Entergy

Entergy Nuclear South
Entergy Operations, Inc
17265 River Road
Killona, LA 70066
Tel 504 739 6440
Fax 504 739 6698
kpeters@entergy.com

Ken Peters
Director, Nuclear Safety Assurance
Waterford 3

W3F1-2002-0099
A4.05
PR

November 27, 2002

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. Also included in this report is a summary of commitment changes for the same time period. This report covers the period from June 1, 2001 through May 31, 2002. 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,

K.J. Peters
Director, Nuclear Safety Assurance

KJP/LBB/ssf
Attachment

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

IE47

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

REPORT OF FACILITY CHANGES, TESTS, AND EXPERIMENTS

PER 10CFR50.59 and COMMITMENT CHANGES

JUNE 1, 2001 THROUGH MAY 31, 2002

WATERFORD 3
10CFR50.59 REPORT
ENTERGY OPERATIONS, INC.

JUNE 1, 2001 THROUGH MAY 31, 2002

TABLE OF CONTENTS

			Page No.
SUMMARY			1
I.	FACILITY CHANGES		
	A.	DESIGN CHANGES	1
	1.	2001-006; DCP-3521, Dry Cooling Tower Sump Pump Piping	1
	2.	2001-029; DCP-3521 Rev. 6, Reroute DCT Sump Pumps Discharge to CW System	2
	B.	MISCELLANEOUS EVALUATIONS	
	1.	2001-028; TRMC-01-008, Revise Technical Requirements Manual Requirements Related to Fire Protection and Control Room Evacuation	3
	2.	2002-008; Revision to Technical Requirements Manual and Offsite Dose Calculation Manual Concerning Radiological Environmental Monitoring	4
	3.	2002-011; EC-S99-005, Cycle 12 Core Reload	4
	C.	ENGINEERING REQUESTS	
	1.	1999-059; ER-W3-99-1037-00-00, Temporary Alteration for New Auxiliary Boiler	5
	2.	2000-026; ER-W3-98-0789-03-00, Modifications to Valves MS-401A & MS-401B	6
	3.	2000-039; ER-W3-99-1130-00-00, Smoke Detector for RAB 27, (+7 RAB), Key Issue Room	7
	4.	2001-001; ER-W3-01-0035-00-01, Repair Leaking Potable Water Piping Near Water Treatment Building	7
	5.	2001-012; ER-W3-00-0991-00-00, Eliminate the Inlet Chilled Water Switches from the Essential Chillers	8

		Page No.
6.	2001-016; ER-W3-99-0744-00-00, HVC-101 and HVC-102 Closure Time Limit	9
7.	2001-019; ER-W3-00-0858-00-00, Compensatory Actions for Fire Rated Assemblies in Containment	10
8.	2001-020; ER-W3-00-1018-00-00; Install More Accurate Feedwater Flow Meter to Support Appendix K Power Uprate	10
9.	2001-021-1; ER-W3-1999-0726-001; Generic Letter 96-06 Penetration Overpressurization	11
10.	2001-022; ER-W3-1999-0411-000; Replace Part Length Control Element Assemblies and Remove Four Element Control Element Assemblies	12
11.	2001-023; ER-W3-1999-0198-004; Replace Reactor Pressure Vessel Top Head Insulation with NUKON Insulation Blankets	13
12.	2001-025; ER-W3-2000-0106-000; Add a Backpressure Control Valve to the Bleedoff Line from the Reactor Coolant Pump Seals to the Volume Control Tank	14
13.	2001-026; ER-W3-2000-1018-002; Authorization for Use of the Leading Edge Flow Meter Check Plus as the Preferred Feedwater Flow input to Core Operating Limits Supervisory System	14
14.	2001-030; ER-W3-1998-1149-001; Replacement of Computer Static Uninterruptible Power Supply Battery Bank	15
15.	2002-002; ER-W3-01-1174-00-00; Deletion of Reactor Coolant Pump Sprinkler as a Credited Fire Protection System	16
16.	2002-003; ER-W3-2000-1009-001, Rev. 1: Replace the Standpipe in the Reactor Coolant System Level Measuring System	17
17.	2002-005; ER-W3-2001-1063-000, TRM Change to Revise Surveillance Frequency	17
18.	2002-006; ER-W3-2001-1133-000; Revise Setpoints for Shield Building Ventilation	18
19.	2002-007; ER-W3-2000-0574, Replace High Pressure Safety Injection Pump Rotors	19
20.	2002-009; ER-W3-2001-0305-00-00 Reactor Head Upgrades	20

		Page No.
21.	2002-0010; ER-W3-2001-0305-01-00; Design/Installation of Permanent Reactor Cavity Seal Ring	21
22.	2002-012; ER-W3-2001-0044-001; Steam Generator Thermal Liner Steel Strip Removal	22
23.	2002-013; ER-W3-2002-0184-000; Change Surveillance Applicability to Fire Barrier Penetration Seals	23
II.	PROCEDURE CHANGES	
A.	PLANT PROCEDURES	
1.	2002-004; STA-001-005, Leakage Testing of Air and Nitrogen Accumulators for Safety Related Valves	24
B.	SPECIAL TEST PROCEDURES	
1.	2001-031; STP 432049; Special Test Procedure to Perform Load Rejection of Emergency Diesel Generator A to Test Functionality of Voltage Regulator	25
III.	COMMITMENT CHANGES	
1.	2001-0014; Containment Closure	26
2.	2001-0015; Operations Shift Turnover	26
3.	2001-0016; Procedures Controlling Activities Affecting Pipe Supports	27
4.	2001-0017; Improper Implementation of the Condition Identification / Work Authorization Process	27
5.	2001-0018; Deficiency Tracking via Condition Identification / Work Authorization	28
6.	2001-019; Control of Painting Activities	28
7.	2001-0020; Prompt Identification of Leakage and Corrective Actions	29
8.	2001-0021; Ineffective Work Controls – Change in Work Scope	29
9.	2001-0022; Revise Work Authorization Procedure to Clarify Retest Change and Scope Change Wording	30
10.	2001-0023; Deficiency Tracking via Condition Identification / Work Authorization	30

		Page No.
11.	2001-0024; Inspection of Work and Restoration	31
12.	2001-0025; Maintenance Package Documentation	31
13.	2001-0027; Responsibility Changes for Planning and Maintenance Departments	32
14.	2001-0028; Post Trip Review Criteria	32
15.	2001-0029; Post Trip Review Task Group	32
16.	2001-0032; Submittal of Reload Analysis	33
17.	2002-0010; Vendor Interface Program	34
18.	2002-0011; Drill Participation Requirements	34

SUMMARY

This report provides a summary of the Waterford 3 Changes made pursuant to 10CFR50.59(c)(1) for the period from June 1, 2001, through May 31, 2002. This report also provides a summary of Commitment changes made during that same period.

Section I of the report identifies 28 facility changes.

Section II of the report identifies 2 procedure changes.

Section III of the report identifies 18 commitment changes.

I. FACILITY CHANGES

A. DESIGN CHANGES

1. 2001-006; DCP-3521, Dry Cooling Tower Sump Pump Piping

DESCRIPTION OF CHANGE

Vent lines with manual isolation valves will be added at two high points on the Dry Cooling Tower (DCT) sump pump discharge piping. In addition, the sump pump discharge Model 400 flow elements will be replaced with Model 500 flow elements, mounted on isolation valves.

REASON FOR CHANGE

This initial revision of this Design Change Package (DCP) provided a means of routing the Dry Cooling Tower sump discharge to the Circulating Water System. However, acceptance testing has revealed that the discharge capacity of the four DCT sump pumps is significantly less than predicted in Calculation EC-M97-029 Revision A. Ultrasonic examinations have confirmed that air is trapped in two high points on the discharge piping. The flow rate is gradually increasing as the air is swept from the piping, but Calculation EC-M99-010 Revision 1 assumes full capacity of the sump pumps immediately upon starting during a Probable Maximum Precipitation (PMP) rainfall event. Therefore, high point vents will be added to ensure the sump pump piping is filled and vented properly. In addition, operating experience has revealed that the DCT sump discharge flow elements collect small metallic particles on the magnetic sensor that must be periodically cleaned to ensure accurate flow measurement. Removal of the flow element requires that all four sump pumps be removed from service and that the system be drained before the instruments can be removed. Therefore, new flow elements will be installed that allow the instrument to be cranked out of the sump pump discharge pipe and an isolation valve will be provided to prevent the need for draining the sump pump piping.

50.59 EVALUATION

This evaluation addresses the potential affect on the integrity of the flood wall by the addition of vent lines to safety related Waste Management piping that is exterior to the flood wall and penetrates the flood wall. In addition, this evaluation addresses the potential affect on the integrity of the non-safety, non-seismic section of sump piping interior to the flood wall by the addition of new model flow elements with isolation valves. This evaluation concluded that the design of the vent lines provides adequate assurance that the ability of the flood wall to perform its design basis flood protection function will not be compromised and the addition of the new model flow elements with isolation valves will not affect the integrity of the non-safety, non-seismic section of sump piping interior to the flood wall. The sump pump system is not an accident initiator; therefore there is no increase in the frequency of occurrence of previously evaluated accidents. The sump pumps are used to mitigate the effects of probable maximum precipitation in the dry cooling tower area to maintain the operability of safety related equipment. This change does not affect the capacity or operating logic of the sump pumps. The integrity of the piping and exterior flood wall will be maintained and no new system interactions are created. Therefore, there are no unreviewed safety questions and this change does not require any Technical Specification changes.

2. 2001-029; DCP-3521 Rev. 6, Reroute Dry Cooling Tower Sump Pumps Discharge to Circulating Water System

DESCRIPTION OF CHANGE

DCP-3521 Rev. 6 revises the FSAR to remove unnecessary details relating to the 300 gpm minimum capacity of the electric motor driven Dry Cooling Tower (DCT) sump pumps. Change 2 to calculation EC-M99-010 Rev. 0 lowered the required minimum capacity of each electric motor driven sump pump that is needed during a Probable Maximum Precipitation (PMP) rainfall event from 300 gpm to 270 gpm.

REASON FOR CHANGE

Acceptance testing for DCP-3521 Rev. 5 revealed that the discharge capacity of the two motor driven sump pumps in DCT A is less than predicted when using the hydraulic model in Calculation EC-M97-029 Rev. A. The calculated capacity using test conditions in the hydraulic model was approximately 350 gpm and the measured capacity was approximately 307 gpm. When the measured capacity of 307 gpm is extrapolated to accident conditions using the hydraulic model it was determined that the flow would be less than the 300 gpm taken credit for in calculation EC-M99-010 Rev. 0, Change 1.

50.59 EVALUATION

This evaluation focused on the results of calculation EC-M99-010 Rev. 0, change 2, which established a new design basis for the portions of the sump pump system which protects safety related equipment in the DCT areas from ponding rainwater. This calculation documented the maximum potential depth of ponding rainwater in the Dry Cooling Tower area for the Probable Maximum Precipitation (PMP) and Standard Project Storm (SPS) rainfall events. This evaluation reflects that the motor driven sump pumps, with a minimum capacity of 270 gpm and supplemented with a diesel powered sump pump with a minimum capacity of 300 gpm in each train, are capable of protecting safety related equipment in the DCT areas during either the PMP or SPS rainfall events. The sump pump system will continue to function as originally designed and adequate pumping capacity is maintained to protect safety related equipment. No new system interactions are introduced. It is concluded that this change does not require prior NRC approval.

B. MISCELLANEOUS EVALUATIONS

1. 2001-028; TRMC-01-008, Revise Technical Requirements Manual Requirements Related to Fire Protection and Control Room Evacuation

DESCRIPTION OF CHANGE

The proposed change replaces the current Technical Requirements Manual (TRM) requirement for a plant shutdown (in conjunction with the Technical Specifications) if the Charging and Component Cooling Water (CCW) pumps, and Essential Services Chilled Water (CHW) chillers and associated pumps are inoperable for greater than 7 days with a requirement to establish a fire watch patrol. This change relaxes a self imposed TRM requirement established to ensure the fire protected pumps and chillers for affected systems remain operable for safe shutdown following evacuation of the control room. The current requirements specifically require the inoperable B train pump (and chillers for CHW) in the CCW, CHW and Charging systems to be restored to operable status within 4 days or enter the associated Technical Specification (TS). The associated TS requires restoration within 72 hours. If the pump cannot be restored within 72 hours, a plant shutdown must be initiated. The proposed change will require a fire watch to be established within one hour if any pump (or chiller for CHW) is inoperable for greater than 7 days. This change encompasses both more restrictive and less restrictive changes to the current TRM requirement. The more restrictive change imposes a requirement for all the affected system pumps (and chillers for CHW) to be Operable. The less restrictive change eliminates the requirement for a plant shutdown when pumps cannot be restored within the allowed outage time with a requirement for deployment of fire watch patrols in the affected fire areas.

REASON FOR CHANGE

The TRM is being changed because the B train Charging and CCW pumps and the B train CHW chiller and pump are the protected trains in the control room and cable spreading rooms. However, the A, B or AB train may be the protected train in other fire areas as documented in EC-F00-026, "Post Fire Shutdown Analysis." Therefore, the TRM change is required to impose requirements when Charging, CCW, and CHW Appendix R protected trains are inoperable in all fire areas. Also, this change relaxes the overly restrictive requirements to shutdown when Appendix R protected components are inoperable with a requirement for hourly fire watch patrols.

50.59 EVALUATION

Revision of the Appendix R designated TRMs does not result in a change requiring prior NRC approval. This change revises the self imposed TRM requirements on Appendix R protected components. This change encompasses both more restrictive and less restrictive changes to the current TRM requirement. The more restrictive changes impose a requirement for Appendix R protected trains to be Operable. This will ensure action is taken when the Appendix R protected train is inoperable for more than 7 days. The less restrictive change replaces a plant shutdown with a requirement to establish an hourly fire watch. This is acceptable based on defense in depth associated with Appendix R (required fire detection and suppression, and manual extinguishing credited in the fire hazards analysis via the fire brigade). This change eliminates the transient of a plant shutdown and replaces it with a requirement to establish fire watch patrols in the affected fire areas.

2. 2002-008; Revision to Technical Requirements Manual and Offsite Dose Calculation Manual Concerning Radiological Environmental Monitoring

DESCRIPTION OF CHANGE

Revise the Technical Requirements Manual (TRM) and Offsite Dose Calculation Manual (ODCM). Add footnote to specify that the normal sampling location for the Turbine Building Industrial Waste Sump for radioactive effluents is at the same location that is currently used for Louisiana Pollutant Discharge Elimination System permit compliance. Revise footnote to allow the contents of the Regenerative Waste Tank or the Filter Flush Tank, when found to contain radioactivity, to be transferred or directed to an already existing and monitored effluent release path instead of requiring them to be transferred to the Liquid Waste Management System. Revise Radioactive Environmental Monitoring Plan (REMP) sampling requirements, frequencies and locations.

REASON FOR CHANGE

This change is being done to clarify the sampling location for the Turbine Building Industrial Waste Sump, revise the specified destination for the Regenerative Waste Tank or the Filter Flush Tank when they are found to contain radioactivity for more operational flexibility, remove obsolete information for REMP samples that no longer exist, and add/change/remove REMP sampling locations and sampling/analysis frequencies to better conform to Entergy Nuclear South's standardized REMP program specification and due to more accurate physical location information.

50.59 EVALUATION

Accurately specifying the sample point for Turbine Building Industrial Waste Sump liquid radioactive effluent sampling will maintain the same level of protection for public safety with respect to liquid radioactivity releases. Diversion of the Regenerative Waste Tank or the Filter Flush Tank to destinations other than the Liquid Waste Management System will not reduce the level of protection for public safety with respect to liquid radioactivity releases. Alterations to the REMP sampling and analysis program scope contained within this change will not decrease the effectiveness of the REMP to adequately monitor radioactivity in the environment surrounding Waterford 3. Thus, all changes evaluated will not adversely impact nuclear safety and will continue to ensure the protection of the public. Prior NRC approval is not required.

3. 2002-011; EC-S99-005, Cycle 12 Core Reload

DESCRIPTION OF CHANGE

The Waterford 3 Cycle 12 core will contain 125 irradiated Batch S and Batch T assemblies from the Cycle 11 core, and 92 new Batch U assemblies. There are minor mechanical design differences between the Batch U assemblies and the Batch S and T assemblies primarily due to the transition of fuel vendor manufacturing operations from Hematite, Missouri to Columbia, South Carolina. The rated thermal power in Cycle 12 will be increased by 1.5% to 3441 MWt in accordance with 10CFR50 Appendix K. The Cycle 12 core was designed on the basis of nominal cycle energy (at this uprated power) of 512 Effective Full

Power Days (at 50-ppm boron) for a nominal Cycle 11 energy of 475 EFPD. The Cycle 12 Safety Analysis Groundrules document was used as input, not the Cycle 12 reload analyses. Changes to the Groundrules for Cycle 12 relative to Cycle 11 were primarily related to fuel management and core design, the planned Appendix K power Uprate, the replacement of Part Length Control Element Assemblies (CEAs) with full length, full strength CEAs and removal of the 4 finger CEAs, and replacement of all incore instrumentation. The Reload Analysis Report was prepared by Westinghouse to document the results of the reload safety analyses. The Core Operating Limits Report for Cycle 12 was prepared based on the results of the safety analyses and the setpoint process of the fuel vendor. Core Operating Limits Supervisory System and Core Protection Calculator system addressable constant and Reload Data Book changes for Cycle 12 were developed to implement the requirements of the safety analysis and Setpoint Process. There were no Technical Specification changes for Cycle 12 that were a direct result of the reload analyses. Technical Specification changes related to part length CEA replacement and 4 finger CEA and Appendix K Power uprate were submitted as part of those respective projects.

REASON FOR CHANGE

The Waterford 3 reactor must be refueled for Cycle 12 operation.

50.59 EVALUATION

All Cycle 12 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. All equipment important to safety will function in the same manner with the Cycle 12 reload core as with the previous core. There is no increase in the probability or consequences of equipment malfunctions and the possibility of a different type of accident or malfunction is not created. Based on a review of the vendor's reload analysis results, the FSAR and the Bases of the Technical Specifications, no design basis limits for a fission product barrier will be exceeded. The Cycle 12 core was designed and evaluated using NRC approved analysis methodology under an approved quality assurance program. No new methodologies were required to verify that previous safety analyses are applicable to Cycle 12 or to perform reanalysis of any events. There is no deviation from the methods of evaluation described in the FSAR. It is concluded that prior NRC approval is not required.

C. ENGINEERING REQUESTS

1. 1999-059; ER-W3-99-1037-00-00, Temporary Alteration for New Auxiliary Boiler

DESCRIPTION OF CHANGE

This change will replace the originally installed auxiliary boiler with a temporary replacement boiler. The replacement boiler will serve the same function as the original; however, piping and wiring changes are required to mechanically and electrically tie in the new boiler.

REASON FOR CHANGE

The temporary auxiliary boiler replacement is required due to a failure of the originally installed boiler. This temporary configuration will be required until a boiler can be permanently installed

50.59 EVALUATION

The auxiliary boiler is non safety related and is used to supply auxiliary steam loads when main steam is not available. The auxiliary boiler is not an accident initiator, nor is it used to mitigate the consequences of any accidents or malfunctions. The 50.59 Evaluation concludes that the temporary replacement of the auxiliary boiler will not result in an Unreviewed Safety Question, nor will it result in a reduction of the margin of safety of any Technical Specification.

2. 2000-026; ER-W3-98-0789-03-00, Modifications to Valves MS-401A & MS-401B

DESCRIPTION OF CHANGE

Valves MS-401A(B) are the steam supply valves to the Emergency Feedwater Pump AB Turbine (Emergency Feedwater Pump Turbine or Terry Turbine). The six inch diameter Anchor Darling gate valves will be replaced with new four inch diameter (6 X 4 X 6) GE Sentinel gate valves. The Limitorque SMB-00 Actuators will be replaced with SMB-0 Actuators. The existing one second opening time delay that allows MS-401B to open first will be eliminated. The existing six second hold at seventeen percent open for MS-401A(B) will be eliminated. The new valves will open on a linear twenty-second ramp. The input to the Terry Turbine control system will be recalibrated to start the turbine ramp up when the new valves reach the 50% open position (the current ramp starts when the valves reach 20% open). New power supply cables will be run to the MS-401A motor to raise the minimum degraded voltage to 90 volts.

REASON FOR CHANGE

The currently installed MS-401A(B) valves were not designed to be installed with the stems horizontal. They have required extensive maintenance due to packing and stem leaks and the valve seats have been reworked to the extent that there is insufficient remaining seat material to allow for additional rework. The currently installed Limitorque SMB-00 motor operators for MS-401A(B) are marginally sized and are limited by the design output torque capability of the DC motors under degraded voltage conditions.

50.59 EVALUATION

The proposed change does not cause the parent systems to be operated outside of their design or test limits, negatively affect any system interfaces or result in an increase in challenges to safety systems or systems important to safety. The proposed activity does not result in a change from one frequency class to a more frequent class or an increase in frequency within a given class or result in an increase in radioactive releases or introduce new release pathways. The proposed change does not involve an unreviewed safety question.

3. 2000-039; ER-W3-99-1130-00-00, Smoke Detector for Reactor Auxiliary Building RAB 27, (RAB +7), Key Issue Room

DESCRIPTION OF CHANGE

Add smoke detectors in RAB +7 (Fire Area RAB 27) and revise Technical Requirements Manual Table 3.3-11 accordingly.

REASON FOR CHANGE

Per National Fire Protection Association (NFPA) requirements, detectors are required in areas on RAB +7.

50.59 EVALUATION

The new detectors in fire area RAB 27 added per NFPA requirements do not have any negative impact on the safe operation of the plant, the ability to safely shutdown or any accident previously evaluated in the FSAR. The change does not introduce any concerns related to a new or unanalyzed accident or equipment malfunction nor does it result in a potential for release of radioactive material to the environment.

4. 2001-001; ER-W3-01-0035-00-01, Repair Leaking Potable Water Piping Near Water Treatment Building.

DESCRIPTION OF CHANGE

During repair of line 8PW3-22 it is necessary to reroute the piping to avoid excavating under the Water Treatment Building. ER-W3-01-0023-00-01 authorizes minor rerouting of the potable water piping to facilitate repairs, and the installation of an isolation valve to facilitate future maintenance. However, installing an isolation valve will require updating FSAR figure 9.2-9 to reflect plant configuration. In addition, FSAR figure 9.2-9 will also be enhanced to identify the Condensate Polisher Building.

REASON FOR CHANGE

This change is necessary to repair an underground potable water leak. This change will allow minor rerouting of potable water piping leading into the Water Treatment Building to facilitate repair of an underground potable water leak. Rerouting the piping will avoid excavating under the Water Treatment Building or routing piping through the building's foundation. Adding the potable water isolation valve will allow potable water to isolate from the Water Treatment Building, Condensate Polisher Building and Turbine Building. This will permit maintenance and repairs of potable water lines in these areas without affecting potable water supply to the rest of the plant.

50.59 EVALUATION

The proposed changes to repair the potable water leak and add an isolation valve will not create an unreviewed safety question. The function of the potable water system will not be affected by this change. The additional ball valve will facilitate future maintenance and will remain open during normal operation. The potable water system is not an initiator of any accidents nor is it relied on for accident mitigation. The final piping configuration will comply with the original piping specification and will not degrade the system operation. Therefore probability and consequences of malfunctions remain unchanged. There are no Technical Specifications relating to the potable water system. In addition, updating FSAR figure 9.2-9 to identify the Condensate Polisher Building is only an editorial enhancement and does not impact plant procedures or configuration

5. 2001-012; ER-W3-00-0991-00-00, Eliminate the Inlet Chilled Water Switches from the Essential Chillers

DESCRIPTION OF CHANGE

In order to improve reliability of the Essential Chillers, this design change will implement four improvements to the existing Essential Chillers: 1) The essential chillers are equipped with two low water temperature switches. One switch monitors the temperature of water entering the chiller and the other monitors the temperature of the water exiting the chiller. These switches are used to trip the chiller if the chilled water temperature becomes too low. The chiller will automatically restart when the chilled water temperature increases past the reset deadband of the switches. This low water temperature switch that monitors the temperature of the water entering the chiller was provided as optional equipment and will be removed by this design change. 2) The low water temperature switches, which monitor temperature of the water exiting the chillers are currently mounted on surfaces which become cooler than the ambient temperature surrounding the chillers when the chillers are operating. As a result, condensation forms in these switches and causes premature switch failures. This design change will relocate these switches to surfaces that remain at the ambient temperature in the areas around the chillers when they are operating. 3) A failure of the Swagelok fitting upstream of an essential chiller condenser drain valve has occurred in the past. The cause of the failure was determined to be fatigue due to vibration. This design change will modify the existing configuration to one that is less susceptible to fatigue failure. 4) There are 3 pressure switches associated with each of the chillers which do not have isolation valves. Subsequently, calibration of any of these pressure switches requires transferring the refrigerant from the affected chiller to the storage tanks. This design change will install isolation valves which will allow the pressure switches to be calibrated without the need to transfer the refrigerant to the storage tanks.

REASON FOR CHANGE

The Essential Chillers have been identified as having an excessive number of functional failures and unplanned Technical Specification Limiting Condition of Operation (LCO) entries. The chillers are not meeting the Maintenance Rule performance goals established for this equipment and were subsequently classified as a(1) in accordance with the Maintenance Rule. The a(1) classification requires an action plan to be prepared that will result in this equipment being returned to a(2) status. The proposed changes are a part of the action plan established for the essential chillers.

50.59 EVALUATION

The proposed changes will increase the reliability of the Essential Chillers. No new system interfaces are created by the proposed changes. Normal and post accident operation of the chillers will be enhanced. In addition, no new failure mechanisms will be introduced. Therefore it is concluded that this modification will not affect the safety or environmental aspects described in any licensing basis documents, will not reduce the margin of safety as defined in the basis for any Technical Specifications and no unreviewed safety questions are created.

6. 2001-016; ER-W3-99-0744-00-00, HVC-101 and HVC-102 Closure Time Limit

DESCRIPTION OF CHANGE

Various discrepancies exist between the In Service Testing (IST) basis document, DBD-038 Safety Related HVAC Control Room, and FSAR Section 6.4.2.2 Control Room Air Conditioning System Design. Currently, DBD-038 states that HVC-101 and 102 have a 2-second stroke time. However, FSAR section 6.4.2.2 states that the closure time for the normal outdoor air isolation valves is less than three seconds. ER-W3-99-0744-00-00 will change FSAR section 6.4.2.2 to clarify that HVC-101 and 102 have a design basis closure time of 2 seconds. This is more conservative than the "less than three seconds" closure time described in the FSAR. There are additional discrepancies between the IST basis document and FSAR section 6.4.2.2 for the Control Room Emergency Outside Air Intake valves HVC-201A(B), HVC-202A(B), HVC-203A(B) and HVC-204A(B). The FSAR states that the closure time for the emergency outdoor air isolation valves is less than five seconds. The IST basis document does not list a stroke time and the maximum allowable testing time is greater than 5 seconds for all the Emergency Outside Air Intake valves. It has been determined that there is no requirement for a specific closure time of these valves after reviewing the Design Basis Document and Design Basis Calculations. The valve's flow path is expected to be closed during normal operations unless testing. The closure time of less than 5 seconds in Section 6.4.2.2. of the FSAR will be removed, since there is no design basis for this value.

REASON FOR CHANGE

The FSAR, Design Basis Documents, and IST Basis Documents should not have discrepancies. This change is being made to correct the problems originally discovered by CR-WF3-1997-2728.

50.59 EVALUATION

Revising FSAR Section 6.4.2.2 to clarify that HVC-101 and 102 have a design basis closure time of 2 seconds instead of the "less than 3 seconds" closure time described in the FSAR, is conservative. Revising FSAR Section 6.4.2.2 to remove the "less than five seconds" closure time of the emergency outside air isolation valves is a clarification to resolve the discrepancy between the FSAR, the Design Basis Documents, and the IST Basis Documents. Currently, the design value came from a vendor specification. The valves have no safety function to close within 5 seconds. There are no physical changes made to the valves or their operation. These valves are not initiators of any accidents but are used for accident mitigation. Since the 2 second closure time is more conservative than the "less than 3 seconds" time, there is no increase in consequences of accidents or malfunctions. This change does not affect the basis for any of the Technical Specifications.

7. 2001-019; ER-W3-00-0858-00-00, Technical Requirements Manual Change - Compensatory Actions for Fire Rated Assemblies in Containment

DESCRIPTION OF CHANGE

This change provides clarification to the ACTION Statement of Technical Requirements Manual (TRM) 3.7.11, regarding compensatory actions necessary for Fire Rated Assemblies located inside the containment. This clarification is based on achieving consistency in TRM actions and meeting ALARA goals.

REASON FOR CHANGE

The current TRM Action Statement lacks specific direction for firewatch activities within the containment as are found in similar TRM sections. This change is administrative in nature and provides extension of previously accepted industry practices.

50.59 EVALUATION

This change involves the compensatory actions associated with fire rated assemblies. This Evaluation has determined no impact to nuclear safety is presented, no new accidents, consequences or probabilities presented and the safety margins presently in place have been maintained. Additionally, the Fire Protection Program as previously approved has been maintained.

8. 2001-020; ER-W3-00-1018-00-00; Install More Accurate Feedwater Flow Meter to Support Appendix K Power Uprate

DESCRIPTION OF CHANGE

The proposed activity represents a change to the plant as described in the FSAR. The change adds equipment that will be used to support an improved feedwater flow measurement accuracy to the Plant Monitoring Computer to ultimately be used in support of a 10CFR50 Appendix K Power Uprate. This configuration change does not approve the new thermal power level or the use of the new Leading Edge Flow Meter (LEFM) to measure mass flow rate of the feedwater. It does, however, authorize the installation of the equipment to support the new power level. All equipment installed is classified as non-safety and non-seismic, with the exception of the spool pieces, which are seismically analyzed to comply with the American National Standards Institute (ANSI) Standard B31.1. The proposed change will install two spool pieces; one digital electronics cabinet connected to the Plant Monitoring Computer, thirty-two ultrasonic transducers and two gage pressure transmitters.

REASON FOR CHANGE

This equipment will be installed to support an approximate thermal power uprate of 1.5%. The new feedwater flow measurement instrumentation will measure the flow of feedwater more accurately than the currently used technology and provide this information to the Core Operating Limit Supervisory System inside of the Plant Monitoring Computer. The new equipment is required to provide support for the Appendix K Power Uprate.

50.59 EVALUATION

All accidents in chapter 15 of the FSAR have been reviewed and it has been determined they are not adversely affected by the proposed change. The addition of this equipment will not increase the possibility or probability of any accident evaluated in the FSAR, or create the possibility of a new accident not previously identified in the FSAR. The impact on plant safety systems has been evaluated and has been determined to be non-significant. The addition of the equipment complies with all design standards currently published and has been evaluated and documented in the calculations identified in the reference section of this evaluation.

9. 2001-021-1; ER-W3-1999-0726-001 Generic Letter 96-06 Penetration Overpressurization

DESCRIPTION OF CHANGE

Changes are proposed to ten piping penetrations to ensure that thermally induced overpressurization resulting from plant heat up or post accident conditions (LOCA) does not affect the integrity of the Containment Isolation System. The changes proposed include: Physical change to the piping system by the addition of a relief valve to Penetration No. 42 (Waste Management), 43 (Boron Management) and 59 (Safety Injection); Administrative procedure control which ensures the penetration/system is flushed or in service eliminating possible overpressurization during a design basis event for penetrations No. 5 & 6 (Blowdown), 28, 29 & 30 (Primary Sampling) and 52 & 68 (Secondary Sampling). These administrative controls have been included in letter W3F1-2001-0061, which requests NRC approval prior to implementation for the seven penetrations that will be controlled administratively.

REASON FOR CHANGE

Generic Letter 96-06 requested licensees to determine whether piping systems that penetrate the containment are susceptible to thermal expansion of fluid between the inboard and outboard isolation valves, such that overpressurization of the piping could occur. The requested actions also stated that in addition to the individual licensee's postulated accident conditions, this item should be reviewed with respect to the scenarios referenced in the generic letter. Design Engineering evaluated all containment piping penetrations and determined that 17 are of concern for this type of overpressurization. Conditions reports were issued to address the subject penetrations and determine plant operability along with assignment of corrective actions. As stated above, this ER in conjunction with letter W3F1-2001-0061 addresses the actions necessary for resolution for ten of the affected penetrations, while ER-W3-99-0726-00-02 previously addressed the final resolution for the remaining seven.

50.59 EVALUATION

The proposed relief valve additions and administrative procedure controls will ensure the identified susceptible penetrations are adequately protected against the affects of potential thermally induced overpressurization. The design function of the penetrations will therefore be preserved. The subject systems will not be operated outside of their design or test limits, affect any system interfaces, or result in an increase in challenges to safety or important to safety systems. The subject activities will not result in a change to accident probability, consequence, type or affect any safety or important to safety equipment. Prior NRC approval is not required for the installation of relief valves on three penetrations. The NRC has provided prior approval for the administrative controls on seven of the penetrations via License Amendment No. 179.

10. 2001-022; ER-W3-1999-0411-000; Replace Part Length Control Element Assemblies and Remove Four Element Control Element Assemblies

DESCRIPTION OF CHANGE

This change will replace part length Control Element Assemblies (CEAs) with full length CEAs and remove the four finger CEAs from the core. In support of these configuration changes, wiring changes in the Control Element Drive Mechanism Control System (CEDMCS) and Core Protection Calculations (CPCs), work on CP-2 and software changes associated with the Control Element Assembly Calculator (CEAC) display and the Plant Monitoring Computer (PMC) are required. This work will be performed during Refueling Outage 11, when the plant is in mode 5 or 6. During this time, CEDMCS, CPCs, CEAC and the Plant Monitoring Computer are not required.

REASON FOR CHANGE

Four finger CEAs have reached the end of usable life and are being removed from the reactor. Justification of the removal will be contained in the Cycle 12 Reload Process as it relates to the FSAR accidents. This configuration change will address the removal of the four element CEAs, those control changes (to CEDMC and CPC) and changes to the operator control panel necessary to affect the removal of the four finger CEAs and the physical removal of the four finger CEAs. Part length CEAs have reached the end of usable life. A like for like replacement was considered for these CEAs, however, consideration was given to replacement with full length CEAs to enhance operational control and to ready the plant for a full power uprate. Justification of the replacement will be contained in the Cycle 12 Reload Process as it related to the FSAR accidents. This configuration change will address the replacement of the part length CEA with full length CEA, those control changes (to CEDMC, CPCs, CP-2, CEACs and PMC), and control panel changes necessary to affect the replacement of the part length CEA with full length CEA and group re-assignments.

50.59 EVALUATION

The changes proposed by this ER do not adversely affect the function of the CEAs, CEDMCS, the CPCs or the control panels in the control room. The proposed changes will re-wire the CEDMCS and the CPCs for the replacement of the PLCEA with FLCEA and the removal of the four element CEAs, and group reassignments. The proposed changes do not affect the qualification of the structures, systems or components (SSCs), nor do the changes affect the design or safety function of the SSCs. The associated change to the Technical Specifications did receive prior NRC approval via License Amendment No. 182.

11. 2001-023; ER-W3-1999-0198-004; Replace Reactor Pressure Vessel Top Head Insulation with NUKON Insulation Blankets

DESCRIPTION OF CHANGE

The reactor pressure vessel (RPV) provides the second fission product barrier which prevents the release of fission products from the reactor core to the containment, and it provides for core cooling during normal plant evolutions and anticipated operational occurrences, to prevent core damage. The RPV transfers heat from the fuel rods to the primary water, which transfers it to the secondary system to produce steam. Currently the RPV is insulated with Transco brand reflective insulation. This evaluation will allow all the RPV top head insulation to be replaced with NUKON insulation blankets in order to facilitate inspections of Control Element Drive Mechanism (CEDM) nozzles for leakage. This ER will also add NUKON insulation to the latrolet branch connection being installed on the pressurizer surge line.

REASON FOR CHANGE

In 2000 and 2001 Oconee 1, 2 and 3 and ANO-1 performed visual inspections of their RPV top heads and found evidence of RCS leakage from CEDM nozzles. The NRC has issued a bulletin (BL-2001-01) mandating inspections of CEDM nozzles to ensure they are not leaking. The ER will authorize installation of thermal blanket insulation on the vessel head to facilitate visual inspections next outage and during subsequent outages. Thermal blankets are easier to remove and re-install, thereby resulting in lower radiation dose. ER-W3-1999-0184 is adding a latrolet and blind flange connection to the surge line to allow for isolation of the line for repairs on the pressurizer during outages.

50.59 EVALUATION

The NUKON blanket type insulation is a quilted, light-density, semi-rigid fibrous glass insulation (a pad). NUKON is attached to the vessel with Velcro quick release straps. NUKON has been evaluated and approved several times for replacing the existing reflective, encapsulated fiberglass insulation within containment. NUKON has been qualified and installed on the reactor head, pressurizer, reactor coolant pumps and steam generators. The NRC in December 1978 found NUKON insulation acceptable. Performance Contracting Inc. (PCI) Test Report ESD-TR-10F dated May, 1991 reflects that the transport velocity (i.e., water velocity to carry insulation in the water stream) of shredded fiberglass insulation is between 0.17 and 0.2 feet per second. Entergy calculation MN(Q)-6-35, revision 1, reflects that the velocity of the water flowing through the Safety Injection Sump screen, at the maximum design flow rate, is 0.136 feet per second. This velocity is less than the minimum transport velocity of the fiberglass as determined by testing performed by PCI. Therefore, the insulation is not expected to be transferred to the Safety Injection Sump screens if it were to be damaged and fall to the floor of the Containment building. This evaluation reflects that changes proposed by this ER, will not reduce the margin of safety as defined in the basis of any Technical Specifications or safety analysis, and there are no unreviewed safety questions.

12. 2001-025; ER-W3-2000-0106-000; Add a Backpressure Control Valve to the Bleedoff Line from the Reactor Coolant Pump Seals to the Volume Control Tank

DESCRIPTION OF CHANGE

The change will add a direct operated backpressure control valve in the controlled bleedoff line leading from the Reactor Coolant Pump (RCP) seals to the Volume Control Tank (VCT). A manually operated throttle valve will be installed in a path parallel to the new backpressure control valve and will maintain a minimal amount of flow. In addition, two manual isolation valves will be provided to isolate the backpressure control valve for maintenance and a pressure gage will be installed to provide local pressure indication.

REASON FOR CHANGE

Backpressure to the RCP seals is not automatically controlled during all Reactor Coolant System (RCS) and VCT operating conditions. Adjustments to RCP Controlled Bleedoff (CBO) to VCT manual isolation valve are currently required during shutdown and startup to ensure reliable RCP seal operation. However, the valve was not originally intended to be used as a throttle valve and does not automatically control backpressure. Since CBO flow discharges to the VCT, CBO backpressure fluctuates proportional to VCT pressure. During VCT purging the pressure in the VCT typically varies from 15 to 50 psig. In addition, backpressure to the RCP seals can fluctuate based on plant operating conditions. Total CBO flow during normal operations is approximately 6 gpm, but reduces to 2 gpm during startups and shutdowns. This flow reduction causes the pressure drop in the CBO line to be significantly lower during startups and shutdowns. As the pressure drop in the CBO line is reduced, the backpressure to the RCP seals is also reduced proportionately. These fluctuations can potentially impact seal reliability.

50.59 EVALUATION

This evaluation concludes that this change will not increase the probability or consequences of an accident or malfunction of equipment important to safety. In addition, margin of safety will not be reduced by this change. The new backpressure control valve and accessories will provide a pressure boundary equivalent to the original piping and will not create an unreviewed safety question.

13. 2001-026; ER-W3-2000-1018-002; Authorization for Use of the Leading Edge Flow Monitor Check Plus as the Preferred Feedwater Flow input to the Core Operating Limits Supervisory System

DESCRIPTION OF CHANGE

This ER has evaluated the balance of plant instrumentation and controls based upon the expected plant parameters following an Appendix K power uprate. The evaluation found that only the steam flow venturi calibration requires an adjustment to compensate for the lower steam header pressure. The ER authorized the use of the Leading Edge Flow Monitor (LEFM) Check Plus as the preferred feedwater flow input to Core Operating Limits Supervisory System (COLSS) at 3390 MWt power. COLSS will not use this input for the Secondary Calorimetric. The proposed change does not authorize an increase in the power of the plant, only those changes necessary to ready the plant for the Appendix K power uprate.

REASON FOR CHANGE

Appendix K power uprate project will change the operating point of the plant to increase the MW output. As a result of this project, several instrument scales, calibrations or control systems may require adjustments. The adjustments will be made during the refueling outage to minimize the impact on the plant when power uprate is implemented.

50.59 EVALUATION

The changes proposed by this ER do not affect the safety function of structures, systems and components as described in the FSAR. Changes to chapter 7 of the FSAR are made to clarify information in the FSAR and revise the span of the main steam transmitter. A change to the calibration span of the main steam flow system will have no impact on the frequency of occurrence of an accident. Steam flow instruments are not accident initiators as they do not have direct control functions. Calibration spans for instruments are designed to ensure proper operation of the instrument loop. This change has no affect on the output devices or on the actual main steam flow of the system. Changes to the main steam flow calibration will not result in a design basis limit for a fission product barrier being exceeded or altered. The change to the scale will ensure that the instrument reading is as accurate as possible during normal plant operations. The ultrasonic flow meter to be utilized as the preferred feedwater flow in the Core Operating Limits Supervisory System (COLSS) will meet or exceed the current instrument accuracies of the feedwater and steam flow venturis. Therefore COLSS will continue to monitor the core limits with the same or better accuracy, and those limits will not be exceeded.

14. 2001-030; ER-W3-1998-1149-001; Replacement of Computer Static Uninterruptible Power Supply Battery Bank

DESCRIPTION OF CHANGE

Replace existing Computer Static Uninterruptible Power Supply (SUPS) battery Exide type EX-13B with Exide type ES-13B.

REASON FOR CHANGE

The existing EX-13B battery cell has reached end of life. The EX-13B is no longer manufactured by Exide and the vendor has recommended replacement type ES-13B.

50.59 EVALUATION

The new battery is a "one for one" replacement for the existing battery. The critical performance characteristics of the replacement battery are adequate for the loading requirements of the computer SUPS. The replacement battery does not adversely impact any system, structure or component considered important to safety. Fission product barriers are not degraded and the new battery does not adversely impact any reactor coolant pressure boundary or containment performance. The change does not impact any Technical Specification or margins of safety. The computer battery and associated SUPS are classified as non-safety and non-seismic. FSAR sections 7.5a, 8.3.2 and Technical Specification section 3.4.8.2 do not describe the computer battery or its associated SUPS. The proposed change replaces the existing obsolete aged battery with a new battery of lesser capacity. The new battery can supply the connected Computer SUPS load for approximately 43 minutes. The existing battery had sufficient capacity to supply the load for approximately 60 minutes. There are no design basis criteria for the battery to supply the computer SUPS for any specified length of time. This change will not impact the function of the connected loads since the normal and alternate class 1E ac busses will be restored via the Emergency Diesel Generator after a loss of offsite power and automatically resume power to the computer SUPS after 2 minutes. In the remote event of a single failure in the A train, the new battery will power the Plant Monitoring Computer SUPS for a shorter duration. This is acceptable as the Plant Monitoring Computer SUPS is not required for safe shutdown of the plant post accident. Therefore, the new battery will adequately provide uninterruptible power to the Computer SUPS.

15. 2002-002; ER-W3-01-1174-00-00; Deletion of Reactor Coolant Pump Sprinkler as a Credited Fire Protection System

DESCRIPTION OF CHANGE

Delete the Reactor Coolant Pump (RCP) sprinkler (and associated detection system) as a credited fire protection system. This involves a change to the Technical Requirements Manual and the FSAR. This change does not physically remove or alter the sprinkler or detection systems. This ER removes credit for the systems and eliminates requirements to test and maintain the systems as functional and operable systems.

REASON FOR CHANGE

NRC guidelines, standards and expectations do not require automatic sprinkler or detection systems on reactor coolant pumps. The NRC requires an oil collection system. Originally the reactor coolant pump sprinkler systems were installed to satisfy property insurer requirements. These requirements no longer exist. The existing sprinkler systems are hydraulically deficient and can not be restored to an adequate design within reasonable and justifiable costs.

50.59 EVALUATION

NRC guidelines and requirements do not require sprinkler/detection systems for the reactor coolant pumps. The protection provided for the reactor coolant pumps consist of an oil collection system as detailed in 10CFR50 Appendix R. The oil collection system is adequate to address the hazards of the area and satisfies NRC requirements. The RCP sprinkler/detection systems are not accident initiators and therefore there is no affect on the frequency of occurrence of accidents. The sprinkler/detection system is a seismic system whose failure was analyzed on initial design and installation. The ER makes no physical change to the plant, therefore there is no affect on the likelihood of occurrence of a malfunction. The RCP sprinkler/detection system is not credited as an accident mitigation system. The only "accident" is a fire event that is addressed by the oil collection system which remains unchanged. Therefore there is no increase in the consequences of accidents or malfunctions. No physical changes are being made and this does not create the possibility for different types of accidents or malfunctions with a different result. The RCP sprinkler/detection system has no impact on design basis limits for fission product barriers.

This change does not involve any methods of evaluation described in the FSAR. NRC approval for not crediting the reactor coolant pump sprinkler/detection system is not required.

16. 2002-003; ER-W3-2000-1009-001, Rev. 1: Replace the Standpipe in the Reactor Coolant System Level Measuring System

DESCRIPTION OF CHANGE

Replace the existing standpipe in the reactor coolant shutdown level measuring system by installing the Mansell Level Monitoring Instrument (MLMI) as the second, independent means of measuring reactor coolant system level. This system will provide control room indication and annunciation using a computer system, indicating equipment and existing control room indication, annunciation and a plant monitoring computer point.

REASON FOR CHANGE

The existing standpipe instrument utilizing the resistance temperature detectors (RTDs) does not provide accurate and reliable level monitoring and it has cost valuable maintenance, operations and refueling hours to compensate for these problems. Due to the slow reaction time of the standpipe instrument, operations has had to suspend RCS drain down to allow the standpipe instrument to "catch up". This has cost refueling time in the past and ultimately slows down refueling operations.

50.59 EVALUATION

All design and administrative control aspects of the original Reactor Coolant System Level Measuring System (RCSLMS) credited by the NRC for approval will be maintained. The new system will be more accurate and reliable, and will have a faster response time than the current standpipe design. An instrument line rupture is an initiating event for a FSAR decrease in reactor coolant inventory scenario. However, the change does not adversely affect the performance or reliability of the system. The new components meet the same design specifications for material and construction as the existing system. The RCSLMS is listed in the FSAR as a method of detecting certain Shutdown Cooling system component failures. However, the proposed change does not change or degrade any actions described or assumed in an accident analysis. The new system will operate during the same operational modes as the original remote indication portion of the RCSLMS and the change will not place additional reliance on any safety system. The change does not create any new system interfaces. The change will not affect any design basis limit, or any method of evaluation of any design basis or safety analysis limit. Therefore the proposed change does not affect the accidents and malfunctions previously evaluated in the FSAR or their potential to cause accidents or malfunctions.

17. 2002-005; ER-W3-2001-1063-000, TRM Change to Revise Surveillance Frequency

DESCRIPTION OF CHANGE

Revision of TRM 4.7.10.1.3a surveillance frequency for Diesel Fire Pump starting batteries from once per 7 days to once per 31 days. This change uses performance based system history and guidance from Nuclear Electric Insurance Limited Loss Control Standards, "Performance Based Analysis for Testing and Maintenance".

REASON FOR CHANGE

System history has indicated that no acceptance criteria failures have taken place within the past two years. Additionally, system design features (one pump meeting the largest system flow demand) provide for dual battery banks, effectively establishing a redundant fire pump arrangement (1 electric driven fire pump and 2 diesel driven fire pumps). Thus, no adverse impact to the level of fire protection provided, the safe shutdown requirements or nuclear safety are presented.

50.59 EVALUATION

The fire protection system is not an accident initiator, therefore there is no increase in the frequency of occurrences of accidents. This is an administrative change to surveillance requirements that has no influence on the probability of a fire occurring. Because no other systems are impacted and because the fire protection water supply system is provided with redundant pumping capability, the likelihood of occurrence of a malfunction of the fire protection water supply system or other systems remains unchanged. As the change in no way affects the operation or design functions of the fire protection water supply system, there is no change in the consequences of accidents, malfunctions or fire events previously evaluated. This change involves only the performance based frequency extension of surveillances associated with the diesel fire pump batteries, it does not create the possibility for a different type of accident or malfunction. There is no interface or impact with any of the fission product barriers or their design limits. This change does not involve a method of evaluation described in the FSAR.

18. 2002-006; ER-W3-2001-1133-000 Revise Setpoints for Shield Building Ventilation

DESCRIPTION OF CHANGE

The setpoints for Shield Building Ventilation will be revised to ensure that the system will maintain a negative pressure of 0.25 inwc in the Annulus during and following a design basis accident. The setpoint to start the exhaust mode of operation is to be revised to -3.0 inwc. In addition, the alarm, which would alert the control room of problems with the Shield Building Ventilation system, will be revised to ensure timely notification of a system malfunction.

REASON FOR CHANGE

Calculation EC-M88-025 was revised to consider the worst case environment of the annulus following a design basis accident. In order to maintain a negative pressure in the entire annulus, a negative pressure of 1.742 inwc should be maintained at the location of the instrument tap. Calculation EC-I01-009 also considers additional process measurement uncertainty due to wind effects on the shield building. This effect has been calculated to be -0.625 inwc. ER-W3-2201-1133-000 documents the new setpoint as -3.0 inwc, which includes the instrument uncertainty per EC-I93-036. A new alarm setpoint of -2.7 inwc has been established to allow for timely notification of a system malfunction.

50.59 EVALUATION

Changing the Shield Building Ventilation System (SBVS) setpoint for the exhaust mode of operation from -1 inwc to -3 inwc will impact the timing and duration of the SBVS discharge flow to the environment and in turn will impact the control room and offsite radiological dose consequences. EC-S96-011 evaluates the impact of this change on the post-LOCA offsite and control room doses. The increase in all the doses are less than 10% of the available margin between the existing doses and the acceptable limits provided in 10CFR100, GDC 19 and SRP 6.4. SBVS operation mitigates the consequences of fission product barrier (fuel clad and RCS) failures but has no protective function for the fission product barriers. The control room alarm changes will not affect the overall safety function of the structure, system or component (SSC). The alarm has been designed to alert the control room when a problem with the SBVS exists. The proposed changes do not affect the qualification of the SSC, nor do the changes affect the design or safety function of the SSC. Note that the current licensing basis (FSAR Section 6.2.3) differs from the original SER (section 6.5.1.3) related to SBVS post-accident operation setpoints and system flow rates. However, the current licensing basis description in the FSAR and operation matches the safety analysis calculations and SBVS function. The basis for approving the original SER, maintaining the annulus at a negative pressure, remains valid.

19. 2002-007; ER-W3-2000-0574, Replace High Pressure Safety Injection Pump Rotors

DESCRIPTION OF CHANGE

The pump rotors for High Pressure Safety Injection (HPSI) Pumps A, B and A/B may be sequentially replaced with a reworked precision balanced rotor. The HPSI A rotor will be replaced first because this pump exhibits vibration that is currently in the ASME Section XI "Alert" level. The proposed change may be installed in any or all of the HPSI pumps.

REASON FOR CHANGE

As part of the Waterford 3 Inservice Test Plan HPSI Pump vibration is monitored in accordance with Section XI of the ASME Boiler and Pressure Vessel Code. On 1/19/98 the HPSI A 4H and 3H vibration levels exceeded the "alert" value and the pump was placed on increased test frequency as required by the Code. Since that time 4H vibration levels have remained above the alert level and 3H vibration levels have decreased to just below the alert level. The current HPSI A pump vibration levels are about twice the vibration levels of the HPSI B and HPSI AB pumps. The three Waterford 3 and the identical 3 ANO-2 HPSI pumps are the only pumps of this design in existence. These pumps have a history of high vibration levels and have historically required significant maintenance after relatively low operating hours. The high vibration levels are the result of a combination of design and operational factors. The purpose of the proposed change is to reduce the high vibration levels in the HPSI pumps by replacing the existing pump rotors with reworked precision balanced rotors.

50.59 EVALUATION

The proposed change to the HPSI pumps will lower pump vibration levels and maintenance and improve the reliability of the pumps. There will be no impact on the remainder of the HPSI system or on any other plant systems, structures or components. There is no significant impact on HPSI pump flow, head or net positive suction head requirements (pump performance is expected to improve slightly). The proposed changes to the HPSI pumps will improve the pump reliability and actually decrease the likelihood of their malfunction. The function of the HPSI pumps is to mitigate the effects of accidents evaluated in the FSAR. The proposed changes will have the effect of improving the operation, availability and reliability of the HPSI pumps. Also, since the existing impellers will be reused, the hydraulic performance of the HPSI pumps will not be significantly affected in terms of discharge head or pressure, nor are the pump NPSH requirements increased (a slight improvement in pump performance is expected). Since the performance of the HSPI pumps is not being significantly changed by the proposed modification their ability to mitigate an accident or malfunction will also remain unchanged. Since the proposed changes improve the operation and durability of the HPSI pumps and do not significantly change the performance of the pumps or their associated systems, there are no changes to any fission product barrier design basis limits. This change does not involve methods of evaluation described in the FSAR. There are no accidents evaluated in the FSAR that are caused by a failure of a HPSI pump. The proposed change does not involve a change in a design basis limit for a fission product barrier or in design basis evaluation methods as described in the FSAR.

20. 2002-009; ER-W3-2001-0305-00-00 Reactor Head Upgrades

DESCRIPTION OF CHANGE

The Control Element Drive Mechanism (CEDM) Cooling System is a subsystem of the Containment Cooling System, and is designed to remove heat generated by the CEDM magnetic jack coil elements. Containment air is drawn through the cooling shroud for the magnetic jack coil elements to the CEDM cooling system. The heated air is cooled by Component Cooling Water cooling coils, and is discharged back to the containment through the cooling fans. The modification will simplify the ductwork configuration, eliminating virtually all of the existing ductwork. This is accomplished by extending the cooling shroud and completely enclosing the CEDM area. The airflow direction through the CEDM assemblies is reversed by this activity. The existing CEDM cooling fans and cooling coils are retained. The modification will provide sufficient airflow to maintain the original design cooling requirements for each CEDM location. The heat removed from the reactor head area and transferred to the cooling coils by the CEDM cooling system is unchanged by the modification. The volume and temperature of the air returned to containment is the same as before the modification. Thus, there will be no effect on the ambient containment temperature or the containment cooling system.

REASON FOR CHANGE

The CEDM Cooling System modification is intended to simplify the disassembly and re-assembly of the reactor vessel head in support of refueling activities. The current system requires many manhours of labor in a high radiation area at a time when the radiation levels are still quite high. The original design of the CEDM cooling system includes the use of closure head exhaust duct assemblies (clamshells) which must be removed along with associated ductwork at the beginning and reinstalled at the end of each refueling outage. Removal and transport of the clamshells and associated ductwork results in excessive exposure and increased safety challenges to personnel. Additionally, a significant amount of polar crane time is required at the beginning and end of each outage. This modification will result in reduced personnel radiation exposure and outage time.

50.59 EVALUATION

The CEDM cooling system modification has been designed for all applicable loading conditions, including seismic, thermal, pressure, and deadweight, using accepted codes and standards. The reactor head lift rig continues to meet NUREG-0612 requirements. The revised flow configuration provides sufficient flow to cool the CEDMs, and to transfer the heat to the Component Cooling Water system and thus has no effect on containment ambient temperature or the containment cooling system. The pressure drop across the CEDMs remains within the cooling fan operating parameters. The effect on containment net free volume is within the established limits. As a result of this evaluation it is concluded that this activity does not meet any of the criteria of 10CFR50.59 paragraph (c)(2), and therefore obtaining prior NRC approval is not required to implement this activity.

21. 2002-0010; ER-W3-2001-0305-01-00; Design/Installation of Permanent Reactor Cavity Seal Ring

DESCRIPTION OF CHANGE

The current reactor cavity seal ring will be replaced with a permanent cavity seal ring. The current seal ring is a temporary metal ring that is installed prior to flooding the refueling pool for refueling the vessel, and then removed following completion of the refueling operation. The temporary seal ring is stored during plant operation on the refueling floor. The permanent cavity seal ring is a permanent installation and as such, remains in place during all modes of plant operation. The permanent cavity seal ring serves as a watertight seal between the reactor vessel seal ring and the embedment plate on the floor of the refueling pool during refueling operations when the refueling pool is flooded, thus allowing no leakage to the reactor cavity. Hatches are provided that allow for airflow through the reactor cavity during normal operation when the hatch covers are removed. The hatch covers are installed prior to filling the refueling pool, providing a water tight seal for flooding the refueling pool. Orientation of the hatch openings on the seal ring allow access to the reactor cavity area for inspection and maintenance, including access to the ex-core nuclear instrumentation. The hatch covers will be removed from containment during plant operation.

REASON FOR CHANGE

The current seal ring can only be installed after shutdown of the reactor and in the beginning of the refueling outage period. This results in excessive exposure and safety concerns for personnel each refueling outage. In addition, installation and removal times for the seal ring add time to the outage period, require the use of the polar crane, and require excessive personnel resources. Implementation of the permanent cavity seal ring will significantly reduce personnel exposure and safety risks for personnel working in the refueling cavity. The installation of the permanent cavity seal ring will also aid in reducing critical path activities at the beginning and end of the refueling outage by eliminating installation/removal activities that were required for the current seal ring.

50.59 EVALUATION

The permanent cavity seal ring has been designed for all applicable loading conditions, including seismic, thermal, refueling water head, and dropped fuel assembly, using accepted codes and standards. Leak before break has been applied to eliminate from the seal ring design the dynamic loads resulting from postulated breaks in the reactor coolant system hot leg and cold leg piping, and is consistent with the Waterford 3 licensing basis. No tributary lines are included in this application of leak before break. Hatches included in the seal ring design provide adequate air flow for reactor cavity ventilation. The pressure drop across the cavity with the seal ring remains within the reactor cavity cooling system operating parameters. Emergency Core Cooling System analyses and Safety Injection Sump recirculation flow paths inside containment are not adversely affected. As a result of this evaluation, it is concluded that this activity does not meet any of the criteria of 10CFR50.59 paragraph (c)(2), and therefore obtaining prior NRC approval is not required to implement this activity.

22. 2002-012; ER-W3-2001-0044-001; Steam Generator Thermal Liner Steel Strip Removal

DESCRIPTION OF CHANGE

The Steam Generator feedwater nozzles are lined with a thermal sleeve to channel feedwater flow from the nozzle to a distribution box, where it splits into two semi-circular feedrings. A mechanical seal prevents leakage between the outside surface of the thermal sleeve and the inside surface of the distribution box. The mechanical seal consists of a stainless steel split o-ring held against a thin stainless steel strip bearing surface by a two-piece, bolted clamp. Inspection of the Steam Generator during Refuel 11 indicated the thermal liner O-ring for Steam Generator #1 was severed and overlapped. In addition, the thermal liner's stainless steel strip on Steam Generator #2 was found out of position. This change will remove the thermal liner's stainless steel strip, O-ring and bolted clamp from Steam Generator #1. Also, the stainless steel strip will be removed from Steam Generator #2.

REASON FOR CHANGE

The thermal sleeve mechanical seal components are being removed to minimize the possibility of steam generator tube damage caused by impingement of loose parts. Removing these items will eliminate the possibility of these degraded parts entering the steam generator tubes and causing steam generator tube damage.

50.59 EVALUATION

Removal of the steam generator thermal sleeve's mechanical seal components does not adversely affect Waterford 3's Licensing Basis. The change may slightly increase leakage from the steam generator distribution box, which conflicts with guidance provided in Branch Technical Position ASB 10-2. However, this leakage will not impact the design basis function of the steam generators. Feedwater leaking from the distribution box may impinge on the feedwater nozzle, which may result in increased thermal fatigue on the nozzle. A feedwater nozzle that has fatigued to the point of failure could result in one of the accidents described in section 5.2.3 of the FSAR, a Feedwater System Pipe Break, or a Loss of Normal Feedwater Flow with an Active Failure in the Steam Bypass System. However, the clearance between the inside diameter of the distribution box and the outside diameter of the thermal sleeve is very small, and the pressure differential between the areas inside the distribution box and the feedwater nozzle is also small. Therefore, the impingement flow is expected to be negligible, and will not affect the original stress analysis of the nozzle. The steam generator vendor concurs with this reasoning and has stated that removing the o-ring and clamp does not affect the original design basis of the steam generators. Therefore, the increase in frequency of occurrence of one of the accidents mentioned is only minimal. This change will not affect the ability of the steam generator to be used as a heat sink. Removing the seal components from the thermal sleeve will not increase the consequences of an accident. Leakage from the steam generator distribution box can result in an increased rate at which water drains from the steam generator when level falls below the ring and feedwater flow is interrupted. Consequently, drain down of the feedring could occur following a loss of feedwater event. This could result in steam within the feedring being condensed once sub-cooled feedwater flow is re-established and causing the feedring to partially collapse. However, this is only an operational issue and does not affect nuclear safety. Combustion Engineering states that severe damage to the feedwater ring will affect plant operation, but will not affect the ability of the steam generator to be used as a heat sink. Thus, even if a severe water hammer event did occur, the steam generator will still be able to perform its design safety function and can be used for a safe shutdown of the plant. This change only involves removing seal components from the steam generator thermal liner. Removing these parts does not introduce any new components or system interactions that could create an accident of a different type than previously analyzed. Eliminating the thermal liner seal components will reduce the possibility of seal parts being carried into the stream generator tubes and causing damage. Hence this change will reduce the risk of damaging a fission product barrier. In addition, the ability of the steam generator to remove heat from the RCS will not be reduced as a result of this change. This change does not involve any methods of evaluation described in the FSAR.

23. 2002-013; ER-W3-2002-0184-000; Revision of TRM Change Surveillance Applicability to Fire Barrier Penetration Seals

DESCRIPTION OF CHANGE

Revision of TRM 4.7.11.1c surveillance to provide clarification of surveillance applicability to those fire barrier penetration seals which are accessible. This change is in response to CR-WF3-2002-00209 and implements elements of the approved fire protection program as described in PEIR 50067.

REASON FOR CHANGE

Condition report CR-WF3-2002-00209 identified discrepancies within the performance methodology of ME-003-006, and specifically the selection of the statistical 10% sample. Evaluation of the CR determined that application of the Technical Requirements Manual surveillance requirements should be only to those seals determined to be accessible based on previous analysis and conversations with NRC contained in PEIR 50067.

50.59 EVALUATION

This is a change to the surveillance requirements of TRM section 4.7.11.1c. This TRM section does not impact the accident scenarios presented in Chapter 15 of the FSAR, but rather affects only a fire event, which is not classified as a design basis accident. The surveillance activity application at the component level is being revised to clarify application to "accessible" fire barrier penetration seals. This surveillance requirement is specific to passive elements of the fire protection system and therefore, in no way influences the probability of a fire event. Because no systems are impacted or altered by this change, the likelihood of occurrence of malfunctions of the fire protection system or other plant systems remains unchanged. The change to the surveillance requirement only provides for a change in application of the surveillance requirement at the individual component (fire barrier penetration seal) level. As this change in no way affects the operation or design functions of the fire rated assemblies, there is no change to the consequences of accidents or malfunctions or of fire events previously considered or presented. This change involves the performance application of a statistical sampling methodology surveillance requirement associated with fire rated assemblies, specifically fire barrier penetration seals. Because the surveillance continues to provide assurance the fire barrier will fulfill its design function, no new or alternative accident scenarios are presented by this change. The accident of plant event associated with this equipment is a fire, with no concurrent design basis accident. Therefore, the surveillance requirement associated with the referenced fire rated assemblies was determined not to affect any current, nor present any new accident scenarios. There is no interface of impact with the plant's fission product barriers or design limits presented by this TRM change. Therefore, no limits are exceeded, challenged or altered which could affect fission product barriers or their design limits. This change does not involve a method of evaluation previously described in the FSAR. This analysis established that no safety impact was presented. This TRM change maintains a consistent level of fire protection and surveillance requirements for systems as described in the TRM. Information contained in PEIR 50067 demonstrates the issue to have previously gain NRR acceptance as an appropriate alternative for meeting the surveillance requirement.

II. PROCEDURE CHANGES

A. PLANT PROCEDURES

1. 2002-004; STA-001-005, Leakage Testing of Air and Nitrogen Accumulators for Safety Related Valves

DESCRIPTION OF CHANGE

This revision to STA-001-005 eliminates a "should" statement for leak testing of safety-related accumulators in Modes 5 or 6. This will allow the testing of certain accumulators in Modes 1-4 as allowed by Technical Specifications. The revision also eliminates an unnecessary obvious precaution which stated tests could not be performed that would place the plant in an unsafe condition. Several editorial changes/enhancements are also incorporated by this revision.

REASON FOR CHANGE

The purpose of this revision is to allow the testing of some nitrogen and instrument air accumulators in Modes 1-4 as allowed by Technical Specifications. This will reduce refueling outage scope and resources. Accumulator testing has been performed mostly during refueling outages in the past, most likely because time was available during outages to do this testing and outages provided more flexibility for performing repairs if a test failed. With shortened refueling outages, not as much time is available to perform this testing, much of which can be done at power with no adverse consequences. Additionally, some Technical Specification Limiting Conditions for Operation, which previously precluded certain accumulator tests on line, have been relaxed such that the testing is not restricted to outages. Accumulator testing is driven by GL 88-14 and several Licensing Commitments, which require periodic testing of the accumulators. These documents do not preclude testing in Modes 1-4. The STA-001-005 procedure sections are already set up for testing in Modes 1-4 as allowed by Tech Specs. Accumulators which must be tested in Modes 5 & 6 (because testing would result in plant transients/shutdowns) will continue to be tested during outages as assured by the limitations of STA-001-005 and plant processes for scheduling/reviewing work.

50.59 EVALUATION

The revision to STA-001-005 does not result in operating any plant systems in a manner which has not been previously analyzed. The testing remains within the bounds of normal system operation as allowed by plant procedures and the Technical Specifications. The revision does not impact the possibility, likelihood or consequences of any accident. This revision also has no impact on the Inservice Testing Plan and STA-001-005 will continue to implement IST requirements for various Instrument Air System and Nitrogen System check valves. Accumulators for safety related valves are discussed in the FSAR, Sections 9.3.1 and 9.3.9, but testing of the accumulators is not described. This revision does not change the testing methodology or bases for the tests.

B. SPECIAL TEST PROCEDURES

1. 2001-031; STP 432049 Special Test Procedure to Perform Load Rejection of Emergency Diesel Generator A to Test Functionality of Voltage Regulator

DESCRIPTION OF CHANGE

The voltage regulator for Emergency Diesel Generator (EDG) A is being replaced with an identical unit. This test will verify the functionality of the replacement unit. EDG A will be loaded between 4000 and 4400 KW. EDG A load will be reduced to 500 KW and 0.5 MVAR. EDG A output breaker will be opened. Verification will be made that EDG A rejects load of greater than 498 KW while maintaining generator voltage less than or equal to 5023 volts. This test sequence will verify the ability of EDG A to accept load, maintain steady state conditions, and reject the largest single load while maintaining acceptable voltage range.

REASON FOR CHANGE

The purpose of the Special Test Procedure is to test the functionality of the EDG A voltage regulator.

50.59 EVALUATION

This special test does not require prior NRC approval. This test does not affect the design basis or configuration of any structure, system or component. The test does not affect safety analysis. The test is a load rejection test. A full load rejection while at power was evaluated by Waterford 3 and the NRC and found acceptable in a Safety Evaluation Report dated July 21, 2000. Review of the full load rejection test data from Refueling Outage 9 indicated that the voltage on the 4140 volt safety bus dropped approximately 2 percent and stabilized in about 0.5 seconds. The test data confirmed a full load rejection was a relative minor transient and well within the capability of the loads on the safety buses. This special test will constitute a significantly less load rejection, approximately 500 KV, than a full load rejection. Therefore, the special test should not have an adverse affect on the safety bus. This test will also demonstrate that the voltage regulator successfully dampens the transient voltages at the output of the EDG. Test criteria will ensure that the voltage transients experienced on the safety bus during the load rejection are within plus or minus 5 percent of the initial test voltage, with stabilization within 1 second.

III. COMMITMENT CHANGES

1. COMMITMENT CHANGE NO. 2001-0014, Containment Closure

ORIGINAL COMMITMENT DESCRIPTION

Waterford 3 will implement the appropriate measures for the condition with water level 23 feet above the fuel assemblies to ensure containment closure can be established prior to the initiation of boiling. Note: the time to boil was determined to be one hour and the time to core uncovery was determined to be 27.74 hours based on decay heat at 4 days after shutdown.

SUMMARY OF CHANGE

This commitment is being deleted. Waterford 3 Technical Specification 3.9.8.2 requires all containment penetrations providing direct access from the containment atmosphere to the outside atmosphere to be closed within 4 hours. Commitments P-25527 and P-25534 establish continuing compliance with commitment A-24489 which was implemented and closed. These commitments require implementation of the appropriate measures to ensure that containment closure is established prior to initiation of boiling (1 hour). This is not consistent with the Waterford 3 Technical Specifications or NUREG-1432, "Standard Technical Specifications Combustion Engineering Plants". The basis for containment closure is to ensure that a release of radioactive material within containment will be restricted from leakage to the environment and the operability and closure restrictions required by Technical Specifications are sufficient to restrict radioactive material release from fuel element rupture based on the lack of containment pressurization potential while in the refueling mode. The time to core uncovery (with subsequent fuel rupture) is 27.74 hours and the requirement for containment closure in the Technical Specifications is 4 hours. Thus, the 4 hour requirement in Technical Specifications is very conservative with respect to time to core uncovery. Also, doses, from any boil-off, prior to closure of penetrations providing direct access from the containment atmosphere to the outside atmosphere within 4 hours is bounded by the fuel handling accident. Therefore, deletion of this commitment and compliance with the Technical Specification required closure time of within 4 hours is sufficient to ensure the containment is closed prior to radioactive release from fuel element rupture.

2. COMMITMENT CHANGE NO. 2001-0015, Operations Shift Turnover

ORIGINAL COMMITMENT DESCRIPTION

The following enhancements will be made to OP-100-007, Shift Turnover: Provide guidance to review the database containing deficiencies /abnormalities associated with watchstation when performing control room shift turnover; Expand control turnover sheets to allow for one entire entry page for abnormal conditions; Expand control turnover sheets to allow for one entire entry page for Technical Specification and Technical Requirements Manual entries. This page should be split up such that one half of the page is for Technical Specifications and the other half for Technical Requirements Manuals; Add the requirement for two board walkdowns per shift per control room watchstation to OP-100-007; Add sign off for one board walkdown per control room watchstander to the appropriate turnover sheets.

SUMMARY OF CHANGE

This commitment change deletes two items related to board walkdowns from the original commitment text. Specifically the commitment to perform two walkdowns per shift and the walkdown signoff for each watchstander. In letter W3F1-98-0029 (response to IR 97-26 Notice of Violation) Waterford 3 stated that OP-100-007 was revised to simplify the control board walkdown for the Control Room Staff by requiring it once per shift by the Primary Nuclear Plant Operator and Secondary Nuclear Plant Operator, and included several other new requirements. The new requirements are captured under commitment P-24954, which supersedes this commitment for board walkdown requirements. The NRC closed out the 97-26 Notice of Violation in Inspection Report 99-20 stating that corrective actions were reasonable and complete.

3. COMMITMENT CHANGE NO. 2001-0016, Procedures Controlling Activities Affecting Pipe Supports

ORIGINAL COMMITMENT DESCRIPTION

Procedure MM-012-001, "Pipe Hanger/Support Installations", will control the installation, removal, modification or repair of pipe supports. Any damage to the support installation or any discrepancy that is identified will be resolved using procedure UNT-5-002, "Condition Identification and Work Authorization".

SUMMARY OF CHANGE

Waterford 3 implements all plant maintenance and construction work in accordance with the system wide work management system (WMS). Work is tracked to completion and documented a Maintenance Action Item (MAI). The WMS requires implementation and review of work during the process. Reviews and inspections are incorporated in the work instructions or are invoked in pre approved procedures. Integral to the WMS is the corrective actions program. A condition report is generated if plant equipment is damaged during the course of MAI implementation. The mindset for work instruction/procedure compliance is prevalent and is reinforced via various directives. Identified noncompliance is investigated and tracked through completion in accordance with the MAI and/or corrective action processes. A commitment to control work specific to pipe supports is not required. This commitment is being deleted.

4. COMMITMENT CHANGE NO. 2001-0017, Improper Implementation of the Condition Identification / Work Authorization Process

ORIGINAL COMMITMENT DESCRIPTION

Procedure UNT-005-002 has recently been revised to clarify requirements and responsibilities related to the corrective maintenance process by subdividing the procedure action into discrete subsections such that individuals performing a portion of the Condition Identification / Work Authorization Process (CIWA) process know exactly what they are required to do.

SUMMARY OF CHANGE

This commitment was initiated in 1985 and resulted from an inspection finding stating that the CIWA process was confusing to the worker. The current philosophy at Waterford 3 is for the worker to stop work when safe and have necessary changes or clarifications made in the work instructions or procedure prior to proceeding. The mindset for work instruction and procedure compliance is prevalent and is reinforced via various mechanisms including pre/post job briefs, walkdowns, department meetings, coaching, STAR, peer checking, procedures and maintenance directives. Identified noncompliance is investigated through the corrective action process. Human performance errors are investigated with face to face investigation and interviews with the workers and a designated human performance trained individual. Since the CIWA process, Entergy Nuclear South has implemented an electronic work management system that automatically tracks work action items. Confusion in the process does not appear to be a recurring problem. There is no requirement to maintain this commitment and it is being deleted.

5. COMMITMENT CHANGE NO. 2001-0018, Deficiency Tracking via Condition Identification / Work Authorization

ORIGINAL COMMITMENT DESCRIPTION

Upon completion of Condition Identification / Work Authorization (CIWA) work, the responsible department supervisor reviews the work package for completeness and forwards the CIWA work package to Planning & Scheduling for closure on the master tracking system. The master tracking system identifies all archived and active CIWAs at the plant site. Tight administrative controls are instituted to assure proper input and extraction of data to/from the master tracking system.

SUMMARY OF CHANGE

All work performed in the operating nuclear plant is performed in accordance with the Work Management System (WMS) and is documented on a Maintenance Action Item (MAI). Reviews and work performed and documented on an MAI is tracked electronically in the WMS database. In addition to tracking work via MAI there are other processes in place to identify, track and control activities at the nuclear facility. Conditions adverse to quality are identified and tracked in accordance with the condition reporting and corrective action process. Engineering technical issues and questions are identified, tracked and documented in accordance with the Engineering Request process. All of these processes are standardized in Entergy Nuclear South plants. A commitment to track MAIs is not required and this commitment is being deleted.

6. COMMITMENT CHANGE NO. 2001-0019, Control of Painting Activities

ORIGINAL COMMITMENT DESCRIPTION

A notification form that will document upcoming painting activity is being developed. This form will be presented to the shift supervisor/control room supervisor each day to inform him of the painting activities scheduled for that day. The form will also include a checklist that will aid the painting supervisor in determining whether a particular painting project will impact other systems or components. If the checklist indicates that temporary covers or screens will be used, engineering input will be obtained and included as an addendum to the work authorization for the painting in question.

SUMMARY OF CHANGE

Painting is a work management system (WMS) process and is implemented in accordance with Maintenance Action Items (MAI) work instructions and implementing procedures. If any painting is performed that impacts the operation of any safety related structures, systems or components, then the control room authorizes the release to work documents in the work package. Painting is controlled and managed in the same manner as other work performed on or near safety related SSCs. Deletion of this commitment does not relieve Waterford 3 of its responsibility to ensure painting activities do not contribute to the failure of any plant equipment. This responsibility is effected through the implementation of the Work Management System and the system of reviews and approvals required prior to beginning any work that can adversely effect the safety function of plant SSCs. There is no requirement to maintain this commitment and it is being deleted.

7. COMMITMENT CHANGE NO. 2001-0020, Prompt Identification of Leakage and Corrective Actions

ORIGINAL COMMITMENT DESCRIPTION

Hydraulic fluid leakage was occurring in the MSIVs without measures having been established to provide for either prompt identification of leakage or to preclude MSIV stem corrosion. Corrective steps will be taken to avoid other violations as well as UNT-005-002 and UNT-005-015 will be changed to require the identification of any necessary interim actions.

SUMMARY OF CHANGE

Both the Work Management System (WMS) and the Corrective Action Program require prompt identification of nonconforming conditions. Inherent to the corrective action program are requirements to take and document immediate corrective actions taken to prevent further degradation of plant structures, systems or components (SSCs). Also, inherent to Waterford 3 is the repetitive task program that requires periodic inspections of vital plant SSCs. Tasks are implemented in the WMS and documented on Maintenance Action Items (MAIs). This program is well established and there have not been repeated instances to indicate that the system in place is ineffective. Maintaining this commitment is no longer required and it is being deleted.

8. COMMITMENT CHANGE NO. 2001-021, Ineffective Work Controls – Change in Work Scope

ORIGINAL COMMITMENT DESCRIPTION

Work controls involving a change in work scope will be evaluated and revised as necessary.

SUMMARY OF CHANGE

The Work Management System (WMS) requires re-review of work instructions if the work scope changes. This is part of WMS and is ensured by the continual review of ongoing work required by the electronic workflow embedded in the WMS electronic tracking and routing database. This includes changes in staged/pledged materials that are necessary to implement the work. Upon work closure, the supervisor reviews and closes the work package. If variations from the work scope are identified at this stage, a condition report would be initiated and appropriate corrective action implemented. There is no requirement to maintain this commitment and it is being deleted.

9. COMMITMENT CHANGE NO. 2001-022, Revise Work Authorization Procedure to Clarify Retest Change and Scope Change Wording

ORIGINAL COMMITMENT DESCRIPTION

A change will be made to the Work Authorization procedure to clarify the retest change and scope change wording.

SUMMARY OF CHANGE

All work performed in the plant is performed in accordance with the Work Management System (WMS) and is documented on a Maintenance Action Item (MAI). The WMS requires implementation and review of work during each phase of the process including work completion. It is an integral part of the WMS process to ensure scope changes are reviewed prior to proceeding with work activities. During the review of the completed work (MAI) maintenance personnel ensure that scope changes did not occur. If scope changes occurred that did not receive the required review, then a condition report is generated and appropriate corrective action is taken. This is basic to the Work Management and Correction Action programs. This program has built in checks and balances to ensure that the work performed does not bypass required reviews when evolving work causes work scope changes. A commitment is not required to track this process.

10. COMMITMENT CHANGE NO. 2001-023, Deficiency Tracking via Condition Identification / Work Authorization

ORIGINAL COMMITMENT DESCRIPTION

The condition identification and work authorization (CIWA) are the primary vehicles through which abnormal plant conditions are identified, evaluated and corrected, as well as the means for implementing routine maintenance.

SUMMARY OF CHANGE

All work performed in the operating nuclear plant is performed in accordance with the Work Management System (WMS) and is documented on a Maintenance Action Item (MAI). Reviews and work performed and documented on an MAI is tracked electronically in the WMS database. In addition to tracking work via MAI there are other processes in place to identify, track and control activities at the nuclear facility. Conditions adverse to quality are identified and tracked in accordance with the Condition Reporting and Corrective Action process. Engineering technical issues and questions are identified, tracked and documented in the Engineering Request process. All of these processes are standardized in Entergy Nuclear South. A commitment to track deficiencies in WMS is not required and is being deleted.

11. COMMITMENT CHANGE NO. 2001-024, Inspection of Work and Restoration

ORIGINAL COMMITMENT DESCRIPTION

Procedures require plant Quality signoff for fulfillment of separation criteria as well as reinstallation of any tray covers or fire barriers that may have been removed in the work package.

SUMMARY OF CHANGE

All work performed in the plant is performed in accordance with the Work Management System (WMS) and is documented on a Maintenance Action Item (MAI). The WMS requires implementation and review of work during each phase of the process including work completion. Reviews and inspections, including hold points, are tracked electronically in the WMS database. This includes inspection hold points required by the work instructions or implementing procedures. The program has built in checks and balances to ensure that the work performed does not bypass required hold points. A commitment is not required to track this process and is being deleted.

12. COMMITMENT CHANGE NO. 2001-025, Maintenance Package Documentation

ORIGINAL COMMITMENT DESCRIPTION

A revision to maintenance procedure MD-001-002 will be made to specifically require that information sheets be retained as part of the work package closure documentation. Maintenance personnel will be required to read the revision to maintenance procedure MD-001-002 when effected.

SUMMARY OF CHANGE

When this commitment was generated, the process for field controlling work package attachments was not clearly defined. Since then the Work Management System and field control processes are more clearly defined and well understood by maintenance personnel. Site technical documents are controlled in accordance with W5.201, Document Control System. Specifically section 5.4.3 describes the requirements for handling field controlled documents. MD-001-040, Maintenance Action Item Performance Documentation, describes the process for building and maintaining a work package. Many documents previously controlled outside of this process are now embedded in the Work Management System / Maintenance Action Item database and are created as part of the MAI. The proper use and disposition of work packages and associated documentation is an integral part of the work management process at Waterford 3 and maintaining this as a commitment is no longer necessary.

13. COMMITMENT CHANGE NO. 2001-027, Responsibility Changes for Planning and Maintenance Departments

ORIGINAL COMMITMENT DESCRIPTION

The maintenance department will revise administrative procedures to reflect the changes in UNT-005-012 when referring to Planners or Lead Discipline Planners

SUMMARY OF CHANGE

Changes to the work management process and station reorganization have caused many responsibility and department changes. The responsibilities of the planning department personnel vs. maintenance department personnel are understood. Maintenance of this commitment is not required.

14. COMMITMENT CHANGE NO. 2001-028, Post Trip Review Criteria

ORIGINAL COMMITMENT DESCRIPTION

Before reclosing the reactor trip breakers, the shift supervisor must sign the completed Post Trip Review (PTR) indicating that the cause of the unscheduled reactor trip has been corrected. The Waterford 3 PTR requires that extensive information as to safety system performance be gathered and assessed prior to restart.

SUMMARY OF CHANGE

The original commitment text is overly restrictive from a verbatim compliance perspective. Generic Letter 83-28 requires a post trip review process to ensure a determination is made that the plant can be restarted safely and has established criteria for determining the acceptability of restart. The original criteria for restart (cause corrected prior to closing reactor trip breakers) does not provide flexibility for dealing with trips resulting from failures in the secondary plant (turbine valve malfunctions, for example). There is no reason to restrict closing of reactor trip breakers for preparation of a plant startup while repairs are ongoing in the secondary system. The revised commitment text maintains the GL 83-28 requirement by requiring the post trip review to be completed prior to startup and requiring equipment to be repaired prior to it being relied upon for plant startup. This revised criteria is consistent with INPO Good Practice OP-211 "Post Trip Reviews".

15. COMMITMENT CHANGE NO. 2001-029, Post Trip Review Task Group

ORIGINAL COMMITMENT DESCRIPTION

In the event of an unscheduled reactor trip, a post trip review task group will be assembled. The group will be headed by the on-shift Shift Technical Advisor and will include the duty event analysis, reporting and response representative, the duty engineering management representative and the duty operations superintendent. This group along with the shift supervisor will be responsible for completing the analysis and evaluations portion of the post trip review and determining the root cause of the trip and the initiating plant protection system signals.

SUMMARY OF CHANGE

Commitment is revised to read "The plant operations review committee (PORC) is responsible for reviewing the post trip review prior to restart if the cause of the reactor trip is not positively known." As stated in the NRC SER (ILN89-0576) associated with this commitment, Waterford 3 committed beyond the requirement of GL 83-28 for independent review by a competent group if the cause of the event cannot be positively identified. The post trip review task group discussed in the commitment is not required and is not necessarily independent since it may be headed by the on-shift STA. The Post Trip Review procedure (OP-100-012) requires the plant operations review committee (PORC) to review the post trip review prior to restart if the cause of the trip is not positively known. This revised commitment text is more in line with and meets the requirement of GL 83-28.

16. COMMITMENT CHANGE NO. 2001-032, Submittal of Reload Analysis

ORIGINAL COMMITMENT DESCRIPTION

This commitment states that Nuclear Services Procedure NSP-102 provide instruction for making FSAR and Technical Specification changes and that the procedure should ensure timely submittal to the NRC of reload analysis which require Technical Specification changes or FSAR amendments.

SUMMARY OF CHANGE

This commitment is being deleted. This commitment is an administrative commitment which meets the criteria of LI-110 for deletion, specifically (1) not within a codified process of 10CFR50.59, 10CFR50.54 or 10CFR50.82, (2) does not have safety significance because it does not impact the ability of a structure, system or component to perform its safety function or impact a safety analysis, (3) is not required to restore compliance with an obligation, (4) the NRC did not rely upon this commitment in lieu of taking other action, and (5) the commitment is not required to minimize an adverse condition. This commitment is an administrative commitment that was made in response to Generic Letter 84-02 via Waterford 3 W3P84-26226 on September 19, 1984. Generic Letter 84-02 communicated that Licensees should allow at least six months prior to restart for NRC approval of licensing submittals that are based on reload analysis and that involve an unreviewed safety question of FSAR Chapter 15 analysis models or methods not approved by NRC. Waterford 3 in letter W3P84-2626 simply communicated that Nuclear Services Procedure NSP-102, replaced by Site Procedure W4.503, provides instructions for making FSAR and Technical Specification changes. There was no safety issue or adverse condition associated with this communication. Although, this commitment was made in response to a Generic Letter, there was no review, approval or reliance of any safety or technical significance placed by the NRC on the communication in lieu of taking other action. Based on today's standards, the communication conveyed by Generic Letter 83-02 would be communicated by a Regulatory Issue Summary (RIS).

17. COMMITMENT CHANGE NO. 2002-0010, Vendor Interface Program

ORIGINAL COMMITMENT DESCRIPTION

To ensure that vendor technical information is kept current and complete, LP&L is currently establishing a more formalized method of vendor contact, proposed as the key vendor contact program. A description of the proposed program was transmitted to the NRC in letter W3P88-1940 dated October 14, 1988. Under this program, the nuclear operations procurement engineering group will maintain a key vendor list that will include the name of the vendor, the type of equipment supplied and pertinent information that will identify the equipment. Once fine-tuning of the proposed system occurs, appropriate controlled program instructions or procedures will be established.

SUMMARY OF CHANGE

Waterford 3 will implement a revised process which will require documented contact with non-NSSS vendors once every other calendar year. This process will also control the list of non-NSSS vendors to be contacted. Generic Letter 90-03 requires licensees to maintain a vendor interface program which is a good faith documented effort to periodically contact the vendors of key non-NSSS safety related components (such as auxiliary feedwater pumps, batteries, inverters, battery chargers, cooling water pumps, and valve operators) to obtain any technical information applicable to this equipment. As documented by letters CEO-98/00079, CEO-99-00086 and CEO-2000-00089, Entergy has contacted approximately 44 vendors per year for the last three years to request updated technical information related to approximately 510 technical publications. In response to these requests, approximately 43 documents were submitted to Entergy as updated information. Only a small percentage of the documents received were found to be applicable to plant equipment. None of the information received resulted in any corrective actions or plant modifications. Changing the frequency of Entergy's periodic contact with key non-NSSS vendors to every other calendar year represents no appreciable difference in meeting the intent of the Generic Letter and therefore should have no adverse effect on plant equipment.

18. COMMITMENT CHANGE NO. 2002-0011, Drill Participation Requirements

ORIGINAL COMMITMENT DESCRIPTION

The drill participation requirements presently maintained in an informal manner will be incorporated as part of procedure EP-003-020, Emergency Preparedness Drills and Exercises.

SUMMARY OF CHANGE

This commitment is being deleted. Waterford 3 no longer tracks all ERO responders. Per NEI 99-02, Waterford 3 tracks and reports results of respondents for all key responders to the NRC as performance indicators. This requirement is satisfied by Procedure EPP-431.