Entergy

Entergy Nuclear South
Entergy Operations, Inc.
17265 River Road
Killona, LA 70066
Tel 504 739 6475
Fax 504 739 6698
aharris@entergy.com

Alan J. Harris
Director, Nuclear Safety Assurance
Waterford 3

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November 29, 2001

U.S. Nuclear Regulatory Commission
ATTN: Document Control Desk
Washington, D.C. 20555

Subject: Waterford 3 SES
Docket No. 50-382
License No. NPF-38
Report of Facility Changes, Tests and Experiments

Gentlemen:

Enclosed is the Report of Facility Changes, Tests and Experiments for Waterford 3, which is submitted pursuant to 10CFR50.59. This report covers the period from June 1, 2000 through May 31, 2001. This letter does not contain commitments.

If you have any questions regarding this report, please contact Lisa Borel at (504) 739-6403.

Very truly yours,

Alan J Harris

A.J. Harris
Director
Nuclear Safety Assurance

AJH/LBB/cbh
Enclosure: 50.59 Summary Report

cc: E.W. Merschoff, NRC Region IV
N. Kalyanam, NRC-NRR
J. Smith
N.S. Reynolds
NRC Resident Inspectors Office

IE47
Rec'd at NRC/DCD
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WATERFORD 3
10CFR50.59 REPORT
ENTERGY OPERATIONS, INC.

JUNE 1, 2000 THROUGH MAY 31, 2001

TABLE OF CONTENTS

ENTERGY OPERATIONS, INC.
WATERFORD 3 SES
DOCKET NO. 50-382
LICENSE NO. NPF-38

REPORT OF FACILITY CHANGES, TESTS, AND EXPERIMENTS

PER 10CFR50.59

JUNE 1, 2000 THROUGH MAY 31, 2001

SUMMARY

This report provides the Waterford 3 Changes made pursuant to 10CFR50.59(c)(1) for the period from June 1, 2000, through May 31, 2001.

Section I identifies acronyms used in the report.

Section II of the report identifies 70 facility changes.

Section III of the report identifies 5 procedure changes.

Section IV of the report identifies 15 commitment changes.

I. LIST OF ACRONYMS

ACRONYM	DEFINITION
ASGT	Asymmetric Steam Generator Transient
ASME	American Society of Mechanical Engineers
BMS	Boron Management System
BRE	Bullet Resistant Enclosure
BRTGM	Broad Range Toxic Gas Monitoring
CAA	Controlled Access Area
CBC	Critical Boron Concentration
CBO	Controlled Bleed Off
CCW	Component Cooling Water
CE	Combustion Engineering
CFC	Containment Fan Cooler
CFM	Cubic Feet per Minute
COLR	Core Operating Limits Report
COLSS	Core Operating Limits Supervisory System
CPC	Core Protection Calculator
CSP	Condensate Storage Pool
DEQ	Department of Environmental Quality
EDG	Emergency Diesel Generator
EFAS	Emergency Feedwater Actuation Signal
EFW	Emergency Feedwater
EFWPT	Emergency Feedwater Pump Turbine
EPA	Environment Protection Agency
FSAR	Final Safety Analysis Report
GPM	Gallons per Minute
HVAC	Heating, Ventilation and Air Conditioning
LCO	Limiting Condition for Operation
LLRT	Local Leak Rate Test
LLRWF	Low Level Radioactive Waste Facility
LOCA	Loss of Coolant Accident
LWMS	Liquid Waste Management System

ACRONYM	DEFINITION
MCC	Motor Control Center
MFIV	Main Feedwater Isolation Valve
MSIS	Main Steam Isolation Signal
MSIV	Main Steam Isolation Valve
MSL	Mean Sea Level
NPSH	Net Positive Suction Head
NRC	Nuclear Regulatory Commission
NSSS	Nuclear Steam Supply System
PC	Protective Clothing
PDP	Power Distribution Panel
PIG	Particulate, Iodine, Gas
PPB	Parts per Billion
PPM	Parts per Million
PWTP	Primary Water Treatment Plant
QA	Quality Assurance
RCB	Reactor Containment Building
RCS	Reactor Coolant System
RG	Regulatory Guide
RTD	Resistance Temperature Detector
RWSP	Refueling Water Storage Pool
SBV	Shield Building Ventilation
SCFM	Standard cubic feet per minute
SGCC	Steam Generator Chemical Cleaning
SGTR	Steam Generator Tube Rupture
SIAS	Safety Injection Actuation Signal
STS	Static Transfer Switch
SUPS	Static Uninterruptible Power Supply
TRH	Temporary Reactor Head
TRM	Technical Requirements Manual
VCT	Volume Control Tank

II. FACILITY CHANGES

A. DESIGN CHANGES

1. 1998-075; DC-3555, Station and Instrument Air Compressor Unloader Valve Replacement

DESCRIPTION OF CHANGE

Replace the loader/unloader valves on each of the Instrument Air and Station Air compressors with more reliable equipment.

REASON FOR CHANGE

The solenoid operated loader/unloader valves used on the Instrument Air and Station Air compressor skids are failing at a rate of 3 to 4 failures per year. The existing compressor skids use a solenoid operated diaphragm valve to load and unload the compressors. Internal to the valve is an orifice that makes use of the process air to change the position of the diaphragm. The predominant failure mechanism is plugging of the internal orifice resulting in the inability of the diaphragm to change position. The air that passes through the valve has high moisture content. As the valve cycles, it heats up and cools down resulting in the valve internals experiencing wet and dry cycles, causing a buildup of scale that eventually plugs the internal orifice. These valves were supplied with the original compressor package and were the standard design at the time. The equipment manufacturer has since changed the standard design and now uses external air pilot solenoid operated valves. The new design uses a 3-way solenoid which actuates an air piston operated angle seat valve. Clean air (taken from downstream of the system dryers) is supplied to the solenoid, which in turn provides air to the piston operated angle valves .

50.59 EVALUATION

The equipment being replaced is non-safety, non-seismic, commercial grade material. The Instrument Air and Station Air compressors serve no safety function and are not required for safe shutdown of the plant or for limiting radiological releases. There are no accidents that credit the Instrument Air or Station Air compressors. The Instrument Air system is designed to maintain a minimum air pressure in air accumulators throughout the plant that serve safety related valves. The modification to change the load/unload valve on the compressor itself does not impact the safety related accumulators or their pressure boundaries. The modification does not increase the probability of occurrence of a malfunction of the safety-related valves or accumulators. The equipment being supplied meets all of the original design specifications for the commercial grade compressor skids. Replacement of the air compressor loader/unloader valves will improve the compressor reliability and reduce the likelihood of malfunction. The replacement valves are functionally equivalent and provide a more reliable valve configuration.

2. 1999-037; DC-8020; Broad Range Gas Monitoring System Replacement

DESCRIPTION OF CHANGE

This modification replaces the main control room broad range toxic gas monitors (BRTGM), and removes the broad range toxic gas chromatograph and the broad range toxic gas recorders from the main control room. The BRTGMs isolate the main control room when potentially toxic gases are detected in the control room intake duct. These monitors assure adequate toxic gas protection for the control room operators.

The toxic chemical analysis was reperformed for the new BRTGMs using the same method as the existing analysis, with the exceptions that the new analysis included chemicals that were previously excluded because they were primarily fire hazards and the new analysis uses updated Immediately Dangerous to Life and Health values. The new units perform an automatic self-calibration or background check with a maximum of two minutes lapse in time per hour during the monitor background check. During the background check, the unit renders itself inoperable. The self-calibration feature is consistent with the surveillance requirements specified in the new Technical Specification 4.3.3.7.3. The Technical Specification change allows the monitors to perform self-calibrations without entering a Technical Specification Action statement and changes the surveillance requirements.

REASON FOR CHANGE

The existing BRTGM has demonstrated problems with reliability. It is necessary to perform calibration testing on a weekly basis and there is a reduced confidence that the devices will be found in calibration. The replacement units, employing different detection principles, have greater sensitivity, higher stability, and provide more timely information than the existing units. The gas chromatograph analyzer is an existing accessory, which is required to determine the nature of and the trend of the concentration of the incoming potentially harmful gases after an alarm/isolation. The gas chromatograph will be eliminated, since the replacement BRTGMs are capable of performing the functions of identification of the gas, and real-time display of its concentration, in addition to providing alarms. The toxic chemical analysis uses BRTGM detection capabilities as input. Since the BRTGMs are changed by this activity, the toxic chemical analysis was reperformed.

50.59 EVALUATION

This change does not affect the design bases of the control room habitability system. This activity is bounded by existing safety analyses with regard to its impact on accident mitigating systems and its ability to mitigate the effects of a toxic gas release as described in the FSAR. Technical Specification 3/4.3.3.7 was previously revised under a License Amendment to accommodate the differences in the BRTGMs. New impacts from the interfacing Instrument Air System have been analyzed and the results approved with the changes to the Technical Specifications. The significant hazard review concluded that this activity is bounded by risks already considered. The new toxic chemical analysis determined that the new BRTGMs would reduce the control room hazard probability associated with toxic chemical releases. This activity is being installed consistent with the new Technical Specification as reflected in the Amendment.

B. MISCELLANEOUS EVALUATIONS

1. 2000-033; Construction Activities Involving Setup of the Waste Processing Facility and Equipment Testing Required to Support the Steam Generator Chemical Cleaning (SGCC) Project

DESCRIPTION OF CHANGE

This evaluation addresses the impact of the temporary equipment and construction activities of the Steam Generator Chemical Cleaning Waste Processing Facility. The activities addressed by this evaluation consist of: 1) radioactive material receipt inspection and storage of sealed containers of radioactive material in the Owner-Controlled Area; 2) opening of sealed containers with radioactive material to support system assembly; and, 3) adequacy of spill control procedures and initial equipment/system checkouts that may involve transporting radioactive materials in fluids that could result in uncontrolled/unmonitored release of radioactive materials to unrestricted areas. The waste processing facility will be located outside of the protected area but inside the owner-controlled area. All equipment that may contain radioactive liquids will be stored within a bermed area that will be lined with a chemical resistant liner that can withstand local weather conditions and is compatible with the chemicals being utilized. The bermed area is designed to contain the potential rupture of the largest possible tank or 10% of the aggregate volume of all waste tanks, whichever is greater, plus 24 hours of the 25-yr. Rainfall.

REASON FOR CHANGE

A facility is required to process waste generated during the Steam Generator Chemical Cleaning Process.

50.59 EVALUATION

The construction activities involved in the Steam Generator Chemical Cleaning Waste Processing Facility will be performed in compliance with all applicable regulatory requirements of Title 10, Parts 20, 40, 50 and 100; Title 40 Part 190; and plant procedures. These regulatory requirements and plant procedures will establish radiological and environmental controls during the construction of the temporary Waste Processing Facility to protect the general public, prevent uncontrolled or unmonitored release of radioactive materials to unrestricted areas, and assure adequate public health and safety. None of the accidents previously considered in the FSAR become more probable due to the construction of the waste processing facility. The consequences of accidents previously evaluated are not increased. Previously evaluated accidents include radioactive waste gas system leak or failure, liquid waste system leak or failure and postulated radioactive releases due to liquid containing tank failures. The analyses performed for all of these accidents would bound any potentially radioactive spill during the temporary equipment testing phase. The Bases for Technical Specifications 3/4-11.1.4 refers to restricting the quantity of radioactive material contained in the outdoor radwaste tanks that are not surrounded by liners, dikes or walls capable of holding the tank contents. The tanks used in the Waste Processing Facility will be within a berm. Samples will be collected and analyzed to verify the quantity of radioactive material to ensure the Technical Specification limit is not exceeded.

2. 2000-045; Calculation EC-S98-098 Rev. 2, Radiological Doses Following a Fuel Handling Accident

DESCRIPTION OF CHANGE

Calculation EC-S98-098, Radiological Doses Following a Fuel Handling Accident, was revised as follows: 1) the source terms used as input to the TRANSACT code used for dose analysis were modified to account for a power peaking factor of 1.8; 2) the assumed unfiltered inleakage to the control room envelope was increased from 3 CFM to 13 CFM; 3)the efficiency of the Fuel Handling Building filtration unit was changed from 99% to 97.02%; 4) additional time steps were used in the plant model files; 5) a case assuming the failure of all fuel rods in one fuel assembly (236 rods) was added, and 6) editorial changes and clarifications as well as minor reformatting was done.

REASON FOR CHANGE

Refer to the item numbers in the description section above: 1) CR-WF3-1999-1225 identified that power peaking had not been accounted for in EC-S98-008 Revision 1. Corrective Action CA-005 required the inclusion of the peaking factor in the calculation; 2) EC-S98-008 Revision 1 did not account for 10 CFM postulated air inleakage through the normal air intake valves to the control room envelope. This amount of leakage is assumed in other dose analyses; 3) EC-S98-008 Revision 1 did not account for airflow from the fuel handling building through the train A/B cross-connect line. Air passing through this bypass flow path is unfiltered when one train of filtration is not in service. In this case, the unfiltered flow is drawn in with the filtered flow from the operating train thus reducing the effective efficiency of filtration; 4) additional time steps were added to the plant model files to address a condition identified in Condition Report CR-WF3-2000-0542 regarding the sensitivity of the total calculated dose to the size of the time intervals used in TRANSACT; 5) the case of 236 failed fuel rods was added for information only to provide a basis of comparison with the NRC's evaluation of a fuel handling accident at Waterford 3 which assumed the failure of all fuel rods in one fuel assembly; 6) editorial and format changes were done to enhance the readability and clarity of the document.

50.59 EVALUATION

Revision 2 of calculation EC-S98-008 documents the offsite and control room radiological dose consequences resulting from a postulated fuel handling accident in the fuel handling building. All calculated doses for a design basis fuel handling accident in the fuel handling building are significantly below NRC acceptance limits. Plant response to the postulated accident is not affected by this calculation and the margin of safety is not reduced. No unreviewed safety questions exist.

3. 2000-046; Technical Specification Basis Change Request 00-001, 25% Surveillance Interval Extension

DESCRIPTION OF CHANGE

Change Technical Specification Surveillance Requirement 4.0.2 Basis to incorporate NUREG-1432 discussions to clarify the acceptability of utilizing Technical Specification Surveillance Requirement 4.0.2 (25% surveillance interval extension) for surveillance requirements located in Technical Specification actions. The change clarifies that the 25% extension is also applicable to the surveillance requirements required in Technical Specification actions following their initial performance.

REASON FOR CHANGE

Clarify the acceptability (Technical Specifications are currently not clear) for utilizing Technical Specification Surveillance Requirement 4.0.2 for surveillance requirements located in Technical Specification actions.

50.59 EVALUATION

The proposed change does not increase the probability or consequences of an accident, does not increase the probability or consequences of a malfunction of equipment, does not increase the probability of a new or different accident or malfunction of equipment, and does not reduce the margin of safety as defined in the basis of the Technical Specifications. Specifically, this change is not an initiator of any accident, does not affect the mitigation capabilities of any equipment, and does not affect the operation of the plant or any equipment. The consequences of an event occurring during the allowable extension time are the same as the consequences of an event occurring within the normal surveillance interval. This change will have no impact on the margin of safety because the most probable result of the performance of a surveillance requirement is its successful completion.

4. 2000-051; Steam Generator Chemical Cleaning - Implementation

DESCRIPTION OF CHANGE

This evaluation is for the implementation portion of the Steam Generator Chemical Cleaning (SGCC). It addresses the effect of the cleaning process on Technical Specifications and FSAR Chapter 15 accident analyses. The effect on the safety related HVAC systems that contain charcoal is also addressed. The SGCC process is conducted in two major parts, iron removal followed by copper/lead removal.

REASON FOR CHANGE

The purpose of the steam generator chemical cleaning is to remove the estimated 8100 lb. of deposits in each steam generator to mitigate potential tube corrosion and restore steam generator thermal performance. The deposits primarily consist of magnetite with approximately five to ten percent copper, approximately one to two percent nickel and trace amounts of other metal oxides, including lead.

50.59 EVALUATION

The plant conditions that will be maintained during this evolution are Reactor Coolant System (RCS) temperature maintained at a maximum of approximately 340°F, RCS pressure in accordance with the Reactor Coolant Pump operating curves, and steam generator pressure and temperature at saturation conditions. Comparison of these plant conditions to the conditions assumed in the FSAR analyses demonstrates that the FSAR analyses represent worst case scenarios and clearly bound the steam generator chemical cleaning evolution. No aspect of the chemical cleaning activity increases the probability or consequences of an accident that was previously analyzed in the FSAR. Corrosion of the steam generators and affected plant systems materials will be within the design corrosion allowances. Foreign material control will be in accordance with existing plant administrative control procedures. The temporary equipment will be operated and discharges sampled until the equipment is proven to be clear of any foreign material. This ensures that no foreign material or loose parts, which could conceivably cause a tube failure during plant operation, are inadvertently injected into the steam generator or other plant systems. The steam generator chemical cleaning will not require any plant structures, systems or components to operate outside of their design bases. During the Steam Generator Chemical Cleaning process the main steam line will also carry some small amounts of the chemical cleaning chemicals, ammonia and hydrazine. These chemicals could have the potential to increase the probability of occurrence of a malfunction of equipment previously evaluated in the FSAR if the new chemicals could cause unanalyzed damage to safety related components that they may come in contact with. The Environmental Qualification program has qualified all safety-related equipment for a pH range of approximately 4.5 to approximately 10.5. The pH range for the chemicals used in the Steam Generator Chemical Cleaning process is approximately 7.6 to approximately 10.0. Equipment that could be exposed to the chemicals, due to a piping rupture will therefore suffer no consequences that are not bounded by the existing analyses in the FSAR. An accident of a different type than those evaluated in the FSAR will not be created. It is concluded that neither the actual chemical cleaning, or the temporary changes associated with the chemical cleaning will degrade the integrity or performance of the steam generators, the connected instrumentation, or the affected systems. All physical changes are temporary and there are no new permanent system interactions created.

5. 2000-052; Steam Generator Chemical Cleaning Process

DESCRIPTION OF CHANGE

Secondary side steam generator chemical cleaning is performed to reduce the potential for secondary side corrosion of the tubes and to maintain the designed thermal hydraulic performance of the steam generators. This evaluation covers the application process of the chemical cleaning within the physical bounds of the steam generators and addresses the following issues: 1) chemical process (affect on Steam Generator (SG) internals and vapor space corrosion); 2) erosion-corrosion effects; 3) chemical process laboratory test results; 4) process corrosion monitoring; 5) faulted tube testing - Inconel 600 corrosion; 6) vacuum affects on Steam Generator; 7) chemical solvent impingement on Steam Generator materials; 8) the effect of air exposure on the corrosion of Steam Generator materials; 9) effect of residual solvent chemicals and decomposition products on steam generator materials and plant operation and 10) flow induced vibration concerns.

REASON FOR CHANGE

The objective of chemical cleaning the steam generators is to remove secondary side deposits from the tubes, tube support structures, the interfaces between the tube support structures and tubes, and from the tube sheet. These deposits can become initiation sites for intergranular attack / stress corrosion cracking. Tube degradation due to such corrosion reduces the margin of tube structure integrity and may require tube plugging or sleeving to prevent the rupture of a tube and possible radiological release. The Steam Generator tubing provides a barrier to prevent fission products and activated corrosion products from entering the secondary steam system and the environment. The chemical cleaning is also designed to remove the deposits that impede primary to secondary heat transfer and degrade the designed thermal performance of the steam generators. The chemical cleaning will reduce upper tube bundle deposits that can cause flow disruption through the bundle and redirect it to the peripherals where erosion of tube supports can occur. This erosion of the tube supports could eventually compromise tube integrity.

50.59 EVALUATION

The proposed change does not cause the Steam Generator portion of the Main Steam System to be operated outside of its design or test limits nor result in any challenges to the Reactor Coolant System (RCS) pressure and fission product boundary or the structures that support the tube material. Although the proposed activity does not make any permanent changes to the plant's operating configuration, it introduces chemicals not normally used into the steam generators. As such, it directly affects the steam generator internals. The effect of exposure to the chemical cleaning solvents and their vapors is the potential degradation of the steam generator internal materials due to accelerated corrosion. The only credible accident previously evaluated in the FSAR, whose probability of occurrence may be affected by the proposed activity is the Steam Generator Tube Rupture (SGTR). To estimate the impact of the chemical cleaning solvent induced corrosion on the steam generator materials of construction extensive corrosion testing was performed. The testing and subsequent evaluation of steam generator materials and reviews of historical material corrosion data conclude that the cleaning solvents and the process sequence will not compromise the integrity or exceed the corrosion allowances of the steam generator internal materials. The integrity of the barrier between the Reactor Coolant System and the Main Steam System is radiologically significant, since a leaking steam generator tube allows transport of the reactor coolant into the Main Steam System. In reviewing the consequences as outlined in the FSAR, the initial conditions and parameters used to determine core and system performance are different than the conditions and parameters placed upon the core and systems while implementing the chemical cleaning. The initial conditions at the time of the event assume the plant is at 100% power operation. The chemical cleaning iron steps will be performed in Mode 4 (Hot Shutdown) and the copper/lead removal step will be performed in Mode 5 (Cold Shutdown). The RCS temperature during the application is well below the design Mode 1 accident analysis and with the reactor shutdown, there is little probability that the steam generator safety valves will automatically lift or the operator would be required to operate the atmospheric dump valves to control RCS temperature. With little probability of these occurring, any potential radiological release will be minimized. The FSAR evaluates the radiological consequences of a Steam Generator Tube Rupture. It considers the most severe release of secondary radioactivity as well as primary system activity leaked from a tube break. The iodine fission product activity available for release to the environment is a function of the primary to secondary coolant leakage rate, the assumed increase in fission product concentration and the mass of steam discharged to the environment. Since the plant will be in Mode 4 at the implementation of the chemical cleaning, the radiological effluent amounts in the FSAR assumed event conditions would not be released. The plant will be in a stable condition with both trains of Shutdown Cooling available. Therefore the radiological consequences of this event are equal to or less than those in the FSAR. The cleaning process will result in the steam generator internals being exposed to the iron and copper/lead removal solvents which along with the corrosion products, will remove some of the base metal from the wetted steam generator surfaces. This base metal corrosion is acceptable as long as the corrosion rates during the application of the process do not result in acute material loss such that when added to the expected lifetime operational loss, could result in exceeding the design allowable. The amount of solvent added to the steam generator is limited to the estimated amount of deposits available for dissolution. There will be no excessive additions of chemical cleaning solvents that would corrode any internal steam generator materials beyond what has been previously tested. The chemical cleaning will also affect the clearances between the tube supports and the tubes. An evaluation was performed to determine the flow induced vibration on the steam generator tubes following the chemical cleaning. The evaluation concluded that the increase in the tube support to tube clearances due to the chemical cleaning are not likely to cause a tube to develop a significant wear indication if the indication did not exist prior to the cleaning. For those tubes that have a pre-existing condition, the wear has a potential to increase marginally by the end of the Steam Generator design life. These tubes have been previously plugged based on the recommendations of the NSSS vendor. Therefore, it is concluded that the proposed activity does not increase the probability of a malfunction of the equipment important to safety. An accident of a different type than those evaluated in the FSAR will not be created. The only change in the plant is the way the unit is operated during the Mode 4 high temperature cleaning sequence. All previously postulated events, whether they be increase or decrease in heat removal events bound the events which could possibly occur during the Steam Generator Chemical Cleaning. The most probable malfunction of equipment important to safety located in the containment that would be impacted by solvent leak would be the ventilation systems using charcoal for radioactive iodine removal. The main decomposition product of the chemical solvents is ammonia that will degrade the charcoal iodine removal efficiency. An ammonia gas monitor will be located in containment and if any ammonia has been detected while a safety related ventilation system is in operation whose charcoal could be compromised, the charcoal will be sampled to ensure it can still provide the efficiency required for the system to provide its safety related function.

6. 2000-053; Steam Generator Chemical Cleaning - Environmental Effects

DESCRIPTION OF CHANGE

The purpose of this evaluation is to address the radiological and non-radiological impact of the steam generator chemical cleaning process (SGCC) and the chemical waste processing on plant safety, plant personnel, and the general public.

REASON FOR CHANGE

During the Steam Generator Chemical Cleaning process and waste processing, the use of large quantities of hazardous chemicals, the generation of substantial quantities of potentially radioactive and hazardous liquids and gases, and the handling of these large quantities of hazardous chemicals and potentially radioactive materials may impact the environment.

50.59 EVALUATION

The SGCC and processing of wastes generated during the process will be performed in compliance with all applicable state and federal laws, permit conditions, and licensing commitments. Amendments to non-radiological permits are required – a) amendment requests to the Louisiana Department of Environmental Quality (DEQ) to add hydrazine and ammonia to the list of chemical cleaning agents used at the site and permit stormwater collected inside the bermed area during rainfall events to be discharged to Outfall 004 via a drainage ditch; b) EPA Permit LAD000757450 declaration of use of Trivalent Chromium Exclusion through the Louisiana DEQ; c) temporary variance from the Louisiana DEQ to operate temporary Steam Generator Chemical Cleaning equipment and discharge ammonia and hydrazine through four emission sources; d) temporary variance from state of Louisiana DEQ to use a diesel generator as a source of backup electricity for SGCC equipment, two diesel pumps to run nitrogen superpumpers used to support Steam Generator draindown and two kerosene vaporizers to convert liquid nitrogen to gas.

The discharge of non-radiological materials in airborne and liquid waste or temporary storage of potentially mixed or hazardous waste will not have any measurable impact on the general public. Radioactive release will be controlled in accordance with 10CFR20 and 50 and 40CFR190. For liquid releases, wastewater will be transferred to the plant and discharged through an existing line with an installed radiation monitor. All sampling and analysis provisions for radwaste tanks in the Offsite Dose Calculation Manual (ODCM) and the Radiological Effluent Controls Program will be applied to the releases from the Batch Release Waste Tanks. Negligible airborne releases of radioactive materials are anticipated during Atmospheric Dump Valve (ADV) venting, Main Condenser Evacuation System Exhaust, filling of the waste tanks, and from the Vent Trailer. Based on this evaluation, the proposed SGCC process and subsequent waste processing will not result in an unreviewed safety question, a significant environmental impact, nor require any Technical Specification or Environmental Protection Plan changes with regards to environmental and radiological considerations.

7. 2000-056; Change to Technical Specification Bases on Containment Systems - Internal Pressure, Air Temperature and Containment Vessel Structural Integrity

DESCRIPTION OF CHANGE

The proposed changes revise Technical Specification Bases Sections 3/4.6.1.4, 3/4.6.1.5 and 3/4.6.1.6. The revisions clarify the Bases and remove the calculated peak containment pressure and containment pressure measurement uncertainty from the Bases.

REASON FOR CHANGE

Technical Specification Bases include certain information, i.e., calculated peak containment pressure for Main Steam Line Break event and instrument error for containment pressure measurement that are old and are inaccurate. These parameters are dynamic in nature and may change due to revised containment analyses or instrumentation and/or calculation changes. These parameters are provided in other documentation, e.g., FSAR and design basis calculations. To avoid having to revise these Bases due to changes in the peak calculated pressure and instrument measurement uncertainty, this information is removed from these Bases. The containment design pressure, which is the limit for post-accident containment pressure, is retained in the Bases. Also editorial changes have been made to Technical Specification Basis 3/4.6.1.5 to provide clarification.

50.59 EVALUATION

The 10CFR50.59 evaluation demonstrates that the proposed changes to these Bases are editorial in nature and do not impact any system, component or equipment or the manner in which the plant is operated. Deletion of the calculated peak pressure from the Technical Specification Bases does not impact any margin of safety. This parameter is documented in other documents e.g., FSAR and calculations. The change does not impact the probability of occurrence or consequences of any accident, does not impact any plant structure, system or component or the manner in which the plant is operated and does not reduce the margin of safety.

8. 2000-061; Cycle 11 Core Operating Limits Report, Rev. 0

DESCRIPTION OF CHANGE

Cycle 11 Reload.

REASON FOR CHANGE

The Cycle 11 fuel management differs from Cycle 10 in the following ways: 1) shorter cycle length; 2) smaller feed batch size; and 3) decreased enrichment. The Cycle 11 safety analyses were performed based on cycle endpoints that bound the cycle operating lengths. The peak pin burnup remains below the limit imposed by the NRC approved Combustion Engineering topical report. The effects of fuel rod bowing on Departure from Nucleate Boiling Ratio (DNBR) margin have been incorporated in the safety and setpoint analyses. All Design Basis Events were evaluated and determined that the consequences of all Cycle 11 non-LOCA transients are bounded by the results already on the Waterford 3 docket. The results of the Cycle 11 Reload Analysis Report analyses indicated no adverse changes to parameters significant to Thermal Hydraulics, Fuel Performance, LOCA and Non-LOCA Safety Analyses except for the revised four Reactor Coolant Pump coastdown curve. The COLSS/CPC setpoints will be established to assure acceptable results with respect to the more adverse coastdown curve.

50.59 EVALUATION

All Cycle 11 design basis events were found to be either bounded by the reference analysis or to be within the appropriate NRC acceptance criteria. The probability and consequences of design basis accidents have not been increased. Technical Specification margin of safety has not been decreased.

9. 2000-064; Waste from Main Condenser Hotwell / Yard Oily Water Separator Sump Temporary Storage Area

DESCRIPTION OF CHANGE

This evaluation addresses the impact of the temporary use of approximately 30 Baker Tanks to store water drained from the Main Condenser Hotwell and the Yard Oily Water Separator Sump containing tritium and chemical constituents from the solvent used to chemically clean the steam generators. The temporary use of Baker Tanks to store tritiated wastewater addressed by this evaluation consists of storage of radioactive material contained in sealed containers in the Owner-Controlled Area and spill control procedures to prevent or minimize the uncontrolled/unmonitored release of radioactive material and metal cleaning chemicals to unrestricted areas.

REASON FOR CHANGE

During the Steam Generator Chemical Cleaning process, higher than expected levels of chemicals were identified in the Turbine Building Industrial Waste Sump and Condensate system. Investigations revealed leakage was occurring from the Steam Generator Blowdown line, which is the chemical injection flowpath to the Steam Generator into the Steam Generator Blowdown Flash Tank. Chemical solvent and water solution contained in the Blowdown Flash Tank entered the Condensate system via vacuum drag either through the Flash Tank vent line or through the Steam Generator Blowdown pumps. Further investigations also revealed that the Condensate relief valve on the #4 Intermediate Pressure Heater was lifted and discharging approximately one gpm to the floor drain and into the Turbine Building Industrial Waste Sump; and it appears that some leakage from the Condensate System to the Condensate Storage Tank also occurred. The actions taken were to isolate the leakage flowpaths and discharge the Turbine Building Industrial Waste Sump, Condensate system and Yard Oily Water Separator Sump into temporary holding tanks.

50.59 EVALUATION

The temporary storage and disposal of water drained from the Main Condenser Hotwell and the Yard Oily Water Separator Sump will be performed in compliance with all applicable state and federal regulatory requirements, permit conditions and plant procedures. These requirements establish radiological and environmental controls during the temporary storage of this wastewater to minimize any uncontrolled or unmonitored releases of effluent containing radioactive materials and higher than normal concentrations of chemicals to unrestricted areas, and assure adequate public health and safety with minimal environmental impact. Any radioactive releases to the environs will be controlled in accordance with the 10CFR20, 50 and 40CFR190, as specified in the Technical Requirements Manual and Technical Specification. All sampling and analysis provisions for radwaste tanks in the Offsite Dose Calculation Manual and Waterford 3's Radiological Effluent Controls Program will be applied to the releases from any Baker Tank Farm container. Analysis has determined the only detectable radioactive material contained in the Baker Tank Farm containers is tritium (hydrogen-3). All releases will be evaluated for significance in accordance with the Offsite Dose Calculation Manual. The proposed change will not result in an unreviewed safety question, a significant environmental impact, nor require any Technical Specification or Environmental Protection Plan changes.

10. 2001-003; TRM Table 3.3-2 Cold Leg Temperature Response Time

DESCRIPTION OF CHANGE

The proposed change increases the TRM Table 3.3-2 and FSAR Table 7.2-6 reactor protective cold leg temperature instrumentation response time from 0.258 seconds to 0.300 seconds based upon the conclusions of Combustion Engineering letter WS-FO-2001-0001. The cold leg temperature instrumentation is used in Core Protection Calculators (CPCs) for a reactor trip protection function.

REASON FOR CHANGE

The increase in allowed Core Protection Calculator cold leg temperature response time is intended to provide more margin between the actual instrumentation response and the surveillance requirements.

50.59 EVALUATION

The proposed change has no affect on the design basis accident results with respect to meeting the design and regulatory acceptance criteria. The proposed change does not physically change any of the reactor trip structures, systems or components and does not change the intended Core Protection Calculator safety function. The cold leg instrumentation and cold leg response times are not initiators of any previously evaluated accidents. The accident consequences remain bounded by the existing analysis due to the available thermal margin or conservatism existing for the current cycle. The margin of safety as defined in the licensing bases is assured by meeting the specified design and regulatory acceptance criteria. The reactor trip system (CPCs) maintains the margin of safety by meeting its intended safety function. The proposed change increases the allowed CPC instrumentation cold leg temperature response time. The limiting accident with respect to the proposed change is the Asymmetric Steam Generator Transient (ASGT). The ASGT analysis demonstrates that the initial available thermal margin exceeds that used up during the transient. The larger cold leg temperature response time would increase the thermal margin degradation and the corresponding initial thermal margin requirements. For Cycle 11, a CPC Berr1 penalty was installed to cover a more adverse Reactor Coolant Pump coastdown. This CPC penalty is sufficient to bound the thermal margin necessary to cover the increased allowed response time. Thus, crediting the CPC BERR 1 penalty ensures that the ASGT consequences remain bounded. Since the accident consequences remain bounded by the existing analysis due to the available thermal margin existing for the current cycle, the increase in allowed response time does not reduce the licensing bases margin of safety. The proposed change also does not increase the probability or consequences of any design basis accident.

11. 2001-015; TRMCR 01-006, Technical Requirements Manual Table 4.3-8

DESCRIPTION OF CHANGE

Remove the quarterly functional surveillance testing requirements for liquid waste discharge flow transmitters BM-IFT-0627 and LWM-IFT-0647. The Boron Management System (BMS) and Liquid Waste Management System (LWMS) are designed to collect and process radioactive and non-radioactive wastewater for discharge from the plant. The radiation monitors for the discharge effluent for each system continuously monitor the liquid being discharged and provide indication, alarm, and flow termination functions in the event a high radiation signal is detected, flow of the monitored fluid through the detector is less than the required minimum or a failure is detected in the monitor. Both the BMS Liquid Waste Discharge Flow transmitter (BM-IFT-0627) and the LWMS Liquid Waste Discharge Flow transmitter (LWM-IFT-0647) are located in the release flow path just downstream of their discharge valves and prior to their respective Liquid Waste Radiation Monitors. Flow signals are sent to meter indication, recorder indication process input and indication on the flow control valve controller, a flow totalizer and indication on the plant monitoring computer. Flow Loop Check and Calibration procedures are performed on these transmitters every 18 months. Conversations with the System Engineer and a review of past calibrations dating from Refuel 5 to Refuel 10 indicate that no adjustments have been necessary in order to return the flow loops into calibration. Since Technical Specifications require that a Channel Calibration include a Channel Functional Test, the past satisfactory Channel Calibrations warrant that quarterly functional tests are not necessary for these transmitters and flow loops. Operations department procedures allow operation of the discharge flow control valves in either Automatic or Manual at the discretion of shift supervision. Therefore these flow transmitters provide no automatic protective functions for these systems and the functional requirement Technical Requirements Manual Table 4.3-8 may be deleted.

REASON FOR CHANGE

Corrective action for CR-WF3-2001-0501. These devices do not serve to terminate effluent flow and should not have any functional requirements in the TRM.

50.59 EVALUATION

The BMS and LWMS liquid discharge flow transmitters perform no protective functions. This change does not add or modify any structure or component in either system. Therefore the probability of an accident previously evaluated in the FSAR will not be increased due to removing the quarterly functional test requirement. All components in the liquid effluent discharge portions of the BMS and WMS are non-seismic Category I and non-safety related. The Liquid Waste System Leak or Failure analysis assumes these systems release their contents to the Reactor Auxiliary Building and the spilled liquids are contained within the building. The only resultant offsite doses will occur as a consequence of released noble gases and of iodines assumed to volatilize from the spilled liquids. Flow signals from BM-IFT-0627 and LWM-IFT-0647 do not function to terminate effluent flow. The associated radiation monitors will function to terminate flow on a high radiation signal, low monitor flow signal or monitor failure. Therefore the consequences of any accident previously evaluated in the FSAR will not be increased due to removing the quarterly functional test requirement.

C. ENGINEERING REQUESTS

1. 1998-081; ER-W3-98-0888-00-00, Containment Fan Coolers Temperature Control Valve Enhancements

DESCRIPTION OF CHANGE

The proposed change will modify the Containment Fan Cooler Temperature Control Valves by replacing the air regulators which control the valve position (and flow) and installing a quick exhaust valve to ensure consistent valve timing (to the open position).

REASON FOR CHANGE

The range of the air regulators currently installed does not allow for adequate control of the valve position and flow. Air regulators with a smaller range and more adjustability will allow for better control of the valve position and flow. During valve maintenance, the positioner was replaced. Due to tolerances of the positioner, the valve timing to open was slowed to an unacceptable value. A quick exhaust valve installed between the positioner and the valve operator will ensure a quick open of the valve, when required. A restriction will be added on the exhaust port of the quick exhaust valve which will slow open to prevent valve damage and perturbations in the Component Cooling Water system.

50.59 EVALUATION

No accidents will have either their probability or consequences affected by this change. Upon receipt of a Safety Injection Actuation Signal, the valves will fail to the open position to allow for full flow to the Containment Fan Coolers. The only equipment potentially affected is the Temperature Control Valves and the Containment Fan Coolers. However, the valves will continue to open within the required 17.5 seconds following installation of the modification. No new system interactions are created and no new failure modes are introduced. No protective boundaries are affected and no margin of safety reduced by this change.

2. 1998-082; ER-W3-99-0013-00-00, Replacement of Westinghouse Model 75RE Recorders

DESCRIPTION OF CHANGE

This modification will replace 39 of the Westinghouse Model 75RE dual pen recorders presently used in the Control Room for displaying various parameters with Westronics Series 1200 dual pen recorders. This change affects RG 1.97, safety-related, and non-safety related recorders.

REASON FOR CHANGE

The maintenance costs associated with the existing Westinghouse Model 75RE recorders is unreasonably high due to lack of Westinghouse support for existing recorders, chart paper alignment problems, ribbon cables shorting out, and replacement expense for the feedback slidewire.

50.59 EVALUATION

The new Westronic recorders will be purchased safety-related, Seismic I and will comply with the requirements of IEEE 323, IEEE 344, IEEE 7-4.3.2-1993, IEC 801 and will be year 2000 compliant. The information displayed will be the same, the recorders will be mounted in the same locations, and the power and signal sources will remain the same. All associated instrument uncertainty, control room, and control room heat load calculations will be revised. No accidents or important-to-safety equipment are affected, no protective boundaries are affected, and no margin of safety is reduced.

3. 1998-093; ER-W3-99-8019-00-00, Radioactive Material Storage Building

DESCRIPTION OF CHANGE

This change will remove the dry waste compactor from the existing Compactor Building, which is currently used to house the dry radioactive waste compaction machine and for storage of compacted dry waste. The dry waste compactor will be decontaminated, disassembled, and salvaged. The existing Compactor Building is not part of the Nuclear Island. The building will continue to be used for radioactive material storage and will be renamed the "Radioactive Material Storage Building" (RMSB). The building will be enlarged to store contaminated PCs and dry radiological consumables in two separate rooms. The rooms will have separate air conditioning units to prevent cross contamination. The existing part of the building will be used to store and repair refueling tools.

REASON FOR CHANGE

The dry waste compactor was installed to compact dry waste into metal boxes to reduce waste volume and disposal costs. The dry waste compactor is no longer used because it is more cost effective to send dry waste to an offsite licensed facility for compaction and disposal. In addition, there is a need for storage space, therefore, the building will be enlarged and used as a radioactive material storage area and for a refueling tool storage and repair area.

50.59 EVALUATION

This change revises the description in the FSAR but does not affect the design or licensing basis of the Solid Waste Management System. There are no analyzed accidents which are initiated by the dry waste compactor. The existing analysis for tornado generated missiles envelops the new building addition if it was to be destroyed by a tornado accident, since the debris is representative of Waterford 3 missiles spectrum. The exiting analysis for flooding is not changed by the modification since all construction is away from the nuclear plant island structure. In a tornado scenario striking the Radioactive Material Storage Building, the general public exposure dosage in 2 hours is limited to less than 10% of the 10CFR100 limit (i.e., 2.5 Rem). Any effluent release would be as a result of an air born, flooding or a fire accident scenario. These are considered secondary release paths and not monitored as long as the release is less than 10% of the instantaneous release limits. The material to be stored in the Radioactive Material Storage Building is enveloped by the material in the Low Level Radioactive Waste Facility (LLRWF). The LLRWF calculations performed for flooding and fire accident scenarios, indicate any releases would be significantly below the 10% limit and would not require monitoring. Therefore LLRWF scenarios envelope the Radioactive Material Storage Building and any effluent releases would be below the 10CFR100 limits. All activities in the Radioactive Material Storage Building are governed by Health Physics plant procedures. Health Physics surveys monitor the level of radiation and contamination and then implement posting requirements. No new methods of failure are created by the modification. No new interfaces with any safety related equipment is created as a result of this change. This change will not impact any effluent release paths. All controls in place ensure that the general public is protected within applicable regulatory limits.

4. 1998-100; ER-W3-98-0946-00-00, Shortening the Existing Suction and Discharge Hydrostatic Test Connections for the Charging Pumps and Removing the Associated Valves.

DESCRIPTION OF CHANGE

The proposed change will shorten the 1-1/2" suction piping on Charging Pump 'A' hydrostatic test connection and cap it, shorten the 1" discharge piping on Charging Pumps A, B, and A/B hydrostatic test connections and cap them, and remove all hydrostatic test connection valves associated with this piping.

REASON FOR CHANGE

A metallurgical evaluation of an identified crack on a cantilevered hydrostatic test connection line during refueling outage 8 concluded that the weld failure was the result of high frequency fatigue failure. A fatigue crack may have initiated at the weld root defect (lack of fusion), propagated by low stress, high cycle fatigue. The shorter cantilevered length with no valve weight would eliminate the possibility of high frequency fatigue failure due to charging pump vibrations. The new piping configuration with a welded cap at the free end should prevent any leakage to atmosphere. Design Engineering evaluated all cantilevered hydrostatic test connections under CR-97-1160 and recommended shortening the cantilevered length and deleting the associated valves. The intent of this change is to shorten the cantilevered length from the run pipe and remove the concentrated valve load acting at the free end.

50.59 EVALUATION

The proposed change will not have any impact on the function of the Chemical and Volume Control System or the charging pumps. This is an enhancement to the present piping design. Removing the valve weight will stabilize the piping vibrations. The welded cap at the free end should prevent any leakage to atmosphere. The 50.59 evaluation has concluded that there is no reduction in the margin of safety as defined in the basis of any Technical Specification or safety analysis and no Unreviewed Safety Question is created.

5. 1998-112; ER-W3-99-3517-00-00, Emergency Feed Water Pump Turbine Main Steam Supply Drip Pot Normal Drain Bypass

DESCRIPTION OF CHANGE

The existing Emergency Feed Water Pump Turbine (EFWPT) Main Steam Supply Drip Pot Normal Drain Bypass valve, MS-407, will be replaced with a fail open air operated valve to enhance draining and improve reliability of the Emergency Feed Water system. The replacement valve will be installed with a seismically qualified solenoid valve and two limit switches. The replacement valve has an air regulator on it, which will be relocated and mounted on a separate support with the solenoid nearby. An existing spare Instrument Air valve will be utilized and new tubing will be added to supply Instrument Air to MS-407. A new non-Class 1E, 120 VAC power supply will be provided for the solenoid circuit. A combination of existing, new, and spare cables will be used. The piping immediately downstream of the new valve will be replaced with stainless steel due to its better resistance to erosion/corrosion. The existing non-safety steam trap and drain piping loop will be reconfigured to minimum dimensions to allow gravity draining of the drip pot under EFWPT standby conditions.

REASON FOR CHANGE

The existing valve is obsolete and unacceptable for further service due to high a leakage rate. Although MS-407 is not required to operate when the EFWPT is needed to perform its safety function, reliability of the Emergency Feed Water (EFW) system can be enhanced with rapid operation of MS-407 in an EFWPT start. The desirable time frame for draining the EFWPT steam supply piping during an EFW turbine start is prior to the governor valve taking control of the turbine. The current drain configuration uses a motor operated valve, which cannot effectively remove condensate during this time frame. The replacement air operated valve will open significantly faster and enhance draining during the start-up period. The existing configuration prevents the automatic draining of the EFWPT steam supply line when the turbine is not operating, which is required to prevent an accumulation of condensate. During this operating mode, condensate drains to a drip pot in the EFW steam supply line and is gravity drained through a steam trap to the floor drain. Due to the existing steam trap and piping configuration, a water level is maintained in the drain piping downstream and the trap is unable to function. Condensate is currently drained manually.

50.59 EVALUATION

An "Increase in Feedwater Flow" accident can be initiated by inadvertent actuation of EFW with the main feedwater system in manual. Reconfiguring the non-safety related drains and changing out the Main Steam Supply Drip Pot Normal Drain Bypass will not increase the probability of occurrence of this accident as no operator or automatic actions are added or affected by this change. Loss of steam into the condenser during a Loss of Offsite Power or Station Black Out, where cooling water is not available, was evaluated. Any potential radioactive release is bounded by the existing off-site dose calculations as the steam is simply taking a different path and exiting through the rupture diaphragm rather than through the Atmospheric Dump Valves. The only failure mode affected by this change is the EFWPT Main Steam Supply Drip Pot Normal Drain Bypass operator. The operator is being changed from fail 'as is' to fail open. Since MS-407 is normally closed and not required to open to perform its safety function, a malfunction of the operator does not affect any important to safety equipment. No protective boundaries are affected by this change and no margin of safety is reduced.

6. 1999-002; ER-W3-98-0841-00-00, Emergency Feed Water Valves Booster Relay Overpressure Protection Replacement

DESCRIPTION OF CHANGE

The proposed change will revise the existing tubing on each of the four Emergency Feed Water flow control valves. Volume booster relays were supplied by the valve manufacturer and installed on each valve. In order to maintain the pressure on these relays below their rating, this change relocates the Instrument Air supply from the Instrument Air supply header to the output of the existing filter/regulator.

REASON FOR CHANGE

The supply pressure from the Instrument Air/Nitrogen system is capable of exceeding the design pressure rating of the volume booster relays on these valves. The setpoint of the filter/regulator is below the pressure rating of the relays, therefore relocating the Instrument Air supply will prevent the possibility of exceeding the design pressure rating.

50.59 EVALUATION

The proposed change will potentially increase the closing times of the valves by several seconds. However, there is no strict requirement for the Emergency Feed Water control valve closing time in the Waterford 3 licensing basis. It is typical in FSAR Chapter 15 accident analyses for a Main Steam Isolation Signal (MSIS) to occur prior to an Emergency Feedwater Actuation Signal (EFAS) or for the Emergency Feed Water actuation to be a manual operator action. With a MSIS present, Emergency Feed Water will only be fed to the intact Steam Generator(s) and the small increase in EFW control valve closure time will have no adverse affect on maintaining Steam Generator level. For events with the Steam Generators intact, the additional inventory will flow to the Steam Generators for residual heat removal. The small Feed Water Line Break scenarios predict EFAS prior to MSIS actuation. A small increase in EFW control valve closing time will have no affect on the Feed Water Line Break results with respect to regulatory or design acceptance limits. The longer closing time will allow more EFW to blowdown to containment prior to isolation, but the additional water inventory does not present a flooding concern since it is bounded by the post-LOCA containment flooding calculation. The EFW temperature range is 70 to 100°F; thus, the potential increase in containment temperature and pressure due to additional EFW inventory is negligible. For the small Feed Water Line Break event, the amount of Condensate Storage Pool inventory lost to containment will not hamper the cooldown/entry to Shut Down Cooling conditions since the EFW inventory used is bounded by the events which require Wet Cooling Tower basin inventory. The FSAR Chapter 15 events credit operator action to control EFW to reach Shut Down Cooling initiation conditions. The Steam Generator Tube Rupture event specifically credits EFW to the affected and unaffected Steam Generators for a 100 °F/hour cooldown until a temperature of 520 °F is met. At this temperature, the affected Steam Generator is isolated. The primary safety function of the EFW control valves is to open on EFAS or as an operator action. The proposed change has no affect on this safety function. The closing function is not explicitly documented in the licensing basis, thus a reasonable closure time of 25 seconds will be used in the acceptance test to verify valve operation.

7. 1999-035; ER-W3-98-0067-00-00, Route Silica Monitor Drains To Sample Recovery Tank

DESCRIPTION OF CHANGE

As part of the Secondary Sampling System (SSL), the silica monitors analyze secondary water samples for water chemistry, which is a non-safety, non-seismic function. The discharged effluent from the silica monitors is currently drained to the chemical waste tank. This change reroutes the silica monitor discharge tubing to the drain header for the sample recovery tank and provides an alternate flow path to the chemical waste tank.

REASON FOR CHANGE

This change will reduce the volume of radioactive liquid waste processed by the Liquid Waste Management system by approximately 139,000 gallons per year and will also extend the life of the resin used in the Liquid Waste Management system. Additionally, it will assist operations in maintaining the pH balance in the industrial wastewater.

50.59 EVALUATION

It is concluded that this modification will not affect the environmental aspects of the FSAR documents, will not reduce the margin of safety as defined in the basis for any technical specification and no Unreviewed Safety Question is created. The failure of the Secondary Sampling system is not postulated to initiate any accident scenario previously evaluated in the FSAR. Therefore this change does not increase the probability of occurrence of an accident previously evaluated in the FSAR. The discharge of the silica effluent to the Sample Recovery Tank will introduce tritium from the secondary system to the Industrial Waste Sump. However, the contents of the Industrial Waste Sumps already contain tritium and are sampled and tested for tritium on a weekly basis. Should unacceptable levels of tritium be detected, the Industrial Waste Sump discharge is isolated and routed to the Reactor Auxiliary Building Waste Tank for processing prior to off-site release. Also, in the event of a Primary-to-Secondary leak, the addition of the silica effluent will increase the radioactivity levels in the Industrial Waste Sumps. However a radiation monitor monitors the discharge of the Industrial Waste Sump prior to release to the environment. In the event of the presence of radioactivity in the Industrial Waste Sump, the discharge is diverted to the Waste Management System. It is noted that the addition of the silica effluent to the Industrial Waste Sump does not increase the discharge to the environment, since the volume of water from the Industrial Waste Sump is dependent on the pump discharge flow. The acids used for the analysis of the silica samples will increase the acid levels of the industrial wastewater. Acid levels will be monitored and reported to the Environmental Protection Agency under Louisiana Permit No. LA0007374. The above safeguards are in place to ensure that 10CFR20 limits are not exceeded during normal and accident conditions.

8. 1999-038; ER-W3-99-0586-00-00,Leak Repair of High Pressure Turbine Diaphragm Drain Line

DESCRIPTION OF CHANGE

This change will modify a 3/8" piping drain on the lower half of the governor end of the High Pressure Turbine to facilitate a leak repair while the Gland Seal System is depressurized. To seal the leaking drainpipe, a leak repair will be installed while the gland seal system is in operation. The area near the drainpipe is not accessible during system operation due to the steam leak and the confined nature of the area. The change will modify the 1/2" tubing connected to the leaking 3/8" piping to allow for repairing the leak. To provide access to the tubing from a safe working area the 1/2" tubing will be cut and rerouted to a nearby accessible area on the Turbine Deck. A leak repair sealant injection valve will be installed on the Turbine Deck end of the 1/2" tubing and a tubing cap will be installed on the remaining free end of the tubing near the High Pressure Turbine case.

REASON FOR CHANGE

The 3/8" drain pipe on the governor end of the High Pressure Turbine is leaking at the connection to the lower half of the High Pressure turbine case.

50.59 EVALUATION

The changes to temporarily eliminate a 3/8" drain connection on the governor end of the High Pressure Turbine will not affect the probability of occurrence of an accident previously evaluated in the FSAR (a Loss of Condenser Vacuum and associated turbine trip) because the integrity of the Gland Seal system will be maintained and no new flow paths or system interactions are created. During plant startup the High Pressure Turbine gland area is pressurized by the Gland Seal steam and at power operations it is pressurized by steam flow through the inner gland, so condenser vacuum would not be affected by leakage or even failure of the new tubing run that extends up to the Turbine Deck. The integrity of the capped main steam tubing which is routed to the condenser will be maintained by installation of a 1/2" Swagelok tube plug which maintains the system design requirements. The consequences of an accident previously evaluated in the FSAR will not be increased because the integrity, function, and operation of the affected systems and components will not be affected. In addition, the Gland Steam System is not relied upon for mitigation of any accident previously analyzed in the FSAR. The gland Steam System is not considered important to safety and the proposed modifications to the Gland Seal System will not increase the probability of occurrence of a malfunction of equipment important to safety previously evaluated in the FSAR. The proposed modification will not create the possibility of an accident of a different type than any previously evaluated in the FSAR because no new system interactions are created and the integrity and function of the Gland Seal System and the High Pressure Turbine are maintained. The proposed modification does not reduce the margin of safety as defined in the basis for any Technical Specification or safety analysis because no margin of safety as defined in the Technical Specification bases is associated with either the Gland Seal System or the High Pressure Turbine.

9. 1999-055; ER-W3-98-0847-00-00 - Essential Chillers Relay Module Replacement

DESCRIPTION OF CHANGE

This proposed change will replace the relay modules used in the control systems of the three essential chillers. The design of the new replacement modules does not incorporate the use of circuit breakers CB1 and CB2, and relay K4, thus they are deleted per this change. This change also adds a time delay relay separate from the relay module, which provides control power for protection circuit interface.

REASON FOR CHANGE

The updated relay modules will perform the same functions with fewer components. The existing relay modules, which are obsolete, are approaching the end of their design life. Currently there are no original model relay module spares available. The original On-Stop pushbutton will be used instead of a two-pole switch (recommended per manufacturer design drawings), therefore a time delay relay is required to provide initial control power for protection circuit interface.

50.59 EVALUATION

The Essential Chiller system furnishes chilled water for space cooling purposes and rejects heat through the Component Cooling system to the Ultimate Heat Sink. The safety function of the Ultimate Heat Sink is to mitigate the consequences of an accident by dissipating the heat removed from the reactor and its auxiliaries after a design basis accident. The Essential Chiller system is not considered an initiator for any accident described in the FSAR. This change enhances the availability of the chillers by providing an upgrade of the original chiller relay module. This modification will not increase the radiological consequences of an accident previously evaluated in the FSAR. The fuel-clad barrier, the reactor pressure boundary and the containment structure will not change due to this modification; therefore, the radiological dose as a result of a design basis accident will not be affected by this change. No new failure modes are presented by the failure of the added time delay relay or the replacement relay module, thus this change cannot increase the probability of occurrence of a malfunction of equipment important to safety. The safety function of the Essential Chiller system, which involves mitigating the consequences of an accident, will not be affected by this design change. The failure mode and associated effect on the system, which are currently in the FSAR for the Essential Chiller system, will not be affected by the changes. The new module and the control relay are qualified to meet the requirements of safety equipment previously installed on the chiller control panel. This change does not create the possibility of an accident of a different type than previously evaluated in the FSAR. The chiller controls will continue to function / operate as per the original design. The new components satisfy the original design requirements and qualifications. An upgrade of the control relay module and the addition of the time delay relay will not change any margin-of safety or impact any protective boundary that is applicable to the Chillers.

10. 1999-057; ER-W3-98-1171-00-00, Shield Building Ventilation System Setpoint Change

DESCRIPTION OF CHANGE

The Shield Building Ventilation (SBV) system is designed to operate during and post accident to maintain the shield building at a negative pressure and filter the annulus such that it remains within the limits of 10CFR100. The SBV is automatically started on a Safety Injection Actuation Signal. After actuation, the operator may secure one train of SBV, with one train in operation. The standby train inlet valve will open to allow a minimum airflow through the standby train and charcoal bed. In addition, should the train in operation secure or trip, the standby train is automatically started. The configuration change will revise the setpoint for the operation of the standby train inlet valve, when only one train of SBV is in operation. The tolerance and reset limit of the start signal for SBV IDPIS5051 A and B will also be revised.

REASON FOR CHANGE

The current setpoint to open the inlet valve of the standby train of SBV allows the valve to open when the differential pressure of the operating train is above the reset of the switch (10.2-inw). If the differential pressure of the running unit is between 8-inw and 10.2-inw on startup, then the standby filter unit inlet valve may not be open, as required. The tolerance and the reset limit of the start signal for SBVIDPIS5051 A and B will be revised to incorporate the current accuracy and deadband information for the instrument from the vendor.

50.59 EVALUATION

The primary function of the Shield Building Ventilation system is to assure that the annulus pressure following a Loss Of Coolant Accident will not become positive, which would permit primary containment outleakage to escape unfiltered directly through the shield building wall to the outside atmosphere. This configuration change does not affect the ability of the system to perform the required safety function. The new tolerance and reset limit for the interlock to start the standby train of SBV when the operating train has been tripped or secured, does not affect the ability of the SBV System to maintain the negative pressure of the annulus. The new tolerance and reset limit specified are within the operation of the system, and will ensure the proper operation of the SBV system. There are no Unreviewed Safety Questions as a result of these configuration changes.

11. 2000-002; ER-W3-99-0426-00-00, Pipe Support for Steam Generator Feed Pump Discharge Line

DESCRIPTION OF CHANGE

The proposed modification will reconfigure the discharge drain line for the Steam Generator Feed Pump. Valve FW-117B will be removed from the facility and the line will be reoriented from a horizontal drain that is perpendicular to the process line to a vertical configuration. Valve FW-117A will also be removed from the facility. The drain, however, will remain horizontal, as its location would present an overhead safety hazard.

REASON FOR CHANGE

During plant startup and power ascension, it was noted that the drain piping for the Steam Generator Feed Pump was vibrating excessively. This is a 1-inch diameter pipe, approximately 3 feet in length, that connects to the bottom of the main feed pump discharge piping immediately upstream of valve FW-118B. The vibration was most severe when the pump was operating at approximately 3900 rpm with a flow of approximately 6000 gpm. When Steam Generator Feed Pump flow increases, the vibration was reduced to normal levels.

50.59 EVALUATION

The proposed modification reconfigures both non-safety non-quality Steam Generator Feed Pump discharge drain valves from a double isolation arrangement with a pipe cap to a single isolation valve arrangement with a pipe cap. The drain lines were originally designed in accordance with ANSI B31.1 and Ebasco specifications. The current existing double valve arrangement was a standard practice by Ebasco when design pressure exceeded 800 psig. However, there is no technical basis or requirements dictated by ANSI B31.1 for such an arrangement. The modified configuration utilizes the existing FW-116A/B drain valves and is constructed of the same carbon steel schedule 80 material. In addition, the modified configuration meets the requirements of ANSI B31.1, which is the original design basis for this piping line. Although the Feed Water system is listed in Chapter 15 of the FSAR as an accident initiator, performing the proposed modification does not increase the probability or consequences of these accidents. The proposed modification reduces both the mass and length of the drain lines and as such reduces the stresses that may ultimately result in a failure of the drain lines. Chapter 15 of the FSAR discusses a Feed Water System Pipe Break as a pipe break resulting in a total loss of normal feedwater and a rapid blowdown of one steam generator with a concurrent loss of normal AC power. The proposed modification to reconfigure the 1-inch diameter feedwater drains is bounded by the current FSAR analysis. This 50.59 Evaluation reflects the fact that reconfiguring the subject drain lines will not reduce the margin of safety as defined in the basis for any Technical Specification or Safety Analysis and that no Unreviewed Safety Question is created.

12. 2000-005; ER-W3-98-1165-00-00, Fire Detection/Plant Computer Room Ventilation Interface

DESCRIPTION OF CHANGE

The proposed change disconnects the HY relay originally installed to enable the computer room purge damper D64 to close automatically during a Halon system actuation.

REASON FOR CHANGE

Damper D64 is normally closed and can only be opened by the computer room purge control switch located on CP-18 and LCP-43. A previous design change prevented computer room purge to be accomplished until a fire was suppressed. Subsequent design changes eliminated the Halon system in the computer room and replaced the fire detection panel. Presently, the detectors have to be disarmed at the fire detection panel and the panel reset in order to accomplish a computer room smoke purge.

50.59 EVALUATION

The smoke purge system and the HY relay cannot be the initiator of an accident. Hence, this change will not increase the probability of an accident. No accidents are impacted by this change. Hence there is no potential to increase the consequences of radiological releases. There is no impact on any safety related or equipment important to safety. This change impacts non-safety related functions of a computer room purge relay. The dampers, the fire protection panel and detectors are not considered equipment important to safety. Removal of the HY relay provides easier operation of the system using control switches and manual overrides. System operation, post computer room fire has not changed. Removal of the HY relay has no negative impact on safe operation of the plant, on the ability to shut down the plant, or on any accident previously evaluated in the FSAR. The changes do not introduce any concerns related to new or unanalyzed accident or equipment malfunction scenarios. The changes do not result in a potential for release of radioactive material to the environment. There is no Unreviewed Safety Question associated with this change.

13. 2000-006; ER-W3-99-0699-00-00, Reverse the Fuel Handling Building West Auxiliary Hoist Orientation

DESCRIPTION OF CHANGE

The primary functions of the Fuel Handling Building crane are to transfer the new fuel to the new fuel elevator and move the empty or loaded spent fuel storage casks between the cask storage area and the railroad cars. This change will reverse the orientation of the 15-ton Auxiliary Hoist on the monorail at the west girder of the Fuel Handling Building Cask Crane. This change will allow the hoist cable/hook to operate at the north end of the hoist/cable drum, which will increase the travel limit of the west auxiliary hoist and improve the handling of the fuel casks in the train bay.

REASON FOR CHANGE

Under the present configuration, the hoist cable/hook operates at the south end of the cable drum, which limits the travel of the hook towards the north end of the Fuel Handling Building. This does not allow for a vertical pick of the northern most fuel casks in the train bay. Since the cable has to be swayed significantly towards the north to reach the northern most casks, there is the potential for the cask to swing, which creates a dangerous situation for personnel and a possible source of damage to the new fuel casks and fuel.

50.59 EVALUATION

This modification will perform a 180-degree reversal of the 15-Ton Auxiliary Hoist at the west girder of the Fuel Handling Building Crane. The Fuel Handling Building crane Auxiliary Hoist is not safety related. This change does not affect the form, fit, or function of the crane and satisfies the original design requirements. There are no new loads imposed by the design change on the monorail beam; therefore, the existing stress analysis is unaffected. The proposed change does not increase the probability of occurrence of an accident previously evaluated in the FSAR because there is no credible accident scenario postulated that is associated with the operation of the Fuel Handling Building Crane. The crane is not credited for limiting the radiological consequences of an accident therefore this change does not increase the consequences of an accident previously evaluated in the FSAR. This change will not affect the safety or environmental aspects of the FSAR and will not reduce the margin of safety as defined in the basis for any Technical Specification and no Unreviewed Safety Question is created.

14. 2000-008; ER-W3-98-0244-01-00, Convert Valves MS-101A(B) from Motor Operated Valves to Manual Valves

DESCRIPTION OF CHANGE

This change converts the existing Steam Generator Vent Valves, MS-101A(B) from Motor Operated Valves to manual operation. Valve operation will be by means of handwheels. The electrical equipment for each valve will be spared or abandoned in place. A change to TRM Table 3.8-1 is also required.

REASON FOR CHANGE

The motor operators for these valves cannot be maintained due to their location and lack of accessibility. The NSSS vendor has confirmed that the vent function of these valves can be performed by the Atmospheric Dump Valves.

50.59 EVALUATION

These valves are not accident initiators and perform no function required for safe shut down. The valves will physically remain in the Main Steam piping and will be manually operated if needed. These valves were designed to vent steam during startup and cold shutdown conditions. The vent function will be performed by the Atmospheric Dump Valves. Steam Generator and Main Steam piping are not affected by this change. No new system interactions or reduction in the operability of existing equipment are created. No Technical Specifications are impacted. Therefore, no Unreviewed Safety Questions will result from this change.

15. 2000-010; ER-W3-98-0605-01-00, Installation of Key Operated Switches

DESCRIPTION OF CHANGE

This change permanently installs key-operated switches which will be used to perform tasks identified in two Operations procedures. These procedures currently require Operations to have maintenance personnel install jumpers and/or lifted leads in various equipment. These actions defeat Safety Injection Actuation Signal contacts in the Safety Injection Tank isolation valve breakers and disable the Shutdown Cooling isolation valves SI-401B and SI-405B low pressure open permissive. The new switches can be actuated by the operators and the contacts will eliminate the need for jumpers in the Shutdown Cooling valves' circuit. In addition, a wiring change will be made to the existing Transfer Switch contacts in Auxiliary Panels 1 and 2 to eliminate lifted leads in the Safety Injection Tank valves' circuit.

REASON FOR CHANGE

The practice of installing jumpers/lifted leads is considered an Operations work around. The use of key-locked switches/Transfer Switch will provide a safer, more reliable method of simulating open/closed device contacts.

50.59 EVALUATION

The key locked switches installed by this change will maintain the present configuration of the affected circuits. The switches will only be actuated during the performance of two procedures: "Evacuation of Control Room and Subsequent Plant Shutdown" and "Isolation Panel Fire". There are no new system interactions. However, the switch contacts will be wired into the control circuits of the equipment. The connections/rewiring will be made in the respective MCC cubicles and Auxiliary Panels 1 & 2. These changes will not affect the operation of the equipment. Quality level L1 switches and wire will be installed inside the cubicles/panel. This change will not alter the function or operation of any plant equipment. The new key-operated switches will maintain the quality class of the affected components and will be controlled by Operations personnel in accordance with approved plant procedures. These procedures involve Appendix R activities which are required to safely shut down the plant, but are not bound by limitations that exist during normal/accident conditions. This 50.59 evaluation determines that no licensing basis documents are affected, no Unreviewed Safety Question is created, and that the margin of safety is not reduced.

16. 2000-013; ER-W3-99-0675-00-00, Motor Operated Valves MS-119A and MS-120A, Actuator Gear Change and Valve Stem Replacement

DESCRIPTION OF CHANGE

This design change will increase the capability of the MS-119A and MS-120A valve actuators by replacing the gear set in the SMB-00 actuator resulting in an increase in Overall Actuator Ratio (OAR) from 36.2 to 87.8. Additionally, the single lead valve stems for the valves will be replaced with double lead stems along with new stem nuts. The thermal overloads for the actuator motors will also be replaced. There is an associated increase in stroke time from approximately 6 to 7 seconds, which remains below the 10-second maximum stroke time for containment isolation.

REASON FOR CHANGE

CR-99-0118 documented that the Limitorque actuator for these valves has an OAR of 36.2 installed instead of the 87.8 ratio used in the design basis calculation. The Limitorque supplied data sheet indicates an OAR of 87.8; however, at some time in the past, the gears were changed without proper documentation and configuration control. The effect of this condition is the actuator develops less output torque than determined by the calculation. CR-99-0118 determined that the valves remained operable but degraded. The valves cannot meet the new guidance for actuator capability in Limitorque Technical Update 98-01. Modification is required to increase actuator output and reestablish adequate margin.

50.59 EVALUATION

The proposed changes will increase the output torque capability of the subject valve actuators; thus providing increased torque margin above the minimum design basis requirements and adding assurance that the valves are capable of performing their safety functions. The valves are used in the mitigation of an accident to ensure containment closure. The changes being made will increase the available output torque and stroke time. These valves can in no way cause an accident as described the FSAR. The change will make the valves capable of operating against higher opening/closing forces and will therefore be less likely to fail as a result of higher operating loads. The new valve stem is the same material, fit, form and function as the single lead stem currently in the valves. The only difference is the double lead stem causes the valve to travel twice as far on one rotation as the single lead stem. The double lead stem is an authorized part for these valves by the manufacturer. The impact of the slight change in weight of the actuator has been evaluated. Although the subject valves will have greater actuator output torque, the torque switch settings and associated Motor Operated Valve setpoints will continue to ensure that the maximum allowable torque values associated with the valve actuators are not exceeded. These physical changes will not alter the basic mechanical or electrical operation of the subject valve actuators. The valves will function as they did prior to implementation of this design change. There are no new failure mechanisms introduced by this design change. There are no new system interfaces created. Valve stroke times will increase from approximately 6 seconds to 7 seconds which is still within the required 10-second stroke time. There are no changes to the basic mechanical or electrical operation of the subject valve actuators due to implementation of this design change.

17. 2000-014; ER-W3-99-0857-00-00, SI-404A(B) Relief Valves Setpoint Change

DESCRIPTION OF CHANGE

This change will revise the setpoint for SI-404A(B) from 2485 psig to 2460 psig to account for 25 psig of backpressure.

REASON FOR CHANGE

Safety Injection Relief valves SI-404A(B) have a design and set pressure established at 2485 psig. The valves discharge to the Reactor Drain Tank, which is protected by a relief device, BM-106, with a set pressure of 25 psig. In accordance with ASME III Code requirements, Subsection NB-7000, the setpoint for SI-404A(B) should be lowered by 25 psig to compensate for this source of potential backpressure.

50.59 EVALUATION

These valves are not initiators of any accidents previously evaluated nor are they credited for accident mitigation. The proposed setpoint change does not change the intended safety function, method of performing the intended safety function, or required design and operating conditions for SI-404A(B). This 25 psig setpoint reduction actually provides increased protection of piping for the protected portion of the Shutdown Cooling system because it merits a backpressure that was not previously accounted for. Any failure of SI-404A(B) will merely result in a discharge of borated water to the Reactor Drain Tank and ultimately to the containment sump, if the Reactor Drain Tank relief valve, BM-106, were to lift. The discharge would be a very minimal amount and would not be any more severe than previously evaluated.

18. 2000-015; ER-W3-00-0105-00-00, Special Test for Safety Injection Drain Header Drain Valve SI-342

DESCRIPTION OF CHANGE

The Safety Injection Drain Header Drain Valve, SI-342, will be opened from its normally closed position for a maximum of 72 hours. Subsequent closure of the valve will end the special test.

REASON FOR CHANGE

The Safety Injection Tank levels are increasing. It is believed that one inleakage path is past valves SI-512B (Hot Leg #2 Injection Check Valve) and SI-302 (Reactor Coolant Loop 2 Hot Leg Injection Leakage Drain Valve) and into the Safety Injection Tanks. Opening SI-342 will route any leakage past SI-512B and SI-302 to the Containment Sump. If this is the inleakage path, Safety Injection Tank levels should no longer increase. During the time the valve is open, various parameters will be monitored to verify this is the leakage path and confirm Safety Injection Tank integrity.

50.59 EVALUATION

The position of SI-342, a non-safety related, non-seismic valve, is not a factor in any accidents evaluated in the FSAR. The valve is part of the Safety Injection Tank drain header piping. The low energy piping is not considered a potential cause for any of the accidents evaluated in the FSAR. The Safety Injection system serves to mitigate the consequences of many of the accidents evaluated in the FSAR. The temporary opening of valve SI-342 will have no effect on the systems ability to perform its mitigative functions. Analyzed events involving the Safety Injection system itself would also be unaffected by the open position of the valve, and it does not effect either the flowpath or flow characteristics of the system. Opening SI-342 will depressurize the Safety Injection Tank drain header, which should be the normal configuration of that piping. Depressurizing the header will increase the differential pressure across the Safety Injection Tank fill and drain valves leading to this header. These valves are designed to operate against a differential pressure in this direction. The Safety Injection Tank leakage drain valves receive a confirmatory close signal upon initiation of a Safety Injection Actuation signal. The Safety Injection Tank fill/drain valves are fail close valves. The test procedure explicitly excludes the simultaneous opening of valve SI-342 and a Safety Injection Tank fill/drain valve, or having SI-342 open while performing any system manipulation which involves flow through the Safety Injection Tank drain header. This will ensure that there will be no inadvertent diversion of Safety Injection flows, and therefore will not increase the consequences of an accident. There is no Unreviewed Safety Question associated with this special test. There will be no adverse effect on plant operations. Should the opened valve fail to close, operations can close manual valve SI-3421, which is in series with SI-342, and would thus restore the plant to pre-test status.

19. 2000-016; ER-W3-99-0184-02-00, Weld Repair of Inconel Instrument Nozzles on the Pressurizer

DESCRIPTION OF CHANGE

This change authorizes repair of instrument nozzles 'B' and 'D' located on top of the pressurizer to prevent future leakage due to primary water stress corrosion cracking. This repair method was previously performed on instrument nozzles 'A' and 'C'. The repair consists of cutting and removing the nozzle approximately one inch into the wall of the pressure vessel, inserting a new Alloy 690 nozzle into the hole, and welding the nozzle to the outside of the pressure vessel. The internal bore and restrictive orifice dimensions of the nozzle are unchanged. To provide access to the nozzles, the ER also authorizes temporary removal of instrument tubing connected to nozzles 'A' and 'C.'

REASON FOR CHANGE

Primary water stress corrosion cracking of Inconel Alloy 600 penetrations in the Reactor Coolant System has become a significant problem in Pressurized Water Reactors over the last ten years. The pressurizer penetrations involve nozzles which are inserted through the opening in the vessel wall and are welded to the inside of the vessel wall by a J-groove weld. Primary water stress corrosion cracking has been found in pressurizer heater sleeves, pressurizer instrument nozzles, hot leg nozzles, and Control Element Drive Mechanism nozzles. Pressurizer instrument nozzles 'A' and 'C' were found leaking during refueling outage 9 and were repaired using this technique.

50.59 EVALUATION

This evaluation addresses the potential effects of the pressurizer nozzle repairs during modification implementation. This includes low temperature overpressurization, rigging concerns, hydrogen gas accumulation, and the potential impact of temporarily removing portions of the instrument sense tubing connected to nozzles 'A' and 'C.' Also evaluated is the potential for the final design to affect pressure boundary integrity, welding issues, thermal stresses, corrosion, erosion, stress corrosion cracking, hydrogen embrittlement, and the effect on connected instrumentation. This evaluation concludes that the proposed change will not degrade the integrity of the Reactor Coolant System pressure boundary or the functional capability of the affected instrumentation. All changes are within the Reactor Containment Building and there are no new system interactions created. There are no USQs and this change does not require any Technical Specification changes.

20. 2000-018; ER-W3-98-1387-01-00, Recover Refueling Water Storage Pool Level Operating Margin

DESCRIPTION OF CHANGE

This change increases the maximum allowed Refueling Water Storage Pool (RWSP) level and reduces Operator burden. The height of eight-instrument stands will be changed so that the lowest point of each instrument will be raised above +0.5 ft MSL flood elevation inside containment. Also, the restriction on maximum RWSP level is changed from 87% to 100% minus instrument uncertainties by changing the appropriate alarms and indicator banding.

REASON FOR CHANGE

The maximum level in the RWSP is currently being limited to 87% to ensure that certain RG 1.97 instruments, their associated conduit seals and electrical cable splices located in the containment building are not submerged following a Loss Of Coolant Accident (LOCA). To increase the maximum RWSP level and provide Waterford 3 Operation with more flexibility to maintain the RWSP level, Calculation EC-M89-004 was revised to determine the maximum post-LOCA containment flood level based on an RWSP water volume of 100% actual level (20 ft. water depth) to bottom of the RWSP. The increase in RWSP level and in turn the increase in post-LOCA flood level in the containment required that certain instruments be raised to a higher elevation to prevent submergence. Calculation EC-M89-004 has determined that with the RWSP at 100% actual level, the maximum flood level in containment after a LOCA will be below +0.50 ft. MSL.

50.59 EVALUATION

The evaluation concludes that the proposed changes: (1) do not increase the probability of occurrence of any accident, since RWSP is not an accident initiator and there will be no change to the operation of instruments that will be raised, (2) do not increase the consequences of any accidents, since there will be no change in the operation of the instruments and the higher water level in the RWSP provides longer cold water injection into the containment that will result in a reduction in containment peak pressure and temperature, (3) the higher flood level calculated for actual 100% RWSP inventory will not impact the operation of any plant SSC important to safety and the Containment Fan Coolers (CFC) located in the containment. The maximum calculated flood level for the post-LOCA CFC performance is lower than the flood level assumed for post-LOCA CFC performance used in the containment analysis. The CFC blades are located above this calculated flood level. Therefore, the proposed change does not increase the probability of failure of any equipment important to safety, does not reduce the margin of safety as defined in the basis for any Technical Specifications, since the maximum RWSP level is not given in the Technical Specifications. Therefore, this evaluation concludes that no unreviewed safety question exist due to the proposed changes.

21. 2000-019; ER-W3-99-0851-00-00, Static Uninterruptible Power Supply (SUPS) 3A-S Replacement

DESCRIPTION OF CHANGE

This modification replaces the existing Class 1E safety-related ELGAR SUPS 3A-S and associated bypass constant voltage transformer located in the Switchgear Room "A". The rating of the SUPS unit and the bypass transformer has been increased. The new SUPS is equipped with an automatic static transfer switch (STS), which is an added design feature not available in the existing SUPS 3A-S. The STS is a zero-break, high-speed electronic, automatic, fail-safe transfer switch. It transfers power to the bypass transformer when the inverter experiences overload, overvoltage or undervoltage. The physical size and weight of the new SUPS unit is larger than the existing unit combined with the bypass transformer. Two new information circuit cables will be installed between Mux cabinet CAB RA0706 and SUPS 3A-S to indicate SUPS power source status. These cables will be non-safety related in application, however, are qualified for safety-related applications.

REASON FOR CHANGE

The inverters made by ELGAR are designed to shut down in microseconds when the output current reaches 165% of their rated current due to short circuit down stream of PDP 390-S. The down stream circuit breaker may not trip if a short circuit occurs and results in a complete loss of power to PDP390-SA. Additionally, bypass transformers provided by ELGAR for the vital AC SUPS are experiencing nuisance blowing of fuses on the transformer primaries. Also, when performing maintenance on SUPS 3A-S, electricians are exposed to energized circuits to install temporary bypass breakers to de-energize the inverter while maintaining power to the inverter loads. The installation of the temporary breakers require connecting cables in a small enclosure next to 480 volt power sources and presents a safety concern.

50.59 EVALUATION

The new SUPS and associated equipment meets or exceeds the same design, material, construction standards and specifications used for the original SUPS equipment. In a few cases where the performance characteristic did not exceed those of the old SUPS, these were evaluated and found to be acceptable. The instrumentation installed on the new SUPS meets or exceeds the accuracy and response characteristics of the original installed equipment being replaced. The replacement SUPS will not cause any system or component to be operated outside of their design, tested, or qualified limits. Although system interface is changed in that the new SUPS includes an automatic static bypass switch, this switch enhances the selective electrical protective features provided for the original design and does not increase the likelihood of any accident. The actions described or assumed in the FSAR accident analyses are not adversely affected. The conclusion previously made in evaluating the radiological consequences of an accident described in the FSAR remains unchanged. The replacement SUPS enhances vital AC reliability and does not adversely impact any system, structure or component considered important to safety. Fission product barriers are not degraded and the new SUPS does not adversely impact any reactor coolant pressure boundary or containment performance. No Technical Specification is changed and margins of safety are not adversely impacted. Thus the proposed change does not constitute an unreviewed safety question.

22. 2000-020; ER-W3-99-0184-01-00, Weld Repair of Inconel Instrument Nozzles on the Hot Legs

DESCRIPTION OF CHANGE

This change authorizes the weld repair of nineteen nozzles on the Reactor Coolant System hot legs, utilizing a welded "partial nozzle replacement" design similar to the repair of two nozzles on top of the pressurizer. The repair removes a portion of the Alloy 600 nozzle and installs a new Alloy 690 replacement nozzle. The repair also moves the partial penetration weld joint from the inside surface of the hot leg to the outside surface. This change authorizes the removal of the three Mechanical Nozzle Seal Assemblies installed on nozzles 1, 4 and 13 and replacement of the associated Nukon insulation blankets to restore the insulation seal around the nozzle areas.

REASON FOR CHANGE

During RF-9, three of the nineteen Alloy 600 instrument / sampling nozzles on the two Reactor Coolant System (RCS) hot legs were found to be leaking. The three leaking nozzles were repaired using Mechanical Nozzle Seal Assemblies (MNSA). These seals were classified as temporary repairs and they received NRC approval for use only until RF-10. In addition to these three nozzles, industry experience has shown that all Alloy 600 RCS nozzles, which are internally welded with partial penetration J welds, are subject to eventual leakage due to Primary Water Stress Corrosion Cracking, and a permanent repair is needed.

50.59 EVALUATION

This evaluation addresses the potential affects of the hot leg nozzle repairs during modification implementation, including low temperature overpressurization, rigging concerns, hydrogen gas accumulation, and the potential impact of temporarily removing portions of the instrument tubing connected to the nozzles. Also evaluated is the potential for the final design to affect the pressure boundary integrity, welding issues, thermal stresses, corrosion, erosion, stress corrosion cracking, hydrogen embrittlement, the maximum diametrical clearance allowed between the pipe bore and the nozzle outside diameter, and the potential affect on connected instrumentation. This evaluation concludes that the proposed change will not degrade the integrity of the RCS pressure boundary or the functional capability of the affected instrumentation. All changes are within the Reactor Containment Building, and there are no new system interactions created. There are no unreveiwed safety questions and this change does not require any Technical Specification changes.

23. 2000-021; ER-W3-99-3550-00-00, Permanent Changes to Condenser Off Gas Discharge

DESCRIPTION OF CHANGE

This change will implement the permanent changes required to disable the automatic transfer of the Air Evacuation pump exhaust from atmosphere to the E-22 fans and the charcoal filters. The proposed change disables the automatic connection to Reactor Auxiliary Building Normal Ventilation, preventing the automatic filtered release via the plant stack.

REASON FOR CHANGE

The condenser vacuum pump exhaust is diverted from the atmosphere to the Reactor Auxiliary Building normal ventilation in the event elevated radioactivity levels are detected due to a postulated primary to secondary leakage potentially resulting from a steam generator tube rupture (SGTR). A sufficiently large SGTR could potentially result in the generation of a safety injection actuation signal (SIAS) from decreasing pressurizer pressure. Reactor Auxiliary Building normal ventilation is designed such that the E-22 ventilation fans are tripped upon an SIAS and are unavailable post-accident. The simultaneous occurrence of diverted condenser gas in the presence of a SIAS could result in the condenser vacuum pumps forcing radioactive gases and vapors throughout the Reactor Auxiliary Building non-safety ductwork. This change will disable the automatic transfer function of the Main Condenser Evacuation System to Reactor Auxiliary Building Normal Ventilation, preventing the automatic filtered release via the plant stack and pumping the Main Condenser Evacuation System gases throughout the Reactor Auxiliary Building.

50.59 EVALUATION

The radiological consequences of disabling the automatic diversion and filtration of condenser vacuum pump exhaust has no impact on design basis accidents since the Accident Analysis of the FSAR does not credit the Air Evacuation exhaust via the Reactor Auxiliary Building normal ventilation. Normal off-site doses will remain within the prescribed limits. The changes implemented remove the automatic diversion and filtration feature permanently. Since the impacted equipment will be placed in their fail-safe positions, no loss of safety can result. The fission product barriers are not impacted. The disabling of automatic filtration of condenser vacuum pump exhaust does not impact any accident initiators or equipment used to safely shutdown the plant or to mitigate the consequences of an accident. The automatic diversion is not relied upon for any safety analysis and its removal does not constitute an unreviewed safety question.

24. 2000-023-2; ER-W3-99-0736-00-02, Generic Letter 96-06 Reactor Containment Building Penetrations Overpressurization

DESCRIPTION OF CHANGE

Changes are made to twelve piping penetrations to ensure that thermally induced overpressurization due to plant accident conditions does not jeopardize the integrity of the containment isolation system. The changes made include physical changes to piping and plant (adding relief valves), interim administrative control (ensures penetration is flushed with fluid greater than 260°F), administrative procedure control (drains or vents pipe segments and controls valve lineups) and insulation replacement on penetration #44.

REASON FOR CHANGE

Generic Letter 96-06 requested licensees to determine whether or not piping systems that penetrate the containment are susceptible to thermal expansion of fluid between the inboard and outboard isolation valves, such that over-pressurization of the piping could occur. Evaluation of all containment piping penetrations determined that seventeen are of concern for this type of over-pressurization. This change addresses the corrective actions for 12 of the affected penetrations.

50.59 EVALUATION

The proposed change does not cause the system(s) to be operated outside of their design or test limits, negatively affect any system interfaces, or result in an increase in challenges to safety or important to safety systems. The changes do not affect the initiating mechanisms for any accident previously evaluated in the FSAR. Adding overpressure protection (relief valves) to the penetrations ensures that the penetrations are not negatively impacted during the occurrence of a design basis accident. All thermal relief valves are added inside the containment and are classified as containment isolation valves. The design, material and construction of the relief valves and associated piping are in accordance with ASME Section III, Class 2 and meet the standards applicable to the existing piping system and components. Acceptance testing and periodic tests and inspections are in accordance with applicable Codes and Standards. Therefore, introduction of the relief valves and associated piping does not increase the probability of occurrence of an accident previously evaluated in the FSAR. The penetrations with the interim administrative controls to ensure that the piping is flushed with high temperature fluid removes an initiating condition of the event. Because the quantity of water trapped in the isolated piping was based on the specific volume corresponding to the elevated temperature, any heatup of the penetration in an accident scenario will only return the trapped fluid to the specific volume that it had when the penetration was isolated. This precludes the penetration from experiencing overpressure conditions. In addition, the piping between the isolation valves is insulated which will minimize heat loss during Mode changes. The low pressure penetrations with the permanent administrative controls to assure that the penetrations are isolated and drained or vented provide overpressure protection by replacing the incompressible fluid in the penetration with air which is much more compressible. Heatup of the trapped air from ambient temperature to the elevated post accident temperature will not yield overpressure conditions. Only a relatively small amount of the fluid in the penetration needs to be replaced with air to provide sufficient elasticity to protect the penetration, so draining the penetration provides adequate assurance that the penetration will not experience overpressure conditions. The changes do not increase the radiological releases postulated to occur in a design basis accident scenario. The systems which mitigate the consequences of an accident are not negatively affected by these changes. The fluid released out of the penetration when the relief valve opens is a very small quantity volumetrically and this has no impact on containment environment or flooding. If the relief valve should fail to reseat in the accident scenario, it will not provide a constant addition of fluid to the containment because the process flow through the containment penetrations is isolated. Additionally, the relief valves will be added to the containment leakage rate testing program to ensure the total containment leakage will not exceed the values assumed in the safety analyses. Therefore, the additions of the relief valves do not increase the consequences of an accident previously evaluated in the FSAR. In the unlikely event that a relief valve opens during normal plant operations, the resulting condition is bounded by existing accident analyses. Administrative controls and adding overpressure protection devices does not affect the design bases or operation of any other equipment important to safety. The relief valves are oriented and installed such that the discharge does not impact any other equipment important to safety. Consequently, the proposed change does not represent an unreviewed safety question or a change to the Technical Specifications.

25. 2000-025; ER-W3-98-0642-02-00, Component Cooling Water Makeup Single Failure Modification

DESCRIPTION OF CHANGE

Component Cooling Water (CCW) Surge tank level switches setpoints will be re-structured by relocating the wire connections from one level switch to another level switch at a different elevation on the surge tank. The subject design change will also remove the automatic CCW Makeup pump start signals from the Emergency Diesel Generator Jacket Water Standpipe and the Essential Chilled Water Expansion Tanks. Makeup to these tanks will require manual operator action if the non-safety condensate makeup system is unavailable or unable to maintain level in the respective tanks.

REASON FOR CHANGE

The CCW Makeup pumps take suction from the Condensate Storage Pool (CSP) and are capable of providing makeup to the Component Cooling Water (CCW) system, the Emergency Diesel Generator (EDG) Jacket Water Standpipes, and the Essential Chilled Water (CHW) Expansion Tanks. The respective makeup pump starts when level in the CCW surge tank, EDG standpipe or chilled water expansion tank reaches a predetermined low level setpoint. Another low level setpoint opens the CCW makeup control valve to the respective tank. Once a tank reaches a predetermined high level setpoint, the CCW makeup isolation valve closes, securing makeup to the respective tank. The makeup pump continues to run until secured by plant operators. If a Loss of Offsite Power occurs, the CCW Makeup Pumps will start after they are sequenced onto the safety buses if a level demand is present. A single failure of any CCW makeup isolation valve in the open position would result in the CCW makeup system continuously providing makeup to their respective tanks. Since CCW Makeup Pumps take suction from the Condensate Storage Pool (CSP), continuous makeup would result in a decrease of CSP inventory. The CSP inventory is credited for the Emergency Feed Water (EFW) system in response to a design basis accident. Also the CCW surge tank, EDG jacket Water standpipes, and the CHW Expansion Tanks are each vented to atmosphere where continuous makeup could result in flooding those areas in the Reactor Auxiliary Building and possibly impact safety related equipment.

50.59 EVALUATION

This evaluation shows that the proposed changes will not create any unreviewed safety questions. The proposed changes will ensure that CSP inventory (Technical Specification Basis 3/4.7.1.3) for the EFW system is preserved, and the affected systems (CCW, CHW and EDG) will still provide accident mitigating functions as required. The new manual actions for providing CCW makeup to the EDG Jacket Water standpipe and the CHW Expansion Tanks has been evaluated using the guidance of NRC Information Notice 97-78 and found acceptable.

26. 2000-031; ER-W3-00-0210-00-00, Instrument Air Backup Compressor

DESCRIPTION OF CHANGE

Install a temporary air compressor as a source of backup air supply to the Instrument Air system. The compressor discharge will be attached at the inlet to Instrument Air Dryer B Prefilters where air can flow to Instrument Air Dryer B and through the Instrument Air Receiver to Instrument Air Dryer A then to the Instrument Air system after it is filtered and dried.

REASON FOR CHANGE

This Temporary Alteration will provide backup capacity to the Instrument Air system while the Station Air (SA) system is out of service during testing and implementation of design change DC-3390, Instrument Air/Station Air Enhancements.

50.59 EVALUATION

The Instrument Air system is not safety-related and is not required for safe shutdown of the plant or for limiting radiological releases. The temporary installation of a backup air compressor for the Instrument Air system does not result in an unreviewed safety question. The Instrument Air system does provide air to numerous safety-related valves in the plant. The safety related valves which are supplied with Instrument Air fail safe, or are provided with safety related backup air or nitrogen accumulators. The addition of a backup compressor only enhances the ability of the Instrument Air system to maintain normal pressure and capacity. This temporary alteration does not involve a change to a protective boundary and will not indirectly have an adverse affect on a boundary. No margins of safety will be directly or indirectly affected. No acceptance limits or safety limits will be exceeded as a result of this temporary alteration.

27. 2000-034; ER-W3-98-1331-01-00, Re-Route Main Condenser Air Evacuation Pump Drains to the Main Condenser

DESCRIPTION OF CHANGE

This change re-routes the condensate discharge of the condenser vacuum pump moisture separators. The separator condensate (seal water) is presently discharged into the 40 Arpent canal by way of the industrial waste sump followed by the oil separator sump. This proposed configuration change will return the separator condensate back to the condensers.

REASON FOR CHANGE

The Main Condenser air evacuation pumps discharge into moisture separators. The condensate overflow from the moisture separators is drained to the Industrial Waste Sump. The pH of the air evacuation condensate is high due to the content of ammonium hydroxide. The Industrial Waste Sump is pumped to the Oil Separator Sump. The contents of the Oil Separator Sump is sampled and acid is added if required to lower the pH prior to discharging the contents to the 40 Arpent Canal. Oil Separator Sump discharge to the canal is no longer an automatic function. Re-routing the separator condensate to the main condenser will eliminate the time involved to measure and correct the pH at the Oil Separator Sump.

50.59 EVALUATION

This evaluation determined that there were no unreviewed safety questions. The worst case scenario associated with this plant configuration change was a condensate return line break or stuck open float valve. Both of these failure modes could potentially result in a loss of condenser vacuum which is classified as moderate frequency incident in FSAR Section 15.2.1.2. The FSAR evaluation and the one performed in FSAR Section 15.2.2.3 which included a single failure of an active component assumed the complete loss of condenser vacuum by three postulated failure modes. A float trap which has become stuck in the open position would result in condenser vacuum degradation, however, the consequence of the incident would not be more severe than the conditions evaluated in the FSAR.

28. 2000-035; ER-W3-00-0157-00-00, Temporary Air Conditioning for Turbine Building

DESCRIPTION OF CHANGE

This change will add temporary air conditioning to the turbine building switchgear during the summer of 2000. The normal ventilation system will be secured and portable air conditioning units will provide cooling to the area utilizing temporary ducting.

REASON FOR CHANGE

The reason for this temporary alteration is to increase the reliability of the turbine building switchgear equipment during the hot summer months. Currently, only ventilation is provided to the switchgear with no cooling. Trending of past summer temperatures in the switchgear area revealed peak room temperatures reached as high as 103°F. In addition, the existing configuration requires a large volume of air to remove heat from the switchgear. This massive quantity of outside air brought into the switchgear has caused problems with excessive particulates in the area. This temporary alteration is sized to maintain the switchgear at a temperature of approximately 83°F during anticipated peak load periods. Furthermore, the temporary ventilation will be 100% recirculated air to limit the introduction of particulates into the area.

50.59 EVALUATION

Adding the temporary air conditioning during the summer will increase the reliability of equipment in the turbine building switchgear and will not adversely affect safety related equipment or equipment important to safety. The turbine building ventilation system is a non-safety related system that is not postulated to initiate any accident previously evaluated in the FSAR. The FSAR states that any failure of the turbine building ventilation system will not affect safety related structures, systems or components. This section further states that turbine building ventilation is not required to mitigate the consequences of an accident or to provide safe shutdown of the reactor. Therefore, this change does not increase the consequences of any accident previously evaluated in the FSAR. This change does not create the possibility of an accident of a different type than any previously evaluated in the FSAR. This change will improve the reliability of the turbine building switchgear by providing a more suitable environment for the equipment. This temporary alteration does not interface with equipment other than that located within or adjacent to the turbine building switchgear. Nevertheless, any possible problem with equipment in the switchgear area could potentially result in a loss of power to the non-safety related plant equipment powered from the turbine building switchgear. Any such event would be bounded by the previously evaluated Loss of Offsite Power. In addition the permanent turbine building switchgear ventilation system will not be altered by this temporary change. This change will secure the normal turbine building switchgear ventilation system by using the existing control switch once the temporary air conditioning is in service. The permanent ventilation system will still be available for operation if desired. However, for the duration of the Temporary Alteration the permanent turbine building switchgear ventilation system will not be operated as described in the FSAR unless required. There are no Technical Specifications relating to turbine building ventilation. Furthermore, turbine building ventilation is not relied upon in any Technical Specification basis defining margin of safety.

29. 2000-036; ER-W3-00-0487-00-00, Temporary Alteration to Re-wire Reactor Coolant Temperature Element

DESCRIPTION OF CHANGE

Temperature element RC ITE0121X is replacing Core Protection Calculator channel "D" RTD element that has failed. The RC ITE0121X element is located on the same side of the hot leg #2, approximately 20 inches upstream from the failed RC ITE0122HD1. The RC ITE0121X element is the same model and was purchased safety related, 1E for installation in either application. Instrument loop RC ITE0121X provides input to CP-2 indication, CP-19 Control Room Annunciator, the Plant Monitoring Computer and CP-12 Reactor Regulating System. This instrument loop control and display functions are non-safety, and provide no accident mitigation function.

REASON FOR CHANGE

Temperature Element RC ITE0122HD1 has failed. Instrument loop RC IT0122HD provides input to Core Protection Calculators Channel D, the plant monitoring computer and indication on CP-7. The core protection calculator, which is part of the plant protection system as defined in Chapter 7.2 of the FSAR, provides pre-trip and trip information to the Reactor Protection System and to the operator in the control room. The Reactor Protection System provides the necessary reactor trips to assist in limiting the consequences for design basis accidents and is a safety-related system. Hot leg temperature is used to calculate thermal power, calorimetric flow (used to calibrate Core Operating Limits Supervisory System and Core Protection Calculator flow), calculating the hot leg temperature correction factor in CPC FLOW algorithm and calculating hot leg temperature as part of the hot leg saturation auxiliary trip function in Core Protection Calculator. These functions are safety related. The RTD is located inside the containment on hot leg piping, which is not accessible during the operation of the plant. Although the failed Core Protection Calculator channel is placed into bypass, which is the safe position, and in accordance with the Technical Specifications of the plant, it is desirable to find a suitable replacement indication for the temperature loop to restore the channel to operation.

50.59 EVALUATION

The effect of this temporary alteration on Core Protection Calculator Channel D, Core Operating Limits Supervisory System and Reactor Protection System are negligible. The RTD or temperature indicating loop is not an initiator of any accident, thus this temporary alteration will not increase the occurrence probability of an accident previously evaluated in the FSAR. Since the replacement element is the same design and is intended for the same application, there is no increase in the malfunction occurrence probability for this equipment. This temporary alteration does not affect the safety of the plant or the ability of the plant to achieve a safe shutdown following a Design Basis Accident.

30. 2000-037; ER-W3-00-0220-00-00, Shield Building Maintenance Hatch Door Seals Temporary Air

DESCRIPTION OF CHANGE

A temporary mechanical jumper (air hose and metering valve) will be installed from a spare connection on the Instrument Air system to a connection on a portion of the Station Air system to provide air to the shield building maintenance hatch door seals while the Station Air system is depressurized. Valve SA-7021 which is upstream of SA-7022 will be closed to prevent the instrument air system from pressurizing the station air system.

REASON FOR CHANGE

During implementation of ER-W3-00-0315-00-00, "Station Air to Instrument Air Equipment Reliability Modification", the Station Air system will be depressurized for some period of time. Additionally, this temporary alteration may be used to support STP-289682, "Instrument Air System Leakage Test" while the Station Air is depressurized. The Station Air system normally provides air to keep the shield building maintenance hatch door seals inflated. Instrument Air may be used to keep the seals pressurized while the Station Air system is unavailable.

50.59 EVALUATION

The Instrument Air and Station Air systems serve no safety functions since they are not required to achieve safe shutdown or to mitigate the consequences of an accident. Providing Instrument Air to the shield building maintenance hatch door seals will help ensure that the seals can remain pressurized for 7 days following a small break LOCA with a loss of Instrument Air. Should the temporary hose break, the maximum break flow is 10 SCFM which does not increase the probability of an accident. This temporary alteration will not affect the safety or environmental aspects of the FSAR and there are no unreviewed safety questions.

31. 2000-038; ER-W3-00-0042-03-00, Modification to the Blowdown System to Support Steam Generator Chemical Cleaning

DESCRIPTION OF CHANGE

This change will install a new 2" diameter flanged branch connection on steam generator blowdown lines. The new branch connections will be located immediately downstream of manual isolation valves, approximately 30" above the -4 Reactor Auxiliary Building floor elevation. Each of the two new branch connections will consist of a 3000 lb. weldolet, a 600 lb. weld neck flange and a 600 lb. blind flange.

REASON FOR CHANGE

Chemical cleaning of the Steam Generators is planned for the beginning of the Refuel 10 plant outage. Hoses will be used to route chemicals to and from the steam generators, and branch connections are needed on each train of the Blowdown System for connecting the hoses. Flanged connections are provided inside the containment building for chemical cleaning, but they are inaccessible because the cleaning is scheduled to begin in Mode 4 when containment integrity is required by Technical Specification 3/4.6.1.

50.59 EVALUATION

This evaluation addresses the potential affects of modification implementation and plant operation. This evaluation concludes that the proposed changes will not degrade the integrity of the Blowdown System pressure boundary or the functional capability of the system. All changes are on the non-safety related portions of the Blowdown System, located within the Reactor Auxiliary Building Wing Area, and there are no new system interactions created. There are no new unreviewed safety questions, and this change does not require any Technical Specification changes. The proposed changes to install two new flanged branch connections on the non-safety related portions of the Blowdown System will not increase the consequences of an accident previously evaluated in the FSAR because the safety related portions of the system are isolated by closure of the containment isolation valves if significant activity is detected, and during other events requiring containment isolation. The non-safety related portions of the Blowdown System are not credited for accident mitigation, although that portion of the system is credited for containing lesser amounts of radioactive materials, as described in the FSAR. As a consequence of allowable steam generator tube leakage, the Blowdown System would concentrate smaller amounts of radioactivity in the blowdown filters and demineralizer resins.

32. 2000-040; ER-W3-00-0315-00-00, Station Air to Instrument Air Equipment Reliability Modification

DESCRIPTION OF CHANGE

The proposed change will provide flanged connections for temporary air compressor hook-ups and modify the interconnection between the Station Air and Instrument Air. An additional cross-tie from Station Air will connect just upstream of the Instrument Air dryer assemblies to provide the capability of doing maintenance on the Instrument Air major components while Station Air fully supports the Instrument Air system demands. The internals of Station Air to Instrument Air pressure control valve SA-125 will be changed in order to decrease the pressure drop across the valve and increase airflow. This will provide added capacity for the Station Air to supplement the Instrument Air system during normal operations. These modifications will provide the capability of doing maintenance on the Instrument Air major components (i.e., compressors, discharge valves, air receiver, etc.) while the Station Air/Instrument Air crosstie supports the Instrument Air system, and will increase the reliability, maintainability, and versatility of the compressed air systems.

REASON FOR CHANGE

The existing compressed air systems are not designed for system outages in order to perform maintenance tasks on main components. At this time it is not possible to take the Instrument Air system out of service in order to do maintenance on components such as the air receiver. The lack of flexibility is the cause for the compressed air systems being considered a high reliability risk. There are also no provisions for temporary air compressors to be connected to the compressed air systems in case an overall system outage is required.

50.59 EVALUATION

The proposed modifications to the Instrument Air and Station Air systems will not have an affect on any accident previously evaluated in the FSAR, nor will it have an affect on the consequences of an accident previously evaluated in the FSAR. The compressed air systems are non-safety, non-seismic, are not needed for the safe shutdown of the reactor or to mitigate the consequences of an accident. During normal plant operation, the Instrument Air system supplies pressure to certain safety related air operated valve accumulators. Equipment which requires compressed air to perform their post accident safety related functions, are equipped with safety related air accumulators. There are no accidents that are initiated by the Instrument Air or Station Air systems. This modification will not affect the safety or environmental aspects of the FSAR documents, and there are no unreviewed safety questions.

33. 2000-042; ER-W3-00-0486-00-00, Condenser Vacuum Breaker Plug

DESCRIPTION OF CHANGE

A plug will be installed in the standpipe for Condenser Vacuum Breaker Valve AE-401A.

REASON FOR CHANGE

Condenser A Vacuum Breaker Valve is leaking excessively past its seat. This valve cannot be isolated at power. Currently demineralized water is being continuously added to the standpipe to makeup for water leaking through the valve in order to prevent sucking air into the condenser and increasing the amount of dissolved oxygen in the condensate.

50.59 EVALUATION

The installation of a pipe plug in the standpipe for AE-401A will functionally disable the valve. It cannot be used to break condenser vacuum until the plug is removed. The vacuum breaker valves for condenser B and C may be used to break condenser vacuum. The condenser air evacuation system has no safety function. The ability to break condenser vacuum is not relied upon in the FSAR accident analysis. Disabling AE-401A does not result in a reduction in any safety margins.

34. 2000-043; ER-W3-99-1118-00-00, Add Additional Radiation Monitoring to the Condenser Off Gas System

DESCRIPTION OF CHANGE

A particulate, iodine, gas (PIG) radiation monitor, sample dryer, and sample pump will be connected to the Air Evacuation pumps discharge to monitor and trend low level primary to secondary leakage. The Drumming Station PIG, ARM-IR-5146 will be relocated from the Hot Tool Room hallway to the Air Evacuation pumps discharge located in the Turbine Building.

REASON FOR CHANGE

This temporary alteration provides a means of continuous monitoring and alarming for low levels of primary to secondary leakage and rate of change as required by Nuclear Energy Institute (NEI 97-06) in accordance with EPRI Primary To Secondary Guidelines, EPRI TR-104788.

50.59 EVALUATION

Loss of Condenser Vacuum and indirectly a Turbine trip are initiated by the air evacuation system. However, installation of the new radiation monitor in the discharge of the air evacuation pumps will not increase the probability of accident occurrence. The monitor will draw a small sample from one air evacuation pump discharge line, and return the sample to the discharge of the remaining air evacuation pumps. Installation and operation of the new monitor will not affect the performance or reliability of the Air Evacuation system and will not require the Air Evacuation system to be operated in any modes outside its design limits. The addition of the sample lines to the drain line off the silencer is passive and therefore will not affect the operation of the Air Evacuation system. The Drumming Station PIG monitor is not needed in its current location due to drumming activities not occurring. The sample for the radiation monitor is being taken from the air evacuation discharge pumps B and C and being discharged to the discharge line of air evacuation pump A. This will allow the discharge from the new monitor to be rerouted through the Condenser Off-gas Wide Range Gas Monitor, before being released through the plant stack. No equipment used to mitigate the consequences of an accident is affected. The removal of the Drumming Station PIG will not increase the consequences of an accident previously evaluated in the FSAR since the monitor is not credited with mitigation of an accident. The new monitor will not replace existing effluent monitors but will provide additional monitoring capability. The new monitor will interface with the Air Evacuation system, the Plant Monitoring Computer and the non-1E electrical system. These systems are not important to safety. The change does not affect any boundary performance parameters as defined in the Technical Specifications.

35. 2000-044; ER-W3-00-0450-00-00, Revision of Technical Requirements Manual Table 3.8-1 to Make Corrections

DESCRIPTION OF CHANGE

This change revises the Technical Requirements Manual (TRM) Table 3.8-1 to incorporate the addition of overcurrent penetration protection for CAR-ISV-200B, the deletion of the overcurrent penetration protection for SSL-ISV-8001A, 1B, 2A & 2B, and correct a typographical error for Safety Injection Tank 1A circuit identification.

REASON FOR CHANGE

Corrective Action Item 6 for Condition Report 99-0844 requires Design Engineering to revise TRM Table 3.8-1 to incorporate the overcurrent penetration protective device for CAR-ISV-200B. In addition, a comprehensive review of all electrical penetrations was conducted to verify the accuracy of the TRM Table.

50.59 EVALUATION

The change to the Technical Requirements Manual does not create an Unreviewed Safety Question. These changes document the addition and omission of protective devices to TRM Table 3.8-1. The required penetration protection existed, however, the appropriate TRM document was not revised to reflect the as built configuration. The devices that were omitted from the table were adequately tested for compliance with the TRM. The devices that are being deleted from the TRM Table do not perform a penetration protection function and hence do not need to be tested per TRM requirements. This change is a documentation change. The change does not represent any physical changes to the plant or its equipment. The power supply (120 Vac, PDP390A, CKT 18) for CAR-ISV-200B meets all the requirements specified in Appendix J and is an acceptable containment penetration overcurrent protective device. The devices that are added to the TRM were tested per an approved Maintenance procedure. Although the procedure was not the specified procedure, the process was the same. The typographical change in the TRM Table reflected the incorrect circuit number identification for Safety Injection Tank 1A. The deletion of the protective devices was a result of the comprehensive review of the TRM Table. The deleted devices are not protecting any containment penetrations.

36. 2000-047; ER-W3-00-0350-00-00, Installation of Bullet Resistant Enclosures

DESCRIPTION OF CHANGE

This change adds Bullet Resistant Enclosures (BREs) on the roofs of the Reactor Auxiliary Building, Turbine Generator Building, and Condensate Polisher Building. BREs are bullet resistant security observation towers, which are to be used in the defense of the plant against acts of aggression. This change includes the addition of plant paging, lighting, telephones, computers and air conditioning to the BREs. Each BRE is an approximately 8' x 10' x 9' high, steel structure. The BREs are to be secured to the Turbine Generator Building northeast corner; Reactor Auxiliary Building West Diesel Tank Room and Condensate Polisher Building southwest corner roofs via steel anchors.

REASON FOR CHANGE

These structures are being added as part of site security enhancements required by NRC Inspection Report 50-383/99-017.

50.59 EVALUATION

This change adds Bullet Resistant Enclosures (BREs) with paging, lighting, telephones and computers, on the roofs of the Reactor Auxiliary Building, Turbine Generator Building and Condensate Polisher Building. The Turbine Generator Building and Reactor Auxiliary Building roofs have been structurally evaluated and found to be adequate to support the BRE (the Condensate Polisher Building roof requires additional steel per this change). All structures or components added by this change are non-safety related. In addition, the systems affected by this change are non-safety related and are not directly involved in any accident scenarios (paging is required during an accident per the Emergency Plan, but does not cause any accidents, nor is it needed for safe shutdown. In addition, the cable/conduit routed per this change will not be tied into the existing plant paging system). The BREs are located such that if they collapse during a seismic event, they would not affect any safety-related structure, system or component. The BRE could potentially become airborne missiles under extreme loading. However, they are very similar to missiles already evaluated in the FSAR. Adding fire protection to the BRE was evaluated, but determined not to be necessary. Defensive positions are not addressed in the Physical Security Plan (PSP). Based on the above discussion, this change does not result in any unreviewed safety question or change to the Technical Specifications.

37. 2000-048; ER-W3-00-0243-00-00, Temporary Power to Power Distribution Panel PDP-390 during SUPS SA Replacement

DESCRIPTION OF CHANGE

SUPS SA will be replaced during RF10 with a unit manufactured by Solid-state Controls, Inc. During this replacement, normal supply power from SUPS SA and bypass supply power from the existing transformer to PDP-390 will not be available. In order to ensure the availability of plant equipment supplied by SUPS SA, a temporary alteration will be installed to provide an alternate source of supply power to PDP-390. The alternate source of power will be from a spare breaker installed in MCC 3A311 Cubicle.

REASON FOR CHANGE

This temporary alteration is required to provide an alternate source of power to PDP-390 during SUPS SA replacement.

50.59 EVALUATION

SUPS 3A-S provides uninterruptible 120 VAC power to safety related loads required to safely shut down the reactor and maintain the reactor in a safe shutdown condition. Power Distribution Panel PDP 390-SA is used to feed a variety of safety related loads including solenoid valves, radiation monitors, control panels and instrumentation. There are no specific analyzed accidents that are initiated by the failure of SUPS 3A-S or PDP 390-SA. FSAR sections analyze accidents associated with the Loss of Normal AC Power. Other accidents evaluated in the FSAR require the components supplied by PDP 390-SA for mitigation. However, the initial conditions for the evaluated accidents are that the plant is operating at power. This temporary alteration will provide an alternate source of power for PDP 390-SA during modes 5 and 6 with the plant on shutdown cooling. During modes 5 and 6, Technical Specifications require only one 120 volt AC SUPS bus energized from its associated inverter connected to its respective bus. During SUPS 3A-S replacement, Train B will be the protected train. All requirements of Technical Specifications will be met by having the B Train components operable and in service. Waterford 3 is able to obtain and maintain shutdown cooling conditions with only one train in operation. Additionally providing an alternate source of power to PDP-390-SA ensures the availability of the A Train components for shutdown cooling requirements. Since the temporary alteration will only be installed in Modes 5 and 6, the consequences of a loss of offsite power are bounded by conditions currently allowed by Technical Specifications. The potential consequences of a loss of shutdown cooling are heat up of the Reactor Coolant System, boil off of RCS inventory and (in the absence of make-up) core uncovery and damage. These potential consequences are unchanged by the installation of this temporary alteration. The temporary transformer and circuit breaker have been sized in accordance with the National Electric Code to protect the PDP and cabling. Existing load circuit breakers will protect the individual loads supplied by the PDP. The temporary transformer and circuit breaker will be located in Switchgear Room A to prevent train interaction. PDP-390 undervoltage and ground indication will be provided in the control room to alert Operations of equipment failure. This evaluation has concluded that providing a temporary source of power to PDP-390 during SUPS SA replacement in Modes 5 and 6 will not have an adverse affect on the ability to maintain shutdown cooling or reactivity control. No unreviewed safety question exists.

38. 2000-054; ER-W3-00-0814-00-00, Correction to FSAR Table 9.4-5

DESCRIPTION OF CHANGE

This change will correct FSAR Table 9.4-5, "Fuel Handling Building Ventilation System Failure Modes and Effects Analysis". The table currently groups all of the Fuel Handling Building Ventilation Isolation and Bypass Dampers together as isolation dampers. However, the dampers should be grouped into two categories. The table incorrectly describes dampers D-36A & B as isolation dampers, rather than bypass dampers. This change will separate the list of isolation dampers in Table 9.4-5 into two separate categories.

REASON FOR CHANGE

The reason for changing FSAR Table 9.4-5 is to correct the FSAR to reflect the system's as-built configuration. This change will ensure that FSAR Table 9.4-5 agrees with the existing Fuel Handling Building Ventilation System configuration and the FSAR system description.

50.59 EVALUATION

This change is limited to the bypass dampers for the Fuel Handling Building Ventilation System. These dampers are required to actuate to their safety related positions following a Fuel Handling Accident signal. The bypass dampers open to ensure that the non-contaminated portions of the Fuel Handling Building are maintained positively pressurized relative to the contaminated areas in the event that the normal ventilation system is started following a Fuel Handling Accident. These dampers are required to mitigate the consequences of a Fuel Handling Accident in order to limit the potential offsite exposures within the acceptable limits of 10CFR100. However, these components do not have any functions that would prevent or potentially cause a Fuel Handling Accident to occur. Furthermore, this change will only correct FSAR Table 9.4-5 to describe the as-built configuration of the Fuel Handling Building Ventilation System. The only accident that requires operation of the HVF Bypass Dampers is a Fuel Handling Accident. Currently, FSAR Table 9.4-5 incorrectly states that bypass dampers D-36A(B) have a failure mode of "fails to close" and a remark stating "redundant damper in series will close". However, during a Fuel Handling Accident dampers D-36A(B) are required to open. Furthermore, these two dampers are arranged in parallel rather than series. This function is in agreement with Waterford 3's Design Basis information. FSAR Section 9.4.2.2.2 and Figure 9.4-2 also reflect the correct function of the bypass dampers which is to open and re-route airflow from the normal supply fan following a fuel handling accident if the operator chooses to start the normal ventilation system following a fuel handling accident.

39. 2000-055; ER-W3-00-0753-00-00, Core Operating Limits Report, Rev. 4

DESCRIPTION OF CHANGE

The ranges of Critical Boron Concentration (CBC) provided in Section 3.1.1.3 and applicable to Figures 2A, B &C of the Core Operating Limits Report (COLR) were revised based on information provided in ER-W3-00-0753-00-00. The range of applicability for Figure 2A was changed from 153 ppm ≥ CBC>100 ppm to 153 ppm ≥ CBC>111 ppm. The range of applicability for Figure 2B was changed from 100 ppm ≥ CBC > 50 ppm to 111 ppm ≥ CBC > 56 ppm. The range of applicability for Figure 2C was changed from CBC ≤ 50 ppm to CBC ≤ 56 ppm.

REASON FOR CHANGE

COLR Figures 2A, B & C were provided to restrict reactor operation near the end of Cycle 10 when Critical Boron Concentration (CBC) is less than 153 ppm. Operation within the bounds of these COLR figures is necessary to ensure that Moderator Temperature Coefficient (MTC) limitations assumed in the safety analyses for Cycle 10 are not exceeded. Each figure is valid for a range of boron concentrations and plant operators are required to use one of the three figures depending on the current measured RCS boron concentration. The figures are based on boron containing the isotope B-10 in an atomic abundance of 19.8%. CR-W3-2000-1061 documented that a Reactor Coolant System boron sample taken on 9/4/00 was analyzed and found to contain an atomic abundance of B-10 equal to 18.7%. As given in ER-W3-00-0753-00-00, by the end of Cycle 10 continued depletion of B-10 is expected to result in an atomic abundance as low as 17.95%. While boron measurement uncertainty was accounted for in the COLR figures, depletion of the B-10 isotope was not considered. Thus, the effectiveness of the RCS boron as a poison is less than that assumed in the development of the COLR figures. Therefore, the CBC ranges given in COLR Section 3.1.1.3 and Figures 2A, B & C have been adjusted so that plant operators will implement the operational restrictions given in the COLR figures at higher measured RCS boron concentrations. The corrected CBC ranges will ensure that operational restrictions are employed when the abundance of B-10, the primary neutron absorber, is consistent with that assumed in the safety analyses and presented in the COLR figures.

50.59 EVALUATION

Adjustment of the CBC ranges given in COLR Section 3.1.1.3 and Figures 2A, B & C ensures that the operating restrictions necessary to avoid exceeding Moderator Temperature Coefficient limitations near the end of Cycle 10 are implemented consistent with safety analyses assumptions. Neither plant response to postulated accidents or accident consequences are affected by this change to the COLR and the margin of safety is not reduced. Hence, no unreviewed safety questions exist.

40. 2000-058; ER-W3-99-0966-00-00, Temporary Reactor Vessel Head

DESCRIPTION OF CHANGE

This change allows a temporary reactor head (TRH) to seal the reactor vessel from the refueling cavity after the reactor, upper guide structure, and fuel are removed. The temporary reactor head will normally be stored in the lower end of the refueling cavity. To be able to place the temporary reactor head, the stairs made of scaffolding and a radiation shield that are currently kept in the lower end of the cavity will be relocated. The stairs and the radiation shield will now be kept adjacent to the permanent stairs in the northwest quadrant of the Reactor Containment Building.

REASON FOR CHANGE

During RF-10, permanent weld repairs are to be made on several small nozzles on the hot legs. The nozzles being repaired are at or below the mid-loop elevation of the pipes. In order to perform these repairs, the Reactor Coolant System hot legs will have to be completely drained. To drain the hot legs, the water level in the reactor will have to be lowered below the Reactor Coolant System nozzles in the reactor itself. The lowered water level in the reactor means that the upper level of the refueling cavity will have to be kept dry and the upper portions of the core barrel and the Upper Guide Structure will be exposed. This results in increased dose rates on the +46 elevation of the Reactor Containment Building.

50.59 EVALUATION

The Temporary Reactor Head (TRH) will be stored in the lower end of the refueling cavity until needed. The storage legs for the TRH are designed for seismic loading. The elevation of the lower end of the refueling cavity is above the maximum internal flood elevation in the Reactor Containment Building (RCB). Therefore, the placement of the TRH in the lower end of the refueling cavity will not raise the flood level in the RCB. The scaffolding stairs and radiation shield will be relocated to the east side of the permanent stairs in the northwest quadrant of the RCB and will be tied to the permanent stairs to prevent movement during a seismic event. These locations were reviewed for jet impingement, and no jet forces were found that would strike the stairs and/or the radiation shield. The TRH can only be used when the reactor is completely defueled, and there are no assemblies in the temporary storage racks, in the upender, or the fuel transfer carriage on either end of the transfer tube. The TRH is designed to remain in place on the reactor if there is a seismic event. This change is also requiring that the Fuel Transfer Tube Isolation Valve be closed prior to lowering the water level in the reactor. By closing this valve, and with all fuel assemblies removed from the RCB, the water level in the Spent Fuel Pool is completely isolated from the water level in the RCB. The movement of the TRH will meet all requirements for control of heavy loads. During the assembly of the TRH some consumables will be used that are already in the Consumable Control Program. The impact of keeping the TRH inside the RCB during a plant operation cycle has been reviewed and shows that the net free volume and heat sink capacity of the containment will remain within the allowable values. This change does not represent a change to the facility or procedure that alters information, operation, function or ability to perform the function of a structure, system or component described in the FSAR. No plant drawings or permanent plant equipment are changed.

41. 2000-065; ER-W3-99-0184-07-00, Repair of Inconel Alloy 600 Pressurizer Heater Sleeve

DESCRIPTION OF CHANGE

This change authorizes the plugging of the penetration of the pressurizer by heater sleeve F-4, utilizing an Inconel 690 plug, weld dam, and a weld pad overlay. The leak repair removes the heater and a portion of the Alloy 600 sleeve, installs a new Alloy 690 plug, and installs a weld overlay to secure the plug. The repair moves the partial penetration weld joint from the inside surface of the pressurizer to the outside surface. The leak repair leaves a small gap between the original and plug material that exposes the pressurizer base material to primary coolant.

REASON FOR CHANGE

Primary Water Stress Corrosion Cracking (PWSCC) of Inconel Alloy 600 penetrations in the Reactor Coolant System has become a significant problem in Pressurized Water Reactors over the last ten years. These penetrations involve sleeves, which are inserted through an opening in the vessel wall and are welded to the inside of the vessel wall by a J-groove weld. One of thirty pressurizer heater sleeves, sleeve F-4, was found leaking during RF-10 and is to be leak repaired by plugging.

50.59 EVALUATION

This evaluation addresses the potential affects of the pressurizer heater sleeve leak repairs during repair implementation activities including low temperature overpressurization, rigging concerns, and hydrogen generation concerns. Also evaluated is the potential for the final design to affect pressure boundary integrity, welding issues, thermal stresses, corrosion, erosion, stress corrosion cracking, and hydrogen embrittlement. The Reactor Coolant System pressure boundary integrity will not be adversely impacted by this leak repair, so the probability of a small break LOCA will not be changed. All changes are within the Reactor Containment Building, and there are no new system interactions created. There are no unreviewed safety questions, and this change does not require any Technical Specification changes.

42. 2000-066; ER-W3-00-0094-01-01, Insulate Reactor Coolant Pump Motor Terminations with Raychem Heat Shrink

DESCRIPTION OF CHANGE

The proposed change is to use Raychem Heatshrink products to insulate the six reactor coolant pump motor terminations (2 per phase) in place of Okonite T95/No. 35 tape. This change will also permit the removal of the cable lug mounting plate for the two outboard terminals and have them connect directly to the busbar on front and back. This will provide for a configuration that is significantly easier to insulate.

REASON FOR CHANGE

The use of insulating and jacket tape to perform the Reactor Coolant Pump motor terminations is a time consuming process because the connections are arranged in pairs that are in close proximity. Insulating with a heatshrink product would take less time, result in providing insulation that is a superior voltage rated product and be easier to install. The change in mounting configuration for the two outboard terminals will provide for a configuration that is significantly easier to insulate without altering the electrical configuration.

50.59 EVALUATION

It has been determined that the use of Raychem Heatshrink to insulate the RCP Motor terminations in place of the originally specified Okonite insulating tapes is considered an improvement to the plant configuration based on the higher voltage rating. Consequently, the use of this product will not increase the probability of loss of power to the RCP motors as evaluated in Chapter 15 of the FSAR. The change in the mounting configuration for the two cable lugs will not subject any connection components to additional electrical stresses different than originally designed. There will be no adverse impact to plant safety as a result of this configuration change. The postulated event of losing power to one or both RCP motors on the same 6.9 KV bus are no more adverse to the core and system performance parameters than those following a total loss of forced reactor coolant flow, which is described in the FSAR. The consequences of this incident are not increased above conditions previously evaluated in the FSAR. The proposed change only involves substituting the insulating materials for the RCP motor terminations to an improved product with higher voltage ratings as compared to the original insulating material and connecting the cable lugs directly to the busbar by eliminating the mounting plate. The cable lug connection remains rigid and seismically qualified. The removal of the mounting plate does not adversely impact containment net free volume. No additional equipment of any classification is involved in the proposed configuration change therefore there will be no increase in malfunction probability to equipment previously evaluated in the FSAR.

43. 2000-067; ER-W3-99-0551-00-02, Feed Water Isolation Valve Actuator Accumulator Bleed Valves

DESCRIPTION OF CHANGE

This minor modification will install a new manifold for the Main Feedwater Isolation Valves between the upper 4-way valve and the main manifold for each hydraulic accumulator (2 accumulators per valve). This new manifold provides two additional flow paths between the hydraulic accumulators and the hydraulic reservoir. One path is considered automatic and is controlled by a thermal relief valve with a manual block valve. The second path is considered manual and is controlled by one needle valve that bypasses the thermal relief valve path.

REASON FOR CHANGE

The Feedwater Isolation valves hydraulic actuator accumulators periodically need to have their pressure reduced in order to ensure design pressures are not exceeded during plant start-ups and ambient temperature transients. The current method employed requires one accumulator at a time to be completely dumped of all hydraulic fluid back to the reservoir and then recharged to its correct design pressure. In the past, this was considered acceptable because only one accumulator was required to maintain the operability of the valves. However, CR 98-0337 determined that both accumulators are required to maintain operability. Therefore, whenever an accumulator is dumped, the valve is declared inoperable and Technical Specification 3.6.3 (4 hour LCO) is entered until pressure is returned to normal.

50.59 EVALUATION

The Main Feed Water Isolation Valves (MFIV) are hydraulically operated and designed to fail-as-is. They are required to remain open during normal plant operations but are required to close on a Main Steam Isolation Signal (MSIS). The proposed minor modification affects the hydraulic portion of the valve actuator but does not create any new failure mechanisms nor increase the probability of any new failure mechanisms that can result in an inadvertent closure of an MFIV. Therefore, there is no increase in the probability of occurrence of a loss of normal feedwater flow as analyzed in the FSAR. The MFIVs are required to close on a MSIS. To ensure closure is achieved, the hydraulic valve operators are furnished with redundant solenoid oil drain valves powered by diverse power to ensure that no single electrical failure will prevent isolation valve closure. The proposed minor modification affects the hydraulic portion of the valve actuator and does create a new internal interface between each hydraulic accumulator and the hydraulic reservoir. This interface includes a valve for manual draining and a thermal relief valve with block isolation valve for automatic draining. The new interconnection paths originate downstream of a new flow restricting orifice. This orifice ensures that the hydraulic pump capacity exceeds the flowing capacity of the thermal relief valve should it lift open and stay open or if the manual bypass valve is left open. This change may cause the MFIV to close slower than the current design basis requirement of 5 seconds. However, this condition is already bounded by the analysis of a failed open MFIV as described in the FSAR. In addition there are no failure scenarios that would cause the MFIV to close too fast. The proposed modification does not increase the consequences of an accident previously evaluated in the FSAR. Any hydraulic fluid released is self contained and returns back to the valve's hydraulic reservoir. There is no new interaction that could create a malfunction of equipment not previously analyzed in the FSAR.

44. 2000-068; ER-W3-00-0853-00-00, Pressurizer Proportional Heater Bank #1

DESCRIPTION OF CHANGE

The pressurizer heaters (D2, F2, G2 and H2) currently associated with backup heater bank #1 will be wired into the control circuitry for pressurizer proportional heater bank #1. Power to one of the four heaters will be disabled to maintain the same loading on the control circuit of proportional heater bank #1. The degraded pressurizer heaters (A1, D1 and G1) currently associated with proportional heater bank #1 will be wired into the control circuitry for backup heater bank #1. This evaluation also addresses the reduction in total pressurizer heater capacity resulting from the removal of the heater element at location F4 on the lower pressurizer head as described in ER-99-0184-07-00.

REASON FOR CHANGE

This control circuit re-assignment will result in full heater output to the proportional heaters, as per original plant design, and will not require an immediate physical replacement of the degraded heaters with new heaters. This will reduce the amount of work done in high radiation areas with a net reduction in radiation exposure.

50.59 EVALUATION

The electrical aspects of this change involve a swap of the feeder cables between a degraded bank of three 50 kW proportional heaters and a bank of 50 kW backup heaters. In order to maintain the same load on the proportional heater bank control circuit, one 50 kW heater breaker will be maintained in the "off" position. This new configuration maintains the same protective relays and breakers for both banks of heaters. The change at the penetration involves exchange of 250 MCM feeder cables between the electrical penetration and the 480V Switchgear 3A-32. The electrical current carrying capability of feeder cables and control circuits is not affected. The protection configuration for electrical penetration 109 is not altered. The overall auxiliary plant load is reduced. The emergency diesel load isn't affected. This change reduces electrical load on the plant auxiliaries and has no adverse impact on the electrical system. The Cycle 11 safety analyses of postulated accidents, and other events potentially impacted by this change were reviewed. Certain Core Protection Calculator setpoint events credit the use of 1500 kW of pressurizer heater capacity and were re-evaluated to determine if reduced heater capacity was acceptable. Only two events had unacceptable consequences when 800 kW of heater capacity was assumed (Cycle 11 will begin with significantly more than 800 kW of heater capacity). For the two events that did not initially have satisfactory results, operating restrictions will be imposed during part of Cycle 11 to account for reduced heater capacity thus preserving safety analyses assumptions and keeping event consequences within acceptable limits. Hence, there is no reduction in the margin of safety and there are no unreviewed safety questions.

45. 2000-069; ER-W3-00-0890-00-00, Main Steam Isolation Valve Design Basis

DESCRIPTION OF CHANGE

The design basis stroke time of the Main Steam Isolation Valves (MSIVs) is being changed from 4 to 7 seconds (previously approved in part in Technical Specification Change NPF-38-224). The Technical Specification Bases for the MSIVs are being changed to clarify that the 4 second surveillance time provided in the Technical Specifications correlates to static stroke time testing. As long as the static test measures less than or equal to 4 seconds, the MSIVs will meet the design basis stroke time during accident conditions. The minimum allowed nitrogen pressure in the actuator domes is being changed from 2100 psig to 2520 psig. This will involve a setpoint change. An incorrect statement regarding MSIV stroke time being independent of differential pressure across the disc is being corrected in the FSAR and design basis document WF3-DBD-006. Incorrect Cv information is being deleted from drawing L-85262 sheets 3 & 4.

REASON FOR CHANGE

As documented in CRs 98-0875 and 98-1033, the MSIVs have been determined to not meet their 4 second design basis closure time under the current configuration, assuming actuation of only one dump valve. Also, the minimum allowed nitrogen pressure to ensure rapid closure is higher than previously calculated. These changes are primarily due to two factors. The first was the use of a non-conservative friction coefficient in valve design. The second was an incorrect statement made in the FSAR that stated differential pressure across the valve did not affect stroke time. New calculations have been developed to document the required actuator nitrogen pressure, a new expected closure time and the instrument uncertainty of the low pressure alarms, using the latest recommended calculation methods and inputs.

50.59 EVALUATION

The conclusion of this evaluation is that the changes reviewed do not impact safe operation of the plant and are bounded by the current safety analyses. The proposed changes to the MSIV design basis will not have an adverse affect on the likelihood or consequences of any accident previously evaluated in the FSAR. The physical operation of the valve has not been changed, only the stroke time assumptions. The changing of the stroke time does not involve any physical valve adjustments, and the minimum allowed nitrogen pressure setpoint change is in the conservative direction (more pressure required). All accidents potentially affected by the stroke time change were reviewed and it was determined that the change would have no adverse impact on the results or consequences of any accidents. One of the accidents potentially affected was the peak containment pressure and temperature of a main steam line break inside containment. This has been analyzed with a 7 second closure time and has been previously reviewed and approved by the NRC in TSCR NPF-38-224. It should be noted that the MSIV would close in less than 4 seconds under accident conditions when both dump valves operate, and in less than 7 seconds with the single failure of one dump valve. Since each safety analysis includes a separate limiting single failure, a 4 second MSIV closure time is acceptable for use in the safety analyses. The minimum allowed nitrogen pressure change involves a setpoint change in the conservative direction. This new allowed pressure has no impact on plant operation or physical valve closure.

46. 2000-070; ER-W3-00-0494-00-00, Modify Hotwell Level Control

DESCRIPTION OF CHANGE

The method of level control of the hotwell (condenser A) will be changed from "on/off" via an existing level switch and solenoid valve to "continuous" via new electronic equipment (controller and positioner). The revised control will utilize the existing level transmitter and valve. The desired level setpoint will be field adjustable and controlled administratively. The emergency level control will still function as before, but the existing level switch will be relocated to actuate at a lower level.

REASON FOR CHANGE

The condenser A Hotwell Level control system requires a wider control band for optimum dissolved oxygen removal.

50.59 EVALUATION

The result of this change will be continuous field-adjustable control of the condenser hotwell level. The new electronic equipment will be designed to the same safety and seismic classification (non-safety and non-seismic) as the existing equipment and the same material, design and construction standards will be used. High condenser level could result in high level in Feedwater Heaters 5 & 6, which is an initiator for a turbine trip, however the setpoints for the existing high level alarms are not being modified. A loss of condensate pumps (resulting from low level) can cause a loss of Feedwater, but the Emergency Makeup valve will still function to maintain adequate level. The limiting condenser initiated accident is Loss of Condenser Vacuum. A failure of the electronic controls could affect hotwell level which would still be bounded by the loss of condenser vacuum scenario. This change does not affect the configuration of the Process Analog Control equipment and no existing Control Room indication or annunciation will be impacted. Since the condenser is not credited in the accident analysis there is no increase in dose consequences. This change improves the plant's ability to maintain dissolved oxygen in the condensate to an acceptable value. The Condensate system/pumps are not important to safety. The resulting NPSHA for the condensate pumps was evaluated and it was determined that he revised NPSHA is acceptable for proper condensate pump operation with a minimum normal hotwell level at 32" at condensate pump runout conditions. This change modifies the components that provide normal and emergency hotwell level control. The change will allow the hotwell to automatically control at lower levels, which will reduce the condensate dissolved oxygen concentration. This will enable improved secondary chemistry and plant performance. If the normal control fails low, a level switch opens the Emergency Makeup valve to preclude a loss of NPSH to the Condensate pumps. A high failure will actuate alarms in the Control Room. However if a turbine trip does occur, this has already been analyzed in the FSAR. This evaluation determined that no unreviewed safety questions exist.

47. 2000-071; ER-W3-99-0947-00-00, QA Audit (Fire Protection): Clarification for Technical Requirements Manual 4.7.10.1.1.f.(3)

DESCRIPTION OF CHANGE

Technical Requirements Manual (TRM) Section 4.7.10.1.1f.(3) states, "Verifying that each fire suppression pump starts (sequentially) to maintain the fire suppression water system pressure greater than or equal to 96.5 psig." This section is being revised to delete the specific requirement to "maintain the fire suppression water system pressure at greater than or equal to 96.5 psig" and require that the "pumps start sequentially on a continued pressure drop in the fire suppression system". This change is to the TRM and to the acceptance criteria of procedure OP-903-056 only and does not change any set points or make physical changes to the plant as presently designed.

REASON FOR CHANGE

The 1999 Fire Protection QA Audit identified the 96.5 psig figure as a requirement that was not being verified during testing. The wording of the requirement is sufficiently ambiguous as to render it difficult to apply or to determine its intent. The most reasonable interpretation is that it referred to a minimum starting pressure. If so, the basis for choosing 96.5 is not reproducible using generally accepted fire protection practices. Also, the starting pressure of the fire pumps is only one of many design features to facilitate proper and safe functioning of the system and is not considered a parameter needed in the Technical Requirements Manual. The Technical Requirements should be limited to the critical items that must be in place for the system to be operable. The 96.5 psig figure is not meaningful as either a starting pressure or a part of a flow and pressure requirement and should be deleted.

50.59 EVALUATION

The 96.5 psig requirement presently specified in the TRM and OP-903-056 has no basis for establishing operability of the sequential start feature of the fire pumps at Waterford 3. The 96.5 psig figure does not support any relevant fire suppression water system function. There is no flow associated with the pressure nor is a gauge elevation given. The pressure figure therefore does not represent a minimum flow and pressure that the system must be capable of maintaining and it does not conform to a minimum starting pressure because the location of the pressure reading is not specified. The fire pumps are set to start sequentially as the failure of one pump to start should not prevent the next pump from starting. Any one of the 3 pumps is capable of supplying the largest design demand of the fire suppression system and having multiple pumps running is not a system design requirement. The operability issue is that the pumps start automatically, not the exact pressure at which they start. This change will not increase the probability or consequences of accidents or malfunctions of equipment important to safety previously evaluated in the FSAR. Also, this change will not create the possibility for an accident or malfunction of equipment important to safety of a different type than any previously evaluated. The fire suppression water system is not addressed by the Technical Specifications.

48. 2001-002; ER-W3-00-1023-00-00, Air Evacuation Pumps Temporary Cooling

DESCRIPTION OF CHANGE

Two 30 ton chillers will be installed to provide chilled water to cool the seal water for Condenser Vacuum Pumps B and C. Turbine Cooling Water will be isolated to each vacuum pump heat exchanger. Chilled water will be supplied to the Turbine Cooling Water side of each vacuum pump heat exchanger.

REASON FOR CHANGE

Condensate dissolved oxygen averages about 3 ppb during the summer to about 9 ppb during the winter. It is desired to maintain dissolved oxygen as low as possible at all times. As condenser back-pressure is reduced, the volume occupied by a mass unit of air is greatly increased. The vacuum pumps always pump the same volume of water vapor and air mixture, but a given volume will contain less mass at lower condenser back-pressure resulting in less air and water vapor mass flow. Providing cooler seal water to vacuum pumps B and C will increase the mass flow rate capacity which should result in removal of more air which will reduce the amount of dissolved oxygen in the condensate.

50.59 EVALUATION

The only accidents that might be affected are “loss of condenser vacuum” and “loss of condenser vacuum with a concurrent single failure”. This temporary alteration will affect two of three available condenser vacuum pumps. The proposed temporary alteration will not increase the probability of occurrence of these accidents because only one vacuum pump is required to maintain condenser vacuum. The air evacuation system has no safety-related function nor is it relied upon to perform any accident mitigating function. The air evacuation system will still perform the same functions of maintaining condenser vacuum and removing non-condensables from the condenser. The air evacuation system is not considered equipment important to safety. The air evacuation system is not electrically interlocked with any systems or components important to safety. The air evacuation system is interconnected to the main condenser which is important to safety. However, the only credible failure mechanism would be the loss of vacuum, which is analyzed by the FSAR and is no more likely to occur. The additional electrical load on the non-safety bus was evaluated and shows that the bus is stable with the additional load provided by two 30-ton chillers. Consequently no safety system or component will be impacted by the proposed temporary alteration. This temporary alteration will not affect the safety or environmental aspects of the FSAR documents, and there are no unreviewed safety questions.

49. 2001-005; ER-W3-01-0128-00-00, Condenser Vacuum Pumps Temporary Cooling

DESCRIPTION OF CHANGE

One 400 ton chiller will be installed to provide chilled water to cool the seal water for Condenser Vacuum Pumps B and C. Turbine Cooling Water will be isolated to each vacuum pump heat exchanger. Chilled water will be supplied to the Turbine Cooling Water side of each vacuum pump heat exchanger.

REASON FOR CHANGE

Condensate dissolved oxygen averages about 3 ppb during the summer to about 9 ppb during the winter. It is desired to maintain dissolved oxygen as low as possible at all times. As condenser back-pressure is reduced, the volume occupied by a mass unit of air is greatly increased. The vacuum pumps always pump the same volume of water vapor and air mixture, but a given volume will contain less mass at lower condenser back-pressure resulting in less air and water vapor mass flow. Providing colder seal water to Vacuum Pump B and C will increase the mass flow rate capacity which should result in removal of more air which will reduce the amount of dissolved oxygen in the condensate.

50.59 EVALUATION

The only accidents that might be affected are "loss of condenser vacuum" and "loss of condenser vacuum with a concurrent single failure". This temporary alteration will affect two of three available condenser vacuum pumps. The proposed temporary alteration will not increase the probability of occurrence of these accidents because only one vacuum pump is required to maintain condenser vacuum. The air evacuation system has no safety-related function nor is it relied upon to perform any accident mitigating function. The air evacuation system will still perform the same functions of maintaining condenser vacuum and removing non-condensables from the condenser. The air evacuation system is not considered equipment important to safety. The air evacuation system is not electrically interlocked with any systems or components important to safety. The air evacuation system is interconnected to the main condenser which is important to safety. However, the only credible failure mechanism would be the loss of vacuum, which is analyzed by the FSAR and is no more likely to occur. The temporary chiller will be powered from offsite power which is completely independent of the in plant power supply. Consequently no safety system or component will be impacted by the proposed temporary alteration. This Temporary Alteration will not affect the safety or environmental aspects of the FSAR documents, and there are no unreviewed safety questions.

50. 2001-007; ER-W3-01-0248-00-00, FSAR Table 9.4-5

DESCRIPTION OF CHANGE

This change will correct FSAR Table 9.4-5, Fuel Handling Building Ventilation System Failure Modes and Effects Analysis. The Table currently shows that each Fuel Handling Building Isolation Damper has a redundant isolation damper located in series. However, isolation dampers D-35A and D-35B are arranged in parallel. This change will list dampers D-35A and D-35B in a separate row and add a remark stating that the system can perform its safety related function with one isolation damper failed open.

REASON FOR CHANGE

The reason for changing FSAR Table 9.4-5 is to correct the FSAR to reflect the system's as-built configuration. This change will ensure that FSAR Table 9.4-5 agrees with existing Fuel Handling Building Ventilation System configuration and the FSAR system description.

50.59 EVALUATION

This change is limited to the isolation dampers for the fuel handling building ventilation system. These dampers are required to actuate to their safety related positions following a fuel handling accident signal. The isolation dampers close to ensure that the contaminated areas in the fuel handling building can be maintained at a negative pressure following a fuel handling accident. These dampers are require to mitigate the consequences of a fuel handling accident in order to limit the potential offsite exposures within the acceptable limits of 10CFR100. However, these components do not have any functions that would prevent or cause a fuel handling accident to occur. Furthermore, the change will only correct the FSAR table to describe the as built configuration of the fuel handling building ventilation system. This change will not impact the margin of safety as defined in the basis for any Technical Specification. The Technical Specification defines the operability and surveillance testing requirements for the fuel handling building ventilation system. This change will reflect the as-built condition of the system. Technical Specification surveillance has been performed in this configuration to verify that the system meets its operability requirements. This correction does not affect the iodine removal efficiency of 99% assumed in the Safety Analysis. Although the dampers are configured in parallel, rather than series, the system will still be capable of performing its safety-related function following a single active failure. Therefore, this change does not present an unreviewed safety question.

51. 2001-008; ER-W3-98-0972-00-00, Modify Secondary Sampling Sodium and Conductivity Monitors

DESCRIPTION OF CHANGE

The proposed change will allow the Secondary Sampling sodium and conductivity monitors SSL-IAIT-7403 and SSL-ICIT-7405 to measure various sample points using the Secondary Sampling patch panel. Also, the pH monitor in the Demineralized Water Analyzer panel is no longer in use and will be made "inactive".

REASON FOR CHANGE

Following the installation of the Demineralized Water Storage Tank and additional piping for the Condensate Transfer Pump, it became difficult to obtain samples of both process streams for the subject Secondary Sampling sodium and conductivity analyzers. Inadequate pressure and fouling prevented proper flow from the makeup demineralizer effluent to the Secondary Sampling lab. Currently, conductivity and sodium monitoring of makeup demineralizer effluent occurs at the Demineralized Water Analyzer Control Panel and makes these instruments obsolete. The pH monitor in DW-EPNL-3000 was never needed and is not used.

50.59 EVALUATION

Analyzers SSL-IAIT-7403 and SSL-ICIT-7405 measure conductivity and sodium levels in the Makeup Demineralizer Effluent and Condensate Transfer Pump sample streams. Monitor DW-IAT-3005 and associated local indicator DW-IAI-3005 measure the pH level of demineralized water samples. None of these functions or activities can affect the initiation of or radiological consequences of an accident. Consequently, altering or removing these functions or activities cannot affect the radiological consequences of an accident. In addition, this proposed change does not create any new release pathways. Only process streams previously routed to the secondary patch panel will be allowed to be directed to analyzers SSL-IAIT-7403 and SSL-ICIT-7405. The outlet from these instruments will be routed to the patch panel and the discharges will be processed in the same manner as previously. There are no margins of safety associated with the subject components or their functions. Technical Specification 3/4.11 and Bases were reviewed for potential impact due to the proposed change. This technical specification discusses liquid and gaseous effluents, and their storage tanks that do not interface with the secondary sampling system. Since only process streams previously routed to the secondary patch panel will be allowed to be directed to the analyzers and since the discharges will be processed in the same manner as previously, no radioactive effluents will be created, changed or directed to new locations for processing. This evaluation concludes that increasing the capability of the Secondary Sampling monitors to sample additional inputs, and inactivating the Demineralized Water pH monitor will not result in an unreviewed safety question, nor will it result in a reduction in the margin of safety of any technical specification.

52. 2001-009-01; ER-W3-01-0261-00-01, Increase Speed Setpoint for Emergency Feed Water Turbine Driven Pump AB

DESCRIPTION OF CHANGE

The proposed activity increases the Emergency Feedwater Terry Turbine (AB Pump) operating speed from 4410 +40/-35 RPM to 4450 +/- 30 RPM and permits the removal of step 18.2 in procedure OP-903-046 to deduct instrument uncertainty from the test gage reading.

REASON FOR CHANGE

The setpoint change is intended to provide additional operating margin for the Emergency Feed Water AB pump. Increasing the speed setpoint will increase pump discharge pressure, creating more operating margin between the minimum requirement and the actual Emergency Feed Water discharge pressure. A minimum Emergency Feed Water design pressure evaluated in ER-W3-01-0261 identifies wider operating margin between the surveillance limit and the minimum pump discharge design pressure. This accommodates an instrument uncertainty as high as 20 psi, eliminating the requirement in OP-903-046 for deducting instrument uncertainty from the value read on the turbine driven pump discharge pressure surveillance instrumentation.

50.59 EVALUATION

This evaluation concludes that the proposed setpoint change will not create an unreviewed safety question. Operation of the Emergency Feed Water AB pump and the Terry Turbine within the limits of the proposed setpoint change is bounded by current system configuration, which achieves minimum design pressures as verified by technical specification surveillance and trips the Emergency Feed Water on overspeeds greater than 4900 RPM. This proposed activity does not change pump discharge minimum design pressure and maintains pump operating speed between 4420 and 4480 RPM. The subsequent increase in discharge pressure is lower than maximum pressure limits allowed by Code. The accident consequences remain bounded and this change will not create a malfunction of equipment important to safety of a different type. Therefore, since the maximum pressure expected during surveillance testing is below the 1470 psig limit, the higher discharge pressure expected as a result of this proposed change is within the design and licensing basis. This proposed activity results in an increase of flow from 2340 to less than 2460 gpm or a net change of less than 120 gpm. This change does not affect the consequences of an accident or malfunction as described in the FSAR.

53. 2001-010; ER-W3-01-0268-00-00, FSAR Section 2.5 Electrical Log Boring Test Figures

DESCRIPTION OF CHANGE

FSAR Figures 2.5-32(a) through 2.5-32(f) will be classified and maintained as historical information. The figures will be removed from the FSAR and placed in Document Control. These figures are plots of electrical log boring tests.

REASON FOR CHANGE

The FSAR will be maintained in an electronic format and updated quarterly. Just a few hard copy volumes will be maintained. The figures cannot be converted to electronic format because of their configuration and size. The figures will be classified and maintained as historical information pursuant to Nuclear Energy Institute (NEI) 98-03 and Waterford 3 Site Procedure W4.504.

50.59 EVALUATION

This change does not affect the configuration or operation of Waterford 3. The change does not affect the safety analysis or FSAR Chapter 6 or 15 accident analyses. This change solely classifies and maintains the specified information as historical. This change is consistent with the guidance in NEI 98-03 and Waterford 3 site procedure W4.504. This change will not affect the configuration, analysis, or operation of Waterford 3. The change does not involve an unreviewed safety question.

54. 2001-011; ER-W3-98-1379-00-00, Change in NPSHA for the Emergency Feed Water Motor and Turbine Driven Pumps

DESCRIPTION OF CHANGE

This calculation revision changes the NPSH available at design flow for the Emergency Feedwater System pumps for the worst case operating condition with the Condensate Storage Pool as the water source. In addition it also calculates the NPSH available at runout flow for Emergency Feed Water pumps for the worst case operating condition. The revised NPSH values are incorporated into the FSAR.

REASON FOR CHANGE

The change was produced as a result of the comments made from the Design Basis Review of the earlier revision of Calculation MN(Q)-10-12 R0. The resistance coefficient values used for fittings, valves, etc. were low in comparison to the published values contained in Crane's Handbook (No. 410) which resulted in the calculation indicating low friction losses. Also, the low water level of the Wet Cooling Tower basin utilized in computing the static head in the system was unusually high and needed to be corrected. The revision of the calculation also provided additional information regarding pump NPSH available for the runout conditions of the Emergency Feed Water pumps.

50.59 EVALUATION

The NPSH available for the Emergency Feed Water Turbine driven pump under worst case scenario was calculated to be higher than the required NPSH for the Design Flow and Runout Flow conditions. Although the NPSH available was shown to be lower than the original values shown in Rev. 0 of the calculation, the available NPSH remains higher than the required NPSH. The Emergency Feedwater System can initiate the Chapter 15 accidents involving an increase in feedwater flow, and an increase in feedwater flow with a single failure, but this change to the calculation, which only confirms that the available NPSH is greater than the required NPSH, does not affect the probability of occurrence of these accidents. No accidents will have radiological release consequences altered by this change. The Emergency Feed Water System will function as necessary. The equipment important to safety that could be affected by the change are the Emergency Feed Water pumps and system. The proposed change has no affect on the system because the NPSH available is still greater than the NPSH required. The likelihood of malfunction will not increase. There exists adequate margin (design and runout conditions) of NPSH above that required. If the Emergency Feed Water pumps malfunctioned there would be decreased makeup water to the steam generators. This affects all accidents. The calculated NPSH available under the worst case scenario is more than that required, so there will be no malfunction of the equipment due to NPSH considerations. The consequences have not changed because the pumps and their performance have not changed. Therefore, no unreviewed safety question exists.

55. 2001-014; ER-W3-00-0799-01-00, TRM Change 01-005, Remove Emergency Feed Water Flow Control Valves from Containment Isolation Valve Table 3.6-2

DESCRIPTION OF CHANGE

This change removes the Emergency Feed Water flow control valves (EFW-223A & B and 224A & B) from the containment isolation valve Table in the Technical Requirements Manual.

REASON FOR CHANGE

Emergency Feed Water Flow Control valves, EFW-223A&B and 224A&B, are not containment isolation valves. The credited isolation valves are EFW-228A&B and 229A&B. The Emergency Feed Water system is a closed system. 10CFR50 Appendix A, GDC 57, "Closed System Isolation Valves" states: Each line that penetrates the primary reactor containment and is neither part of the reactor coolant pressure boundary nor connected directly to the containment atmosphere shall have at least one containment isolation valve which shall be either automatic, or locked closed, or capable of remote manual operation. This valve shall be outside containment and located as close to containment as practical. A simple check valve may not be used as the automatic isolation valve." This is also supported by the Standard Review Plan Section 6.2.4 and FSAR Section 6.2.4.1.2. Therefore, EFW-228A&B and 229A&B meet the requirements for closed system isolation valves as defined in 10CFR50 Appendix A, GDC 57.

50.59 EVALUATION

One containment isolation valve per line is required for the Emergency Feed Water system as discussed in 10CFR50 Appendix A, GDC 57, Section 6.2.4 of the Standard Review Plan (NUREG-0800), and FSAR Section 6.2.4.1.2. EFW-228A and229A, and EFW-228B and 229B are the credited containment isolation valves for penetrations 3 and 4, respectively. Therefore EFW-223A&B and 224A&B can be deleted from Technical Requirements Manual Table 3.6-2. The Emergency Feed Water flow control valves alone are not initiators of any accident described in the FSAR. An Emergency Feed Water pump would have to be running along with a Flow Control Valve further opening in order for an increase in feedwater flow event to occur. However, this event is already postulated to occur in the FSAR with acceptable results (consequences bounded by other accidents). The penetrations in which these valves are located are required to be isolated remote manually by Containment Isolation Valves if required following an accident to ensure consequences of accidents posing radiological hazards are limited by the primary containment. The credited containment isolation valves associated with these penetration are unchanged. Deletion of the EFW Flow Control Valves from the TRM table will not prevent the penetrations from being isolated to mitigate the consequences of an accident. The change has no effect on the design or operation of the Emergency Feed Water system. No physical change is being made to the plant. This change does not affect the ability to isolate the penetrations.

56. 2001-017; ER-W3-98-0821-01-00, Inactivation of Part of the Primary Water Treatment Plant

DESCRIPTION OF CHANGE

This change inactivates part of the primary water treatment plant (PWTP). The part being inactivated is referred to as the Boze Water Treatment System. The remaining part of the water treatment plant consisting of the clearwell tank and transfer pumps and associated piping, valves, instrumentation and control will remain functional. The primary Water Treatment Plant is designated as a non-safety related, non-quality related and non-seismic system.

REASON FOR CHANGE

The function of the primary water treatment plant is to provide a clean water source for various plant systems and components. The original design of the PWTP considered two methods for producing clean water. The primary method uses water from the Potable Water System that is diverted to the clearwell tank by a level control valve. The back up method draws water from the Circulating Water System (raw water from the Mississippi River) and processes it through the Boze Water Treatment System which removes suspended solids and then pumps it to the clearwell tank. Plant experience has shown that the Boze Water Treatment System has been difficult to operate, requires constant attention and has been plagued with maintenance problems since its installation. A decision has been made that the reliability of the Potable Water System is high enough to warrant inactivating the Boze System.

50.59 EVALUATION

The inactive status of the Boze Water Treatment System will not affect the availability of clean water for plant usage or the function of any safety related structure, system or component needed for plant operation or accident mitigation. The Boze System is considered a backup system for clean water. The availability and reliability of the Potable Water System eliminates the need for a backup system. The Boze Water Treatment System is located in the water treatment building and has no direct or indirect interface with those safety related structures, systems or components that are needed to safely shut down the reactor or to mitigate an accident postulated in the FSAR. The postulated accidents described in the FSAR make no reference to the Primary Water Treatment System or the Boze System. The postulated accidents described in the FSAR make no reference to the Boze System or the need to process river water for plant make up. The make up water source for the various systems and components are the primary water storage tank, condensate storage tank and the demineralized water storage tank during normal plant operations. The Refueling Water Storage Pool, Condensate Storage Pool and Wet Cooling Tower Basins are the safety related systems that are available for accident mitigation and post accident recovery. The safety related storage pools and cooling tower basins have sufficient volumes to account for water usage and system losses during and following accidents. The inactive status of the Boze System will isolate the system from interfacing systems and components by closing process valves and opening circuit breakers. The isolation of the Boze system will not impact the normal operation of the Potable Water System or the Circulating Water System. The transfer of water from the clearwell tank to the demineralized water storage tank, condensate storage tanks and the fire water storage tanks will not be affected.

57. 2001-018; ER-W3-00-0300-00-00, Update Sprinkler System Hydraulic Calculations

DESCRIPTION OF CHANGE

This change incorporates the affect of updated hydraulic calculation of sprinkler systems protecting safety related plant areas. This in turn will revise the minimum required volume contained in the fire water storage tanks as depicted in the FSAR and TRM. This change also eliminates the calculated value associated with the now abandoned system and interconnection to Circulating Water. Because this was eliminated by a previous design change there is no need to maintain this value as a tank level value.

REASON FOR CHANGE

Hydraulic calculations performed for plant sprinkler systems have indicated a change in the highest demand sprinkler system and thus the minimum required water volume. The minimum required volume was reduced indicating a conservative increase in tank volume margin.

50.59 EVALUATION

Fire protection is non-safety quality related. No adverse nuclear safety impact was identified. Further, because this change was resultant of increased accuracy of the hydraulic calculation methods the reduced contained volume is a more accurate value and represents additional margin with respect to storage capacity. The accident or plant situation directly impacted by this change would be a fire event. The change to the calculated contained water supply does not affect the consequences of a fire because the volume is based on the revised highest sprinkler system flow and hose stream demands which are designed to adequately contain and control the fire events postulated. The plant's capabilities to address the consequences of a fire have been maintained consistent with that previously analyzed and approved. The change to the calculated contained water supply does not affect the probability of occurrence of a malfunction within the fire protections system nor the circulating water system. The fire water storage tanks volume would only be called into affect once the fire event has taken place. The amount contained is sized to adequately address the fire events postulated and includes a 100% redundant supply. This ER does not alter the performance characteristics of the fire protection system nor place additional or differing demands on the system's performance. Because there are no physical system interconnections and dependencies related to this change, the possibility of another or differing accident are not presented.

III. PROCEDURE CHANGES

A. PLANT PROCEDURES

1. 2000-041; W4.503, Control of Changes to the Operating License and Selected Licensing Basis Documents, Rev. 5

DESCRIPTION OF CHANGE

Site procedure W4.503 "Control of Changes to the Operating License and Selected Licensing Basis Documents" has been revised to clarify administrative controls that govern changes to the Operating License, Technical Specifications, Technical Requirements Manual, Technical Specification Bases and Core Operating Limits Report. Also, the Technical Requirements Manual (TRM) Introduction section on page I was revised to remove reference to W4.503 as the site procedure to follow in making changes to the TRM. The change to the TRM will specify that the administrative controls prescribed in the Waterford 3 Site Procedures should be followed.

REASON FOR CHANGE

W4.503 was rewritten to clarify the administrative controls and minimize the confusion in processing licensing basis document changes to the Operating License, Technical Specifications, Technical Requirements Manual, Technical Specification Bases and the Core Operating Limits Report. This procedure was revised to incorporate NRC approved Amendment 161, Technical Specification Bases Control Program, into the TS as section 6.16 (implementation date 7/8/00). In addition, this procedure revision resolved several concerns identified in the following condition reports: (1) CR-98-0003, clarify steps in processing TRM changes; (2) CR-97-1455, to clearly identify the reason for the change to the COLR and any mode applicability. To avoid future unnecessary changes to the TRM, the specific reference to a site procedure was removed and left as a general statement.

50.59 EVALUATION

The revisions to Site Procedure W4.503 and the TRM Introduction section do not create an Unreviewed Safety Question. W4.503 is an administrative change that clarifies the administrative controls that govern changes to licensing basis documents. The TRM change is also administrative in nature by removing the specific reference to W4.503 by procedure number and title. No changes are being made to the plant that could adversely affect the probability of either an accident or a malfunction of equipment important to safety. No physical changes are being made that could create either a new accident or a new equipment failure mode. No protective boundary is affected by this change and no margin of safety is reduced by this change.

2. 2000-050; PLG-009-015, Temporary Facilities, Rev. 1

DESCRIPTION OF CHANGE

This evaluation addresses the impact of the radioactive material storage in areas outside the Controlled Access Area (CAA) and Plant structures. These storage areas are inside the Protected Area such as the temporary Waste Storage area west of the Turbine Building, area behind the Radioactive Material Storage Building and the Warehouse, etc. The Radiation Protection Group will select and establish the location of any temporary radioactive material storage area inside the Protected Area. These areas will be controlled in accordance with applicable regulatory requirements and plant procedures to protect members of the public, prevent uncontrolled or unmonitored release of radioactive materials to unrestricted areas, and assure adequate public health and safety with minimal environmental impact.

REASON FOR CHANGE

Radioactive waste materials "in transition" stored between processing and shipping are normally packaged in their final containers. Containers are kept in temporary holding areas prior to being shipped. Examples of stored radioactive materials are tools or equipment not being discarded or awaiting final disposition, special outage tooling or equipment used during outages that may be awaiting transition to a final storage location or shipment to an outside vendor and miscellaneous containers containing radioactive material.

50.59 EVALUATION

The interim storage area inside the Protected Area as described in FSAR section 11.4.10.1 and Figure 11.4-7 provides sufficient space for over two months of shielded storage for waste which is normally stored before shipment. Other radioactive material storage areas established outside the CAA / plant structures within the Protected Area are maintained under control of the Radiation Protection Group. 10CFR20.1301 (b) states that if members of the public are permitted access to controlled areas, the limits of 10CFR20.1301(a) still apply. To meet this requirement of 10CFR20.1301, these storage areas are enclosed and public access is restricted. Radiation Protection personnel periodically survey designated radioactive material storage areas according to the same administrative Radiation Protection procedures that are used in the CAA. The dose rates at the Restricted Area fence boundary will be below the limits of 10CFR20.1301(a), thereby ensuring them to be below the dose limits at the site boundary. Stored radioactive materials are surveyed prior to leaving the CAA and are packaged, housed or enclosed to contain any contaminant. Containers will remain closed and controlled in accordance to the plant procedures. In the event of a natural disaster such as a hurricane, any container that could be damaged by high winds will be moved into a protected location or anchored to avoid being ejected and damaged. Therefore, any radioactive material stored inside the Protected Area will have no potential for uncontrolled/unmonitored releases to unrestricted areas that could exceed any effluent limit specified in 10CFR20, TRM and Offsite Dose Calculation Manual. Based on this safety evaluation, radioactive material storage areas inside the Protected Area will not result in an unreviewed safety question nor require any Technical Specification changes with regards to environmental and radiological considerations. There will be negligible impact on the health and safety of the public.

3. 2000-062; Change Operating Procedures to Allow Extended Equipment Hatch Closure Time

DESCRIPTION OF CHANGE

Changed procedures OP-001-003 RCS Drain Down, R19C9; OP-901-131, Shutdown Cooling Malfunction R1C6; OP-901-403, High Airborne Activity R1C3 and OP-901-405, Fuel Handling Incident R1C4 to allow extended equipment hatch closure time while moving large equipment that may preclude closure within one hour in Mode 5.

REASON FOR CHANGE

Clarify containment closure times required in Mode 5.

50.59 EVALUATION

This evaluation addresses the potential affects of the loss of extended equipment hatch closure times in Mode 5 when moving heavy equipment through the hatch. The change in time to close the containment equipment hatch would not alter the probability of occurrence of an accident. The operation of equipment is not changed by these procedure changes. The time allowed to close the equipment hatch does not affect how equipment is operated. Since there is no change in how equipment is operated, the change in time to close the containment equipment hatch would not alter the occurrence of a malfunction of equipment. The consequences of a malfunction of equipment important to safety remains unchanged. Requiring two trains of High Pressure Safety Injection and two Emergency Diesel Generators to be operable while moving large equipment through the equipment hatch does not increase the consequences of a malfunction of equipment. The change in the allowed time to close containment in mode 5 does not create any new system interactions or operate systems in a different manner. The configuration and response of the Shut Down Cooling system are the same as analyzed for other accidents. There are no technical specifications associated with closure of the equipment hatch in mode 5. The requirement for both diesels to be operable provides greater defense in depth than is required in mode 5. This evaluation concludes that the proposed change will not degrade the required integrity of containment during Mode 5 or the functional capability of the equipment hatch. All changes are within the Reactor Containment Building, and there are no new system interactions created. There are no unreviewed safety questions, and this change does not require a Technical Specification change.

B. SPECIAL TEST PROCEDURES

1. 2000-030; STP-289682, Rev. 1, Instrument Air System Leakage Test

DESCRIPTION OF CHANGE

This test will determine if valves SA-126 and SA-127 can be relied upon to isolate the Instrument Air system while the Station Air system is secured to implement a design change.

REASON FOR CHANGE

Valve SA-126 and SA-127 will be Danger Tagged Closed during the implementation of design change DC-3390 to ensure that the Instrument Air system remains fully pressurized while the Station Air system is depressurized. There has been some indication in the past that these valves may leak. This test will provide assurance that these valves can be relied upon as a system boundary prior to the installation of DC-3390.

50.59 EVALUATION

The Instrument Air system is not safety-related and is not required for safe shutdown of the plant or for limiting radiological releases. The Instrument Air system does provide air to numerous safety-related valves in the plant. These valves fail-safe or are provided with safety related backup air or nitrogen supply accumulators to ensure they can perform their safety functions on loss of air. This test will be secured and the system returned to normal if the Instrument Air system pressure falls below 100 psig. This test does not involve a change to a protective boundary. No margins of safety will be affected. This test cannot cause safety limits to be exceeded. There are no unreviewed safety questions created by the performance of this test.

2. 2000-057; STP 420689, CVC-403 Response Evaluation, Rev. 0

DESCRIPTION OF CHANGE

This test will evaluate the operating characteristics of valve CVC-403 and quantify system effects due to valve repositioning. The results of this test will provide data to determine volume control tank (VCT) and Controlled Bleedoff (CBO) conditions that will allow the most efficient VCT degassing.

REASON FOR CHANGE

To support VCT pressure changes required for degassing operation, CVC-403 must be throttled to maintain minimum required Controlled Bleedoff (CBO) pressure. Experience has shown that CVC-403 is sensitive to operate and can cause vendor recommended CBO pressure operating parameters to be exceeded. In addition, the Reactor Coolant Pump (RCP) seal vendor has provided an expanded pressure band that can be utilized during short periods of operation. This change to the vendor recommended operating parameters from a range of 40 to 65 psig to an increased range of 30 to 120 psig is justified since these parameters are based solely upon the vendors recommendation for RCP seal operation. To avoid possible RCP seal degradation, the vendor has recommended the introduction of an additional operating parameter to maintain CBO pressure changes to less than or equal to 4 psig/min when utilizing the new pressure band. This Special Test Procedure will utilize these new pressure values to evaluate valve positions that will facilitate the most efficient venting operations. The first objective of the test is to identify adverse responses associated with CVC-403 movement to establish methods for mitigating any negative effects. The second objective is to determine if CVC-403 can be positioned to establish system conditions, which will allow the performance of the most efficient venting of the Volume Control Tank.

50.59 EVALUATION

The special test procedure being evaluated directs the variation of VCT pressure within the vendor recommended operating parameters and the manipulation of CVC-403 to establish Controlled Bleed Off pressures within Reactor Coolant Pump seal vendor recommendations. The Volume Control Tank and Reactor Coolant Pump Controlled Bleed Off are not required for safe shutdown of the plant or for limiting radiological releases. Therefore, since the performance of the Special Test Procedure being reviewed only varies the parameters of these systems within allowable limitations, there are no unreviewed safety questions created by the performance of this test.

3. 2000-059; STP 404676, Station Air Supplying Instrument Air, Rev. 0

DESCRIPTION OF CHANGE

This Special Test Procedure will align the Station Air system such that it will be the sole source of air pressure and flow to the plant Instrument Air system. The Instrument Air compressors and the Instrument Air Receiver will be secured and isolated from the plant.

REASON FOR CHANGE

This is a Special Test Procedure which will be used as the acceptance test for ER-W3-00-0315-00-00, Station Air to Instrument Air Equipment Reliability Modification.

50.59 EVALUATION

This test will only affect the instrument air and station air systems. There are no accidents that are initiated by the instrument air or station air systems. Neither the station air or instrument air system serve a safety function and a complete failure of either system will not initiate an accident. The compressed air systems are not needed for the safe shutdown of the reactor or to mitigate the consequences of an accident. Equipment that requires compressed air to perform their post accident safety functions are equipped with safety-related air or nitrogen accumulators. This test will not have an affect on any accident previously evaluated in the FSAR, nor will it have an affect on the consequences of an accident previously evaluated in the FSAR. The compressed air systems are non-safety, non-seismic and are not needed for the safe shutdown of the reactor or to mitigate the consequences of an accident. This test will not affect the safety or environmental aspects of the FSAR documents and there are no unreviewed safety questions.

4. 2001-004; STP 424982, Local Leak Rate Test of PSL-303/304

DESCRIPTION OF CHANGE

The special test is generated to perform post repair local leak rate testing on valves PSL-303/304. The test also addresses potential leakage past test boundary Reactor Coolant System isolation valves RC-319 and PSL-301 by monitoring for test boundary pressure increase for 15 minutes prior to initiating the Local Leak Rate Test (LLRT). During the LLRT, RC-319, PSL-301 and PSL-306 remain closed. The inboard and outboard test connection valves, PSL-302 and PSL-305 are opened. A minimum of one containment isolation valve (PSL-303 or PSL-304) remains closed during performance of the test.

REASON FOR CHANGE

CR-WF3-2001-0118 and 0243 identified that valves PSL-303 and PSL-304, Pressurizer Steam Space Sample Inside and Outside Containment Isolation Valves, failed to properly close. Subsequent investigations determined that the manufacturing tolerances may not provide adequate diametrical clearance to ensure that binding between the plug and cage would not affect proper closure of the valves. This configuration change may increase leakage through the valve. A post repair LLRT is required to demonstrate the valve's capability to provide containment integrity. FSAR Section 6.2.6.4 describes the scheduling and reporting of periodic tests. This section indicates that Type B and C periodic tests must be performed during reactor shutdown for refueling. The special test is not considered a periodic test, thus the shutdown requirements are not applicable.

50.59 EVALUATION

The proposed change does not physically change any of the Primary Sample Line Structures, Systems or Components and does not change the intended PSL-303/304 containment isolation function. A break in the pressurizer steam space sample line is a potential initiator for the FSAR section 15 primary sample or instrument line breaks. These breaks are analyzed because they have the potential to release reactor coolant system activity outside of containment. For the special test, valves RC-319 and PSL-301 will be closed. These valves isolate the reactor coolant system from the test boundary and eliminate the outside containment leakage path. With the sample line isolated the FSAR sample line break consequences remain bounded. The PSL-303 and 304 valves are also credited as containment isolation valves for events that initiate a Containment Isolation Actuation Signal. During the special test, Technical Specification 3.0.5, 3.6.1.1. and 3.6.3 will be entered. In addition administrative controls defined in the technical specification bases to maintain containment integrity will be performed by ensuring that PSL-306 is closed for the duration of the special test and that the penetration will be isolated on reactor trip by gagging PSL-304 closed and closing PSL-305. The administrative controls consist of an operator stationed locally for valve operation and in constant communication with the control room and the operator will close the valves on a reactor trip. Environmental conditions will not preclude the operator from accessing the valve and this action will limit the potential release of radioactivity. Therefore, the special test does not reduce the margin of safety as defined in the licensing basis. The proposed change also does not increase the probability or consequences of any design basis accident.

IV. COMMITMENT CHANGES

1. COMMITMENT CHANGE NO. 2000-0010, Operating Procedure Change to Include any Necessary Human Factoring Enhancements

   ORIGINAL COMMITMENT DESCRIPTION

   Review Reactivity Related Sections of OP-002-005 for human factor considerations. Any necessary changes will be incorporated into Rev. 12 of this procedure. Procedure OP-002-005 will be updated by 12/15/94 to include any necessary human factoring enhancements.

   SUMMARY OF CHANGE

   This commitment is being closed. Human factors enhancements were incorporated into Rev. 12 of OP-002-005 as a result of a comprehensive formal assessment. Procedural human factors controls are now institutionalized by site procedures W2.109 and W2.110 and are applicable to all site procedures, including OP-002-005. Additional human factors reviews for Operations Department owned procedures, including OP-002-005, are dictated by Operating Instruction OI-019-000. Various Waterford 3 Commitments against procedures OI-019-000, W2.109 and W2.110 will ensure appropriate human factors controls are retained. Therefore, a specific commitment against OP-002-005 for human factor control is redundant and unnecessary.

2. COMMITMENT CHANGE NO. 2000-0011, Change to Procedure OP-002-005 Chemical and Volume Control

   ORIGINAL COMMITMENT DESCRIPTION

   Change procedure OP-002-005 Chemical and Volume Control to provide a single section of the procedure for routine blends to the Volume Control Tank.

   SUMMARY OF CHANGE

   This commitment is being closed to Commitment P-22047. This commitment is redundant to Commitment P-22047, (commitment made via letter W3F1-94-0180 in reply to violation IR 94-24-01) of which the Commitment Text states, "Procedure OP-002-005, Chemical and Volume Control has been changed to provide a single section of the procedure for routine blends to the Volume Control Tank.

3. COMMITMENT CHANGE NO. 2000-0013, Evaluate Procedure Requirements for Pre- and Post- Job Briefs

   ORIGINAL COMMITMENT DESCRIPTION

   Observation: CR 93-103 Corrective Actions/ brief Maintenance personnel & evaluate procedural requirements.

   SUMMARY OF CHANGE

   This commitment is being closed. P-21089 resulted from a condition in which Maintenance personnel inadvertently performed maintenance on an incorrect plant component. Current

Waterford 3 procedures and maintenance directives, including pre and post job briefs reinforce the use of STAR (Stop, Think, Act and Review) process whenever working on plant equipment. Use of the STAR process and various verifications performed prior to, during and after the performance of maintenance has significantly reduced the probability that work will be performed on incorrect components. Commitment P-21089 was generated when the STAR process was initially implemented at Waterford 3. STAR is now basic to all maintenance performed.

4. COMMITMENT CHANGE NO. 2000-0014, Termination/Determination Sheets Documentation Requirements

   ORIGINAL COMMITMENT DESCRIPTION

   Lineout and initials will not be used on Termination/Determination Sheets; new sheets will be used when changes are made.

   SUMMARY OF CHANGE

   This commitment is being closed. This commitment resulted from a single inspection finding and the technician admitted that he merely reacted to a mistake in the installation work package instructions. The current philosophy at Waterford 3 is for the technician to stop work when safe and have necessary changes made in work instructions or procedure prior to proceeding. The mindset for procedure compliance is prevalent and is reinforced via various mechanisms including pre/post job briefs, procedures and maintenance directives. Identified noncompliance is investigated through the corrective action process.

5. COMMITMENT CHANGE NO. 2000-0015, Component Cooling Water Design Pressure Rerate

   ORIGINAL COMMITMENT DESCRIPTION

   Resolve long standing issues - implement modification to provide Component Cooling Water design pressure rerate.

   SUMMARY OF CHANGE

   This commitment is being closed. The original commitment was to issue a Design Change Package (DCP) to rerate the design pressure of the Component Cooling Water (CCW) system to envelope the hydraulic transients caused during surveillances of system valves. DCP-3534 was initiated to rerate the system below the +21 elevation to a design pressure of 150 psig. While DCP-3534 was in development, DCP-3493 was implemented to minimize the CCW system hydraulic transients by modifying the logic and/or stroke times for certain system valves. The acceptance test for DCP-3493 recorded system pressures while perturbating the system by performing the routine system valve surveillances. The results recorded during acceptance testing indicated that the hydraulic transients in the CCW system had been reduced to below the current design pressure of the system. Therefore, a modification to rerate the CCW design pressure is not required. The commitment can be closed based on actions taken when DCP-3493 was implemented.

6. COMMITMENT CHANGE NO. 2000-0019, Guidance on the Proper Sequence of Nozzle Dam Installation

ORIGINAL COMMITMENT DESCRIPTION

The shift supervisor will be required to approve a nozzle dam configuration prior to initiating installation. Guidance on the proper sequence of Nozzle dam installation: a) complete installing or removing all nozzle dams on one loop; b) always install cold leg nozzle dams on one loop before installing that loop's hot leg dam; c) always remove hot leg nozzle dams prior to the cold leg nozzle dams on the same loop. This guidance will also be included in procedure RF-003-002. Procedure changes for the nozzle dams will be implemented prior to their next use at Waterford 3.

SUMMARY OF CHANGE

The requested change is the deletion of the committed requirement to complete installation and removal of nozzle dams on one loop (inferred as "one loop at a time"). Other requested minor text changes provide better clarity of the resultant requirements.

The revised commitment text reads: "The Shift Manager will be required to approve a nozzle dam configuration prior to initiating installation. Guidance on the proper sequence of nozzle dam installation: a) always install cold leg nozzle dams on a loop before installing that loop's hot leg dam; b) always remove hot leg nozzle dam prior to any cold leg nozzle dam on the same loop. This guidance will also be included in procedure RF-003-002. Procedure changes for the nozzle dams will be implemented prior to their next use at Waterford 3.

7. COMMITMENT CHANGE NO. 2000-0023, Revision to Procedure UNT-007-006, Housekeeping

ORIGINAL COMMITMENT DESCRIPTION

Revise Procedure UNT-007-006, Housekeeping, to identify the ponding concern in the cooling tower areas and the Fuel Handling Building.

SUMMARY OF CHANGE

Remove reference to the procedure UNT-007-006, Housekeeping, in the commitment text. A new procedure was written to incorporate the requirements of this commitment. The procedure will be referenced as an implementing document. There is no need to have the procedure number identified in the commitment text. The change will not affect the intent of the commitment.

The revised commitment text reads: "Reporting of Licensee Event Report (LER 99-010-00) -- Inadequate Pumping Capacity in the Dry Cooling Tower Areas Due to Inadequate Design Control -- Develop procedural guidance to identify the ponding concern in the cooling tower areas and the Fuel Handling Building."

8. COMMITMENT CHANGE NO. 2000-0024, 10CFR50.54(q) Review Requirements for Emergency Plan Training Procedures

ORIGINAL COMMITMENT DESCRIPTION

The following training procedures will receive a 10CFR50.54(q) review by Emergency Planning Department prior to approval by Training Department Management for future revisions: NTP-203, Emergency Plan Training; NTC-216, Emergency Plan Training Initial; NTC-217, Emergency Plan Continuing Training. Future Revisions of these procedures will be submitted to the NRC in the same manner as Emergency Plan Implementing Procedures.

SUMMARY OF CHANGE

Rev. 3 of NTP-203 incorporated the requirements previously included in NTC-216 and NTC-217. Therefore, NTC-216 & NTC-217 were deleted. The NRC was notified of the revision to NTP-203 and the deletion of the NTC procedures in Letter W3F1-99-0101 dated June 22, 1999. As part of the approval process for these procedures a 10 CFR 50.54(q) review was conducted and it was determined that the deletion of these procedures did not constitute a decrease in the effectiveness of the Emergency Plan.

The revised commitment text reads: "NTP-203 Emergency Plan Training, will receive a 10 CFR 50.54(q) review by the Emergency Planning Department prior to approval by Training Department for future revisions. Future revisions of this procedure will be submitted to the NRC in the same manner as Emergency Plan Implementing Procedure. "

9. COMMITMENT CHANGE NO. 2000-0026, Records Quality Review

ORIGINAL COMMITMENT DESCRIPTION

Quality related station modification packages are reviewed by the Operations QA group before final closure and transmittal to project files. A Quality Reviewer completes a QA review checklist on the modification package to ensure that records establishing proper review and other necessary records are retained. The QA review scope ensures that documents required by the modification package index and controlling procedures are included, proper review and approval is indicated on the records, applicable codes and quality standards are identified, test and inspection requirements are documented, and safety evaluation and design verification is performed. Comments from this review are tracked and closed out on a standard procedure review comments sheet, ensuring completeness of the modification package. The checklist, comments sheet and any additional records generated by the review are filed for storage. Similarly, Quality Related documents generated by the Plant Quality and Quality Assurance Groups in the performance of their duties are reviewed and retained in project files. These records include audit reports, inspector certification, hold tags, conditional release tags, various NDE documents, calibration records and NDE personnel qualification and training records.

SUMMARY OF CHANGE

This commitment is being closed. The quality assurance closure review of Station Modification Packages, Plant Changes and Engineering Requests is no longer required. The Documentation review is accomplished by Engineering during closure of the Design Change. There are other programs and processes in place to ensure that Station

Modification Packages, Plant Changes and Engineering Requests get the proper reviews (ERD process).

10. COMMITMENT CHANGE NO. 2000-0027, Wet Cooling Tower Overspray

ORIGINAL COMMITMENT DESCRIPTION

Implement Modification to reduce Wet Cooling Tower Overspray

SUMMARY OF CHANGE

Inspections in the Wet Cooling Tower area have resulted in corroded equipment being repaired. The modification originally identified to resolve the Wet Cooling Tower overspray issues was concluded to be very costly, could potentially reduce the airflow to the fans, and may not completely eliminate overspray. Therefore, it was determined to be more effective to inspect equipment (accumulators, valves, piping, supports, structural steel, cabinets, etc.) in the areas periodically (refueling interval) for corrosion. Any items found corroded will be cleaned and painted as necessary. If identified equipment's material condition cannot be restored by cleaning and painting then a corrective MAI will be generated.

The revised commitment text reads: "Implement a plan to mitigate the consequences of the Wet Cooling Tower Overspray."

11. COMMITMENT CHANGE NO. 2001-0002, Tracking of Equipment Out of Service

ORIGINAL COMMITMENT DESCRIPTION

Administrative Procedure OP-100-010 "Equipment out of Service" will be revised as follows: a) A Shift Reactor Operator, Senior Reactor Operator (other than the shift supervisor), or the Shift Technical Advisor will be required to perform and document an initial screening of LCO Applicability; b) The documentation for components taken out of service which have associated Technical Specification Actions will be separated from the documentation for components taken out of service which do not have Technical Specification Actions; c) Improvements will be made to facilitate tracking of plant conditions, which if changed, could affect the Limiting Condition for Operation requirements and d) Improvements will be made to allow for tracking of all Technical Specification Actions entered, not just Technical Specification Actions entered due to declaring equipment out of service.

SUMMARY OF CHANGE

The original commitment was made due to removing a Technical Specification related component from service without entering the appropriate Technical Specification. Part (A) Committed to performing and documenting an initial screening for Technical Specification applicability, which is presently done on the OP-100-010 TS/TRM Review Checklist. Operations is revising OP-100-010 to replace the existing TS/TRM Review Checklist with a Briefing Guideline, which will not require a signature. The procedure will still require an initial screening for TS/TRM applicability and will also require a shift briefing prior to removing TS/TRM equipment from service, which will establish shift concurrence and Shift Manager approval. The removal of the equipment from service and the applicable TS/TRM actions will be documented in the Station Log. Part (B) of this original commitment will also go away with this procedure change. Tracking and documentation of TS/TRM related equipment out of service for longer than a shift will still be performed by the use of an Equipment Out of Service (EOS) checklist. The Station Logs will provide the only documentation needed for TS/TRM related equipment out of service for less than a shift. The station Log identifies the individuals on shift; therefore accountability for TS/TRM entry decisions remains unchanged. Since the level of review required for TS/TRM applicability remains the same with this procedure change, this commitment change will preserve the original intent of the commitment. The purpose of the commitment change is to minimize the documentation burden associated with the EOS process. The effectiveness of the EOS process and identifying applicable TS/TRM actions will not be impacted.

The revised commitment text reads: "Administrative Procedure OP-100-010 "Equipment Out Of Service" will be revised as follows: a) A shift Reactor Operator, Senior Reactor Operator (other than the shift supervisor), or the Shift Technical Advisor will perform an initial screening of Limiting Condition of Operation applicability; b) Removal from service of equipment which have Technical Specification or Technical Requirements Manual Actions will be documented in the station log; c) Improvements will be made to facilitate tracking of plant conditions, which if changed, could affect the Limiting Condition for Operation requirements and d) Improvements will be made to allow for tracking of all Technical Specification Actions entered, not just Technical Specification Actions entered due to declaring equipment out of service."

12. COMMITMENT CHANGE NO. 2001-0003, Calibration of Wrenches

ORIGINAL COMMITMENT DESCRIPTION

Procedure MM-006-012 will be revised to require a calibration of the wrenches from once each working day to twice each working day. The calibration of the automatic cutoff impact wrench in step 8.2.5.1 of MM-006-012 will be the same as identified in ANSI N45.2.5-1974.

SUMMARY OF CHANGE

Commitment text is being revised to remove reference to procedure number and specific steps in the procedure because the potential for changing the number sequence occurs during each procedure revision which would require a commitment change evaluation form to be prepared each time the step number is changed during the revision process.

The revised commitment text reads: "The appropriate procedure will be revised to require a calibration of the wrenches from "once each working day" to "twice each working day". The calibration of the automatic cutoff impact wrench detailed in this procedure will be the same as identified in ANSI N45.2.5-1974."

13. COMMITMENT CHANGE NO. 2001-0004, Revise Operations Procedures

ORIGINAL COMMITMENT DESCRIPTION

Operations procedures will be revised to maintain CVR-401A&B closed. CVR-401A will only be opened for surveillance testing.

SUMMARY OF CHANGE

Commitment being deleted. Design Change DC-3502 deleted CVR-401B from penetration 53, the line was cut and capped. CVR-401B was placed in line with CVR-401A. The names were changed to CVR-400 and CVR-401. With this arrangement, both valves close on a Containment Isolation Actuation Signal isolating the instrument line. The concerns of the commitment were removed by implementation of the design change.

14. COMMITMENT CHANGE NO. 2001-0005, Proper Interpretation of Technical Specification 3.1.3.6

ORIGINAL COMMITMENT DESCRIPTION

Follow Up Item: Procedure OP-903-005 rev. 6. The inspector will follow up on proper interpretation of Technical Specification 3.1.3.6 to determine for future reference if all Control Element Assemblies in a group must be above 145 inches by all indications. The interpretation from NRR was that TS 3.1.3.6 required the entire group (all Control Element Assemblies in the group) to be above the TIL, due to the TS bases that state the basis for the TIL is to ensure minimum shutdown margin is maintained and the potential effects of a CEA ejection are limited to acceptable levels. As a result of NRR's interpretation, all the Control Element Assemblies in a group must then be in the position required to the group. In a separate but related issue, there was discussion with NRR regarding whether on February 22, 1991, CEA 68 could have been considered at or above the TIL of 145 inches if one CEA control display indicated 145.50 inches and the redundant CEA control display indicated 144.75 inches. The response from NRR was that the licensee should determine which indicator is correct, based on other redundant and/or diverse indications, if possible. If sufficient evidence does not exist to allow the lower indication to be discarded, then conservative call should be made as deemed appropriate by the licensee.

SUMMARY OF CHANGE

For single CEA misalignments, NUREG-1432 ACTIONS include provisions for both single and multiple CEA misalignments. Current Technical Specifications also include actions for both single and multiple CEA misalignments. Both versions of Technical Specifications also contain provisions for CEA Group misalignments. The Bases for NUREG-1432, ER-W3-99-0455, and ER-W3-98-021, provide justification for applying separate specifications for single CEA and Group misalignments. For CEA groups, operation beyond the transient insertion limit may result in a loss of Shut Down Margin and excessive peaking factors. When the regulating groups are inserted beyond the transient insertion limits, actions must be taken to either withdraw the regulating groups beyond the limits or to reduce Thermal Power to less than or equal to that time allowed for the actual CEA insertion limit. The Technical Specification allowed Action time provides a reasonable time to accomplish this, allowing the operator to deal with current plant conditions while limiting peaking factors to acceptable levels. Misalignment of a single CEA causes Xenon redistribution in the core to occur as soon as a CEA becomes misaligned and may result in excessive local linear heat rates, a distortion in radial power, a decrease in Departure from Nucleate Boiling Ratio (DNBR), and a small effect on Shutdown Margin. The effect of any misoperated CEA on the core power distribution will be assessed by the CEA calculators, and an appropriately augmented power distribution penalty factor will be supplied as input to the core protection calculators (CPCs). As the reactor core responds to the reactivity changes caused by the misoperated CEA and the ensuing reactor coolant and Doppler feedback effects, the CPCs will initiate a low DNBR or high local power density trip signal if specified acceptable fuel design limits are approached. The NRC has approved and therefore accepts the interpretations of NUREG-1432, accordingly, the latest interpretation accepted by the NRC would be that Technical Specification 3.1.3.1 applies to CEA misalignments within Groups and Technical Specification 3.1.3.6 applies to CEA Group misalignments.

The revised commitment text reads: "Follow-up Item: Procedure OP-903-005 Revision 6 "Control Element Assembly Operability Check". For the case in which a single rod is found inserted past the TILs, TS 3.1.3.1 (Action C) should be entered. As stated in ER-W3-98-0021, TS 3.1.3.6 should not be entered since TS 3.1.3.6 only applies to CEA group insertion, i.e. all Control Element Assemblies in Action A of LCO 3.1.3.6 should be entered, Regulating CEA Insertion Limits. This condition has been recognized by the NRC and the industry and thus is the reason why the two separate Tech Specs (3.1.3.1 and 3.1.3.6) are present. In conclusion, when a CEA group is inserted past the TILs, Action A of TS 3.1.3.6 should be entered since, although it is not known to be violated, shutdown margin can not be ensured. In a separate, but related issue, there was discussion with NRR regarding whether on February 22, 1991, CEA 68 could have been considered at or above the TIL of 145 inches if one CEA control display indicated 145.5 inches and the redundant CEA control display indicated 144.75 inches. The response from NRR was that the licensee should determine which indicator is correct, based on other redundant and/or diverse indications, if possible. If sufficient evidence does not exist to allow the lower indication to be discarded, then conservative call should be made as deemed appropriate by the licensee."

15. COMMITMENT CHANGE NO. 2001-0008, Enhancements and Clarifications to the Procedure for Modifying Clearances

ORIGINAL COMMITMENT DESCRIPTION

Procedure UNT-005-003 Clearance Request, Approval and Release will be reviewed for enhancements and will be revised to clarify review requirements for modified clearances. The corrective step to revise UNT-005-003 will be completed by September 30, 1997, at which time Waterford 3 will be in full compliance.

SUMMARY OF CHANGE

Commitment being deleted: Change 5 to revision 14 of UNT-005-003 accomplished implementation of this commitment. Since that procedure change, the tagging program has evolved to utilize a computer program system and UNT-005-003 has undergone two revisions. A new Nuclear Management Manual Procedure, OP-102, is currently in review which will assume implementation of the protective tagging program for all nuclear sites. UNT-005-003 will concurrently be scoped down such that it will only provide details specific to Waterford 3 in meeting the requirements and processes specified in OP-102. Since, adequate procedure change controls are in place to ensure procedural requirements of UNT-005-003 are not arbitrarily altered without proper technical and nuclear safety bases, this commitment provides no additional level of assurance above what already exists. Additionally, since OP-102 will soon implement the protective tagging program, this commitment will no longer be able to be implemented by UNT-005-003. It is, furthermore, not appropriate for OP-102 to implement this commitment since Waterford 3 is the only nuclear site currently bound by the commitment.