periodic cleaning to remove dirt, debris and foreign material to prevent clogging or partial blockage of ECCS suppression pool suction strainers. Several methods have been considered to accomplish this task, however, many of the options do not represent optimum cost efficiency or meet the intent of ALARA principles. The proposed method for cleaning the suppression pool floor will remove one of two P11 suppression pool suction strainer screens, and use suction from refueling water transfer pumps to vacuum affected areas. This task will be performed during Reactor Mode 4 or 5.

REASON FOR CHANGE: Cleaning the suppression pool and annular area between the drywell and containment is required to ensure removal of debris and foreign material from these areas that could possible foul ECCS pump suppression pool suction strainers. Fouling of the suppression pool suction strainers could create conditions that would result in degraded ECCS performance.

SAFETY EVALUATION: Seismic qualification of the screens will not be compromised by cleaning because total weight for cleaning attachments is less than permanent strainer assembly weight including mounting flange. Each suction strainer screen assembly weighs approximately 50 pounds. Hose weight is approximately 30 pounds (dry) per 20 foot section, and the hose connection point is approximately 1.75 feet from suppression pool floor. Accordingly, hose weight will not exceed approximately 2.75 pounds because the hose will be resting on suppression pool floor. The flange that will connect to suction strainer housing and hose will be identical to permanent suction strainer screen flanges, and will not exceed 40 pounds. Total weight of hose, hose connections and flange connected to suppression pool suction strainer housing will not exceed 50 pounds.

The cleaning attachments will be equipped with strainer screens identical to the screen size installed on the permanent suppression pool suction strainers. The appliance on the suction end of the hose will be covered with screen which will prevent all particles larger than 1650 microns from entering the refueling water transfer pumps. Use of this screen is acceptable since material passing through this screen size is too small to damage pumps.

Specific controls must be implemented and followed to prevent damage to the refueling water transfer pumps and comply with technical specifications and design requirements. These controls ensure that cleaning resembles normal system operation and design as closely as possible, and allows some flexibility in work process.

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The safety evaluation determined that the proposed cleaning method described does not require a change to the GGNS Technical Specifications or represent an unreviewed safety question. The controls prescribed will provide necessary safeguards to ensure that the system components will be operated in a normal manner, will not be subjected to unusual or abnormal conditions and will comply with technical specification and design requirements. The safety evaluation also concluded that use of hose and associated attachments in the suppression pool and weir wall areas during Reactor Modes 4 and 5 did not introduce any conditions that constitute unreviewed safety questions. Controls placed in the evaluation will ensure that there will be no challenges to operable safety related components during cleaning.

Serial Number: 93-141-NPE Document Evaluated: SCN 93-0001A

DESCRIPTION OF CHANGE: SDDR 93-0058 requests acceptance of changes to the locking screw of the containment transfer tube closure hatch (Q1F11E015). The changes include: 1) Deletion of ASME classification of the locking screws and the bar attached to the locking screw; 2) Use of ASTM material in lieu of ASME material for the above parts; 3) Change to the surface examination procedure for the above parts; 4) Adding straightness requirements for the locking screw. These changes do not affect the Fuel Transfer System or any of its components as described in the FSAR. SCNs to the design and purchase specifications have been issued to allow shipment of the material. MCP 93/1074 will be issued to allow installation of the parts.

REASON FOR CHANGE: During a review of the subject components for purchase of replacement piece parts, the code classification was investigated for ASME code applicability. Certain parts were identified as not required to be within the ASME Section III, Subsection NE, for Class MC boundary.

SAFETY EVALUATION: The requested changes per SDDR 93-0058 have been evaluated and determined not to affect the functionality, seismic qualification, or adequacy of the containment transfer tube closure hatch. The locking screw and the bar attached to the locking screw have been determined as non-ASME, but the safety related classification has been maintained.

Serial Number: 93-144-NPE Document Evaluated: MCP 93/1074 R00

DESCRIPTION OF CHANGE: MCP 93/1074 was initiated to provide replacement parts for the locking screw of the containment transfer tube closure hatch (Q1F11E015) per SDDR 93-0058. The MCP is required to address changes in the material technical code requirements. The changes include deletion of ASME certification of the locking screw and the bar attached to the locking screw and use of ASTM material in lieu of ASME material for the above parts. These changes do not affect the Fuel Transfer System or any of its components as described in the FSAR.

REASON FOR CHANGE: During a review of the subject components for purchase of replacement piece parts, the code classification was investigated for ASME code applicability. Certain parts were identified as not required to be within the ASME Section III, Subsection NE, for Class MC boundary.

SAFETY EVALUATION: The changes per MCP 93/1074 have been evaluated and determined not to affect the functionality or adequacy of the containment transfer tube closure hatch. The locking screw and the bar attached to the locking screw have been determined as non-ASME, but the safety related classification has been maintained.

Serial Number: 93-145-PSE Document Evaluated: 07-S-186 R15 TCN 5

DESCRIPTION OF CHANGE. This safety evaluation assesses the removal and reinstallation of the reactor steam separator in the reactor vessel with the reactor cavity drained. It has been past practice to remove/reinstall the steam separator underwater in the upper containment pool, except for the brief period of time necessary to clear the weir wall. The UFSAR will be changed to clarify that the timing of the flooding with respect to separator lifts is not critical. Revision 1 of this safety evaluation includes separator removal (Revision 0 included only the evaluation of separator installation). Also, consequence responses referring to compliance with NUREG 0612, Section 5.1.1 are improved by citing NPE's review of potential consequences documented by GIN-93/04568, Engineering Report GGNS 930037.

REASON FOR CHANGE: This evolution allows scheduling flexibility during refueling outages.

SAFETY EVALUATION: During previous refueling outages at GGNS, the steam separator has been removed/installed while remaining submerged for the most part under water. This transport method is being changed to permit steam separator removal and reinstallation while not submerged. This change in transport method for the steam separator is dependent on meeting the applicable criteria of NUREG-0612, Section 5.1.1 (Phase I requirements) for handling heavy loads over irradiated fuel. A review of the referenced documents has determined that this change continues to satisfy the applicable requirements of NUREG 0612.

Serial Number: 93-146-NPE   Document Evaluated: Cycle 7 Reload Analysis

DESCRIPTION OF CHANGE: This safety evaluation assesses operation with the Cycle 7 core configuration. Adetailed description of the Cycle 7 core and the issues considered in this evaluation is attached to this evaluation.

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: 1) will require no changes to the current GGNS Technical Specifications, and 2) will not constitute an unreviewed safety question.

Serial Number: 93-148-NPE Document Evaluated: DRR 88/36

DESCRIPTION OF CHANGE: This change documents the SWO correction made to drawing M-1067C. The change made was to the service line number associated with valves G18F085 and G18F086. The line number was changed from JDD-182 to JDD-132.

REASON FOR CHANGE: An editorial error in a drawing was corrected by changing the number designating the service lines for valves G18F085 and G18F086. This is an evaluation of that change.

SAFETY EVALUATION: This is a drawing change only and will not affect the operation or function of any plant component. This is an editorial change and will not increase the probability or consequences of an accident or malfunction, and will not decrease any margins of safety.

Serial Number: 93-149-NPE Document Evaluated: MNCR 87-0079

DESCRIPTION OF CHANGE: Although rework activities as a result of a discovered non-conformance have been implemented, some Hydraulic Control Units (HCU) mounting bolts are still missing or loose and cannot be installed or tightened due to interferences and inaccessibility. In addition, various weld deviations and broken or missing washers have been identified.

REASON FOR CHANGE: This evaluation was performed to review continued operation with the present condition of the HCUs.

SAFETY EVALUATION: The operability, structural integrity and seismic qualifications of the HCUs are not affected by the missing/loose bolts and washers and weld deviations. Therefore, continued operability with the HCUs in their present condition will not pose an unreviewed safety question.

Serial Number: 93-150-NPE Document Evaluated: MNCR 86-0131

DESCRIPTION OF CHANGE: A 1" dielectric union in the Instrument Air System was found leaking. A threaded gate valve was installed into this union as a mean of stopping the leak without shutting down the system. This new valve will remain permanently installed and has been assigned a valve number of N1P53FA05. This valve will be indicated as normally open on drawings. Although this valve serves no function, it could be used as an isolation valve for its open ended line. This new valve will not impact the structural integrity or function of the piping or pipe supports system.

REASON FOR CHANGE: This evaluation was performed to review the addition of a threaded gate valve on 1" line in the Instrument Air System. The valve was added because of leaks found in a dielectric union on that line.

SAFETY EVALUATION: The added valve meets all applicable design requirements and is designated as non-safety related. The valve was added to a portion of the Instrument Air System which has no safety-related function. The addition will not affect any safety-related system or component, or prevent safe reactor shutdown. Likewise, a failure of the affected portion will not effect any safety-related component or system, or prevent safe reactor shutdown. The addition of this valve will not increase the probability or consequences of an accident or malfunction, and will not decrease any margin of safety.

Serial Number: 93-151-NPE   Document Evaluated: MNCR 88-0133

DESCRIPTION OF CHANGE: A determination was made to remove the internals (i.e., screens) from the strainer installed in the drain line from the precoat addition and body feed tanks to radwaste. Each of these tanks has an overflow line which ties into the drain line upstream of the strainer to provide protection against overfilling; however, when liquid containing the body feed or precoat material is introduced into the drain, via the overflow lines, the strainer becomes blocked very quickly causing the liquid in the tanks to backup and spill onto the Radwaste Building floor. The internals of the strainer were removed to eliminate the blockage which was causing the tanks to overflow and spill onto the Radwaste Building floor.

REASON FOR CHANGE: The internals of the strainer were removed to eliminate the blockage which caused the tanks to overflow. This evaluation reviews the potential effects of operation with the strainer internals removed.

SAFETY EVALUATION: The removal of the strainer internals will allow some body feed and precoat material to enter the floor drain collection tank. This material, however, will be processed through the floor drain filter, and will have no effect on the Radwaste System. In addition, the Radwaste System is non-safety related and is not required to mitigate the consequences of an accident. This change will not increase the probability or consequences of any accident or malfunction; nor decrease any margins of safety.

Serial Number: 93-152-NPE Document Evaluated: MNCR 87-0214

DESCRIPTION OF CHANGE: The south and east walls of the elevator machine room are delineated to be 2-hour fire rated barriers. Room 1T503, located on Turbine Building Elevation 186'3", has penetrations in its ceiling, floor, south and east walls which are not provided with fire rated closures. It is conservatively assumed that a fire may propagate into Elevator 2 by way of Room 1T503.

Appendix A to Branch Technical Position APCSB 9.5-1, Section D.4.f indicates that stairwells, elevators and chutes should be enclosed in fire rated masonry towers or appropriate access routes should be established by pre-fire plan and practiced in drills. Staircases serve as escape routes and access routes for fire fighting while elevators should not be used during fire emergencies. Fire Protection Evaluation 87/0025 concludes that fire protection measures pertaining to elevators, as described in Appendix A, are intended to prevent fire propagation between floors by way of the elevator shaft. Elevators are not intended to play an active role during a fire emergency. Fire rated doors protecting elevator openings on Elevations 93'0", 133'0" and 166'0" will prohibit fire propagation between Turbine Building floors by way of Room 1T503.

REASON FOR CHANGE: It was discovered that walls designated as two-hour fire barriers had penetrations which were not provided with fire rated closures. This evaluation was performed to enable continued operation without installing the barriers.

SAFETY EVALUATION: The staircases serve as the escape routes and access routes during fire emergencies. The elevators are not intended to play an active role in fire emergencies. The fire-rated doors will prevent fire propagation between floors. The unprotected penetrations, therefore, will not be necessary. Operation in this manner will not pose an unreviewed safety question.

Serial Number: 93-154-NPE Document Evaluated: MNCR 86-0781

DESCRIPTION OF CHANGE: A stainless steel pin has been documented inside the reactor vessel. This evaluation addresses the possibilities of mechanical damage and flow blockage.

REASON FOR CHANGE: This evaluation was performed to allow continued operation with the stainless steel pin loose inside the reactor vessel.

SAFETY EVALUATION: The size of the pin allows an upper estimate of the possible flow blockage of 2%. The plant transient analyses were performed with a flow uncertainty of 2.8%, which envelopes damage from flow blockage. Due to the geometry and dimensions of the pin, it was determined that it could not be lodged in the control rod blade to cause malfunction. Finally, damage to the recirculation pump impeller is not expected due to the relatively large flow passages within the pump casings compared to the size of the pin. These considerations support a conclusion that operation with the pin in the reactor does not constitute an unreviewed safety question.

Serial Number: 93-156-NPE Document Evaluated: MNCR 86-1067
Interim #4

DESCRIPTION OF CHANGE: Damage recently discovered to Residual Heat Removal (RHR) to Containment Valve "A" has been attributed to throttling the valve beyond design limits. Restrictions have been imposed on RHR system flows that will prevent an excessive differential pressure across Valves A&B. Also, an upper bound on system flow is imposed to prevent RHR pump damage. An inspection of the "B" valve will be performed to determine if replacement is required.

REASON FOR CHANGE: This evaluation was performed to review the actions proposed to deal with the discovered condition. The proposed actions were to restrict RHR system flow to prevent valve damage and to place an upper limit on system flow to prevent pump damage.

SAFETY EVALUATION: The flowrate restrictions will not prevent RHR shutdown cooling mode operation nor RHR return to the upper containment, and will help ensure the operability of the subject valves, thus ensuring their ability to function as isolation valves. Therefore, the GGNS Technical Specifications are not affected. The flowrate limitations will only affect the RHR shutdown cooling and RHR return to the upper containment modes of operation, which are not required for accident mitigation. In addition, the restrictions will help prevent damage to the valve, which will ensure their function as a pressure boundary. Finally, the maximum flow limitation will help prevent damage to the pump. These actions will not increase the probability or consequences of any accident or malfunction, and will not decrease any margins of safety.

Serial Number: 93-157-NPE   Document Evaluated: MNCR 86-1090

DESCRIPTION OF CHANGE: Damage recently discovered to Residual Heat Removal (RHR) Shutdown Cooling Return to Feedwater Valves A & B has been attributed to throttling the valves beyond design limits. As a result restrictions were imposed on RHR system flows that will prevent an excessive differential pressure across the valves. Also, an upper bound on system flow is imposed to prevent RHR pump damage.

REASON FOR CHANGE: This evaluation was performed to review the actions proposed to deal with the discovered condition. The proposed actions were to restrict RHR system flow to prevent valve damage and to place an upper limit on system flow to prevent pump damage.

SAFETY EVALUATION: The flowrate restrictions will not prevent RHR shutdown cooling mode operation, and will help ensure the operability of the subject valves, thus ensuring their ability to function as isolation valves. Therefore, the GGNS Technical Specifications are not affected. The flowrate limitations will only affect the RHR shutdown cooling mode of operation, which is not required for accident mitigation. In addition, the restrictions will help prevent damage to the valve, which will ensure their function as a pressure boundary. Finally, the maximum flow limitation will help prevent damage to the pump. These actions will not increase the probability or consequences of any accident or malfunction, and will not decrease any margins of safety.

Serial Number: 93-161-NPE Document Evaluated: EAR M-353-92

DESCRIPTION OF CHANGE: Some fuel pool cooling and cleanup piping is routed in close proximity to containment ventilation exhaust monitors, and requires lead shielding to prevent monitor activation from radiation in the piping. The lead shielding will be placed at various locations in appropriate amounts according to the radiation levels.

REASON FOR CHANGE: The shielding is required to prevent unnecessary alarms from the monitors from radiation not contained in the monitored system. This evaluation reviews the use of the temporary lead shielding.

SAFETY EVALUATION: Stress analysis of the piping with the added weight of the shielding showed that the structural integrity of the pipes will be maintained. The temporary lead shielding will not result in any permanent changes to location, routing, or type of supports, nor does it alter any component performance characteristics, design parameters, or operational parameters of the affected system after the shielding is removed. This shielding presents no unreviewed safety question.

Serial Number: 93-162-NPE   ›cument Evaluated: Temp Lead Shielding

DESCRIPTION OF CHANGE: The Residual Heat Removal (RHR) piping in the Auxiliary Building requires lead shielding to reduce radiation exposure to personnel working in the area. The lead shielding will be installed in Operating Modes 4 and 5 only, and must be removed prior to restart. The lead shielding will be placed at various locations in appropriate amounts according to the radiation levels.

REASON FOR CHANGE: The shielding is required to reduce radiation exposure from RHR piping to personnel working in the area. This evaluation reviews the use of the temporary lead shielding.

SAFETY EVALUATION: Stress analysis of the piping with the added weight of the shielding showed that the structural integrity of the pipes will be maintained. The temporary lead shielding will not result in any permanent changes to location, routing, or type of supports, nor does it alter any component performance characteristics, design parameters, or operational parameters of the affected system after the shielding is removed. This shielding presents no unreviewed safety question.

Serial Number: 93-163-NPE Document Evaluated: Mech Standard MS-37

DESCRIPTION OF CHANGE: To assure long and reliable service from the Division I and II diesel generators, GGNS committed to the maintenance and surveillance requirements set forth in the Transamerica Delaval, Inc. DR/QR Report for GGNS. We also committed to utilize the 10CFR50.59 process to evaluate and implement any changes made to the maintenance and surveillance requirements specified in the DR/QR report, with some exceptions. Design Engineering (NPE) developed Mechanical Standard MS-37 to incorporate these requirements into a document controlled by Entergy Operations, under procedures, as required by ANSI 18.7. Since MS-37 was developed directly from the DR/QR program and since GGNS committed to evaluate and implement any changes to the DR/QR requirements under the 10CFR50.59 program, a 10CFR50.59 safety evaluation must be performed for all revisions made to MS-37.

REASON FOR CHANGE: MS-37 is controlled by NPE under the NPE administrative procedures, which requires only that a Safety Evaluation Applicability Review Form be completed, not that a 50.59 safety evaluation be performed. To assure that the commitment to utilize the 10CFR50.59 process to evaluate and implement any changes made to the maintenance and surveillance requirements specified in the TDI DR/QR report for GGNS is met, NPE is revising MS-37 to specifically state that a 50.59 safety evaluation is required for all revisions made to the standard.

SAFETY EVALUATION: The requirement to evaluate all changes to MS-37 by performing a 50.59 evaluation will not increase the probability or consequences of any accident or malfunction, and will not decrease any margins of safety. Increasing the required level of review is a conservative measure.

Serial Number: 93-164-NPE Document Evaluated: MNCR 91-0109

DESCRIPTION OF CHANGE: MNCR 0109-91 documents a fluid boundary failure of pressure switches N1N21-PSL-N204A and N1N21-PSL-N264A. These pressure switches control operation of the AC powered oil pumps for reactor feed pump turbine (RFPT) "A" by automatically starting the standby oil pump if the oil pressure reaches 222 psig and is decreasing. After an unisolable oil leak developed in the pressure switches, a clamp was placed on the oil supply line to the switches to stop the oil leak. The final MNCR disposition provides for reworking the oil line and replacement of the pressure switches.

REASON FOR CHANGE: Since the failed pressure switches were unisolable, the oil supply line had to be crimped in order to stop the oil leak. The RFPT trips on low oil pressure, therefore the loss of oil would have eventually resulted in a trip of the RFPT on low oil pressure. Non-radioactive, potentially oily drains from the RFPT area are collected in floor drain sumps and are pumped to the Radwaste Building for processing. Because of deleterious and, consequently, undesirable effects of oil on the Liquid Radwaste System processing components, it is good practice to minimize the amount of oil entering the liquid waste management system influent streams.

SAFETY EVALUATION: A pressure boundary failure of pressure switches N1N21-PSL-N204A and N1N21-PSL-N264A, will cause the standby AC powered oil pump to automatically start. Once the standby pump has been automatically started due to a sensed low oil pressure, the pump may either be manually shut down, or may be left in the operating mode. In the event of a false low oil pressure signal, operation of two AC powered oil pumps is not necessary in order to provide sufficient oil pressure to the reactor feed pump and to the reactor feed pump turbine. Operation of both oil pumps could increase the oil temperature due to the excess pump energy. Under manual control, with one pump running, a low lube oil pressure of 130 psig would be indicated in the Control Room by annunciation of alarm PAL-L652. Low oil pressure alarm annunciation at 130 psig provides some margin of time to manually start the second oil pump from the P680 panel in the Control Room since the RFPT does not trip until the oil pressure falls to 4 psig.

Upon implementation of the final disposition of MNCR 0109-91, the lube oil pressure switches and oil supply lines will be restored to the original design conditions.

The Condensate and Feedwater System serves no safety related function. Systems analysis has shown that failure of this system will not compromise any safety related systems or prevent safe shutdown of the reactor. The Condensate and Feedwater System is not required to effect or support the safe shutdown of the reactor or perform in the operation of reactor safety features. A single reactor feed pump may be removed from service while maintaining system operability with the remaining reactor feed pump.

Serial Number: 93-165-NPE Document Evaluated: MNCR 90-0197

DESCRIPTION OF CHANGE: MNCR 0197-90 documented non-conforming conditions found upon disassembly of safety relief valve Q1E21F018. This valve is a Lonergan LCT-20 1-1/2" x 2" screwed cap pressure relief valve. A later submittal requested approval and justification for operating the LPCS system during Modes 4 and 5 without Q1E21F018 and its associated piping spool piece installed.

This solution was determined acceptable only during Mode 5 with the reactor vessel head removed since it has been determined that with Q1E21F018 removed and a blind flange installed, the Low Pressure Core Spray (LPCS) System can perform all required design functions. Although the containment isolation function of Q1E21F018 and line 2"-HBB'18 cannot be met using this temporary solution, containment integrity is not required during the period applicable to this temporary solution.

REASON FOR CHANGE: This evaluation was necessary to review a non-conforming condition that was discovered on RHR Safety Relief Valve F018. The proposed action in response to this condition was to remove the valve and install a temporary blind flange in its place while in Operational Mode 5.

SAFETY EVALUATION: The intended safety function of this valve during Mode 5 is to act as a passive process pressure boundary. This flange will meet that intended requirement, and will satisfy all seismic requirements. This temporary arrangement will not constitute any unreviewed safety question.

Serial Number: 93-166-NPE Document Evaluated: MNCR 88-0199

DESCRIPTION OF CHANGE: The Fire Hazards Analysis describes the walls around Fire Zone OC110 to be 2-hour fire rated barriers maintained for good fire protection practice. MNCR 199-99 documents the fact that a portion of the subject walls are not constructed in such a way as to perform as a fire barrier. A fire protection evaluation concluded that the aforementioned walls are not required to be fire rated. Consequently, the walls will have the original 2-hour fire rating deleted.

REASON FOR CHANGE: This change is necessary to remove the 2-hour fire rating from walls around Fire Zone OC110. These walls were determined not to have been constructed to act as a fire barrier and an analysis has shown that they are not required to be fire-rated.

SAFETY EVALUATION: The fire protection evaluation has shown that these walls do not separate safety-related fire areas or safety-related or safe-shutdown components. Therefore, the deletion of the fire-rating for these walls will not increase the probability or consequences of an accident or malfunction and will not decrease any margins of safety.

Serial Number: 93-167-NPE Document Evaluated: MNCR 89-0199

DESCRIPTION OF CHANGE: Main Steam Line Drain Valves F119C and F120C are presently experiencing seat leakage. The valves are located on a manual drain line from the "C" train main steam line to a drain header. The valves are positioned in series to provide double isolation for the drain line. To reduce the seat leakage, the seats of the valves will be temporarily sealed using on-line sealant injection techniques. The sealant injection is made through an injection nozzle, which is inserted into the valve body using a th ded connection.

The change to the existing lesign is limited to the threaded installation of the sealant injection nozzle. The injection nozzle is designed to withstand the injection pressure as well as the design operating pressure of the reactor pressure boundary. The valve body is compatible with the injection nozzle for temporary installation.

The injection nozzle is to be installed on the shoulder of the socket welded piping connection on the inlet end of Valve B21-F119C. The injection nozzle location on the valves has been evaluated to verify that the valve body allowable stress limits are not exceeded due to the hole location. The sealant compound injection pressure has been limited to values less than the hydrostatic design pressure for the valve and connecting pipe. The sealant compound volume has been limited by the MNCR disposition such that excessive compound volume will not be injected into the valve. Since the sealant injection is being made on the main steam side of the valves, any sealant leakage past the valves will be into the equipment drains.

REASON FOR CHANGE: Seat leakage experienced on the above valves will be reduced by temporary sealing using on-line injection techniques. This evaluation is being performed to review the injection and necessary equipment installations.

SAFETY EVALUATION: After the installation of a sealant injection nozzle on the valve, the valve will still be within allowable stress limits, so the structural integrity of the valve is assured. The possibility of the nozzle becoming a missile was considered, but was found to be negligible. The weight of the sealant in the line and the injection nozzle will not adversely affect the structural integrity of the pipe. The drain lines are not included in the mitigation analysis for equipment important-to-safety, and they will not affect any equipment important to safety. The amount of sealant has been predetermined, and the sealant will migrate away from the reactor steam boundary. Due to the above considerations, the proposed change will not constitute an unreviewed safety question.

Serial Numbe 93-169-NPE Document Evaluated: UFSAR CR
NPE 89-031

DESCRIPTION OF CHANGE: A new analysis of Fuel Handling Accidents (FHAs) inside the Auxiliary Building and the containment has been prepared which incorporates the following topics:

1. The consequences of dropping non-fuel objects onto the core during refueling,

2. A revised analysis for the drop of a fuel bundle onto the core to include a more conservative figure for the maximum height of a fuel bundle over the core,

3. An analysis for loads handled over spent fuel storage areas with a greater potential energy than that of a fuel assembly and its associated handling tool at normal handling height.

The offsite dose consequences of this analysis are in excess of the current discussion in UFSAR Section 15.7.4, yet are still well within the guidelines of 10CFR100.

REASON FOR CHANGE: This evaluation was performed to review the results of a revised analysis of fuel handling accidents. The previous analysis was revised to include new topics listed above.

SAFETY EVALUATION: The new topics that were considered will not result in any handling system being subjected to loads greater than its design capacity, and no changes to the design specifications of any load handling systems were proposed. The possibilities of various accident scenarios were reviewed; core criticality accidents, spent fuel pool criticality, and piercing the liner of the spent fuel pool were all reviewed and were not considered to be increased. The potential consequences of accidents or equipment malfunctions were also reviewed. It was determined that while the potential consequences are greater than those currently discussed, they were still well within 10CFR100 limits, and thus are not considered increased. The revised analysis did not indicate an increased probability of any accidents or equipment malfunctions, and did not indicate a reduction of any margins of safety. The potential consequences of accidents or equipment malfunctions were not considered increased, and no margins of safety will be reduced.

Serial Number: 93-170-NPE Document Evaluated: UFSAR CR 91-023

DESCRIPTION OF CHANGE: Revise wording in UFSAR Section 8.2.4 concerning 115 kV line capacity computer and annunciator alarms which are discussed, but not presently in the system. The UFSAR revision will reflect plant conditions as tested and "as-built".

REASON FOR CHANGE: To correct wording concerning computer and annunciator alarms being sounded if the 115 kV line capacity decreases below the UFSAR stated limit, while no such alarms exist in the plant. It was determined during startup test that the 115 kV Port Gibson line has sufficient capacity to start and operate the required loads with either the South Vicksburg or Lorman line out of service. Consequently there was no need for a capacity monitoring system for the 115 kV line, however this was not updated in Section 8.2.4 of the UFSAR.

SAFETY EVALUATION: UFSAR Section 8.2.4 discusses system operating limits for the 115 kV lines. Paragraph 2 (Bases for the selection of the operating limits), Part 3 discusses the system capacity. It notes that the 115 kV line has a minimum capacity decreases below this limit. The 115 kV Port Gibson line was tested with the South Vicksburg line (the stronger of the two lines feeding the Port Gibson substation), out of service and being fed by the Lorman line only. It was found that the line has sufficient capacity to start and operate the required loads for Unit 1 at GGNS. A line monitoring system was originally designed and purchased but after test was conducted, it was determined that the system would not be needed because the 115 kV Port Gibson line has the required capacity. The MP&L dispatchers monitor line integrity and load flows rather than capacity. All switching done on the line by MP&L is cleared with GGNS operators. GGNS operators are kept informed of the line status by MP&L dispatchers verbally. Operation of the 115 kV line will not be changed as a result of this change. In addition, no new interfaces or capabilities are created by this change. Thus, equivalent capability has been established by test and no unreviewed safety question has been introduced.

Serial Number: 93-171-NPE   Document Evaluated: B33F060A,B
Flow Control Valve

DESCRIPTION OF CHANGE: The Recirculation Flow Control Valve (FCV) has an interlock installed to prevent system startup or transfer from 25 percent (slow recirculation pump speed) to 100 percent (fast recirculation pump speed) unless the valve is in the minimum position. The objective of this interlock is to prevent scrams due to a rapid flow increase resulting from an operator failure to close the FCV prior to the start of speed transfer. The FCVs presently will not move from the minimum position after the recirculation pumps have been switched from slow speed to high speed. This temporary alteration will bypass the above described interlock to allow manual pre-positioning the FCVs up to an additional 7 percent open prior to the start of speed transfer of the recirculation pumps.

REASON FOR CHANGE: The FCVs presently will not move from the minimum position after the recirculation pumps have been switched from slow speed to high speed. This evaluation reviews a bypass of the interlock to allow manual pre-positioning of the FCVs up to an additional 7% open prior to the start of speed transfer of the recirculation pumps.

SAFETY EVALUATION: Switching the speed at 7% above the existing minimum flow will create the possibility of a recirculation flow increase of up to 17%. The switch at a higher flow may also increase the potential neutron flux transient. Both of these increases, however, are bounded by existing analyses in the UFSAR, and are not considered to be unreviewed safety questions. The margins of safety will remain substantially higher than the safety limit.

Serial Number: 93-175-NPE Document Evaluated: Mech Standard MS-38

DESCRIPTION OF CHANGE: Revise SERI-MS-38, Revision 0 to delete the requirements for the testing frequency of compressed air systems dryer assembly instruments and dew point monitors. The standard was also revised to change the document number from a SERI designation to a GGNS designation.

REASON FOR CHANGE: EER 91/6378 requested that the revision be made since plant repetitive tasks are being revised per the Reliability Centered Maintenance (RCM) program. The requirements of SERI-MS-38 conflict with the RCM designated calibration frequency. The document designation for the standard was also changed from a SERI identifier to a GGNS identifier in accordance with current NPE administrative procedures.

SAFETY EVALUATION: AECM-89/0032 was written in response to Generic Letter 88-14. The AECM committed to develop a GGNS instrument air system quality standard based on overall system and component needs. SERI-MS-38, Revision 0, "Quality Standard for Instrument Air System and for Diesel Generator Starting Air" was developed to meet the AECM commitment for an instrument air system quality standard.

SERI-MS-38, Revision 0 established the requirements for compressed air monitoring frequency and preventative maintenance requirements. The revision to SERI-MS-38, Revision 0, to GGNS-MS-38, Revision 1 maintains all previously established standards for the quality of air based on dew point, oil content, particulates, and contaminants. GGNS-MS-38, Revision 1 also maintains all previously established sample frequencies and operational surveillances for monitoring the quality of air.

Serial Number: 93-177-NPE Document Evaluated: QDR 92-0296

DESCRIPTION OF CHANGE: QDR 0296-92 documents the failure of CFR 91/00263R01 to recognize and evaluate the impact of DCP 91/0026-3 on the requirements of Emergency Procedure (EP) 4 - 'Auxiliary Building and Radioactive Release Control'.

DCP 91/0026-3 added a two minute time delay to the low negative pressure alarm function of annunciator 1T42-L608, deleted the high negative pressure alarm function of this annunciator and modified the annunciator wording to show that only the low negative pressure alarm remained. As stated in the QDR, these changes were not evaluated with respect to EP-4 which references the 1T42-L608 annunciator for an entry condition.

Of the changes made to the annunciator, only the addition of the time delay would affect the operator response required by EP-4. This change would delay entry to EP-4 in the event of a loss of normal negative pressure in the fuel handling area.

REASON FOR CHANGE: No change, test or experiment is proposed in the NPE disposition of QDR 0296-92. No change to the as-built configuration of the subject annunciator is proposed.

SAFETY EVALUATION: The T42-L608 annunciator is used to indicate a loss of normal negative pressure in the fuel handling area. If negative pressure were lost, an unmonitored release from the fuel handling area could occur. However, for a loss of building negative pressure to create an immediate radiological hazard, the radiation levels within the fuel handling area would have to be initially hazardous. This condition would itself require EP-4 entry on high FHA HVAC exhaust radiation indications. Therefore, the time delay added to the pressure annunciator circuit would not delay the EP-4 emergency response to conditions which would require immediate operator action.

If the fuel handling area radiation were not above the alarm points of the FHA vent exhaust or the FHA pool exhaust monitors, no immediate radiological hazard would be created by a loss of building negative pressure. In this case, EP-4 entry after 2 minutes would not excessively delay the subsequent monitoring activities required by the EP.

The as-built configuration of the T42-L608 annunciator circuit is acceptable. This annunciator will still function to prevent a long term unmonitored release and with the addition of the time delay, it will also eliminate spurious alarms caused by normal pressure transients in the fuel handling area.

Serial Number: 93-178-NPE Document Evaluated: MNCR 93-0165

DESCRIPTION OF CHANGE: This evaluation addresses the safety significance of power operation in Cycle 7 and beyond with several small, unsecured foreign objects remaining in the reactor vessel. These are primarily metal items such as nuts, wire, parts of broken tools, and small pieces of the damaged Jet Pump #10 assembly which were located in the vessel annulus region during RF06 inspection activities. The exact parts which have been observed and their locations are documented in MNCR 0165-93, Supplements 2 and 3 (Reference 13).

REASON FOR CHANGE: The shapes and positions of the objects are such that their retrieval would be very difficult. Some attempts at recovery have been made, however further efforts would prove very time consuming and could even result in unintentionally displacing the objects to other areas of the vessel which are less benign in terms of safety and operating effects. Thus, operation with the loose parts remaining in the vessel is under consideration.

SAFETY EVALUATION: Evaluation of the loose parts shows that the identified parts do not present a safety concern for operation in Cycle 7 and beyond. The primary safety concern with leaving the parts in the vessel is the potential for local flow blockage of a fuel assembly. This has been determined not to be significant for the particular items being evaluated. Other safety considerations include possible control rod interference, plugging of jet pumps and damage to the recirculation system. For the items of concern, none of these issues could impact nuclear safety by increasing either the probability or consequences of accidents or malfunctions. Leaving the items in the vessel does not create a new type of event which has not been evaluated, and margins of safety are preserved. Thus, the presence of the identified loose parts does not involve an unreviewed safety question. Technical specifications do not require modification as a result of the items remaining in the vessel.

The presence of some of the smaller objects, such as the wire, do present the remote possibility that fretting wear of a fuel rod could occur, resulting in fuel failure. Existing monitoring requirements would provide clear indication of any such fuel failure prior to coolant or offgas radioactivity levels exceeding technical specification limits. While there is some commercial risk in operating with such parts in the vessel, no increase in nuclear safety risk is involved.

Serial Number: 93-179-PLS Document Evaluated: 08-S-03-10 and 08-S-04-950

DESCRIPTION OF CHANGE: The Post Accident Sampling System (PASS) drywell atmosphere sample pathway will be isolated and disconnected to allow for repairs for an extended length of time. This PASS sample pathway will be unavailable for use until repairs are completed.

REASON FOR CHANGE: As a result of the latest integrated leak rate test (ILRT) it was identified that the sample pathway from the drywell to the post accident sample panel was leaking excessively.

Subsequent review after completion of the ILRT indicated the need to isolate this pathway in order to be in compliance with Technical Specification 6.8.3.a.

The isolation of this sample pathway for an extended period of time is necessary in order to allow adequate planning for and to obtain the materials necessary for making repairs. This will remove the PASS drywell atmosphere sampling line from service and will preclude the capability to draw a sample through PASS of the drywell atmosphere. This deviates from the described conditions in UFSAR Section 9.3.2.2.3 and Table 9.3-3s.

The net effect of this condition will not prevent the ability to draw an atmospheric sample from containment, but may impede the core damage assessment for a reactor accident when using PASS atmospheric samples due to the inability to sample the drywell atmosphere directly. Other sample pathways will remain available to allow core damage assessments to be performed as required by GGNS licensing bases.

SAFETY EVALUATION: The Post Accident Sampling System provides no safety function or accident mitigation functions. The isolation of a single PASS sample pathway has no impact on any described accident and is not the initiator of any new accident. The results of this evaluation concludes that there is no unreviewed safety question created.

Serial Number: 93-180-NPE Document Evaluated: TSTI 1B21-93-012-O-S

DESCRIPTION OF CHANGE: The mode 1/2 TSTI is written to allow testing during power operation. During testing, several reactor trip channels will be inhibited to prevent inadvertent trips, and enabled after testing. While the trip channels are inhibited, appropriate LCOs will be entered. Mode 2 testing will be used to confirm backfill performance during heat up and SRV testing. The testing will be performed on one reference leg with the most historically sensitive instruments, D004C and HPCS will be put in "Maintenance" during the SRV testing to prevent inadvertent actuation because it receives start signals from multiple channels. Change to Mode 1 will occur with D004C backfill in service. At 100% power, the remaining backfill lines will be put in service one at a time with trips inhibited while monitoring transmitter signals. If no unexpected anomalies occur, trips will be restored.

REASON FOR CHANGE: The mode 1/2 TSTI is written to monitor the effects on RPV level instruments by the changing conditions of coming up on power. Instructions are also provided for inhibiting various trip channels which could trip spuriously due to a disturbance on the RPV reference legs. Testing completed during Mode 4 envelopes the conditions of Mode 1/2. Concern for system operation is related to failure resulting in high flow conditions, which would result in thermal and cyclic stresses greater than those analyzed. Testing during Mode 4, with the reactor shutdown, provided the highest DP possible, since reactor pressure was at zero psi. This provides the highest possible purge flow rate, which was measured to be within the established acceptance criteria. System operation during power operation is completely enveloped by testing during Mode 4. No credible failure mode exists which could result in flow rates greater than those measured during Mode 4 testing. Thus the possible RPV level instrument perturbations are also bounded by the Mode 4 test results.

SAFETY EVALUATION: The Mode 1/2 TSTI will operate the RVLIS under conditions which were bounded by Mode 4 testing. Although certain reactor trip channels will be temporarily inhibited during implementation of this TSTI, appropriate LCOs will be initiated. Therefore, no system will be operated in an abnormal manner, or in a manner not described by the FSAR.

Serial Number: 93-181-PSE Document Evaluated Temp Alt 93/0025

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, [illegible] particles and other impurities from water being transferred into the equipment drain sample tanks for make up to condensate storage tank (CST).

This temporary alteration 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 the hoses 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&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. Chemistry sampling of equipment drain sample tanks will be taken prior to any transfer of tank contents.

A bypass line will be provided to bypass flow around the filter.

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 (SF17A003A&B) respectively. A design change package will be pending on the effectiveness of this filter.

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SAFETY EVALUATION: The routing of hoses and location of the new filter and valves will be contained within the same area/rooms as existing piping that have been analyzed by the FSAR (all new spill/releases mechanisms by temporary alteration are bounded by existing analysis). The hoses, filter and valves that will be installed per this temporary alteration are rated above system design FSAR 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. The safety evaluation of this temporary alteration concludes that there are no unreviewed safety questions.

Serial Number: 93-182-PSE Document Evaluated: Temp Alt 93/0024

DESCRIPTION OF CHANGE: This safety evaluation assesses the effect of installation of Temporary Alteration 93/024 on plant safety. Temporary Alteration 93/024 caps and isolates the Post Accident Sampling System (PASS) drywell atmosphere sample line from the Fission Products Monitoring System drywell atmosphere sampling line.

REASON FOR CHANGE: The fission products monitor is required to be operable and capable of providing drywell atmosphere radioactivity monitoring per Technical Specification 3.4.3.1b. Leakage in the Post Accident Sampling System (PASS) drywell atmosphere sample line was detected during the recent ILRT. The temporary alteration is required to maintain the fission products monitor operable while troubleshooting is performed to pinpoint and correct the Post Accident Sampling System (PASS) drywell atmosphere sample line leak.

SAFETY EVALUATION: The safety evaluation concluded that Temporary Alteration 93/024 did not require a change to GGNS Technical Specifications or represent an unreviewed safety question.

Serial Number: 93-183-NPE Document Evaluated: SCN-93-0002 to GGNS-M-141.1

DESCRIPTION OF CHANGE: The SCN for Specification GGNS-M-141.1, Revision 37 will update the valve data sheets (Appendix F) to correct the information determined form EERR 92/6090 Interim #1 through Interim #23 and Interim #25 through Interim #28. In addition the SCN will add a section to each valve data sheet that provides the ability to include design basis references and will add P&ID information as required. Although not part of the SCN it was determined that the technical information for some relief valves identified in the specification was incorrectly reflected in the FSAR. This document will also evaluate the changes necessary to correct the FSAR.

REASON FOR CHANGE: EERR 92/6090 Interim responses were issued to resolve discrepancies identified in the EER, NPE Engineering Report Number 90/0025 and SIMS Database information for safety relief valves. In resolving these discrepancies it was determined that some of the information on the valve data sheets (Appendix F) of Specification GGNS-M-141.1 Rev. 37, "Design Specification For Pressure Relief Valves-Nuclear Service", required changes as a result of the review. The reasons for specific changes to the data in Appendix F of the Specification are as follows:

Valve	Reason
P41F061	The technical data being changed was incorrectly transferred into Revision 35 of the Specification
E21F018	The valve setpoint does not accurately reflect the elevation differential as required by the GE Specification
C11F029A,B	Cold differential test pressure does not compensate for back pressure
E12F029A,B	Cold differential test pressure does not compensate for temperature
B21F125J	Cold differential test pressure does not compensate for temperature
B21F125A-H	Cold differential test pressure does not compensate for temperature

B21F124A-H	Cold differential test pressure does not compensate for temperature

The following are the changes to the relief valve technical information presented in the FSAR required to be changed but not as a direct result of the SCN:

Valve	FSAR Section/Table	Reason
E12F017A,B,C	Table 5.4-2a	The setpoint as presented in the FSAR does not take into consideration the design pressure of the jockey pump casings or elevation of relief valve
E12F055A,B	Table 5.4-2a	Capacity of valves is not correctly reflected. The correct capacity is indicated in Section 5.4.7.1.3 of the FSAR
E51F090	5.4.6.2.2.2 d	Piping design pressure is 1500 psig. valve is used to protect piping

SAFETY EVALUATION: The changes to Specification GGNS-M-141.1 and the FSAR will not affect any present or impose any new Technical Specification requirements. It has been determined that the changes to Specification GGNS-M-141.1 and the FSAR will not result in an increased probability for failure of the relief valves or their associated piping or equipment. The changes will not result in the failure of any safety related component or system to perform its intended function as previously analyzed or introduce any new failure modes to any equipment or systems.

Serial Number: 93-184-PLS Document Evaluated: Part 20 TS Changes

DESCRIPTION OF CHANGE: The Standards for Protection Against Radiation (10CFR20) have been revised by the Nuclear Regulatory Commission (NRC) with mandatory compliance as of January 1, 1994. This safety evaluation is limited to the impact of the necessary changes on the GGNS FSAR. The following FSAR changes are necessary to implement the revised 10CFR20 at GGNS:

- Changes involving changes to references to revised and new regulatory guides and 10CFR to reflect revisions to these documents.

- The proposed change in FSAR Section 12.3.1.2 deletes part of the definition of a radiation area in sentence eight (8) an adds the "Locked High Radiation Area" posting in Sentence 12.

- The proposed change in FSAR Section 12.5.3.3 changes the term "restricted area" to "protected area" deletes Potential Airborne Radioactivity Areas and adds Locked High Radiation Areas in Sentence 3.

- The proposed change in GGNS FSAR Section 12.5.3.5 changes the reference to Appendix B, Table I, Column 1 of 10CFR20, in Sentence 1, to 10CFR20 Subpart C; changes the airborne radioactivity limit, in Sentence 2, from 25% to 30% and changes the reference to 10CFR20.103, in Sentence 7, to 10CFR20 Subpart H.

- The proposed change to FSAR Figure 12.5-1 deletes the Safety Coordinator and changes the other second level supervisor positions to Radiation Control Supervisor(s) and Health Physics Coordinator(s).

REASON FOR CHANGE: These changes are necessary to reflect compliance with the revised 10CFR Part 20.

SAFETY EVALUATION: The changes to references to new and revised regulatory guides and 10CFR are administrative and do not affect plant hardware or operation, and will neither create the possibility of a new or different kind of accident from any previously evaluated nor create new radiological hazard to plant staff or the public.

The changes to the definitions of radiation area, very high radiation area, and locked high radiation area are administrative changes which are necessary to implement the requirements of the revised 10CFR20. Since these changes are administrative and do not affect plant hardware or operation, these changes will

neither create the possibility of a new or different kind of accident from any previously evaluated nor create a new radiological hazard to the plant staff of public.

Regulatory Guides 8.34 and 8.36 were written to provide guidance that may be used by licensees to develop procedural methods to implement the requirements of the revised 10CFR20. Incorporation of these regulatory guides is an administrative change necessary to implement the requirements of the new 10CFR20 and does not reduce the existing requirements. Since this change is administrative and does not affect plant hardware or operation, this change will neither create the possibility of a new or different kind of accident from any previously evaluated nor create a new radiological hazard to the plant staff or public.

Redefining the protected area as the restricted area, an "area access to which is limited or controlled to protect individuals from exposure to radiation or radioactive materials" and changing the term "restricted area" to "protected area" are administrative changes which provides consistency with GGNS administrative procedure and the revised 10CFR20. Revising the VHRA posting and adding locked high radiation area are administrative changes which will not decrease the ability of GGNS radiation protection programs to provide control of exposure from external sources in restricted areas. These administrative changes are necessary to implement the requirements of the revised 10CFR20. Since these changes are administrative and do not affect plant hardware or operation, these changes will neither create the possibility of a new or different kind of accident from any previously evaluated nor create new radiological hazard to plant staff or the public.

The proposed change of the posting limit from 25 percent to 30 percent will maintain consistency and compliance with 10CFR20. These administrative changes are necessary to implement the requirements of the revised 10CFR20. Since these changes are administrative and do not affect plant hardware or operation, these change will neither create the possibility of a new or different kind of accident from any previously evaluated nor create a new radiological hazard to plant personnel or the public.

The proposed organizational changes are administrative. The Safety Coordinator was reassigned to the Manager, Operations and the other changes amount to title changes brought on by the Span and Control process. The functional areas of Health Physics will continue to receive the same level of management oversight. These changes do not affect any plant hardware or operation and will neither create the possibility of a new or different kind of accident from any previously evaluated nor a new radiological hazard to plant personnel or the public.

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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.

DCP 89/0034-01, Revision 0 provides the design and installation of additional intrusion detection monitoring equipment in alarm stations and the Control Room. Additionally, a new panel will be installed in the Control Building to route video information to all alarm stations. The added raceways require opening of existing penetrations in the Control Room envelope and existing fire rated penetrations in the Control Building. The new signal processing equipment will require that some rooms be totally reworked to replace all present security monitoring equipment.

The CNs address FSAR figure changes to reflect changes associated with this DCP to Uninterruptable Power Supply Breakers.

REASON FOR CHANGE: These changes were initiated as part of a 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 DCP will 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 which is non-safety related and whose function will not change due to the equipment and monitoring capabilities added by this DCP.

All cabling and raceway modifications will be performed in accordance with the separation requirements of Regulatory Guide 1.75 and, where required, seismic supports have been provided to preclude creation of any Seismic II/I concerns. Opening of raceway penetrations in the Control Room envelope and fire rated penetrations in the Control Building have been evaluated for impact on the Control Room environment and the fire rated barriers. Penetration requirements have been provided to ensure the Control Room envelope and fire rated penetration boundaries are maintained.

Combustibles are added to some rooms, however the fire duration in those rooms remains considerably below the 3-hour rated fire area boundaries.

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The carpet installed was evaluated for its effects on our fire load evaluations. Fuel contribution is taken into account by the combustible heat load calculation. The most stringent rating for interior floor finishes requires a minimum critical radiant flux (CRF) rating of .45 W/cm2. The carpeting installed has a CRF rating of .92 W/cm2, which is considerably better than the minimum required and is therefore acceptable. A maximum rating of 450 is prescribed for smoke development for interior finishes. The carpeting installed has a smoke development rating of 138, which is considerably better than the maximum required and is therefore acceptable.

No evaluated accident is affected by any change to the Security Monitoring System, the addition of combustibles, or the addition of the remote facility. The components of the Security System to be changed by this DCP 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.