Entergy
Operations

Entergy Operations, Inc.

W. T. Cottle

June 4, 1991

U.S. Nuclear Regulatory Commission
Mail Station P1-137
Washington, D.C. 20555

Attention: Document Control Desk

Subject: Grand Gulf Nuclear Station
Unit 1
Docket No. 50-416
License No. NPF-29
Report of 10CFR50.59 Safety Evaluations - June 1, 1990 through December 31, 1991

GNRO-91/00091

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 June 1, 1990 through December 31, 1990. A summary of these changes, tests, and experiments is contained in the attachment.

It has been the practice of Entergy Operations, Inc. to submit the 10CFR50.59 reports semi-annually. In accordance with 10CFR50.59(b), Entergy Operations, Inc. will in the future submit the reports on an annual basis. Annual reports covering the safety evaluations for GGNS for each calendar year will be submitted prior to July 1st of the following year.

Yours truly,

W T Cottle

WTC/GWR/ams
attachment:
cc: (See Next Page)

G9105311/SNLICFLR - 1

9106110325 910604
PDR ADOCK 05000416
R PDR

IE47

cc: Mr. D. C. Hintz (w/o)
Mr. J. L. Mathis (w/o)
Mr. R. B. McGehee (w/o)
Mr. N. S. Reynolds (w/o)
Mr. H. L. Thomas (w/o)
Mr. F. W. Titus (w/a)

Mr. Stewart D. Ebneter (w/a)
Regional Administrator
U.S. Nuclear Regulatory Commission
Region II
101 Marietta St., N.W., Suite 2900
Atlanta, Georgia 30323

Mr. L. L. Kintner, Project Manager (w/a)
Office of Nuclear Reactor Regulation
U.S. Nuclear Regulatory Commission
Mail Stop 11D21
Washington, D.C. 20555

bcc:

Mr. R. W. Byrd (w/o)
Mr. L. F. Daughtery (w/o)
Mr. M. A. Dietrich (w/o)
Mr. J. O. Fowler (w/o)
Mr. W. K. Hughey (w/o)
Mr. C. R. Hutchinson (w/o)
Ms. F. K. Mangan (w/o)
Mr. M. J. Meisner (w/o)
Mr. G. W. Muench (w/a)
Mr. D. L. Pace (w/o)
Mr. T. E. Reaves, Jr. (w/o)
Mr. J. L. Robertson (w/o)
Mr. G. W. Rogers (w/2)
Mr. M. J. Wright (w/o)
Mr. G. A. Zinke (w/o)
File (LCTS) (w/a)
File (Hard Copy) (w/a)
File (RPTS) (w/a)
File (NL) (w/a)
File (Central) (w/a) (215)

TABLE OF CONTENTS

OF

10CFR50.59 SAFETY EVALUATIONS FOR THE PERIOD

JUNE 1, 1990 THROUGH DECEMBER 31, 1990

TABLE OF CONTENTS

OF

10CFR50.59 SAFETY EVALUATIONS FOR THE PERIOD

JUNE 1, 1990 THROUGH DECEMBER 31, 1990

TABLE OF CONTENTS

OF

10CFR50.59 SAFETY EVALUATIONS FOR THE PERIOD

JUNE 1, 1990 THROUGH DECEMBER 31, 1990

TABLE OF CONTENTS

OF

10CFR50.59 SAFETY EVALUATIONS FOR THE PERIOD

JUNE 1, 1990 THROUGH DECEMBER 31, 1990

SRASN: NPE-90-022 DOC NO: DCP-88-0005-S00-R00 SYSTEM: G36

DESCRIPTION OF CHANGE: This change replaced the Reactor Water Clean-Up (RWCU) resin metering pump with a new larger capacity pump equipped with a flow monitoring sight glass. A new piping and backwash system was also installed.

REASON FOR CHANGE: The previous peristaltic pump experienced frequent ruptures of the Tygon Tubing, and was not of adequate capacity. The new RWCU resin metering pump provides higher capacity and greater flexibility with precoat resin injection rates. The new system includes backwash capability on the resin pump suction and discharge lines for cleaning purposes.

SAFETY EVALUATION: The safety evaluation concluded that the change did not involve an unreviewed safety question. The pump, piping, supports, and associated equipment replaced by this change are non-safety related, seismic Category II/I. This change meets all requirements of the original RWCU resin metering system and does not compromise any safety related systems or components or prevent a safe reactor shutdown. No safety related circuits or interfaces are added or affected by this change. Operation of the RWCU resin metering pump and associated equipment are not required to mitigate the consequences of an accident.

The RWCU resin metering pump or associated equipment are not addressed by the GGNS Technical Specifications, nor does any of this equipment impact the margin of safety of any systems addressed in the Technical Specifications.

SRASN: NPE-90-023 DOC NO: MNCR-90-0083 SYSTEM: E22

DESCRIPTION OF CHANGE: This evaluation identified a condition in which the accident load profile for the Division III batteries exceeded the profile described in FSAR Table 8.3-8.

The current load profile is:

≥76 amperes for the first 60 sec.,
≥16 amperes for the next 59 min.,
≥18 amperes for the last 60 min.

As a result of efforts to review the design basis of the electrical systems, the Division III battery load profile was revisited and installed as-built loads were calculated. The resulting minimum required test profile, with margin built in, was calculated to be:

≥65 amperes for the first 60 sec.,
≥20 amperes for the next 59 min.,
≥20 amperes for the last 60 min.

REASON FOR CHANGE: Calculations were performed to ensure the batteries capacity to deliver the energy. These calculations use the methodology presented in IEEE 485-1978 and shows that the existing battery is sized adequately to deliver the required energy.

The FSAR change is a change to the profile presented in FSAR Table 8.3-8 and section 8.3.2.1.7.2.

SAFETY EVALUATION: This safety evaluation concluded that the change did not involve an unreviewed safety question. The change being evaluated is a revision of the load profile to reflect the actual emergency loads imposed on the Division III batteries. This change does not reflect a physical hardware change to the facility, but imposes the proper requirements on the existing battery system. Calculations have been performed in accordance with IEEE 485-1978 which ensure the capability of the existing battery banks to meet the newly calculated load profile. This industry standard is the governing document for determination of battery sizing. Compliance with this standard ensures that the battery system can perform its intended function.

The load profile was developed from the postulated design basis accident scenario and the capability of the battery bank was determined using IEEE 485-1978 battery capacity methodology. This ensures that the new profile is greater than the actual emergency load and that the installed battery system is sized properly to carry the load. No change is being made to the installed plant hardware and the capabilities of the existing hardware to meet the design requirements have been verified.

With the imposition of the load profile specified, additional margin to actual accident load profiles has been included. This additional load value has been verified to be within the capabilities of the installed hardware and above the load imposed on the batteries by the postulated accident scenario. The margin provided by the current technical specification surveillance has been increased by the profile provided. Therefore, the margin of safety as defined in the basis for any technical specification will not be reduced.

SRASN: NPE-90-024 DOC NO: DCP-85-4007-S00-R00 SYSTEM: N71

DESCRIPTION OF CHANGE: This change provided the design change necessary to install a Condenser Tube Cleaning System (CTCS) on the Circulating Water (CW) system which provides an on line cleaning method for the condenser tubes. The operational principle for the CTCS is to continuously inject sponge cleaning balls into the CW flow on the inlet side of the LP condensers and to collect the cleaning balls from the CW flow on the discharge side of the HP condensers. The cleaning balls are designed to be randomly distributed throughout the CW Flow. A control panel will provide annunciation in the control room for specified CTCS malfunctions. The CTCS control panel, recirculation pump, and ball collector tank are to be located in an area of the Turbine Building which is accessible during normal plant operation.

Prior to operation of the CTCS, the condenser tubes were cleaned to remove excessive tubeside fouling, in order to achieve free tube passage for the CTCS cleaning balls. The tubes were cleaned during the second refueling outage by implementing a NALCO chemical cleaning process which requires the circulation of tannin solution, sulfuric acid, citric acid, and iron dispersants through the tubes. NALCO representatives provided continuous coverage during the cleaning process. The cleaning process has been laboratory tested by Entergy to ensure that the process is benign to the materials of construction used in the condensers and CW piping components. The waste water generated by the process was transported to the Unit 2 cooling tower basin for storage until such time as approval had been granted by the Mississippi Dept. of Natural Resources for discharge to the environment. GGNS Operating License Condition No. 2.c.(27) contains a provision which prohibits filling the Unit 2 cooling tower basin. The NRC has been contacted regarding the discharge of waste water into the Unit 2 cooling tower basin. The NRC response stated that the operating license condition does not apply since Unit 2 is not in operation.

REASON FOR CHANGE: This change provided for an automatic CTCS on the CW system to prevent fouling and thus reduce flow restriction and improve heat transfer.

SAFETY EVALUATION: There is no increase in the probability of occurrence or in the consequences of an accident or malfunction of equipment important to safety previously evaluated in the Safety Analysis Report. CW system accidents previously evaluated in the FSAR are limited to the potential flooding of safety related equipment due to the failure of a CW system component. Two possible sources of CW system failure have been previously identified as: 1) failure of the expansion joints, and 2) failure of the butterfly valves. CTCS installation will add an additional source by the installation of flanged strainer sections on the CW piping discharge lines inside the Turbine Building. The flanged connections are designed to withstand 95 PSIG which exceeds the 90 PSIG design pressure of the Condensers and far exceeds the CW pump maximum shutoff pressure of approximately 66.1 PSIG. The added weight of the CTCS strainer sections does not exceed the adequacy of the existing supports per the applicable calculation. The CTCS recirculation piping is designed per ANSI B31.1 Power Piping Code requirements to withstand 150 PSIG to compensate for the added pressure required to reinject the cleaning balls into the CW flowpath at the LP condenser inlet. CTCS installation does not install new safety related equipment, alter the location of existing safety related equipment, or add volume to the CW system fluid inventory. Implementation of the design change will provide an enhancement to the CW systems ability to maintain design backpressures inside the main condensers.

The CW system serves no safety function. Systems analysis has shown that failure of the CW system will not compromise any safety-related systems or prevent safe reactor shutdown.

Installation of the CTCS will enhance the ability of the main condenser to maintain design backpressure and will enhance the reliability of the condenser tubes by reducing the possibility of pitting corrosion. Implementation of the change will not adversely affect the functional characteristics of the CW system. CW system component fluid boundary failure has been previously evaluated in the FSAR and no additional modes of failure are postulated. Therefore, there is no creation of a possibility for an accident or malfunction of a different type than any evaluated previously in the Safety Analysis Report.

The design bases for the CW system as defined in the GGNS Technical Specifications does not contain provisions for any specified margins of safety regarding the failure of a CW system component. Therefore, implementation of the design change does not reduce the margin of safety as defined in the basis for any Technical Specification.

SRASN: NPE-90-025 DOC NO: MCP-89-1042-S00-R00 SYSTEM: G17

DESCRIPTION OF CHANGE: This change removed the legend plates from the listed annunciator windows and replaced them with blank mylars. The alarm cards were permanently pulled and so noted on all of the associated drawings.

Liquid Radwaste Filters Trbl. Ann./SG17-UA-L602
Floor Drain Waste Evap. Trbl. Ann./SG17-UA-L604
Solid Radwaste Sys. Trouble Ann./SG18-UA-L600
CNDS Coll. Tk. Level High-High Ann./SN12-LAHH-L610
CNDS Rtn. Stg. Tk. Level High Ann./SN12-LAH-L658
Chorin. System Trouble Alarm Ann./SN72-UAHL-L600
Makeup Wtr. Trtmt. Sys. Tbl. Ann./SP21-UA-L601

REASON FOR CHANGE: The annunciators listed are connected to non-safety related equipment which is not being utilized with the exception of the liquid radwaste alarms. These alarms are provided in the Radwaste Control Room and are not required to be in the Main Control Room. These annunciators are not required per IEEE 279 and there is no requirement for the annunciators for equipment protection. The alarm cards were removed under an Operations Nuisance Annunciator Program.

SAFETY EVALUATION: The safety evaluation concluded that the change did not involve an unreviewed safety question. These annunciators serve no safety function or support equipment important to safety. Failure of these annunciators will not compromise any safety related system or component and will not prevent safe reactor shutdown. There is no probable accident associated with this equipment. The annunciators are not connected to equipment which is related to any plant safety function. Disabling these annunciators creates no new failure modes not already enveloped by present FSAR analysis.

These annunciators are not addressed in any Technical Specification nor are they essential in monitoring the plant for compliance with the Technical Specifications.

SRASN: NPE-90-026 DOC NO: MCP-90-1079-S00-R00 SYSTEM: E22

DESCRIPTION OF CHANGE: Modifications were made to certain power circuits to reduce voltage drops within long power and control circuit runs. Spare conductors within some power cables were utilized for a parallel feed on the positive lead to reduce the voltage drop on these circuits. For control circuit 1CA701, spare conductors of existing Division 3 cables were utilized to reconfigure the HPCS Diesel Generator Breaker 152-1701 autoclose circuit in order to eliminate an excessively long control circuit route. Also, certain conductors within the control circuit cables were paralleled to further aid in voltage drop reduction. Spare conductors within existing Division III cables were utilized to ensure both divisional separation in accordance with Reg. Guide 1.75 and proper cable qualification. The conductors utilized are of adequate ampacity for their application.

REASON FOR CHANGE: Material Nonconformance Report 0083-90 identified certain Division III 125 VDC circuits whose devices may not receive manufacturer's minimum voltage values during a Design Basis Accident (DBA). The deficient voltages have been attributed to voltage drops within long power and control circuit runs.

SAFETY EVALUATION: The safety evaluation concluded that the change did not involve an unreviewed safety question. No system function has been altered and no new equipment was installed. Only spare conductors within existing Division III cables were utilized to ensure both divisional separation in accordance with Reg. Guide 1.75 and proper cable qualification. Since proper separation is maintained, a failure in Division III circuits cannot propagate into another safety system thus limiting the failure to Division III. The conductors utilized are of adequate ampacity for their application.

This design change does not affect the HPCS system in consideration to items addressed in GGNS Technical Specifications such as flow, chemistry, setpoint, capacity, level, or pressure. This change is limited to termination/sparing of existing conductors and deletion of jumpers to improve voltage conditions of these Division III circuits.

SRASN: NPE-90-027 DOC NO: CN-90-0105 SYSTEM: L21

DESCRIPTION OF CHANGE: This change replaces resistors in 125 VDC ground detection circuits with lower value resistors to allow full scale meter deflection.

REASON FOR CHANGE: The change to the lower resistance value resistors allow full scale meter deflection. The change to replace the contact block of pushbutton switch PB2 allows isolation of the test circuit for the meter from ground to ensure that any existing grounds on the system will not interfere with testing of the meter.

SAFETY EVALUATION: The safety evaluation concluded that the change did not involve an unreviewed safety question. The modification will have no physical impact to any safety related components, structures, or systems described in the FSAR. The changes are confined to the interior of non-safety related panels and do not affect the function of the systems. Existing fuse protection on the control circuit ensures that malfunctions will not propagate to the DC Distribution Panel. The consequences of failure of the distribution panels, however, are enveloped by accidents or occurrences already evaluated.

The modification will have no impact on systems, components, or functions that could alter any technical specification safety margins. Isolation and separation of the circuit per Reg. Guide 1.75 will prevent propagation of failures to any other equipment.

SRASN: NPE-90-028 DOC NO: DCP-84-0149-S00-R00 SYSTEM: R60

DESCRIPTION OF CHANGE: Safeguards DCP

REASON FOR CHANGE: Safeguards DCP

SAFETY EVALUATION: Safeguards DCP

The safety Evaluation is available for review at Grand Gulf Nuclear Station.

SRASN: NPE-90-029 DOC NO: DCP-88-0042-S00-R00 SYSTEM:

DESCRIPTION OF CHANGE: This change provided for the erection of an enclosed structure in the Motor Control Center Area (MCC) at Elevation 133' of the Turbine Building.

REASON FOR CHANGE: This change provided Nuclear Operation 'B' personnel a permanent workbase to use for planning and scheduling activities.

SAFETY EVALUATION: The safety evaluation concluded that the change did not involve an unreviewed safety question. This facility is located in the MCC Area at Elevation 133' of the Turbine Building and is not in close proximity to any safety related components. Additionally, the minimal amount of safety related components in the Turbine Building are designed to fail safe or in a manner that does not compromise any required safety function. In accordance with the original design criteria for structures located within the Turbine Building, the facility was designed to satisfy Uniform Building Code (UBC) requirements, including seismic. The facility is constructed and finished with non-combustible materials and contains a smoke detector to provide early warning detection. The facility itself, however, is not required to have fire rated boundaries. The loads associated with this facility, including live load, are well within the live loads specified for this portion of the Turbine Building and therefore do not adversely impact the Turbine Building 133' floor slab or structural steel. Additionally, this facility does not house and is not located in close proximity to any equipment or component used in mitigating the consequences of an accident.

The new structure does not degrade the ability of any Fire Protection System to perform its intended function, does not introduce new or different failure criteria, and does not adversely affect or invalidate existing analyses for postulated design basis fires.

SRASN: NPE-90-030 DOC NO: CALCULATION NPE-E22F004 SYSTEM:

DESCRIPTION OF CHANGE: This calculation revised the maximum stem thrust that can be applied to valve E22F004 while maintaining all components within code allowable limits. The maximum stem thrust provides a maximum upper bound limit for the Mechanical Specification for torque switch setting on motor operated valves. The torque switch is used to stop the motor from providing a stem thrust higher than the valve design will allow.

REASON FOR CHANGE: The original seismic stress calculations determined a required thrust value based on the expected change in pressure in the valve. This value was only based on empirical formulas. This supplemental calculation determined the maximum thrust based on the actual valve design. The calculation reanalyzes only those components which are affected by stem thrust.

SAFETY EVALUATION: The safety evaluation concluded that the change did not involve an unreviewed safety question. Increasing the stem thrust value for valve 1E22F004 in the seismic stress analysis does not physically change the valve or modify the use of the valve. This calculation only shows the maximum stem thrust which can be obtained while maintaining all valve components, both pressure retaining and non-pressure retaining, within allowable code limits. This calculation shows all stresses are within the code allowable limits and that pressure integrity and structural integrity is maintained for operational loads, internal pressure loads and seismic loads. This thrust value will only be used as a maximum total thrust limit in Mechanical Specification SERI-MS-25.0 for the testing of motor operator valves and the setting of the torque switches. The torque switch is used to stop the valve motor operator at a thrust lower than the maximum thrust determined in the supplemental calculation in order to maintain the integrity of the valve. The supplemental stress calculation is performed to show the valve can maintain ASME code allowables for pressure retaining components with a larger stem thrust than was previously evaluated in the original calculation.

Providing supplemental seismic stress calculations for a valve will not affect the basis for any GGNS Technical Specification. The calculation is performed to show the valve will still perform its intended function during normal operation or any accident condition and the valve component stresses are still within the original design basis. Therefore, the margin of safety as defined in the GGNS Technical Specification has not been changed.

SRASN: NPE-90-031 DOC NO: EERR NO. 90-6162 SYSTEM:

DESCRIPTION OF CHANGE: Engineering Evaluation Request Response (EERR) No. 90/6162 was issued for the installation of a temporary snubber testing facility on Elevation 166'-0" in the Southeast Quadrant of the Auxiliary Building (Area 7). The testing facility consists of an 8' x 17' room, which houses a computer and printer, control console, two desks or tables with chairs, and file cabinet; and an adjacent 12' x 32' test room which houses the snubber test bench, a work table, and storage cabinets.

REASON FOR CHANGE: This EERR specified the requirements for the temporary snubber testing facility that was installed to support snubber testing during RF04.

SAFETY EVALUATION: The safety evaluation concluded that the change did not involve an unreviewed safety question. There are no design basis events (anticipated operational occurrences and accidents) described in the UFSAR that are applicable to the installation of the temporary snubber test facility or its supporting equipment. Appendix 9C of the UFSAR requires that "a single exposure fire cannot affect redundant safe shutdown-related components". The temporary testing facility was installed in Fire Zone 1A403 of Fire Area 19. Per the Fire Hazards Analysis (FHA) for GGNS Unit 1, this fire zone contains only Division 1 safe shutdown components. Sufficient physical separation is provided from adjacent Division 2 safe shutdown components to ensure that a postulated fire in Fire zone 1A403 does not affect nor propagate to affect more than one safe shutdown train/division. The analysis of safe shutdown in the event of a fire, as described in Appendix 9C of the UFSAR, is not adversely affected.

The temporary testing facility does not adversely affect the existing operation of plant systems, structures, or components required for the mitigation of a postulated event. The potential radiological dose rates postulated for accident conditions described in the UFSAR and as limited by 10CFR20 and 10CFR100 are not increased.

The temporary snubber test facility does not introduce intervening combustibles which would compromise the separation of Division I and II safe shutdown components as described in the Fire Hazards Analysis (FHA). The temporary electrical power to the facility is supplied from non-safety related BOP power receptacles. To ensure that equipment important to safety is not affected, the power feed is installed to provide physical separation from safety related equipment per the requirements of Reg. Guide 1.75. The temporary test facility is a non-seismic structure and a seismic II/I walkdown has been performed to ensure that no safety related systems, structures, or components are affected. The integrity of the Auxiliary Building structure for the additional loads created by the temporary test facility and its related equipment was verified. A partial blockage of an existing emergency light is created by the temporary construction. This partial blockage has been evaluated to ensure that sufficient lighting will be maintained along the affected ingress and egress routes in accordance with 10CFR50, Appendix R, Section III.J.

The installation of these temporary power supplies and the selection of cable sizes performed in accordance with Reg. Guide 1.75 and Article 310.15 of the National Electric Code to ensure that possible accidents remain within the bounds of existing analyses evaluated in the UFSAR. Proper sizing of cable for the temporary power feeds to the snubber testing facility ensures that there are no adverse effects to existing plant equipment. The power feeds are electrically isolated and physically separated from existing safety related components. Clamping devices and a support restraint are installed to components of the snubber test machine, in conjunction with special requirements for operation, to prevent potential missiles which could compromise existing plant safety related equipment. Therefore, there is no creation of a possibility for an accident or malfunction of a different type than any evaluated previously in the Safety Analysis Report.

The snubber testing does not modify, delete, or add any new or unanalyzed loads to existing plant electrical or mechanical systems or components that could change the operational or functional characteristics of the plant that could result in a change to the safety limits of conditions of operation as defined in the bases for the Technical Specifications. The Auxiliary Building Structure has been qualified for the added loads from the test facility and its equipment. Therefore, the construction and use of the temporary snubber testing facility does not reduce any of the margins of safety defined in the bases for any Technical Specification.

SRASN: NPE-90-032 DOC NO: CN-90-0125 SYSTEM: E12

DESCRIPTION OF CHANGE: This change adds two new manual vent valves to the ADHR system and deletes vent valves E12-F427 and E12-F418. These valves are the new safety to non-safety boundary of the vent system and will perform the function of venting and isolation of the Alternative Decay Heat Removal (ADHR) system.

REASON FOR CHANGE: The previous two valves, E12-F418 and E12-F427, were difficult to access.

SAFETY EVALUATION: The safety evaluation concluded that the change did not involve an unreviewed safety question. The safety related piping and pipe supports designs meet ASME Section III requirements and are qualified as seismic category I. The non-safety related piping and pipe support meet ANSI B31.1 requirements and are qualified as seismic category II/I. The addition of the piping and pipe supports does not affect the integrity of the interfacing piping systems or any safety system. The piping and pipe supports will function in their intended manner. This design change will allow easier venting of the system. The operation or function of the E12 system, as analyzed in the FSAR, is not affected by the modifications of this change.

The installation of the piping and pipe supports to the system will not change the system function or operation as defined by any bases for the Technical Specifications.

SRASN: NPE-90-034 DOC NO: CALC. EC-Q1L21-85001,R02 SYSTEM:

DESCRIPTION OF CHANGE: Revision 2 to this calculation is revising the Division I and II battery load profiles.

REASON FOR CHANGE: Calculation EC-Q1L21-85001, Rev. 1 was issued to verify the adequacy of the Division I and II 125V DC batteries during a worst case scenario (Loss of Offsite Power and associated diesel generators in conjunction with a LOCA). Calculation EC-Q1L21-85001, Rev. 1 identified a diesel generator field flashing load of 70 amps during the first and third 1 minute periods of the batteries duty cycle. Per IEEE 485-1978, if a discrete sequence of momentary loads can be established, the load for the 1 minute period shall be assumed to be maximum current at any instant. Since the field flashing circuit for the diesel generator is opened prior to the generator and first sequencing load group breaker's spring charging motors energized, and the load for the breakers spring charging motor envelops the generator's field flashing load, the generator's field flashing load will not be listed for the first and third cycle of the battery loading tables. Also, since the duty cycle of the 4.16KV spring charging motors is 2 seconds and a 5 second delay exist for one of the two first sequencing load group loads, two concurrent switchgear operations will be considered (diesel generator and one first sequencing load group breaker). Also the load identified in calculation EC-Q1L21-85001, Rev. 1 for the Uninterruptible Power Supplies (UPS) will be increased to allow for an added margin between the existing UPS load and future load additions. UFSAR Tables 8.3-6 and 8.3-7 will be revised to reflect the results of this calculation for the existing loads on the 125 VDC ESF batteries A and B. This calculation is based on the methodology described in IEEE 485-1978 'Recommended Practice For Sizing Large Lead Cell Batteries For Generating Stations and Substations'.

SAFETY EVALUATION: The battery load as determined by this calculation is lower than or equal to the Technical Specification load utilized for the operational surveillance for ESF Batteries A and B. These batteries have demonstrated the capacity to maintain the minimum allowed terminal voltage of 105V using the testing load thus, demonstrating the capacity for the load determined by this calculation. The calculation also shows that the battery chargers are adequately sized to recharge the batteries in less than 12 Hrs. as presently required per UFSAR 8.3.2.2.1. The UFSAR table changes performed for this calculation require no modification to the 125 VDC ESF Division I or II batteries to accommodate the revised load calculation. The modification to UFSAR tables 8.3-6 and 8.3-7 is only a software change required to update the UFSAR to reflect the results of the revised load calculation.

NPE-90-034
Page 2

The actual worst case load as determined by this calculation is within the capacity of the Division I and II ESF batteries according to the methodology specified in IEEE 485-1978, 'Recommended Practice For Sizing Large Lead Storage Batteries For Generating Stations and Substation'. Also, the load is less than or equal to the Technical Specification load during all time periods. The batteries have demonstrated the ability to accommodate the Technical Specification load and thus, have demonstrated the ability to accommodate the worst case load as determined by this calculation.

SRASN: NPE-90-035 DOC NO: CN-90-0185 SYSTEM: E12

DESCRIPTION OF CHANGE: This change notice will add an annunciator on control room panel 1H13-P601-17A for the Alternative Decay Heat Removal System (ADHRS). The annunciator will alarm on HI ADHR Heat Exchanger Inlet Temperature or LO ADHR System Flow. Should this alarm occur, the ADHR heat exchanger inlet temperature indicator or the ADHR system flow indicator, both mounted on control room panel 1H13-P601-17B, will provide indication to allow operations to determine which parameter caused the alarm.

REASON FOR CHANGE: This design change is an enhancement to the existing ADHR system, providing an audible alarm in the control room and thus this change will not have any impact on the existing design functions or operation of the ADHR system.

SAFETY EVALUATION: The safety evaluation concluded that the change did not involve an unreviewed safety question. The recommended alarm setpoints are consistent with existing technical specification requirements for the applicable reactor operational condition. The control room annunciator system is a non-safety related system. The design will utilize existing transmitters to provide input to alarm cards which will provide input to the annunciator logic. The alarm cards will be installed in an existing card rack. Proper separation will be maintained within the panels for the alarm cards, transmitters, and annunciator logic. Since there are no ESF devices within panels 1H13-P84 & 1H13-P63, this design will not create any seismic II/I concerns. One cable will be routed in non-divisional floor cable ducts within the control room, maintaining proper separation. Failure of any component added or modified by this change will not initiate any transient or accident previously evaluated in the UFSAR. The changes made by this design will not prevent any equipment relied upon to mitigate the consequences of any evaluated accident from performing its safety function. No equipment important to safety is affected by this change. All necessary requirements and commitments are met by the new design and no new accident precursors are created.

The addition of this annunciator does not change the original design intent of any equipment and all applicable design and installation requirements are met.

SRASN: NPE-90-036 DOC NO: W.O. 19998 SYSTEM:

DESCRIPTION OF CHANGE: Work Order 19998 provides directions necessary for the application of Induction Heating Stress Improvement (IHSI) on 34 Reactor Pressure Vessel Nozzle Weldments. The work order provides direction for the location of major components of the IHSI process and will establish temporary sources for electrical power and cooling water necessary to the process. IHSI was implemented with the reactor in Mode 5.

REASON FOR CHANGE: The IHSI process is intended to be applied to welded joints of austenitic stainless steel which are the primary materials for which the stress improvement process was developed. The IHSI treatment is being performed to mitigate the susceptibility of the inconel weld materials to intergranular stress corrosion cracking (IGSCC).

SAFETY EVALUATION; The safety evaluation concluded that the process does not involve an unreviewed safety question. Since IHSI changes only the residual stress state at the inside surface of the piping weldment from tensile to compressive and the existing design is unchanged, no modes of failure are introduced. With the elimination of a major stress factor, the incident of IGSCC is significantly reduced and therefore the probability of an accident is reduced.

The implementation of IHSI on the Reactor Vessel Nozzle Weldments does not change the existing design, physically or operationally, therefore existing safety evaluations remain unchanged. With the elimination of a major stress factor, the incident of IGSCC is significantly reduced, therefore, reducing the probability of a failure of the Nozzles.

The application of IHSI will ensure that the structural integrity of the Reactor Pressure Vessel is maintained by eliminating a major stress factor as a contributor to IGSCC.

SRASN: NPE-90-037 DOC NO: CN-90-0182 SYSTEM: G41

DESCRIPTION OF CHANGE: This change disables or removes the stop check valves currently utilized on the return lines to the spent fuel pool and provides redundant, passive anti-siphon vents.

REASON FOR CHANGE: The stop check valves were a frequent maintaince item, and this change provides siphon protection for the subject lines through passive anti-siphon vents.

SAFETY EVALUATION: The safety evaluation concluded that the change does not involve an unreviewed safety question. Siphon protection is provided on the supply lines which terminate below the minimum pool level as required by FSAR section 9.1. The active siphon protection system previously provided is being removed from the system and replaced by a passive system. Being passive, it does not rely on active components and thus increases the reliability of the system. No other equipment is affected by this change. This change does not affect the compliance of the overall design to 10CFR50 Appendix A criteria 61 and 62 as discussed in FSAR paragraphs 3.1.2.6.2 and 3.1.2.6.3. All of the limits for stored fuel shielding, cooling, and reactivity control as described in FSAR paragraph 15A.6.2.3.14 are unaffected by this change. The cask drop in the spent fuel pool accident described in FSAR subsection 15.7.5 and the fuel handling accidents described in FSAR subsections 15.7.4 and 15.7.6 are also unaffected by this change.

The design provided by this has been evaluated against the applicable design criteria, installation, and operational requirements. It was determined that all necessary requirements and commitments are met by the new design and that no new accident precursors are created. The overall capabilities of the spent fuel pool as described in FSAR sections 7.1, 7.4, 7.6, and 9.1 are not reduced by this change. Therefore, there is no creation of a possibility for an accident or malfunction of a different type than any evaluated previously in the Safety Analysis Report.

The referenced technical specifications and bases have been reviewed to determine if the margin of safety will be reduced by the implementation of the change. Technical Specifications require a minimum pool level to be maintained. The siphon protection method or function is not specifically addressed in the bases. The sipon protection is provided in the design to prevent inadvertent draining of the pool below elevation 202'5-1/4". No reduction in the margin of safety results because of the the alternate method of siphon protection provided by this change.

SRASN: NPE-90-039 DOC NO: EER-90-6228 SYSTEM:

DESCRIPTION OF CHANGE: This evaluation allows temporary lead shielding to be attached to certain Reactor Pressure Vessel nozzles. The lead shielding will be installed during Operating Modes 4 and 5 only, and must be removed prior to restart.

REASON FOR CHANGE:: This change was made in order to reduce radiation exposure to personnel performing work in this area.

SAFETY EVALUATION: The change does not involve an unreviewed safety question. All applicable ASME Code allowable stresses are met. The probability of occurrence of an accident resulting from a seismically initiated pipe break is not increased. There will be no change to existing designs after the lead shielding is removed. These temporary changes do not affect the structural integrity of the nozzles or associated piping during cold shutdown or for Operating Modes 4 and 5. Installation of lead shielding temporarily does not change the limiting conditions for operation, applicability, or surveillance requirements as defined in the basis for the Technical Specifications.

SRASN: NPE-90-040 DOC NO: CALC: MC-Q1E30-90112 SYSTEM:

DESCRIPTION OF CHANGE: NPE Calculation MC-Q1E30-90112 was performed to determined the effect an Upper Containment Pool (UCP) available water volume reduction would have on the containment analysis and to "as-built" UFSAR Table 6.2-50, "Suppression Pool Geometry - 251 Plant". The values resulting from the calculation were then applied to Table A-10, "Drywell and Suppression Pool Geometry" of Appendix 6A to the UFSAR. Table A-10 contains the numerical values for parameters utilized for the GGNS Containment Analyses. The engineering evaluation analyzes any resulting differences to verify that the current "as-built" conditions are bounded by the existing analyses.

REASON FOR CHANGE: Design Change Package 86/0083 added an 18 inch extension to the Upper Containment Pool (UCP) Dryer Separator Wall. The extension has two gates supplied which are removed during plant operation. The extension required the addition of a 2 7/8" sill to the top of the existing wall. This sill reduced the UCP volume available for suppression pool make-up.

SAFETY EVALUATION: This safety evaluation concluded that the change did not involve an unreviewed safety question. The calculation was performed to provide a basis for the parameters utilized in the various calculational models for the Mark III containment. The differences identified between the parameters derived in the calculation and those utilized in the containment analyses were evaluated in the engineering evaluation. The conclusion of the evaluation is that all identified differences are bounded by the current analyses. No adverse effects on systems, structures, or components previously evaluated will result due to the conclusions reached in the calculation. The calculational results were evaluated as to the impact of each parameter change on the various containment analyses. This evaluation demonstrated that the parameters are still within the design capabilities of the affected safety related structures and equipment of the "as-built plant configuration. There is no adverse impact on systems, structures, and components necessary to mitigate a postulated accident affecting the drywell or containment or to safely shutdown the plant.

The referenced technical specifications and bases have been reviewed to determine if the margin of safety was reduced. A limit for submergence of the top row of vents was identified as a design variable to verify during the calculation. Even with the reduced UCP make-up volume, the required 2 foot submergence is maintained. Based on this fact and the results of the engineering evaluation, the plant design is bounded by the current accident analysis for the "as-built" configuration of the Suppression pool and Drywell. No reduction in the margin of safety results from the values determined by calculation no. MC-Q1E30-90112, Rev. 0.

SRASN: NPE-90-041 DOC NO: DCP 82-0056-S00-R00 SYSTEM: P75

DESCRIPTION OF CHANGE: DCP 82/0056 changes the orientation of the standby Diesel Generators Starting Air Storage tank relief valves Q1P75F025 A, B, C, and D from the horizontal to the vertical position. An elbow and an additional piece of pipe were used to reorient the valves to the vertical position. The valves were previously attached in the horizontal position with a single piece of pipe.

REASON FOR CHANGE: The current vertical position of the relief valves is less susceptible to inadvertent actuation and improves valve reseating.

SAFETY EVALUATION: The change does not involve an unreviewed safety question. The operation and function of the affected system will not be altered. The valve orientation and piping supplied by the DCP meets all applicable design requirements and will function in their intended manner. The mounting of the valves in the vertical position will enhance the reliability of Division I & II Diesel Generators and will not impact the capability of the Diesel Generators to mitigate the consequences of an accident. The valves were reoriented to the vertical position as recommended by the manufacturer and will not affect the structural integrity of the starting air storage tank.

Because this DCP does not change the limiting condition for operation, applicability of surveillance requirements as defined in the basis for technical specifications, there is no reduction in the margin of safety.

SRASN: NPE-90-042 DOC NO: DCP 82-4178-S00-R00 SYSTEM: E31

DESCRIPTION OF CHANGE: DCP 82-4178 replaces the leak detection turbine meters used for the upper containment pool liner, spent fuel pool liner, and refueling bellows with sight glasses. The sight glasses will be periodically checked by operations.

REASON FOR CHANGE: The turbine meters tend to clog when leakage flow occurs. The use of sight glasses is intended to alleviate this problem.

SAFETY EVALUATION: The change does not involve an unreviewed safety question. There is no accident evaluated in the FSAR that postulates a failure of the existing leak detection instruments. There is no automatic safety function associated with this equipment. All piping and supports used in this DCP meet all design requirements and will function in their intended manner.

Failure of this flow monitoring system could result in an initially undetected leak in the refueling bellows or pool liners. Any leakage in excess of the 25 gpm identified leakage limit will be detected by sump fill annunciators or by pump out timers.

No margin of safety as defined for any technical specification is affected by this flow monitoring subsystem.

SRASN: NPE-90-043 DOC NO: DCP-84-0250-S00-R00 SYSTEM: N31

DESCRIPTION OF CHANGE: DCP 84-0250 changes the operating ranges of the turbine bearing pedestal and shaft vibration measuring instrumentation. The DCP will also modify all associated computer, annunciator, and recorder scales ranges and setpoints as appropriate.

REASON FOR CHANGE: The input circuit boards of the sensor amplifier cards will be modified to narrow the operating range as per vendor recommendations. This will provide greater resolution and readability of vibration values in the lower ranges associated with normal operations.

SAFETY EVALUATION: The change does not involve an unreviewed safety question. There is no safety related function associated with the turbine generator control system. No accident previously analyzed in the FSAR relies on the turbine generator bearing and shaft vibration monitoring system to mitigate the consequences of an accident. No malfunction of equipment important to safety previously evaluated in the FSAR is predicated on a failure of the turbine generator bearing and shaft vibration monitoring equipment.

This DCP does not affect the operation or function of the turbine generator bearing and shaft vibration monitoring system components.

Because there is no affect on the operation or function of the turbine generator bearing and shaft vibration monitoring system components, there is no reduction in the margin of safety as defined in the basis for any Technical Specification.

SRASN: NPE-90-045 DOC NO: DCP-85-4051-S00-R01 SYSTEM: P64

DESCRIPTION OF CHANGE: DCP-85/4051 installs a pressure switch in the pneumatic actuation piping in order to provide auxiliary trip functions on manual actuation. This DCP provides a means for shutting down the HVAC inlet fan motors and for closing the associated fire dampers after a manual initiation of the N1P64D006-N Halon suppression system. The manual actuation control panel (N1P64D006D) is being relocated to outside the hazard area.

REASON FOR CHANGE: To enhance fire protection performance after a manual initiation of the Computer and Control Panel Room Halon 1301 fire suppression system.

SAFETY EVALUATION: The modifications performed by this DCP meet all applicable requirements of system specifications and fire protection standards. Modifications are consistent with the original system design and vendor recommendations/requirements. The CMU wall from which the Automan II-C panel and pressure switch are supported has been analyzed to assure its structural integrity. This DCP does not change the sequence of events or the consequences of a failure of the Halon 1301 fire suppression systems for the computer and control panel room. Further, this design change will not reduce the capability of this equipment to performing its intended function. The operation of safety related equipment will not be affected by the implementation of this DCP. No new interfaces with other equipment will be created.

Normal automatic system actuation is not affected. System power, instrumentation and control are such that reliability is not reduced.

This DCP does not alter the ability of this equipment to meet fire protection standards and system specifications. Further, it does not affect the ability of the equipment to perform in accordance with the original design and vendor recommendations. No system or components will be expected to operate outside of design or Technical Specification limits.

SRASN: NPE-90-046 DOC NO: DCP-86-0073-S00-R00 SYSTEM: P71

DESCRIPTION OF CHANGE: DCP 86/0073 replaces the lubricating oil pump assemblies on Plant Chillers N1P71B001A-N, B-N, and C-N.

REASON FOR CHANGE: Parts of the installed lubricating oil pumps are no longer available from the manufacturer.

SAFETY EVALUATION: The change does not involve an unreviewed safety question. The Plant Chilled water system is a non-safety related system whose failure will in no way compromise any safety related systems or components or prevent a safe reactor shutdown. Further, the Plant Chilled water system does not function to mitigate the consequences of an accident. The new and old lube oil pump assemblies are very similar in design and construction with the new lube oil pump assemblies being vendor supplied equivalents. The centrifugal compressors are the only potential missile source on the plant chillers evaluated in the FSAR.

Because of the similarity of the old and new designs, any analyses of the old lube oil pump assemblies with respect to missile hazards would be valid for the new lube oil pump assemblies while some piping modifications are required to facilitate the installation of the new lube oil pumps, none are safety related or seismic. No new failure modes are being introduced.

The plant chilled water system is not addressed by the GGNS Unit 1 Technical Specification.

SRASN: NPE-90-047 DOC NO: DCP-87-0034-S00-R00 SYSTEM: L21

DESCRIPTION OF CHANGE: DCP 87/0034 installs fuses on the load side of the four safety related GE model AK breakers.

REASON FOR CHANGE: The fuses will provide the necessary circuit protection for the 125 VDC Bus feeders. The existing breakers will serve as disconnects only. This was done on response to SER #28-83, which pertains to failure of breakers of this type.

SAFETY EVALUATION: The change does not involve an unreviewed safety question. The addition of the fuses added by the DCP provides additional short circuit protection without changing the circuit function. The use of these fuses ensures a high reliability in the prevention of spurious trips of the 125 VDC system.

However, the GGNS Electrical Distribution System Functional Inspection (EDSFI) documented in GNRI-91/033 contained one notice of violation (NOV) which addressed a concern the NRC had with DCP 87/0034, NOV-50-416/90-24-01. The violation cited the lack of an adequate engineering evaluation of the 125 VDC Distribution System Fuse/Breaker coordination.

In the response to Notice of Violation, GNRO-91/00054, Entergy Operations has taken the following steps to correct the problem:

1. A design review of the breaker coordination associated with the DCP was performed. We are in the process of determining an approach to resolve the design deficiency.

2. A memorandum was issued to Design Engineering personnel involved in the application and coordination of protective devices. The purpose of this memorandum was to make appropriate personnel aware of this violation and the potential consequences of failure of fully coordinate all breakers associated with a modification.

After review of the modification, it was determined that this deficiency (NOV) is not considered safety significant in that the fuses provide full protection of the feeder cables and the DC Distribution System is not designed for operation with a fault.

SRASN: NPE-90-048 DOC NO: DCP-87-0048-S00-R00 SYSTEM: P75

DESCRIPTION OF CHANGE: This DCP replaced the carbon steel starting air manifolds with stainless steel manifolds for the Division I Standby Diesel Generator.

REASON FOR CHANGE: This change will eliminate corrosion caused by moisture in the manifold.

SAFETY EVALUATION: The change does not involve a USQ. Replacing the carbon steel piping will not change the operability, function, or surveillance requirements of the Diesel Generator starting air subsystem. The piping supplied by this DCP meets all applicable design requirements and will function in its intended manner. This change in no way impacts the Diesel Generator capability for mitigating the consequences of an accident.

There is no change in the limiting condition for operation, applicability of surveillance requirements, operation or function of the Diesel Generator and consequently, there is no reduction in the margin of safety as defined in the basis for any Technical Specification.

SRASN: NPE-90-049 DOC NO: DCP-88-0027-S00-R00 SYSTEM: R11

DESCRIPTION OF CHANGE: DCP 88-0027 adds direct indication that selected transformers are energized and available for a power feed. This indication will be added directly to the panel mimics for Busses 11HD, 12HE, 13AD, 14AE, 15AA, 16AB & 17AC to ensure the operator has indication within close proximity of the breaker handswitches. The status lamps for safety related busses 15AA, 16AB & 17AC are being installed in parallel with the existing breaker synchronization handswitches, fed by a potential transformer on the incoming feed. For non-safety related busses 11HD, 12HE, 13AD & 14AE, the status lamps will utilize spare potential transformers and will have no effect on safety related equipment.

REASON FOR CHANGE: During plant operation, the electrical busses mimicked on the P807, P864 and P601 panels do not have readily available indication to determine whether they are energized. The operator must locate the proper meter on the vertical section of the panel to determine if the bus is energized. Due to the spatial relationship between the control and its associated indication, switching to a dead bus for power feed could occur. A dead bus transfer could cause undesirable plant effects and possibly a plant scram.

SAFETY EVALUATION: This change does not involve an unreviewed safety question.

This design change installs neon status lamps to indicate when voltage is present at feeder breakers for Busses 11HD, 12HE, 13AD, 14AE, 15AA, 16AB & 17AC. The voltage present status lamps are being added to aid the operator during breaker alignment changes. The lamps are solid state passive components, which use lamp holders that have been seismically tested and qualif[illegible] Divisional separation requirements for each lamp being installed is maintained.

For non-safety related busses 11HD, 12HE, 13AD & 14AE, these status lamps utilize spare potential transformers and will have no effect on safety related equipment.

For the status lamps monitoring safety related busses 15AA, 16AB & 17AC, their failure could cause loss of synchronization capability for the Feeder Breaker associated with the faulted bus status circuit. However, this failure will not cause Bus De-energization nor prohibit bus sync. or transfer with other available power sources. The preferred system lineup uses an offsite power feed for Busses 15AA, 16AB and 17AC. If synchronization capability is lost and the preferred power source is maintained during an evaluated event, these busses continue to be fed from the preferred source. In this situation, loss of synchronization will not prevent any equipment form completing their intended safety function.

If the preferred power source for Busses 15AA & 16AB degrades or is lost, the Load Shedding and Sequencing system (LSS) automatically trips the incoming breakers and ties the diesel generator to the appropriate bus.

For bus 17AC, if the preferred power feed degrades or is lost, the undervoltage protection system automatically trips the incoming breakers and ties the HPCS diesel generator to the bus.

Loss of synchronization capability is not considered as an initiating or sequence event in any UFSAR accident analysis. Failure of the status lamps will thus not affect any safety related function.

SRASN: NPE-90-050 DOC NO: CN-90-0318 SYSTEM: E12

DESCRIPTION OF CHANGE: This change to the manual override logic for ECCS injection valves 1E21F005, 1E12F042B, 1E12F042A and 1E12F042C provides a time delayed contact in the closing circuit such that when the injection override circuit is sealed in while the valve is stroking open two seconds must pass after the valve limits open before it will cycle closed.

REASON FOR CHANGE: The breaker of valve 1E12F042A had tripped while testing the manual override logic. The logic allowed the operator to seal in the override logic while valve was in midstroke. The breaker tripped when the actuator tried to reverse itself while still coasting in the open direction.

SAFETY EVALUATION: This change does not involve an unreviewed safety question. The change in the low pressure ECCS automatic injection logic bypass circuit meets all applicable design requirements. The change will not cause any system or component to operate beyond its design limits nor will it affect overall system performance in a manner which could lead to an accident. The LPCS and LPCI manual override control requirements to prevent opening of the injection valves without the RPV high/low pressure interlock permissive for protection against intersystem LOCAs are unaffected by this change. The shutdown cooling event is likewise unaffected since the design requirements for the valve control handswitches are maintained.

The postulated loss of a division of ECCS considered in the determination of the most limiting failure for the various applicable UFSAR Chapter 15 accidents is completely unaffected by this design change. No accident precursors evaluated in the UFSAR are affected by this change. The effect of a component failure or single error in the operation of the manual override as modified by this DCP and CN remains bounded by the most limiting divisional failure. The accident mitigation functions associated with the use of the manual override are addressed in the EPs. Thus, since this design change does not alter any of the assumptions or degrade any of the required actions and barriers relied upon for mitigating an accident, the consequences of previously evaluated accidents are not affected. Therefore, there is no creation of a possibility for an accident or malfunction of a different type than any evaluated previously in the Safety Analysis Report.

The changes in the automatic injection override circuits for LPCI-A, B, C and LPCS do not affect any existing bases for the Technical Specification requirements and do not introduce any new requirements. Although this change increases the capabilities for manually disabling automatic low pressure ECCS injection functions, the existing coincident system initiation signal logic and override annunciation features and requirements are not reduced. The open and close valve stroke times associated with the automatic and manual active safety related functions (e.g., injection and isolation) are unaffected by the new design.

SRASN: NPE-90-051 DOC NO: DCP-88-0029-S00-R00 SYSTEM: N23

DESCRIPTION OF CHANGE: Install redundant Low-Low Level Switch on existing Heater Drain Tank Instrument Strongback.

REASON FOR CHANGE: To reduce scram frequency by increasing reliability of the Low-Low Level Trip Circuit.

SAFETY EVALUATION: This change does not involve an unreviewed safety question. The installation of the redundant low level switch as proposed by this DCP will increase the reliability of the Heater Drain Pump trip signal. This will reduce the probability for a loss of Feedwater flow which is addressed in the UFSAR section 15.2.7.

The presently installed switch (1N23-LSLL-N081) does not have any affect on the consequences of an accident evaluated in the FSAR. Section 10.4.7.3 states "The condensate and feedwater system serve no safety function. System analysis has shown that failure of this system will not compromise any safety-related systems or prevent safe shutdown." Also in this section the UFSAR states "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." The new redundant switch will have the same function and design bases as the original switch and will serve only to improve the reliability of the Heater Drain Pump trip signal.

The switch added by this DCP and all associated piping and supports are in the Turbine Building and have no seismic qualifications, therefore no II/I hazards will be created by modifications made to the Instrument Strongback. The power for the new switch will be supplied from the same bus which supplies power to the presently installed switch. This bus contains no safety related equipment.

Neither the Heater Drain Tank Level nor Heater Drain Pump Control is addressed in the Technical Specification. The DCP does not change the original function or design bases and therefore does not affect the margin of Safety defined in the Technical Specification.

SRASN: NPE-90-052 DOC NO: DCP-88-0056-S00-R00 SYSTEM: M31

DESCRIPTION OF CHANGE: This change replaced the 12 ton capacity drywell valve handling crane with a 5 ton hoist.

REASON FOR CHANGE: MSIV/SRV valve maintenance at the crane inner and outer tramrails was extremely difficult due to the size of the hoist.

SAFETY EVALUATION: This change does not involve an unreviewed safety question. DCP-0056 changes the drywell valve handling crane from a 12 ton hoist to a 5 ton hoist. The drywell valve handling crane is the only equipment affected. The crane is only used for maintenance activities in the shutdown or refueling modes. This crane is Seismic Category I for structural integrity. Both the 12 ton hoist and the 5 ton hoist are non-safety related Seismic Category II/I. There are no structural changes required to the crane to accommodate the 5 ton hoist. The 5 ton hoist will not adversely impact the crane seismic qualification as the new hoist capacity is less than half of the existing capacity. The new hoist is compatible for its anticipated service environment and there are no changes to the crane's function. Plant procedures are in place to assure the crane and hoist are inspected and maintained as appropriate and are only operated by qualified personnel. The new hoist will be fed from the existing power supply.

The drywell valve handling crane is not used in any technical specification to define the margin of safety. Nor do the changes herein require that it be used as such a base.

SRASN: NPE-90-053 DOC NO: DCP-88-0057-S00-R00 SYSTEM: R61

DESCRIPTION OF CHANGE: DCP 88-0057 provides three additional public address stations and four sound powered telephone stations inside the drywell.

REASON FOR CHANGE: To provide additional personnel safety and to improve the efficiency of operations conducted inside the drywell.

SAFETY EVALUATION: This change does not involve an unreviewed safety question. This change will not affect the operation of any safety related equipment nor modify the operation of existing safety related systems. Seismic supports are provided for raceway and equipment to ensure no II/I seismic hazards are created. The added BOP raceway, equipment and cable will be installed to meet the Regulatory Guide 1.75 separation requirements.

No accident parameters or existing safety functions are being modified.

The added sound powered phones and PA will be added to the existing sound powered phone and to the existing PA system respectively. Each will meet all applicable design, seismic, and separation requirements.

The PA system is not addressed in the Technical Specifications, nor does the added capability adversely affect any system addressed in Technical Specifications.

SRASN: NPE-90-054 DOC NO: DCP-89-0343-S00-R00 SYSTEM: R93

DESCRIPTION OF CHANGE: DCP 89-0343 installs a lightning dissipation system. The purpose of the system is to dissipate the charge in the area of protection before a lightning strike occurs. The system consists of a variety of dissipation systems including Paragon Arrays, pyramid arrays, parapet arrays, rim arrays, spline ball ionizers, hemisphere arrays and DDS lines. This patented lightning protection system was installed on the following plant structures:

PLANT STRUCTURE	ARRAY TYPE
Cooling Tower	RIM ARRAY
Turbine Building	PARAGON ARRAY
Enclosure Building	PYRAMID ARRAY
Auxiliary Building	PARAPET ARRAY
Radwaste Building	PARAPET ARRAY
Water Treatment Building	PARAPET ARRAY
Radial Well Switchgear House	HEMISPHERE ARRAY
Radial Wells 1, 3, 4&5	HEMISPHERE ARRAY
Meteorological Towers	SPLINE BALL IONIZER
High Mast Halo Lights	HEMISPHERE ARRAY
High Mast Ballfield Lights	SPLINE BALL IONIZER

REASON FOR CHANGE: To help prevent various transients to plant equipment due to lightning strikes. Induced current caused by lightning can cause plant monitoring systems to give erroneous information resulting in spurious trips or plant scrams.

SAFETY EVALUATION: This change does not involve an unreviewed safety question. To prevent equipment damage from occurring after a lightning strike, protective relay schemes automatically disconnect electrical sources and loads to mitigate damage and regain electrical grid stability. Reducing the potential for lightning strikes will reduce the risk of creating plant monitoring system instability and thus reduce the risk of damaging plant equipment. Failure of the system (lightning strike) will not have a detrimental effect since the dissipation structures will fail as air terminals.

Lighting could cause physical damage to protection components. Reducing the potential for lightning strikes will reduce the risk of damaging or degrading the performance of protection system components.

The dissipation system will also reduce the risk of transients in plant monitoring systems during high storm activity. This in turn will provide the operators with more reliable monitoring instruments.

The dissipation structures have been designed to withstand a wind load of 110 MPH. The additional loads imposed on existing plant structures is small and has no effect on their structural integrity. Failure of the dissipation structures under tornado loads has been reviewed.

Since tornado generated missiles were considered as the limiting natural phenomena in the design of all structures that are safety related, failure or collapse of the lightning dissipation structures will not affect the ability of the Seismic Category I structures, systems, or components to perform their intended function. This system is a passive system and is tied into the existing plant grounding system. It will neither interact with nor have any effect on other plant systems.

SRASN: NPE-90-055 DOC NO: DCP-89-0343-S01-R00 SYSTEM: P47

DESCRIPTION OF CHANGE: DCP 89-0345-1 installs surge protection on the Hard wired instrument power and data lines that run between the radial wells and the switchgear house. Surge protection was also added between the Meteorological Monitoring station and the plant. The base package of this DCP added a plant wide lightning dissipation array system which included the radial wells, radial well switchgear house, and the meteorological monitoring station.

REASON FOR CHANGE: Surge protection reduces the detrimental effect on plant equipment caused by electrical transients induced on the plant grounding system.

SAFETY EVALUATION: This change does not involve an unreviewed safety question. The radial well system has no safety related function. It provides makeup to the standby service water system cooling tower basins through the PSW system, but this makeup capability is not required to safely shutdown the reactor following a LOCA. Failure of the surge protectors at the radial wells will not have a detrimental effect on plant safety. The surge protectors will also reduce the risk of damage to meteorological monitoring equipment due to transients during high storm activity. This in turn will provided the operators with more reliable monitoring instruments. The meteorological monitoring system serves no safety related function. Failure of this system (i.e. failure of the surge protectors at the meteorological monitoring station) will have no adverse effect on plant safety.

Addition of this system will not create any new failure modes of plant systems due to electrical failure. It is a passive system which will be tied to the existing grounding system only. It will not interact or affect the operation of other plant systems. It is designed to reduce the probability of lightning induced transients causing damage to and/or inadvertent actuation of plant equipment.

Surge protection is not addressed in the GGNS-1 Technical Specifications nor does the installation affect any safety related system addressed in the Technical Specifications.

SRASN: NPE-90-056 DOC NO: NPE-FSAR-89-0041 SYSTEM:

DESCRIPTION OF CHANGE: This change adds section 3.2.6 to the UFSAR which will allow the replacement and modification of ASME Section III, Class 1, 2 and 3 components which are not code stamped but meet code requirements. This is allowed by Generic Letter (GL) 89-09.

This change also provided for the addition of TABLE 3.2-5, ASME Section III Component or Component Parts obtained to the Guidance of Generic Letter 89-09.

REASON FOR CHANGE: Numerous ASME accredited suppliers/ manufacturers have allowed their Certification of Authorization to expire. This has created difficulty for licensees to obtain replacements in full compliance with the licensing commitments. The Nuclear Regulatory Commission has responded to the issue with the issuance of Generic Letter 89-09 (GL). The GL provides generic relief allowing the use of non-code stamped replacements for items that were originally code stamped. The GL contains numerous staff positions that require consideration and compliance by the licensee for its implementation. One of the staff positions requires the licensee's FSAR be revised to address the GL and identify the items obtained using its guidance.

SAFETY EVALUATION: This change does not change involved an unreviewed safety question. As indicated in Generic Letter 89-09, the replacement of Section III items using the guidance of the GL provides an acceptable level of quality and safety. The requirement imposed by ASME Section XI and III to use stamped items with accompanying documentation imposes undue hardships on the licensees without a compensating increase in the level of quality and safety over that provided by the alternatives contained in the guidance of the GL. The technical requirements of the code are still maintained and assured by the use of a 10CFR50, Appendix B Quality Assurance Program in lieu of an NCA-4000 Quality Assurance Program. Use of the GL does not alter any evaluations that depend on the function of a Section III component.

Items affected by the FSAR change request are in accordance ASME Section III, Classes 1, 2 or 3 and are also classified as Quality Group A, B or C, respectively. Because of compliance with the rules associated with these classifications, margins of safety can be established and maintained. As recognized by the NRC, when replacements are not available in full compliance with the code stamping and documentation requirements of ASME Section III, the result is an undue hardship on the Licensee without a compensating increase in the level of quality and safety over that provided by the alternatives contained in the guidance of GL 89-09. The generic letter guidance provides an acceptable level of quality and safety to ensure continued function of the components that is commensurate with those obtained having an ASME Section III code symbol stamp with accompanying documentation. The existing margins of safety are not reduced by the use of GL 89-09.

SRASN: NPE-90-057 DOC NO: DCP-90-0005-S00-R00 SYSTEM: B21

DESCRIPTION OF CHANGE: DCP 90-005 replaces the carbon steel air accumulators and receivers used to supply the Main Steam Isolation Valves (MSIVs) and Main Steam Safety/Relief Valves (MSRs) with stainless steel accumulators.

This DCP also changes valve P53F012 from a stop check valve to a manual globe valve.

REASON FOR CHANGE: The air accumulators used to supply the MSIVs and the MSRs were fabricated out of carbon steel with a protective coating on the interior surface. The coating failed and was replaced. The replacement of the accumulators with stainless steel version will alleviate this problem.

The stop check valve was replaced due to problems in performing the automatic depressurization system (ADS) drop test. The test requires the valve to be in the open position. The only way to accomplish this was to manually prop open the disc. The manual valve will alleviate this problem.

SAFETY EVALUATION: This change does not involve an unreviewed safety question. The replacement of the accumulators with stainless steel will not affect the operation or the function of the MSIV's since the accumulator size and location has not changed. The replacement will in fact increase the reliability of the MSIV's by reducing the possible introduction of particles to the valve accumulators. The replacement of the P53 F012 with a globe valve will not affect the function of the ADS air supply since as a stop-check valve its only safety function was in the open position.

All piping and pipe supports changes meet ASME Section III requirements and are qualified as seismic category I. The safety related piping and pipe supports changes will function in their intended manner. The design provided by this DCP has been evaluated against all applicable design criteria as well as applicable installation and operational requirements. It was determined that all necessary requirements and commitments are met by the new design and no new equipment failure modes are introduced. The design change will not result in an operational or functional change to the systems involved or to any other safety related system.

The design change does not affect the operation or function of the MSIV's. The changing of the P53 F012 valve will not impact the operation of the ADS as defined in the bases for any Technical Specification.

SRASN: NPE-90-058 DOC NO: DCP-90-0060-S00-R00 SYSTEM: E12

DESCRIPTION OF CHANGE: DCP 90-0060 changes the manual override logic for valves 1E12F042A, B, C and 1E21F005. These valves are the Low Pressure coolant injection (LPCI)-A, B, C and Low Pressure Core Spray (LPCS) injection valves.

This logic change also affects the manner in which the override can be disabled and the automatic functions reenabled. With the existing design, the automatic valve control can be reset by either: 1) placing the valve handswitch to "OPEN"; 2) resetting the system initiation logic following the clearing of the initiation signals; or 3) the automatic resetting of the RPV high/low pressure interlock by a subsequent increase in reactor pressure above the permissive for valve opening. The override seal-in installed by this DCP will effectively prevent item (3) from disabling the override and will allow enabling of the override feature prior to the RPV pressure interlock being cleared.

REASON FOR CHANGE: The mitigation of certain accident events involving operator actions governed by the Emergency Procedures (EPs) requires the disabling of the automatic low pressure ECCS injection functions. By the existing GGNS design, the automatic injection functions for the low pressure ECC systems (e.g., LPCI-A, B, C and LPCS) can be manually overridden by turning the applicable handswitch (E12-HS-M609A, B, C for LPCI injection valves E12-F042A, B, C and E21-HS-M601 for LPCS injection valve E21-F005) to the "CLOSE" position following the receipt of a system initiation signal. However, since the RPV pressure interlock must first be cleared before the override signal will seal in, the override attempt will not interrupt the initial automatic opening of the valve. When the valve limits open, the override can then be sealed in and the valve will close. The override will remain sealed in and will inhibit all further automatic initiation signals until the logic is manually reset or automatically reset from the unusual case of an increase in reactor pressure above the permissive (which would then reenable the injection valve interlock, reclose the valve, and reset the logic).

As a result of the EP requirements, this DCP will modify the override logic for the LCPI-A, B, C and LPCS automatic injection circuits to permit the override function for the initial as well as all subsequent injections. This logic change is being accomplished by relocating the override relay contact from downstream of the RPV pressure interlock relay to upstream of that relay such that the valve automatic open signal can be inhibited following receipt and seal in of the ECCS initiation signal (with bus power available) but prior to the RPV pressure interlock clearing. The valve handswitches are also being replaced with new handswitches having additional contacts which will be used in associated with the momentary contacts in the manual override logic circuit. These additional contacts are used to seal in the override circuit when the handswitch is placed to "CLOSE" the valve and to allow the operator to break the override circuit seal-in and open the injection valve when the applicable handswitch is taken to "OPEN" position. In conjunction with the handswitch contacts, new relay contacts are being added in each valve closure circuit to seal in the "CLOSE" signal to the valve motor. If the valve is stroking open at the time the override function is sealed in, these relay contacts will reclose the valve after it limits open.

SAFETY EVALUATION: This safety evaluation concluded that the change did not involve an unreviewed safety question. The change in the low pressure ECCS automatic injection logic bypass circuit meets all applicable design requirements. The change will not cause any system or component to operate beyond its design limits nor will it affect overall system performance in a manner which could lead to an accident. The LPCS and LPCI manual override control requirements to prevent opening of the injection valves without the RPV high/low pressure interlock permissive for protection against intersystem LOCAs are unaffected by this DCP. The probability of a loss of shutdown cooling event is likewise unaffected since the design requirements for the valve control handswitches are maintained. The postulated loss of a division of ECCS considered in the determination of the most limiting failure for the sequence of events of the various applicable UFSAR Chapter 15 accidents is completely unaffected by this design change. No accident precursors evaluated in the UFSAR are affected by this change. Implementation of this DCP will not affect the low pressure ECCS initiating circuits, logic, sequencing, bypassed, or interlocks, other than permitting the override function (and override seal in) prior to the clearing of the RPV high/low pressure interlocks and thus prior to the initial injection. The logic requirement for an ECCS initiation signal prior to an automatic injection override is not altered by this design change. Inadvertent or improper operation of the override function is thus minimized by this coincident logic. Annunciation of the override function is also not changed by this DCP.

In evaluating the low pressure ECCS injection valve logic circuit design as modified by this DCP, the following single failures and operator errors were postulated:

1) any operator errors in the use of a single injection valve handswitch;

2) failure of the override seal-in contacts to make in the handswitch "CLOSE" position;

3) failure of the override seal-in break contacts to make in the handswitch "OPEN" position;

4) failure of any of the new logic circuit relay contacts to make or break when required.

It should be noted that since the new handswitches are seismically qualified, failures which would cause any of the new handswitch contacts to inadvertently make or otherwise fail as a result of a seismic event need not be postulated. Inadvertent operation of an injection valve handswitch (e.g., to "CLOSE") would only enable the override function if that action occurred with a system initiation signal present. The worst case failure resulting from this action would be a disabling of the associated automatic ECCS injection function. Each of the other malfunctions could result in a failure of the associated ECCS injection function or a failure to override the automatic injection depending on the exact failure mode. Whether by a single component malfunction or by a single operator error, the postulated failure of a single automatic ECCS injection function is still bounded by the limiting divisional failure. In the event of a failure of the injection valve override, injection can still be stopped by closing other valves in the process stream or by shutting down the applicable ECCS pump. The effect of a component failure or single error in the operation of the manual override as modified by this DCP remains bounded by the most limiting divisional failure. The accident mitigation functions associated with the use of the manual override are addressed in the Emergency Procedures.

This DCP only changes the logic and associated handswitches for overriding the automatic injection function for LPCI-A, B, C and LPCS as described above. The replacement switches are seismically qualified and meet all of the design and installation requirements of the original switches. The design provided by this DCP has been evaluated against the applicable design criteria, installation, and operational requirements. It was determined that all necessary requirements and commitments are met by the new design and that no new equipment failure modes are introduced. The potential for disabling the automatic injection function for more than one low pressure ECC system by the use of the additional override capabilities provided by this DCP when not operating by the Emergency Procedures is beyond single failure and single operator error criteria.

The new logic circuits and switches meet all applicable design and installation requirements. The existing system and component design functions are not affected. Although this DCP increases the capabilities for manually disabling automatic low pressure ECCS injection functions, the existing coincident system initiation signal logic and override annunciation features and requirements are not reduced. All margins of safety as defined for any Technical Specification thus remain unchanged.

SRASN: NPE-90-059 DOC NO: QDR-323-89 SYSTEM:

DESCRIPTION OF CHANGE: The auxiliary building isolation damper limit switch trips the normal auxiliary and fuel building area HVAC systems upon initiation of the Standby Gas Treatment System (SGTS). The purpose of the subject trips, however, is not to mitigate the consequences of an accident. Instead, the primary purpose is that of providing basic equipment protection for the normal auxiliary and fuel building HVAC system fans.

REASON FOR CHANGE: This change clarified the FSAR.

SAFETY EVALUATION: The review that was conducted after the issuance of this QDR confirmed the integrity of previous design bases and found that existing probabilities of occurrence remain valid. The subject UFSAR revision is indicative of the fact that while clarification was needed, no changes to existing analysis were necessary.

This UFSAR revision does not change system operation, nor does it imply a reduction in the safety-related capability or classification of existing systems/system components. The reason for this revision is to provide clarification of existing operation characteristics and not to describe any change from what has already been used in evaluating the consequences of an accident. The safety function of the SGTS and the capability of the secondary containment isolation valves to perform their safety function have not changed. This evaluation reconfirmed that the existing system design does not require any modifications. It also confirms the integrity of the evaluation of various accidents addressed in the UFSAR.

The text of the subject UFSAR change emphasizes that the existing design, utilizing the non-safety limit switches, does not constitute a deviation from required design considerations. The switches provide the intended function of equipment protection for non-safety related fans. As confirmed by the previously described review, the margin of safety provided in the original design has not changed.

SRASN: NPE-90-061 DOC NO: MNCR-90-0032 SYSTEM: Z51

DESCRIPTION OF CHANGE: This change revises UFSAR Table 18.1-2 to remove the four second closure requirement for the post-accident fresh air makeup valves (Z51F007, F016).

REASON FOR CHANGE: The post-accident makeup air is separated from normal fresh air makeup. It is located such that intake air is filtered prior to distributing in the Control Room. The boundary valve is normally closed motor operated valve. It is opened only post-accident to admit fresh air to replenish the oxygen for the Control Room operators and has no 4 second closure requirement.

SAFETY EVALUATION: This safety evaluation concluded that the change did not involve an unreviewed safety question. The Control Room emergency filtration system functions to mitigate the consequences of an accident, not to prevent an accident. These valves are normally closed and are interlocked closed for 10 minutes post-accident to admit fresh air and replenish oxygen for the Control Room operators. The IST program will be revised to require stroking of the valves during cold shutdowns or per ASME Sections XI rather than quarterly. Surveillances on the Z51F007 and Z51F016 standby fresh air valves will be performed only in Operation Conditions 4 or 5 when core alterations are suspended, i.e., handling of irradiated fuel in the primary or secondary containment and operations with a potential for draining the reactor vessel are not in progress. Under these circumstances, the possibility of design basis accidents and abnormal operations transients that can affect the Control Room environment are not deemed credible, and the risks associated with an inoperable filtration system are negligible.

SRASN: NPE-90-062 DOC NO: DCP-90-0344-S00-R00 SYSTEM: E30

DESCRIPTION OF CHANGE: DCP 90-0344R00 adds Reg. Guide 1.97, type C, Category 2, wide range containment water level monitoring instrumentation to support the Emergency Procedures (EPs). Two separate channels are provided (Div I and Div II) with each consisting of two probes. The probes are Fluid Controls Inc. Model CL 86 level transmitters. Two ranges are required to be monitored. The first or lower range will be from 20 to 35 ft. level (113 to 128 ft. elevation). This provides an overlap with the upper end of existing instrumentation to a level above the upper limit of the Safety Relief Valve Tailpipe Level Limit (SRVTLL) as addressed in the EPs. The second or upper range spans from a point below the elevation of the top of active fuel to a point above the elevation of the containment pressure instrumentation tap. This range is from 60 to 75 ft. level (153 to 168 ft. elevation). The indication for the probes will be in the control room.

REASON FOR CHANGE: Emergency Operating Procedures EP-2, EP-2A and EP-3 require the operator to take action at containment water levels beyond the range of existing instrumentation. The presently installed suppression pool level indication available in the control room only provides level indication over the range of 10.5 ft. (103.5 ft. elev.) to 25.5 ft. (118.5 ft. elev). The Emergency Procedures Figure 1, Maximum CTMT Water Level Limit (MCWLL) and Figure 3, SRV Tailpipe Level Limit (SRVTLL) require the operator to make decisions based on containment levels outside this range. Without this instrumentation, a potential exists for prematurely terminating injection systems from sources external to containment. Because of this SERI has committed to the NRC to install containment wide range water level monitoring.

SAFETY EVALUATION: This safety evaluation concluded that the change did not involve an unreviewed safety question. This DCP provides the Control Room Operators with indication of Containment water level during the performance of the site specific Emergency Operating Procedures EP-2, EP-2A and EP-3 thus enabling an operator to identify when an EP level setpoint or decision point is reached. Failure of the instrumentation installed per this DCP will not compromise any existing safety related system or component nor will it prevent safe reactor shutdown. No new interface is created which would affect components, equipment or systems which perform safety functions.

The changes made by this DCP do not prevent any equipment relied upon to mitigate the consequences of a malfunction of equipment important to safety from performing its safety function. These changes do not affect any Seismic Category I system, structure or component. The circuits and raceways installed per this DCP are associated Div. I and II and are routed as divisional cables in seismically supported divisional raceways. The instrumentation is to be installed Seismic II/I in the containment for the probes and in the Control Building for the indication components. The probes are considered to be structurally adequate to preclude electrical failure and all Seismic II/I concerns based on calculations performed and are considered functionally adequate to withstand a LOCA event and still be operable. There are no ASME Section III, Class 1, 2 or 3 piping or components added or modified by this change. Divisional separation, per Reg. Guide 1.75, of electrical components added or modified by this DCP is not compromised by the implementation of this change.

SRASN: NPE-90-063 DOC NO: DCP-90-0547 SYSTEM: B21

DESCRIPTION OF CHANGE: DCP 90-0547 modifies the instrument loops used for the Safety Parameter Display System (SPDS).

REASON FOR CHANGE: The input instruments affected by this DCP are being changed to be consistent with Reg Guide 1.97. The only exception will be the input for suppression pool level. SPDS is presently supplied a suppression pool level input from E30 LT N003A[B] while the Reg Guide 1.97 instruments are E30 LT N003C[D]. Although the SPDS and Reg Guide instruments are not the same they monitor the same level on the suppression pool, they have the same range and are QF1 instruments. Therefore nothing would be gained by having the Reg Guide 1.97 instruments provide the SPDS inputs.

The following are the SPDS points to be added by this DCP:

RPV Level	B21N027A
	B21N027B
Drywell Temperature	M71N013A
	M71N013B
	M71N013C
	M71N013D
Containment Temperature	M71N007A
	M71N007B
	M71N007C
	M71N007D
Suppression Pool Temperature	M71N012A
	M71N012B
	M71N022A
	M71N022B
	M71N023A
	M71N023B
	M71N024A
	M71N024B
	M71N025A
	M71N025B
	M71N026A
	M71N026B

SAFETY EVALUATION: This safety evaluation concluded that the change did not involve an unreviewed safety question. The instrument loops which are to provide the new inputs to the SPDS are indication instrument loops only and do not provide any control or trip function. Because these instrument loops do not provide any control or trip function they could not be the direct cause for the occurrence of an accident.

Additionally, the instruments are not associated with any safety related equipment and do not provide any mitigating action. They will reduce the possibility of the control room operator receiving conflicting information during an event.

The new inputs to the SPDS are being obtained by using spare points on existing R Mux units therefore no new seismic considerations are being created. In addition the design of the new inputs maintains the proper 1E to non 1E isolation.

Because the affected instrument loops do not affect any trip or control function there will be no reduction in the margin of safety as defined in the basis for any Technical Specification.

SRASN: NPE-90-064 DOC NO: FSAR-CR-90-0032 SYSTEM:

DESCRIPTION OF CHANGE: This change provides for the deletion from FSAR Table 3.10-1 certain Category II/I and QP components that are seismically qualified mechanical, I&C and electrical devices.

REASON FOR CHANGE: The equipment is removed from Table 3.10-1 only because special seismic operability considerations beyond that normally performed to ensure structural and preserve integrity are not required.

SAFETY EVALUATION: There is no increase in the probability of occurrence or in the consequences of an accident or malfunction of equipment important to safety previously evaluated in the Safety Analysis Report. The design requirements for the subject equipment are not modified. The equipment is not modified. The equipment is removed from Table 3.10-1 only because special seismic operability considerations beyond that normally performed to ensure structural and pressure integrity are not required. UFSAR Section 3.10.2.3.1 allows the deletion of Category II/I and QP components from the table. JS-08 designates all of these components as passive. This change does not degrade the ability of the subject components to perform their required functions since design standards are not relaxed.

SRASN: NPE-90-066 DOC NO: CN-90-0101 SYSTEM: P44

DESCRIPTION OF CHANGE: This change relocates the existing Temperature Control Valve (TV) valves N1P44F531B/532B with the existing butterfly valves N1P44F466B/467B to the condenser inlet piping for Drywell Chillers N1P72B001B-B/B002B-B.

REASON FOR CHANGE: The valves are switched so that the butterfly valve will now isolate the TV valve to provide for proper cleaning and maintenance.

SAFETY EVALUATION: This safety evaluation concluded that the change did not involve an unreviewed safety question. This modification provides for the removal of valve N1P44F925 to eliminate the flow obstruction problem and replacing it with a flanged branch line for hydrolyzing. The relocating of the existing TV valves N1P44F531B/532B with the existing butterfly valves N1P44F466B/467B will allow for easier maintenance of the TV valves. The existing flow point FP-N413A was relocated 15" to allow the annubar to be installed. 8" JBD 378 pipe south of fourway valve was replaced because of the high velocity erosion resulting from the bypass flow from the fourway valve. The flanged branch line on the 8" JBD-43 was moved north 23" to accommodate pipe installation. The piping and pipe supports design meets ANSI B31.1 Code requirements. The piping is supported to dead weight loads only, since it is installed in a portion of the Auxiliary Building where no II/I hazards exists. The Plant Service Water System serves no safety function. Systems analysis has shown that failure of the Plant Service Water System will not compromise any safety related systems or prevent reactor shutdown. The operation or function of the Plant Service Water System, as analyzed in the FSAR, is not affected by these modifications.

This modification to the Plant Service Water System does not change the function of operation as defined in any Bases for the Technical Specifications, therefore, the margin of safety is not reduced.

SRASN: NPE-90-067 DOC NO: SERI-JS-08 SYSTEM:

DESCRIPTION OF CHANGE: This revision to JS-08 incorporates the information developed in Phase II and Phase III of the instrument Q-List.

REASON FOR CHANGE: This revision provides design functions, safety application, Q-criteria, and references any QEVAL associated with an instrument. Also the as-built information identified by Configuration Management as of 9/89 and the identified discrepancies between JS-08 Rev. 1 and Rev. 97 of Bechtel Instrument Index are incorporated in this revision of JS-08 using the new format. As a result of changing the Q-classification of some instruments a revision to FSAR Table 3.10-1 is required.

SAFETY EVALUATION: This safety evaluation concluded that the change did not involve an unreviewed safety question. The changes made by this revision of JS-08 do not constitute a change to any intended design function for any instrument, system, or structure. The change in classification is consistent with the As-Built design documents referenced in the FSAR and with the description of system operations, when given, in the FSAR. The instrument classifications being changed are consistent with their function in accident mitigation. The devices deleted from UFSAR Table 3.10-1 are either non-essential to nuclear safety or do not require specialized dynamic qualification to maintain the currently postulated level of functionality or are previously exempted from specialized dynamic qualification in UFSAR Section 3.10.1.4.1.5. No physical modification to any equipment is being made. No changes to operational process, equipment function, safety level, or system parameters or characteristics are made with this change. The safety function of the instruments are as documented in the As-Built plant design. This is the function performed in the appropriate safety analysis.

The categorization of these instruments is consistent with the current bases of Technical Specifications and plant accident analysis. The change to Table 10.1 is consistent with the current As-Built design documents.

SRASN: NPE-90-068 DOC NO: MNCR-0124-90-R02 SYSTEM: E22

DESCRIPTION OF CHANGE: MNCR 0124-90 was written to document Nuclear Plant Engineering's analysis of a torque switch adjustment required for High Pressure Core Spray (HPCS) valve Q1E22-F004.

REASON FOR CHANGE: This adjustment was necessary as a result of recalculation of the maximum allowable stem thrust (MAST) value based on the ASME Code material stress allowables as part of the continuing motor operated valve program in response to NRC Generic Letter 89-10. The existing torque switch trip point in the closing direction had been previously set such that the new (recalculated) MAST would be exceeded. An Operability Review was conducted to verify that the valve had not been subjected to damaging conditions and to state that the valve was fully capable of performing all required design functions.

The torque switch setting was required to be adjusted to provide valve actuator output thrust between the minimum required stem thrust (MRST) and the MAST. The torque switch adjustment was implemented as required to lower the thrust delivered to the valve. However, subsequent to that adjustment, the required local leak rate test (LLRT) was performed. This test failed to meet the leakage criteria for that valve (most likely due to compression setting of the valve gate and seat surfaces from previous operation at the higher thrust values). The torque switch setting for this valve was then reset to the higher pre-MNCR value to provide the closing force necessary to pass the LLRT. The LLRT was then successfully passed.

NPE reevaluated this situation for acceptability. This evaluation has determined that allowable valve stresses for pressure retaining valve components have not been exceeded and the valve is fully capable of performing all required design functions. In addition, this evaluation concluded that the stress condition for the non-pressure retaining valve part which would only occur during accident conditions is minimal and will not impact the ability of the valve to perform its intended design function. Therefore, the closing torque switch setting may remain at the higher as-left value until startup from the fifth refueling outage.

SAFETY EVALUATION: This safety evaluation concluded that the change did not involve an unreviewed safety question. The FSAR considers various events (accidents) which are postulated to occur in order to determine the plant's capability to control or accommodate potentially damaging process disturbances and component failures. The accidents whose probability may be increased involve only those events which are related to the ability of the subject valve to provide its passive function of reactor coolant pressure boundary integrity or its active functions of emergency core cooling, reactor isolation, and containment isolation.

NPE-90-068
Page 2

Concerning the active functions, the ability of the valve to open or close is not a precursor to any Design Basis Accident or transient. These events include the analyzed loss of coolant accidents, unexpected process or system perturbations, and reactivity events (FSAR 15A.6). The only event discussed in the FSAR directly caused by a malfunction associated with HPCS is an inadvertent HPCS injection. This event is assumed to be the result of an unintended manual pump start via an operator error. Therefore, the stress condition of the valve stem does not change the probability of occurrence for this event (FSAR 15A.6.3.3).

Subparagraph NB-2121(b) of the Code stipulates that the Code requirements do not apply to items not associated with the pressure retaining function of a component such as valve stems. However, subparagraph NB-3546.2(a) establishes that valve stems, stem retaining structures, and other significantly stressed valve parts whose failure can lead to gross violation of the pressure boundary shall be designed so that their primary stresses do not exceed Code allowable values. Thus, the code allowable limits must be applied to that portion of the valve stem penetrating the valve body but need not be applied to the portion outside the body.

It has been determined by calculation that the worst case stresses in the portion of the stem penetrating the valve body are within the Code allowable limits. Only the threaded portion of the stem located outside the body in and just below the actuator were calculated to be stressed above the Code allowable values during worst case conditions. However, these stress levels were determined to be well below the actual material yield strengths such that damage is not predicted. The passive function of the valve is thus maintained during all conditions and the applicable design margins required by the Code for limiting the probability of a passive pressure boundary failure are assured. There are no credible failures of the active functions of this valve which can affect the probability of an accident (e.g., LOCA).

The body-bonnet bolts were determined to be overstressed when using the nominal yield stresses provided in the ASME Codes by Calculation NPE-Q1E22F004, Rev. 1. A review of the valve code data package showed that the originally supplied bolts had a minimum yield stress of greater than 120 ksi, which is higher than the nominal value of 105 ksi. A visual inspection of the valve verified the original bolts were still installed in the valve. Using this information, Revision 2 to this calculation demonstrated that the maximum expected stress in the body-bonnet bolts will be less than the allowable stress provided in the ASME Codes when considering normal operating conditions or abnormal accident conditions. The use of the higher yield stress of the original body-bonnet bolts adds no additional accident precursors which could affect the probability of an accident.

The Q1E22-F004 valve must actively function to mitigate events which require injection of HPCS to prevent fuel or containment damage. These events include loss of coolant accidents as well as loss of normal feedwater supply or reactor cooling systems requiring makeup with HPCS (FSAR 15A.6.3). In such events, the HPCS system must provide design flow at up to design pressure to ensure that FSAR accident analyses assumptions are met (FSAR 6.3). This action requires that the injection valve's active automatic opening function remain available and that, if closed, the valve will reopen when called upon to do so. The valve must also be capable of being closed manually from the control room to isolate the HPCS injection line, if necessary, to minimize radioactive release paths (FSAR Table 6.2-44). Q1E22-F004 also has an active function of automatic closure on a high reactor water level following injection; however, this function is not assumed in any event analysis. Additionally, Q1E22-F004 must maintain its passive integrity so that the HPCS flow path is intact and surrounding equipment necessary to mitigate accident consequences is not damaged.

The increased stress which occurred with the higher torque switch setting was analyzed to have had no detrimental effect on the ability of the valve to perform the above functions. This will continue to be true with the torque switch setting remaining at the higher value. A visual inspection of the valve has confirmed that no damage is present.

To date, no stress limits have actually been exceeded. Body-Bonnet bolt stresses have been determined to be below their allowable stress when actual bolt material yield stresses were evaluated. The stem stresses, while predicted to slightly exceed (2%) the allowable code stress under the most severe accident environment, will remain well below their yield point. Further, use of the ASME code allowable stress is considered only a guideline in this situation. The ASME code (NB-3546.3) states that for components whose failure will result in gross violation of the pressure retaining boundary, the allowable primary membrane stress should be used as the limit. This is not the case in this situation and standard engineering practice allows a higher allowable stress to be applied (AISC, 1.5.1.3.1). Thus, no catastrophic failure, buckling or other deformation is predicted to occur as a result of a the potential overstress. The calculated force required to open the valve remains within the capability of the valve actuator, and the maximum stresses expected to occur during opening remain well below the material allowable. The ability of Q1E22-F004 to perform its active function will not be affected.

Loss of valve integrity from the standpoint of the consequences of a resulting HPCS line break LOCA was evaluated. The conditions identified are not subject to endangering pressure boundary integrity, therefore this analysis is not affected.

The FSAR assumes the availability of the HPCS system (and associated injection valve) to mitigate the consequences of failures of equipment important to safety under various postulated scenarios. These situations include design basis events such as small or large break LOCAs, and events of higher frequences like main steam line isolation or loss of feedwater in which Q1E22-F004 must open to provide spray coolant flow (FSAR 6.3 and 15A.6). These may be coupled with failures of other ECCS or makeup systems. They also include situations under which the valve is required to close manually to mitigate the consequences of equipment malfunctions which could result in the release of radioactive material outside of the containment. In any event, the consequences of such failures will be no more severe under the higher torque switch setting applied to this valve per the MNCR. Therefore, there is no creation of a possibility for an accident or malfunction of a different type than any evaluated previously in the Safety Analysis Report.

The GGNS Technical Specification Bases discuss the function of the HPCS system and the Q1E22-F004 valve in particular. The valve must open and close as required to provide its active function for emergency coolant injection control (TS B3/4.5.1; TS B3/4.5.2) and remote manual isolation of the HPCS injection line (TS B3/4.6.4). It must also satisfy its passive function of pressure retention (TS B3/4.4.3.2).

The margin of safety associated with the valve's primary active function involves the ability to open to provide cooling to prevent exceeding fuel cladding integrity limits. Also implied in the Bases, and assumed in the FSAR (Table 6.3-2; Section 6.3.2.2.1), is that the valve will open in a time period consistent with meeting the required overall system response time of 27 sec., or faster. Because the valve will still perform its intended functions adequately, margins of safety are not affected. The valve will open when required to do so, and surveillance procedures which verify the HPCS system's ability to respond in the necessary timeframe are unchanged. The HPCS system will therefore remain capable of providing its design flow rate within the bounds assumed in the TS Bases and FSAR analyses and the margin of safety to exceeding fuel cladding integrity limits is not reduced.

The margin of safety associated with the passive pressure retaining functions are also not impacted. The Bases discuss the requirement for the valve to remain intact to reduce the possibility of an intersystem LOCA (TS B3/4.4.3.2). Evaluation of the resulting valve stresses, both operating and seismic, under accident conditions shows that the valve is no more likely to fail with the torque switch set as recommended. There are no changes to the surveillance procedures used to verify valve integrity.

Thus, all intended functions of Q1E22-F004 will continue to be performed without degradation, and there is no reduction in the margin of safety discussed or implied in the Bases for the GGNS Technical Specifications.

SRASN: NPE-90-069   DOC NO: DCP 90-0551-S0 & S1-R00   SYSTEM: P41

DESCRIPTION OF CHANGE: This DCP corrects a potential common mode failure of SSW loop A and loop C return header to the SSW cooling tower. Additionally the DCP corrects a lack of design loop C SSW cooling flow to the High Pressure Core Spray (HPCS) room coolers. Loop A and C will be separated by disconnecting the 10" Loop C return pipe from the 24" Loop A return pipe in the Loop A valve room. The Loop C return line will be re-routed to the SSW cooling tower previously designed for Unit 2, Loop A service. The modified Loop C return line will essentially be routed along the existing path for the Unit 2, Loop A return line from the Loop A valve room to the existing Unit 2, Loop A distribution header. All submerged, Unit 2, Loop A, 24" return piping will be replaced with 10" piping for Loop C service.

The SSW cooling tower previously identified for Unit 2, Loop A service will be re-defined for Unit 1, Loop C service. The cooling tower for Unit 1, Loop C service will consist of two cells operating as a natural draft cooling tower. The existing cooling tower fans, gear reducers, drive shaft, and all associated drive shaft components will be removed from the Loop C cooling towers. Modifications to the distribution headers will include replacing the existing spray nozzles with small nozzles, and the repair of all construction welds not previously Code stamped.

The existing leak detection will be modified to create independent loop detection systems. The flow restricting orifice plates in the SSW Loop C piping will be modified to provide design flow to the HPCS Diesel Generator cooling water jackets and the HPCS room cooler. The scope of modifications to the restricting orifice plates is provided in Supplement 1 of DCP 90/0551.

The existing Loop C restricting orifice plates are located on the main supply and return lines. These orifice plates restrict the total loop flow to all components served by Loop C. The flow resistance caused by these plates will be reduced by enlarging the bore of these plates. The core of Q1P41-D014 will be enlarged to line size, which will eliminate all resistance caused by the plate. The plate will remain in the loop to prevent necessitating removal of the flanges. The bore of Q1P41-D013 will be enlarged to cause less flow resistance, but will not be enlarged to line size since some resistance is still needed to limit the total loop flow rate.

New restricting orifice plates will be installed in the parallel branch supply lines to to the HPCS Diesel Generator jacket water coolers. The new orifice plates will serve to reduce the flow to the jacket water coolers in order to force additional flow to the HPCS room cooler.

Drain holes are provided in the modified header to provide passive freeze protection for the header piping. The drain hole size has been reduced from 3/4" (as previously designed for the 16" and 24" pipe) to 1/4". The smaller drain hole size is designed to minimize the amount of hot return water bypassing the cooling tower nozzles. The 1/4" hole size is considered large enough to prevent clogging under normal conditions since the SSW pump suction screen is fabricated from perforated plate with 1/8" perforations. As with the previous design for the existing loops, two drain holes are provided for redundancy in the remote event that a single drain hole should become clogged.

The nozzles selected for the modified header are hollow cone spray nozzles fabricated from brass which is similar to the existing type of nozzles in the loop A & B headers. Hollow cone spray nozzles create smaller size water droplets and have larger internal flow clearances than full cone spray nozzles. The smallest internal passage of the hollow cone replacement nozzles in 1/4", which is larger than the 1/8" perforations in the pump suction screen.

Two types of nozzles will be used in the modified cooling towers. One nozzle type is designed for installation about the header perimeter, while a different type of nozzle is designed for installation on the internal locations of the header. The perimeter nozzles feature a spray cone angle designed to minimize spray against the tower wall which could result in tower fill material bypass. The nozzles located in the interior of the header feature a wide spray cone angle designed to maximize spray coverage overlap. The spray cone angle for both nozzle types is designed to provide sufficient spray area coverage of the fill material in the event of a loss of an adjacent nozzle due to clogging.

A thermal performance calculation has been performed for utilizing the existing tower cells as natural draft cooling towers. The performance calculation indicates that the limiting maximum flow to the tower is approximately 800 GPM per cell, for a total loop flow rate of approximately 1600 GPM. With a total loop flow rate of 1600 GPM, the cooled water temperature is calculated to approach 90°F. The design of the modified Loop C piping will limit the total loop flow to approximately 1000 GPM, which is far less than the allowable 1600 GPM, and will result in cooled water temperatures less than 90°F. Although the design will limit the maximum flow to approximately 1000 GPM, operation between 1000 GPM, operation between 1000 GPM and 1600 GPM is acceptable. The flow difference between cells is calculated to be balanced to within approximately 4 percent such that neither cell will approach the 800 GPM per cell limit under the designed loop flow rate of approximately 1000 GPM.

NPE-90-069
Page 3

REASON FOR CHANGE: Two separate problem areas have been identified concerning the HPCS SSW. The first problem deals with the potential common mode failure of SSW Loop A and Loop C return header to the SSW cooling tower. In a postulated LOCA scenario where the single failure is ESF Electrical Division I, the Low Pressure Core Spray (LPCS) system and the Standby Service Water (SSW) system Loop A would not be available. This would leave the High Pressure Core Spray (HPCS) system (ESF Electrical Division III) as the only available core spray, in addition to the two Low Pressure coolant Injection (LPCI) pumps for long term core cooling.

In the GGNS design, the HPCS service water and the division I service water both return to the SSW cooling tower through the common Loop A spray header. The relatively small return flow of the HPCS service water, without the added SSW return flow from Division I components, would be insufficient to provide effective spray over the SSW Loop A cooling tower fill. After approximately 50 to 60 hours, HPCS service water temperature could exceed the design temperature of 90°F, and the availability of the HPCS system may not be assure.

GE has performed an evaluation which demonstrates ECCS criteria are met assuming no credit for core spray cooling after the initial 50 hours of HPCS operation. Entergy Operations considered this evaluation adequate for interim operation. However, Entergy Operations committed to implement system modifications to attain adequate long term HPCS service water cooling prior to start-up from RF04.

The second problem deals with the lack of design Loop C SSW cooling flow to the HPCS room coolers. The HPCS room cooler and the two HPCS Diesel Generator cooling water jackets are designed in parallel flow paths for SSW Loop C flow. The component with the highest flow resistance is the HPCS room cooler. The high room cooler flow resistance is caused primarily by the lengthy run of 2" and 2-1/2" piping. The HPCS Diesel Generator cooling water jackets and branch piping create very little flow resistance therefore the vast majority of the Loop C flow passes through the cooling water jackets.

The HPCS room cooler is designed to operate with a minimum flow rate of 40 GPM. A Pre-Operational Test documented a measured SSW Loop C flow rate to the HPCS room cooler of 22.2 GPM during 1982. The Pre-Operational test documented a start-up exception to the lack of design SSW flow to the HPCS room cooler based on an evaluation which used the Log Mean Temperature Difference (LMTD) to characterize the performance of the room cooler at the lower flow-rate. The LMTD method was non-conversative for evaluating the HPCS room cooler performance.

The HPCS pump room temperature could reach 166 °F with an SSW Loop C flow rate to the HPCS room cooler of only 20 GPM. Based on a postulated Post-LOCA room temperature of 166 °F, the expected operating life of the HPCS pump motor windings was determined to be approximately 64 days. The documentation of the HPCS Room Cooler flow problem was reported to the NRC.

The system modifications were implemented prior to start-up from RF04 and will provide for the original design SSW flow of 40 GPM to the HPCS room cooler.

SAFETY EVALUATION: This safety evaluation concluded that the change did not involve an unreviewed safety question. The SSW system, containing the plant ultimate heat sink (UHS), is an essential auxiliary supporting system which is designed to remove heat from plant auxiliaries that are required for a safe reactor shutdown.

The modifications made per DCP 90/0551 do not affect the integrity of the PSW makeup line, nor do the modifications affect the 30 day basin inventory. DCP 90/0551 does not alter the existing configuration of the Loop A or Loop C pump seal. DCP 90/0551 does not alter the existing configuration of the Loop A or Loop C return valve located in the Loop A valve room. Supplement 1 to DCP 90/0551 does modify the bore of the previously installed flanged restricting orifice plates in the HPCS SSW supply and return line and provides for additional flanged orifice plates in the supply lines to the HPCS diesel generator jacket water cooler. However the orifice plant modifications made are designed in accordance with ASME Section XI, which meets the original construction code for the HPCS SSW piping. Modifications shall be in accordance with the original construction code except that NA symbol stamping is not required. The safety related portions of the SSW system were originally designed and constructed to the requirements of ASME Section III, Division 1, Class 3. Modifications required by DCP 90/0551 are designed, and installation requirements are specified, to meet the requirements of ASME Section III, Class 3. Compliance with ASME Section XI and the original construction code (ASME Section III) ensures continued pressure boundary integrity of the SSW piping and components.

ECCS electric power loads are rigorously divided into Division 1, Division 2 and Division 3 so that loss of any one group will not prevent the minimum safety functions from being performed. No interconnection modifications are being made per DCP 90/0551 which could compromise redundant power sources.

Separation within the ECCS is such that controls, instrumentation, equipment and wiring is segregated into three separate divisions designated 1, 2, and 3.

Loss of control power or bypass of any piece of equipment in the SSW system is continuously indicated in the control room.

SSW system redundancy will be maintained following implementation of DCP 90/0551. Instrumentation and controls associated with the three logic trains of the SSW system are physically and electrically separated and meet all separation requirements imposed upon redundant safety related circuits. The separation criteria for redundant Class IE circuits and equipment within the ECCS assures that the failure of equipment of one redundant system cannot disable circuits or equipment essential to the operation of the other redundant systems.

The modifications made per DCP 90/0551 will provide for complete separation of Loop A from Loop C therefore providing for independence of operation between the two loops. The failure of either Loop A or Loop C pump will not affect the operation of the remaining basin A pump. DCP 90/0551 does not alter the configuration of SSW Loop B.

DCP 90/0551 does not alter the existing provisions for nonessential system intertie isolation valves or the redundant intertie isolation valves. Valve modifications per DCP 90/0551 are designed per the requirements of ASME Section XI and per the requirements of the design specification for the SSW system. DCP 90/0551 does not alter the design of the Loop A cooling tower fan. The revised design of the HPCS SSW per DCP 90/0551 utilizes natural draft circulation through the Loop C cooling tower, therefore, the revised HPCS SSW does not utilize fans for mechanical draft cooling of the Loop C return water. DCP 90/0551 does not alter the existing diesel generator loading sequence, nor does the DCP add or delete any loads from any diesel generators. The increased HPCS SSW pump flow does alter the actual HPCS SSW pump motor electrical load, however the increased electrical load is bound by existing Division III generator load and fuel consumption analysis.

The modifications per DCP 90/0551 have been evaluated for consequences of moderate energy line breaks.

The components and supporting structures of all modified equipment that are not seismic Category I and whose collapse could result in loss of a required function of the SSW system through either impact or flooding were evaluated to ensure that they will not collapse when subjected to seismic loading.

Radiation monitors are provided on Loops A & B to detect contamination resulting from RHR heat exchanger tube leaks. The provisions for radiation detection in Loops A & B will not be altered by DCP 90/0551. Loop C does not interface directly with contaminated systems, therefore Loop C is not included in the post accident sampling system. However, the modified Loop C will be provided with a local sample station for contamination monitoring in the event that basin A becomes contaminated from RHR heat exchangers via Loop A requiring Loop A isolation while Loop C operation is still required.

The SSW cooling towers , basins, and pump houses are designed to withstand, without a loss of functional capability, the following natural phenomena:

a. Earthquake

b. Maximum probable flood elevation of +103 feet above mean sea level.

c. Tornado wind forces and tornado-borne missiles.

The SSW cooling towers, basins, and pump houses are constructed of concrete walls and roofs at least 2 feet thick. The SSW cooling tower fans are provided with debris protectors to prevent simultaneous failure of the fans from tornado entrained debris. The Loop C cells will retain the missile protection previously afforded to Loops A & B. Flood protection is not altered by implementation of DCP 90/0551 since the Standby Service Water Pump Houses are not affected by the DCP.

Since DCP 90/0551 maintains the structural integrity of the SSW basins and towers, the factor of safety for buoyancy for seismic Category I structures is not altered. The buildings containing ESF components have been designed:

a. to withstand all credible meteorological events and tornados

b. seismically and will remain functional during and following a safe shutdown earthquake (SSE).

c. to protect the ESF systems in the event of a postulated fire.

d. for protection against dynamic effects associated with the postulated rupture of piping.

e. for protection against missiles

Failures associated with the SSW system do not result in the initiation of accidents. The UHS structure is capable of withstanding the effects of the most severe natural phenomena associated with the plant location, other applicable site-related events, reasonable probable combinations of less severe phenomena, and any credible single failure of any man-made structural features without loss of the sink capability to provide the necessary heat rejection. Where protective action is required under adverse environmental conditions during postulated accidents the SSW system components are designed to function under such conditions.

Control room annunciation is provided for leakage from the SSW system. Leakage can also be detected by a high level alarm from any one of the sumps located throughout the plant. Both high alarms and standby sump pump operation signals are monitored by the plant computer.

There are no existing multiple set points in the SSW system. DCP 90/0551 does not create any multiple set points.

Access to all means for bypassing the SSW system is under administrative control. DCP 90/0551 does not alter the administrative control or automatic system design logic. DCP 90/0551 does not alter the total heat rejection loads to the UHS.

The SSW system is designed to perform its required function for all modes of system operation. Previous analyses of system operation for the various modes have determined that Mode IV is the critical mode for evaluating the capability of the SSW system to perform its safety function during single unit operation. Mode IV is defined as a LOCA in Unit 1 coincident with worst single active failure and total loss of offsite power; with Unit 2 non-operational.

The Safety Evaluation for the SSW is affected by implementation of DCP 90/0551 by changing the heat loads delivered to the existing loop A cooling tower. Mode IV cooling requirements for shutdown of Unit 1 have been previously evaluated and are satisfied by SSW Loop B and HPCS Service Water Loop C. Therefore the modification was evaluated using mathematical techniques previously, used for modeling Mode IV heat rejection.

The heat rejection evaluation for the Loop C natural draft cooling tower shows a cold water return temperature of approximately 89.5°F with an ambient air wet bulb temperature of 79°F and an ambient air dry bulb temperature of 100°F. The evaluation was based on a basin water temperature of 90°F, a Loop C flow rate of 1000 GPM, with a constant peak Loop C heat load.

The worst one day of UHS cooling tower demand occurs on the first day of the 30-day period following a LOCA. The actual basin water temperature is at its lowest on this day and will be at a maximum initial temperature of 75°F, assuming highest PSW temperature. The cooled SSW return water mixes with the large basin water volume, resulting in a lower actual SSW pump suction temperature than the 90°F assumed in the evaluation. The cold water return temperature will therefore not actually be as high as 89.5°F.

As the basin level decreases due to drift and evaporation, the Loop C flow rate will decrease. The decrease in Loop C flow will result in a higher Loop C return water temperature which will improve the performance of the natural draft cooling tower. The improved performance of the natural draft cooling tower will result in a lower Loop C cold water temperature returning to the basin. The HPCS SSW performance analysis verifies that with a gradually depleting basin inventory, the basin water will not exceed the design temperature of 90°F.

Losses from the SSW basin inventory result from cooling tower drift and evaporation. The evaporative loss is a function of the meteorological conditions and the return water temperature. Since the applicable meteorological conditions are identical for both the HPCS SSW and the Standby Diesel Generator SSW, the evaporative losses determined in the Mode IV analyses are considered valid for the separate utilization of the - HPCS SSW natural draft cooling tower and the Standby Diesel Generator SSW mechanical draft cooling tower.

The Mode IV estimate for a 0.02 percent drift loss from the Grand Gulf SSW cooling towers has been supported by drift eliminator performance tests conducted by an independent testing firm. The tests were performed on both a test cell and an actual operating cell whose size is similar to those for the Grand Gulf SSW towers. The drift eliminators used on the cells tested were of the same zig-zag design as those used on the Grand Gulf towers. The test results found that drift losses to be less than 0.000018 percent. Based on the results of the tests conducted on the drift eliminators, the 0.02 percent estimated drift loss from the Grand Gulf towers is conservative.

The HPCS SSW natural draft cooling tower will have a lower air velocity through the tower than the previously analyzed existing mechanical draft cooling towers. The lower air velocity should result in a lower drift rate, therefore the drift rate preciously documented for the mechanical draft cooling towers is considered to be conservative for the natural draft cooling tower.

For a 30-day period of operation following plant shutdown a design conditions, the existing Mode IV total integrated water losses resulting from cooling tower drift (of approximately 103,000 gallons) are considered to represent the sum of the losses from the Loop C cooling tower and the Loop A cooling tower since the losses due to the Loop C flow were previously included in the Mode IV analyses.

The HPCS SSW natural draft cooling tower design is such that variations in wind speed and direction may temporarily affect the tower's performance. However variations in wind speed and direction are considered to be sufficiently random and dynamic thus precluding any consistent, extended degradation in tower performance. During the peak heat load period of SSW cooling tower operation, the basin water is at a temperature lower than the maximum allowable water temperature. Because the basin contains a large volume of water, any short period of wind affects or air temperatures higher than the design air temperatures is not expected to cause the basin water temperature to exceed the design limit. There is therefore no reduction in the margin of safety as defined in the basis for any Technical Specification.

SRASN: NPE-90-070 DOC NO: CN-90-0391 SYSTEM: N64

DESCRIPTION OF CHANGE: This change provided for new top connections at the bottom of the 6" pipe EBD-25.

REASON FOR CHANGE: Due to piping interference, a new location was found to install the Fluid Components, Inc. (FCI) model LT-81-4 flow instruments.

SAFETY EVALUATION: There is no increase in the probability of occurrence or in the consequences of an accident or malfunction of equipment important to safety previously evaluated in the Safety Analysis Report. FSAR Section 15.7.1 states the equipment and piping are designed to contain any hydrogen oxygen detonation which has a reasonable probability of occurring. A detonation is not considered as a possible failure. The new flow elements have been purchased to the original design drawings and specifications, therefore the original safety analysis is not compromised. The piping designs issued by this change meet ANSI B31.1 requirements and are qualified for the appropriate deadweight and thermal loads. The piping will function in its intended manner. No other accident precursors evaluated in the UFSAR are affected by this change. Therefore, there is no creation of a possibility for an accident or malfunction of a different type than any evaluated previously in the Safety Analysis Report.

The design change does not affect the operation or function of the offgas system as defined in the bases for any Technical Specification. Therefore, all margins of safety as defined for any Technical Specification thus remain unchanged.

SRASN: NPE-90-071   DOC NO: MCP-89-1098-S00-R00   SYSTEM: M41

DESCRIPTION OF CHANGE: The exhaust valves and ASCO solenoid valves of eighteen (18) Bettis air operated valve actuators were replaced with different Asco solenoid valves. The actuators are an integral part of several butterfly valves. The associated isolation valves are identified as follows; (a) 1M41F007, F008, F036, F037, (b) 1T41F006, 007,
(c) 1T42F003, F004, F019, F020, F011, F012, (d) SZ51 F001, F002, F003, F004, F010, F011.

REASON FOR CHANGE: Three secondary containment isolation damper/valves failed to close within their Technical Specification limit of ≤ 4.0 seconds. The cause of the failure was attributed to the quick exhaust valve (Parker-Hannifin Model 0R50 OR 0R50B) installed on the Bettis air operated valve actuators.

Several system M41 air operated valve actuators have been successfully modified without the quick exhaust valves that had been the cause of the failure. Based on these modifications, the installed ASCO solenoid valves and exhaust valves were replaced with different Asco solenoid valves successfully used before. The new solenoid valves have a larger orifice than the original equipment and use larger actuator air exhaust tubing than the original installation so that the exhaust air flow rate is limited to a large degree by the pressure port size of the air operated valve actuator.

SAFETY EVALUATION: This safety evaluation concluded that the change did not involve an unreviewed safety question. No isolation damper/valve control logic/circuitry has been changed by the MCP. The MCP implementation maintains the required maximum operating time while eliminating a potential valve failure source (quick exhaust valves). The modifications which consist of a different size solenoid valve and larger diameter tubing has been analyzed and determined to be seismically satisfactory. Since the remaining pneumatic components are standard items for this type of installation, successful implementation of the MCP will improve the damper/valve operational reliability. In order to compensate for the exclusion of the quick exhaust valve the resistance of the flow path through the tubing and solenoid valve will be decreased by increasing the tubing and solenoid valve orifice diameters. The use of a solenoid valve with a larger orifice does not increase the likelihood of any failure. The tubing modifications have also been analyzed for seismic concerns and are satisfactory.

A potential source for malfunction of the isolation damper/valves has been identified by the actuator supplier via a 10CFR21 report concerning the use of Mobile 28 grease and Ethylene-Propylene seals within the basic actuator assembly. The affected actuators were rebuilt to eliminate the problem. The solenoid valves, although a potential source for malfunction, have not been identified with any generic failure mode in the present applications.

The isolation damper/valves can only fail to isolate within the ≤ 4 seconds required if the initiation circuitry fails or the pneumatic and mechanical components directly associated with the valve malfunction. The MCP affects only the pneumatic or electro-pneumatic components associated with the valve actuator. Implementation of the MCP will result in the removal of the quick exhaust valves. The remainder of the pneumatic components will be the same as those presently installed except for internal diameters and therefore their potential failure modes will be identical.

Since the only changes involve the pneumatic control flow path for the isolation damper/valves and no changes are made in any isolation control logic or electrical components (other than Asco solenoid valve size), only the performance of the isolation valves requires evaluation. The MCP implementation provides the same valve actuator function while excluding a potential isolation damper/valve malfunction source (quick exhaust valve).

A Calculation was performed to assure that the design does not increase the pneumatic damper/valve blow down time when compared with the present exhaust valve installation. The modification will result in a less complex closure scheme which will maintain the Technical Specification required closure time of < 4 seconds.

SRASN: NPE-90-072 DOC NO: MCP-89-1102 SYSTEM: C91

DESCRIPTION OF CHANGE: This MCP provides electrical details for the installation of a permanent power cable (routed in conduit) between Panel SC91-P890 and disconnect switches 08-1Y91-24 & 26 in the 120-240VAC uninterruptible power panel 1Y91. This involves removal of the existing power cables and installation of new power cables routed in conduit between the two panels.

REASON FOR CHANGE: To provide a permanent power supply in the place of the temporary supply previously being used.

SAFETY EVALUATION: This safety evaluation concluded that the change did not involve an unreviewed safety question. This design change installs permanent power to non-safety related computer panel SC91-P890 fed from disconnect switches 08-1Y91-24 &26 in the 120-240 VAC BOP uninterruptible power supply (UPS) panel 1Y91.

Distribution panel 1Y91 is fed from the station 125 VDC Non-Class 1E battery and battery charges which are connected to one of the class 1E busses. Failure of any of the equipment in the 125 VDC supply circuit enables the static switch to transfer the power source automatically to an alternate source fed from a 480 volt Class 1E AC bus through a transformer. When a LOCA occurs, the Class 1E feed from the load center that feeds the chargers is tripped. Therefore the malfunction of loads to 1Y91 will thus be bounded by a LOP-LOCA.

The implementation of this design change will not affect any equipment identified as the basis for any technical specification. The design adds permanent power to BOP computer panel SC91-P890 from Non-Class 1E uninterruptible power distribution panel 1Y91. The BOP computer and the 120-240 VAC BOP uninterruptible power supply system are not addressed in the GGNS-1 Technical Specifications.

SRASN: NPE-90-073 DOC NO: MCP-89-1103-S00-R01 SYSTEM: P81

DESCRIPTION OF CHANGE: The installed air regulators for starting air regulator valves P81 PCV-F505 A [B] and PCV-F506 A [B] for the HPCS diesel generator are Norgren model # R02-200-RGS-AU. This model regulator has been replaced by a model R11-200-RGSA or a model R08-200-RGSA.

REASON FOR CHANGE: The old model has been discontinued.

SAFETY EVALUATION: This safety evaluation concluded that the change did not involve an unreviewed safety question. The replacement regulators maintain the same form and function as the original model. Mounting hardware will be modified to allow the new model to be installed. Evaluation has shown that the models and the revised mounting hardware will not compromise the original seismic qualifications. The new model number and installation will meet the original form, fit, and function. Therefore the start air pressure regulator valves P81 PCV F505 A[B] and PCV F506 A[B] will function as originally designed. Evaluation has shown that the models and the revised mounting hardware will not compromise the original seismic qualifications.

SRASN: NPE-90-074 DOC NO: MCP-89-1135-S00-R00 SYSTEM: B33

DESCRIPTION OF CHANGE: This MCP disables the non loop manual modes by hardwiring the "Reset Flow Control to Manual" trip. No equipment is to be physically removed or electrically disconnected from it power source.

The only mode of operation of the Recirc Flow Control valves used at GGNS is Loop Manual. This MCP disables all other "non loop manual" modes.

SAFETY EVALUATION: This safety evaluation concluded that the change did not involve an unreviewed safety question. This MCP only disables the non loop manual modes. Continuous operation in loop manual may reduce the probability of a Recirc Flow Control Failure (Increasing Flow) and a Recirc Flow Control Failure (Decreasing Flow). This is because a large proportion of the flow control instrumentation is bypassed when the system is in loop manual mode. With less active equipment, there should be a mathematical decrease in the probability of an equipment failure that could cause these events to occur. The classifications of these events (incidents of moderate frequency) will not be changed. No other evaluated accidents are predicated on a failure of the recirc flow control system and no other systems/system components are affected by this change. Operating in loop manual may decrease the consequences of these events because only one loop is postulated to fail instead of two as in non loop manual modes.

No safety related equipment is affected. This MCP simply forces the recirc flow control system to loop manual by placing jumpers across the contacts of non safety related relays. There is no addition, deletion or modification of any ASME component or pressure boundary involved. There is no addition, deletion or modification of any class 1E component or circuit involved.

Loop manual and non loop manual modes of recirc flow control have been separately evaluated. Both modes of operation have been approved. This MCP simply disables the non loop manual modes. Therefore, there is no reduction in the margin of safety as defined in the basis for any Technical Specification.

SRASN: NPE-90-075 DOC NO: MCP-89-1126-S00-R0-R1 SYSTEM: B21

DESCRIPTION OF CHANGE: MCP 89-1126 replaces the solenoid pilot valves used on the inboard and outboard Main Steam Isolation Valves (MSIV) B21-F022A, B, C, D and B21-F028A, B, C, D. The ASCO dual solenoid valve model NP8323A20E is being replaced with two ASCO model NP8320A185V solenoid valves.

REASON FOR CHANGE: ASCO no longer manufacturers the old model solenoid valves. In addition, GGNS has experienced problems with this solenoid valve model not going to their deenergized position. These failures have been caused by extrusion of the EPDM seating material into the valve body. The mechanism for this failure has been attributed to degradation of the EPDM due to elevated temperature. The valves are not only subjected to a high ambient temperature there are also exposed to a higher temperature rise because both coils are continuously energized. Adding to this failure mechanism is the high seating force continually applied to the seating material by the B solenoid.

A calculation was done and the results indicate that an ASCO NP8320A185 with a 3/32" orifice would minimize the impact on MSIV response time. The new model solenoid valves use viton seating material. Replacing the dual solenoid valves with two single solenoid valves will reduce the expected heat rise by 30 degrees centigrade.

SAFETY EVALUATION: This safety evaluation concluded that the change did not involve an unreviewed safety question. The change performed by this MCP will not alter the MSIV trip logic. Therefore all safety action required by the MSIVs will not be altered. The evaluated event in the FSAR is an increase in reactor pressure due to a MSIV closure. The failure mechanism within the replacement valves for a MSIV closure would be a failure of the seating material to maintain it's pressure seal thus allowing the MSIV to go to the close position. It would be expected that this kind of failure would be similar to the failures of the seating material experienced throughout the industry and in particular here at GGNS. One of the causes for this failure of the seating material is it's exposure to elevated temperatures. Replacing the single dual solenoid valve with two single solenoid valves will reduce the expected valve temperature rise by ≅ 30 °C. An expected contributing factor to the failure of the seating material to maintain it's seal would be the seating forced experienced by the seating material due to the "B" solenoid. Replacing the NP8323A20 valves with two NP8320A185 valves will greatly reduce this seating force. A reduced qualified life for the replacement SVs is expected. This new qualified life is based on the use of viton seating material. Analysis shows that the use of viton will not impair the valve's ability to perform it's safety function when exposed to the expected radiation dose over the qualified life. In addition the new solenoid valves and tubing arrangement has been analyzed to ensure that seismic qualifications have not been compromised. A Stress calculation has been performed to ensure the new tubing configuration will not loose it's pressure retaining capability before during and after a Safe Shutdown Earthquake (SSE).

The absence of the second coil will also reduce the seating force on the valve. In addition the new SVs will use viton not EPDM seating material and have a reduced qualified life. Hiller has noted that the closing spring force is slightly greater for the NP8320 valve. Hiller has also noted that the NP8320 has been successfully used on Hiller operator applications similar to the MSIV. The NP8320 valves are fully qualified and the revised tubing configuration has been fully analyzed to ensure it will function before during and after an SSE. The probability of a failure of the pneumatic/Hydraulic unit to operate their respective MSIV has been decreased by this change. Therefore, there is no creation of a possibility for an accident or malfunction of a different type than any evaluated previously in the Safety Analysis Report.

The operating time of the MSIV is a minimum of 3 seconds and a maximum of 5 seconds. A calculation was performed and has demonstrated that the installation of two NP8320A185 valves with a 3/32" orifice will provide a slightly faster response time of the pilot pneumatic circuit then the presently installed NP8323A20 valve with a 1/16" orifice. Therefore, the maximum time of 5 seconds will still be achievable. From review of past MSIV time response data it can be determined that sufficient adjustment is available to compensate for the slight increase in pilot operating time. Therefore, the minimum operating time is achievable.

SRASN: NPE-90-076 DOC NO: MCP-90-1004-S00-R00 SYSTEM: P41

DESCRIPTION OF CHANGE: The purpose of this MCP is twofold: provide a removable spool piece in the SSW makeup water line to Basin "A" and install the injection line in Basin "A" for the future SSW chemical injection system.

The SSW makeup supply line, 8" JBD-174, will be modified by the installation of 2 pair of flanges just downsteam of valve NSP41F504A. Also, line 3/4" JBD-1205 will require minor design changes to allow installation of the flanges.

The installation of the injection line, JZD-40, for the future chemical injection system, is 2" diameter pipe made of carpenter 20 alloy. It originates outside of the pump house on the north side. This outside portion consists of a blind flange and a plug valve. The line enters and exits the pump house through two new penetrations. It descends to elevation 76' passing through the debris screen to a point between the SSW pump Q1P41C001A-A and the HPCS SW pump Q1P41C002-C. It is located and supported as to preclude any possible failure that could affect the operation of the SSW system.

REASON FOR CHANGE: The removable spool piece in the basin makeup water supply line was provided to allow for the installation of a temporary filter system during the refilling of the basin following a drain down.

Provision was made for the installation of an SSW system. Chemical injection system by a future DCP. The change was made at this time because the "A" basin was drained.

SAFETY EVALUATION: This safety evaluation concluded that the change did not involve an unreviewed safety question. The safety function of the SSW system, containing the plant ultimate heat sink (UHS), is to provide a reliable source of cooling for plant auxiliaries that are essential to a safe reactor shutdown. The SSW system is designed to perform this cooling function following a design basis loss of coolant accident (LOCA) automatically and without operator action assuming a single active failure coincident with a loss of offsite power.

The SSW system original design as described in the UFSAR has not changed as a result of the installation of the described flanges or the injection line. The piping and pipe supports installed by this MCP have been designed to ANSI B31.1 requirements and are qualified as seismic category II/I. Plant operation with this piping installed in the SSW system will have no adverse effect on the functionality of system required to mitigate the consequences of postulated accidents evaluated in the UFSAR.

NPE-90-076
Page 2

The modification of the SSW makeup line by the installation of the flanges will not require a change in operating the system. The installation of the injection line will not impact operation of the system. The addition of the flanges to the SSW makeup line will not change or affect its function. The design of the chemical injection line's discharge sparger, which will be located in the SSW basin sump, is consistent with the design of the debris screen over the sump with respect to preventing particles greater than 1/8" diameter from entering the SSW pump suction. Also, the discharge sparger is located and supported as to preclude any possible failure that could affect the operation of the SSW system.

SRASN: NPE-90-077 DOC NO: MCP-90-1007-S00-R00 SYSTEM: E21

DESCRIPTION OF CHANGE: This MCP changes the makeup water supply to the reference leg of suppression pool level transmitters C61-LT-N402A, E30-LT-N003A, 3C, and 4A. The new supply is from instrument valve E21FX020 located on the Low Pressure Core Spray (LPCS) jockey pump discharge line. The old supply was from E21FX013 located on the LPCS pump discharge line. The supply tubing was rerouted to the new location. E21FX013 will be capped off and abandoned in place.

REASON FOR CHANGE: A LPCS pump start would cause the suppression pool level monitoring transmitters to go into an alarm state, thus making up half the logic required to dump the upper containment pool. This was caused by the pressure surge in the supply water line when the pump started. The new supply is not susceptible to this problem and will keep the reference legs full.

SAFETY EVALUATION: This safety evaluation concluded that the change did not involve an unreviewed safety question. Presently, the seal pots can receive makeup water from either the LPCS pump or the LPCS jockey pump. When implemented, the design change will prevent the LPCS pump from being used for this purpose. This is acceptable, because of LPCS jockey pump is environmentally and seismically qualified. Also, the suppression pool makeup system consists of two independent, 100 percent capacity subsystems which are divisionally separated. Thus, the failure of a single active component (including the LPCS jockey pump) in either subsystem will not cause a loss of suppression pool makeup capability. The valve, tubing and tubing support changes meet all applicable seismic/ASME Section III Class 2 design requirements.

The E21, E30, C61 system operation and function will not change. The instrument valve, tubing and tube supports supplied by this MCP meet all applicable seismic/ASME Section III Class 2 design requirements and will function in their intended manner.

The modification of the valves, tubing and tube supports does not change the limiting conditions for operation applicability or surveillance requirements. The setpoints of the suppression pool level transmitters are not affected.

SRASN: NPE-90-078 DOC NO: MCP-90-1017-S00-R0-R1 SYSTEM: N71

DESCRIPTION OF CHANGE: This design change will install manually-operated "Mud Valves" (N1N71F384 through N1N71F395) in the lateral flumes of the natural draft cooling tower.

REASON FOR CHANGE: The new design facilitates on-stream flushing of the flumes by allowing accumulated sediments to be flushed directly into the cooling tower basin during station operation. Additionally, this capability will help to alleviate structural concerns relative to the flumes and supports structures due to accumulated sediments. The implementation of this design has provided a significantly less laborious and time consuming method of draining and cleaning the flumes during maintenance outages.

SAFETY EVALUATION: As postulated in the UFSAR, gross failure of the circulating water system butterfly valves and/or expansion joints results in flooding inside the Turbine Buildings, Radwaste Buildings, Control Building, and the Unit 1 radwaste pipe tunnel. The Circulating Water System is a closed loop system, and failure of the "mud valve" design would result in water passing directly from the cooling tower flumes into the tower basin, and no additional water would be added to the system. Therefore, no increase in area flooding would occur should these system components fail. The GGNS UFSAR also evaluates the Circulating Water System for potential flooding of safety-related equipment due to failure of a system component. The only safety-related equipment in the vicinity of the condenser room below elevation 116 feet is valve Q1-P44-F116, a secondary containment isolation valve. Failure of this valve due to area flooding will not adversely affect attaining and maintaining a cold safe shutdown. Failure of the flumes or the "mud valves" inside the cooling tower will not increase the probability of flooding, and consequently cannot increase the probability of valve Q1-P44-F116 malfunctioning.

Design installation will be in accordance with required standards and specifications and will enhance flume cleaning and drainage. Area flooding due to system or component failure is the only postulated accident evaluated for the N71 system. Gross failure of this design will result in water and debris being deposited in the tower basin. This situation could cause a decrease in system performance, but will not create a possibility for an accident or malfunction of a different type than any evaluated previously in the Safety Analysis Report.

In addition, the design does not involve installed instrumentation that is used to detect, and indicate in the control room a significant abnormal degradation of the reactor coolant pressure boundary. The design does not involve a process variable that is an initial condition of a Design Basis Accident or Transient Analyses that either assumes the failure of, or presents a challenge to the integrity of a fission product barrier. The design does not affect a structure, system, or component that is part of the primary success path and which functions or actuates to mitigate a Design Basis Accident or Transient that either assumes the failure of, or presents a challenge to the integrity of a fission product barrier.

SRASN: NPE-90-079 DOC NO: MCP-90-1020-S00-R00 SYSTEM: N22

DESCRIPTION OF CHANGE: This MCP replaces a leaking elbow downstream of valve N22F098 and relocates a restricting orifice into a straight section of piping.

REASON FOR CHANGE: The leak in the elbow was caused by erosion from the restricting orifice which was located between the elbow and valve N22F098. Moving the restricting orifice into a straight section of pipe will eliminate the erosion effect on the piping system.

SAFETY EVALUATION: The modifications provide for the repair of a leaking elbow and the relocation of a restricting orifice to eliminate the existing erosion problem. The piping is supported to dead weight loads only, since it is installed in the portion of the Turbine Building, containing no safety related equipment. The Condensate Cleanup System serves no safety function. Systems analysis has shown that failure of the Condensate Cleanup System will not compromise any safety related systems or prevent reactor shutdown. The operation or function of the Condensate Cleanup system, as analyzed in the FSAR, is not affected by the modifications of this MCP. The design change by this MCP is non-safety related. The modifications made by this MCP will not affect the N22 system. The piping designs have been designed to ANSI B31.1 code requirements. The system will function in its intended manner.

SRASN: NPE-90-080 DOC NO: MCP-90-1042-S00-R00 SYSTEM: E22

DESCRIPTION OF CHANGE: An annunciator was installed for alternative visual indication of both "HPCS Initiated" and "HPCS High Water Level Seal-In". These Hi Pressure Core Spray annunciators will utilize spare contacts of existing Division 3 relays as signal sources. The power through the relay contacts will be 125 VDC from a Division 3A circuit and will be wired into existing division 3-to-Nondivision isolators. All wiring on the Q-side of the isolator will be marked as Division 3A when installed and will be routed and separated the same as Division 3 circuits. The power source for the output side of the isolator and the annunciator input is non-divisional. This wiring will be designated as non-divisional upon installation and will be separated from all divisional circuits.

REASON FOR CHANGE: The only indications available to identify these two plant conditions (i.e HPCS initiated, HPCS hi water level seal-in) were single element incandescent lamps. Should either of these conditions have occurred and should the indicator lamps be blown, it could not be easily determined if the condition(s) had been reset.

SAFETY EVALUATION: This annunciator will be utilized as alternative visual indication of "HPCS Initiated" and "HPCS High Water Level Seal-In". Only existing equipment will be utilized for this design change. Only spare conductors of existing cables or jumper wires added inside control room panels will be utilized. The appropriate divisions of electrical power have been utilized for both the input and output of the electrical isolator. The added electrical loads will be intermittent in nature and the input to the divisional side of the isolator is current limited. Failure of the Division 3A circuit utilized in this design change has been previously concluded to have no adverse affects on the safety performance of the HPCS System.

All wiring will be routed, separated, and identified in accordance with the appropriate reg guide. A failure in this power circuit can not propagate in Division 1 or 2 nor can electrical failure in Division 1 or 2 propagate into this power circuit due to this design change.

This design change will not affect the HPCS System in consideration to items such as flow, chemistry, setpoint, capacity, level, or pressure. Failure of the Division 3A circuit utilized for this MCP has been previously concluded to have no adverse affects on the safety performance of the HPCS System. None of the affected equipment will be required to operate outside of their designed ratings.

SRASN: NPE-90-081 DOC NO: MCP-90-1054-S00-R00 SYSTEM: E12

DESCRIPTION OF CHANGE: MCP 90-1054 adds 0-100% valve position indication in the Control Room for valve 1E12-F424. This valve is the flow control valve for the Alternate Decay Heat Removal System (ADHR). This MCP also provides for routing/termination of Balance of Plant (BOP) cables for future flow monitoring instrumentation for ADHR with indication in the Control Room.

REASON FOR CHANGE: To enhance the long term viability of the ADHR system.

SAFETY EVALUATION: Connecting the position indication on this valve and routing/termination of BOP cables for system flow indication do not change the original design intent of any component, system or structure. The addition of the position indication is not required to support the safe shutdown of the reactor or to perform in the operation of reactor safety features nor is the addition of flow indication. The changes made by this MCP do not prevent any equipment relied upon to mitigate the consequences of any evaluated transient or accident from performing its safety function.

All structures, systems and components added or modified by this MCP have been designed to meet all applicable requirements and thus no new failure modes are created.

The margins of safety as defined in the bases for the Technical Specifications are not changed by the addition of this position indication. The addition of the position indicator to the valve does not change the original design intent of any equipment.

SRASN: NPE-90-082 DOC NO: MCP-90-1064-S00-R00 SYSTEM: E12

DESCRIPTION OF CHANGE: MCP 90/1064 adds the equipment necessary to provide monitoring capabilities for Alternate Decay Heat Removal (ADHR) system flow, pump suction pressure, and pump discharge pressure. System flow will be provided in the control room. The control room indication will be driven from a differential pressure transmitter connected to an annubar flow sensor. The pressure indication will be provided by local instrumentation. All instrumentation is powered from non divisional Balance Of Plant (BOP) power.

REASON FOR CHANGE: To enhance the long term viability of the ADHR system.

SAFETY EVALUATION: The instrumentation installed will provide a monitoring function only. The instrumentation installed in a pressure boundary application is seismically analyzed for structural integrity and is acceptable for use in safety related pressure boundary. All instrument tubing is installed to seismic category I design requirements. The modifications implemented by this MCP will not change any design criteria or functions of the ADHRS. Failure of the components modified or added by this change will not initiate or prevent initiation of any seismic category I component, system, or structure.

The technical specification contains the administrative controls for the operation of the ADHRS. The addition of the flow and pressure indication will not impact the operating controls of the ADHRS. Therefore, the margin of safety as defined in the basis for any technical specification will remain unchanged.

SRASN: NPE-90-083 DOC NO: MCP-90-1055-S00-R00 SYSTEM: E12

DESCRIPTION OF CHANGE: MCP 90/1055 adds two manually operated valves which in effect, relocates the Alternate Decay Heat Removal system high point vent valves from an elevation of approximately 17' above floor elevation to approximately 6" above floor level. The new valves will be the new safety to non-safety boundary of the vent system. The old valves (E12F418 and E12F427) will be left in place but will remain open during system operation.

REASON FOR CHANGE: To increase accessibility of the manually operated vent valves, thereby enhancing ADHRS operability and increasing plant worker safety.

SAFETY EVALUATION: This safety evaluation concluded that the change did not involve an unreviewed safety question. The installation of the vent valves at an accessible location will not affect the operation or function of the E12 system as described in the FSAR. This design change provides an easier method of venting the system and will not result in the creation of any new failure modes. The piping and pipe support changes will function in their intended manner.

The safety related piping and pipe supports designs meet ASME Section III requirements and are qualified as seismic category I. The non-safety related piping and pipe supports meet ANSI B31.1 requirements and are qualified as seismic category II/I. The addition of the piping and pipe supports does not affect the integrity of the interfacing piping systems or any safety system. The piping and pipe supports will function in their intended manner.

The installation of the piping and pipe supports made by this MCP to the E12 system will not change the function or operation as defined by any bases for the Technical Specification, therefore, the margin of safety is not reduced.

SRASN: NPE-90-084 DOC NO: MCP-90-1056-S00-R00 SYSTEM: E12

DESCRIPTION OF CHANGE: MCP 90/1056, Rev. 0 will allow liquid sampling capabilities on the RHR and PSW (P44) portions of ADHRS. Sample element, P44-SE-N093 and sample element, E12-SE-N195 will be routed to a new sample sink located in the Auxiliary Building, Elevation 93'-0", Area 10. The E12 sample will require a new penetration, AJ-86A, in the north wall of Room 1A116 and a sample cooler to be utilized. Cooling water to the sample cooler will be supplied from the CCW system. The sample sink drain will be routed to an existing DRW drain.

REASON FOR CHANGE: To make the sampling required by Technical Specifications easier to obtain.

SAFETY EVALUATION: This design change provides a method of obtaining liquid samples of the E12 and P44 portions of ADHRS. This change to the systems (E12 and P44) will not affect their normal operation or function. The safety related piping and tubing designs meet ASME Section III requirements and are qualified as Seismic Category I. The non-safety related piping, pipe support and tubing meet ANSI B31.1 requirements and are qualified as Seismic Category II/I. The sample sink support has been designed to withstand the applicable seismic loads to preclude any II/I hazards. No seismic II/I hazards or pipe break concerns will be created by the implementation of this MCP. The addition of the pipe, pipe supports, tubing, and tubing supports does not affect the integrity of the interfacing systems or any safety system. The piping, pipe supports, tubing and tubing supports will function in their intended manner.

No seismic II/I hazards or pipe break concerns will be created by the implementation of this MCP. No new failure modes are being created. Therefore, there are no unresolved safety questions associated with this change.

The bases for Technical Specification 3/4.7.7 is to limit fire damage by preventing a single fire from involving more than one safety related fire area prior to detection and extinguishment.

The aforementioned penetration provides a 3-hour fire rated closure which is an equivalent rating to the affected barriers.

The implementation of MCP 90/1056 involving E12, P44, and P42 systems will not change the function or operation as defined by any bases for the Technical Specifications, therefore, the margin of safety is not reduced.

SRASN: NPE-90-085 DOC NO: MCP-90-1063-S00-R00 SYSTEM: P44

DESCRIPTION OF CHANGE: MCP 90/1063 increased the size of the piping immediately downstream of Plant Service Water Flow Control valve P44F513 from 14" to 24". This valve is the temperature control valve for the Turbine Building Cooling Water System (TBCW).

REASON FOR THE CHANGE: A pin hole leak had developed downstream of the valve and significant erosion was discovered upstream and downstream of the valve. This erosion appears to be a result of high velocity flow because of the line size reduction. The increase in piping size from 14" to 24" was made to reduce the flow velocity to an acceptable level.

SAFETY EVALUATION: The modifications provide for the repair of a leaking reducer and to increase the pipe line size to eliminate the existing erosion problem. The Plant Service Water System serves no safety function. System analysis has shown that failure of the Plant Service Water System will not compromise any safety related systems or prevent reactor shutdown. The operation or function of the Plant Service Water System, as analyzed in the FSAR, is not affected by the modification of this MCP.

Tho designs installed by this MCP meet ANSI B31.1 code requirements. The piping is supported to dead weight loads only since it is installed in the portion of the Turbine Building containing no safety related equipment. Increasing the pipe size will not impact operation of the Plant Service Water System (P44) and will eliminate the erosion effect on the piping. The system will function in its intended manner.

SRASN: NPE-90-086 DOC NO: MCP-90-1073-S00-R00 SYSTEM: P44

DESCRIPTION OF CHANGE: The objective of this MCP will be to remove valve N1P44F925 and replace it with a flanged branch line for hydrolyzing.

REASON FOR CHANGE: The fourway valve, P44F925 on the supply/return Plant Service Water (PSW) piping to the Drywell Chillers is obstructing flow. This valve was originally installed to provide on-line flow reversal capabilities for an automatic tube cleaning system on the cold side (PSW) of the Drywell Chillers. However, performance of the cleaning system was suspect and the cleaning system was subsequently removed and the valve was abandoned in place.

SAFETY EVALUATION: This safety evaluation concluded that the change did not involve an unreviewed safety question. The removal of valve P44F925 will not affect the operation or function of the system since the automatic tube cleaning system for the drywell chillers has been previously deleted and the valve's flow reversal function is no longer required. The affected system in this MCP is non-safety related. The failure of the affected system will not compromise any safety related system or component and will not prevent reactor shutdown. The modification made by this MCP will not affect the analysis of the system as described in the FSAR.

The design installed by this MCP meets ANSI B31.1 Code requirements. The piping is supported to dead weight loads only, since it is installed in a portion of the Auxiliary Building where no II/I hazards exist. Removing the valve (N1P44F925) will not impact operation of the Plant Service Water System (P44) and will eliminate the flow obstructions on the system.

The modification made by this MCP to the P44 System will not change the function of operation as defined in any Bases for the Technical Specifications; therefore, the margin of safety is not reduced.

SRASN: NPE-90-087 DOC NO: MCP-90-1097-S00-R00 SYSTEM:

DESCRIPTION OF CHANGE: MCP 90/1097 provides for the inspection and repair, as necessary, of pipe supports in SSW "B". Additionally, this MCP provides for the removal of non-essential basin piping. MCP 90/1097 was being developed to: 1) provide inspection requirements and required repair procedures for pipe supports in SSW Basin B; and 2) as an alternate, remove piping and supports in SSW Basis B which do not impact Unit 1 operations. More specifically, the piping to be removed is as follows:

1). Portions of the following Unit 2 SSW Basin B piping and associated supports:

   a. Loop C supply from pump discharge to basin wall

   b. Loop B return from basin wall to Q2P41G014A01

   c. Loop B return from Q2P41G014A01 to cooling tower cell

   d. Q2P41G014A01 can only be removed if both partials listed as "b" and "c" above are removed.

2). Portions of the Unit 2 SSW Basin B small piping, instrumentation and associated non-standard supports.

3). Basin B Sodium Hypochlorite and acid piping, supports, and spray headers downstream of valves SP41AVF505B and SP41AVF506B.

REASON FOR CHANGE: Inspection of the SSW "A" basin indicated the potential for corroded pipe hangers in "B" SSW basin.

SAFETY EVALUATION: This safety evaluation concluded that the change did not involve an unreviewed safety question. The Sodium Hypochlorite System is not safety related and has never been utilized. The removal of components as identified in MCP 90/1097 will not compromise any safety related system or components or prevent a safe reactor shutdown.

The chlorination system (N72) is not safety related and the only safety related system which it is connected to is the SSW system. The design function of the SSW system is not changed by the implementation of this MCP and no new failure modes are created.

The GGNS Unit one Technical Specifications do not mention the Sodium Hypochlorite System and the requirements specified in the Technical Specifications are not impacted by the implementation of this MCP.

SRASN: NPE-90-088 DOC NO: MCP-90-1098-S00-R00 SYSTEM: E51

DESCRIPTION OF CHANGE: MCP 90/1098 replaces 1E51N052 due to equipment malfunction. Originally, this device was a Rosemount 1151GP7D52T0003PB transmitter. A Rosemount 1151GP7D22T0003PB transmitter is being installed in its place. These transmitters have all the same characteristics except the "D22" has a stainless steel process flange versus a nickel plated carbon steel process flange on the original.

Rosemount 1151GP7D22T0003PB transmitters have been qualified for use inside or outside containment. The 1151 transmitters are commercial grade transmitters purchased by General Electric who dedicated them for nuclear power applications. Qualification of these transmitters was accomplished by testing performed on 1151 transmitters and similarity arguments to 1152 transmitters.

REASON FOR CHANGE: The old transmitter is no longer available.

SAFETY EVALUATION: An engineering evaluation was done which concluded that the device cannot fail in such a manner as to degrade the Class 1E power source. Therefore, these devices can be classified as Category C (equipment that will experience environmental conditions of design basis accidents through which it need not function for mitigation of such accidents and whose failure is deemed not to be detrimental to plant safety or accident mitigation; it need not be qualified for any accident environment). Further, there are no unresolved safety questions associated with this change.

The engineering evaluation done indicated that the electrical failure modes and effects and concludes that no electrical failure of this device would degrade the Class 1E power supply. Therefore, there is no reduction in the margin of safety as defined in the basis for any Technical Specification.

SRASN: NPE-90-089 DOC NO: MCP-89-1112-S00-R00 SYSTEM: P41

DESCRIPTION OF CHANGE: MCP 89/1112 caps the SSW basin overflow drain lines and documents the acceptability of the slight movement of the missile shield wall.

REASON FOR CHANGE: The missile shield structures on the SSW pumphouses and valve rooms had settled allowing the shield structures to separate from the main SSW structure. This movement damaged the overflow drain lines.

SAFETY EVALUATION: This safety evaluation concluded that the change did not involve an unreviewed safety question. The movement of the shield wall is minor and the shield walls were designed as separate structures to provide missile protection for the doors. The angle required for any small missile to enter through the crack is such that the missile would hit the concrete wall or slab and pose no safety concerns. Since the shield walls have been designed as a separate structure, this movement does not impose any additional loads to the SSW structures. Capping of the overflow line will not, by itself, create the possibility of an accident since the basin level is automatically maintained. However, if a malfunction of the basin level controller was to occur causing excessive make-up, the basin could overflow. This condition is bounded by the probable maximum precipitation (PMP) event.

Since shield walls function is maintained, there is no reduction in the margin of safety. Capping of the basins overflow line will not affect the minimum basin water level of 130' 3" MSL as required by Technical Specifications because the only path now available for overflow on the basins will be at the basin slab of 133' MSL.

SRASN: NPE-90-090 DOC NO: NPEAP-807, 320, 332 SYSTEM:

DESCRIPTION OF CHANGE: Nuclear Plant Engineering Administrative Procedure (NPEAP) 807, 320, and 332 will govern the dispositioning and corrective action via the 10CFR50.59 Safety Evaluation/Applicability Screening for all drawing changes made in response to a QDR, Drawing Revision Notices (DRNs) and Drawing Revision Requests (DRRs).

REASON FOR CHANGE: The categories for the drawings changes addressed by this safety evaluation for the above documents are as follows:

1) Editorial changes

2) Device numbers (valves, breakers, penetrations, etc.) except for those specifically addressed in Technical Specifications.

3) Valve position Identifiers - except for those specifically indicated in Technical Specifications.

4) Electrical contact position identifiers - except for those specifically identified in Technical Specifications.

5) Increase in level of detail shown on drawings, i.e., addition of instrument root valves of Piping and Instrumentation Diagrams (P&ID[s]).

SAFETY EVALUATION: This safety evaluation concluded that the change did not involve an unreviewed safety question. The actions described are drawing changes only and have no physical impact on plant components, structures or systems. These changes have no affect on the operations or functions of plant facilities nor its reliability.

SRASN: NPE-90-091 DOC NO: NPEFSAR-90-0044 SYSTEM:

DESCRIPTION OF CHANGE: NPEFSAR 90-044 corrects the specified maximum closure times in UFSAR table 5.2-5. This change deletes the specified maximum closure times which are not based on an analytical limit. Specifically the change to table 5.2-5 will bring it into agreement with Table 6.2-44 of the UFSAR and with Table 3.6.4-1 of the Technical Specifications. The maximum stroke times for valves without analytical limits are governed by the ASME Section XI Inservice Testing (IST) program.

REASON FOR CHANGE: UFSAR Table 5.2-5 gives a description of pumps and valves which are part of the reactor coolant pressure boundary (RCPB). Maximum closure times are listed for those valves equipped with motor operators. UFSAR Table 6.2-44 gives a description of containment isolation valves and lists the maximum closure time if based on an analytical limit. Valve stroke times with no analytical limit are not included in table 6.2-44.

Valves which are containment isolation and part of the RCPB are listed in both Table 5.2-5 and Table 6.2-44. Some discrepancies existed between these tables with regard to the stroke times listed. Specifically, the "maximum closure tines" in Table 5.2-5 did not agree with the "analytical isolation times" in Table 6.2-44 for the following valves:

RHR Shutdown Cooling	E12F009
Suction	E12F008
Main Steam Isolation	B21F022
	B21F028
RWCU	G33F001
	G33F004

Table 5.2-5 contains two valves which have incorrect maximum closure times listed. These valves are not containment isolation valves and are not listed in Table 6.2-44. The subject valves are:

RWCU	G33F250
	G33F251

Table 5.2-5 also contains maximum closure times for certain valves which have no analytical isolation time. The following valves have non-analytical closure times listed in Table 5.2-5:

RHR Head Spray	E12F023
	E12F394
Main Steam Drain	B21F016
	B21F019
	B21F067
RCIC Steam Supply	E51F076
RWCU Pump Discharge	B33F019
	B33F020

These valves are also listed in Table 6.2-44 but are not included in Note (d) as having analytical isolation times. Since the stroke times for these valves do not represent an analytical limit, the stroke times should not be listed in Table 5.2-5.

SAFETY EVALUATION: This safety evaluation concluded that the change did not involve an unreviewed safety question. The changes to FSAR Table 5.2-5 will not result in any change to the design or function of the associated valves. The analytical stroke times are revised to reflect the correct values as per UFSAR Table 6.2-44 and TS Table 3.6.4-1. Non-analytical stroke times which are being deleted from Table 5.2-5 are not used in any accident analyses. No physical modification to any plant equipment is involved.

The changes to FSAR Table 5.2-5 do not require any new safety analyses or impact any existing safety analyses. The analytical stroke times which are affected by this change are revised in order to reflect the correct values which are listed in TS Table 3.6.4-1 and FSAR Table 6.2-44. The non-analytical stroke times which are being deleted from FSAR Table 5.2-5 are not used as a basis for any safety analysis and are governed by the IST Program. The ability of the valves to perform their required active function will continue to be verified by testing in accordance with the IST Program. Therefore, this change will not reduce the margin of safety as defined in the basis for any Technical Specification.

SRASN: NPE-90-092 DOC NO: MNCR-89-00293 SYSTEM: D17

DESCRIPTION OF CHANGE: This safety evaluation reevaluated the environmental conditions for certain post accident monitoring equipment. The equipment consists of the Eberline AXM-1 accident range monitors and the Air Monitor Corporation (AMC) redundant stack flow monitors. The equipment was established to be in a mild environment and are therefore exempt from being environmentally qualified per 10CFR50.49/NUREG 0588. The equipment was deleted from the GGNS Environmental Qualification Program on this basis. No physical change was made to any equipment.

REASON FOR CHANGE: To correct the environmental condition designation of this equipment and delete the equipment from the GGNS Environmental Qualification Program.

SAFETY EVALUATION: This safety evaluation concluded that the change did not involve an unreviewed safety question. No change to the plant or plant procedures is being made. The equipment performs no safety function and is merely being deleted from the GGNS EQ Program because the environmental conditions have been determined to meet the definition of "Mild Environment". No change is being made to the equipment and appropriate separation from safety systems already exists.

Deleting these items from the GGNS EQ Program will not reduce the margin of safety as defined in the basis for any Technical Specifications since the items perform no safety function, are separated from Class 1E power, and are located in mild environments post accident.

SRASN: NPE-90-093 DOC NO: NPEFSAR 90-0056 SYSTEM:

DESCRIPTION OF CHANGE: This changes UFSAR table 3.2-1 to accurately reflect the as-built configuration of High Pressure Core Spray (HPCS) Diesel Generator auxiliaries.

REASON FOR CHANGE: During the licensing of the Division III Diesel Generator the NRC established that, as a minimum, all piping and valves in the engine skid mounted portions of the lube oil subsystem, the jacket water subsystem, the starting air subsystem and the fuel oil subsystem which were not designed in accordance with ASME Section III (i.e. Quality Group C) be upgraded from Quality Group D to Quality Group D Augmented. MP&L committed to impose Quality Group D Augmented design requirements on the engine skid mounted components (i.e. piping, valves, pumps, etc.) associated with the HPCS Diesel Generator starting air, lube oil and fuel oil subsystems. MP&L also committed to hydrostatically leak test the engine skid mounted piping in the lube oil, fuel oil and starting air subsystems in accordance with ANSI B31.1 (i.e. 1.5 times the design pressure) even though the NRC required a hydro of only 1.25 times the design pressure which is consistent with Section III of the ASME code. In addition MP&L committed to impose the design requirements of ASME Section III on: (A) the engine skid mounted components associated with the HPCS Diesel Generator jacket water subsystem and (B) the off-engine piping and accessories (i.e. exhaust silencers, intake air silencers and intake air filters) in the combustion air intake and exhaust subsystem. Although MP&L committed to design the jacket water and combustion air intake and exhaust subsystems to the Codes applicable to Quality Group C (i.e. ASME Section III, Class 3), the NRC had agreed to accept Quality Group D Augmented (Ref. MAEC 75/36). The design of the piping in jacket water subsystem has been evaluated against the guidance of ANSI B31.1 and it has been concluded that the ANSI B31.1 requirements have been satisfied. The piping in the combustion air intake and exhaust subsystem are designed and installed in accordance with ANSI B31.1 as Seismic Category I piping.

SAFETY EVALUATION: This safety evaluation concluded that the change did not involve an unreviewed safety question. The engine skid mounted auxiliaries (i.e. piping, valves, pumps, heat exchangers, tanks, etc.) on the HPCS Diesel Generator have been evaluated/analyzed against each of the Quality Group D Augmented requirements, as specified in MAEC 75/36, and it has been concluded that the design of the subject systems comply with the intent of the Quality Group D Augmented requirements. Changing UFSAR Table 3.2-1 to accurately reflect the as-built configurations of HPCS Diesel Generator auxiliaries will not jeopardize the ability of the HPCS Diesel Generator to perform its design safety function. No new failure modes have been introduced since the skid mounted auxiliaries are in compliance with the Quality Group D Augmented requirements.

The GGNS Unit One Technical Specifications do not address the quality group classification nor the codes and standards used to design and install the skid mounted auxiliaries on the HPCS Diesel Generator. Therefore, there is no reduction in the margin of safety as defined in the basis for any Technical Specification.

SRASN: NPE-90-094 DOC NO: NPE'SAR 90-0021 SYSTEM:

DESCRIPTION OF CHANGE: UFSAR 6.2.1.1.5.8 addresses a failure in the LPCI injection check valve (E12F041A/B) during transfer from injection into the vessel (LPCI) mode to the containment spray mode. The analysis was conducted to determine the amount of containment pressurization which could occur due to postulated backflow through a failed open LPCI check valve into the containment spray piping. Backflow can only occur during the time it takes the LPCI injection valve to close and while the containment spray header valve is opening. The analysis assumed 18.5 seconds for the E12F042A/B to close and resulted in a 0.8 psi increase in containment pressure. Initial containment pressure was assumed to be 9.0 psig, therefore, the total pressure as a result of the check valve failure is 9.8 psig which is well below the containment design pressure of 15 psig.

An analysis was done using the same methodology, but assuming a stroke time of 30 seconds to ensure that the conclusions reached in the original analysis were still valid.

REASON FOR CHANGE: Valve stroke times associated with the Inservice Testing Program potentially conflicted with times in the UFSAR.

SAFETY EVALUATION: The revision of UFSAR 6.2.1.1.5.8 clarifies that the 18.5 second closure time is an assumption based on GGNS startup data, and specifies that this analysis has been further evaluated up to a stroke time of 30 seconds. The analysis in section 6.2.1.1.5.8 is an evaluation of the failure of the LPCI injection check valve (E12F041A/B) during transfer from injection into the vessel to the containment spray mode. An increase in the time allowed to close E12-F042A/B from 18.5 seconds to 30 seconds will increase containment pressure from 9.8 psig to approximately 10.8 psig, which is bounded by the maximum calculated accident pressure of 11.5 psig as listed in UFSAR Table 6.2-13.

The revision to UFSAR 6.2.1.1.5.8 does not reduce the margin of safety as defined in the basis for any technical specification. The margin of safety is established by a more bounding analysis resulting in a higher accident containment pressure as specified in UFSAR Table 6.2-13. Technical Specifications do not include a time requirement for the close direction for E12-F042A/B since these valves do not receive an automatic containment isolation signal.

SRASN: NPE-90-095 DOC NO: CN-90-0268 SYSTEM:

DESCRIPTION OF CHANGE: CN 90-0268 provides for an interface device between the Plant Paging System and the telephone system in the second level of the M&E building.

REASON FOR CHANGE: The installation will allow M&E building personnel the use of Plant Paging System through their telephones.

SAFETY EVALUATION: This safety evaluation concluded that the change did not involve an unreviewed safety question. All work associated with the Plant Paging System and the telephone system are non-safety related. All installation in M&E Building are non-seismic since this building does not contain any safety related equipment. Power to the P.A. system are from BOP batteries D & E. The Plant P.A. and telephone systems are independent and electrically separated from all other class 1E circuits.

SRASN: NPE-90-096 DOC NO: NPEFSAR 90-0042 SYSTEM:

DESCRIPTION OF CHANGE: The stroke times for the motor operated valves in the Main Steam Isolation Valve Leakage Control System (MSIV-LCS) are given in the UFSAR section 6.7.1.3.1 as about 5 seconds. The actual stroke times based on operating history are between 7 and 10 seconds. A revision of the UFSAR is required to correct this discrepancy. The specific valves covered by these requirements are:

E32-F001A, E, J, N
E32-F002A, E, J, N
E32-F003A, E, J, N
E32-F006
E32-F007
E32-F008
E32-F009

REASON FOR CHANGE: The maximum stroke times of 15 and 30 seconds are based on a system process limit which will cause the inboard system to trip if adequate flow is not established within 30 ± 5 seconds. The outboard system has no low flow trips associated with it's control circuitry. Therefore, a maximum valve open stroke time of 15 seconds will allow flow to develop in the system before the minimum trip setpoint of 25 seconds is reached. The 5 second stroke time currently in subsection 6.7.1.3.1 of the UFSAR has no analytical basis. Therefore, revision of this subsection is necessary to eliminate this incorrect valve stroke time.

SAFETY EVALUATION: This change revises the MSIV-LCS motor operated valve stroke times from about 5 seconds to 15 to 30 seconds. The correct valve stroke times of 15 to 30 seconds have always been specified in the MSIV-LCS Design Criteria and General Electric Process Diagram. For this reason, this UFSAR change request only corrects an error in the UFSAR and does not change any operational parameters or design requirements of the MSIV-LCS.

This change does not introduce any new operational parameters or design requirements to the MSIV-LCS or any other system. No change to any physical system will be made.

Actuation of this system will be by operator action no sooner than 20 minutes following a postulated design basis LOCA. In addition, Table 3.6.4-1 of the Technical Specifications has not included any maximum valve isolation time for any of the motor operated valves in the MSIV-LCS. This change to subsection 6.7.1.3.1 of the UFSAR also will not result in any change to the required valve stroke times specified in the MSIV-LCS Design Criteria and General Electric Process Diagram. For these reasons, this change will not reduce the margin of safety as defined in the basis for any technical specification.

SRASN: NPE-90-097 DOC NO: EER-90-6388 SYSTEM: B21

DESCRIPTION OF CHANGE: Engineering Evaluation Request (EER) 90-6388 requested that temporary lead shielding be attached to certain portions of the packing leak-off lines from the B21F028 valves going into the standpipe drain which runs into the Reactor Core Isolation Cooling (RCIC) room. Calculations were performed on the subject piping with the added weight of the lead shielding. These calculations show that the structural integrity of the subject piping with the temporary lead shielding will be maintained in the unlikely event of an operating basis earthquake (OBE) or a safe shutdown earthquake (SSE). All applicable ANSI code stress allowables are met. Therefore, the operability of the system in Operating Modes 4 and 5 is not affected by the temporary lead shielding attached to the pipe.

Based on the above analysis, the temporary lead shielding was installed on the pipe during Operating Modes 4 and 5. No other lead shielding or any other additional weight could be attached to the piping out to the first anchor while this shielding was attached. This temporary shielding was installed during Operating Modes 4 and 5 only, and was removed prior to restart after RF04. Temporary addition of lead shielding does 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 temporary lead shielding is removed.

REASON FOR CHANGE: To reduce radiation exposure to personnel performing work in this area. The lead shielding will be installed during Operating Modes 4 and 5 only, and must be removed prior to restart.

SAFETY EVALUATION: This safety evaluation concluded that the change did not involve an unreviewed safety question. Since these temporary changes do not affect the structural integrity of the subject piping during cold shutdown, since all applicable ANSI code allowable stresses are met, the probability of occurrence of an accident resulting from a seismically initiated pipe break is not increased. There will be no change to existing designs after the lead shielding is removed. No new failure modes are created.

Structural integrity of the subject piping has been confirmed with temporary lead shielding for Operating Modes 4 and 5. Installation of lead shielding temporarily does not change the limiting conditions for operation, applicability, or surveillance requirements as defined in the basis for the Technical Specifications. There will be no permanent changes made to existing designs or operational parameters after the affected shielding is removed.

SRASN: NPE-90-098 DOC NO: Engineering Report GGNS-90-0028 R00 SYSTEM:

DESCRIPTION OF CHANGE: Engineering Report GGNS-90-0028 R00 evaluated upper containment pool single failure and siphon protection requirements. This report determined that the siphon protection vacuum breakers (G41-F042A through H and G41-F060A through D) in the Fuel Pool Cooling & Cleanup (FPCCU) System return lines to the upper containment pool (UPC) need not be classified as active safety-related components.

REASON FOR CHANGE: Reclassification of the vacuum breakers allows safe elimination of ASME Section XI testing requirements in association with overall efforts to replace the FPCCU system siphon breakers with a more reliable, passive form of protection. The existing design meets the intent of the siphon protection requirements specified in GE and GGNS design documents. These requirements do not specify a degree of protection which will prevent any drop in UCP water level nor are they intended to maintain the levels above the T/S minimum limits following single failures. All postulated single failures resulting in a UCP draindown below specified minimum levels have been evaluated to be acceptable in that the capabilities of the plant systems to perform and maintain a safe reactor shutdown or mitigate an accident are not reduced. Therefore, the existing piping design meets the applicable requirements and the active function of the UCP siphon breakers is nonsafety-related. Although the design requirements for the active function of these components are being changed, implementation of these changes do not constitute a design change as defined by GGNS procedures.

SAFETY EVALUATION: This safety evaluation concluded that the change did not involve an unreviewed safety question. The Engineering Report evaluated the following types of events which could lead to an UCP draindown: 1) actuation of the SPMU system during a LOCA; 2) an inadvertent UCP dump; 3) a moderate energy line crack in piping connected to the UCP; 4) siphoning of water from the UCP; and 5) an operator error which results in loss of water through piping connected to the UCP. Each of these potential causes of a draindown was considered as a single initiating event and was thoroughly evaluated against the UCP design criteria. The report concluded that there will be no impact on the theoretical minimum UCP water level if no credit is taken for these siphon breakers. The only scenario which would result in lower water levels is that of an operator error which results in the isolation of the fuel storage area diffuser line by closing valve G41-F254. The theoretical minimum UCP water level following this event is at El. 195'-8" which is 4'-4" lower than the minimum theoretical water level if credit is taken for the siphon breakers (i.e., El. 200'-0"). The probability of this operator error is extremely remote and need not be postulated to occur coincident with a passive piping failure since there is no mechanistic relationship between these two failures. Although certain failures resulted in UCP water levels below the T/S minimum requirements as previously described, the consequences were evaluated to be within the existing licensing bases for all reactor OCs. For a "DBA" LOCA concurrent with the design basis assumptions including a loss-of-offsite power and a single limiting failure, the elimination of the vacuum breakers would have no impact on the capabilities of the suppression pool makeup system in performing the required safety functions. In addition, the evaluation concluded that there are no UCP draindown events involving the lines containing these siphon breakers which may occur concurrent with a fuel handling accident which would prevent the UCP from performing the required fission product removal functions within the applicable time frame as currently analyzed.

The only active function of these siphon breakers is to limit the severity of an inadvertent UCP draindown. The reclassification of this function as not safety-related by the Engineering Report has no effect on the probability of a draindown event. The passive safety function of these siphon breakers for maintaining the associated safety-related pressure boundaries is not changed by this reclassification of the active vacuum relief function. Thus, the probability of a passive siphon breaker failure which could lead to an inadvertent draindown event is not increased. All other applicable design requirements are not changed by this report. The results of this report and the changing of the active safety function requirements for these siphon breakers will not cause any system or component to operate beyond its design limits nor will it affect overall system performance in a manner which could lead to an accident. No accident precursors evaluated in the UFSAR are affected by this change.

The design requirements for the passive function of these siphon breakers are not changed by this report. The report results support the elimination of ASME Section XI testing requirements and the eventual removal of the valves by establishing that the active function to prevent siphoning of the UCP is not safety-related. As evaluated in the report, all applicable design, analysis, and installation requirements are met and that no new equipment failure modes are introduced by the elimination of the active function of these siphon breakers. The changes in the classification and testing requirements for the UCP siphon breakers do not affect any existing bases for the Technical Specifications and do not introduce any new requirements. By the evaluation presented in this Engineering Report, all applicable requirements for the existing UCP water level specifications are met. The margin of safety provided by the minimum UCP water levels specified in the Technical Specifications are applicable to a LOCA and a fuel handling accident. No siphoning event postulated to occur following a LOCA or a fuel handling accident would result in any significant reduction in UCP water inventory during the period when this water level is required to achieve safe shutdown or to limit the release of radioactivity.

SRASN: NPE-90-099 DOC NO: EER-90-6231 SYSTEM: G33

DESCRIPTION OF CHANGE: EER-90-6231 evaluated the addition of temporary lead shielding to certain portions of the Reactor Water Cleanup (RWCU) system. The lead shielding was installed only during Operating Modes 4 and 5. Calculations were performed on the subject piping with the added weight of the lead shielding. These calculations show that the structural integrity of the subject RWCU piping with the temporary lead shielding and supports will be maintained in the unlikely event of an operating basis earthquake (OBE) or a safe shutdown earthquake (SSE). All applicable ASME code stress allowables are met. Therefore, the operability of the RWCU system in Operating Modes 4 and 5 is not affected by the temporary lead shielding attached to the pipe. Temporary dead weight supports were installed on the system before the lead shielding was added and was not removed until all the shielding was removed. During the time the temporary supports are being utilized, the change in temperature of the RWCU system was not to exceed 50°F. Also, no other lead shielding or any other additional weight can be attached to the piping out to the first anchor while this shielding is attached. The temporary lead shielding and supports were installed during Operating Modes 4 and 5 only and were removed prior to restart after RF04.

REASON FOR CHANGE: To reduce radiation exposure to personnel performing work in this area.

SAFETY EVALUATION: This safety evaluation concluded that the change did not involve an unreviewed safety question. Structural integrity of the RWCU piping has been confirmed with temporary lead shielding and supports for Operating Modes 4 and 5. There are no permanent changes made to existing designs after the affected shielding and temporary supports are removed.

Since all applicable ASME code allowable stresses are met, the probability of occurrence of an accident resulting from a seismically initiated pipe break is not increased. No new failure modes are created.

Installation of lead shielding temporarily does not change the limiting conditions for operation, applicability, or surveillance requirements. Therefore, there is no reduction in the margin of safety as defined in the basis for any Technical Specification.

SRASN: NPE-90-100 DOC NO: MNCR-90-0176 SYSTEM:

DESCRIPTION OF CHANGE: MCP 90/1095, Rev. 0 was issued to repair damage received by the shroud head in RF04. In addition, a previously issued design, MCP 90/1090, Rev. 0 was used to replace the locking bolt at location 34 due to spline wear. This safety evaluation addresses the modifications made in the above design documents. Also, this safety evaluation addresses those areas of damage where it was determined that the as found condition was acceptable. The repairs included removal of a damaged separator assembly, removal of existing shroud head bolt locking assemblies at bolt locations 12 through 28, installation of new bolt assemblies at locations 12, 14, 16, 19, 21, 23, 25, 27, 34, and a weld repair to a gusset in the vicinity of bolt 14. An engineering evaluation was performed to evaluate the damage and repairs performed on the separator. A summary of the evaluation results is provided below:

The upper guide ring need not be restored to its original condition. The function of the ring is to provide alignment and support for the locking bolts. The design loads for the guide ring are small and well below the capability of the ring. The structural integrity of the ring is maintained in the bent position. The extension bolts and the retainer cans in the damaged areas are being removed and replaced, as required, to meet minimum bolting requirements.

The function of the tie bars is to interconnect the separators in order to reduce flow induced vibration, and to provide support against horizontal loads during a seismic event. The tie bars can still perform this function in the deformed condition. The structural adequacy of the tie bars is maintained.

A weld repair was performed on the gusset torn from the separator. Gussets that pushed into separator tubes are acceptable in that position. The dimpling is minor and does not adversely affect the structural integrity of the gusset or separator and does not adversely affect the performance of the shroud head/separator.

The locking bolts that were bent in the damaged area were removed to facilitate stud detensioning.

The retainer cans that were damaged will be removed and replaced, as required. The associated bolts will also be removed as stated above. The retainer cans perform no function if the bolt is removed.

The elevated separator assembly will be removed. There are 301 separators on the shroud head with the requirement to have only 280 to ensure adequate separator performance. Therefore, removal of the separator assembly will have no adverse affect on the shroud head.

REASON FOR CHANGE: The GGNS Unit 1 reactor steam separator received damage during RF04 vessel disassembly. The damage occurred when the upper guide ring was contacted during removal of the dryer from the vessel. The damage was confined to the area from approximately Azimuth 110° to azimuth 200°. A summary of the damage is as follows:

- The upper guide ring was bent vertically upward a maximum of approximately 30°
- Several tie bars were slightly buckled
- Several gussets used to attach the guide ring to the separator had pushed into the separator tubes
- Several locking bolts were bent
- Several retainer can were partially detached or bent
- One separator was elevated approximately 1.5" higher than the others

The reactor shroud head consists of a flange and a dome onto which is welded an array of standpipes, with a steam separator on top of each standpipe. The shroud head mounts on the flange at the top of the top guide and forms the cover of the core discharge plenum region. The stainless steel fixed axial flow type steam separators have no moving parts. The shroud head is bolted to the top guide flange by shroud head studs that have an extension to the top of the separators for access during refueling. The separator/shroud head is not a pressure retaining component. It is nonsafety-related, safety class other, and nonseismic.

SAFETY EVALUATION: This safety evaluation concluded that the change did not involve an unreviewed safety question. The damage that is "accept as is" and the repaired damage does not adversely affect the structural integrity or performance, nor does it create the potential for a loose part. The bolting requirements for the shroud head are maintained within the design limits. The actions described will not cause a decrease in reactor coolant temperature, an increase in reactor pressure or a decrease in reactor coolant system flow rate. The actions will have no affect on reactivity or power distribution. In addition, the actions will not cause an increase or decrease in reactor coolant inventory, affect the radioactive release from a subsystem and component, or affect the control rods from performing their function.

The function and structural integrity of the separator/shroud head is maintained. The separator/shroud head does not serve a safety function nor will the actions described adversely affect any safety related systems or components, or prevent safe shutdown.

SRASN: NPE-90-101 DOC NO: EER-90-6385 SYSTEM: F41

DESCRIPTION OF CHANGE: EER-90-6385 evaluated the possibility of deferring the removal of certain reactor internal vibration instrumentation until RF05.

The UFSAR lists the equipment used in the Reactor Internal Vibration Monitoring System (F41), along with the location of equipment inside the reactor vessel. A partial description of this startup test equipment is included in GE Specification 21A3854, which states "it is intended that the equipment above the shroud support plate and above the core support plate be removed during the first refueling outage". Most of the incore vibration instrumentation was removed during RF01, RF02, and RF03. The vibration instrumentation remaining in vessel at the start of RF04 is listed below:

Group 1 Guide rod with associated vibration instrumentation string

Group 2 Four (4) transition blocks as foll ws:

1 at 90° associated with Jet Pump 6
1 at 150° associated with Jet Pump 12
1 at 200° associated with Jet Pump 14
1 at 270° associated with Jet Pump 19

Group 3 Vibration equipment as follows: (90° to 180°)

Fourteen (14) clamps
One (1) coupling
Seven (7) conduits approximately 14 feet long with lead offs

Group 4 Vibration equipment as follows: (180° to 270°)

Fourteen (14) clamps
One (1) coupling
Four (4) conduits approximately 14 feet long with lead offs

Group 5 Vibr..ion equipment as follows: (Bottom grid to top of shroud support plate at 180°)

Ten (10 clamps
One (1) coupling
Four (4) conduits approximately 20 feet long with lead offs

NPE-90-101
Page 2

A review has been performed to allow vibration instrumentation Groups 2-5 (or any combination of Groups 2-5) to remain within the reactor vessel until RF05. The review concluded that this deferral is acceptable. The basis for the acceptance is the results of a vibration instrumentation residence time evaluation which concluded that the degradation of the vibration instrumentation equipment would be unlikely for up to 130 months of operation.

REASON FOR CHANGE: To reduce the impact of reactor vessel vibration instrumentation removal on the RF04 schedule.

SAFETY EVALUATION: This safety evaluation concluded that the change did not involve an unreviewed safety question. The presence of the subject equipment within the reactor vessel in Cycle 5 will not have any affect on the response of the plant to any of the analyzed accidents. There is no credible mechanism to force any of the subject parts off their mountings. It was shown that the only conceivable mechanism for detachment of this equipment (stress corrosion cracking) is not a credible event during Cycle 5.

Because the equipment coming loose and circulating in the reactor vessel has been evaluated not to be a credible event, there is no concern for interference with control rod operation or fuel performance. Reactor coolant chemistry will not be affected by this equipment due to the use of stainless steels which are suitable for use inside the reactor vessel. The subject equipment no longer serves any function. Furthermore, evaluations have shown that the structural integrity of the equipment will be maintained for at least another cycle of operation ensuring that no safety related systems or components will be affected. Therefore, the actions described will not reduce the margin of safety as defined in the basis for any Technical Specification.

SRASN: NPE-90-102 DOC NO: MNCR-90-0093 SYSTEM:

DESCRIPTION OF CHANGE: The dual coil solenoid valves for the main steam isolation valves (MSIV) have been replaced or rebuilt as an interim measure until the valve design can be modified in RF04. The only rebuild kits available are equipped with Viton seating material. Viton is an acceptable material for the period in question and is used as the material of choice by several utilities. GGNS is not currently using the Viton material in the drywell due to its radiation tolerance performance. An evaluation has been performed to ensure acceptable performance of the Viton material until RF04. Both Viton and the EPDM material have similar thermal aging performance.

REASON FOR CHANGE: The ASCO HTX- series single and dual coil solenoid valves for the main steam isolation valves (B21) were replaced with ASCO NP- series solenoid valves by DCP 84/3084. The HTX- series were not environmentally qualified valves while the NP- series are an environmentally qualified valve. Both valves are functionally similar. MNCR 265-89 later reworked the inboard and one outboard MSIV to replace the internal elastomer. The EPDM elastomer was deteriorating at a faster rate than previously expected. MNCR 0093-90 documents another case where the EPDM elastomer has deteriorated at a faster rate than expected.

SAFETY EVALUATION: This safety evaluation concluded that the change did not involve an unreviewed safety question. The implementation of this MNCR will not increase the probability of occurrence of an accident. The subject valves are functionally similar to those they replaced. The NP- series valves are environmentally qualified per 10CFR50.49. Viton is thermally equivalent to EPDM (EQDP EQ6.3, Tab H1), however, Viton is more radiation sensitive. The use of Viton until RF04 has been evaluated and its radiation threshold is acceptable for greater than one year of service which will not be exceeded prior to replacement in RF04. The elastomer materials can be considered equivalent materials for the period of time they will be installed.

Because of the functional similarity of the replacement ASCO solenoid valves to the ones that were replaced, no change in Plant Technical Specifications are required. Replacement of the internal elastomer within the solenoid valves with Viton will not impact the valve function. Therefore, there is no reduction in the margin of safety as defined in the basis for any Technical Specification.

SRASN: NPE-90-103 DOC NO: EER-90-6401 SYSTEM: G36

DESCRIPTION OF CHANGE: EER90-6401 evaluated the addition of temporary lead shielding to certain portions of the Reactor Water Cleanup (RWCU) drain lines from the regenerative heat exchangers. The lead shielding will be installed during operating modes 4 and 5 only and must be removed prior to restart.

REASON FOR CHANGE: To reduce radiation exposure to personnel performing work in these areas.

SAFETY EVALUATION: This safety evaluation concluded that the change did not involve an unreviewed safety question. These temporary changes do not affect the structural integrity of the subject piping during cold shutdown. There will be no change to existing designs after the lead shielding is removed. Since all applicable ANSI Code allowable stresses are met, the probability of occurrence of an accident resulting from a seismically initiated pipe break is not increased. No new failure modes are created.

Structural integrity of the subject piping has been confirmed with temporary lead shielding for Operating Modes 4 and 5. Installation of lead shielding temporarily does not change the limiting conditions for operation, applicability, or surveillance requirements as defined in the basis for any Technical Specification. There will be no permanent changes made to existing designs or operational parameters after the affected shielding is removed.

SRASN: NPE-90-104 DOC NO. EER-90-6417 SYSTEM:

DESCRIPTION OF CHANGE: EER-90-6417 request that temporary lead shielding be attached to certain portions of the RWCU system. The lead shielding will be installed during Operating Modes 4 and 5 only, and must be removed prior to restart. Reactor pressure cannot be increased above 280 pounds while shielding is installed. This evaluation does not cover reactor hydrolyzing.

REASON FOR CHANGE: To reduce radiation exposure to personnel performing work in this area.

SAFETY EVALUATION: This safety evaluation concluded that the change did not involve an unreviewed safety question. These temporary changes do not affect the structural integrity of the RWCU piping during cold shutdown. Structural integrity of the RWCU piping has been confirmed with temporary lead shielding for Operating Modes 4 and 5. There are no permanent changes made to existing designs after the affected shielding is removed.

Calculations were performed on the subject piping with the added weight of the lead shielding. These calculations show that the structural integrity of the subject RWCU piping with the temporary shielding will be maintained in the unlikely event of an operating basis earthquake (OBE) or a safe shutdown earthquake (SSE). All applicable ASME code stress allowables are met. Inadvertent pressurization due to loss of shutdown cooling (SDC) in Mode 4 was considered. However, due to the nature of the errors of failure required to cause the event, pipe breaks are not required to be analyzed.

Structural integrity of the RWCU piping has been confirmed with temporary lead shielding for Operating Modes 4 and 5. Installation of lead shielding temporarily does not change the limiting conditions for operation, applicability, or surveillance requirements as defined in the basis for the Technical Specifications.

SRASN: NPE-90-105 DOC NO: CN-90-0523 SYSTEM:

DESCRIPTION OF CHANGE: Above normal seat leakage was identified for the Automatic Depressurization System (ADS) air accumulator stop check valves during leakage testing for valves. (Q1B21F036D, F036F, F036H, F036J, F036P, F036R through F036T, F036U, F039D, F039F, F039H, F039J, F039P, and F039R through F039T) for the .)S air accumulators. Additionally, a nonconforming condition fo. .he ADS air supply was identified. The old piping analysis assumed a peak piping temperature of 240°F. Since ADS is required for accident conditions the piping must be analyzed for the peak post accident drywell temperature of 330°F. The evaluated document provides approval and justification for closing one of the accumulator stop check valves and replacement of the remaining accumulator stop check valve with a resilient seat check valve. In addition, this document provides for the necessary piping support modifications to qualify the ADS air supply piping for 330°F.

REASON FOR CHANGE: Closing of one of the inlet stop check valves for each ADS S/RV will not prevent the accumulators from initially charging or prevent S/RV leakage makeup following actuation since a common two inch discharge line connects both accumulators to the S/RV actuator. Furthermore, there would be negligible pressure drop across the remaining stop check valve in the common one inch supply to the two accumulators considering the leakage makeup requirement of 1 scfh. Finally, closure of one of the inlet stop check valves for each ADS S/RV directly prevents accumulator depressurization through that valve upon depressurization of the common distribution header. Replacement of the remaining ADS S/RV accumulator stop check valves with a resilient seat check valve improves the seating characteristics of the valves. The seating surface of the existing stop check valves is metal-to-metal which results in above normal seat leakage during required surveillance testing for the ADS S/RV accumulators. The resilient seat check valves meet all applicable code and design requirements, including environmental considerations. The current piping analysis shows that the piping was analyzed for a temperature of 240°F. Since ADS is required for accident conditions the piping must be analyzed for the peak post accident drywell temperature condition of 330°F.

SAFETY EVALUATION: This safety evaluation concluded that the change did not involve an unreviewed safety question. The UFSAR considers various accidents and transients which are postulated to occur in order to determine the capability of the plant to operate within regulatory guidelines without undue risk to the public health and safety. Those accidents and transients whose probability of occurrence may be increased due to closing one accumulator stop check valve and replacing the remaining stop check valve with a resilient seat check valve involve only those accidents which are dependent on the ability of the passive air supply system to support the ADS, Low Low Set (LLS) and relief functions. For the previously postulated accidents and transients dependent on the ADS, LLS, and relief functions, the required safety functions of the stop check valves will be maintained; therefore, the passive air supply system supports the ADS, LLS and relief functions. In addition, the ADS air supply piping system and pipe supports designs meet ASME Section III requirements for the required accident and transient scenarios and are qualified as seismic category I. The piping and pipe supports will function in their intended manner. The proposed changes do not adversely affect any fission product barrier, the ability to mitigate accidents and transients, or the radiological consequences of accidents and transients.

The ADS air supply piping system and pipe supports designs meet ASME Section III requirements for the required accident and transient scenarios and are qualified as seismic category I. The ADS, LLS, and relief functions are no more likely to fail when required to function than before.

The ADS and non-ADS air accumulator stop check valves are not explicitly discussed in the bases for TS 3/4.5.1. The bases assume the operability of the passive air supply system to ensure that the ADS function to depressurize the reactor vessel so that the low pressure ECCS can inject water into the reactor vessel for core cooling following a small primary system line break if the HPCS system fails or cannot maintain reactor water level. The margin of safety associated with the ADS function involves the ability to depressurize the reactor to prevent exceeding fuel cladding integrity limits. As discussed above, operation with the proposed modifications has been evaluated for its effect on the ADS function during postulated accidents. Evaluation results demonstrate that the passive air supply system supports the ADS function with no impact on fuel cladding integrity limits.

The margin of safety associated with the LLS function involves the ability to minimize the induced loading on the containment/suppression pool boundary by ensuring no more than one relief valve opens subsequent to the initial blowdown on an overpressure transient. As previously described, the proposed changes have been evaluated for their effect on the LLS function during postulated transients. Review results demonstrate that the passive air supply system supports the LLS function with no impact on the ability to prevent more than one relief valve from opening subsequent to the initial blowdown on an overpressure transient.

The margin of safety associated with the relief function involves the ability to protect the reactor vessel from overpressure during upset conditions. As previously described, the proposed changes have been evaluated for their effect on the relief function during postulated overpressure transients. Review results demonstrate that the passive air supply system supports the relief function with no impact on reactor coolant pressure boundary safety limits. Since operation with the proposed changes has been found to be acceptable, the passive air supply system is capable of supporting the ADS, LLS, and relief functions and the margin of safety as defined in the basis for the Technical Specifications is not reduced.

SRASN: NPE-90-106 DOC NO: CN 90-0537 SYSTEM:

DESCRIPTION OF CHANGE: CN 90-0537 requested the removal of pressure regulator SP21F438.

REASON FOR CHANGE: To reduce the pressure drop in the Make-up Water Treatment (MWT) system supply to the Circulating Water (CW) pump lube water pumps and motor coolers.

SAFETY EVALUATION: This safety evaluation concluded that the change did not involve an unreviewed safety question. The removal of the regulator will not adversely affect the function of the lube/cooling water supply system. Control of which system is the supply for the lube/cooling water may be obtained by throttling other valves in appropriate lines or by adjustment of the pressure regulator in the DW supply piping. The change does not compromise any safety related system or prevent a safe shutdown of the plant. It has no effect on the function or reliability of any equipment important to safety. The design change does not create any interface with equipment important to safety.

No credit is assumed for the DW and CW systems in the Bases of the Technical Specifications. The design change does not affect that part of the MWT system which is addressed in the Technical Specifications, specifically, valves forming a part of containment boundary. Therefore, there is no reduction in the margin of safety as defined in the basis for any Technical Specification.

SRASN: NPE-90-107 DOC NO: OPS w/o Purge Flow To Reactor Recirc Pump SYSTEM:

DESCRIPTION OF CHANGE: This Safety Evaluation discusses the implications relative to operation with zero seal purge flow to the reactor recirculation pump shaft seal assemblies.

REASON FOR CHANGE: Operation with Zero seal purge flow to reactor recirculation pump shaft seal assemblies will reduce this source of cyclic thermal stress responsible for crack initiation in the shaft and heat exchanger.

SAFETY EVALUATION: This safety evaluation concluded that the change did not involve an unreviewed safety question. The UFSAR considers various accidents which are postulated to occur in order to determine the capability of the plant to operate within regulatory guidelines without undue risk to the public health and safety. Those accidents whose probability of occurrence may be increased due to operation with zero seal purge flow involve only those accidents which are dependent on the passive pressure boundary of the recirculation system. Operation with zero seal purge supports the passive pressure boundary since cyclic thermal stresses will be reduced. Furthermore, there are no events postulated in the UFSAR directly caused by a reduction in the seal purge flow and operation with zero seal purge flow would not create such an event. Therefore, since the recirculation system passive pressure boundary is not affected in a manner that could lead to an accident or cause an accident previously evaluated to shift to a higher frequency category, there is no increase in the probability of occurrence or in the consequences of an accident or malfunction of equipment important to safety previously evaluated in the Safety Analysis Report. Furthermore, operation with zero seal purge flow will not prevent the recirculation system from performing its design functions consistent with the assumptions of the UFSAR accident and transient analyses.

Since operation with zero seal purge flow supports the passive pressure boundary as originally designed and since the reactor recirculation system is no more likely to fail when required to function than before, there is no creation of a possibility for an accident or malfunction of a different type than any evaluated previously in the Safety Analysis Report.

Since the seal purge flow is not explicitly discussed in the bases for T/S 3/4.4.1 and since operation with zero seal purge flow is found to be acceptable for the UFSAR accident and transient analyses the margin of safety as defined in the basis for any Technical Specifications is not reduced.

SRASN: NPE-90-108 DOC NO: EER-90-6466 SYSTEM:

DESCRIPTION OF CHANGE: The pump shaft and impeller will be replaced on the reactor recirculation pump "B". Although it is considered a "like-for-like" replacement, the replacement impeller has minor dimensional differences in the impeller diameter from that of the existing impeller. However, the difference is minor and will not affect the performance of the pump. This evaluation will address the differences in the currently installed impeller and the replacement impeller.

REASON FOR CHANGE: The impeller will be replaced on the "B" reactor recirculation pump due to excessive vibration.

SAFETY EVALUATION: This safety evaluation concluded that the change did not involve an unreviewed safety question. Replacement of the pump internals will not result in a change to the operation or performance of the pump or its associated system. The minor difference in impeller diameter will not adversely affect the pump capacity. There will be no impact to any interfacing system as a result of the replacement.

The replacement has been evaluated against the applicable design criteria, installation requirements, and operational requirements. It was determined that all necessary requirements and commitments are met by the new component and that no new accident precursors are created.

The existing system and component design functions are not affected. Therefore, this change will not reduce the margin of safety as defined in the basis for any Technical Specification.

SRASN: PLS-90-011 DOC NO: UFSAR Appendix 3A SYSTEM:

DESCRIPTION OF CHANGE: This change deletes the reference in UFSAR Appendix 3A which indicates that SERI will comply with Regulatory Guide 8.14 (1976), which addresses personnel neutron dosimeters.

REASON FOR CHANGE: Grand Gulf no longer uses a separate dosimetry for monitoring neutron exposure, and therefore this Regulatory Guide does not apply to our dosimetry system. GGNS meets the ANSI N13.11 and 10CFR20.202 requirements for dosimetry.

SAFETY EVALUATION: This safety evaluation concluded that the change did not involve an unreviewed safety question. Personnel monitoring for radiation exposure is unrelated to any accidents previously evaluated in the FSAR. Personnel dosimetry has no effect on or interface with any systems related to plant safety. This change has no effect on or interface with equipment important to safety previously evaluated in the UFSAR. It has no effect on the limiting condition for operation, applicability, action or surveillance requirements as defined in any Technical Specification.

SRASN: PLS-90-012 DOC NO: FSAR C/R 90-0005 SYSTEM:

DESCRIPTION OF CHANGE: This UFSAR change takes exception to Regulatory Guide 1.137 step C.2.d(3) that requires removing condensate in the Diesel Generator Fuel Oil Storage Tanks one day after adding new fuel oil.

REASON FOR CHANGE: Chemistry samples and analyses are performed on new fuel oil prior to discharging to the Fuel Oil Storage Tanks. The sampling requirements are very stringent ($\leq$ .05 volume percent) thus controlling the amount of water added to the fuel oil tanks. Therefore the sample required one day after adding new fuel is not necessary.

SAFETY EVALUATION: The change does not involve an unreviewed safety question. The Technical Specification sample requirement for water (.05 volume percent) in new fuel precludes putting any significant amount of water into the Emergency Diesel Generator Fuel Oil Storage tanks. In addition, the Fuel Oil Transfer pump suction line(s) are located 8" above the bottom of the Fuel Oil Storage Tanks. Water accumulation in the bottom of the storage tank would have to be significant (approximately 2000 gals) before the Fuel Oil Transfer pump would pump water into the fuel oil system of the Emergency Diesel Generators. Presently water is removed quarterly and less than one gallon is routinely removed.

Because of the stringent Technical Specification sampling requirements of new fuel (prior to adding to the fuel oil storage tanks), the probability and consequence of equipment malfunction due to water intrusion into the fuel oil system of the Emergency Diesel Generators is not increased.

Taking exception to the Regulatory Guide 1.137 Step c.2.d(3) does not reduce the margin of safety as defined in the basis for Technical Specifications because the exception doesn't alter the surveillance frequencies or acceptance criteria for water content in the fuel oil.

SRASN: PLS-90-013 DOC NO: TSTI-1G17-90-003-0-S SYSTEM: G18

DESCRIPTION OF CHANGE: This change allows the addition of sodium hypochlorite to a condensate phase separator tank to stop microbiological activity in the tank.

REASON FOR CHANGE: There are methane-producing bacteria present in the tank which cause pressurization of the radioactive waste liner when the liner is dewatered. The addition of sodium hypochlorite to obtain a free chlorine residual of 0.5 ppm for thirty minutes is necessary to prevent the gas formation from occurring

SAFETY EVALUATION: The change does not involve an unreviewed safety question. The performance of this activity does not change the operation of the phase separators, resin transfer, or dewatering equipment. The chemical to be used will not be detrimental to the equipment in the concentrations to be used. Inadvertent spillage of hypochlorite into the radwaste system would result in the early changeout of a demineralized bed, but would not have any effect on the integrity of the piping or components.

Accidents evaluated in the UFSAR involving the radwaste system are leaks/tank ruptures in the system (15.7.2 and 15.7.3). System operation is not changed and the chemical is not detrimental to the equipment. No different failure would be caused by this activity, which is bounded by these analyses of whole tank ruptures.

This activity meets the requirements of the PCP addressed in Technical Specifications and does not affect the activity of radwaste shipments.

SRASN: PLS-90-015 DOC NO: OQAM FSAR 17.2 SYSTEM: N/A

DESCRIPTION OF CHANGE: This change reassigns the responsibilities for audits and evaluations of suppliers, review of procurement documents ard receipt inspection as delineated in various policies of the OQAM to the Manager, Quality Seivices due to the transfer of the current Manager, Quality Systems to the Manager, Quality Services position.

REASON FOR CHANGE: This transfer of responsibilities will allow consistency in the administration of those activities and facilitate anticipated changes in the Quality Programs area due to consolidation.

SAFETY EVALUATION: The change does not involve an unreviewed safety question. These changes are administrative in nature only and have no affect on any component or system. Since these changes do not delete any responsibilities there is no reduction in program requirements. These transferred functions are still being performed and the managerial changes have no effect on the safety of the plant.

The changes do not effect any basis in the Technical Specifications.

SRASN: PLS-90-016 DOC NO: 04-1-01-N19-1-TCN 25 SYSTEM:

DESCRIPTION OF CHANGE: This procedure change allows removal of the water seal from around the High Pressure Condenser rubber expansion joint while the plant is operating.

REASON FOR CHANGE: The removal of the water seal fr vice is being performed as an interim measure to reduce leakag. to radwaste from the seal.

SAFETY EVALUATION: The change does not involve an unreviewed safety question. The removal of the water seal from service could result only in increased air in-leakage into the condenser and reduce the ability to detect the loss or gross degradation of the rubber expansion joint during plant operation. The removal of the seal from service will not cause deterioration of the rubber joint above and beyond normal expected service life. The water seal does not directly or indirectly affect any components other than the rubber joint. There is no equipment important to safety which could be affected by the removal of the water seal from service.

The removal of the seal from service does not reduce the margin of safety as defined in the basis for any of the Technical Specifications, because there are no safety functions or safety limits which are associated or affected by the water seal.

SRASN: PLS-90-017 DOC NO: WO#-00014194 SYSTEM: P44

DESCRIPTION OF CHANGE: This temporary change installed a supply and return pipe for the drywell chiller cooling water which originates from the Plant Service Water (PSW) piping.

REASON FOR CHANGE: The four-way valve on the normal supply/return PSW piping to th Drywell Chillers was obstructing flow. This temporary change bypasses the valve.

SAFETY EVALUATION: The change does not involve an unreviewed safety question. This change does not affect the overall flow balance of the PSW system. The potential flows to CCW and Drywell Chillers during normal and LOP conditions have been evaluated and determined acceptable. The effects of the piping addition have been evaluated and determined to be acceptable.

Standby Service Water (SSW) flow balance will not be adversely affected by this change. The remaining components are non-safety related and not required to mitigate the consequences of an accident. The temporary four way valve bypass does not adversely affect any system as described in the basis of any Technical Specification.

SRASN: PLS-90-018 DOC NO: Deleting Operator Actions SYSTEM:

DESCRIPTION OF CHANGE: This change removes references to specific operator actions found in Chapter 15 that are not required by the safety analysis basis and are not safety actions required to bring the plant to a stable condition. A statement is added to reference operator actions to the Site Specific Operating Procedures and their programmatic control. Operator actions are also removed from the Operating Modes of RCIC System section of Chapter 5. The operator action found in Chapter 7 is deleted and reference to the guidance found in the Site Specific Emergency Procedures is added. Specific operator actions at the Remote shutdown Panel found in Appendix 9C are deleted and reference to the guidance found in the Site Specific Emergency Procedures is added.

REASON FOR CHANGE: Regulatory Guide 1.70, Rev. 3(15.x.s, 2a) requires that the Event Evaluation section of the FSAR Chapter 15 include a sequence of events and systems operations. This listing must include a step-by-step sequence of events from the event initiation to the final stabilized condition including all required operator actions. The required operator actions must include any operator action assumed in the safety analysis and any actions that are not part of the safety analysis basis, but are safety actions required to bring the plant to a stable condition. FSAR Paragraph 15.0.3.2.1.4 (s8) states "For all anticipated operational transients cited in Chapter 15, no operator corrective action is required to prevent the plant from exceeding safety design basis limits." (s9) states "In no case would the operator's action or non-action result in an unacceptable effect on the health and safety of the general public."

The change clarifies that the operator actions previously identified were not required operator actions assumed in the safety analysis nor actions required to bring the plant to a stable condition. The change is consistent with Regulatory Guide 1.70 for Event Evaluations.

SAFETY EVALUATION: This safety evaluation concluded that the change did not involve an unreviewed safety question. Operator actions are not taken until after an event has occurred and therefore have no effect on the probability of occurrence. This change will not create the possibility of an accident of a different type than any evaluated in the FSAR because it only affects operator actions. The effect of single operator error is already analyzed in the FSAR and therefore bounds the scope of this change. The Site Specific Emergency Procedures are symptomatic in nature and provide guidance to mitigate the symptoms and to maintain the plant in a safe condition regardless of what event occurred to generate the symptom or regardless of what equipment is available to combat the symptom.

PLS-90-018
Page 2

This change will not reduce the margin to safety as defined in the basis for any Technical Specification because the Site Specific Emergency Procedures are provided to the operator for mitigating any symptom regardless of the initiating event and therefore actually increase the margin to safety over the operator actions presently found in the FSAR.

SRASN: PLS-90-019 DOC NO: TSTI-1E51-90-002-0-S SYSTEM: E51

DESCRIPTION OF TEST: TSTI-1E51-90-002-0-S places the RCIC system in service in the Test Return Mode of operation in accordance with SOI 04-1-01-E51-1 to obtain differential pressure thrust data on 1E51-F022 and 1E51-F059, test return flow path isolation valves. Once in service, the automatic opening function of the RCIC minimum flow valve, 1E51-F019, will be defeated to allow determination of peak differential pressures across the two valves during performance of the test and to allow a higher pressure differential to be developed across the valves. Failure of the minimum flow valve was assumed in the Maximum Expected Differential Pressure calculations for these valves. The automatic closure of the minimum flow valve will remain effective during performance of this test. Minimum flow control valve operation in the open direction will be controlled via the Main Control room handswitch. Thrust data will be obtained at a series of four independent differential pressure data points.

REASON FOR TEST: The subject data is being obtained in an attempt to address the issues of GL 89-10 and GL 89-10 Supplement 1.

SAFETY EVALUATION: This safety evaluation concluded that the change did not involve an unreviewed safety question. With the exception of minimum flow valve automatic opening, the RCIC system is operated in a normal system configuration, test return mode. Operation of the RCIC system in this mode is a normal plant activity and does not increase the consequences of an accident. The system/plant has been evaluated for this mode of operation in the original plant design safety evaluation. The RCIC system is not Operable (as defined in the Technical Specifications) during performance of this test and as such no credit can be taken for RCIC system operation in a capacity to mitigate events. The Technical Specifications provide the necessary flexibility for operation with the RCIC system inoperable (provided HPCS is operable) for mitigation of analyzed events. Operation of the RCIC system in the test return mode is a previously evaluated mode of operation for the RCIC system.

The RCIC system is declared Inoperable during performance of this test. The HPCS system remains Operable during performance of this test. The HPCS system provides the necessary protection when the RCIC system is Inoperable for the RCIC associated event analyses in the SAR. As this is the case, adequate capability exists to maintain event/accident mitigation margin for events analyzed in the SAR. Manual control will take the place of automatic open control of the RCIC minimum flow valve and therefore RCIC pump integrity will be maintained.

Provided the HPCS system is Operable during performance of this TSTI, the margin of safety is consistent with that discussed in the BASES of the Technical Specifications.

SRASN: PLS-90-020 DOC NO: FSAR C/R 90-0008 SYSTEM: N64

DESCRIPTION OF CHANGES: The following sentence was deleted from FSAR Section 11.3.2.1.6.2 - "During transfer of the charcoal into the charcoal adsorber vessels radial sizing of the charcoal will be minimized by pouring the charcoal (by gravity or pneumatically) over a cone or other instrument to spread the granules over the surface."

REASON FOR CHANGE: The deleted sentence did not describe how the adsorber vessels were actually filled.

SAFETY EVALUATION: This safety evaluation concluded that the change did not involve an unreviewed safety question. The adsorber vessels were filled during construction and have performed as designed. Through construction experience General Electric has determined that the method of filling the adsorber vessels does not affect adsorber performance. The charcoal is intended to last the life of the plant. During construction the charcoal was just poured in. This method would be reused if change out is required in the future. If the adsorber vessels ever had to be filled again the post treatment radiation monitor would confirm the charcoal adsorber performance.

The Offgas System outlet vent valve is interlocked to the offgas post treatment radiation monitor that monitors radiation levels at the outlet of the adsorber vessels and upon receipt of a predetermined high high radiation alarm the offgas system is isolated from discharging to the environment.

The charcoal adsorber vessels are design to withstand a hydrogen detonation. The composition of the charcoal fill does not affect the process system boundary therefore this change does not create the possibility of an accident of a different type than any evaluated in the FSAR.

The only safety significance of the adsorber vessels is the pressure boundary and the ability of the vessels to be isolated by the post treatment radiation monitors. The charcoal fill does not affect either the pressure boundary nor the ability of the radiation monitor to isolate the vessels.

This change will not reduce the margin of safety as defined in bases for any Technical Specification because isotopic analysis has verified the ability of the charcoal adsorbers to delay release of fission gases and keep the effluent release to atmosphere within prescribed limits. The charcoal adsorber fill does not effect the offgas pressure boundary and the offgas process will be isolated from the offgas and radwaste vent upon receipt of a high-high radiation signal.

The design bases for the Circ. Water system as defined in the GGNS Technical Specification does not contain provisions for any specified margin of safety regarding the failure of a circulating water system component. Therefore, implementing this work order does not reduce the margin of safety as defined in the basis for any Technical Specification.

SRASN: PLS-90-022 DOC NO: UFSAR 7.7.1.11.4.2.b SYSTEM: P33

DESCRIPTION OF CHANGE: This change allows for chlorides to be analyzed via the Post Accident Sampling System (PASS) within 4 days (96 hours) instead of the current requirement of 24 hours.

REASON FOR CHANGE: This change will bring the UFSAR in compliance with NUREG 0737 Attachment I, II.B.3. This change allows the sample to decay for 96 hours which reduces personnel exposure.

SAFETY EVALUATION: This safety evaluation concluded that the change did not involve an unreviewed safety question. There is no accident evaluation on 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. This UFSAR change does not reflect any change to the PASS system or interfaced systems.

The only equipment associated with PASS that is important to safety are the containment and drywell isolation valves of the reactor coolant, suppression pool and atmospheric sample lines. These valves are not impaired by PASS sampling and are able to perform their required isolation functions in the event of an accident while a scheduled sampling evolution is in progress. Any PASS sampling evolution in progress during the occurrence of an accident would be terminated by the load shedding and sequencing system and automatic sample line isolations. A manual reset is required before sampling could resume. This UFSAR change concerns sample analysis which is performed on a PASS grab sample and does not change the operation of the PASS panel but only clarifies the analysis requirements which are performed after the sample is collected.

There are no Technical Specifications bases applicable to PASS. It is a non-safety related, non-seismic, and non-environmentally qualified system. PASS was constructed by principal construction code B31.1. There is no direct or indirect impact to any other margins of safety as defined in the bases for any Technical Specifications.

SRASN: PLS-90-023 DOC NO: UFSAR CR 90-010 SYSTEM: E12

DESCRIPTION OF CHANGE: This change added the following statement to UFSAR 7.7.1.11.4.3: "The Suppression Pool, RHR-A and RHR-B, shall be sampled through the Post Accident Sample System separately in consecutive six-month intervals, rotating sampling personnel for training purposes, such that all three points are sampled on an 18-month interval." This will increase the use of the PASS system and require occasional operation of the RHR system pumps for the sole purpose of taking samples.

REASON FOR CHANGE: This was a mandated change by the NRC and a documented licensing commitment, LCTS ID No. 15799.

SAFETY EVALUATION: This safety evaluation concluded that the change did not involve an unreviewed safety question. There is no accident evaluation in the UFSAR for the Post Accident Sample System (PASS). However, sampling of the Suppression Pool via PASS causes a loss of Division 2 Suppression Pool level indication. This instrument functional loss is temporary, lasting only while sampling is actually occurring. Loss of this instrument function places the plant in a 7-day LCO condition as per Technical Specification Table 3.3.7.5-1 (3.), Action 80. The PASS connection for sampling the Suppression Pool taps off of the Division 2 Suppression Pool sensing line. This line is equipped with a restricting orifice near the Suppression Pool connection point. PASS samples from downstream of this orifice and, while sampling, removes water faster than make-up can occur through the restricting orifice. This causes the instrument to indicate a false low-low Suppression Pool level. This inputs one of the two required low-low level indications required for Suppression Pool make-up to occur. Therefore, if a single instrument failure along with a LOCA signal were to occur while sampling Suppression Pool via PASS, two Suppression Pool Low-Low Level signals would occur. This would initiate the Suppression Pool Make-up (SPMU) system, dumping the Upper Containment Pool into the Suppression Pool. As a safety measure, a step is included in Chemistry Section Instruction 08-S-04-954, which directs the taking of PASS liquid samples, that requires Chemistry to have Operations to place the SPMU Division 2 Mode Selector handswitch, on Control Room Panel 1H13-P-870 Section 10B, in the "OFF" position prior to taking a PASS Suppression Pool sample. This overrides the SPMU function of the Division 2 Suppression Pool level instrumentation, preventing an inadvertent dump from occurring. Therefore, the probability of an inadvertent SPMU dump is not increased. The action of placing the Division 2 SPMU Mode handswitch to "OFF" is acceptable by entry into an LCO condition as per Tech Spec Sections 3.3.8 and 3.6.3.4.

The PASS system is a non-safety related system required by NUREG-0737 to operate after a design basis accident. Although PASS is used for providing information regarding the extent of core damage following an accident, PASS plays no part in mitigating the consequences of an accident. The only equipment associated with PASS that is important to safety are the containment and drywell isolation valves of the reactor coolant, suppression pool and atmospheric sample lines. Those isolation valves required to be operated for performance of these samples (Suppression Pool, RHR-A and RHR-B) are not impaired by these sampling events and should be able to perform their required isolation functions in the event of an accident while a scheduled sampling evolution is in progress. PASS and condensate cooling water (CCW) (used for sample coolers in PASS) are shed in the event of an accident. These systems are not required to mitigate the consequences of an accident and are not required for a safe shutdown of the reactor. Any PASS sampling evolution in progress during the occurrence of an accident would be terminated by the load shedding and sequencing system and automatic sample line isolations. A manual reset is required before sampling could resume. Therefore, the routine samples described in this UFSAR change will not create the possibility of an accident of a different type than any already evaluated in the UFSAR (UFSAR references 1.2.2.8.2 and 7.7.1.11.4.2.L).

PASS itself is a non-safety related system. The additional scheduled samples described in this UFSAR change requires running of the RHR-A and RHR-B systems and pumps to sample from the respective sample points. It is unlikely that all of the scheduled PASS samples will coincide with scheduled running of these systems. Therefore, RHR-A and RHR-B will need to be started and run in either reactor cooling or suppression pool cooling modes, depending upon plant conditions, for the purpose of collecting routine PASS samples. This will add a proportionally small amount of run hours and wear on the systems. However, assuming the worst case that all of the required samples over the remaining life of GGNS operating license required the start-up and running of the RHR systems for the sole purpose of PASS samples with a conservative estimate of 3 hours of run-time per sample, would require only a maximum of 24 samples (18 month frequency) and 72 additional hours of run-time for each of RHR-A and RHR-B pumps. This additional run-time is not significant over the lifetime of the RHR pumps and is therefore acceptable.

There are no Technical Specifications bases applicable to PASS. It is a non-safety related, non-seismic, and non-environmentally qualified system. PASS was constructed by principal construction code B31.1. The operation of PASS is principally for operability verification and training with the intent that it be available for assessment of core conditions following a design base accident. PASS is not used to mitigate the consequences of an accident and is not required for safe shutdown of the reactor. There is no direct or indirect impact to any other margins of safety as defined in the bases for any Technical Specifications.

SRASN: PLS-90-024 DOC NO: 01-S-06-2 SYSTEM:

DESCRIPTION OF CHANGE: This change adds the Plant Supervisor (SRO) duties and responsibilities to the conduct of Operation Administrative Procedure 01-S-06-2.

REASON FOR CHANGE: The addition of the third SRO to each shift contributes to the experience and knowledge level to further enhance the safe operation of the unit.

SAFETY EVALUATION: This safety evaluation concluded that the change did not involve an unreviewed safety question. The added experience and knowledge of the third SRO to each shift improves overall shift performance and reduces the probability of occurrence of an accident. The added talent of a third SRO improves the performance of the shift such that if any abnormality occurs, event evaluation and proper response tend to minimize the consequences of the accident. The established control room command structure remains in effect ensuring continuity during normal and abnormal conditions. The presence of the third SRO improves equipment monitoring therefore detecting symptoms relating to malfunctions earlier. This earlier detection could minimize the consequences of equipment malfunction.

The additional knowledge and experience provided by the third SRO can only improve compliance to Technical Specifications and related bases. The third SRO provides a valuable resource to discuss and evaluate conditions relating to Technical Specification concerns.

SRASN: PLS-90-025 DOC NO: MWP-90-1151 SYSTEM: L11

DESCRIPTION OF CHANGE: This safety evaluation addresses operability of the Battery Room Hydrogen Detector Panel H22-P535 for all plant modes of operation (Modes 1 through 5).

Relocation of the hydrogen detector panel will require the battery room hydrogen detector circuits to be inoperable for approximately seven days. This is considered a conservative number to allow completion of work and subsequent re-calibration of the detector circuits. During this period, ventilation systems will be verified operable on a daily basis. If ventilation is found to be not operating, ventilation will be restored, or portable hydrogen samples will be taken daily and upon every access into a battery room where the detector circuit is inoperable. In addition, a weekly portable hydrogen detector sample will be taken on all battery rooms where the detector is inoperable until H22-P535 is restored.

REASON FOR CHANGE: Replacement of UPS Inverters 1Y87, 1Y89, 1Y95 and 1Y96 will require relocation of Hydrogen Detector Panel H22-P535 to facilitate maintenance on the new inverters. This relocation will result in the Hydrogen Detector Panel being inoperative during the disconnection, relocation and reconnection.

SAFETY EVALUATION: This safety evaluation concluded that the change did not involve an unreviewed safety question. The battery hydrogen detector panel serves no safety function, nor is it required to be operable as part of the fire protection system. None of the accidents previously evaluated in the FSAR are affected by the battery room hydrogen detector panel. The battery room hydrogen detectors play no role in mitigating the consequences of any accidents described in the FSAR. The hydrogen detector panel performs an information function only. The hydrogen detector panel does not affect malfunction of any equipment important to safety.

No margins of safety as defined in the basis for any Technical Specifications are associated with the Hydrogen Detector Panel H22-P535, therefore there is no reduction in the margin of safety.

SRASN: PLS-90-026 DOC NO: DCP-88-0051 SYSTEM: L62

DESCRIPTION OF CHANGE: This safety evaluation addresses the operability concerns associated with supplying temporary power in place of the normal inverters 1Y87, 1Y88, 1Y95 and 1Y96.

REASON FOR CHANGE: The subject inverters are to be replaced.

SAFETY EVALUATION: This safety evaluation concluded that the change did not involve an unreviewed safety question. A temporary inverter will be supplied by a non-operable (not the declared operable, but energized), class 1E bus, thus failure of the temporary inverter could not affect or degrade a class 1E power source. The output of each temporary inverter will be isolated from its respective distribution panel via the panel circuit breaker and fused disconnects for each panel branch circuit. The inverter itself is considered non-essential since none of the circuits that power is being supplied to require power in order to perform their safety function. The temporary power supplied is a known capacity and quality, i.e., regulated to maintain voltage at 118 Vac plus 3 1/2% to 2 1/2% with less than 5% harmonic distortion. Therefore, there is no degradation in quality of power by using the temporary supply. The inverter itself is not essential, therefore failure does not increase the consequences of any accident. Further, the class 1E circuits are isolated from the temporary inverter output via a circuit breaker and fused disconnect.

The only failure postulated is a failure of the temporary inverter itself or the temporary cable supplied to the UPS distribution panel. The power supply, i.e., the temporary inverter is non-essential, and failure does not prevent any safety function. The circuits and instruments that are being supplied power to, do not require that power to perform their safety function. Sensors, sensor channels and trip logics of the reactor protection system are not used directly for automatic control of process system. Therefore, failure in the controls and instrumentation of process systems cannot induce failure in any portion of the protection system.

Since the power supply is not relied on to perform any safety function, the margin of safety as defined in the basis for any Technical Specifications will not be reduced.

SRASN: PLS-90-027 DOC NO: CR-90-011 SYSTEM: N11

DESCRIPTION OF CHANGE: This change makes the inspection interval of the Turbine Stop and Control Valve at least once in 40 months rather than the previous once each year.

REASON FOR CHANGE: This change makes the UFSAR agree with the Technical Specifications for the inspection interval.

SAFETY EVALUATION: This safety evaluation concluded that the change did not involve an unreviewed safety question. Turbine Stop and Control Valve operability testing for overspeed protection is addressed in both the UFSAR and the GGNS Technical Specifications. These Stop and Control Valves as well as the two overspeed devices are tested for operability every 14 days. This requirement remains unchanged. The UFSAR is only being changed to make the valve inspection cycle consistent with the requirements of the GGNS Technical Specification 4.3.9.2c.

The design function of the overspeed protection system is not affected and will perform in its intended manner. Overspeed protection for the Turbine/Generator is not compromised and will function in its intended manner to protect safety related components, equipment, and structures from damage induced by Turbine generated missiles. ATT testing of the overspeed system provides periodic testing to ensure operability and integrity of the Turbine Stop and Control Valves. Requirements of the UFSAR Section 10.2.3.6 for demonstrating the integrity of the overspeed protection system by exercising of the HP, LP, and Bypass Stop and Control Valves through ATT program testing remains the same. The overspeed trip system will perform its design function.

The margin of safety as defined in the basis for any Technical Specification remains unchanged. No GGNS Technical Specification is affected. Only the requirements for valve inspection intervals as listed in the UFSAR Section 10.2.3.6 is being changed to make it consistent with Technical Specifications.

SRASN: PLS-90-028 DOC NO: T.S. 3.0.4, ACTION C SYSTEM:

DESCRIPTION OF CHANGE: This safety evaluation addresses the use of Technical Specification 3.0.4 to enter Operational Condition 5 (High Water Level) from Operational Condition 5 (Low Water Level) while complying with Action Statement c for Technical Specification 3.5.3.

REASON FOR CHANGE: This safety evaluation documents the analysis of entry into Operational Condition 5 (High Water Level) from Operational Condition 5 (Low Water Level) when one suppression pool level instrumentation is inoperable due to removing ECCS jockey pump from service.

SAFETY EVALUATION: This safety evaluation concluded that the change did not involve an unreviewed safety question. The Suppression Pool is required to be OPERABLE as part of the ECCS to ensure that a sufficient supply of water is available to the HPCS, LPCS and LPCI systems in the event of a LOCA. This limit on suppression pool minimum volume ensures that sufficient water is available to permit recirculation cooling flow to the core. The OPERABILITY of the suppression pool in OPERATIONAL CONDITIONS 4 and 5 is not required by Specification 3.6.3.1 for pressure suppression. In OPERATIONAL CONDITIONS 4 and 5 the suppression chamber minimum required water volume is reduced because the reactor coolant is maintained at or below 200°F. Since pressure suppression is not required below 212°F, the minimum required water volume is based on NPSH, recirculation volume and vortex prevention plus a 1 foot 2 inches safety margin for conservatism. The UFSAR evaluates several accidents (events) which are considered to be applicable during OPERATIONAL CONDITIONS 4 and 5. The majority of these are unrelated to the proposed application of TS 3.0.4 for TS 3.5.3 and thus the probability of occurrence of these events does not increase. The reactor drain down event is not specifically addressed in the UFSAR during OPERATIONAL CONDITIONS 4 and 5; however, events that result in reactor vessel inventory loss (i.e., reactor drain down) are most directly affected by the use of TS 3.0.4 while in ACTION c of TS 3.5.3. In accordance with TS 3.5.2 and 3.5.3, ECCS and the suppression pool are not required to be OPERABLE provided that the reactor vessel head is removed, the cavity is flooded or being flooded from the suppression pool, the reactor cavity and transfer canal gates in the upper containment pool are removed when the cavity is flooded and the water level is maintained within the limits of TS 3.9.8 and 3.9.9. Therefore, provided TS 3.5.3, ACTION c is complied with (verify suppression pool level at least once per 12 hours by an alternate indicator) the flexibility provided by the provisions of TS 3.0.4 does not increase probability of an accident previously evaluated in the UFSAR.

The accidents considered by the UFSAR during shutdown conditions are not changed by the use of TS 3.0.4. This application of TS 3.0.4 neither adds or removes systems or components, nor does it change present system design features or plant operating procedures. No new mechanism for draining the reactor vessel is created.

The bases for TS 3.5.3 discusses the need for suppression pool volume during OPERATIONAL CONDITIONS 4 and 5 is to prevent NPSH concerns, provide recirculation cooling volume and vortex prevention. Complying with TS 3.5.3 ACTION c will ensure that these concerns and the margin preventing these concerns are adequately addressed during flooding of the reactor cavity.

SRASN: PLS-90-029 DOC NO: CR-90-006 SYSTEM: N11

DESCRIPTION OF CHANGE: The purpose of this safety evaluation is to evaluate the impact on plant safety of deleting sentence 8 of the UFSAR Section 10.2.2.4.

REASON FOR CHANGE: UFSAR 10.2.2.4 sentence 8 states "All motor-operated valves will be bench tested or in-place tested", referring to motor-operated valves associated with the Turbine Generator system. The sentence does not specify the type of testing or the basis for such testing. No other requirements exist which reference this sentence or explain the type or basis for the testing. Due to these facts, and based on a review of the plant programs currently in place to monitor the condition of the motor operated valves, it has been determined that Sentence 8 of UFSAR Section 10.2.2.4 serves no useful purpose and should be deleted from the UFSAR.

SAFETY EVALUATION: This safety evaluation concluded that the change did not involve an unreviewed safety question. If any of the motor-operated valves malfunction, such that steam flow could not be stopped, the steam would be exhausted into the condenser, and only result in a loss of plant megawatt output. Also, the steam supply to these valves will be isolated in an accident. The motor operated valves do not directly or indirectly affect any components necessary for safety. There is no equipment important to safety which could be affected by deleting the bench or in-place testing of all motor-operated valves in the turbine generator system.

Deleting in-place or bench testing of all motor-operated valves in the turbine generator system does not reduce the margin of safety as defined in the basis for any of the Technical Specifications, because there are not safety functions or safety limits which are associated or affected by the testing of these valves.

SRASN: PLS-90-030 DOC NO: TEMP ALT 90-0004 SYSTEM: P47

DESCRIPTION OF CHANGE: This change abandons the prelube system on radial wells 1, 3 and 5.

REASON FOR CHANGE: The subject prelube systems were unnecessary for proper operation of the radial wells and were high maintenance items.

SAFETY EVALUATION. This safety evaluation concluded that the change did not involve an unreviewed safety question. This temporary alteration (TA) does not affect the operation or reliability of any safety related system. No accident evaluated in the UFSAR is affected by this TA. This TA does not affect the operation or reliability of the radial well system as described in the FSAR.

SRASN: PLS-90-031 DOC NO: W.O. 27751 SYSTEM: D17

DESCRIPTION OF CHANGE: This change provides temporary BOP power for the Auxiliary Building Fuel Handling Area Ventilation Exhaust, the Auxiliary Building Fuel Handling Area Pool Sweep Exhaust, the Containment and Drywell Ventilation Exhaust and the Control Room Ventilation Radiation Monitoring Systems.

REASON FOR CHANGE: The reason for the need for temporary BOP power was due to a Bus 15 and a Bus 16 outage.

SAFETY EVALUATION: This safety evaluation concluded that the change did not involve an unreviewed safety question. The loss of the radiation monitoring systems' temporary BOP power or degradation of that power will cause the initiation of the intended safety function. Loss of power to the radiation monitoring systems will actuate the appropriate annunciator in the main control room. Degradation of the power will cause initiation because decreasing voltage will cause a radiation monitor high voltage (downscale) inop trip. An increase in the temporary power's voltage will cause an increased radiation indication. A constant voltage transformer will be used to condition the BOP temporary power to maintain reliability of the radiation monitor power supplies. The constant voltage transformer's output will be held to 120 VAC with its input voltage varying from 95 to 130 VAC. With temporary power applied to the radiation monitoring system an isolation will occur upon a high-high radiation or an inop signal. A high radiation signal will cause an alarm in the main control room. Loss of power will cause an inop signal and thus an isolation will occur. Per the GGNS FSAR the safety functions of these radiation monitors is to isolate ventilation systems and or start the appropriate filtration system and to provide indication and alarm in the main control room.

This change does not reduce the margin of safety as described in the basis for any Technical Specification because the margin of safety is maintained by the initiation of the intended safety function. Any failure of the temporary power supply will result in the initiation required to ensure safety.

SRASN: PLS-90-032 DOC NO: MWO 26063 SYSTEM: R21

DESCRIPTION OF CHANGE: MWO 26063 provides temporary power from ESF Bus 16AB and BOP Buses 11HD and 13AD to loads normally supplied by Bus 15AA. The additional power requirements being placed on Buses 11HD and 13AD are negligible and no loading calculations were required. No additional load is being placed on Bus 16AB. No components being supplied temporary power will be considered operable. In all cases temporary power was being supplied as a matter of convenience and not plant safety. Required LCOs were entered when normal power was removed. All work was done while in Reactor Mode 5. Temporary power was supplied as shown in Table 1.

TABLE 1

Temporary Loads

Loads	Normal Power Supply	Temporary Power Supply
Battery Charger 1K4	52-15104	52-16106
Battery Charger 1D4	52-15102	52-132249
Lighting XFMR 1X113	52-154224	52-111217
Refuel Platform	52-154223	52-111217
SLC Operating Heater	52-154221	52-111219

REASON FOR CHANGE: To allow required maintenance and cleaning of the 15AA ESF Bus.

SAFETY EVALUATION: This safety evaluation concluded that the change did not involve an unreviewed safety question. Temporary power will be supplied in a similar manner as the normal power supply. Cable sizing and breaker selection will be such that adequate circuit protection is maintained. The loads being supplied temporary power will not be relied upon to perform a safety function. Review of the load shedding tables in the FSAR shows that all loads being supplied temporary power are non-essential. In any accident situation in which load shedding were to occur all loads listed in Table 1 would be shed, by either their normal or temporary supply. The only possible failure of the circuits supplying temporary power is their loss of power. Regardless of how that loss occurs, the end result is failure of component to function. Loss of power to all components listed in Table 1 has already been considered. Using Buses 11HD, 13AD and 16AB does not diminish the quality of power to the temporary loads, nor does it decrease the reliability of the power available to the loads normally supplied by Buses 11HD, 13AD and 16AB. Breaker 52-16106 which normally supplies power to battery charger 1L4 will be disconnected and reconnected to battery charger 1K4. This does not constitute a violation of divisional separation and it does not increase the load on ESF Bus 16AB since battery chargers 1K4 and 1L4 are identical. None of the loads being supplied power are required to perform safety functions, and in the case of any load shedding accident appropriate load shedding of loads in Table 1 will be accomplished.

Since Technical Specification requirements will be met with Division II and/or Division III operability, and none of the loads being supplied temporary power will be required to perform any safety function, the margin of safety as defined in the basis for any Technical Specification will not be reduced.

SRASN: PLS-90-0?? DOC NO: MWO 26064 SYSTEM: R21

DESCRIPTION OF CHANGE: MWO 26064 provides temporary power from ESF Bus 15AA and BOP Buses 12HE and 21HD to loads normally supplied by Bus 16AB. The additional power requirements being placed on Buses 12HE and 21HD are negligible and no loading calculations were required. No additional load is being placed on Bus 15AA. 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 were entered when normal power was removed. All work was done while in Reactor Mode 5. Temporary power was supplied as shown in Table 1.

TABLE 1

Temporary Loads

Loads	Normal Power Supply	Temporary Power Supply
Battery Charger 1E4	52-16102	52-124118
Battery Charger 1L4	52-16106	52-15104
Lighting XFMR 1X114	52-164211	52-125125
Drywell Floor Drain Sump Recorder	52-1P66111	Control Room Wall Socket
Unit 1 Inst. Air Dryer	52-1P64218	52-213104

REASON FOR CHANGE: To allow required maintenance and cleaning of the 16AB ESF Bus.

SAFETY EVALUATION: This safety evaluation concluded that the change did not involve an unreviewed safety question. Temporary power will be supplied in a similar manner as the normal power supply. Cable sizing and breaker selection will be such that adequate circuit protection is maintained. The loads being supplied temporary power will not be relied upon to perform a safety function. Review of the load shedding tables in the FSAR shows that all loads being supplied temporary power are non-essential. In any accident situation in which load shedding were to occur all loads listed in Table 1 would be shed, by either their normal or temporary supply. The only possible failure of the circuits supplying temporary power is their loss of power. Regardless of how that loss occurs, the end result is failure of component to function. Loss of power to all components listed in Table 1 has already been considered. Using Buses 12HE, 15AA and 21HD does not diminish the quality of power to the temporary loads. Nor does it decrease the reliability of the power available to the loads normally supplied by Buses 12HE, 15AA and 21HD. Breaker 52-15104 which normally supplies power to battery charger 1K4 will be disconnected and reconnected to battery charger 1L4. This does not constitute a violation of divisional separation and it does not increase the load on ESF Bus 15AA since battery chargers 1K4 and 1L4 are identical.

None of the loads being supplied power are required to perform safety functions, and in the case of any load shedding accident appropriate load shedding of loads in Table 1 will be accomplished.

Since Technical Specification requirements will be meet with Div I and/or Div III operability, and none of the loads being supplied temporary power will be required to perform any safety function, the margin of safety as defined in the basis for any Technical Specifications will not be reduced.

SRASN: PLS-90-043 DOC NO: TSTI-1G17-90-004-0-S SYSTEM: G17

DESCRIPTION OF CHANGE: This activity will add Dearborn 702 biocide to a condensate phase separator tank to stop microbiological activity in the tank and allow burial.

REASON FOR CHANGE: There are methane-producing bacteria present in the tank which cause pressurization of the radioactive waste liner when the liner is dewatered. The addition of 600 ppm Dearborn 702 is necessary to prevent the gas formation from occurring. The following steps were taken to develop this process:

- Samples of the tank were taken to determine the dosage necessary to kill the bacteria.
- The radwaste liner supplier was contacted to ensure that the use of Dearborn 702 would not adversely affect the liner.
- The ChemNuclear burial facility in Barnwell, South Carolina was contacted to ensure the treatment process was in accordance with the burial regulations at the site.
- The resin manufacturer was contacted to ensure the treatment process would not adversely affect the waste resin and filter media.
- The solidification process vendor was contacted to ensure the addition of this chemical would not adversely affect the solidification process.

This safety evaluation was applicable for the treatment of both the condensate phase separators and the Reactor Water Cleanup phase separators with Dearborn 702.

SAFETY EVALUATION: This safety evaluation concluded that the change did not involve an unreviewed safety question. As stated in UFSAR 11.4.1.1, the radwaste system "... is designed so that failure or maintenance of any frequently used component shall not impair system or plant operation." The performance of this activity does not change the operation of the phase separators, resin transfer, or dewatering equipment. The chemical to be used will not be detrimental to the equipment in the concentrations to be used. Inadvertent spillage of Dearborn 702 into the radwaste system could result in the early changeout of the demineralizer bed, but would not have any affect on the integrity of the piping or components.

The radwaste system is not necessary for the safe operation or shutdown of the plant. A failure in the radwaste system would have no applicable effect on the core or NSSS performance (15.7.2 and 15.7.3). Accidents involving the radwaste system are bounded by the UFSAR whole tank rupture analyses.

Accidents evaluated in the UFSAR involving the radwaste system are leaks/tank ruptures in the system. System operation is not changed and the chemical is not detrimental to the equipment. No different failure would be caused by this activity, which is bounded by these analyses of whole tank ruptures. This activity meets the requirements of the PCP addressed in Technical Specifications. This activity will not affect the activity of radwaste shipments.

SRASN: PLS-90-044 DOC NO: WO #29996 SYSTEM:

DESCRIPTION OF CHANGE: WO #29996 connects a mobile demin water trailer to temporarily supply demineralized water to the Demin Water Storage Tank. This supply of water will be made through the manual "modified" valve NSP21F077. The valve was modified to facilitate a connection for the temporary mobile demin water trailer through the valve bonnet via a hose connection from the trailer. Temporary valve connections controlled by Temp Directive 04-S-01-P21-1-TEMP 17 Rev. 0, will allow chemistry sampling and analysis of the water supply prior to connection through the modified valve to the Demin Water Storage Tank.

REASON FOR CHANGE: To provide a temporary demin water source to the Demin Water Storage Tank.

SAFETY EVALUATION: Since no FSAR accident is postulated on any demin water failure, this safety evaluation concluded that the change did not involve an unreviewed safety question. The subject activity represents a temporary change to P21 demin water system as described in the FSAR only because valve NSP21F077 will be in effect removed (will not serve as a valve) during the duration of the activity. Final supply water quality will be well within the conductivity parameters given in FSAR 9.2.3.1.2.

P21 demin water has no safety-related function. P21 serves no system or component in a way vital to reactor shutdown. There is no reduction in the margin of safety as defined in the basis for any Technical Specification, because no Tech Spec governs the filling of P21 demin water or the manipulation of valve NSP21F077.

SRASN: NLS-90-002 DOC NO: SERI Operations Manual to Operations Mgmt Manual SYSTEM:

DESCRIPTION OF CHANGE: This change changes the title of the SERI Operating Manual to the Entergy Operations Management Manual.

REASON FOR CHANGE: This manual title change reflects the new name of the company.

SAFETY EVALUATION: With this title change all the responsibilities and commitments being performed in the existing operating manual will continue to be performed. The title change will have no effect on plant design or operations; therefore, there will be no increase in probability of occurrence or consequences of accidents previously evaluated in the UFSAR; nor will there be any increase in probability of occurrence or consequences of a malfunction of equipment important to safety previously evaluated in the UFSAR; nor will there be created the possibility of an accident or malfunction of equipment important to safety different than any previously evaluated in the UFSAR.

SRASN: NLS-90-003 DOC NO: GGNS Emergency Plan Section 6.6.S6 SYSTEM:

DESCRIPTION OF CHANGE: The word "drinking" is deleted from the sentence in the Emergency Plan Section 6.6.S6 that stated: The requirements of 10CFR20, Appendix B, are met for air and drinking water.

REASON FOR CHANGE: 10CFR20, Appendix B is not applicable to drinking water.

SAFETY EVALUATION: This safety evaluation concluded that the change did not involve an unreviewed safety question. With this change, the Radioactive Liquid and Gaseous Waste Sampling and Analysis Program will continue to be performed as required in 3/4.11 of the GGNS Unit One Technical Specifications. The change will have no effect on plant design or operations; therefore, there will be no increase in probability of occurrence or consequences of accidents previously evaluated in the FSAR; nor will there be any increase in probability of occurrence or consequence of a malfunction of equipment important to safety previously evaluated in the FSAR; nor will there be created the possibility of an accident or malfunction of equipment important to safety different than any previously evaluated in the FSAR.

The BASES section of the Technical Specifications provide general requirements applicable to each of the Limiting Conditions for Operations and Surveillance Requirements within Section 3/4, and the justification for Safety System Settings. The bases for the Radioactive Effluents LCOs will not be altered as a result of this change; thus the margin of safety as defined in the bases for any Technical Specification is not reduced.

SRASN: NLS-90-004 DOC NO: TS 3.7.2, ACTION b.1 SYSTEM:

DESCRIPTION OF CHANGE: The safety evaluation addresses the use of TS 3.0.4 for entry into Operational Condition 4, 5, or * when one control room emergency filtration (CREF) subsystem is inoperable. The specified condition * is defined as "when irradiated fuel is being handled in the primary or secondary containment."

REASON FOR CHANGE: During refueling outages, situations may arise due to maintenance, implementation of modifications, or surveillances such that it is necessary to enter into one of the subject OPERATIONAL CONDITIONS or specified condition with a CREF subsystem inoperable. This evaluation assumes one CREF subsystem remains OPERABLE and is operating in the isolation mode of operation.

SAFETY EVALUATION: This safety evaluation concluded that the change did not involve an unreviewed safety question. The CREFS is not a component precursor to any of the accidents evaluated in the UFSAR. Additionally, the operation of one subsystem of the CREFS in the isolation mode as required by TS 3.7.2, Action b.1, is in accordance with safety design basis defined in the UFSAR. Operation of one of the CREFS subsystems in the isolation mode prior to or following OPERATIONAL CONDITION changes or specified condition changes does not affect its operations or the operation of equipment that could be precursors to accidents. The Safety design basis of the CREFS, in conjunction with other control room design provisions, is to ensure that the control room will remain habitable for operations personnel during and following all design basis conditions, and that the radiation exposure to the personnel will be 5 rem or less whole body in accordance with GDC 19 of Appendix A, 10 CFR Part 50. Operation of one subsystem of the CREFS in the isolation mode is in accordance with its safety design bases as defined in the UFSAR, and is valid for all OPERATIONAL CONDITIONS including 4, 5, and when handling irradiated fuel. The system is also designed to allow for isolation mode fresh air makeup to allow for dilution of carbon dioxide (C02) buildup. The system design allows for manual initiation of the fresh air makeup 10 minutes following initiation of the isolation mode; however, fresh air makeup is not required until approximately 72 hours following isolation based upon C02 buildup from respiration of 12 persons as described in UFSAR Section 6.4. Due to the potential buildup of C02 with the CREFS operating in the isolation mode for extended periods of time during OPERATIONAL CONDITIONS 4, 5, and handling of irradiated fuel in accordance with TS 3.7.2, Action b.1, and TS 3.0.4, C02 levels could potentially buildup to higher than normal levels. In accordance with Station Operating Instruction No. 04-S-01-Z51-1, Health Physics will sample the control atmosphere every 8 hours to ensure that the oxygen levels remain above 20% and the C02 levels remain below 1%. The fresh air makeup would then be utilized as required and is available for use within ten minutes after the isolation signal. The only accidents which are evaluated in the UFSAR that rely upon the CREFS for consequence mitigation are LOCA - inside containment (UFSAR 15.6.5, NSOA Event 37, Figure 15A.6-37) and steam line break outside containment (UFSAR 15.6.4, NSOA Events 38, 39, & 40, Figure 15A.6-38). The LOCA and steam line break accidents are not applicable to the OPERATIONAL CONDITIONS 4, 5, and when handling irradiated fuel being evaluated for this technical specification action statement. The fuel handling accident (UFSAR 15.7.4, NSOA Event 36, Figure 15A.6-36) and the liquid radwaste system failures (UFSAR 15.7, NSOA Events 43 & 44, Figures 15A.6-41, 42) were also reviewed for potential consequence impact. Again, operation of the CREFS as evaluated during OPERATIONAL CONDITIONS 4, 5, and when handling irradiated fuel, or when changing between these, does not impact these accidents. The operation of the CREFS as described in TS 3.7.2, Action b.1, is in accordance with the safety design basis of the CREFS, and does not impact the control room HVAC failure analysis and Control Room Atmospheric Control and Isolation System (CRACIS) failure analysis presented in UFSAR Table 9.4-2 and habitability of the control room and does not negatively impact the mild environment equipment qualification requirements for equipment in the control room. Operating the CREFS in the isolation mode during OPERATIONAL CONDITIONS 4, 5, and when handling irradiated fuel, or when entering into one of these conditions, does not have any control interface to other equipment important to safety. The change in OPERATIONAL CONDITIONS or specified condition with the CREFS operating in accordance with TS 3.7.2, Action B.1, is in accordance with and does not reduce the margin of safety described in Technical Specification Bases for TS 3.0.4. In addition, the TS Bases for TS 3.7.2 requires operation of the CREFS so that the control room will remain habitable for operations personnel during and following all design basis accident conditions, and limit the radiation exposure to 5 rem or less whole body. This requirement is met by one subsystem of the CREFS being placed in the isolation mode of operations and does not reduce the safety margin of the CREFS. This is applicable whether operating in one of the applicable conditions or entering into one of the conditions. Finally, the operation of the CREFS does not adversely impact or interact with other plant systems as described in the Technical Specification and their bases.

SRASN: NLS-90-005 DOC NO: TS 3.6.4, Actions b & c SYSTEM:

DESCRIPTION OF CHANGE: This evaluation addresses the safety implications of commencing core alterations and/or handling of irradiated fuel in the primary or secondary containment with containment and/or drywell penetrations already isolated by an acceptable method as allowed by TS 3.6.4 Action b or c as compared to taking these actions after beginning core alterations or the handling of irradiated fuel.

REASON FOR CHANGE: During refueling outages, various isolation valves must be made inoperable to perform maintenance, conduct surveillance tests and inspections, or implement design changes. TS 3.0.4 allows the plant to begin core alterations or the handling of irradiated fuel without having all required isolation valves OPERABLE provided that the requirements of the applicable Action Statements are met.

SAFETY EVALUATION: This safety evaluation concluded that the change did not involve an unreviewed safety question. The function of the containment and drywell isolation valves is to ensure that drywell and containment penetrations are isolated in the event of a radioactive release inside the containment. This assures that an environmental release of radioactive material is controlled to within the design leakage rate of the containment systems, thereby preventing offsite doses from exceeding those determined by plant safety analyses. During core alterations or the handling of irradiated fuel in OPERATIONAL CONDITION 5, certain containment, and drywell isolation valves (Groups 5, 6A, 6B, 7, 8, 10) are required to be OPERABLE as specified in TS 3.3.2 to mitigate radioactive releases which might occur. The UFSAR considers events which may potentially result in a radioactive release during refueling while performing core alterations or the handling of irradiated fuel. These include inadvertent criticality, failures of various plant systems and components, loss of offsite power, and fuel handling accidents. Of these, only the fuel handling accident inside containment generated a radiological release which results in the need for automatic isolation of containment and drywell penetrations. Should isolation valves become inoperable while performing core alterations or the handling of irradiated fuel, Action b or c may be entered to indefinitely provide an equivalent level of protection by isolating the affected penetrations. Under TS 3.0.4, Action b or c will be taken prior to beginning core alterations or the handling of irradiated fuel for those penetrations with inoperable isolation valves. Footnote * is also present to assure that isolation valves remain closed to provide a level of safety equivalent to the LCO when beginning core alterations or the handling of irradiated fuel. This flexibility has no affect on the methods or equipment used for fuel handling or the monitoring and control of refueling activities. The flexibility of TS 3.0.4 as applied in TS 3.6.4 also does not change or affect the number of activities defined as core alterations. There are no changes in refueling interlocks, so the probability of an inadvertent criticality is not increased.

Isolating any penetrations having inoperable isolation valves before beginning core alterations or the handling of irradiated fuel completely fulfills the safety function of the valves. The radiological consequences of a fuel handling accident will thus be no worse than analyzed. Also, none of the analyzed accident sequences are changed by isolating the affected penetrations prior to beginning core alterations or the handling of irradiated fuel rather than at some later time. Fuel handling techniques and equipment are not altered, monitoring and control methods are not modified, nor are the types of activities defined as core alterations changed. Refueling interlocks remain unchanged. No radioactive material release mechanism or path is created where none previously existed. Exercising the provisions of TS 3.0.4 in this case maintains the plant in an acceptably safe condition relative to the radiological consequences of potential accidents during core alterations or the handling of irradiated fuel.

This application of TS 3.0.4 may directly affect equipment important to safety in two ways. Firstly, the isolation valves and penetrations themselves will be affected due to the requirement to close and/or deactivate valves or affix blind flanges in order to isolate penetrations. Secondly, systems and equipment served by the penetrations may also be affected due to the blocking of various flow paths. Refueling equipment and other plant components are not impacted by this use of TS 3.0.4.

The containment/drywell penetrations will be isolated under the provisions of TS 3.6.4, Action b or c in the event that their isolation valves are made inoperable for outage activities prior to or while core alteration or irradiated fuel handling activities were underway. There is no additional effect on the valves and penetrations themselves as a result of performing the isolation prior to beginning core alterations or the handling of irradiated fuel. The method accomplishing the required isolation is identical in either case, and the maintenance or testing of the valves or penetrations will also be unchanged. Similarly, systems whose flow paths are altered as a result of isolated penetrations will be impacted in the same manner regardless of the timing of the Actions.

Having performed the required isolations prior to beginning core alterations or the handling of irradiated fuel may actually reduce the probability of an equipment malfunction by reducing the amount of system and component manipulation otherwise required. Without the relief of TS 3.0.4, each time core alterations or the handling of irradiated fuel were to commence, any isolation valves undergoing maintenance or testing would have to first be made OPERABLE. Then core alterations or irradiated fuel handling could begin and the valves subsequently declared inoperable and TS 3.6.4 Actions taken.

The equipment important to safety which may be directly affected by this application of TS 3.0.4 includes the isolation valves and penetrations required to be OPERABLE by Specification 3.6.4 during core alterations or the handling of irradiated fuel as well as the equipment and components in systems served by those penetrations. The radiological consequences of a malfunction of such equipment is not increased by taking the required actions to isolate penetrations prior to beginning core alterations or the handling of irradiated fuel rather than after core alterations or the handling of irradiated fuel have begun. The degree of isolation and the maintenance and testing to be done on the valves is the same in either case. There are also no changes to refueling procedures or monitoring capabilities. For other plant equipment important to safety not directly affected by this use of TS 3.0.4 but which may malfunction due to unrelated events, the radiological consequences of any such malfunction would be no more severe under TS 3.0.4. Should a release of radioactive material take place inside the containment during core alterations or the handling of irradiated fuel while under the requirements of TS 3.6.4 Action b or c, those requirements provide the necessary isolation for penetrations with inoperable isolation valves. The safety function of the valves has already been fulfilled by isolating the penetrations. This is true whether these Actions were taken before or after beginning core alterations or the handling of irradiated fuel.

The Bases for Technical Specification 3.6.4 discusses the necessity for the OPERABILITY of the containment and drywell isolation valves to prevent the release of radioactive material to the outside environment under postulated accident scenarios. During core alterations or the handling of irradiated fuel, the accident of concern for this Specification is a fuel handling accident inside containment and the margin of safety of interest as addressed in the UFSAR analysis, is the margin of 25% of 10CFR100 limits. Inadvertent criticality and other accidents considered during core alterations are either not possible or have no radiological consequences. Also of concern in the Bases are the closure times of the isolation valves to ensure that any release is terminated in a time frame consistent with safety analysis assumptions.

Under the flexibility of TS 3.0.4, any penetrations with required, but inoperable, isolation valves may be isolated in accordance with the requirements of Action b or c prior to beginning core alterations or the handling of irradiated fuel. Taking these actions at that time as compared to taking them after core alterations or the handling of irradiated fuel have begun does not impact any of the above considerations regarding the margin of safety. Since the isolation is already accomplished the safety function of the isolation valves is fulfilled and the penetration will be no more susceptible to allowing a radioactive release than if relying on automatic isolation. The isolation times are no longer of concern and penetration leak rates are unaffected.

Under this application of TS 3.0.4, the affected penetrations remain as capable of preventing a release as if the necessary isolations were taken after beginning core alterations or the handling of irradiated fuel. There are no changes to procedures, controls, or interlocks associated with core alterations or the handling of irradiated fuel which could result in a larger release of material inside the containment than previously calculated. Thus, the margins of safety described above are not reduced by the flexibility of TS 3.0.4 as applied to TS 3.6.4, Actions b and c.

SRASN: NLS-90-006 DOC NO: TS 3.6.6.2, Actions b & c SYSTEM:

DESCRIPTION OF CHANGE: This evaluation addresses the safety implication of commencing core alterations and/or the handling of irradiated fuel in the primary or secondary containment with secondary containment penetrations already isolated by an acceptable method as allowed by TS 3.6.6.2, Actions b or c as compared to taking these actions after beginning core alterations or the handling of irradiated fuel. This relief has been previously approved by the NRC for a limited-time exception.

REASON FOR CHANGE: During refueling outages, various isolation valves must be made inoperable to perform maintenance, conduct surveillance tests and inspections, or implement design changes. TS 3.0.4 allows the plant to begin core alterations or the handling of irradiated fuel without having all required isolation valves OPERABLE provided that the requirements of the applicable Action Statement are met.

SAFETY EVALUATION: This safety evaluation concluded that the change did not involve an unreviewed safety question. The function of the secondary containment isolation valves and dampers is to isolate secondary containment penetrations when necessary. This function, along with that of the Standby Gas Treatment System (SGTS), ensures that secondary containment integrity assures that environmental releases of radioactive material are minimized, thereby preventing offsite doses form exceeding those determined by plant safety analyses. During core alterations or the handling of irradiated fuel in OPERATIONAL CONDITION 5, all secondary containment isolation valves and dampers are required to be OPERABLE to mitigate radioactive releases which might occur. The UFSAR considers events which may potentially result in a radioactive releases during refueling while performing core alterations or the handling of irradiated fuel. These include inadvertent criticality, failures of various plant systems and components, loss of offsite power, and fuel handling accidents. Of these, only the fuel handling accident inside primary or secondary containment generates a radiological release which results in the need for isolation of secondary containment penetrations. Should isolation valves or dampers become inoperable while performing core alterations or the handling of irradiated fuel, Action b or c may be entered to indefinitely provide equivalent level of protection by isolating the affected secondary containment penetrations. Under TS 3.0.4, Action b or c will be taken prior to beginning core alterations or the handling of irradiated fuel for those penetrations with inoperable isolation valves or dampers. This flexibility has no affect on the methods or equipment used for fuel handling or the monitoring and control of refueling activities. The flexibility as applied in TS 3.6.6.2 also does not change or affect the number of activities defined as core alterations. There are no changes to refueling interlocks, so the probability of an inadvertent criticality is not increased.

For other plant equipment important to safety not directly affected by this use of TS 3.0.4 but which may malfunction due to unrelated events, the onsite or offsite radiological consequences of any such malfunctions would be no more severe under TS 3.0.4.

The margin of safety is not reduced by the flexibility of Technical Specification 3.0.4 as applied to Technical Specification 3.6.6.2.

SRASN: NLS-90-007 DOC NO: TS 3.4.9.2, Action a SYSTEM:

DESCRIPTION OF CHANGE: The Safety Evaluation documents the use of TS 3.0.4 for entry into OPERATIONAL CONDITION 4 from OPERATIONAL CONDITION 5 when one loop of RHR (either A or B) shutdown cooling is inoperable while in Action a of GGNS Technical Specification 3.4.9.2.

REASON FOR CHANGE: During planned refueling outage activities, situations may arise where one shutdown cooling loop (A or B) is inoperable in order to perform maintenance activities, surveillance tests, or design change implementation.

SAFETY EVALUATION: This safety evaluation concluded that the change did not involve an unreviewed safety question. The UFSAR evaluates several accidents (events) which are considered to be applicable during OPERATIONAL CONDITIONS 4 and 5. The majority of these events are unrelated to the proposed application of TS 3.0.4 for Technical Specification 3.4.9.2 in that their probability of occurrence is unaffected by the shutdown cooling system status or method by which shutdown cooling is provided.

If the alternate method is in service while tensioning the reactor vessel head closure bolts, the probability of a complete loss of shutdown cooling is not increased since an OPERABLE RHR shutdown cooling loop remains available in standby just as it would under full LCO compliance. If the alternate method were to fail, the standby loop could still be placed in operation as described in the UFSAR. This would be the case during a loss of offsite power as well, since the OPERABLE shutdown cooling loop is associated with an OPERABLE diesel generator.

Provided that TS 3.4.9.2 Action a is complied with (operability of one shutdown cooling loop of RHR with associated diesel generator and an alternate cooling method provided) the flexibility provided by the provisions of TS 3.0.4 does not increase the probability of the accident previously evaluated in the UFSAR. This application of TS 3.0.4 neither adds or removes systems or components, nor does it change present system design features or plant operating procedures. No new mechanism for draining the reactor vessel is created. No new or different procedures or equipment are used for tensioning the reactor vessel head closure bolts.

There are minimal radiological consequences to this event provided the appropriate mitigating actions are taken. These include 1. establishing shutdown cooling with alternate means or, if necessary, using Emergency Core Cooling modes available under TS 3.5.2 to maintain reactor water level. These actions may still be taken while under the requirements of Action a whether Action a was entered prior to tensioning the reactor vessel head closure bolts or after having done so. Further, tensioning the reactor vessel head closure bolts has no affect on the degree of decay heat generation by the reactor core. Should a complete loss of shutdown cooling occur while tensioning the bolts under Action a requirements, the consequences would therefore be no more severe since the amount of decay heat to be removed is unchanged.

Thus, tensioning the reactor vessel head closure bolts while already under the provisions of Action a as allowed by TS 3.0.4 has no affect on the consequences of accidents previously analyzed.

The Bases for Specification 3.4.9.2 do not specifically discuss margins of safety associated with the LCO. Discussions of the ability of only one shutdown cooling train to provide adequate decay heat removal capability imply that if a complete loss of shutdown cooling is prevented, there is no negative impact on plant safety. UFSAR 15.2.9 does discuss the failure of both redundant RHR shutdown cooling trains resulting in a complete loss of shutdown cooling event. Available mitigating actions such as injection from ECCS provide adequate cooling to prevent any temperature or pressure transients in excess of the criteria for which the fuel, pressure vessel, or containment are designed. These actions may be taken just as readily under the application of TS 3.0.4 described in this evaluation. The release of radioactivity to the environment is therefore not increased and remains bounded by more severe accidents such as a complete MSIV closure. This application of TS 3.0.4 does not change the protection against a complete loss of shutdown cooling capability provided by the requirements of Action a of TS 3.4.9.2. Should such an event occur, the necessary mitigating actions may still be taken to prevent damaging conditions. Thus, the margin of safety defined in the TS Bases is not reduced.

SRASN: NLS-90-008 DOC NO: TS 3.7.1.1, Action b SYSTEM:

DESCRIPTION OF CHANGE: This Safety Evaluation addresses the application of TS 3.0.4 when either SSW subsystem A or B is inoperable while entering OPCON 4 from 5.

REASON FOR CHANGE: During refueling outages, situations may arise where one service water subsystem (A or B) is made inoperable in order to perform maintenance or implement design changes. It may also be necessary to change plant OPERATIONAL CONDITIONS while in this situation to facilitate other planned outage activities. Provided that the above individual system LCOs are fully satisfied via the cooling capability of the remaining OPERABLE service water train, those LCOs do not impact any such changes. If reliance on any Action Statements of these Specifications is necessary, however, further consideration is required with regard to the flexibility allowed by TS 3.0.4 for the OPERATIONAL CONDITION or specified condition change contemplated. The provisions of TS 3.0.4 allow entry into an OPERATIONAL CONDITION or specified condition while complying with the requirements of an Action Statement only if those requirements allow continued operation in that situation for an unlimited period of time. Specifically, the case examined in this evaluation is entering OPERATIONAL CONDITION 4 from OPERATIONAL CONDITION 5 by tensioning the reactor vessel head closure bolts with either SSW A or B inoperable. Each Specification impacted by TS 3.7.1.1 Action b requirements must then be considered with respect to this change in OPERATIONAL CONDITION. There is no creation of a possibility for an accident or malfunction of a different type than any evaluated previously in the Safety Analysis Report. This application of TS 3.0.4 to TS 3.7.1.1, Action b does not change the protection against a complete loss of shutdown cooling capability provided by the requirements of TS 3.4.9.2, Action a. Should such an event occur, the necessary mitigating actions may still be taken to prevent damaging conditions. Thus, the margin of safety defined in the Technical Specification Bases is not reduced.

SAFETY EVALUATION: This safety evaluation concluded that the change did not involve an unreviewed safety question. This evaluation only addresses the entry into OPERATIONAL CONDITION 4 while under the requirements of Action a of TS 3.4.9.2 as directed by TS 3.7.1.1, Action b. Since all other LCOs directly or indirectly related to service water OPERABILITY in OPERATIONAL CONDITION 4 are satisfied, no other Specifications require consideration of TS 3.0.4 provisions relative to an increase in accident probability. Thus, the question of an increase in probability of occurrence of previously analyzed accidents must only be addressed relative to TS 3.4.9.2, Action a. The UFSAR evaluates several accidents (events) which are considered to be applicable during OPERATIONAL CONDITION 4. The majority of these events are unrelated to the proposed application of TS 3.0.4 for Technical Specification 3.4.9.2 in that their probability of occurrence is unaffected by the shutdown cooling system status or method by which shutdown cooling is provided. Consequently the probability of occurrence of these events does not increase. These are:

a. Losses or failures of various plant systems or components (other than f. below)
b. Inadvertent operation of various plant systems or components
c. Loss of AC power
d. Inadvertent criticality events

The UFSAR also considers two specific events related to shutdown cooling while shutdown:

e. Inadvertent increase in shutdown cooling. This accident is only significant near unit criticality and thus does not apply for this case (OPERATIONAL CONDITION 5 to 4).

f. Loss of shutdown cooling. TS 3.4.9.2 exists to ensure long term cooling capability while the reactor is shutdown. OPERATIONAL CONDITION 4 is entered from OPERATIONAL CONDITION 5 by tensioning the reactor vessel head closure bolts. Under the flexibility of TS 3.0.4, TS 3.4.9.2, Action a requirements will first be met by demonstrating an alternate method of decay heat removal prior to tensioning the head closure bolts. Tensioning the head closure bolts has no affect on decay heat generation or the alternate method provided. Since the alternate method provides for adequate decay heat removal capability, a complete loss of shutdown cooling is no more likely under these conditions than if the head closure bolts were tensioned with both RHR loops OPERABLE and Action a subsequently entered.

Also, if the alternate method is in service while tensioning the reactor vessel head closure bolts, the probability of a complete loss of shutdown cooling is not increased since an OPERABLE RHR shutdown cooling loop remains available in standby just as it would under full LCO compliance. If the alternate method were to fail, the standby loop could still be placed in operation as described in the UFSAR. This would be the case during a loss of offsite power as well, since the OPERABLE shutdown cooling loop is associated with an OPERABLE diesel generator. This application of TS 3.0.4 to TS 3.7.1.1, Action b does not change the protection against a complete loss of shutdown cooling capability provided by the requirements of TS 3.4.9.2, Action a. Should such an event occur, the necessary mitigating actions may still be taken to prevent damaging conditions. Thus, the margin of safety defined in the Technical Specification Bases is not reduced.

SRASN: NLS-90-009 DOC NO: TS 3.4.9.2, Action b SYSTEM:

DESCRIPTION OF CHANGE: This safety Evaluation addresses the application of TS 3.0.4 to enter OPERATIONAL CONDITION 4 from OPERATIONAL CONDITION 5 while relying upon an alternate reactor coolant circulation method as allowed by Action b of TS 3.4.9.2.

REASON FOR CHANGE: During planned refueling outage activities, situations may arise where no RHR shutdown cooling loop or recirculation pump is in operation due to maintenance or surveillance activities, and/or due to use of the RHR system in other designed modes. TS 3.4.9.2, Action b allows an alternate coolant circulation method to be established, along with appropriate monitoring to verify proper mixing. It may also be necessary to change OPERATIONAL CONDITIONS from 5 to 4 in this situation to facilitate preparation for plant startup. This will result in entering the jurisdiction of TS 3.4.9.2 under the requirements of Action b once the reactor vessel head closure bolts are tensioned.

SAFETY EVALUATION: This safety evaluation concluded that the change did not involve an unreviewed safety question. The UFSAR evaluates several accidents (events) which are considered to be applicable during OPERATIONAL CONDITIONS 4 and 5. The majority of these events are unrelated to the proposed application of TS 3.0.4 for Technical Specification 3.4.9.2 in that their probability of occurrence is unaffected by the status or method of reactor coolant circulation. There are also no changes to any system's design configuration, or operating procedures that would affect these events. Consequently, the probability of occurrence of these events does not increase. These are:

a. Losses or failures of various plant systems or components (other than those discussed separately below)

b. Inadvertent operation of various plant systems or components

c. Loss of AC power

d. Inadvertent criticality events

The UFSAR also considers some events related to reactor coolant circulation while shutdown:

e. Inadvertent increase in shutdown cooling. This event is only significant near unit criticality and thus does not apply for this case.

f. Loss of shutdown cooling. OPERATIONAL CONDITION 4 is entered from OPERATIONAL CONDITION 5 by tensioning the reactor vessel head closure bolts. Adequate shutdown cooling methods are provided by adherence to TS 3.9.11.2 when lowering reactor cavity level to prepare for head placement and tensioning, and by adherence to TS 3.4.9.2 after tensioning. The alternate reactor coolant circulation method established for Action b of TS 3.4.9.2 may also serve as an approved decay heat removal method if one RHR shutdown cooling loop is inoperable (TS 3.9.11.2, Action a). However, adequate decay heat removal capability is still provided whether it is put into place prior to or after head bolt tensioning. Tensioning the head bolts has no effect on decay heat generation nor does it result in different methods being used for decay heat removal. A complete loss of shutdown cooling is no more likely while under reliance upon Action b than if OPERATIONAL CONDITION 4 were entered and Action b subsequently taken. This would also be the case during a loss of AC power, since an OPERABLE RHR shutdown cooling loop associated with an OPERABLE diesel generator remains available. (Note that a separate evaluation is required for entry into OPERATIONAL CONDITION 4 under TS 3.4.9.2, Action a requirements.)

g. Recirculation Loop Pump Trips. Loss of recirculation pump flow results in loss of forced reactor coolant circulation within the vessel. This safety evaluation already assumes that no recirculation pump is in operation so that the requirements of Action b apply. This event is therefore not applicable to this evaluation. Further, for shutdown conditions, the UFSAR considers a loss of recirculation flow to be within the bounds of normal operation.

The UFSAR does not specifically consider the loss of reactor coolant circulation while shutdown as an accident or event requiring mitigating actions to prevent potential radiological consequences. However, the occurrence of such events in the industry has led to establishing operational procedures and restrictions to prevent and mitigate the consequences of a loss of reactor coolant circulation while shutdown. Under Action b requirements, an alternate method is established to provide the required circulation, and reactor temperature and pressure monitoring are implemented to verify the effectiveness of the alternate method. The methods available have been shown to provide the necessary mixing and monitoring capability. Whether alternate reactor coolant circulation methods and system monitoring are put into place prior to tensioning the reactor vessel head closure bolts under Action b, or after doing so has no affect on the likelihood of a loss of coolant circulation. The alternate method put into place is the same in either case, and once the vessel head is positioned, the tensioning procedure has no affect on reactor coolant circulation or the ability to monitor temperature and pressure.

The application of TS 3.0.4 also does not increase the probability of a vessel draindown, although this event is not specifically considered in the UFSAR while shutdown. The alternate reactor coolant circulation method will be the same regardless of the timing of its implementation, and the tensioning of the reactor vessel head closure bolts has no affect on draindown potential.

The consequences of other events related to coolant circulation will also be no more severe under this proposed application of TS 3.0.4. The same mitigating actions may still be taken as described in the UFSAR for a loss of shutdown cooling. These include re-establishing shutdown cooling with alternate means or, if necessary, using Emergency Core Cooling modes available under TS 3.5.2. Tensioning the reactor vessel head closure bolts to enter OPERATIONAL CONDITION 4 while under TS 3.4.9.2, Action b has no affect on the availability of these methods or the amount of decay heat generated. Should a complete loss of shutdown cooling occur, the consequences would therefore be no more severe.

The equipment important to safety under consideration for this evaluation involves systems or components associated with alternate reactor coolant circulation methods, or equipment which may be affected by failure to provide adequate coolant circulation. Since Action b requirements of TS 3.4.9.2 provide for sufficient circulation and monitoring of vessel condition, the probability of a malfunction of such equipment remains essentially the same whether Action b is entered after tensioning of the reactor vessel head closure bolts or prior to tensioning. The method and equipment used for tensioning are also no different in either case, so the tensioning procedure would not affect equipment important to safety in a different way.

The UFSAR also discusses the possible harmful effects of thermal stratification on the vessel during periods of low coolant circulation. Limits are placed on recirculation pump restart during such conditions to prevent thermally induced stresses in excess of vessel design. These limitations are unchanged by this application of TS 3.0.4, therefore the possibility of vessel damage due to thermal stratification from a loss of reactor coolant circulation is also not increased.

The Bases for Specification 3.4.9.2 do not specifically discuss margins of safety associated with the LCO. Discussions of the ability of only one shutdown cooling train to provide adequate coolant mixing imply that if a complete loss of coolant circulation for a time period sufficient to induce thermal stratification is prevented, there is no negative impact on plant safety. Other TS Bases discuss the need to prevent thermal stratification in order to assure accurate temperature indication and allow for proper mixing of neutron poison solution should it be needed, as well as to prevent undue thermal stresses on the vessel when mixing is reestablished.

Establishing an alternate coolant circulation method under TS 3.4.9.2, Action b requirements prior to tensioning the reactor vessel head closure bolts as opposed to after doing so does not negatively impact the ability to prevent thermal stratification. The time to reestablish coolant circulation is not increased. Available coolant circulation methods are no different and monitoring instruments are unaffected. Tensioning the head closure bolts to enter OPERATIONAL CONDITION 4 does not affect coolant circulation or the actions to be taken in the event circulation is lost. Also, the limitations imposed to prevent thermal stresses when restarting a recirculation pump are unchanged should this be the method selected to reestablish coolant circulation. This application of TS 3.0.4 thus does not reduce the margin of safety as defined in the Technical Specification Bases.

SRASN: NLS-90-010 DOC NO: TS 3.6.4, Actions b & c SYSTEM:

DESCRIPTION OF CHANGE: This evaluation addresses the safety implication of commencing operations with a potential for draining the reactor vessel (OPDRVs) during OPERATIONAL CONDITION 4 or 5 with containment and/or drywell penetrations already isolated by an acceptable method as allowed by TS 3.6.4 Action b or c as compared to taking these actions after beginning OPDRVs. This relief has been previously approved by the NRC for a limited-time exception.

REASON FOR CHANGE: During refueling outages, various isolation valves must be made inoperable to perform maintenance, conduct surveillance tests and inspections, or implement design changes. TS 3.0.4 allows the plant to begin OPDRVs without having all required isolation valves OPERABLE provided that the requirements of the applicable Action Statements are met.

SAFETY EVALUATION: This safety evaluation concluded that the change did not involve an unreviewed safety question. The function of the containment and drywell isolation valves is to ensure that drywell and containment penetrations are isolated in the event of a radioactive release inside the containment. This assures that an environmental release of radioactive material is controlled to within the design leakage rate of the containment systems, thereby preventing offsite doses from exceeding those determined by plant safety analyses.

During OPDRVs in OPERATIONAL CONDITIONS 4 or 5, certain containment and drywell isolation valves (Groups 5, 6A, 6B, 7, 8, 10) are required to be OPERABLE as specified in TS 3.3.2 to mitigate radioactive releases which might occur. The UFSAR considers events which may potentially result in a radioactive release during shutdown and refueling. These include inadvertent criticality, failures of various plant systems and components, loss of offsite power, and fuel handling accidents. Of these, only the fuel handling accident inside containment generates a radiological release which results in the need for automatic isolation of containment and drywell penetrations. The UFSAR does not specifically consider a vessel draindown while shutdown.

Should isolation valves become inoperable while performing OPDRVs, Action b or c may be entered to indefinitely provide an equivalent level of protection by isolating the affected penetrations. Under TS 3.0.4, Action b or c will be taken prior to beginning OPDRVs for those penetrations with inoperable isolation valves. The flexibility of TS 3.0.4 as applied in TS 3.6.4 also does not change or affect the number of activities defined as OPDRVs. There are no changes in refueling interlocks, so the probability of an inadvertent criticality is not increased.

Isolating any penetrations having inoperable isolation valves before beginning OPDRVs completely fulfills the safety function of the valves. The radiological consequences of a fuel handling accident will thus be no worse than analyzed. Also, none of the analyzed accident sequences are changed by isolating the affected penetrations prior to beginning OPDRVs rather than at some later time. Fuel handling techniques and equipment are not altered, monitoring and control methods are not modified, nor are the types of activities defined as OPDRVs changed. Refueling interlocks remain unchanged. No radioactive material release mechanism or path is created where none previously existed. Exercising the provisions of TS 3.0.4 in this case maintains the plant in an acceptably safe condition relative to the radiological consequences of potential accidents during OPDRVs.

This application of TS 3.0.4 may directly affect equipment important to safety in two ways. Firstly, the isolation valves and penetrations themselves will be affected due to the requirement to close and/or deactivate valves or affix blind flanges in order to isolate penetrations. Secondly, systems and equipment served by the penetrations may also be affected due to the blocking of various flow paths. Refueling equipment and other plant components are not impacted by this use of TS 3.0.4.

The containment/drywell penetrations will be isolated under the provisions of TS 3.6.4, Action b or c in the event that their isolation valves are made inoperable for outage activities prior to or while OPDRVs were underway. There is no additional effect on the valves and penetrations themselves as a result of performing the isolation prior to beginning OPDRVs. The method accomplishing the required isolation is identical in either case, and the maintenance or testing of the valves or penetrations will also be unchanged. Similarly, systems whose flow paths are altered as a result of isolated penetrations will be impacted in the same manner regardless of the timing of the Actions.

Malfunctions of equipment important to safety have been considered in the UFSAR for plant conditions associated with shutdown and refueling. The equipment which could be affected by this application of TS 3.0.4 includes the isolation valves and penetrations addressed in Specification 3.6.4, as well as the systems and components served by these penetrations. The possibility of failure of isolation valves is considered in that redundant isolation capability is provided. While under the requirements of Action b or c, this protection is preserved because any penetrations having inoperable isolation valves will already be isolated prior to the initiation of OPDRVs. Performing this action prior to beginning OPDRVs rather than after as allowed under TS 3.0.4 does not create the opportunity for a new or different type of malfunction of the isolation valves.

The Bases for Technical Specification 3.6.4 discusses the necessity for the OPERABILITY of the containment and drywell isolation valves to prevent the release of radioactive material to the outside environment under postulated accident scenarios. During OPDRVs, the accident of concern for this Specification is a fuel handling accident inside containment and the margin of safety of interest, as addressed in the UFSAR analysis, is the margin to 25% of 10CFR100 limits. Inadvertent criticality and other accidents considered during shutdown and refueling are either not possible or have no radiological consequences. The UFSAR does not specifically address vessel draindown events while shutdown. Also of concern in the Bases are the closure times of the isolation valves to ensure that any release is terminated in a time frame consistent with safety analysis assumptions.

Under the flexibility of TS 3.0.4, any penetrations with required, but inoperable, isolation valves may be isolated in accordance with the requirements of Action b or c prior to beginning OPDRVs. Taking these actions at that time as compared to taking them after OPDRVs have begun does not impact any of the above considerations regarding the margin of safety.

SRASN: NLS-90-011 DOC NO: TS 3.6.6.2, Actions b & c SYSTEM:

DESCRIPTION OF CHANGE: This evaluation addresses the safety implication of commencing operations with a potential for draining the reactor vessel (OPDRVs) during OPERATIONAL CONDITION 4 or 5 with containment and/or drywell penetrations already isolated by an acceptable method as allowed by TS 3.6.4 Action b or c as compared to taking these actions after beginning OPDRVs. This relief has been previously approved by the NRC for a limited-time exception.

REASON FOR CHANGE: Technical Specification 3.6.6.2 identifies operability requirements for secondary containment automatic isolation valves and dampers in OPERATIONAL CONDITIONS 1, 2, and 3 and at other times including during operations with a potential for draining the reactor vessel (OPDRVs). During refueling outages, various isolation valves must be made inoperable to perform maintenance, conduct surveillance tests and inspections, or implement design changes. TS 3.0.4 allows the plant to begin OPDRVs without having all required isolation valves OPERABLE provided that the requirements of the applicable Action Statements are met.

SAFETY EVALUATION: This safety evaluation concluded that the change did not involve an unreviewed safety question. The function of the secondary containment isolation valves and dampers is to isolate secondary containment penetrations when necessary. This function, along with that of the Standby Gas Treatment System (SGTS), ensures that secondary containment integrity is maintained when required. Secondary containment integrity assures that environmental releases of radioactive material are minimized, thereby preventing offsite doses from exceeding those determined by plant safety analyses.

During OPDRVs in OPERATIONAL CONDITIONS 4 or 5, all secondary containment isolation valves and dampers are required to be OPERABLE to mitigate radioactive releases which might occur. The UFSAR considers events which may potentially result in a radioactive release during shutdown and refueling. These include inadvertent criticality, failures of various plant systems and components, loss of offsite power, and fuel handling accidents. Of these, only the fuel handling accident inside primary or secondary containment generates a radiological release which results in the need for isolation of secondary containment penetrations. The UFSAR does not specifically consider a vessel draindown while shutdown.

Should isolation valves or dampers become inoperable while performing OPDRVs, Action b or c may be entered to indefinitely provide an equivalent level of protection by isolating the affected secondary containment penetrations. Under TS 3.0.4, Action b or c will be taken prior to beginning OPDRVs for those penetrations with inoperable isolation valves or dampers. This flexibility has no affect on the methods or equipment used for fuel handling or the monitoring and control of refueling activities. The flexibility as applied in TS 3.6.6.2 also does not change or affect the number of activities defined as OPDRVs. There are no changes to refueling interlocks, so the probability of an inadvertent criticality is not increased. The SGTS is unaffected and remains able to provide its mitigating function. Exercising the provisions of TS 3.0.4 in this case maintains the plant in an acceptable safe condition relative to the radiological consequences of potential accidents during OPDRVs.

This application of TS 3.0.4 may affect equipment important to safety in two ways. Firstly, the isolation valves, dampers and penetrations themselves will be directly affected due to the requirement to close and/or deactivate valves/dampers or affix blind flanges in order to isolate penetrations. Secondly, systems and equipment served by the penetrations may also be affected due to the blocking of various flow paths. Refueling equipment and other plant components are not directly impacted by this use of TS 3.0.4.

The Bases for Technical Specification Section 3.6.6 discusses the necessity for the OPERABILITY of the secondary containment isolation valves and dampers to prevent the release of radioactive material to the outside environment under postulated accident scenarios. During OPDRVs, the accident of concern for this Specification is a fuel handling accident inside primary or secondary containment and the margin of safety of interest, as addressed in the UFSAR analysis, is the margin to 25% of 10CFR100 limits. This is derived from the interpretation of the NRC Standard Review Plan. Inadvertent criticality and other accidents considered during shutdown and refueling are either not possible or have no radiological consequences. The UFSAR does not specifically address vessel draindown events while shutdown. Also of concern in the Bases are the closure times of the isolation valves and dampers to ensure that any release is terminated in a time frame consistent with safety analysis assumptions.

Under this application of TS 3.0.4, the affected penetrations remain capable of preventing a release as if the necessary isolations were taken after beginning OPDRVs. There are also no changes to procedures, controls, or interlocks associated with OPDRVs that could result in a larger release of material inside the primary or secondary containment than previously calculated.

Thus, the margins of safety described above are not reduced by the flexibility of TS 3.0.4 as applied to TS 3.6.6.2.

SRASN: NLS-90-012 DOC NO: CR-NL-90-009 SYSTEM:

DESCRIPTION OF CHANGE: This UFSAR change revised the outage data for the offsite 500 kV transmission lines. A decrease (0.90 to 0.94 outages/year/100 miles) in the overall performance of the 500 kV system was realized. Also the data was changed for the 115 kV transmission line between Natchez SES and Baxter Wilson SES and to GGNS. The 115kV transmission line has experienced an overall outage rate of 1.79 outages/year/100 miles compared to an overall rate of 1.65 currently in the UFSAR.

REASON FOR CHANGE: The line outage data is revised annually to update the UFSAR. No physical changes to the transmission lines were made under this evaluation.

SAFETY EVALUATION: This safety evaluation concluded that the change did not involve an unreviewed safety question. The changes to the UFSAR consist of revisions to MP&L transmission line outage data. Information on outages for the period from June 1, 1989 to May 31, 1990 was added to Chapter 8. Statistics on the transmission line outage rate are routinely updated based on the new data. In addition, data for previous years is corrected based on information received from MP&L. The changes to the UFSAR reflect the actual performance of the MP&L transmission system. No physical change to GGNS, the operation of GGNS, or the three offsite power sources has occurred.

The outage rate has increased slightly over the values currently in the UFSAR. UFSAR Section 15.2.2 identifies grid disturbances that cause closure of the turbine control valves as events of moderate frequency (1 to 0.05 events per year). The slight increase in outage rate does not change the classification as an event of moderate frequency. Thus the Chapter 15 analysis is not affected. The probability or consequences of an accident or malfunction in the Chapter 15 analysis are not changed.

A loss of all grid connections has been analyzed in the UFSAR. No change to the plant design or operation are being made so no possibility of an accident or malfunction different than previously evaluated is created.

There is no reduction in the margin of safety as defined in the basis for any Technical Specification. The action requirements specified in the Technical Specifications assume a loss of offsite power and are intended to provide assurance that a loss of offsite power will not result in a complete loss of safety function of critical systems.

SRASN: NLS-90-013 DOC NO: TS 3.9.11.2, Action a SYSTEM:

DESCRIPTION OF CHANGE: This safety evaluation addresses the use of TS 3.0.4 to enter the APPLICABILITY of LCO 3.9.11.2 by tensioning the reactor pressure vessel head closure bolts and entering OPERATIONAL CONDITION 5 from 4 while complying with Action Statement a.

REASON FOR CHANGE: Should one or more of the required shutdown cooling systems become inoperable, ACTION a allows the plant to remain in this condition indefinitely provided that an OPERABLE alternate method of decay heat removal is made available for the inoperable system. During planned outage activities, situations may arise where one or more shutdown cooling systems are inoperable in order to perform maintenance activities, surveillance tests, or design change implementation. This condition is allowed by TS 3.9.11.2; however, such situations require compliance with Action a of TS 3.9.11.2 to ensure adequate plant protection while the reactor cavity water level is less than 22 feet 8 inches in OPERATIONAL CONDITION 5.

SAFETY EVALUATION: This safety evaluation concluded that the change did not involve an unreviewed safety question. The UFSAR evaluates several accidents (events) which are considered to be applicable during OPERATIONAL CONDITION 5. The majority of these events are unrelated to the proposed application of TS 3.0.4 for Specification 3.9.11.2 in that their probability of occurrence is unaffected by the status of shutdown cooling or the mechanism by which shutdown cooling is being provided. Thus, the probability of occurrence of these events is not increased. These are:

a. Loss of plant instrument air

b. Fuel loading and handling errors

c. Radwaste system malfunctions

d. Rod withdrawal errors

e. Inadvertent pump starts (HPCS, Recirc.)

f. Feedwater controller failure

g. Loss of onsite or offsite AC power

The UFSAR also considers two specific events related to shutdown cooling while shutdown:

h. Inadvertent Increase in Shutdown Cooling - moderator temperature decrease

As stated in the UFSAR (Reference 5) this accident is only of concern during startup or cooldown near critical which is not the case in this situation.

i. Loss of Shutdown Cooling (References 6 and 9)

Even though event (i) is not considered a Design Basis Accident, the Technical Specification LCOs (including 3.9.11.2) are provided to maintain the probability and consequences of such previously evaluated events consistent with analyses by ensuring that equipment and systems assumed in the analyses remain operable. When TS 3.9.11.2 is not met due to one RHR shutdown cooling mode train and ADHRS being inoperable, Action a allows continued operation for an unlimited period of time by providing for an alternate method capable of decay heat removal. This alternate method provides protection in the event the remaining RHR train also becomes inoperable.

Under the provisions of TS 3.0.4, detensioning the reactor vessel head closure bolts is allowed provided an alternate method of decay heat removal has been demonstrated. The method of detensioning the closure bolts does not change whether detensioning is being performed while:

a. already under reliance of an alternate method of decay heat removal in accordance with Action a; or,

b. in full compliance with the LCO and subsequently entering Action a once the reactor vessel head closure bolts are detensioned.

Also, if the alternate method is in service while detensioning the closure bolts, the probability of a complete loss of shutdown cooling is not increased since an OPERABLE RHR shutdown cooling train remains available in standby as it would under full LCO compliance. If the alternate method were to fail, the standby RHR train could still be placed in operation as described in the UFSAR. This would be the case during a loss of offsite power as well, since the OPERABLE RHR shutdown cooling train is associated with an OPERABLE diesel generator.

The likelihood of a complete loss of shutdown cooling may in fact be decreased by taking steps to demonstrate an alternate method prior to beginning closure bolt detensioning. Without the flexibility of TS 3.0.4, outage activities would have to be interrupted to make the affected shutdown cooling system operable prior to bolt detensioning. After a closure bolt is detensioned, the RHR loop and/or ADHRS would again be made inoperable and Action a entered. Should the remaining OPERABLE RHR shutdown cooling loop fail before the alternate method has been adequately demonstrated, the time to provide alternate cooling is less since no method of shutdown cooling remains OPERABLE.

This application of TS 3.9.4 also does not affect the potential for draining the reactor vessel since no procedures are changed or equipment modified, although vessel draindown is not specifically addressed in the UFSAR for refueling.

As stated above, the only UFSAR analyzed accident requiring consideration for this application of TS 3.0.4 is a loss of shutdown cooling during refueling. There are no radiological consequences to this event provided the appropriate mitigating actions are taken. These include re-establishing shutdown cooling with alternate means or, if necessary, using Emergency Core Cooling modes available under TS 3.5.2 to maintain reactor water level. These actions may still be taken while under the requirements of TS 3.9.11.2, Action a whether Action a was entered prior to closure bolt detensioning or after. Further, closure bolt detensioning has no affect on the degree of decay heat generation by the reactor core. Should a complete loss of shutdown cooling occur while under Action a requirements, the consequences would therefore be no more severe since the amount of decay heat to be removed is unchanged.

The equipment important to safety under consideration for this evaluation involves systems or components associated with RHR or the alternate methods of decay heat removal, or equipment which may be affected by failure to provide adequate shutdown cooling while refueling. Since the Action a requirements of TS 3.9.11.2 provide sufficient shutdown cooling capability, the probability of a malfunction of such equipment remains essentially the same whether Action a is entered after closure bolt detensioning or prior to closure bolt detensioning. The method of bolt detensioning is the same in either case, so this process would also not affect equipment important to safety in a different way.

Technical Specification 3.9.11.1 and 3.9.11.2 Bases discuss the requirement for the RHR shutdown cooling system and ADHRS to provide sufficient cooling capability to remove decay heat to maintain the average reactor coolant temperature below 140°F. The TS 3.9.11.2 requirement to have RHR shutdown cooling trains and/or ADHRS OPERABLE when reactor cavity water level is less than 22 feet 8 inches ensures that a complete loss of shutdown cooling capability will not occur. By demonstrating the OPERABILITY of an approved alternate decay heat removal method should one of the required RHR shutdown cooling trains or ADHRS become inoperable per Action a, adequate shutdown cooling capability is maintained.

Under TS 3.0.4, the OPERABILITY of the alternate method is demonstrated prior to detensioning the closure bolts, as opposed to detensioning the head closure bolts and subsequently demonstrating its OPERABILITY with an RHR loop and ADHRS inoperable. Both situations provide essentially equivalent decay heat removal sufficient to maintain the average reactor water temperature below 140°F. In fact, under reliance on TS 3.0.4, having demonstrated the availability of a backup decay heat removal method before bolt detensioning may reduce the amount of time necessary to place such a system in service.

Thus, the proposed application of TS 3.0.4 does not decrease the margin of safety as discussed in the Technical Specification Bases.

SRASN: NLS-90-015 DOC NO: CR-NL-90-006 SYSTEM:

DESCRIPTION OF CHANGE: These changes in the areas of organization, communications and related fields were made over the past year. These changes affected senior management down to plant staff management.

REASON FOR CHANGE: These changes updated the UFSAR.

SAFETY EVALUATION: This safety evaluation concluded that the change did not involve an unreviewed safety question. The administrative changes are intended to reflect the current structure of both onsite and offsite organizations, which are not being handled/addressed by other Change Requests. Each substantive change is designed to consolidate and strengthen management and administrative functions, and provide a more effective management chain-of-command. Individuals assigned to any newly created positions are required to meet the qualifications specified in the UFSAR.

The changes to the communication system reflect recent transfers in ownership and equipment upgrades, which are designed to enhance system coverage and reliability. There are no general design criteria or regulatory guides that directly apply to the safety-related performance requirements for the design and use of the communication system during normal plant operations and transient conditions.

The organizational changes have no impact on the margin of safety due to their administrative nature. Qualification requirements for newly created positions meet any applicable standards represented in UFSAR Section 13.1.3. The changes to the communication system reflect equipment upgrades, which are designed to enhance system coverage and reliability. There are no general design criteria or regulatory guides that directly apply to the safety-related performance requirements for the design and use of the communication system during normal plant operations and transient conditions.

SRASN: NLS-90-016 DOC NO: TS POS STMT 128, Rev. 0 SYSTEM:

DESCRIPTION OF CHANGE: This safety evaluation documents the evaluation of the effect of the Division II ESF Switchgear room coolers being out of service during a refueling outage with temporary ventilation provided to maintain temperature below the Technical Specification (TS) limit of 104 degrees F.

REASON FOR CHANGE: To allow for flexibility in outage work that requires the Division II ESF Switchgear room coolers to be removed from service.

SAFETY EVALUATION: This safety evaluation concluded that the change did not involve an unreviewed safety question. The alternate method of room cooling provides air flow rates to maintain room temperatures below 104°F. This evaluation considers that the Div II ECCS pumps and EDG 12 will not be required during this time and that offsite power is available. Since Div I or III ECCS will be available to meet Technical Specification ECCS requirements, the relevant concern is maintaining power distribution available for operation of Div II Primary Containment Isolation Valves. An additional concern is to ensure that a loss of shutdown cooling does not occur due to a failure of equipment in an ESF Switchgear room causing an inadvertent isolation of the Div II SDC isolation valve.

The failure of the alternate cooling equipment does not increase the likelihood of the occurrence of loss of SDC. The maximum temperature calculated to occur with no room cooling available is less than the calculated temperature at which the limiting equipment failure will occur.

UFSAR section 15.A.6.3.3 assumes that LPCI, LPCS, or HPCS can be used with the reactor vessel head off in the event of a loss of SDC. If the reactor vessel head is on and the system can be pressurized, the ADS or manual operation of relief valves in conjunction with any of the ECCS and the RHR suppression pool cooling mode can be used to maintain water level and remove decay heat. ECCS Div I and III will not be affected by Div II ESF switchgear room temperature. Therefore, the consequences of a postulated loss of SDC are not increased.

The alternate room cooling is provided to maintain Div II primary containment isolation valves operable while the redundant Div I isolation valves are also operable. The alternate cooling equipment will be connected to BOP power supply and therefore will have no effect on power supplies important to safety.

The proposed activity does not create the possibility of a malfunction of a different type since only a single train failure would occur as a result of excessive temperatures in the Div II ESF Switchgear room. The flow rate requirements for the alternate room cooling equipment are sufficient to maintain room temperature below the environmental qualification TS limit of 104°F. The possibility of a malfunction due to excessive temperatures is not likely due to the maximum calculated temperatures rise with no room cooling being less than that at which the limiting equipment failure would occur.

There is no reduction in the margin of safety as defined in the basis for any Technical Specification. The Technical Specification limit of 104°F is established by TS 3.7.8 to ensure that the temperature remains below the environmental qualification temperatures of the equipment located in the ESF switchgear rooms. The alternate method of room cooling establishes sufficient air flow to maintain room temperatures below this limit during this period of time when Div II heat loads are minimal.

The 12 hour surveillance requirement of Technical Specification 4.7.8 will be maintained to confirm that the temperature in the ESF Switchgear rooms remains less than the 104°F TS limit.

SRASN: NLS-90-017 DOC NO: TSPS 128 R00 SYSTEM:

DESCRIPTION OF CHANGE: This safety evaluation documents the evaluation of the effect of the Div 1 ESF switchgear room coolers being out of service during a refueling outage (Operational Conditions 4 or 5) with temporary ventilation provided to maintain temperature below the Technical Specification (TS) limit of 104°F.

Previous tests and calculations have shown that equipment in those rooms will remain functionally capable of performing the safety functions at temperatures well in excess of 104°F.

REASON FOR CHANGE: To allow the Div. 1 ESF switchgear room coolers to be out of service during a refueling outage with temporary ventilation provided.

SAFETY EVALUATION: This safety evaluation concluded that the change did not involve an unreviewed safety question. The alternate method of room cooling provides air flow rates to maintain room temperatures below 104°F. This evaluation takes into consideration that the Div 1 ECCS pumps and EDG 11 will not be required during this time and that offsite power is available. Since Div II or III ECCS will be available to meet Tech Spec ECCS requirements, the concern is maintaining power distribution available for operation of Div I primary containment isolation valves. An additional concern is to ensure that a loss of shutdown cooling does not occur due to a failure of equipment in an ESF switchgear room causing an inadvertent isolation of the Div I SDC isolation valve.

Although the alternate method of room cooling is not designed to the same requirements of the ESF switchgear room cooling equipment, the failure of the alternate cooling equipment does not increase the likelihood of the occurrence of loss of SDC. The maximum temperature calculated to occur with no room cooling available is less than the calculated temperature at which the limiting equipment failure will occur.

UFSAR Section 15.A.6.3.3 assumes that LPCI, LPCS, or HPCS can be used with the reactor vessel head off in the event of a loss of SDC. If the reactor vessel head is on and the system can be pressurized, the ADS or manual operation of relief valves in conjunction with any of the ECCS and the RHR suppression pool cooling mode can be used to maintain water level and remove decay heat. ECCS Div II and III will not be affected by Div I ESF switchgear room temperature. Therefore, the consequences of a postulated loss of SDC are not increased. Primary containment isolation system is designed to be single failure proof and the redundant Div II equipment will be operable when the alternate room cooling is provided to the Div I ESF switchgear room.

The alternate room cooling is provided to maintain Div I primary containment isolation valves operable while the redundant Div II isolation valves are also operable. The alternate cooling equipment will be connected to BOP power supply and therefore will have no affect on power supplies important to safety and does not create a possibility for an accident or malfunction of a different type than any evaluated previously in the Safety Analysis Report.

There is no reduction in the margin of safety as defined in the basis for any Technical Specification. The alternate method of room cooling for the Div I ESF switchgear room is intended to maintain the room temperature below the limit of 104°F established by Technical Specification 3.7.8. The Technical Specifications do not address operability requirements of the Div I ESF switchgear room coolers nor specify the method of maintaining room temperature below the 104°F TS limit. Maintaining the temperature limit below 104°F ensures that the environmental qualification limits are not exceeded. Technical Specifications require the temperature in Div I ESF switchgear rooms to be determined to be within the 104°F limit at least once per 12 hours whenever the equipment in the rooms are required to be operable. This surveillance will continue with the room coolers out of service.

SRASN: NLS-90-018 DOC NO: UFSAR CR-NL-90-014 SYSTEM:

DESCRIPTION OF CHANGE: The FSAR change adds a description of present administrative controls which restrict the handling of loads to ensure that in the unlikely event of a load drop into spent fuel, the radiological results are well within the guidelines of 10CFR100.

REASON FOR CHANGE: To provide consistency and clarification in the discussions of controls for hanlding loads over spent fuel.

SAFETY EVALUATION: This safety evaluation concluded that the change did not involve an unreviewed safety question. The described controls on load handling have no adverse affect on the integrity of the handling system. Therefore, the probability of a load drop onto spent fuel is not increased by the existing administrative controls set forth by Technical Specification Position Statement 126 and Plant Administrative Procedure 07-S-05-300. The proposed UFSAR change simply summarizes these controls. The GGNS SER sets the acceptance criteria for the consequences of a fuel handling accident to be "well within the guidelines of 10CFR100" (less than 25 percent of 10CFR100 limits). The described administrative controls limit the potential impact energies (by weight and height restrictions) such that the radiological consequences of a postulated drop onto spent fuel assemblies is within the acceptance criteria (less than 25 percent of 10CFR100 limits).

The load handling systems are not being subjected to a different application than previously used. The administrative controls do not involve any handling equipment not previously considered in UFSAR for the handling of loads over the core or the spent fuel storage areas. The described controls do not subject the equipment to different application than previously used and therefore the margin of safety as defined in the basis for any Technical Specification were not reduced since the described controls place additional conservative restrictions when handling loads over spent fuel assemblies.

SRASN: NLS-90-019 DOC NO: OpCon 4 Entry While in TS 3.5.3 SYSTEM:

DESCRIPTION OF CHANGE: This safety evaluation addresses the use of Technical Specification 3.0.4 to enter Operational Condition 4 from Operational Condition 5 while complying with Action Statement c or d of Technical Specification 3.5.3. When one or more suppression pool level instrumentation divisions are inoperable. The evaluation includes the following considerations:

a. Either or both suppression pool level instrumentation divisions (A or B) may be inoperable.

b. An alternate indicator of suppression pool water level is used at least once per 12 hours to verify suppression pool water level is greater than or equal to 12 feet 8 inches.

c. There are no operations that have a potential for draining the reactor vessel in progress.

d. With no suppression pool level instrumentation OPERABLE, there are no evolutions with the possibility of depleting suppression pool inventory (e.g., suppression pool cleanup) in progress.

REASON FOR CHANGE: During planned refueling outage activities, situations may arise where suppression pool level instrumentation is inoperable in order to perform maintenance activities, surveillance tests, or design change implementations.

SAFETY EVALUATION: This safety evaluation concluded that the change did not involve an unreviewed safety question. TS 3.5.3 requires a suppression pool water level of at least 12 feet 8 inches in Operational Conditions 4 and 5. The suppression pool provides a primary source of water for the ECCS in the event of an accident to provide cooling water for irradiated fuel. The required pool level is sufficient to provide the required heat sink capability and water supply to the ECCS. The OPERABILITY of the suppression pool in Operational Conditions 4 and 5 is not required by TS 3.6.3.1 for pressure suppression.

In Operational Conditions 4 and 5 the suppression pool minimum required water volume is reduced because the reactor coolant is maintained at or below 200°F, the minimum required water volume is based on NPSH, recirculation volume and vortex prevention.

The suppression pool water level instrumentation provides control room visual confirmation of pool level. TS 3.5.3 Actions c and d require the suppression pool level to be verified by an alternate indicator at least once per 12 hours in the event of inoperable suppression pool water level instrumentation.

The UFSAR evaluates several accidents (events) which are considered to be applicable during Operational Conditions 4 and 5. The majority of these are unrelated to the proposed application of TS 3.0.4 for TS 3.5.3 in that their probability of occurrence is unaffected by suppression pool level instrumentation status. Consequently, the probability of occurrence of these events does not increase.

These are:

a. Losses or failures of various plant systems or components (other than g. below).

b. Inadvertent operation of various plant systems or components.

c. Loss of AC power.

d. Inadvertent criticality events.

e. Fuel handling accident.

f. Inadvertent increase in shutdown cooling.

g. Loss of shutdown cooling.

A reactor vessel drain down event or a loss of suppression pool inventory event are not specifically addressed in the UFSAR during OPERATIONAL CONDITIONS 4 and 5. However, both events are a concern during the use of TS 3.0.4 while in Action c or d of TS 3.5.3 during the period of time changing from Operational Condition 5 to 4.

Tensioning of the reactor vessel head closure bolts has no affect on either the reactor vessel or suppression pool water inventories. An alternate indication method will be used every twelve hours. In addition, no planned operations with the potential to drain either the reactor or suppression pool (when no suppression pool level instrumentation is OPERABLE) will be in progress. Therefore, the intent of the TS basis is met even with suppression pool water level instrumentation inoperable.

The use of TS 3.0.4 does not alter the function or operation of either ECCS or suppression pool. Complying with Action c or d of TS 3.5.3 will ensure that suppression pool level is maintained such that the minimum water level based on NPSH, recirculation volume and vortex prevention is maintained.

The bases for TS 3.5.3 discusses the need for suppression pool volume during Operational Conditions 4 and 5 and is based on NPSH recirculation volume and vortex prevention. Complying with TS 3.5.3 Action c or d will ensure that these concerns and the margin preventing these concerns are adequately addressed during tensioning of the reactor vessel head closure bolts. Therefore, the use of TS 3.0.4 will not result in a decrease of any safety margin as defined in the bases of any Technical Specification.

SRASN: NLS-90-020 DOC NO: OpCon 4 Entry While in TS 3.3.6 and Bases SYSTEM:

DESCRIPTION OF CHANGE: Technical Specification 3.3.6 governs the OPERABILITY of the control rod block instrumentation. Two channels of the instrumentation associated with the Reactor Mode Switch shutdown position rod block are required OPERABLE to prevent withdrawal of a control rod in OPCON 4. If one or more of the required channels are inoperable, a rod block must be initiated in accordance with ACTION 63. This safety evaluation documents the analysis of the use of TS 3.0.4 for entry into OPCON 4 from 5 when one or more channels of the Reactor Mode Switch shutdown position trip function are inoperable.

This evaluation considers the following:

a. One or more channels of the Reactor Mode Switch shutdown position trip function are inoperable.

b. A rod block is initiated in accordance with ACTION 63.

REASON FOR CHANGE: During planned refueling outage activities, situations may arise where one channel of the Reactor Mode Switch shutdown position trip function is inoperable in order to perform maintenance or surveillance activities and/or due to divisional bus outages affecting the normal power supply to the associated instrumentation.

SAFETY EVALUATION: This safety evaluation concluded that the change did not involve an unreviewed safety question. The control rod block instrumentation supplies input to the Rod Control and Information System. The function of these inputs is to inhibit control rod movement or selection to prevent reactivity changes in OPCONs 3 and 4. Two channels provide input to this trip function. Technical Specification 3.3.6, ACTION b, establishes requirements through Table 3.3.6-1 for the minimum number of operable channels. If the minimum number of operable channels cannot be met, Action Statement b refers to Table 3.3.6-1, which specifies required ACTION 63 to be taken. ACTION 63 requires a rod block to be initiated thereby positively fulfilling the safety function.

The UFSAR evaluates several accidents (events) which are considered to be applicable during OPERATIONAL CONDITIONS 4 and 5. The majority of these are unrelated to the proposed application of TS 3.0.4 for TS 3.3.6 and thus the probability of occurrence of these events does not increase. The reactivity insertion event is not specifically analyzed in the UFSAR for OPCONs 3 and 4 because the core is assumed to be subcritical. With the control rod block initiated in accordance with ACTION 63, a control rod cannot be inadvertently withdrawn. The core remains subcritical and the assumptions of the accident analyses are preserved.

Technical Specification 3.0.4 presently allows entry into an OPCON or specified condition when LCOs are not met if the plant is in conformance with the LCO Action requirements and those requirements permit continued operation of the facility for an unlimited period of time. This is in accordance with the NRC's stated position and has been accepted for GGNS. Although not specifically addressed in the Bases for TS 3.3.6, compliance with the requirements of ACTION 63 while changing from OPCON 5 to 4 will provide the same level of safety as compliance with the LCO. Therefore, the use of TS 3.0.4 will not result in a decrease of any safety margin as defined in the bases of any Technical Specification.

SRASN: NLS-90-021 DOC NO: OpCon 4 Entry While in TS 3.3.7.5 SYSTEM:

DESCRIPTION OF CHANGE: This safety evaluation documents the analysis of the use of TS 3.0.4 for entry into Operational Condition 4 from Operational Condition 5 while complying with Action 81 of Table 3.3.7.5-1 for the following instruments of Table 3.3.7.5-1:

a. Containment/Drywell Area Radiation Monitors (Item 13)
b. Containment Ventilation Exhaust Radiation Monitor (Item 14)
c. Offgas and Radwaste Building Ventilation Exhaust Radiation Monitor (Item 15)
d. Fuel Handling Area Ventilation Exhaust Radiation Monitor (Item 16)

The evaluation takes into consideration that the preplanned alternate method of monitoring the appropriate parameter(s) is initiated within 72 hours.

REASON FOR CHANGE: During planned refueling outage activities, situations may arise where instruments are inoperable in order to perform maintenance activities, surveillance tests, or design change implementation.

SAFETY EVALUATION: This safety evaluation concluded that the change did not involve an unreviewed safety question. The Accident Monitoring Instruments covered in this evaluation do not perform any automatic functions to mitigate a DBA or transient. These instruments ensure that sufficient information is available during a DBA or transient. Action 81 requirements provide an acceptably safe alternative means of meeting the LCO. First, the Action requires that an alternate preplanned method of monitoring the appropriate parameter be initiated. This ensures that in the event of a DBA or transient, an alternative method of monitoring the parameter is available which will allow assessment of important variables following an accident. Secondly, the Action requires that a special report be prepared and submitted to the NRC outlining the cause of the inoperability and the plans and schedule for restoring operability. This ensures timely attention and resolution of the inoperability as well as an additional review by the NRC of the specific conditions involved in the inoperability.

The requirements of this Specification are applicable in various Operational Conditions dependent upon the instrument. Since the action requirements set up conditions equivalent to those required by the LCO, none of the evolutions involved in changing to Operational Condition 4 from 5 result in any change to the level of safety.

SRASN: NLS-90-022 DOC NO: UFSAR Appendix 13A SYSTEM:

DESCRIPTION OF CHANGE: The revision to Appendix 13A primarily reflects changes to update the Resumes of key personnel associated with the operation of GGNS. Other changes include: addition of a few key positions, deletion of the Technical Assistant to the Operations Superintendent, and removal of the resumes of those positions reporting to the Radiation Control Superintendent. These changes are explained as follows:

- The positions added are: Director, Fuels; Manager, Nuclear Fuels Supply; Manager, Nuclear Fuels Planning.
- The functions of the Technical Assistant to the Operations Superintendent have been assumed by the Operations Assistants per MTO-90/0386.
- The positions currently reporting to the Radiation Control Superintendent include two Radiation Control Supervisors and a Technical Assistant. The Nuclear Plant Safety Coordinator reports to the Technical Assistant. With the exception of certain positions reporting to the Operations Superintendent, Appendix 13A generally does not capture "Supervisors" unless they happen to be on the "Superintendent" reporting level.

REASON FOR CHANGE: The revision of resumes contained in UFSAR Appendix 13A is an administrative change which provides (1) the most current listing of positions supporting the operation of GGNS, (2) names of personnel filling these positions, and (3) the most current status of experience for these individuals.

SAFETY EVALUATION: This safety evaluation concluded that the change did not involve an unreviewed safety question. Individuals assigned to new positions are required to meet the qualification requirements specified in the UFSAR. Analyses in the UFSAR which resume operator error would remain unchanged based on these individuals meeting the UFSAR requirements. No system functions or designs are being changed.

Individuals assigned to new positions are required to meet the qualification requirements specified in the Technical Specifications; therefore, there is no reduction in the margin of safety as defined in the basis for any Technical Specification.

SRASN: NSP-90-003 DOC NO: Change of Executive Director, SYSTEM:
Operations Support to
Vice President, Operations Support

DESCRIPTION OF CHANGE: This evaluation changes the title Executive Director of Operations Support (EDOS) to Vice President, Operations Support (VPOS).

REASON FOR CHANGE: This title change reflects the correct level of management who reports to the Executive Vice President and Chief Operating Officer.

SAFETY EVALUATION: With this title change all the duties, responsibilities and commitments being performed in the existing organizational structure will continue to be performed. The title change will have no effect on plant design or operations; therefore, there will be no increase in probability of occurrence or consequences of accidents previously evaluated in the UFSAR; nor will there be any increase in probability of occurrence or consequences of a malfunction of equipment important to safety previously evaluated in the UFSAR; nor will there be created the possibility of an accident or malfunction of equipment important to safety different than any previously evaluated in the UFSAR.

SRASN: NSP-90-004 DOC NO: Onsite Storage of New Fuel for GGNS Cycle 5 SYSTEM:

DESCRIPTION OF CHANGE: This evaluation is for those activities concerning the new fuel bundles produced for Cycle 5 by Advanced Nuclear Fuels Corporation:

a. The movement of new fuel to either the new fuel vault or the spent fuel rack,

b. The storage of ANF-1.4 fresh reload fuel in the new fuel vault,

c. The storage of ANF-1.4 fresh reload fuel in the spent fuel pool.

REASON FOR CHANGE: The introduction of the new fuel design at GGNS is to improve the fuel cycle economics and increase the operational flexibility of the reactor core.

SAFETY EVALUATION: Confirmatory analyses have been performed to show that the ANF-1.4 reload fuel bundles have weights and geometries similar to those of the GE fuel bundles on which the analyses described in the SAR are based. No new activities are required for the movement of ANF-1.4 fuel bundles to the new fuel vault or the spent fuel pool. Precursors to any accident previously evaluated will not be affected.

Confirmatory analyses have been performed to show that the ANF-1.4 reload fuel is compatible with, and similar to, the reload fuel stored in the new fuel vault during previous reload activities.

The NRC has approved a revision to the licensing basis for storage of the ANF-1.4 reload fuel in the spent fuel storage racks. Because of the similarity of the ANF-1.4 reload fuel to the reload fuel stored in the spent fuel pool during previous reloads, the storage of the new fuel types in the spent fuel storage racks will not affect the precursors to any accident previously evaluated. Therefore, performing the activities in connection with onsite storage of new fuel for Cycle 5 will not increase the probability of occurrence of an accident previously evaluated in the FSAR.

The fuel handling accident is evaluated in the FSAR. The radiological consequences of dropping an unirradiated fuel bundle on the spent fuel racks was evaluated and found to meet the applicable acceptance criteria. This analysis includes fuel parameters applicable to ANF-1.4. The radiological consequences of dropping an unirradiated fuel bundle are determined by the performance of the irradiated fuel in the spent fuel rack. Therefore, the consequences of dropping an ANF-1.4 fuel bundle on the spent fuel racks are unchanged.

Confirmatory analyses have shown that the reactivity of the ANF-1.4 reload fuel in the new fuel vault is within the acceptance criteria established for previous reloads for new fuel. Therefore, as for previous reloads, the occurrence of inadvertent criticality is precluded for ANF-1.4 reload fuel.

The analyses performed in support of the revised basis show that the maximum reactivity of the racks when loaded with ANF-1.4 reload fuel is within the acceptance criteria for the spent fuel pool criticality analysis. The ANF-1.4 reload fuel has a similar static and dynamic response and therefore, the consequences of a seismic event remain unchanged. Therefore, performing the activities in connection with onsite storage of new fuel for Cycle 5 will not increase the consequences of an accident previously evaluated in the FSAR.

The equipment required to be used for the onsite storage and handling of the new fuel bundles is similar to that required to be used for previous reloads; no additional loads will be imposed on any equipment; no increase in frequency of operation of the equipment will result. The precursors to any malfunction of equipment important to safety will not be affected. Therefore, performing the activities in connection with onsite storage of new fuel for Cycle 5 will not increase the probability of a malfunction of equipment important to safety previously evaluated in the FSAR.

The fuel handling and storage equipment will not be subjected to operational conditions different from those during previous reloads; changes to the equipment protection features will not be required. Therefore, performing the activities in connection with onsite storage of new fuel for Cycle 5 will not increase the consequences of a malfunction of equipment important to safety previously evaluated in the FSAR.

The activities associated with the onsite storage and handling of ANF-1.4 reload fuel are unchanged from those associated with the onsite storage and handling of new fuel for previous reloads; no new operational modes will be required; no plant modifications will be required. Therefore, performing the activities in connection with the onsite storage and handling of new fuel for Cycle 5 will not create the possibility of an accident of a different type than any already evaluated in the FSAR.

Based on the operational requirements for the fuel handling equipment, no new equipment is required for the storage or handling of the ANF-1.4 reload fuel. No new fuel handling activities are required in connection with the onsite storage of ANF-1.3 reload fuel; no modifications to the existing equipment are required; no changes in operational setpoints are required. Therefore, performing the activities in connection with the onsite storage and handling of new fuel for Cycle 5 will not create the possibility of malfunction of equipment important to safety different than previously evaluated in the FSAR.

The fuel handling accident has been evaluated. An analysis of the radiological consequences of dropping unirradiated fuel on the spent fuel racks, with and without secondary containment, was performed. This evaluation established height/weight restrictions on the movement of objects above the spent fuel racks which were implemented. These restrictions assure that the radiological consequences of a fuel handling accident for unirradiated fuel meets the acceptance criteria of 25% of 10CFR100 dose rate limits. This evaluation included the fuel design parameters applicable to the ANF-1.4 fuel design. Therefore the margin of safety remains unchanged.

Analyses have been performed to determine the reactivity for the ANF-1.4 reload fuel. The acceptance criterion stated in the FSAR for K-effective in the new fuel vault is 0.95. The corresponding licensing basis value for maximum in-core reactivity is 1.31 (K-infinity), as determined for previous reloads. The analyses described in the FSAR, and which form the bases for the Technical Specification, are based on this value of K-infinity. The maximum in-core reactivity was calculated to be 1.1847 (K-infinity). This value is below the acceptance criterion of 1.31 established for previous reloads.

The NRC has approved a revision to the licensing basis for the storage of the ANF-1.4 reload fuel in the spent fuel pool storage racks. The maximum reactivity for the storage of ANF-1.4 fuel as stated in the NRC safety evaluation is 0.9452 (K-effective). This is below the acceptance criterion of 0.95 (K-effective). The acceptance criterion remains unchanged from that for previous reloads.

The static and dynamic response of ANF-1.4 reload fuel is similar to that for the fuel used in previous cycles. The margin of safety for seismic events is therefore unaffected by the use of ANF-1.4 reload fuel. Therefore, performing the activities in connection with onsite storage of new fuel for Cycle 5 will not result in a reduction in the margin of safety as defined in the basis for any Technical Specification.

SRASN: NSP-90-005 DOC NO: RF04 Fuel Management SYSTEM:

DESCRIPTION OF CHANGE: This evaluation is for the movement of fuel bundles and the shuffling of fuel assemblies in the reactor core.

REASON FOR CHANGE: This fuel movement was to support Refueling Outage Number Four (RF04).

SAFETY EVALUATION: The safety evaluation concluded that the change did not involve an unreviewed safety question. Confirmatory evaluations have shown that the accident analyses described in the UFSAR, which were applicable to previous reloads, continue to remain applicable and bounding for RF04. The precursors to any accident previously evaluated will not be affected.

A confirmatory evaluation has been performed to show that the ANF fuel assemblies have weights, geometries, and seismic response characteristics similar to those of the GE fuel assemblies, on which the analyses described in the UFSAR are based. Because the masses and drop heights are essentially the same, the momentum and kinetic energy effects of dropping an ANF fuel assembly are similar to those for previous reload fuel types. A bounding evaluation has shown that the dose rates resulting from the drop of an ANF fuel assembly are within the dose rates acceptance criterion stated in the GGNS-1 Safety Evaluation Report. Using the same analysis assumptions for the GE and ANF fuel types, it has been shown that the radiological consequences resulting from the drop of an ANF fuel assembly are bounded by the consequences that would result from the drop of a GE fuel assembly. The precursors to any accident previously evaluated will not be affected.

Calculations have been performed to show that adequate shutdown margin exists during fuel shuffling. Restrictions applicable to fuel shuffle activities have been provided to GGNS-Reactor Engineering for inclusion in the appropriate procedures in a manner similar to previous reloads. The precursors to any accident previously evaluated will not be affected.

The equipment required to be used during RF04 is similar to that used for previous reloads; no additional loads will be imposed on any equipment as a result of handling the ANF fuel assemblies; no increase in frequency of operation of the equipment will result; no new operational modes are required; no plant modifications are required; no changes in operational setpoints are required. The precursors to any malfunction of equipment important to safety will not be affected. The consequences of a malfunction of equipment important to safety are bounded by the consequences evaluated in the UFSAR.

Accident analyses applicable to previous reloads during operational modes 4, 5, and * either continue to remain applicable for RF04, or are cycle-specific. The acceptance criteria applicable to the accidents analyzed for previous reloads continue to be satisfied. Accident analyses/evaluations have been performed to show that cycle-specific events meet the acceptance criteria. A bounding evaluation, using conservative assumptions, has shown that the radiological consequences of dropping an ANF fuel assembly continue to satisfy the Safety Evaluation Report acceptance criterion (25% of 10CFR100 limits); the dropping of a GE fuel assembly, on which the UFSAR analyses are based, continues to be the limiting event for determining the radiological consequences of the Fuel Handling Accident. The Shutdown Margin determined by the analyses is within the acceptance criterion (1%) established in the UFSAR. Therefore, performing the activities in connection with refueling activities during RF04 for Cycle 5 fuel will not result in a reduction in the margin of safety as defined in the basis for any Technical Specification.

SRASN: NSP-90-006 DOC NO: Refueling Operations with Revised Core Loading Plan SYSTEM:

DESCRIPTION OF CHANGE: The refueling operations in Modes 4, 5 and * were previously evaluated assuming fuel bundle XNB-487 would remain in the core for Cycle 5 operation. This safety evaluation addresses refueling operations with fuel bundle XNB-529 replacing XNB-487 in its beginning of cycle (BOC) location (21,58) for the following proposed activities:

1. The movement of fuel bundles, and

2. The shuffling of fuel assemblies in the reactor core.

REASON FOR CHANGE: During the course of loading the GGNS-1 Cycle 5 core, fuel bundle XNB-487 was dropped from slightly above the core into its designated location (21,58). This bundle was initially inserted in the core during Cycle 3 and reinserted in Cycle 4. It has similar reactivity characteristics to fuel planned for discharge during RF04. Therefore, replacing the bundle with a bundle planned for discharge was considered more practical than requalifying the dropped bundle for an additional cycle of operation. Fuel bundle XNB-529 was identified as the appropriate replacement bundle. This bundle has similar reactivity performance to XNB-487. Both fuel bundles have the same nuclear design but the replacement bundle (XNB-529) has a slightly higher burnup and therefore slightly lower reactivity.

SAFETY EVALUATION: The safety evaluation concluded that the change did not involve an unreviewed safety question. A confirmatory evaluation has been performed to show that the ANF fuel assemblies have weights, geometries, and seismic response characteristics similar to those of the GE fuel assemblies, on which the analyses described in the UFSAR are based. Because the masses and drop heights are essentially the same, the momentum and kinetic energy effects of dropping an ANF fuel assembly are similar to those for previous reload fuel types. A bounding evaluation has shown that the dose rates resulting from the drop of an ANF fuel assembly are within the dose rates acceptance criterion stated in the GGNS-1 Safety Evaluation Report. Using the same analysis assumptions for the GE and ANF fuel types, it has been shown that the radiological consequences resulting from the drop of an ANF fuel assembly are bounded by the consequences that would result from the drop of a GE fuel assembly. The change in the Cycle 5 core loading plan only replaces one ANF fuel bundle with a similar bundle of the same design. The precursors to any accident previously evaluated will not be affected.

Calculations have been performed to show that adequate shutdown margin exists during fuel shuffling. These calculations bound the revised core configuration. Restrictions applicable to fuel shuffle activities have been provided to GGNS-Reactor Engineering for inclusion in the appropriate procedures in a manner similar to previous reloads. The precursors to any accident previously evaluated will not be affected.

The fuel that will be handled during RF04 is similar to, and compatible with, the fuel that was handled for previous reloads. The equipment required to be used during RF04 is similar to that used for previous reloads; no additional loads will be imposed on any equipment as a result of handling the ANF fuel assemblies; no increase in frequency of operation of the equipment will result. The refueling activities associated with Cycle 5 fuel will not subject the equipment to operational conditions different from those during previous reloads; changes to the equipment protection features will not be required.

A bounding evaluation, using conservative assumptions, has shown that the radiological consequences of dropping an ANF fuel assembly continue to satisfy the Safety Evaluation Report acceptance criterion (25% of 10CFR100 limits).

The Shutdown Margin (SDM) determined by the analyses is within the acceptance criterion (1%) established in the UFSAR. The revised core configuration has the same or slightly less SDM and therefore the previous analysis results remains applicable. Therefore, performing the activities in connection with refueling activities during RF04 for Cycle 5 fuel will not result in a reduction in the margin of safety as defined in the basis for any Technical Specifications.

SRASN: NSP-90-007 DOC NO: UFSAR 15.5.1 SYSTEM:

DESCRIPTION OF CHANGE: A revision of the UFSAR was made to address the inadvertent startup of the High Pressure Core Spray (HPCS) system. The revision describes two alternative event sequences that could result from inadvertent HPCS startup.

REASON FOR CHANGE: To describe in the UFSAR an event sequence that was observed during the inadvertent HPCS actuation that occurred on October 10, 1988. The reactor level control system was unable to compensate for the level increase resulting from HPCS injection. In spite of operator actions to mitigate the level increase, the reactor vessel level increased, resulting in a trip signal in one of two reactor protection system instrumentation channels.

SAFETY EVALUATION: The safety evaluation concluded that the change did not involve an unreviewed safety question. Of the two alternative event sequences resulting from the inadvertent HPCS actuation (IHA), the sequence that results in a new equilibrium power level has been analyzed previously and is described in the FSAR. The sequence leading to the high level trip is bounded by the Feedwater Controller Failure - Maximum Demand transient. The two alternative event sequences do not require the use of any new equipment or the use of existing equipment in any new functional capacity. No changes to plant operational modes are required. No plant modifications are required.

The delta-Critical Power Ratio (CPR) for the event sequence resulting in a new equilibrium power level has been analyzed previously. The delta-CPR for the event sequence resulting in the high level trip is bounded by the delta-CPR for the Feedwater Controller Failure - Maximum Demand transient; this transient has been analyzed on a cycle-specific basis as one of the limiting transients that causes increase in reactor vessel inventory and decrease in reactor coolant temperature. Consequently, there is no reduction in the margin of safety as defined in the basis for any Technical Specification.

SRASN: NSP-90-008 DOC NO: Cycle 5 OPS With Revised Core Configuration SYSTEM:

DESCRIPTION OF CHANGE: This safety evaluation was written to demonstrate the acceptability of Cycle 5 operation with a revised core configuration.

REASON FOR CHANGE: The new configuration was necessitated by the replacement of an Advanced Nuclear Fuels (ANF) 8x8 fuel assembly (XNB-487) with a similar, less reactive ANF 8x8 fuel assembly (XNB-529) in core location (21,58). The old assembly was dropped during refueling.

SAFETY EVALUATION: The safety evaluation concluded that the change did not involve an unreviewed safety question. The replacement fuel assembly is of a design similar to the assembly that was to be present in the NRC-approved Cycle 5 core configuration. The supporting analyses for the NRC-approved Cycle 5 core configuration continue to remain applicable for the revised configuration.

The replacement fuel assembly is less reactive and has been placed in a non-limiting core location. The postulated accidents for the revised Cycle 5 core configuration have been shown to be no more severe than the postulated accidents for the NRC-approved Cycle 5 core configuration. Because the replacement fuel assembly is similar to, and compatible with, the fuel assembly it has replaced, no new equipment will be required; no new activities are required; no modifications to the existing equipment are required; no changes in operation setpoints are required.

An evaluation of the impact of the revised core configuration on the fuel mechanical design limits, plant transients, and postulated accidents has shown that the supporting analyses that were performed for the NRC-approved Cycle 5 core configuration remain applicable for the revised Cycle 5 core configuration. The analytically determined limits applicable to the NRC-approved Cycle 5 core configuration continue to be applicable to the revised Cycle 5 core configuration; the available margins to their respective acceptance limits are unaffected.

SRASN: NSP-90-009 DOC NO: Cycle 5 OPS With 9X9.5 Reload SYSTEM:

DESCRIPTION OF CHANGE: This safety evaluation addresses those issues associated with Cycle 5 operation with ANF 9x9-5 fuel assemblies that have not already been evaluated under other 50.59 safety evaluations or in the Cycle 5 reload PCOL. Items evaluated included:

1) A confirmatory analysis to verify that the baseline analyses continue to remain applicable to the ANF 8x8 core from the standpoint of energy releases to the containment.

2) An analysis comparing the energy release from a ANF 8x8 fuel assembly with that of and ANF 9x9-5 fuel assembly.

3) An analysis to confirm adequate recombiner capacity for cycle 5.

4) A Fire Scenario Evaluation for 9x9-5 Reload Fuel.

5) An analysis to ensure compliance with the Anticipated Transients Without Scram (ATWS) rule. The baseline analysis, which assume a GE 8x8 fueled core, were reevaluated for applicability to the ANF fuel types.

6) The Emergency Procedures were reviewed to ensure no changes to the fuel related inputs to the supporting analyses for the Emergency Procedures were necessary.

REASON FOR CHANGE: To assess all other fuel dependent issues for Cycle 5 operation not previously addressed.

SAFETY EVALUATION: There is no increase in the probability of occurrence or in the consequences of an accident or malfunction of equipment important to safety previously evaluated in the Safety Analysis Report, because:

a) The events that could result in a design basis LOCA (DBLOCA) are based on certain a prior assumptions. They are independent of fuel stored energies.

b) The events that could result in a DBLOCA are based on certain a prior assumptions. They are independent of active clad volume.

c) The events leading to a major fire that could affect safe shutdown capability are a function of plant operational conditions. They are independent of the fuel types resident in the core.

d) The events leading to an ATWS are determined by the response of the reactor shutdown systems to abnormal plant conditions. They are independent of the fuel types resident in the core.

e) The stored energies in the fuel assemblies, which are the only significant fuel-dependent parameters used in determining containment response to a DBLOCA, have been compared for the GE and ANF fuel types. The comparison has shown that the maximum stored energy in the ABF 9x9-5 fuel assembly is bounded by that in the ANF 8x8 fuel assembly; the difference in the maximum stored energy between the ANF and GE 8x8 fuel assembly is insignificant. Furthermore, the fuel stored energy is a small part of the total energy released to the containment. The parameters used to determine containment response during the DBLOCA are unchanged.

f) The active clad volume that was used in sizing the hydrogen recombiners bounds the active clad volume that will be present in the Cycle 5 core.

g) The peak clad temperatures (PCTs) during a major fire have been shown to be well below the temperature of incipient clad deformation for all ANF fuel types that will be present in the Cycle 5 core.

h) The ANF fuel designs are compatible with the GE fuel design, on which the FSAR analyses are based. The core-wide response to an ATWS event resulting from the insertion of Cycle 5 fuel has been determined to be no more severe than that for previous cycles.

The postulated accidents for Cycle 5 have been shown to be no more severe than the postulated accidents for previous cycles. There is no creation of a possibility for an accident or malfunction of a different type than any evaluated previously in the Safety Analysis Report.

The Cycle 5 fuel is similar to and compatible with the fuel inserted into the core during previous reloads. The design of the Cycle 5 fuel does not require any activities different from those associated with previous cycles; no new operational modes are required; no plant modifications are required. Additionally no new equipment will be required; no new activities are required; no modifications to the existing equipment are required; no changes in operational setpoints are required.

a) The fuel stored energy constitutes a small part (6.8%) of the total energy released to the containment during a DBLOCA. The impact of changes in the stored energy (0.38% higher for ANF 8x8 fuel, compared to GE 8x8 fuel and 12.2% lower for ANF 9x9-5 fuel, compared to ANF 8x8 fuel) results in a decrease in stored energy for the Cycle 5, as compared to the GE 8x8 core.

b) The active clad volume for the Cycle 5 core (2693 cubic inches) is less than that used to size the hydrogen recombiners (2696) cubic inches). The design basis criter , for sizing the , drogen recombiners continues to be satisfied for Cycle 5.

c) The PCT during the major fire event for GE fuel (700 degrees F) provides for a margin of 1190 - 700 = 490 degrees F to incipient cladding deformation. The corresponding margins for ANF 8x8 and 9x9-5 fuels are 1500 - 870 = 630 degrees F and 1500 - 801 = 699 degrees F, respectively. The available margin for ANF 9x9-5 fuel is greater than that for ANF 8x8 fuel; both ANF fuel types have increased margin to incipient clad deformation than GE fuel.

d) The core average response and vessel pressurization effects for the Cycle 5 core during an ATWS have been determined to be no more severe than those for previous cycles because the ANF and GE fuel designs are similar. The actions required to mitigate the effects of the limiting ATWS event for Cycle 5 are unchanged; the ability to maintain critical plant parameters within the limits established previously is unchanged.

The acceptance criteria applicable to previous cycles continue to be adequately satisfied for the issues described.

Therefore, by implementing or performing the actions described, a reduction in the margin of safety as defined in the basis for any technical specifications will not result.