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

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

Gentlemen:

Enclosed is the Report of Facility Changes, Tests and Experiments for Waterford 3 which is submitted pursuant to 10CFR50.59. This report covers the period from June 1, 1994 through November 30, 1995.

If you have any questions regarding this report, please contact me at (504) 739-6242 or G.E. Wilson at (504) 739-6357.

Very truly yours,

James J. Fisicaro

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

REPORT OF FACILITY CHANGES, TESTS AND EXPERIMENTS

PER 10CFR50.59

JUNE 1, 1994 TO NOVEMBER 30, 1995

SUMMARY

This report provides the Waterford 3 Facility Changes made pursuant to 10CFR50.59(a)(1). The report covers the period from June 1, 1994 through November 30, 1995. None of the items in the report represent an unreviewed safety question.

Section I of the report identifies 166 Facility Changes; 65 Design Changes (DC), 34 Condition Identification/Work Authorizations (CI/WA), 17 Temporary Alterations (TAR), 14 Document Revision Notices (DRN), 16 License Document Change Request (LDCR), and 20 Miscellaneous Evaluations. Section II of the report identifies 58 Procedure Changes; 37 Plant Procedures and 21 Special Test Procedures (STP).

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

JUNE 1, 1994 THROUGH NOVEMBER 30, 1995

TABLE OF CONTENTS

B. CONDITION IDENTIFICATIONS/WORK AUTHORIZATIONS (CI/WA)

C. TEMPORARY ALTERATION REQUEST (TAR)

D. DOCUMENT REVISION NOTICES (DRN)

E. LICENSE DOCUMENT CHANGE REQUESTS (LDCR)

F. MISCELLANEOUS EVALUATIONS

II. PROCEDURES

A. PLANT PROCEDURES

B. <u>SPECIAL TEST PROCEDURES (STP)</u>

I. FACILITY CHANGES

A. DESIGN CHANGES (DC)

1. DC-3018, Modifications to the Charging Pump Packing Cooling System (Revision 0)

DESCRIPTION OF CHANGE

The DC replaces the high-low level sensor with a float switch, the existing automatic fill valve is replaced with a solenoid valve, the existing manual fill valve is replaced with a ball valve and three new isolation valves are added to allow for isolation of each individual pump for maintenance.

REASON FOR CHANGE

The Packing Cooling system has not proven to be reliable, causing inadequate cooling which results in premature packing wear. Malfunction of the automatic fill valve and tank level sensing probes have resulted in the tank overflowing.

SAFETY EVALUATION

According to the safety evaluation the Packing Cooling system components that are modified by this DC are non-safety related. The Packing Cooling system is not required for operation of the charging pump, the pump is only required to run approximately four hours for accident mitigation. The evaluation states that similar pumps have been qualified to run for 100 hours without packing cooling water. No margin of safety was affected by the DC.

2. DC-3096, Moisture Separator Reheater (MSR) Tube Bundle Replacement and Temperature Control (Revision 0, Revision 1 and Revision 2)

DESCRIPTION OF CHANGE

This DC will re-tube the MSRs with stainless steel (type 439) tube bundles, replace carbon steel chevrons with high capacity Peerless stainless steel chevrons. Scavenging steam lines above the Turbine deck will also be replaced with stainless steel lines. Revision 1 of the DC installs positioners on the ten inch diameter MSR temperature control valves and rewires the ten inch diameter MSR temperature control shut off valve to actuate under control of the limit switches on the three inch diameter MSR temperature control valves (currently they are controlled by limit switches on the ten inch diameter temperature control valves). Revision 2 of the DC increases the opening time of the MSR temperature control valves from approximately three hours to approximately five hours.

REASON FOR CHANGE

The DC will improve the performance and reliability of the MSRs. It will bring MSR performance into agreement with the Waterford 3 licensing basis heat balances.

SAFETY EVALUATION

According to the safety evaluation no unreviewed safety question exists because of the DC. The evaluation states that there are no accidents in the FSAR that are caused by the MSRs. The only effect that the MSRs have on other accidents is the mitigating effect on turbine generated missiles, this passive function will not be affected because the shells of the MSRs are unchanged and all internal components that are removed are replaced with essentially identical components. The evaluation also notes that no new system interactions or connections are created by the DC, that the MSRs and adjacent piping are classified as non-safety and there is no affect on any accident response or protective boundary.

3. DC-3125, Secondary Vacuum Degasifier (Revision 2)

DESCRIPTION OF CHANGE

Revision 2 of this DC deletes all acceptance testing requirements associated with the DC.

REASON FOR CHANGE

The DC was unsuccessful in correcting the numerous problems associated with the Secondary Vacuum Degasifier. It is not cost beneficial to pursue additional efforts. The equipment has been isolated and vendor equipment is performing the degasification function during the water demineralization process.

SAFETY EVALUATION

According to the safety evaluation the Secondary Vacuum Degasifier has been isolated from the Demineralized Water (DW) System and flow is being bypassed around the affected equipment by utilizing existing valves and piping intended for this purpose. The system is not being operated in an abnormal manner, portions of the DW system located in the "yard area" are physically separated from all safety related and seismic I equipment and structures. The system serves no safety function and is not required for operation during the safe shutdown of the plant following an accident or to mitigate the consequences of any accident. Isolating the Secondary Vacuum Degasifier has no impact on nuclear safety.

4. DC-3136, Stiffening of Dry Cooling Tower Fan Motor Supports (Revisions 0, 1, and 2)

DESCRIPTION OF CHANGE

DC-3136 will add stiffening plates to only one of the DCT Fan motor pedestals. Vibration measurements will then be conducted to determine "fine tuning" of the pedestal. Revision 1 of the DC will then modify the remaining twenty-nine pedestals.

Revision 2 of the DC deleted the modification to the remaining twenty-nine fan pedestals.

REASON FOR CHANGE

Reliability of the DCT Fan motors has been low due to failure of the motor bearings. The high failure rate is attributed to excessive vibration of the motor pedestals. Stiffening of the pedestals is expected to reduce or eliminate the excessive vibrations. Vibration data taken on the stiffened structure showed the structure behaving as predicted by computer analysis, a significant reduction in the overall displacement of the structure.

Improved maintenance practices and the use of shims under the motor base has achieved reduction in motor bearing failures, thus the need to complete the modification to the remaining motors was to determined to be unnecessary.

SAFETY EVALUATION

According to the safety evaluation there is no unreviewed safety question associated with this DC. Accidents identified by the safety evaluation are the Loss of Coolant Accident (LOCA), Main Steam Line or Feedwater Line Break (inside containment), OBE, and SSE. The evaluation states that the number of fans that can be declared inoperable without affecting the operation of the plant is specified by the Waterford 3 Technical Specifications and that only one fan will be inoperable at a time during performance of the DC. According to the safety evaluation the DC does not diminish the ability of the DCT fans to remove heat from the containment. The margin of safety inherent in the ultimate heat sink is maintained by the limiting condition for operations (LCO) specified in the Technical Specifications in regards to the fans' operability. The DC will not affect the margin of safety.

5. DC-3196, Installation of Instrumentation to Measure Moisture Separator Reheater Efficiency (Revision 0 and Revision 1)

DESCRIPTION OF CHANGE

Revision 0 of the DC installs instrumentation on Moisture Separator Reheater (MSR) "B" only, to measure flows at both ends of MSR "B" for main steam flow to the reheater section and heater drain flow from the heater drain collection tanks.

Revision 1 of the DC supersedes Revision 0 and reflects the as-built condition with the flow elements removed and blind flanges installed at the branch connections on the heater drain piping. The transmitters, cables, and power supplies associated with the heater drain instrumentation will be spared in place.

REASON FOR CHANGE

MSRs have experienced generic problems industry wide which have caused loss of plant thermal efficiency, poor unit availability, and high maintenance costs. The MSRs require additional instrumentation to perform operating efficiency tests. The MSR efficiency tests are required to determine the optimum time frame for MSR tube bundle replacement.

Following implementation of DC-3196, Revision 0, steam leaks developed at the isolation valves for the flow instrumentation for MSR "B." In addition, accurate flow measurement could not be obtained by the instrumentation installed in the heater drain piping because the piping contained two phase flow. Efforts to correct these problems were unsuccessful. All flow elements were eventually removed from the piping and blind flanges were installed to "cap" the two inch branch connections.

SAFETY EVALUATION

As indicated in the Revision 0 safety evaluation the MSRs and associated piping and instrumentation are non-safety related components. The addition of a two inch branch line for new instrumentation will have no impact on existing analysis. The possibility of an accident which is different than already evaluated in the FSAR will not be created because the main steam line break analysis would bound any possible pipe break or failure of the branch lines. The probability or consequences of malfunction of equipment important to safety previously evaluated in the FSAR will not be increased because the components are not safety-related and are not used in any safety-related application. Similarly, the addition of non-safety branch connections will not create the possibility of a malfunction of equipment important to safety The margin of safety as defined in the design basis will not be reduced.

The safety evaluation for Revision 1 also determined that the DC will not reduce the margin of safety as defined in the basis of any technical specification or safety analysis and no unreviewed safety questions are created. The safety evaluation also notes that failure of the affected piping could cause a decrease in flow to the shell side of #1 Feedwater heater, resulting in a "decrease in feedwater temperature." According to the safety evaluation the blind flanges used to cap the branch connections meet the requirements of ANSI B31.1 (piping category 5), thus, a gross failure of the flanges is no more likely than the failure of an installed valve or the piping itself. Therefore, the probability of a "decrease in feedwater temperature" is not increased.

6. DC-3203, Stator Cooling Water (SCW) Surveillance System Enhancement

DESCRIPTION OF CHANGE

This DC installs instrumentation to measure and record conductivity and dissolved oxygen content of the main generator stator cooling water.

REASON FOR CHANGE

The DC is an enhancement to monitor conductivity and dissolved oxygen content of the SCW system.

SAFETY EVALUATION

According to the safety evaluation there are no unreviewed safety questions associated with this DC. There are no accidents listed in the FSAR that can be caused or affected by this modification, The DC impacts only the SCW System on the secondary side of the plant. SCW is a closed loop and has no common boundaries with safety related systems, is totally isolated from the primary side and does not contribute to any plant effluents. The evaluation states that a review of Technical Specifications indicates no LCO's or limiting safety system settings are associated with the SCW system. And, because of the system's isolation from the safety-related aspects of the plant, no reduction in the defined margin of safety will result from the implementation of this DC.

7. DC-3251, Essential Chilled Water Pumps Replacement (Revision 0)

DESCRIPTION OF CHANGE

DC-3251 replaces the existing Essential Chilled Water (CHW) Pumps with qualified, reliable equivalent pumps. The existing Class IE qualified motors will be retained.

REASON FOR CHANGE

The CHW pumps have experienced a long history of packing leakage and multiple occurrences of shaft failure. The original supplier of the pumps can no longer supply qualified replacement parts.

SAFETY EVALUATION

According to the safety evaluation the replacement of the CHW pumps will not cause or affect any accidents described in the SAR. The new pumps serve the same function and operate the same way as the existing pumps. The new pumps are supplied with mechanical seals instead of packing as the existing pumps use. This is expected to result in enhanced operation because packing leakage has been an ongoing problem with the existing pumps. The evaluation states that the DC will have no impact on any protective boundaries.

8. DC-3253, Upgrade Telephone System (Revision 3)

DESCRIPTION OF CHANGE

This DC installs an upgraded telephone system switch in the communication room, Elevation +7.00' Reactor Auxiliary Building (RAB), and supplies power to the switch from an uninterruptable power source. The DC also installs a fiber optic cable from this switch to the Generation Support Building.

REASON FOR CHANGE

The existing telephone system is longer adequate and reliable and maintenance costs for the system are increasing. The present system is not compatible with other Company communication systems.

SAFETY EVALUATION

According to the safety evaluation the communication system is a non-safety related system that has no impact on reactor safe shutdown. There is no safety related equipment associated with this DC. A review of Station Blackout and HVAC calculations concluded that the added heat load is within acceptable design limits.

9. DC-3259, Supplemental Chiller Condensing (SCC) System Improvements

DESCRIPTION OF CHANGE

The DC replaces the mechanical cooling tower level control valves with float operated diaphragm valves and replaces the SCC cooling water chemistry controller. The DC adds time delay relays to bypass the SCC circulating pump trip during pump startup. Also, it adds a water softener at the cooling tower inlet and makes potable water (PW) the primary make-up to the SCC via the water softener.

REASON FOR CHANGE

The existing level control valves have proven to be unreliable resulting in cooling tower overflow. This results in diluting the chemical treatment of the water. TAR 90-021 (W3F-91-0039, dated December 12, 1991, Item 59) installed a temporary water softener to reduce the hardness of the water supplied to the cooling towers. Modifications listed in the above description are expected to result in reduced maintenance and improved reliability of the SCC.

SAFETY EVALUATION

According to the safety evaluation the SCC is a non-safety system and does not affect the probability of occurrence or the consequences of any previously evaluated accident. The evaluation also states that there are no new system interactions involved in the DC that would create the possibility of an accident of a different type than previously evaluated. The DC does not impact any safety related systems and does not reduce the margin of safety as defined in the bases for any technical specification.

10. DC-3268, Fire Detection System Upgrade (Revision 3 and Revision 4)

DESCRIPTION OF CHANGE

This DC provides for the replacement of the existing fire detection/suppression system and installs a new fire detection system. The DC will be implemented in two phases. The first phase will install control panels, cables and raceways. Phase two will involve interconnection of new detectors, suppression system alarm devices and other external connections to fully install the system and transfer all responsibility from the existing system. Revision 4 of the DC will add a strobe light to the Fire Detection Master Control Panel in the Control Room and connect the Service Building extension to the Fire Detection System.

REASON FOR CHANGE

This DC will correct problems involving obsolescence of detectors and system reliability. Installation of the new state-of-the-art system provides more capabilities than the current system, reduces maintenance costs because of integral system sensitivity monitoring from the control panels rather than local sensitivity tests at the detectors.

SAFETY EVALUATION

The safety evaluation concluded that this DC will not have any affect on any accidents. Engineering calculations provide assurance that the Emergency Diesel Generators will not be adversely impacted by the new loads added by the DC. The evaluation identified that there is no adverse impact on Control Room heat load because of the DC.

11. DC-3330, Miscellaneous Hoist Enhancement Inside the Reactor Containment Building (RCB)

DESCRIPTION OF CHANGE

This DC provides for four different activities associated with lifting requirements inside the RCB during refuel and maintenance outages. 1) A one ton manual Jib Crane with manual trolley is designed for the -4 elevation on the west side "D" wall to provide enhanced movement of materials and tools from -11 elevation to -4 elevation. 2) The second activity involves procuring four, 2-ton, quality related jibs/hoists which can remain in place on the Steam Generator framing during normal plant operation. 3) Activity three will qualify a currently existing pedestal crane to remain on its pedestal during normal plant operation, this activity will also include a cab access platform. 4) The fourth activity consist of modifying the padeyes and lifting slings of the Control Element Drive Mechanism (CEDM) Cooling system cooling shrouds.

REASON FOR CHANGE

The activities listed above will result in improvement of handling of materials and tools, improved worker safety, reduced exposure, and reduced preparation time for refuel/maintenance activities during outages.

SAFETY EVALUATION

According to the safety evaluation there are no unreviewed safety questions associated with this DC. The newly designed jibs and hoists are classified as Seismic Category II, they are all seismically supported. During normal plant operations the jib will be restrained in its parked position by means of a locking device and a secondary restraint on the "D" wall.

A review of jet impingement maps showed that the modifications performed for activities 1, 3 and 4 (see DESCRIPTION OF CHANGE above) are not in the direct path of any jet stream created by a pipe break. One jib crane over Reactor Coolant Pump 1A is in the direct path of several vertical jet streams, however, this jib crane will be restrained by a secondary restraint at the top of the "D" wall and would not affect essential components or become a missile. This DC is structural, the Net Free Volume of Steel being added by this change is a small amount and is within the margins of previously calculated values that it will not have any significant effect on the containment pressure in the event of an accident. Thus the margin of safety is maintained.

12. DC-3337, Fire Protection System Containment Isolation Valve Leakage

DESCRIPTION OF CHANGE

DC-3337 adds coalescing filters to the Station Air (SA) System and drain traps and spectacle blind flanges to the Fire Protection (FP) system.

REASON FOR CHANGE

Installation of the DC will reduce the probability of internal corrosion of the carbon steel components and reduce valve leakage during LLRT testing.

SAFETY EVALUATION

According to the safty evaluation no new system interactions are created by the DC. Neither the FP or SA systems are postulated to initiate any accident previously evaluated in the FSAR. The DC will reduce the probability of of interior surface corrosion which will increase the reliability of the FP system.

The portions of the FP and SA systems t be modified are non-safety and the portions which fall within the seismic analysis of the Reactor Containment Building will analyzed and supported accordingly. All new materials utilized in the FP system will be QC-2.

13. DC-3354, Spent Resin Transfer System Enhancements (Revision 0 and Revision 2)

DESCRIPTION OF CHANGE

This DC provides a modified liquid level transmitter and relocates the High Pressure (HP) leg tap for the Spent Resin Tank (SRT), it also provides a new short cycle recirculation line and a clean water flush line. Revision 2 of the DC provides instruction to abandon in place Spent Resin Level Detector (LI-WM-5103), Spent Resin Level indicator (RWM-ILT-5103), and "Resin to Water Ratio Unacceptable" light. Revision 2 also re-scales RWM-ILT-0644 to show the tank overflow point equivalent to 100%.

REASON FOR CHANGE

Addition of the short cycle recirculation line allows recirculation of the resin/water mixture prior to transfer of the spent resin. Installation of the clean water flush will allow flushing of the spent resin piping after a transfer of spent resin, this will reduce radiation levels inside the piping and the pump room itself. SRT level indicator (RWM-ILT-5103) has been out of service due to a broken float cable. Modification of the water level transmitter (RWM-ILT-0644) will provide accurate and reliable liquid level measurement in the SRT.

SAFETY EVALUATION

The safety evaluation determined that there are no unreviewed safety questions associated with this DC. The evaluation indicates the only accident discussed in the FSAR that may be affected by this DC is "Liquid Waste System Leak." The evaluation concluded that the DC will not affect this previously evaluated accident since the components are designed in accordance with ANSI B31.1-1973 and the design requirements of R.G. 1.143, Radiological doses that could result from this accident remain within the allowable limits as analyzed in the FSAR.

According to the "Radioactive Waste Systems Additional Safety Evaluation" performed for this DC no changes are made to the amount or type of radioactive waste generated. The DC will not increase or change the radiological consequences of an unplanned and uncontrolled release of radioactivity.

14. DC-3360, Replacement Cable/Connector Assemblies for CEDM Exhaust Fan Motors (Revision 0)

DESCRIPTION OF CHANGE

To provide spare replacements for BIW cable/connector assemblies modified with an additional connector handling device to aid in installation and removal of the CEDM Exhaust Fan Motors cable/connector assembly.

REASON FOR CHANGE

The existing BIW cable assemblies are very cumbersome due to their length, size and weight. In addition, installation of the assembly plug mounted on the missile shield is difficult because personnel are required to hang over the refueling pool to accomplish the task. This DC should result in reduced exposure and simplify installation/disconnection of the CEDM Exhaust Fan Motors cable assemblies.

SAFETY EVALUATION

According to the safety evaluation the DC has no direct or indirect interaction with the Control Element Drive Mechanism (CEDM) System power to the CEDMs. Thus, the DC has no effect on Control Element Assembly (CEA) withdrawal accident occurrences. The assemblies are non-class IE and seismically designed. The cable/connector assemblies are associated with the power feed to the CEDM Exhaust Fan Motors and are not for the CEDM System

The CEDM Exhaust Fan Motors are non-nuclear safety, seismic Category I. Because the power feeds to the motors are designed as associated circuits, the replacement assemblies do not need to be procured as Class IE, Seismic Category I, as were the original assemblies, but as non-Class IE, seismic designed. During plant operation the assemblies (i.e., power feeds) serve no safety function.

15. DC-3362, Replacement of 3A-S, 3B-S and 3AB-S Class IE Station Batteries and Associated Equipment (Revisions 0 and 2)

DESCRIPTION OF CHANGE

This DC replaces each of the plant's Class IE battery banks. The cells of 3A-S and 3B-S will be replaced with ones of nearly twice the capacity. In addition, the bus circuit breakers will be replaced because of the higher short circuit current produced by the new batteries. Battery 3AB-S will be replaced with cells of marginally smaller capacity. The battery bank will be modified by the addition of two more cells to enhance the voltage profile and the capacity. The battery racks will be either replaced (3A and 3B) or modified (3AB) to accept the new batteries. Revision 2 of the DC replaces the equalize timers with an equalize toggle switch.

REASON FOR CHANGE

This DC is required to ensure that the Class IE batteries continue to satisfy their Design Bases Accident (DBA) and Station Blackout (SBO) coping requirements.

SAFETY EVALUATION

The safety evaluation did not identify any unreviewed safety questions associated with this DC. According to the evaluation the replacement cells with higher capacity provide power for a longer duration and will enable the DC system (A&B) trains to meet current DBA requirements for connected loads while allowing sufficient margin for growth. The batteries provide back-up power to the DC system, with the increased capacity of the new batteries this will be for a longer duration and a better voltage profile, a desired response for coping with an event. Calculations EC-E91-058, 059, 061, and 062 document the improved performance of the replacement batteries.

16. DC-3364, Feedwater Isolation Valve Enhancements (Revisions 2, 3, 4 and 5)

DESCRIPTION OF CHANGE

The DC relocates pressure switches from the Feedwater Isolation Valves (FWIV), FW-184 A & B, to an instrument stand. Instrument block and bleed valves are added to facilitate maintenance, calibration, and reliability. Revision 2 replaces the installed 8.5 gallon accumulators with 11 gallon accumulators with integral piston stop tubes. Revision 3 of the DC removes welding between the missile shield grating (located above the FWIVs) and supporting steel beams to facilitate removal of the actuators. Revisions 4 and 5 of the DC revise the no load closure test times for stroking the valve to the closed position.

REASON FOR CHANGE

Relocation of the pressure switches and installation of the instrument valves will increase the reliability of the FWIVs. Installation of the instrument valves will also facilitate the pre-charging of the accumulators and the calibration of the pressure switches. Replacing the 8.5 gallon accumulators with larger (11 gallon) accumulators with integral piston stop tubes will allow more effective monitoring of the nitrogen pressure in the accumulator. (Larger accumulators are utilized to compensate for the volume occupied by the integral piston stop tubes.)

Testing of the new accumulators with reduced needle valve settings resulted in changing the no load closure test times for stroking the valve to the closed position from 1.75 to 2.75 seconds. The new needle valve settings assures that the actuator will not close the valve in less than 1.5 seconds using both accumulators and in not more than 5.0 seconds using one accumulator.

SAFETY EVALUATION

According to the safety evaluation the DC increases the reliability of the FWIVs. The new accumulators with the integral piston stop rings will ensure that the accumulator pressure can be reliably measured thus ensuring adequate nitrogen to close the valve.

Table 3.3-5 of the Technical Specifications, Engineered Safety Features Response Times, requires main feedwater isolation to occur in less than or equal to 6.0 seconds. This time limit exists to limit the mass and energy released to the containment during a postulated main steam line break accident. With an assumed 1.0 second signal processing time valve closure must then be achieved with 5.0 seconds of receipt of the isolation signal. Correspondence from the valve vendor confirms that the valve will continue to close within the required limit by following the new precharge graph. Since the valve will continue to close within the required limit the DC is bounded by the current accident analysis in the SAR.

17. DC-3374, Plant Monitoring Computer (PMC) Replacement/Upgrade Phase I, Cable Installation (Revision 0 and Revision 1) Phase 2 Hardware/Software Replacement /Upgrade (Revision 2), Deletion of the Bypassed/Inoperable Status Indicating System (BISIS)

DESCRIPTION OF CHANGE

Phase I of the DC installs the necessary cables (coaxial and fiber optic) for the replacement PMC. The cables will not be connected during this phase of the DC. Low voltage communication cables for the Information Systems Local Area Network (LAN) will also be installed during this phase and connected as required. Revision 1 adds clarification of air boundaries and changes acceptance test criteria to narrow the scope of the test to specific criteria.

Phase 2 of the DC provides for the replacement of computer hardware/software, upgrade of the EOF and simulator equipment and the installation of existing applications programs from the existing system to the new hardware.

Revision 3 of the DC provides for the removal of the BISIS from the PMC (software) and removal of the indicating panel from CP-2

REASON FOR CHANGE

The current PMC is 17+ years old and has fallen well behind the current state of computer systems. Replacement of the PMC is needed to maintain reliability, operability, and growth capability for plant needs and requirements. PMC failures have resulted in power reductions. Phase I of the DC will install necessary cabling for the replacement PMC, during a Refueling Outage in order to maintain the integrity of the Control Room Envelope.

SAFETY EVALUATION

According to the safety evaluation addition of the cables to the area beneath the Shift Supervisor's office does not impact existing accident scenarios, nor does it create the potential for new accidents. Neither the probability of malfunction of equipment important to safety or the consequences of previously evaluated accidents are increased by this phase of the DC. The evaluation notes that combustible loading calculations are maintained in accordance with the 10CFR50 Appendix R requirement for a Fire Hazards Analysis. Fire protection features provided in the Control Room are sufficient to provide adequate protection for the additional combustible loading.

The safety evaluation states that there are no accidents listed in the FSAR that may be caused or affected by this DC. The Plant Monitoring Computer (PMC) is a non-safety system that runs application programs. These programs interface with other plant control systems that are required for reactor control, but are not essential for the safety of the plant. The safety evaluation for Revision 2 of the DC also included the supplemental guidance of EPRI TR-102348 for 10CFR50.59 evaluations of digital upgrades.

During the installation process a parallel run (both new system and old system will be in operation) will be completed to verify proper operation of the new system. This parallel operation will result in increased load for the Emergency Diesel Generators but is within the analyzed margin of safety and does not affect any Technical specification bases for fuel or load rating of the equipment. The final PMC configuration will reduce the EDG load. No changes will occur regarding any protective boundaries as a result of this DC.

According to the safety evaluation for Revision 3 of DC-3374 removal of the BISIS panel will not affect any accidents defined in the SAR. This panel is a non-safety monitoring operator aid for which there are means of determining system operability for mitigating any accident postulated. BISIS does not control process systems so it cannot initiate an accident. Removal of the RG 1.47 BISIS Display Panel and software logic will not affect the bases for the Technical Specifications or the safety analysis for the facility.

18. DC-3375, Cutting and Capping of HPSI/LPSI Drain Lines (Revision 0)

DESCRIPTION OF CHANGE

This DC will cap individual drain lines, in the Safety Injection System (SIS), to eliminate possible leak paths from the SIS and make the detection of future leaks past the main SIS header valves in the High Pressure Safety Injection (HPSI) and Low Pressure Safety Injection (LPSI) headers possible. Where practical drain headers and supports will be removed if all the drain lines that tie into the header are capped. The DC will also cap one Containment Spray (CS) drain line which ties into a drain header with SIS drains that will be capped. This will allow removal of the common drain header.

REASON FOR CHANGE

The Safety Injection Tanks have been losing water and the leakage has been determined to be coming from drain valves in the HPSI and LPSI systems. Many of the drain valves are linked into common headers which makes detection of leaky drain valves difficult, and make it impossible to leak check the main SIS header valves. Capping these drain valves will eliminate possible leak paths from the SIS and allow for detection of leaks past the main SIS header valves.

SAFETY EVALUATION

The DC involves a non-safety related portion of a safety related system, however, it does not alter or degrade the isolation boundary with the safety related system or alter any of the performance parameters of the safety related system. The drain lines to be capped are normally isolated and are only used for maintenance or testing. Thus the DC will not affect any existing accident analysis.

19. DC-3376, Replace Packing and Cap Leakoff Lines on Safety Injection Valves

DESCRIPTION OF CHANGE

To reduce or eliminate the possibility of packing leakage, the existing packing in several Safety Injection System (SIS) valves will be replaced with packing of an improved design and proven record of performance. The valve leakoff line will be cut and capped to maintain the pressure boundary.

REASON FOR CHANGE

Packing leakage on several SIS valves is believed to be contributing to overall system leakage. The leakoff piping routes the leakage to the Equipment Drain System where it eventually collects in the Reactor Containment Building (RCB) Sump. The original packing design consist of a lantern ring which diverts any leakage past the lower valve packing to the leakoff line. This design makes it difficult to determine which valves are leaking and to determine the extent of the packing leakage. The improved packing to be installed is essentially leak free and the valve leakoff connection will be capped to maintain the pressure boundary.

The following SIS valves are affected by this DC:

SI-302	(Deleted from scope of DC by Revision 1)
SI-303 A&B	
SI-304B	
SI-307 A&B	
SI-308	
SI-331 A&B	
SI-332B	
SI-342	
SI-343	(Added to scope of DC by Revision 1)
SI-405 A&B	(Deleted from scope of DC by Revision 1)

SAFETY EVALUATION

The safety evaluation identifies three accidents of interest regarding this DC; the Loss of Coolant Accident (LOCA), Steam Generator Tube Rupture (SGTR), and Main Steam Line Break (MSLB). The evaluation states that the DC will not increase the probability of occurrence of an accident because the integrity of the SIS has been improved. No new system interactions are created by the implementation of this DC and the packing material itself is not an ASME code component. The new gland bolting materials will be corrosion resistant QC-1 or QC-3 materials which will prevent damage of the pressure retaining components should a leak occur. Failure (leakage) of the new packing could not initiate a previously evaluated accident. Leakage would be identified through the Containment Sump level and flow monitoring system (Technical Specification 3/4.4.5, "Reactor Coolant System Leakage").

20. DC-3379, Secondary Metal Transport Program

DESCRIPTION OF CHANGE

DC-3379 will install new sample taps on the non-safety, non-seismic portions of the Feedwater Main Steam and Blowdown Systems. The sample taps will be connected to three new sample panels which will cool the sample, reduce and regulate the pressure, and route a small portion of the flow through a 0.45 micron Millipore filter. The filter will be periodically analyzed to evaluate erosion/corrosion rates in addition to other chemistry analysis.

REASON FOR CHANGE

DC-3379 was developed because Waterford 3 does not presently have an adequate method of monitoring metal transport throughout the secondary system. Phase 1 of the DC will consist of only the new tie-ins or branch connections and isolation valves for the systems affected. Phase 2 will implement the balance of the DC.

SAFETY EVALUATION

According to the safety evaluation the DC will not increase the probability of occurrence of a Steam Generator Tube Rupture (SGTR) event. Should a SGTR occur the Blowdown radiation monitor would alarm and flow from the Blowdown sample panel could either be isolated or diverted to the radioactive waste system. System modifications will all take place on non-safety, non-seismic portions of the systems. System operation, function and individual component integrity will not be degraded by the DC. No new radioactive systems or new release paths will be created by the DC. Should the Blowdown system become radiologically contaminated an alternate drain path for the Blowdown sample panel is provided to the Liquid Waste Management (LWM) System. The added flow to the LWM system (approximately 38 gallons per day) will be insignificant to the LWM system (minimum design capacity of the LWM system is 18,675 gallons per day).

21. DC-3382, Control Room Annunciation Alarm Reduction (Revision 0)

DESCRIPTION OF CHANGE

DC-3382 will eliminate seven identified Control Room annunciation "nuisance alarms" and modify five Control Room annunciation alarms so that they will no longer be "nuisance alarms."

REASON FOR CHANGE

Elimination of the above "nuisance alarms" will reduce unnecessary distraction of the Operations staff and allow them to focus on valid alarms. This DC also allows Waterford 3 to continue its efforts to attain a "black board" in the Control Room.

SAFETY EVALUATION

According to the safety evaluation the DC will not alter the ability of affected systems to perform their functions. All of the alarms affected by the DC are for annunciation only. The evaluation states that other methods of indication or control (associated with these annunciation alarms) are used by Control Room personnel to monitor operation of the plant. The DC does not reduce the margin of safety as defined in the bases for any Technical Specification or the appropriate safety analyses.

22. DC-3383, Spent Fuel Handling Machine Upgrade (Revision 0)

DESCRIPTION OF CHANGE

DC-3383 upgrades the Spent Fuel Handling Machine (SFHM). The control console, power center, and hoist will be replaced with state-of-the-art equipment.

REASON FOR CHANGE

Upgrade of the SFHM will reduce maintenance and result in improved availability of the SFHM.

SAFETY EVALUATION

The SFHM is a non-safety, seismic category 1 machine that is used to handle new and spent fuel. Operation of the SFHM is not required for any safety functions. No credit is taken for components or subsystems of the fuel handling equipment to either prevent or mitigate the consequences of a Design Basis Fuel Handling Accident (DBFHA). The machine's design basis, operational functions, operational interlocks and procedures as described in the SAR are not changed by this DC. The likelihood of a DBFHA occurring is not affected by this DC

23. DC-3384, Emergency Diesel Generator (EDG) Overhead Rigging Enhancements (Revision 0)

DESCRIPTION OF CHANGE

This DC modifies the EDG Rigging System by raising the outside monorails from elevation 37' 6" to elevation 39' 6" and installing a 3 ton bridge crane spanning from the outer rails.

REASON FOR CHANGE

Insufficient clearance between the EDG and the existing overhead rigging systems makes the overhead rigging systems unusable. This DC will provide adequate vertical headroom for component installation/removal and would minimize outage manpower requirements. The bridge crane will allow full range of motion in all directions which will aid in the removal and installation of heavy diesel engine components.

SAFETY EVALUATION

The safety evaluation states that there are no postulated accidents in the FSAR affected by this DC. The modified system is designed as a seismic Class 1 structure capable of functioning under Design Basis Earthquake (DBE) load conditions. Although the bridge crane and hoist is classified as Seismic Category II, they are designed with a safety factor of 5 times their rated capacity. Since this produces stress values far lower than the allowable for Seismic Category I requirements, the II over I criteria is satisfactorily addressed. There are no protective boundaries affected by the DC.

24. DC-3385, SI Sump Isolation Valves Maintainability Modifications (Revision 0)

DESCRIPTION OF CHANGE

The DC adds flanges to the downstream side of Safety Injection (SI) valves SI-602A&B to allow access to the valves internals. The valves' leakoff lines will be cut and capped and the valves' stems will be modified to allow the use of the Valve Operation Test & Evaluation System (VOTES).

REASON FOR CHANGE

SI-602A&B, Safety Injection Sump Outlet Isolation Valves, are welded in place causing a lack of access to the valves internals. This lack of access hampers the ability to perform maintenance activities on the valves. Currently the valve stem length is to long to allow use of the VOTES torque plug. The DC will reduce the valve stem length to eliminate the interference with the VOTES testing equipment. Revision 1 added an acceptance test to the DC.

SAFETY EVALUATION

According to the safety evaluation the affected portion of the SI system is used during SI Recirculation Actuation for the mitigation of a Small or Large Break Loss of Coolant Accident (LOCA). The addition of the flanges and a drain to the downstream piping will not prevent any component from performing its safety function while the plant is operating. The evaluation notes that a pipe plug will be utilized to isolate containment atmosphere during performance of the DC. The plug mounting was evaluated by calculation EC-C95-001 and it will not fail during a seismic event.

All new piping components will be procured ASME Safety Class 2 (QC-1) and seismic category I, all the current seismic qualifications of the existing equipment and/or components will be maintained.

Modifications associated with DC-3385 will be performed during Mode 6 when the plant is in cold shutdown. Thus the consequences of a malfunction of equipment important to safety previously evaluated in the SAR will not be affected or changed.

25. DC-3387, Installation of Versa Vent (Revision 0 and Revision 1)

DESCRIPTION OF CHANGE

DC-3387 installs the Combustion Engineering "Versa-Vent" system on the Control Element Drive Mechanism (CEDM) pressure housings.

REASON FOR CHANGE

The present method for venting the CEDM pressure housings takes approximately 20 to 30 minutes per CEDM (91 total) and involves disassembly and maintenance of the ball seal housings. This takes a large part of critical path time. The Versa-Vent system allow for quick and efficient venting of the CEDMs during refueling outages.

SAFETY EVALUATION

The Loss of Coolant Accident (LOCA) is identified by the safety evaluation as the accident that would be affected by this DC. The evaluation determined that the DC has no effect on this previously evaluated accident. The safety evaluation notes that the Versa-Vent replaces the upper housing nut on the CEDM ball seal housing and that it has no interaction with the operation of the CEDM. The Versa-Vent is installed outside of the Reactor Coolant System pressure boundary and serves no safety function. Addition of the Versa-Vent system actually will help indicate a malfunction such as leakage past the ball seal because of the installed leak detection device in the Versa-Vent. The Versa-Vent system is seismically mounted and will have no effect on any other equipment. According to the evaluation there are no unreviewed safety questions associated with this DC.

26. DC-3389, Alternate Chemical Addition for Secondary System (Revision 0)

DESCRIPTION OF CHANGE

This DC will provide the ability to inject alternate amines and buffers into the secondary system. The DC installs two new chemical feed skids in the Turbine Generator Building. Both skids are designed "generically" to the extent practical to accommodate alternate chemicals. The DC will also allow the use of four new chemicals in the Waterford 3 secondary cycle; ethanolamine, morpholine, boric acid and ammonium chloride.

REASON FOR CHANGE

The existing chemical feed system was designed to add ammonia for pH control and hydrazine to scavenge trace levels of dissolved oxygen. The new chemical feed skids will provide the ability to inject a variety of chemicals into the secondary system.

SAFETY EVALUATION

The safety evaluation reflects that the DC does not result in an unreviewed safety, radwaste, or environmental question and confirms that all appropriate criteria for modifications to the affected systems have been met. The evaluation identifies the following FSAR accident analyses that could be affected by adverse secondary cycle chemistry:

Steam Generator Tube Rupture
Steam Line Break
Feedwater System Pipe Break

The introduction of ethanolamine, morpholine, boric acid and ammonium chloride into the secondary cycle was evaluated with regard to materials compatibility, steam generator effects, balance of plant effects, turbine effects and industry experience and testing. No adverse effects were identified as discussed in the "Evaluation of the Application of Alternate Chemical Control for the Waterford 3 Secondary Cycle" which was prepared for this DC. The application of these chemicals is expected to improve steam generator tube life, thus reactor coolant pressure boundary performance will be unaffected.

The DC only affects non-safety, non-seismic portions of the Chemical Feed, Condensate, Condensate Make-up, Feedwater and Potable Water systems.

27. DC-3394, Installation of Equipment and Tie-Ins for the Maintenance Support Building (MSB) (Revision 0)
(See also Item 117 of Waterford 3 Letter W3F2-94-0051, Report of Facility Changes, Tests and Experiments, dated October 20, 1994)

DESCRIPTION OF CHANGE

DC-3394 will enclose the MSB inside the protected area and provide all of the tie-ins necessary to render the building functional.

REASON FOR CHANGE

The MSB is a three story office building constructed outside of the Protected Area, on the north side of the plant and west of the Administration Building. To make the building functional this DC will:

Extend the physical security barrier to surround the MSB.

Add new security equipment to support extension of the physical security barrier.

Provide required tie-ins fc · electrical, fire protection, sewer, water, communications, lighting, and accountability.

SAFETY EVALUATION

According to the safety evaluation no SAR postulated accidents are affected by this DC. The modified equipment is non-safety related. Where the non-quality MSB fire protection system is connected to the Quality Related plant fire protection system, isolation valves are provided. This will allow for isolating the MSB in the unlikely event a failure of the MSB fire protection system is degrading the performance of the plant fire protection system. Electrical power for the MSB is provided from an off-site source except for the security system which will be powered from the security bus.

The evaluation concludes that the DC does not reduce the margin of safety as defined in the basis for any technical specification or safety analysis. The systems affected serve no safety function since they are not required for operation during safe shutdown of the plant following an accident or to mitigate the consequences of an accident.

28. DC-3396, Hydrogen Purity Meter and Pressure Monitor Replacement (Revision 0)

DESCRIPTION OF CHANGE

The DC replaces the existing pneumatic Hydrogen Purity Meter with an electronic Hydrogen Purity Meter.

REASON FOR CHANGE

The existing system is difficult to calibrate and maintain and due to the age of the equipment replacement parts are difficult to obtain.

SAFETY EVALUATION

The safety evaluation states that the Hydrogen Purity Monitor is a Q-4 system located entirely in the Turbine Generator Building and it is neither safety or quality related. It is not connected to any safety or quality related system thus, the system can not increase the probability of occurrence of any accident addressed in the SAR. The function of the monitor remains the same. The DC does not affect any protective boundary or margins of safety.

29. DC-3398, In-Core Instrumentation Thimble Replacement

DESCRIPTION OF CHANGE

This DC will replace up to 20 In-Core Instrument (ICI) thimbles with a more wear resistant thimble. It will also provide the capabilities to replace additional thimbles during subsequent refueling outages.

REASON FOR CHANGE

Currently eight ICI positions are not available due to missing or blocked thimbles. Another nine thimbles are heavily worn, and normal refueling activities could cause them to break or bend. By replacing the heavily worn and missing thimbles, the positions that are unavailable will be recovered, and risk to lose an additional position will be reduced.

SAFETY EVALUATION

According to the safety evaluation the ICI thimbles are a passive component and perform no specific safety function. The interface between the replacement thimbles and both the safety related Core Exit Thermocouples and the reactor internals is within the design analysis of the original thimbles. There is no impact to the safety related function of those components. The evaluation determined that there is no unreviewed safety question associated with the DC and it does not reduce the margin of safety as defined in the SAR or Technical Specification bases.

The replacement thimbles have been evaluated to meet the same form, fit, and function of the original thimbles. Thus the replacement of the ICI thimbles will not cause or affect any accidents listed in the SAR. The major difference between the original and the replacement thimbles is the method of connecting the replacement to the stub from the original thimble. The replacement uses a swagelok connection consisting of a nut with one end welded to the replacement thimble and swaged to the original stub. The nut is crimped to prevent it from becoming loose. The nut is also small enough such that it will not block nor affect Reactor Coolant Flow through the instrument tube.

30. DC-3401, Integrated Leak Rate Test (ILRT) Piping and Support Redesign/Reinstallation (Revision 0)

DESCRIPTION OF CHANGE

The ILRT portion of the Station Air (SA) system is designed to allow temporary air compressors to be connected to the Reactor Containment Building (RCB) to pressurize the containment vessel during periodic testing in accordance with Technical Specifications. This DC will reinstall SA line 7SA10-102 between the exterior wall of the Reactor Auxiliary Building Wing Area and RCB penetration #63. It will also install new and or redesigned seismic supports for the line and remove existing non-seismic supports.

REASON FOR CHANGE

During Refuel Outage 5 it was determined that portions of non-seismic line 7SA10-102 were routed over safety related components of the Component Cooling Water System. As an interim solution, sections of this piping were removed where it was routed over safety related components. To eliminate the non-seismic over seismic (II/I) concerns this DC seismically redesigns the affected portions of line 7SA10-102.

SAFETY EVALUATION

According to the safety evaluation no SAR postulated accidents are affected by the DC, the affected system is not safety related and no new system interactions are introduced. As noted in the evaluation the ILRT piping will be reinstalled with seismic I supports to eliminate II/I concerns where it is routed over safety related equipment, so the probability of occurrence of a malfunction of equipment important to safety is reduced.

31. DC-3402, Turbine Generator Building Battery Installation (Revision 0)

DESCRIPTION OF CHANGE

The DC removes and relocates part of the non-safety dc loads from the AB battery to a new non-safety Turbine Generator Building (TGB) battery. For increased reliability, the new non-safety TGB battery will have two chargers, one powered from each train (A & B). The AB battery will be reconfigured from 62 cells to 60 cells.

REASON FOR CHANGE

The DC will improve the ability of the AB battery to perform its safety related functions because the majority of the non-safety related loads are removed and re-assigned to the new TGB battery.

SAFETY EVALUATION

According to the safety evaluation no unreviewed safety question exists because of the DC. The evaluation determined that the ability of the AB battery to provide continuous dc power for safety functions will be improved by the removal of the large amount of non-safety loads. The AB battery will have a greater capacity than required by the sizing criteria in IEEE-Std-485-1983. The removal of the two cells from the AB battery will result in a lower battery float voltage and a higher volts per cell float voltage. Lowering of the battery voltage will help improve the life expectancy of the AB battery loads. The higher volts per cell will allow the battery to operate closer to the manufacturer's preferred recommendations. The seismic qualification of the AB battery rack will be maintained by the DC. The AB battery will continue to perform required safety functions for normal plant operations, safe shutdown, and Station Blackout.

32. DC-3405, Motor Operated Valve Modifications (Revision 0 and Revision 1)

DESCRIPTION OF CHANGE

The DC rewires the Shutdown Cooling (SDC) isolation loop no. 1 & 2 valves (SI-401 A & B) to provide adequate torque switch bypass during valve operation (flow and static conditions). It will also rewire the Control Room Outside Air Intake (CROAI) valves (HVC-201 A & B, HVC-202 A & B, HVC-203 A & B, and HVC-204 A & B) to use the limit switch to open and close the valve in lieu of the torque switch. Revision 1 of the DC added the Emergency Feedwater Pump Turbine (EFPT) Steam Line emergency and normal drain valves (MS-407 and MS-408) and the Reactor Water Storage Pool to Charging Pump suction valve (CVC-507. These valves will also be rewired to provide adequate torque switch bypass during valve operation.

REASON FOR CHANGE

Action required by NRC Generic Letter 89-10 determined that the torque switch bypass in the open stroke for the SDC, EFPT, and CHV valves listed above was inadequate. Additionally, the circuitry for the CROAI valves torque closed the valves which is not recommended by the valve vendor and could result in damage to the valves.

SAFETY EVALUATION

The safety evaluation states the DC rewires the SDC, EFPT, and CHV valves torque bypass limit switch to allow up to 100% bypass in the open direction and that this assures the capability for the valve to open in the event the torque switch fails to operate. This allows redundant capability to open the valves, thus, the operation of these valves is enhanced. The evaluation also notes that the change to the CROAI valves decreases the probability of failure or malfunction of the CROAI valves by limit closing, in lieu of torque closure, the valves to prevent damage to the mechanical stops. The evaluation concluded that the function of the valves is not changed by the DC and that no unreviewed safety question is created by the DC.

33. <u>DC-3407 (Plant Change) Diesel Generator Air Dryer Assembly Replacement (Revision 0)</u>

DESCRIPTION OF CHANGE

This DC removes each of the Emergency Diesel Generator (EDG) existing air dryer assemblies, which contain precoolers, and installs new air dryer assemblies without precoolers.

REASON FOR CHANGE

Each EDG has two starting air systems, two compressors and two air receivers. Air is discharged from the compressor through an air dryer assembly to the receiver. The air dryer assembly is a desiccant type and includes a precooler, prefilter and afterfilter. Emergency Diesel Generator's "B1" air dryer has developed a leak in its precooler inlet header and like-for-like replacement parts are unavailable.

SAFETY EVALUATION

The safety evaluation states that the EDG air dryer assemblies are non-safety related and are not needed for the safe shutdown of the plant. The assemblies are seismically supported which will eliminate any possibility of affecting any adjacent safety related equipment during a seismic event.

The evaluation determined that the precooler is not needed in order to obtain dry air. Comparison of the dew point temperatures between a dryer assembly with its precooler bypassed and one that used its precooler were identical. Moisture removal capability of the air dryers will be maintained.

The evaluation also discusses the automatic drain valves associated with the new air dryers and concludes that they will not create the possibility of a malfunction of equipment important to safety. The valves are spring loaded to close and fail in the closed position. A failed close valve would be easily identified by the gross moisture indicator located on the air dryer skid.

There is no margin of safety associated with the air dryers and precoolers.

34. DC-3409, Containment Spray Operation (Revision 0 and Revision 1)

DESCRIPTION OF CHANGE

The DC will modify the close control circuit of both Containment Spray (CS) Pumps by delaying the pump start signal for 1.9 seconds. It also modifies the Train "B" CS isolation valve (CS-125B), by adding an additional vent valve, to allow faster air venting.

Revision 1 of the DC installs jumpers around the relays installed, by Revision 0, to delay the pump start.

REASON FOR CHANGE

This DC is an enhancement to ensure proper valve operation in the presence of a significant volume of entrapped air and/or degraded valve conditions. Delaying the pump start will allow the valve to partially open prior to the resultant pressure surge. The additional solenoid vent valve will provide additional air bleedoff which will provide increased assurance of proper valve operation.

Revision 1 installs jumpers around the time delay installed by Revision 0. This will override the delay and allow the equipment to operate per the original design basis and satisfy current Technical Specification. These jumpers will remain installed until the submitted Technical Specification Change is approved and received.

SAFETY EVALUATION

The safety evaluation identifies Main Steam Line Break (MSLB), Loss of Coolant Accident (LOCA), Inadvertent CS, and Main Feedwater Line BreaK (MFLB) as accidents affected by this DC. The MFLB is bounded by the MSLB. The limiting containment pressurization event is identified as 75% power MSLB (with Main Steam Isolation Valve failure) with Off-site Power available. Calculation EC-S94-014 demonstrates containment peak pressure would be 43.6 PSIG, the FSAR reports containment design pressure as 44 PSIG. Calculation EC-S94-002 demonstrates that existing licensing basis analyses for MSLB remains valid for a 1.9 second time delay between Containment Spray Actuation Signal (CSAS) and CS pump start. This delay corresponds to delaying spray flow to containment by 0.945 full-flow-seconds. Calculation EC-S94-002 demonstrates that sufficient margin exists in analysis assumptions for CSAS timing and processing to account for a deprivation of 1.3 full-flow-seconds. Thus, the existing MSLB peak pressure and temperature calculations remain valid. Also per EC-S94-002 there is no reduction in the reported 0.4 PSI margin to containment design pressure (44 PSIG) associated with this DC.

The safety evaluation for Revision 1 addresses the installation of jumpers to bypass the time delay on starting the CS pumps. With the jumper installed the pumps will start upon receipt of CSAS even if the time delay relay coils fail. This operation is similar to the original design, the jumpers will not degrade the CS System performance.

According to the safety evaluations there is no unreviewed safety question created by this DC. Also, protective boundary and margin of safety as related to the performance of the containment are unaffected by the DC.

35. DC-3414, Enhancements to the Environmental Monitoring System (Revision 0)

DESCRIPTION OF CHANGE

The DC will replace the functional equipment in the Meteorological Monitoring System that is located in the Environmental Monitoring building and tower. The DC will be accomplished in two phases; 1) new system will use the existing wiring connecting the Meteorological instruments to the Plant Monitoring Computer (PMC), and, 2) after installation of the new PMC the Meteorological instruments will communicate by modem over two existing twisted shielded pair wiring.

REASON FOR CHANGE

Current meteorological monitoring equipment is obsolete and no longer supported by the vendor. Lack of available spare parts makes it very difficult and expensive to maintain the equipment.

SAFETY EVALUATION

According to the safety evaluation there are no accidents in the SAR that could be caused or affected by this DC. The only interface between the plant and this system is through the non-safety related PMC and the circuit breaker protected power distribution system. This DC has no affect on any of the protective boundaries, i.e., fuel cladding, reactor coolant system or containment.

36. DC-3415, Replacement of L&N Recorders for the Radiation Monitoring System (RMS) (Revision 0)

DESCRIPTION OF CHANGE

Existing 1, 2, and 3 pen analog Leeds & Northrup (L&N) recorders on CP-14A, CP-14B, and CP-52 will be replaced with multi-point digital Johnson Yokogawas (J/Y) recorders. The plant stack normal and accident process flow channels will no longer be recorded. The power supply for all the recorders on CP-52 will be changed to Non-Class 1E uninterruptible 120 vac on the new recorders.

REASON FOR CHANGE

L&N no longer supplies the existing recorders and there are several recorders which need replacement. The plant stack normal and accident process flow recorders are not used and are not required, therefore, these channels will no longer be recorded.

SAFETY EVALUATION

According to the safety evaluation there are no accidents listed in the FSAR that may be caused or affected by this DC. The recorders perform no automatic functions. The recorders provide historical records of radioactivity and process flow values. Recorders installed in CP-14A&B are seismically qualified and seismically mounted. Recorders installed in CP-52 are seismically qualified for II over I concerns (Calculation EC-C95-002 addresses all three panels).

The new recorders on CP-14A&B will be fed from the same power supply as the old recorders, CP-52 recorders will be powered from a non-safety related power supply. This will have no significant impact on the power supply margin.

EPRI Document TR-102348 guidance was also included in the evaluation to address the digital upgrade aspect of the DC.

The evaluation states that the recorders do not interface with any protective boundary and there will be no change to any protective boundary as a result of the DC.

37. DC-3416, CVC Deborating Ion Exchanger Redesignated Purification Ion Exchanger "C" (Revision 0)

DESCRIPTION OF CHANGE

DC-3416 is a "documentation and labeling" change that redesignates the "Deborating Ion Exchanger" as a third "Purification Ion Exchanger." There is no physical change to the Chemical and Volume Control System (CVCS) associated with this DC.

REASON FOR CHANGE

The letdown portion of the CVCS routes letdown flow through one of three identical ion exchangers. Two of these ion exchangers are called "Purification Ion Exchangers A & B" and the third is called the "Deborating Ion Exchanger." Redesignating the "Deborating Ion Exchanger" as a third "Purification Ion Exchanger" will result in the Operations Department having more flexibility during normal plant cycles and also during the end of a fuel cycle when more than one ion exchanger could be used for deborating.

SAFETY EVALUATION

The safety evaluation confirmed that the DC does not reduce the margin of safety as defined in the basis of any Technical Specification or safety analysis and no unreviewed safety questions are created. According to the evaluation the "CVCS Malfunction (Inadvertent Boron Dilution)" is the only accident analyzed which involves the CVCS. Redesignating the Deborating Ion Exchanger could allow for faster deboration, however, this "inadvertent dilution" is bounded by the moderate frequency incident analyzed in FSAR section 15.4.1.5. In this analysis pure, demineralized water is supplied to the suction of the charging pump via the Volume Control Tank. Initial conditions for this analysis require a 140 ppm boron dilution to reach criticality. The deborating capability of the anion resin requires two charges to reduce reactor coolant boron concentration from 30 ppm to 0 ppm (each ion exchanger volume can remove about 15 ppm boron). Thus the worst case dilution, all three ion exchangers in service containing anion resin, could only dilute the boron concentration by 45 ppm. The evaluation concludes that the DC does not increase the probability of an accident.

Operations procedure OP-002-005 will require verification of the correct resin for the intended function prior to placing a ion exchanger in service. This procedure along with administrative controls by the Operations and Chemistry departments ensure that correct resin is utilized for the desired application.

38. DC-3418, High Pressure (HP) Turbine Gland Steam Drain Improvement (Revision 0 and Revision 1)

DESCRIPTION OF CHANGE

DC-3418 will provide for the following:

1) increase the size of the upper drain connection from HP turbine gland chamber "X" from 1/2" to 2" diameter, 2) route new 2" O.D. tubing drains to connect directly in the 6" diameter gland steam "spillover piping" which is routed to the main condenser.

Also, the drain connection on the "Y" gland chamber will be changed from 1/2" to 1 1/2", new 2" O.D. tubing will be routed to the gland steam "exhaust piping" which is routed to the gland steam condenser.

New supports will be added for the new drain tubing and a new pressure test connection will be installed upstream of spillover pressure regulating valve GS-112.

REASON FOR CHANGE

Steam leakage from the outer glands of the HP turbine has persisted despite Maintenance Department efforts to correct the leakage. The turbine vendor has recommended several enhancements to minimize and eliminate the leakage.

SAFETY EVALUATION

The safety evaluation confirmed that the DC will not reduce the margin of safety as defined in the basis of any Technical Specification or safety analysis and no unreviewed safety questions are created. The Gland Steam System serves no safety function since it is not required to achieve safe shutdown or mitigate the consequences of an accident. According to the evaluation no SAR postulated accidents are affected by the DC. The "Loss of Condenser Vacuum" accident scenario is not affected because no new flow paths or system interactions are created.

39. DC-3421 (Plant Change), Nitrogen Accumulator Outlet Header Check Valves (Revision 0)

DESCRIPTION OF CHANGE

DC-3421 removes the internals from nitrogen accumulator check valves; NG-617, NG-618, NG-717, NG-718, NG-817, NG-818, NG-917, and NG-918. The valves are located in the outlet headers of safety related nitrogen accumulators.

REASON FOR CHANGE

The check valves affected by this DC serve no safety function except to open to allow full flow for the operation of downstream Air Operated Valves (AOV) if Instrument Air (IA) is unavailable. The check valves prevent backflow of IA to the affected nitrogen accumulators if they should become depressurized. However, assurance of nitrogen flow to the AOVs is more important. The In-service Testing Program includes testing of the downstream check valves to verify their ability to allow full flow. Testing of these check valves is difficult to accomplish.

SAFETY EVALUATION

According to the safety evaluation the loss of nitrogen will not initiate any accident which has been evaluated in the SAR. The safety function of the check valves is to open and allow full flow, by removing the valves' internals the system will not depend on the function of the check valves. This change will increase the reliability of the nitrogen system by assuring full flow downstream of the nitrogen accumulators.

40. DC-3426, WCT Basins Chemical Addition & Filtration System

DESCRIPTION OF CHANGE

This DC will install a new non-safety related independent pump and filtration system for each Wet Cooling Tower (WCT) basin to allow for water filtration without operating the existing safety related Auxiliary Component Cooling Water (ACCW) Pumps. Included in this DC is the installation of seismically designed suction screens for the suction piping of the ACCW pumps in each WCT basin. It will also install permanent facilities for adding and distributing water based corrosion inhibitor and biocide type chemicals in the WCT basins.

Elevation of the WCT basin water level will be raised from the present level of -9' 9" to a new level of -9' 5 1/8".

REASON FOR CHANGE

There are presently no permanent provisions for filtering or treating the water in the WCT basins to maintain proper pH and purity. Waterford 3 Significant Occurrence Report No. 92-008 documented that a large plastic bag was inadvertently dropped into WCT "A" basin during routine chemical addition. The primary concern being the potential to damage the ACCW pump or clog the pump suction.

Increasing the WCT basin level provide approximately 3 7/8" of water over the siphon breaker holes placed in the Filtration pump suction piping at elevation -9' 9".

SAFETY EVALUATION

According to the safety evaluation the failure of support system components included in this DC cannot initiate a LOCA or any other accident previously evaluated in the SAR. No new system interactions will be created by the DC and use of specific chemicals will be controlled by Chemistry Department procedures. Engineering calculation EC-C94-018 considered the effects of seismically induced wave actions, on the suction piping screens, during LOCA induced low water levels. The new filtration system and pumps are classified as non-safety and are not needed to mitigate the consequences of any accident.

The setpoint change for the basin level is in the conservative direction and will result in an increased water inventory for accident mitigation.

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42. DC-3430, Replacement Valve Assembly for Letdown Heat Exchanger Temperature Control Valve (CC-636), (Revision 0)

DESCRIPTION OF CHANGE

The DC replaces the existing six inch butterfly Component Cooling Water to the Letdown Heat Exchanger (LDHX) flow control valve (CC-636) with a globe valve with equal flow characteristics. The DC will also change the fail safe position of the valve by making it a "Fail Closed" valve.

REASON FOR CHANGE

CCW flow through the LDHX will be much easier to control, especially at low flow rates, with a higher flow accuracy. With the valve as a "fail closed" valve there will no longer be surveillance requirements for the valve's accumulator because the valve will fail in its desired position during an accident, and the valve will not be dependent upon the accumulator pressure to maintain its closed position.

SAFETY EVALUATION

According to the safety evaluation there are no accidents in the SAR which are attributed to failure of CC-636. The valve is part of the safety related, non-essential loop of the CCW system which is isolated during accident conditions. It is required to remain closed to conserve CCW flow to the essential, safety related components which are used to mitigate an accident. The valve will continue to serve the same function and there are no new system interfaces created. The "fail closed" position will make it less likely that a malfunction will occur, making it more reliable for its post-accident function.

43. DC-3432, Boronometer and Process Radiation Monitor Deletion (Revision 0)

DESCRIPTION OF CHANGE

This DC removes the annunciators supplied by the Boronometer and Process Radiation Monitor from the Chemical and Volume Control System (CVCS) and the Process Radiation Monitoring System (PRM).

REASON FOR CHANGE

The boronometer and the process radiation monitor are presently declared Out of Service and will not be made functional. This DC allows the annunciator windows to be permanently de-energized by sparing the connecting cables. The Radiation Monitoring System (RMS) Database Manual will be revised to prevent the RM-80 from activating the high activity summary alarm for non-safety monitors.

SAFETY EVALUATION

The safety evaluation determined that removing these annunciators does not create any unreviewed safety questions. The evaluation states that there are no accidents listed in Chapter 15 of the FSAR that may be caused by or affected by this DC. FSAR Table 7.5-3 lists the sampler and indicator (not the annunciator) of the boronometer as Regulatory Guide 1.97 Type B components. These components are classified as Category 3 (commercial grade quality) which are not required to function following an accident. The letdown flow path (sample point for these instruments) is isolated by engineered safety features actuation signals, these annunciators perform no role in mitigating any accidents. These instruments serve as a backup to analysis results from the primary sample system or the PASS. Emergency Operating procedures do not employ these annunciators so control room operator actions in response to accident precursors will not change as a result of this DC.

44. DC-3434, EH System Constant Pressure Pumps (Revision 0)

DESCRIPTION OF CHANGE

The DC replaces the existing Westinghouse furnished Electro Hydraulic (EH) fluid pumps with new Westinghouse furnished constant pressure pumps. Also, flow meters will be provided in the EH pump casing drain lines and in the pump discharge lines. Existing unloader valves will be removed and cover plates will be installed in their place.

REASON FOR CHANGE

The current EH system uses unloader valves to regulate fluid pressure between 1800 and 2150 psig, operation of these valves results in pressure impulse vibrations. These vibration accelerate tubing fatigue and cause unnecessary leaks in the system. The unloader valves also have a history of failures.

SAFETY EVALUATION

The evaluation confirmed that the DC does not reduce the margin of safety as defined in the basis for any Technical Specification or safety analysis and no unreviewed safety question was identified. The evaluation identified the turbine trip event and turbine trip with a single active failure as events that may be caused by the DC. The evaluation notes that the constant pressure pumps have a proven performance and reliability record, and that the DC provides additional monitoring instrumentation to permit evaluation of pump performance and degradation. The DC will not increase the probability of occurrence of an accident previously evaluated because the integrity of the existing non-safety related EH system will be maintained and the reliability will be improved.

The EH system is non-safety related and non-seismic and does not connect to or interface with any system that is safety related or needed for safe shutdown.

45. DC-343?. (Plant Change) Valve Actuator Quick Disconnects for Air Operated Valves (AOV) (Revision 0)

DESCRIPTION OF CHANGE

This plant change installs permanent test connections at the required locations for ten (10) AOVs. Each test connection will consist of a quick-connect coupling and an isolation valve. Valves affected by this plant change are; FW-166A&B (Startup Feedwater Regulating valves), FW-173A&B (Feedwater Regulating valves), MS-319A, B, & C, and MS-320A, B, & C (Main Steam Bypass valves).

REASON FOR CHANGE

The new test connections will eliminate the requirement to disassemble the tubing to accomplishing testing of the valves.

SAFETY EVALUATION

According to the safety evaluation the plant change will not increase the probability of an accident previously evaluated in the FSAR because the integrity of the existing non-safety related valve actuators will be maintained, the operation of the valves is not affected, valve testing will not be accomplished with the valves in service and reliability will be improved. All tubing and fittings will be procured Q-2 to ensure the integrity of the Instrument Air System pressure boundary. The valvop portion of the Startup Feedwater Regulating and the Feedwater Regulating valves are non-safety, seismic. The test connection additions have been analyzed to ensure there is no impact on the seismic qualifications of the valves. The valvop portion of the Main Steam Bypass valves is also non-safety and non-seismic.

46. DC-3438, Main Turbine Lube Oil Filtration Enhancement (Revision 0)

DESCRIPTION OF CHANGE

This plant change will replace the Main Lube Oil Centrifuge with a Turbine Oil Conditioner skid within the oil containment curb. The skid provides a filter unit for particulate filtration and a coalescing filter for water removal.

REASON FOR CHANGE

Existing lube oil purification filters are not capable of maintaining the turbine lube oil at desired cleanliness levels. The Main Lube Oil Centrifuge is a high maintenance item and has become unreliable and ineffective.

SAFETY EVALUATION

According to the safety evaluation the "Turbine Trip" event could be related to lube oil and seal oil problems. This plant change will improve the cleanliness of the Main Turbine Lube Oil System which is expected to enhance the availability of the system and lessen the probability of a Turbine Trip due to poor oil quality.

The Main Turbine Lube Oil system is non-safety and non-seismic and is not required for mitigation of any accident described in the FSAR. The system does not connect to nor does it interface with any system that is safety related or needed for safe shutdown.

47. DC-3441, SI 502A&B Actuator Replacement (Revision 0)

DESCRIPTION OF CHANGE

This DC replaces the existing Limitorque SMB-000 actuators and 5 ft-lbs motors on the High Pressure Safety Injection (HPSI) Isolation valves with Limitorque SMB-00 actuators and 10 ft-lbs motors. The overall actuator ratio (OAR) will be increased from 68.0 to 101.3 to maximize the capability of the new actuator.

REASON FOR CHANGE

The existing motor operators are considered marginally sized based on their ability to deliver the thrust and torque required by the valve to perform its design basis safety function. The capability of the existing motor operators is limited by their design torque rating of 90 ft-lbs which cannot be increased.

SAFETY EVALUATION

The safety evaluation determined that no unreviewed safety questions exist as a result of this DC. The DC will significantly increase the margin above the minimum design basis requirements thus providing added assurance that the valve is capable of performing its safety function. The Emergency Diesel Generator and Fuel Oil Storage Tanks were evaluated with the increased motor loads and determined to be within their respective design ratings and capacities. The final EDG electrical loading and fuel requirements will only slightly increase as a result of this DC. (Calculation EC-E90-006.)

Two accidents identified in the evaluation, Loss of Coolant Accident (LOCA) and LOCA with Loss of Power (LOOP) are not affected by the DC. The DC results in increasing the stroke time of the valves from approximately 30 seconds to approximately 50 seconds. This slight delay in initiation of HPSI hot leg injection does not affect any existing accident analysis.

48. DC-3442, Oil Separator Waste Oil Piping Improvements (Revision 0)

DESCRIPTION OF CHANGE

This Plant Change (DC-3442) will raise the Oil Recovery Header piping approximately 5-1/2 inches inside the separator polishing bay, as recommended by the manufacturer. It will also install a 1 inch drain on the oil recovery header to allow for manually checking for excess oil accumulation.

REASON FOR CHANGE

The oil separator has a history of unsatisfactory performance and it has been determined that the elevation of the oil recovery header in the polishing bay is too low for obtaining the proper influent, separator/coalescing pack and weir relationship. The existing oil recovery header elevation allows clean water to be needlessly discharged to the Waste Oil Collection Tank. In addition, waste oil buildup can not be monitored to determine removal necessity.

SAFETY EVALUATION

According to the safety evaluation the Oil Separator is not used to mitigate the consequences of any FSAR accident scenario. The Oil Separator is non-seismic, non-safety, Quality Class 4 and in no way affects any safety-related system or any equipment important to safety. All plant effluent passes through radiation monitor PRM-IRE-6778 prior to entering the Oil Separator, if activity is detected the flow stream is automatically isolated.

49. <u>DC-3444, DCT Sump Pump Timer (Revision 0)</u>

DESCRIPTION OF CHANGE

This DC adds run time totalizers for the Dry Cooling Tower (DCT) area sump pumps (one for each of the four pumps).

REASON FOR CHANGE

The DC provides a means of easy retrieval of field data (elapsed run time of the DCT area sump pumps) which is used to develop data required for the plant's discharge reporting requirements.

SAFETY EVALUATION

The safety evaluation did not identify any unreviewed safety questions associated with this DC. There are no accidents identified by the evaluation that are impacted by this DC. Circuits impacted by this DC are electrically isolated from the safety related portion of affected Motor Control Centers (MCC). As stated in the evaluation calculation EC-C94-017 evaluated the seismic qualification of MCCs due to the addition of the totalizers, seismic requirements are satisfied The DC does not cause any change to any protective boundaries or equipment important to safety, thus the DC will not reduce any margin of safety as defined for the bases of the Technical Specifications or safety analysis.

50. <u>DC-3449 (Plant Change), Modification of "Battery Station Supplying Load" and "Battery Drain" Alarms (Revision 0)</u>

DESCRIPTION OF CHANGE

This Plant Change removes the "Battery Station Supplying Load" alarm from the safety related Static Uninterruptable Power Supplies (SUPS) MA, MB, MC, and MD and non-safety related SUPS AB. The setpoint for safety-related SUPS A and B "Battery Drain" alarm will be lowered to effectively delete the alarm.

REASON FOR CHANGE

Safety related SUPS are required to provide a continuous supply of regulated power for safety-related equipment during all plant conditions. The function of the alarms is to warn the operator that the inverter is being supplied by the dc source (no ac input power). There is no automatic shutdown feature associated with the alarms. There is no scenario which the alarms are necessary.

SAFETY EVALUATION

According to the evaluation this Plant Change will not affect the operation of the SUPS or their ability to perform their safety functions. Affected circuits are for "alarm only" and there are no automatic shutdowns associated with them.

Status of the SUPS will be monitored by operations on the shift logs. Other alarms are available in the control room should the battery charger fail and the battery starts to drain. These alarms are: Charger; High Voltage Shutdown, No Charge, AC Power Failure, and DC System; Undervoltage.

51. DC-3451, Modification of In-Core Instrument (ICI) Flanges to Quickloc Design (Revision 0)

DESCRIPTION OF CHANGE

The DC replaces all ten of the ICI flanges with a new ABB Combustion Engineering Quickloc design. Also, the existing ICI assemblies will be replaced with instruments compatible with the Quickloc design.

REASON FOR CHANGE

Installation of the Quickloc ICI flange will eliminate the need to remove the 56 ICI assembly castle nuts and the 10 Grayloc clamps each refuel. Each one of these connections is a Reactor Coolant System (RCS) seal and requires torquing to complete installation. By reducing the number of times they are assembled/disassembled, the probability of human error and risk of an RCS leak is reduced.

SAFETY EVALUATION

According to the safety evaluation a Loss of Coolant Accident (LOCA) is the accident in FSAR that may be caused or affected by this DC. The ICI flanges are part of the RCS pressure boundary, however, the Quickloc flanges have been designed to meet the same seismic category 1 and ASME Class 1 requirements as the original flanges. Therefore installation of the Quickloc ICI flanges and installation of the ICI assemblies will not cause nor affect a LOCA or any other accidents listed in the FSAR.

The ICI flanges are a passive component and their only safety related function is to provide an RCS pressure boundary. Installation will not create any new or unique requirements/loading on the reactor vessel instrument nozzles. Installation of the Quickloc will not degrade the RCS pressure boundary, nor reduce any margin of safety as related to boundary performance.

52. DC-3452, Main Steam Isolation Valve (MSIV) Enhancements

DESCRIPTION OF CHANGE

DC-3452 will replace the existing Hydraulic Power Unit (HPU) pump start counters with electromechanical counters, replace the HPU reservoir rupture disc with a vacuum reinforced rupture disc, and modify the HPU skid to allow connection of a self contained recirculation and filtration unit, when required.

REASON FOR CHANGE

The existing pump start counters (used for trending purposes) have proven unreliable and the currently installed rupture disc is weakened by occasional negative pressure transients in the reservoir. The HPU skid has no filtering capability and over time the fluid (FYRQUEL) becomes increasingly acidic which greatly increases corrosion rates throughout the MSIV fluid system.

SAFETY EVALUATION

According to the safety evaluation the concern associated with connecting and operating the purification skid is that it may cause an inadvertent closure of the MSIV. However the evaluation determined that the DC will not increase the likelihood of an inadvertent closure of the MSIV because the purification skid recirculates oil from the HPU reservoir which is outside the safety boundaries of the MSIV. If a leak should occur there is sufficient time available for the skid to be disconnected from the reservoir and the leak stopped. Should a leak occur an annunciator will alarm in the control at 12.25" level in the reservoir, the HPU pump will continue to start, if required, until the reservoir level reaches 10.25" (this 2" drop is approximately 8.8 gallons). According to the evaluation the worst case leak will take approximately 6.7 minutes for the 2" level decrease to occur. Thus with immediate action taken to isolate the leak there is no impact on the MSIV.

No equipment important to safety will be affected by this DC, the safety function of the MSIV, to shut, will not be impacted. The DC does not affect any protective boundary or margins of safety.

53. DC-3453 (Plant Change), Low Level Lock-out Switch, Continuos Air Vents and Transfer System Changes (Revision 0)

DESCRIPTION OF CHANGE

This plant change modifies the Electrohydraulic (E-H) Pump 1 and 2 control circuit to allow continued pump operation with a failed low-low level lockout switch or an actual low-low level lock out switch actuation. The change also provides for alarming the failure of the low-low level lockout switch, modifies the EH High Pressure fluid control system reservoir filter housings to install automatic air eliminators, and modifies the E-H High Pressure fluid control system reservoir to provide an extension of the existing fill connection for adding EHC fluid.

REASON FOR CHANGE

Modification of the low-low level lockout switch is in response to a re-evaluation of the trip and lockout scheme by the turbine manufacturer, Westinghouse Operation and Maintenance Memo 122. Not tripping the operating pump but continuing to lockout the standby pump will preclude inadvertent turbine trips. Other modifications associated with the change will improve filter efficiency and reduce operator maintenance activities.

SAFETY EVALUATION

According to the safety evaluation the E-H System is non-safety and non-seismic. The only exception is the seismic mounting of the E-H Pump control switches on Control Panel (CP)-1. Seismic mounting of the control switches on CP-1 will be maintained. The evaluation states that the changes will not affect accidents evaluated in the SAR.

54. DC-3455 (Plant Change), Modification of the Drain Line for the Heat Exchangers for the Condenser Wide Range Gas Monitor (WRGM)

DESCRIPTION OF CHANGE

The Condenser WRGM (PRM-IRE-0002) monitors the exhaust of the Air Evacuation pumps for activity and diverts the discharge from atmosphere to the Reactor Auxiliary Building Filter Unit on detection of activity above it's setpoint. The Plant Change adjusts the setpoint of the WRGM Moisture Control Unit (MCU) chiller from 49 degrees F. to 39 degrees F. It also changes the drain configuration from a loop seal (installed by TAR-94-019, Item I.C.4 of this report) to a drain trap, check valve and three-way valve. The three-way valve will allow sampling of the chiller condensate and constant discharge to a local floor drain.

REASON FOR CHANGE

Original installation of the MCU drained to a common line containing a drain trap, check valve and three-way valve. Debris from the heat exchangers of the MCU caused the 5/64" orifice of the drain trap to clog. TAR-94-019 corrected this problem by installing a loop seal in the heat exchangers drain line. This change removes the loop seal and installs separate drain lines. In addition to the drain line modification the change also changes the setpoint of the chiller temperature switch to ensure that most moisture is removed in the heat exchangers.

SAFETY EVALUATION

According to the results of the safety evaluation there is no unreviewed safety question associated with the change and the change does not impact a radioactive waste system. The safety evaluation was performed because the change results in a change to FSAR Figure 10.4-2.

The safety evaluation notes that the equipment is used for the detection of activity and it does not affect the initiation of any event and will not increase the probability or consequences of any accident previously evaluated in the SAR. The change only affects the removal of moisture from the sample stream of the condenser WRGM. The WRGM will continue to operate and detect any condenser activity without any changes to the alarm setpoint of the WRGM. Therefore the margin of safety is not reduced or affected.

55. DC-3457 (Plant Change), Fire Protection Service for the Low Level Radioactive Waste Storage Facility (Revision 0)

DESCRIPTION OF CHANGE

DC-3457 add a new 6 inch tee in valve pit No. 10, between valve FP-338 and fire hydrant No. 10. This connection will be routed to a new fire hydrant (No. 10A) to provide protection for the new Low Level Radwaste Storage Facility (LLRWSF).

REASON FOR CHANGE

Because of continued delays in opening a regional radwaste disposal facility Waterford 3 has constructed an on-site facility to provide interim storage space for a total of five years. The LLRWSF is located outside the protected area on the west side of the plant. Fire protection for the facility is accomplished passively through the use of physical barriers and administrative controls which are required to eliminate the introduction of ignition sources in the building. One fire hydrant, connected to the plant fire protection water supply system, and appropriately equipped with a hose house and appropriate equipment will be located within NFPA recommended distances from the facility.

SAFETY EVALUATION

According to the safety evaluation the integrity of the fire protection system will be maintained. The modification will be performed in accordance with NFPA 24, NFPA 801, and ANSI B31.1 requirements. The Fire Marshal for the State of Louisiana and American Nuclear Insurers (ANI) approved the fire protection for the LLRWSF. No new methods of failure are introduced by this DC and no new type of plant system interactions are caused.

56. DC-3459, Loss of Remote Shutdown Capability During a Control Room Fire (Revision 0 and 1)

DESCRIPTION OF CHANGE

This DC adds a new limit switch contact to the open control circuit and electrically relocates the torque switch and bypass limit switch of the close control circuit of Reactor Coolant System (RCS) loop Shutdown Cooling (SDC) System isolation valves SI-407A&B.

REASON FOR CHANGE

The Waterford 3 evaluation associated with NRC Information Notice 92-18, "Potential for Loss of Remote Shutdown Capability During a Control Room Fire," determined that physical damage could occur to the valve actuators of Safety Injection (SI) valves SI-407A&B. A locked rotor condition could occur from a hot short that bypasses the limit and torque switch, and energizes the motor circuit. The change prevents physical damage to the valve actuators when a postulated hot short occurs that bypasses the limit switch/torque switch and then energizes the MOV control circuit.

SAFETY EVALUATION

According to the safety evaluation the DC will not impact any present Loss of Shutdown Cooling analyses. The operation and function of the valves will remain unchanged. The change will reduce the probability of actuator damage to the motor operated, fail-as-is valves, due to a postulated hot short that could originate from a control room fire.

The evaluation states that the valves have two primary functions; 1) provide outside containment isolation during normal operations, and 2) provide SDC suction to the Low Pressure Safety Injection pumps during shutdown cooling operations. The valves are normally key locked closed and de-energized during normal operations.

57. DC-3460, Heater Drain system Alternate Level Control Valves (Revision 0)

DESCRIPTION OF CHANGE

This DC replaces the alternate drain valves that serve the #1 High Pressure Feedwater Heaters, the #2 Intermediate Pressure Feedwater Heaters, and the Moisture Separator Reheater (MSR) Drain Collector Tanks.

REASON FOR CHANGE

The alternate drain valves are being replaced because they do not provide a positive shutoff and seat leakage has been identified as a contributor to reduced megawatt generation.

SAFETY EVALUATION

The safety evaluation addressed Feedwater System events (temperature decrease, flow increase, pipe breaks) but determined that these events are not affected by the DC because reliability and integrity of the valves being replaced will not be degraded. The new valves and piping modifications necessary to install them will be performed in accordance with ANSI B31.1. Long term integrity of the new valves is expected to be better than the original because of the corrosion resistant materials selected for the components. Load changes associated with the DC do not exceed the acceptable criteria contained in EPRI Report No. 5939, dated May, 1988, thus there are no support modifications or additions associated with the DC.

The new valves have the same capabilities as the existing valves and they have better sealing capabilities which will reduce seat leakage. Controls for the alternate level control valves are not altered by the DC.

58. DC-3461, Eliminate SIT Leakage into the Reactor Drain Tank (Revision 0)

DESCRIPTION OF CHANGE

This DC eliminates two potential sources of non-Reactor Coolant System (RCS) leakage into the Reactor Drain Tank (RDT). The Safety Injection Tank (SIT) recirculation header drain valve and the SIT drain header relief valve discharge will be routed to the containment sump

REASON FOR CHANGE

There are three possible sources of leakage into the Reactor Drain Tank which are not valid sources of RCS leakage. Leakage from these sources could result in identified RCS leakage being greater than actual. Because of the limitation associated with unidentified RCS leakage it important that all sources on non-RCS leakage be eliminated from the RDT.

SAFETY EVALUATION

The safety evaluation determined that the DC does not result in an unreviewed safety question and does not reduce the margin of safety as defined in the bases of any technical specification. According to the evaluation no SAR postulated accidents are affected by the DC. The change re-directs two sources of SI leakage from the RDT to the containment sump. The piping and valves and fittings affected are not safety related and non-seismic. The functions of SI-339 (SIT recirculation header relief valve) and SI-342 (SIT recirculation header drain) are not affected by the DC.

59. DC-3463, Service Building Extension (Revision 0)

DESCRIPTION OF CHANGE

This activity consists of constructing a two story building on the north side of the Service Building within the protected area. The DC will expand the existing Service Building by 40 feet, adding approximately 9600 square feet of working area. The DC also provides all necessary tie-ins to render the building functional.

REASON FOR CHANGE

The Service Building is located on the West side of the plant, inside the protected area, and houses a warehouse and three maintenance groups. Work space for the three groups has become overcrowded and maintenance shop space is not sufficient to meet the needs of the three groups.

SAFETY EVALUATION

The safety evaluation states that there are no postulated accidents in the SAR that would be affected by the expansion of the Service Building. Electrical power for the extension will be provided from a non-safety related Motor Control Center. Safety related plant structures are designed to withstand impact effects from the design basis missiles, missiles selected to be representative of construction site debris. Thus, if parts of the Service Building extension become airborne in a tornado the resulting missiles would be enveloped by the design basis spectrum. The new extension is not designed as a Seismic Class I structure, however, its location is far enough from the Nuclear Plant Island that if it were to collapse it would not affect any safety related equipment. According to the evaluation systems to be modified for the Service Building extension are non-safety related and no new system interactions are created. The DC does not affect any protective boundaries.

60. DC-3464, Boric Acid Make-up (BAM) Tank Level Instrument Replacement (Revision 0)

DESCRIPTION OF CHANGE

Replace the existing BAM Tank Foxboro air bubbler type level instrument with a Rosemount model 1152GP transmitter. The local indication associated with the Foxboro instrument will not be available with the new level instrument (Rosemount transmitter).

REASON FOR CHANGE

The DC is expected to increase the reliability of the BAM Tank level indication because the new instruments have liquid filled impulse tubing which should reduce the possibility of crystallization in the tubing. Also, operation of the transmitters will no longer be dependent upon Instrument Air (IA).

SAFETY EVALUATION

The safety evaluation determined that an unreviewed safety question does not exist as a result of this DC. The BAM Tank level Transmitters do not perform a safety function other than to provide a pressure boundary to prevent leakage of Boric Acid Solution. They are, however, used for Technical Specification compliance. The new installation will a more accurate and more reliable source of BAM Tank level measurement.

Steam Line BreaK, SG Tube Rupture, and LOCA are accidents identified in the evaluation that may be affected by this DC because BAM Tank inventory has an effect on the events. The evaluation states that all tubing for the transmitter will be installed as safety class 3, seismic category 1 to prevent leakage of boric acid solution. The Rosemount 1152GP transmitter is seismically qualified to maintain the BAM Tank pressure boundary during a seismic event. The evaluation notes that the FSAR does not identify the transmitters as "Safety Related Display Instrumentation" or "Accident Monitoring Instrumentation." Replacement of the instrumentation does not change the consequences of a BAM Tank malfunction. The new instrumentation will provide a more reliable and accurate source of BAM Tank level.

61. <u>DC-3467, CVC-101, 103, and 109 Solenoid Valve Relocation/Replacement (Revision 0)</u>

DESCRIPTION OF CHANGE

The DC relocates the existing solenoid valves, and regulators, for Chemical and Volume Control (CVC) valves CVC-101 and CVC-103 to a new location outside of the Regenerative Heat Exchanger Cubicle. Also, the volume boosters for CVC-103 and CVC-109 will be removed. Instrument Air (IA) service valves IA-94831 and IA-94841 will be spared in place and new valves IA-94831 and IA-94841 will be added as service to the relocated solenoid valves.

REASON FOR CHANGE

The harsh environment of the present location for these solenoids is one of the contributing causes of premature failure identified in Condition Report (CR)-95-0489 and the follow-up root cause analysis. The new location will result in a lower radiation and temperature environment. Removal of the volume booster (CVC-103 and CVC-109)will result in the removal of an unnecessary component.

SAFETY EVALUATION

According to the safety evaluation CVC-101 has no closure time requirements per the FSAR. The safety analysis does not credit the closure of CVC-101. The safety analysis credits Safety Injection Actuation Signal (SIAS), which includes a contact for Containment Isolation Actuation Signal (CIAS), and achieves isolation with CVC-103 and CVC-109.

There will be additional air volume to be removed when CVC-103 is required to close. This delay in closure (due to frictional forces of the air through the tubing and the additional air volume) will be a maximum of 2 seconds. The valves will close within the 10 seconds required limit. (IST data indicates a current closure time of 3 to 4 seconds.) Removal of the volume booster from CVC-103 and CVC-109 will result in the venting of the air through the solenoid port, which is larger than the vent path of the booster. This will not result in increasing the valve closure time. Seismic qualifications of the affected components will be maintained in the new location.

62. DC-8001, (Plant Change) Extraction Steam Isolation Logic Change to Improve Reliability (Revision 0)

DESCRIPTION OF CHANGE

This plant change will remove the "seal-in" logic currently associated with Extraction Steam (ES) isolation valves ES-109 and ES-205 control logic circuits. This will result in the interruption of the closure signal (without operator action) to the valve if the High or High-High level conditions, in the #1 or #2 Feedwater Heaters, clears.

REASON FOR CHANGE

The plant change will keep the ES isolation valves (to the #1 #2 Feedwater Heaters) from unnecessarily closing, without any operator action, once a high level condition has cleared. This will increase the system reliability. The function of the valves is not explicitly mentioned in the FSAR. The implicit function is to automatically shut during valid high level conditions in the associated feedwater heaters. This change will not alter that function. It will alter the manner by which this automatic function is terminated (by operator action) once the initiating high level condition clears. Operator action will be required to reposition the valve as before.

SAFETY EVALUATION

According to the safety evaluation the plant change will have no impact on the consequences of any accidents previously evaluated in the SAR. The ES isolation valves are not required to function for radiological releases or accident mitigation, and are not credited in any accident analyses. Turbine Trip due to high water level in one of the Moisture Separator Reheaters (MSR) Shell Drain Tanks (SDT) was considered because the change will enable the automatic closure signal of ES-205, due to high level conditions in any of the #2 Feedwater Heaters, to be manually overridden. Since the four MSR-SDTs drain to the shell side of the #2 Feedwater Heater, overriding this signal could theoretically impact levels in one or more SDTs. However, this scenario was considered negligible for the following:

control switch for ES-205 would have to be continuously in the open position while high level indication persisted,

Reverse Current Valves in the MSR-SDT drain lines would prevent a backflow scenario of this nature, SDT Alternate Level Control Valves would open to correct level deviations in the associated SDT, and

MSR-SDT High Level alarms would provide another warning to the operator, indicating that overriding the ES isolation valve logic is inappropriate and should be terminated.

63. DC-8003, (Plant Change) Bearing Lube Water Strainer for CW Pumps (Revision 0)

DESCRIPTION OF CHANGE

Plant Change 8003 adds a duplex strainer to the cooling water supply line for the Circulating Water (CW) pumps.

REASON FOR CHANGE

The existing strainer arrangement includes twelve (12) (3 for each CW pump) strainers (75 microns). The plant has experienced problems with these filters clogging every few months which then requires the filter to be changed. The filter must be isolated to be changed which means that the respective CW pump must be secured.

SAFETY EVALUATION

According to the safety evaluation none of the equipment affected by this plant change is safety related, quality related or important for any accident mitigation. There are no new system interactions as a result of this plant change and the function of the system remains unchanged. The change provides a large duplex strainer in place of 12 smaller strainers and allows for on-line maintenance and eliminates the need to secure the affected CW pump for strainer cleaning. None of the equipment is a boundary to any safety related system.

64. DC-8005, (Plant Change) Fuel Sipping System Upgrade (Revision 0)

DESCRIPTION OF CHANGE

This plant change will abandon the existing "dry" fuel sipping system and install provisions on the fuel hoist box for connecting a new on-line refueling mast "wet" fuel sipping system. The Nitrogen and Gaseous Waste Management (GWMS) systems will be affected by this change only because of labeling changes for those systems, there will be no physical changes to these systems.

REASON FOR CHANGE

The existing fuel sipping system does not provide an on-line sipping capability and the system has never been used because performance was determined to be unacceptable at other plants using the same design.

The dry fuel sipping system interfaced with the Nitrogen and GWMS systems, however, this interface is not required with the wet fuel sipping system. Thus fuel sipping will no longer be a source of gaseous waste to the GWMS.

SAFETY EVALUATION

According to the safety evaluation two events are of interest related to this plant change; a) Fuel Handling Accident and, b) Radioactive Waste Gas System Leak or Failure. The refueling machine is non-safety, Seismic Category I and changes made by this plant change will not affect the function or operation of the refueling machine. The seismic qualification of the refueling machine will not be affected by the negligible weight increase associated with the plant change.

FSAR Section 9.1.4.3.2 states that no credit is taken for components or subsystems of the fuel handling equipment to either prevent or mitigate the consequences of the postulated accident.

There are no physical changes to the GWMS because of this change, the only effect will be to eliminate a source of gaseous waste to the GWMS. The wet sipping system will vent gas to the containment atmosphere and the system will draw water from and return it to the Refueling Pool. Gases that are vented from the system are normally introduced into the containment atmosphere during fuel movement. No new release paths will be created by this plant change. The change does not affect a protective boundary and will not reduce the margin of safety as defined in the bases for any of the Technical Specifications.

65. DC-3076, Reactor Coolant Pump (RCP) Vibration Monitoring

DESCRIPTION OF CHANGE

DC-3076 provides additional vibration and speed monitoring instrumentation for the Reactor Coolant Pumps and Motors.

REASON FOR CHANGE

The instrumentation being added provides additional monitoring provisions. The new information will allow trending which will provide more advance warning of potential component failure.

SAFETY EVALUATION

According to the safety evaluation the equipment being added is seismically designed to meet II over I criteria. The new system has no connections to existing systems. The design change does not alter any existing margins of safety or protective boundaries.

66. DC-3347, CEDMCS Hold Bus Power Supply Upgrade

DESCRIPTION OF CHANGE

The DC modifies the Control Element Drive Mechanism Control System (CEDMCS) by adding the capability of supplying momentary high voltage to the Control Element Assembly (CEA) Drive Mechanism upper gripper coils from the hold bus power supply.

REASON FOR CHANGE

The DC will enhance the reliability of the hold bus transfer operation (i.e., reduce the potential for dropping a control rod). The application of high voltage will ensure the upper gripper is engaged before the lower gripper is subsequently deenergized.

SAFETY EVALUATION

The safety evaluation states that the new high voltage timer circuits have no direct or indirect interaction with the CEDMCS logic circuits associated with CEA withdrawal, and therefore have no effect on the accident frequencies. The new circuit will be powered from the existing relay power supply and will also interface with the existing hold bus SCR gate circuits. There are no other points of electrical interaction. Loss of the relay power supply would not cause a CEA withdrawal accident. The hold bus is only capable of supplying power to the upper gripper coils, there is no hold bus failure that could result in CEA withdrawal. The modification has been specifically designed to reduce the frequency of occurrence of a CEA drop accident.

According to the evaluation the consequences of the CEA withdrawal and CEA drop accidents are not increased because the modification does not alter the limiting accident analysis assumptions. More specifically, the positive reactivity insertion rate has not been altered since the modification can not affect rod speed.

The modification is completely confined within the CEDMCS and has no effect on any other plant equipment. The modification does not alter the functional requirements of any equipment important to safety.

The modification does not affect the operation or accuracy of the CEA position indication system, or any of the CEA motion inhibit interlocks. The DC does not alter the CEA insertion limits or the ability to comply with those limits. It does not affect the CEA drop times or the ability to measure the rod drop times. Therefore, the DC does not reduce the margin of safety as defined in the bases for any technical specification.

B. CONDITION IDENTIFICATIONS/WORK AUTHORIZATIONS (CI/WA)

1. CI-260068, 260071, 260080, 260083, 260084, 260085, and 260095, Incorrect Installation of Penetration Seals

DESCRIPTION OF CHANGE

The listed CI/WAs accept the associated non-conforming penetration seals as "use-as-is," because the seals are capable of performing their intended function.

REASON FOR CHANGE

During a 100% penetration seal surveillance several penetration seals were identified as being inaccessible (inspection would involve greater detail than the scope of the 100% surveillance). These CI/WAs were generated to inspect the affected seals. The inspection determined that voids existed in the seals and they were declared inoperable and Fire Impairments were generated for the seals. (Penetration seals affected: VIA0051, VIA0052, VIA0054, VIA0060, VIA0064, VIA0086, and VIA0107.)

SAFETY EVALUATION

According to the safety evaluation accepting the seals as "use-as-is" will not affect any accidents listed in the FSAR. An evaluation of the seals conducted in accordance with UNT-007-053 determined that the seals will perform their intended function. The seals do not provide separation of safety related equipment and do not perform any safety related function.

2. <u>CI-268573/WA-01055319 and CI-289478/WA-01121188, Reroutes Auto Start Feature Pressure Sensing Lines for Fire Pumps</u>

DESCRIPTION OF CHANGE

Pressure sensing lines (used as part of the auto-start feature) will be rerouted for the Motor driven, Jockey, and Diesel driven fire pumps.

REASON FOR CHANGE

Current installation of the pressure sensing lines is not in accordance with NFPA-20.

SAFETY EVALUATION

According to the safety evaluation the consequences of a fire event as described in section 9.5.1 of the FSAR remain the same as previously described. The margin of nuclear safety is not affected by these CI/WAs. Fire Protection in non-safety, quality related, provisions of the approved fire protection program are maintained as is the plant's ability to achieve and maintain safe shutdown following a fire event.

3. CI-275562/WA-01138291, Damaged Radiation Seal Between the Concrete and Containment Liner Plate

DESCRIPTION OF CHANGE

This CI/WA addresses keeping the damaged radiation seal between the concrete and containment liner plate for the fuel transfer tube penetration "as-is."

REASON FOR CHANGE

This portion of the seal was damaged due to ILRT during Refuel 4. Design Engineering has determined that leaving the seal in it's current condition will provide an adequate release path through the gap during ILT depressurization and thus will prevent further degradation of the seal; also, the thermal flexibility of the containment vessel will not be impaired. Additional HP control will be established to ensure that no unnecessary exposure will occur with the seal in it's current configuration.

SAFETY EVALUATION

According to the safety evaluation the consequences of a Fuel Handling Accident will not be altered due to the damaged seal. Also, the damaged seal does not affect onsite doses in a way that restricts access to vital areas or impedes mitigating actions. Survey results of the damaged seal area confirmed that dose rates in the building changed slightly (40 mrem/hr as opposed to 25 mrem/hr stated in the FSAR) and could potentially cause inadvertent exposure to plant personnel during fuel transfer. To prevent personnel from getting unexpected exposure radiological posting/barricades will be used adjacent to the Fuel transfer tube. Also, additional HP control to monitor the condition of the seal and radiation levels during fuel movement are established. The evaluation concludes that the seal is part of the Fuel Transfer Tube Shield Structure and protects plant personnel from inadvertent exposure from fission product gamma radiation. The seal is not a fission product barrier, therefore, the damaged seal will not impact the safety margin defined in the Technical Specifications, 10CFR50, or 10CFR100.

4. CI-281257, Evaluation of Thermo-Lag on Seven (7) Fire Dampers and Ductwork

DESCRIPTION OF CHANGE

This is a "Use-As-Is" CI which addresses Thermo-Lag installations which have been evaluated using criteria stated in Generic Letter (GL) 86-10. The Thermo-Lag was determined acceptable "Use-As-Is."

REASON FOR CHANGE

The "Use-As-Is" CI approves the use of Thermo-Lag for 3 hour fire separation. The Thermo-Lag included in the CI provides a 3 hour barrier around dampers and associated ductwork that is located outside a 3 hour fire wall.

SAFETY EVALUATION

According to the safety evaluation there are no accidents that will be affected by this "Use-As-Is" CI. All the Thermo-Lag installations have been reviewed using criteria as stated in GL 86-10 and determined to be acceptable.

5. CI-287957/WA-01116256, Uprate Component Cooling Water (CCW) Supply and Discharge Headers to each Shutdown Cooling Heat Exchanger (SDCHX) to 150 PSIG

DESCRIPTION OF CHANGE

The CI/WA uprates the design pressure on the CCW supply and discharge headers to each SDCHX to 150 PSIG and resets relief valves CC-958A&B accordingly. The relief valves will be replaced with an equivalent relief valve as part of this activity.

REASON FOR CHANGE

Hydraulic transients created during system surveillance testing cause SDCHX CCW relief valves to lift because the system's normal operating pressure is close to the 125 PSIG design pressure. The lift pressure of relief valves CC-958A&B is set above the design pressure of the protected system.

SAFETY EVALUATION

The safety evaluation concluded that an unreviewed safety question does not exist. The CI/WA will restore the plant back into ASME code compliance by setting relief valves CC-958A&B to the design pressure of the system being protected. The intended functions of the CCW, SDC and Containment Spray (CS) systems will not be impacted by this activity. Uprating the CCW design pressure to 150 PSIG on the supply and discharge headers to each SDCHX will not decrease safety margin.

Uprate of the system is based on:

the uprated pressure at current design temperature does not exceed the design limits of the components in accordance with ASME Section III,

pipe minimum wall thickness' at the uprated design pressure and current design temperature do not exceed current pipe wall thickness' for the material listed in the ASME Section III Allowable Stress Tables,

all components are seismically qualified at the proposed design pressure, and

the SDCHX shellside (CCW) is designed and certified to 150 PSIG.

6. CI-287970/WA-01115110, Condenser Inlet Waterboxes Common Vent Header (Revision 0)

DESCRIPTION OF CHANGE

This Repair CI/WA is to document, evaluate, and accept the presently installed condenser inlet waterboxes common vent header piping located in the Circulating Water (CW) system. One new pipe support will be added to the line.

REASON FOR CHANGE

The vent header from inlet waterboxes to isolation valve CW-1161 has been installed in the plant since startup. This CI/WA will document the installation. Calculation EC-P-94001 was prepared to qualify the piping stresses and supports and resulted in the addition of one new pipe support.

SAFETY EVALUATION

The safety evaluation states that the common vent header is classified as a non-safety, non-radioactive system located outside the nuclear island in the Turbine Building. The vent header will not impair or affect any plant system or component in performing its intended function. There are no accidents affected by the vent header.

7. CI-288510/WA-01116768, Hot Machine Shop Power Hacksaw Deletion

DESCRIPTION OF CHANGE

Delete the unrepairable and unneeded power hacksaw located in the Reactor Auxiliary Building Hot Machine Shop. FSAR Figure 1.2-18 will be updated to reflect this deletion.

REASON FOR CHANGE

The power hacksaw has not been used, is not repairable and is not needed. Power hand tools are now used to perform metal shaping activities.

SAFETY EVALUATION

Deletion of the power hacksaw does not reduce the margin of safety, affect a protective boundary, or impact any acceptance limit. The deletion is confined to a tool in the Hot Machine Shop with no impact on the ability of any system or equipment to perform its intended design function.

8. CI-289619/WA-01121421, Remove Support #TB-T82 on Line #6CD2-243

DESCRIPTION OF CHANGE

This CI/WA removes support TB-T82 which is located on Condensate Line 6CD2-243. Removal of the support will result in the "As Built" condition as reflected in the "As Analyzed" condition as shown in Stress Analysis Calculation SA-1279, Revision 4.

REASON FOR CHANGE

During maintenance activities on a Condensate system valve the base plate anchor bolts for support TB-T82 were found to be sheared off. The support was located on a 2" drain line (6CD2-243) for Feed Water Pump "B" suction line (24" Condensate pipe). The support apparently broke because it acted as a thermal restraint as the 2" line experienced displacements due to the thermal growth of the 24" line.

SAFETY EVALUATION

The safety evaluation confirmed that no unreviewed safety questions are created by this CI/WA. According to the evaluation no SAR postulated accidents are affected by the CI/WA. Review of Stress Analysis Calculation SA-1279, Revision 4 indicates that functional and structural integrity of the 24" Condensate line along with its 2" drain line remain unaffected without the support on the 2" line.

9. CI-289966 (Non-conforming), Main Steam Safety Valve (MSSV) Seat Leakage

DESCRIPTION OF CHANGE

This CI revises the MSSV seat leakage requirements from no audible or visible leakage on air or steam at 90% (+/-1%) of set pressure to a maximum of 65 lbs/hr of steam seat leakage at 900 PSIG for each valve.

REASON FOR CHANGE

The MSSVs are ASME Section III Class 2, Seismic Category 1 valves that provide overpressure protection for the secondary side of the steam generator. The original specification required that the valves have no audible or visible leakage at 90% (+/-1%) of set pressure when tested on steam or air. Waterford 3 contract WF3-1485-0001 required seat leakage testing to be performed using steam.

After refurbishment, a spare set of safety valves failed to meet the no audible or visible seat leakage requirement on steam. Rework by the vendor was unable to achieve the leakage requirement. The vendor was able to quantify the leakage. The highest leakage rates found at 900 psig were 61 lbs/hr for the worst valve, others were generally lower than 40 lbs/hr.

This CI recommends that the MSSVs be "used as is" and that the specification be revised to limit the amount of steam leakage from each MSSV to 65 lbs/hr of steam at 900 psig.

SAFETY EVALUATION

The safety evaluation notes that the only accident analysis described in the FSAR that may be impacted by the seat leakage through the MSSVs is a Steam Generator Tube Rupture (SGTR) event. However, seat leakage through the MSSVs is independent of the occurrence of a STGR; therefore, this leakage will not influence the occurrence of SGTR or any other accident analyzed in the FSAR. In addition, seat leakage will not affect the set pressure or opening characteristics of the MSSVs. There are no physical changes made to the MSSVs by this CI.

The evaluation demonstrated that even with a leak rate as high as 4980 lb/hr the resulting radiological releases are still within the NRC acceptance limit. Therefore the consequences of a SGTR event with a 65 lb/hr steam leak rate per safety valve (390 lb/hr per generator) is less than the acceptable limits for this event.

The change does not reduce the margin of safety as defined in the bases for any technical specification or the appropriate safety analysis.

10. CI-290081/WA-01121813, Magnetic Actuator Change Out

DESCRIPTION OF CHANGE

This CI/WA replaces the installed magnetic actuator for the "Upender Vertical" indication on the fuel transfer system with a larger magnet.

REASON FOR CHANGE

DC-3088 (reported in the 1990 Report of Facility Changes, W390-1579, dated December 17, 1990, page 8) replaced the fuel transfer machine limit switches and cable assemblies. The switches were replaced by maintenance free proximity switches. This CI/WA will replace one of those proximity switches with a larger magnet. This will provide a greater magnetic flux for the proximity switch which will ensure the switch remains made up the Upender vertical.

SAFETY EVALUATION

According to the safety evaluation no credit is taken for components or subsystems of the fuel handling equipment to mitigate the consequences of the postulated fuel handling accident. This CI/WA will not change the operation or possible failure mode of the transfer machine.

The transfer machine is a non-safety, non-seismic system that does not contribute to the probability of a malfunction of equipment important to safety. The transfer machine does not contribute to any margin of safety as defined in the bases of the Technical Specifications.

11. CI-290154/WA-01122119, Simulated Incore Instrumentation (ICI) for Locations E09, L09, and T09 and CI-290405/WA-01122206, NCR Repair of Broken ICI Thimble L04

DESCRIPTION OF CHANGE

CI-290154 installs simulated ICI's at locations E09, L09, and T09 instead of actual incore detectors due to broken ICI sections blocking insertion of a new ICI into the ICI thimble. CI-290405 provides instructions for cutting the broken ICI thimble, at location L04, approximately 4.5 inches below the instrument plate and installs a dummy ICI in this location.

REASON FOR CHANGE

CI-290154 addresses the ICI thimbles that contain parts of old ICI detector assemblies in them, restricting the installation of new ICI detectors. CI-290405 addresses a bent ICI thimble at location L04, the thimble was bent while lowering the instrument plate. The use of simulated ICIs is required per vendor manual, Reactor Vessel Internal Instruction Manual (Waterford 3 Technical Manual 457000072).

SAFETY EVALUATION

The safety evaluations for the CI/WAs note that no accidents are found associated with the incore nuclear instrumentation system. The probability for reactor coolant leakage around the Hydrostatic Test Plug in the simulated ICI is not increased. The Hydrostatic Test Plug is of the same design as the seal plug of the ICI except that there are no rhodium detector or core exit thermocouple cables passing through it. Thus, the Reactor Coolant System pressure boundary is maintained.

12. CI-290155/WA-01122120, Replacement of E02 Incore Nuclear Instrumentation (INI) with R02 INI

DESCRIPTION OF CHANGE

Placement of INI designed for location R02 into location E02.

REASON FOR CHANGE

The incore instrument in location E02 was inadvertently removed while pulling the INI's to be replaced during Refueling Outage 6. The only difference between the two instruments is the difference in length, the replacement INI is 2 3/4" shorter.

SAFETY EVALUATION

According to the safety evaluation no accidents in the SAR are associated with the incore nuclear instrumentation system. The probability for Reactor Coolant (RCS) leakage around the seal plug portion of the INI designed for location R02 is the same as for the original INI, the seal plugs are of the same design. The integrity of the RCS pressure boundary is maintained.

There are no new system interactions created by using the INI designed for R02 as a replacement for the INI in location E02. The replacement is 2 3/4" shorter than the original and as such would provide a closer match to the vibration expected from the original INI.

The screening for this evaluation noted that adjustment of the rhodium detector signal will be required because of the offset of the replacement detector, this does not change the systems' operation.

13. CI-290368/WA-01122154, ACCW System Uprate to 125 psig Design Pressure (Revision 0)

DESCRIPTION OF CHANGE

This CI/WA uprates the design pressure of the Auxiliary Component Cooling Water (ACCW) System to 125 psig design pressure and resets relief valves ACC-121A&B and ACC-1012A&B accordingly.

REASON FOR CHANGE

Waterford 3 Corrective Action Document CI-289221 identified the ACCW system was operating above the current design pressure of 75 psig. The NCI Engineering Evaluation concluded there were no adverse impacts on the structural integrity of the ACCW system and the ACCW system was declared operable.

SAFETY EVALUATION

The CI/WA does not alter the operation or function of the ACCW system and can not cause or affect any accidents described in the SAR. The evaluation concludes that all components and structures are acceptable for this uprate in design pressure (125 psig) and at the current design temperature (125 degrees F.) according to ASME Section III. As noted in the evaluation, engineering calculations and design specifications supported this conclusion by determining that:

A) the uprated pressure at current design temperature does not exceed the design limits of the components in accordance with ASME Section III,

B) the required pipe minimum wall thickness at the uprated design pressure and current design temperature do not exceed the current pipe wall thickness for the material listed in ASME Section III Allowable Stress Tables,

C) all components are seismically qualified at the uprated design pressure, and

D) uprating the CCW Heat Exchanger shellside design pressure to 125 psig does not exceed the ASME Section III Code stress limits.

14. CI-290405/WA-01122206, NCR Repair of Broken ICI Thimble L04

DESCRIPTION OF CHANGE

This CI/WA provides for the repair of a broken Incore instrumentation (ICI) thimble at location L04 and the installation of a dummy ICI in this location.

REASON FOR CHANGE

The ICI thimble at location L04 was bent while lowering the instrument plate. This CI/WA provides for cutting the thimble a maximum of 4.5 inches below the instrument plate and installing a dummy ICI in this location.

SAFETY EVALUATION

According to the safety evaluation there are no accidents associated with the Incore Nuclear Instrumentation (INI) system and the integrity of the Reactor Coolant System is maintained because the Hydrostatic Test Plug of the dummy ICI is of the same design as the seal plug of the ICI. The only difference in the two plugs is that the Hydrostatic Test Plug has no rhodium detector or core exit thermocouple cables passing through it.

15. CI-290918/WA-01123090, Electrical Separation for the Class IE Power Supplies to CMU-407A and CMU-407B

DESCRIPTION OF CHANGE

This CI/WA will provide a fuse in series with the existing circuit breaker for the control circuits of Condensate Make-up (CMU) valves CMU-407A and CMU-407B.

REASON FOR CHANGE

Waterford 3 Corrective Action Document, CR-94-357, identified that the non-safety valves CMU-407A & B control circuits were powered from Class IE power supplies without double isolation from the Class IE power. Installation of the fuse in series with the existing circuit breaker will satisfy Regulatory Guide 1.75 guidance for electrical separation/isolation between non-class IE and class IE components.

SAFETY EVALUATION

According to the safety evaluation there are no unreviewed safety questions associated with this CI/WA. The evaluation states that the changes to the control circuits of CMU-407 A & B will enhance the availability of applicable class IE power supplies. It will provide electrical isolation between the non-safety related, non-seismic valves and the class IE power supplies. The electrical function of the circuit will not change. There is no impact to a protective boundary as a result of this CI/WA.

16. CI-291830/WA-01125865, Repair of Weld Crack on Main Steam Bypass Penetrations to Condenser

DESCRIPTION OF CHANGE

Main steam by-pass penetrations to Condensers "A," "B," and "C" from both the north and south sides will be stiffened. The thickness of the split ring on the sleeved nozzle on lines 6MS20-83, 84, 85; and 6MS20-75, 76, 77 will be increased from 3/8" to 1/2" Reinforcing pads will be added to the transition shell.

REASON FOR CHANGE

CI-291830 identified a crack in the weld between the split ring and the sleeve for 20" Main Steam bypass line 6MS20-85 going into Condenser "A" from the north side.

SAFETY EVALUATION

According to the safety evaluation no SAR postulated accidents are affected by the proposed modification of the Main Steam bypass penetrations to the Condenser. A review of stress analysis calculations SA-1030, Revision 4, IM-1030, Revision 1 and EC-P94-010, Revision 0 indicates that the functional and structural integrity of lines 6MS20-75, 76, 77, 83, 84, and 85 remain assured under design conditions. The safety evaluation notes that the CI does not affect any protective boundary. The Condenser is non-safety and non-seismic. Therefore, the modification does not reduce the margin of safety as defined in the bases for any technical specification or safety analysis.

17. CI-292623/WA-01127369, Resin Intrusion Into Condensate Make-up (CMU)

DESCRIPTION OF CHANGE

The CI/WA provides for sending a diver into the Condensate Storage Pool (CSP) while in Mode 1 to inspect for and cleanup any Blowdown Ion Exchanger resin that may have accumulated in the CSP.

REASON FOR CHANGE

Waterford 3 Corrective Action Document CR-94-774 identified that resin was introduced into the CMU header which supplies makeup water to the CSP. This CI/WA provides for a diver to enter the CSP to inspect for resin and remove it if found.

SAFETY EVALUATION

According to the safety evaluation no accidents are affected by this activity. Administrative controls will be in place to terminate the activity should the CMU system become unavailable for make-up. In the highly unlikely event that an accident occurs during this dive, systems are such that the diver will not be able to affect the margin to safety as demonstrated in the FSAR. The diver will be unable to block both EFW suctions in the CSP, and there is sufficient time available to retrieve the diver should CCW make-up be required during the diving operation.

18. CI-293078(WA-0112802) Instrument Air Line Cut and Cap

DESCRIPTION OF CHANGE

The CI/WA cuts and caps Instrument Air (IA) line 7IA1-100 upstream of a punctured elbow adjacent to 7IA2-51. All IA piping and valves downstream of the cap will be abandoned in place.

REASON FOR CHANGE

While preparing the yard south of the Maintenance Support Building (MSB)for application of asphalt, a 1 inch brass IA line (7IA1-100) was punctured by a concrete chipper. This line originally supplied air from header 7IA2-51 to FP-332. FP-332 was abandoned under DC-3386 (DC-3386 was reported in W3F2-94-0051, dated October 20, 1994, Report of Facility Changes, Tests and Experiments, Item #44).

SAFETY EVALUATION

According to the safety evaluation no unreviewed safety question exists. The IA System is not required to achieve safe shutdown or mitigate the consequences of an accident. All valves which are safety related fail to a safe position on loss of IA. Valves which are required to operate are supplied by accumulators. The repair does not affect a protective boundary.

19. CI-293269, 2SI2-80 A/B TEST CONNECTION ADDITION
(See also OP-009-008, Change A, Revision 11, Item II.A.20 of this report)

DESCRIPTION OF CHANGE

This CI installs a 1/2 inch test connection in Safety Injection (SI) line 2SI2-80 A/B, outside of the containment, between containment penetration 59 and containment isolation valve SI-344.

REASON FOR CHANGE

Installation of the test connection will facilitate local leak rate testing of valves SI-343 and SI-344. The test connection is on the Reactor Auxiliary Building (RAB) side of containment penetration 59.

SAFETY EVALUATION

According to the safety evaluation there is no unreviewed safety question associated with this CI. It notes that the SI system functions to mitigate a Loss of Coolant Accident or a Main Steam Line BreaK accident and that the portion of the SI system affected provides a recirculation path back to the Refuel Water Storage Pool (RWSP) while filling the SITs. The test connection maintains containment isolation by using a closed valve and a threaded cap.

20. CI-293401/WA-01129167, Low Volume Metal Waste and McCubbins Pond Transfer Pump Strainers

DESCRIPTION OF CHANGE

The CI/WA will fabricate and install stainless steel strainers on the suction piping for the Low Volume Metal Waste Pond (LVMWP) and the McCubbins Pond Transfer Pumps.

REASON FOR CHANGE

The McCubbins Pond Waste Transfer Pump is used to transfer liquid to the Waterford 3 LVMWP. The LVMWP Transfer Pump is used to transfer liquid from the Waterford 3 LVMWP to the Waterford 1 & 2 Low Volume Metal Waste Pond. The current suction piping configuration for these pumps does not prevent foreign material from entering the pumps.

SAFETY EVALUATION

The safety evaluation notes that the strainers will be added to non-safety, non-quality, Class 7 lines. There are no accidents previously evaluated that are affected by addition of the strainers. The pumps are not required for safe shutd wn or for mitigation of accidents. According to the safety evaluation there are no unreviewed safety questions associated with this CI/WA and no margins of safety are affected.

21. CI-294541, Addition of COLSS Steam Calorimetric

DESCRIPTION OF CHANGE

The Core Operating Limit Supervisory System (COLSS) secondary calorimetric will have an additional method based on a measured steam mass flow rate as well as the original method based on feedwater mass flow rate.

REASON FOR CHANGE

The COLSS steam calorimetric will use steam mass flow rate corrected for pressure in conjunction with the blowdown mass flow rate to determine a feedwater mass flowrate to perform an energy balance on the steam generator. The steam calorimetric power calculation will also help address the generic industry problem with feedwater venturi fouling as it is independent of the feedwater venturi.

SAFETY EVALUATION

According to the safety evaluation there are no accidents associated with the COLSS. The addition of another core power indication in the COLSS will not increase the consequences of an accident previously analyzed in the SAR. COLSS is not required for plant safety since it does not initiate any direct safety-related functions during anticipated operational occurrences or postulated accidents. This change does not create any new inter-relationships between COLSS and equipment important to safety.

22. CI-295102, Control Room Loose Items

DESCRIPTION OF CHANGE

Personnel lockers, storage cabinets, file cabinets, and book cases in the Control Room that are in the vicinity of safety related equipment will be restrained with the use of Hilti bolts.

REASON FOR CHANGE

A Waterford 3 corrective action document was written to document and evaluate the loose items stored in the vicinity of safety related cabinets in the Control Room. The evaluation determined that no operability concerns existed; however, the recommended corrective action was to restrain the items to prevent interactions during a seismic event. FSAR Figure 1.2-8 will be updated to reflect results of this CI.

SAFETY EVALUATION

According to the safety evaluation the lockers/cabinets will be restrained to prevent them from interacting with any control panels; therefore , this decreases the probability of occurrence of any accident previously evaluated in the SAR. Calculation EC-C95-005, performed to evaluate the floor loading from the items and the anchor bolts used to restrain them, show that the items will remain in place during a seismic event. Thus, the modification does not reduce the margin of safety as defined in the bases for any technical specification or the appropriate safety analysis.

23. CI-295858/WA-01134391, Supplementary Chilled Water Cooling Towers Potable Water Make-up Backflow Preventer Addition (Revision 0)

DESCRIPTION OF CHANGE

Adds a code specified backflow preventer to the Potable Water System at the make-up for the Supplemental Chilled Water Cooling Towers.

REASON FOR CHANGE

The backflow preventer is added to prohibit cross contamination of the Potable Water system, which provides the plant with water suitable for personnel consumption, from the Fire Protection or Circulating Water Systems. Potable Water check valve PW-503 will be replaced with the code specified backflow preventer.

SAFETY EVALUATION

The safety evaluation states that the backflow preventer is added to the Potable Water system, a non-safety, non-radioactive system located outside the nuclear island in the Chiller Building. The backflow preventer will have no affect on any plant systems or components therefore no accidents can be postulated or affected by this change. No new methods of failure are introduced and no new system interactions are created by installation of the backflow preventer.

24. <u>CI-297943/WA-01138326, EDG "A" Lube Oil Filter Drain Valve Replacement/Addition</u>
<u>CI-297944/WA-01138327, EDG "B" Lube Oil Filter Drain Valve Replacement/Addition</u>

<u>DESCRIPTION OF CHANGE</u>

The CIs will replace the clean oil drain valve and associated pipe fittings with ASME Class 3 components, install a new dirty oil drain valve and associated pipe fittings and update the system drawing to accurately reflect continuous flow vents from the lube oil filter and the circulating and pre-lube pump to the engine crank case.

<u>REASON FOR CHANGE</u>

Current clean oil drain valve on the lube oil filter for the Emergency Diesel Generators does not meet ASME Section III, Class 3 requirements, and the diesel vendor has recommended that a valve also be provided at the dirty oil drain connection to aid in the draining process during filter cartridge replacement.

<u>SAFETY EVALUATION</u>

According to the safety evaluation the original filter design has two drain connections furnished by the vendor with threaded pipe plugs. ASME Code qualified valves with pipe nipples and caps will be added to these connections in place of the plugs to aid in the draining process. The integrity of the filter housing as a pressure vessel will therefore be maintained and will not affect the lube oil system or any other plant system.

The drain valve addition/replacement on the lube oil filter housing and the lube oil schematic drawing revision to as-build a vendor furnished vent line does not reduce the margin of safety, affect a protective boundary, or impact any acceptance limit. The changes are confined to the EDG lube oil system with no impact on the systems' ability to perform its intended design functions.

25. CI-298804/WA-1140681, CI-298805/WA-1140682, Removal of Sudden Pressure Relay Trip for The Main Transformers (Repair)

DESCRIPTION OF CHANGE

This CI/WA Repair removes the Sudden Pressure (SP) relay trip signal to the 86G2 Generator Lockout relays. The Main Transformer SP relay will provide local annunciation only,

REASON FOR CHANGE

See Item I.C.8 of this report, TAR-95-008. This CI/WA will allow the removal of TAR-95-008.

SAFETY EVALUATION

The safety evaluation did not identify any unreviewed safety questions associated with the CI/WA. The evaluation identifies the Loss of Offsite Power (LOOP) as an accident that may be affected, however, it notes that the CI/WA does not increase the probability or the consequences of a LOOP. The evaluation states that the function of the Main Transformers will not be affected, they are not safety-related and no nuclear accidents are related to the transformers. In the event of an internal transformer fault, two independent zones of differential relaying protection are available for fault detection. The sudden pressure relays provided a third zone of protection.

26. CI-298934/WA-011410447, RCP Lube Oil Cooler Drain Valve

DESCRIPTION OF CHANGE

The CI/WA replaces the existing 1/2" drain plug of the Reactor Coolant Pump (RCP) Upper Oil Cooler with 3/4" tubing and a ball valve. The oil collection system enclosures will be modified to permit access to the new drain valve.

REASON FOR CHANGE

The oil in the RCP motors must be removed each refueling outage and the RCP Oil Collection system enclosures must be partially removed to provide access to a drain plug on the bottom of each RCP lube oil cooler. In addition, oil is inadvertently spilled into the RCP oil collection system when the drain plug is removed and the drain hose is connected.

SAFETY EVALUATION

According to the safety evaluation the RCP Upper Oil Cooler is classified as non-safety, non-seismic equipment and is not required to achieve safe shut down or to mitigate the consequences of an accident. The CI/WA will not reduce the margin of safety as defined in the basis of any Technical Specification or safety analysis and no unreviewed safety questions are created. The CI/WA maintains the Oil Collecting system function and integrity.

27. <u>CI-299424/WA-01141458, RCP, Repair Package for the Cutout Opening of the Oil Drip Pan Enclosures</u>

DESCRIPTION OF CHANGE

The CI/WA package repairs the existing cutout opening of the Reactor Coolant Pump (RCP) oil collecting drip pan enclosures.

REASON FOR CHANGE

Existing drip pan enclosures will be enhanced by repairing the cutout opening with a stainless steel, 16 gage sheet metal cover. The installation of the sheet metal cover plates will assure that the integrity of the oil collection system is maintained and that a oil leak will not lead to a fire during normal or design basis accident.

SAFETY EVALUATION

The RCP oil collecting drip pan is classified as non-safety related, seismically supported equipment and is not required to achieve safe shutdown or to mitigate the consequences of an accident. The evaluation did not identify any unreviewed safety question associated with the CI/WA. The additional weight added by the cover plates will have negligible impact on the seismic considerations for the RCP. The CI/WA maintains the oil collecting system function as originally set forth, it enhances the existing oil drip pan by reducing the leakage of oil, minimizes spillage and potential risk.

28. CI-299494/WA-01141620, Addition of Tubing Support Clamps on Main Steam (MS) Instrument Sensing Lines

DESCRIPTION OF CHANGE

The CI/WA installs four (4) additional 2 directional tubing support clamps, one on each MS instrument sensing line, to reduce or minimize line vibration while still maintaining sufficient flexibility for thermal, seismic and other dynamic movements.

REASON FOR CHANGE

Installation of the tubing supports will enhance the existing tube routing supporting system by adding stability to the system. Installation will assure that the integrity of the sensing line system is maintained in normal, upset or faulted load conditions.

SAFETY EVALUATION

The safety evaluation confirmed that the CI/WA will not reduce the margin of safety as defined in the basis of any Technical Specification or safety analysis and no unreviewed safety questions are created. Integrity of the sensing lines will be maintained by the use of standard 2 directional tube clamps and meeting tubing stress in accordance with ASME Section III code criteria. The sensing lines are classified as safety and seismic category 1 tubing, and are not required during safe shutdown of the plant following an accident or to mitigate the consequences of an accident.

29. CI-300062/WA-01142290, Drain Valves Connected to the Charging Pump Crankcase and Speed Reducer

DESCRIPTION OF CHANGE

The CI/WA documents the as-built condition of the drain piping and the associated valve of the Speed Reducer and Pump Crankcase for Charging Pump "B."

REASON FOR CHANGE

The "Use-As-Is" CI/WA documents the use of the piping and valve arrangement as installed. The "as-built" condition does not conform with the Technical Manual (457000093) which required the CI/WA to be developed. No physical changes are caused by the CI/WA.

SAFETY EVALUATION

According to the safety evaluation the piping and valves have been analyzed in accordance with ASME Section III, 1971 edition and qualified to safety related and seismic category 1 with no designated code class. The piping and valves have been commercially dedicated by CGE 00415. The engineering evaluation, field walkdown, and Stress Analysis and Support Calculation, EC-P95-007, indicates that the existing arrangement will have minimum impact due to thermal, dead weight, and seismic cases. The charging pump oil lubricating and draining system function remains the same as it was originally intended.

30. CI-300196/WA-01142585, Liquid Waste Management - Chemical Addition Funnel Rigs

DESCRIPTION OF CHANGE

This "Use-As-Is" CI/WA documents the chemical addition funnel rigs installed on valves LWM-312A and LWM-311B in the Liquid Waste Management System (LWMS).

REASON FOR CHANGE

The funnels are currently installed and are used in accordance with procedure ‹ -002-018 to facilitate pH control of the Waste Condensate Tanks. The CI/WA will only add the funnels to plant drawings to reflect as-built conditions.

SAFETY EVALUATION

According to the safety evaluation the FSAR discusses the analysis for a LWMS Leak or Failure. The analysis assumes a complete failure of all non-safety and non-seismic equipment to occur as a res of a safe shutdown earthquake. The funnels are installed in the non-safety, non-seismic portion of the LWMS and are normally isolated by valves LWM-312A and LWM-311B. The funnels do not increase the probability that these valves will fail during a safe shutdown earthquake. There are no technical specifications or protective boundaries that are affected by the CI/WA.

31. CI-300596/WA-01143238, TCCW Pumps A&B Suc. ; 'Discharge Piping Vent Additions (Repair)

DESCRIPTION OF CHANGE

This CI/WA Repair package adds vents to the suction and discharge lines of Turbine Closed Cooling Water (TCCW) systems pumps A & B.

REASON FOR CHANGE

The present piping configuration for the TCCW Pumps A & B does not allow adequate post maintenance venting of the suction and discharge lines. The suction and discharge valves are located in vertical piping at an elevation above existing vents.

SAFETY EVALUATION

According to the safety evaluation the addition of vents will not alter the function of the non-safety TCCW system. No unreviewed safety questions or changes to the approved design basis exist as a result of the CI/WA. The TCCW system serves no safety function during plant shutdown, operation or accident scenarios. The protective boundary or margin of safety described in any technical specification or safety analysis is unaffected.

32. CI-301157/WA-01144162, Dry Cooling Tower Tube Sheet Drain Repair

DESCRIPTION OF CHANGE

This WA repair replaces the existing Dry Cooling Tower (DCT) tube bundle drain line, which is threaded into the bottom of the tube sheet and includes a manual valve, pipe cap, fittings, and heat tracing, with a threaded plug. Identical tube plugs are currently installed on the bottom of all of the tubes except for the one affected tube on each DCT bundle (20 bundles) which currently contains the drain line.

REASON FOR CHANGE

Existing DCT tube bundle drain line configuration does not agree with design drawings. The affected drain lines are ASME Section III, Class 3.

SAFETY EVALUATION

Replacing the drain line with a tube plug will not increase the probability of the occurrence of an accident previously evaluated in the FSAR. The new tube plug will be identical in size and material to the existing, vendor supplied tube plugs which are currently installed on all of the adjacent tubes. The new plugs will be Quality Class (QC) 1 and ASME SA-105 material. The overall weight of the DCT will decrease as a result of this change. Therefore, the existing seismic qualification of the DCT which is contained in SQRT File SQ-MN-273 bounds the new configuration. Therefore, the likelihood of an accident occurring will not be increased as a result of this change.

33. WA-01138830, Refuel 7 Fuel Reconstitution

DESCRIPTION OF CHANGE

This WA provides for the repair of leaking fuel assemblies by removing the leaking pin and inserting a stainless steel dummy rod.

REASON FOR CHANGE

To perform fuel reconstitution for assemblies identified as leaking by UT and Sipping before returning to the core for Cycle 8.

SAFETY EVALUATION

The safety evaluation notes that fuel reconstitution requires removal of the upper end fitting from the fuel assembly. However this temporary configuration change of the assembly will not affect the likelihood of the fuel handling accident occurring because the fuel assembly will not be moved from its location without the upper end fitting in place. Other mitigating factors are that no fuel movement is planned for the time period when reconstitution will be performed and during reconstitution rod pulling templates will cover the assembly under reconstitution most of the time.

The FSAR analysis for the Fuel Handling Accident assumes 236 broken pins, all pins in the assembly, the dropping of a single pin during reconstitution is well below the 236 broken pins limit.

The evaluation states that the fuel bundles are passive components in the reactor core. The small change in power distribution due to the stainless steel replacement rods will be accounted for in the CECOR coefficient library and will have no effect on LPD or DNBR margins.

34. WA-01141193, High Speed Loaded Grapple Over the Upender

DESCRIPTION OF CHANGE

The WA addresses the use of high speed hoisting of a fuel assembly by the Refueling Machine (RM) while inserting or removing fuel from the Upender fuel transfer can.

REASON FOR CHANGE

The WA allows the control system to allow high hoisting speed to be used when over the Upender cans with the RM, this will allow refueling to occur quicker.

SAFETY EVALUATION

The safety evaluation did not identify any unreviewed safety questions associated with the WA. Increasing of the hoist speed was reviewed against the design of the Fuel Handling System and the Fuel Handling Accident. No information in the FSAR is affected by the WA and no credit is taken for any components or subsystems of the fuel handling equipment to either prevent or mitigate the consequences of an accident. The capability of the machine to lift a load is unchanged and the controls for hoist load interlocks is unchanged by the WA. The "high hoist speed" will be limited to 18 fpm (0.3 fps) while the speed analyzed for the fuel handling accident is approximately 30 fps.

C. TEMPORARY ALTERATION REQUEST (TAR)

1. TAR-92-015, Feedwater Regulating Valve

DESCRIPTION OF CHANGE

This TAR provides for sealing the discharge vent for the "B" Feedwater Regulating Valve (FRV) solenoid valve. This will allow for normal operation of the "B" FRV.

REASON FOR CHANGE

The FRVs are equipped with a pressure switch which detects loss of Instrument Air (IA) and sends a signal to a solenoid valve on the operator which fails the control valve as is. The solenoid valve has developed a leak which requires sealing to allow normal operation of the "B" FRV.

SAFETY EVALUATION

According to the safety evaluation the FRV will operate normally in all conditions except loss of IA. On a loss of IA the valve would lose function regardless of this TAR. The FRV is non-safety and only provides a backup feature for the Feedwater Isolation Valve (FWIV), this function will not be affected by this TAR. Receipt of an ESFAS signal will continue to close the valve.

2. TAR-94-008, Containment Sump Pump "A" Bypass

DESCRIPTION OF CHANGE

The TAR will connect a temporary sump pump and a check valve with a flexible 2" hose to valve SP-103A. The temporary sump will be installed under the Weir Box to catch all incoming water into the Containment Sump. The sump water will take the normal path out of the Containment, routed to the Radioactive Waste Tanks in the Reactor Auxiliary Building through radiation monitor PRM-IREE-6777, located outside containment.

REASON FOR CHANGE

The TAR will allow for cleaning of the Containment Sump and calibration of the sump level instrument. This activity will be performed during mode 6 of Refueling Outage 6.

SAFETY EVALUATION

The safety evaluation states that the TAR can only be performed in modes 5 and 6 (when the Containment Sump System is not required) and has no effect on accidents listed in the SAR. The Reactor Coolant System Leakage Detection System is the only system affected by this TAR and it is only required in Modes 1 through 4 and is not a safety related system. All sump influent will be routed and processed via a normal radwaste system operation (other than the temporary pump and flexible hose connected to SP-103A). Containment isolation could still be achieved because valves SP-105 and SP-106 will close on a Containment Isolation Actuation Signal.

3. TAR-94-011, Temporary Power for Containment Atmospheric Purge (CAP) Valves CAP-102, CAP-203, and CAP-204

DESCRIPTION OF CHANGE

The TAR provides a source of 120VAC non-safety power to CAP-102, CAP-203, and CAP-204 to allow for operation of the CAP system. CAP-102 serves as a Containment Isolation valve and allows make-up air to enter Containment, CAP-203 and 204 also serve as Containment Isolation valves and are utilized for Purging Containment.

REASON FOR CHANGE

Normal power to the subject valves will not be available during the "B" safety bus outage and this TAR will allow the valves to function as designed. "Containment-to-Ambient dp" (CAP-IPAC-5258(B)) will not be available during this bus outage, CAP-IPAC-5258(A) will be available. The TAR includes instructions for Operations to secure Containment Purge should containment pressure fall to -10 in. wg.

SAFETY EVALUATION

According to the safety evaluation the TAR will not affect any accident evaluated in the FSAR. Safety features will continue to function as required. The radiation monitoring system and the Containment Purge Isolation Signal (CPIS) will continue to operate as designed. The evaluation states that the non-safety power supplied to the valves in lieu of the normally supplied safety-related power supply will have no adverse impact on the safety-related function of the CAP system. The system will perform its purge function during refueling activities during the "B" safety bus outage.

4. TAR-94-019, Modification of the Drain Line for the Heat Exchangers for the Condenser Wide Range Gas Monitor (WRGM)

DESCRIPTION OF CHANGE

Associated with the Condenser WRGM is a Moisture Control Unit (MCU) which removes moisture from the sample stream prior to the stream entering the detection skid. The MCU consists of two major subassemblies, a chiller and dryer. The chiller has two Basic heat exchangers in parallel. The TAR will replace the heat exchangers drain line trap and check valve with a loop seal.

REASON FOR CHANGE

The trap in the heat exchangers drain line has become clogged with debris from the shell side of the heat exchangers. Installation of the loop seal should prevent clogging of the drain line. This TAR will be replaced by a plant change (see Item I.A.54 of this report, DC-3455).

SAFETY EVALUATION

According to the safety evaluation the TAR will not affect the function or operation of the Condenser WRGM. Modification of the drain line will improve operation of the WRGM. The Condenser WRGM is one of three types of radiation monitors designed to detect a Steam Generator Tube Rupture. If the Condenser WRGM were to fail the Steam Generator Blowdown and the Main Steam Line monitors are also designed for this event.

5. TAR-95-005, Shell Drain Tank Normal Level Control Valves

DESCRIPTION OF CHANGE

This TAR will fail-open the Moisture Separator Reheater (MSR) Shell Drain Tank (SDT) Normal Level Control Valves (NLVCs) by isolating the Instrument Air to the valves positioners and then gagging the valves open. The SDT Low Level alarm inputs will also be disabled by the TAR.

REASON FOR CHANGE

The purpose of the SDT Normal Level Control Valves was to maintain a desired level in the associated MSR-SDT. During system design this was considered necessary to prevent the direct communication of steam between the SDT and the #2 Feedwater Heater. System operation has proven that this consideration is not necessary

SAFETY EVALUATION

The safety evaluation determined that there is not an unreviewed safety question associated with this TAR. The TAR will have no impact on safety-related equipment. The SDT NLCVs are placed in a more conservative line-up (failed open) and will, therefore, lower the probability of a turbine trip due to high SDT levels. Disabling the SDT Low Level Alarms only impacts a system parameter indication, this is an operator aid and provides no automatic protective functions.

6. TAR-95-006, Separation of the 3A2 Unit Auxiliary Transformer (UAT) and the Startup Transformer (SUT) 4160 Bus

DESCRIPTION OF CHANGE

TAR-95-006 will electrically isolate the Unit Auxiliary Transformer (UAT) "A" 4.16kV winding from the 4.16kV non-safety 3A2 bus. This will allow the 3A2 bus to be re-energized from the Startup/Standby Transformer (SUT) "A."

REASON FOR CHANGE

The UAT 4.16kV cable bus duct was damaged by a fire event that occurred on June 10, 1995. This TAR will allow SUT "A" to supply the plant auxiliaries via 3A1 and 3A2 busses with Waterford 3 supplying power to the grid. This is contrary to the "normal" lineup in which UAT "A" supplies the plant auxiliaries with Waterford 3 supplying power to the grid.

SAFETY EVALUATION

According to the safety evaluation the TAR will not reduce the margin of safety as defined in the basis of any Technical Specification or safety analysis and no unreviewed safety questions are created. The evaluation discusses Loss of Condenser Vacuum, Loss of Normal A/C Power, Partial Loss of Forced RCS Flow, Total Loss of RCS Flow and Loss of Normal Feedwater Flow as being potentially affected by the TAR. Prior to the TAR an off-site event which removes all offsite power to the plant might not cause the events since Turbine Runback and Steam Bypass may allow the Main Generator to continue powering the non-safety busses. However, no credit is taken for Turbine Runback for any accident analysis. After installation of the TAR the non-safety busses will be powered from the off-site source. The overall effect is that there is no discernible increase in the probability of occurrence of the accidents evaluated in the SAR.

The 3A2 and 3A1 busses are non-safety and non-seismic. The result of these busses failing is no different prior to or post modification. Either prior to or post modification a loss of either or both these busses will result in the same type of failures and resulting consequences. Therefore, there is no change to a protective boundary.

7. TAR-95-007, Additional Cooling for the Regenerative Heat Exchanger Room

DESCRIPTION OF CHANGE

The TAR installs flexible duct from the non-safety HVAC cooling air register located inside the Regenerative Heat Exchanger Room to provide additional cooling for the solenoid valves that serve CVC-101 and CVC-103. Sheet metal blanks will be installed over two HVAC registers outside the Regenerative Heat Exchanger Room. (See Item I.A.61, of this report, for related information.)

REASON FOR CHANGE

Additional cooling for the solenoid valves is required to assure that the ambient air temperature around the valves remains below the EQ qualification temperature.

SAFETY EVALUATION

According to the safety evaluation the modification affects a non-safety branch of the Containment Cooling System. This branch is not required for accident mitigation and does not affect any accidents previously evaluated in the SAR. The evaluation notes that the temporary duct and associated metal plates are considered seismically supported. The temporary duct is not qualified for extreme environmental conditions which are encountered during LOCA and/or MSLB. However, due to the location within the Regenerative Heat Exchanger Room debris created from the duct by these conditions will be confined to the Heat Exchanger Room. Therefore, there is no concern about debris being transported to the Safety Injection Recirculation Sump and potentially clogging the screens.

8. <u>TAR-95-008, Removal of Sudden Pressure Relays for Main Transformers</u>

DESCRIPTION OF CHANGE

The TAR will disable the sudden pressure relay from the 86G2 Generator Lockout relays. Annunciation will be provided at the transformers' control cabinet to monitor sudden pressure relay actuation. (See Item I.B.25, CI-298804/WA-1140681 and CI-298805/WA-1140682, of this report.)

REASON FOR CHANGE

TAR-91-034, "Removal of Sudden Pressure Relay Inputs to Generator Trip," Item 31 of the Waterford 3 10CFR50.59 Annual Report for 1992, W3F2-92-0033, dated December 10, 1992, also performed the same modification as this TAR. At that time the plant used oil monitoring sudden pressure relaying operations. The sudden pressure relaying was replaced with gas pressure monitoring equipment during Refuel 5. Until the June 10, 1995 fault which eventually resulted in the loss of the 3A2 4.16kV bus, the plant had not experienced any spurious trips of the sudden pressure relays. The sudden pressure relay on Main Transformer "A" has been cited to have incorrectly actuated during the June 10, 1995 event.

SAFETY EVALUATION

The safety evaluation did not identify any unreviewed safety questions associated with the TAR. The evaluation states that the function of the Main Transformers will not be affected, they are not safety-related and no nuclear accidents are related to the transformers. In the event of an internal transformer fault, two independent zones of differential relaying protection are available for fault detection. The sudden pressure relays provided a third zone of protection.

9. <u>TAR-95-009, Installation of Blind Flange on Line 7AE20-21 During Repair AE-117</u>

DESCRIPTION OF CHANGE

The TAR adds a blind flange on line 7AE20-12 to allow operation of the Air Evacuation (AE) system while AE-117 is removed for repair.

REASON FOR CHANGE

The AE System removes non-condensable gasses from the condenser shell during plant operation. Exhaust from the system is monitored by the Condenser Wide Range Gaseous Monitor (WRGM) and discharged through motor operated valve AE-117. AE-117 failed to fully close when the WRGM was declared out of service and inspection revealed that the valve seat is deteriorated. Repair requires removal of the valve from the line, continued operation of the AE system requires the installation of the blind flange to prevent an unisolable path in the event of a primary to secondary leak.

SAFETY EVALUATION

Installation of a blind flange in line 7AE20-12 will not cause or affect the probability of occurrence of a previously evaluated accident. During the temporary alteration, the AE exhaust will be manually diverted to the Reactor Auxiliary Building (RAB) Normal Exhaust System. This is the required path in the event of a primary to secondary leak. Installation of the flange does not affect a protective boundary and no margins of safety are affected.

10. TAR-95-010, Reconfiguration of the Primary Access Point (PAP)

DESCRIPTION OF CHANGE

This TAR revises the layout of the PAP and adds a new x-ray machine to the PAP.

REASON FOR CHANGE

Traffic through the PAP is very congested during shift changes during outages. This TAR is expected to expedite personnel and package monitoring during these high traffic periods.

SAFETY EVALUATION

The safety evaluation confirmed that the revised layout of the PAP will not reduce the margin of safety as defined in the basis of any Technical Specification or safety analysis and no unreviewed safety questions are created. There are no postulated accidents that are affected by the TAR. The Security System is non-safety related, existing equipment will continue to be used so the equipment operation and function will not be affected. The addition of the new X-ray equipment was evaluated and determined to be within acceptable limits (calculation EC-E91-090).

11. TAR-95-012, Hose Connection for Removing Water from RWSP Directly to a HUT

DESCRIPTION OF CHANGE

This TAR adds a hose between the Fuel Pool System and the Boron Management (BM) System to permit removing water from the Refueling Water Storage Pool (RWSP) directly to a Hold-up Tank (HUT).

REASON FOR CHANGE

Removing water from the RWSP will create room to add boric acid from the BAM tanks to raise the RWSP boron concentration for refueling. The TAR will utilize that part of the Fuel Pool System normally used to purify, recirculate and sample the RWSP.

SAFETY EVALUATION

According to the safety evaluation the TAR creates another connection between the BM system and the Fuel Pool System. The only possible new incident would be reducing the RWSP water volume below the Technical Specification limit. However, operators will be monitoring RWSP and HUT levels during this evolution. Failure of a HUT filled with water from a Reactor Coolant System having 1% failed fuel has been calculated not to cause any off-site limits to be exceeded. The water being drained has far less radioactivity than in the calculated case. Therefore no off-site dose limitations are exceeded. This TAR is for a very short duration evolution monitored by operators, reaction time to any equipment failure will be short. All water from a failure will be collected in floor drains and directed into the BM system or Liquid Waste Management system. Since the only equipment affected by Technical Specifications or safety analysis is the RWSP and this TAR does not affect the safety posture of the RWSP, no margins of safety are changed by the TAR.

12. <u>TAR-95-014, Containment Sump Pump "A" Bypass</u>

See item I.C.8, TAR-94-008, of this report.

13. TAR-95-017, Containment Atmosphere Purge Valves CAP-103, 104, and 205

DESCRIPTION OF CHANGE

TAR-95-017 provides temporary power to Containment Atmosphere Purge Valves (CAP) CAP-103, 104, and 205 and Plant Stack Radiation Monitor PRMIRE0100.1 from auxiliary wall receptacles.

REASON FOR CHANGE

The TAR provides a temporary source of power to allow operation of the valves and radiation monitor during a maintenance outage of the primary source of power (safety bus "A").

SAFETY EVALUATION

The safety related functions of the CAP valves to isolate containment and the plant stack radiation monitor to generate a Containment Purge Isolation Signal (CPIS) are not affected by the TAR. It provides an alternate source of non-safety electrical power which will not alter the function of the equipment. If the alternate source of power is lost the CAP valves fail-safe which is closed and the radiation monitor fails in the tripped position which would generate a CPIS and isolate CAP.

14. TAR-95-019, Temporary Power to Fuel Pool Pump "A"

DESCRIPTION OF CHANGE

The TAR provides temporary power for non-safety Fuel Pool Cooling (FPC) Pump "A" from motor control center (MCC) 3B314-S.

REASON FOR CHANGE

The TAR will be used when the normal supply for the pump (MCC-3A314-S) is de-energized for a scheduled bus outage and the plant is shutdown.

SAFETY EVALUATION

According to the safety evaluation the TAR is required to allow continuous full capacity cooling of the fuel pool during the scheduled maintenance outage of Bus "A." A spare associated circuit of MCC 3B314-S will be connected to the line side of the normal supply breaker for FPCP "A." The normal supply breaker will be racked out and isolated from de-energized MCC 3A314-S. In this configuration the FPCP is another non-safety load connected to bus B, as is FPCP "B." Both pumps operate in parallel as part of the same Fuel Pool Cooling system. The motor feeder cable is run in its own non-safety conduit. The motor control circuits are run in non-safety raceways. The TAR will not increase the probability of a Loss of Off-site Power because an upstream associated breaker provides electrical isolation of 1E and non-1E loads in the event of a down stream fault. The TAR will not impact any protective boundary or margin of safety.

15. TAR-95-020, CC-958A & CC-958B

DESCRIPTION OF CHANGE

Shutdown Heat Exchanger Outlet thermal relief valves (CC-598A & CC-598B) will be gagged closed to prevent them from lifting during testing of the Component Cooling Water (CCW) Train A (CC-598A) and CCW Train B (CC-598B).

REASON FOR CHANGE

The valves are provided as thermal relief valves in the event the heat exchangers are isolated. During testing of the CCW isolation valves these two valves have been noted to lift and fail to reseat without operator action.

SAFETY EVALUATION

According to the safety evaluation the CCW system and the Shutdown Heat Exchangers are required during and following a Loss of Coolant Accident (LOCA) and a Main Steam Line Break (MSLB) inside the containment. Gagging of the valves will not affect the overall system performance or reliability in a way which could lead to an accident occurring. The valves are required only as thermal reliefs in the event the shellside of the Shutdown Heat Exchangers are isolated. During operation, the Shutdown Heat Exchangers are not isolated. The isolation valves are normally locked open and caution tags will be placed on the valves to warn against closing them with CC-598A & B gagged closed.

16. TAR-95-023, Removal of CEA #34 RSPT #1 Inputs to CPC "A" and CEAC #1

DESCRIPTION OF CHANGE

The TAR affects Control Element Assembly (CEA) #34 Reed Switch Position Transmitter (RSPT) #1 signal which is used as an input for Core Protection Calculator (CPC) "A" and Control Element Assembly Calculator (CEAC) #1. The TAR changes the RSPT signal from one which intermittently drifts low to a constant full out signal. This affects only the indication of CEA #34 position provided from RSPT #1 to CPC "A" and CEAC #1.

REASON FOR CHANGE

CEA #34 RSPT #1 was sending intermittent signals to CPC "A" and CEAC #1 resulting in nuisance alarms. The TAR will end the nuisance alarms while corrective action is determined. The TAR will result in a constant fully withdrawn CEA #34 indication to CPC "A" and CEAC #1.

SAFETY EVALUATION

CEACs are designed to provide protection against Anticipated Operations Occurrences (AOOs) which involve the insertion or withdrawal of a single full length or part length CEA. The TAR does not degrade the ability of the CPC/CEAC to provide this protection. Waterford 3 operates in an All Rods Out (ARO) condition and while the TAR is installed , Operations will administratively require that ASI control be done utilizing Regulating Group 6, thus making the possibility of an outward deviation of CEA #34 essentially zero. Installation of the TAR provides a hardware mimic of the actual position of CEA #34 as long as it is maintained at the full out position. The constant input signal to CPC "A" and CEAC #1 is within the normal range of input signals and does not increase the probability that the CPC or CEAC will malfunction. No other RSPT signals are affected including redundant indication of CEA #34.

The response of CPCs and CEACs to CEA AOOs is maintained with this TAR installed. Prompt operator action per the ACTION statements in Technical Specifications LCOs are also credited in the bases as limiting the effect of CEA misalignments. None of these prompt actions are affected by the TAR. Thus, the margin of safety as defined in the basis of any Technical Specification is not reduced.

17. TAR-96-002, Instrument Air Backup Compressor

DESCRIPTION OF CHANGE

The TAR provides a backup source of air to the Instrument Air (IA) system while the Station Air (SA) system is out of service.

REASON FOR CHANGE

During performance of STP-289682 (Item 162, Report of Facility Changes, Tests, and Experiments - 1994, W3F2-94-0051, dated October 20, 1994) the SA system will be isolated from the IA system and depressurized to verify that SA-126 and SA-127 are adequately leak tight to serve as isolation boundaries for implementation of DC-3390, Instrument Air/Station Air Enhancements. The addition of the temporary compressor will assist the IA compressors to maintain normal system pressure and operation if required.

SAFETY EVALUATION

The safety evaluation addresses the following events of concern for this TAR; Decrease in Feedwater Temperature, Increase in Feedwater Flow, Increase in Main Steam Flow, and Inadvertent Atmospheric Dump Valve Opening. The evaluation concludes that installation of a backup compressor enhances the ability of the IA system to maintain normal pressure and capacity for system loads related to the preceding events. The evaluation states that the IA system is not safety related and is not required for safe shutdown of the plant or for limiting radiological releases. Results of the evaluation were that no unreviewed safety question is associated with the TAR

D. DOCUMENT REVISION NOTICES (DRN)

1. DRN-C-9401921 and DRN-C-9401923, Low Level Radioactive Waste Storage Facility

DESCRIPTION OF CHANGE

Updates drawings and the FSAR to reflect the addition of the Low Level Radioactive Waste Storage Facility (LLRWSF)

REASON FOR CHANGE

Waterford 3 has shipped low-level radioactive wasted to off-site waste disposal facilities, however, closure of these facilities and delays in opening a regional waste disposal facility resulted in the decision to construct an on-site interim storage facility for low-level radwaste. The LLRWSF will provide storage space for a total of five years based on current Waterford 3 estimates of waste generation. This is in accordance with Generic Letter 81-38 guidelines.

SAFETY EVALUATION

The safety evaluation states that storage of low level radwaste in the LLRWSF does not involve or impact any of the systems or components in the gaseous and liquid waste systems subject to failure as addressed in the FSAR. Radwaste systems will continue to operate as before. Utilization of plant radwaste processing systems is no different than when wastes were being shipped off-site.

Four operations-related accidents were postulated for the facility and reviewed: 1) a HIC (High Integrity Container) drop, 2) a heavy load drop onto stored HICs, 3) a malfunction of the crane during handling of the payload and, 4) a mishap during transfer of the wastes from the plant to the new facility.

HIC Drop: NRC approved testing documents the toughness of the HICs. The maximum lift height of the HICs will be less than the 25 feet free fall the HICs are designed for. Thus, loss of integrity of the HIC due to being dropped during transfer operations will not occur.

Heavy Load Drop onto HICs: This accident postulates dropping a vault top skyshine concrete panel onto a stored HIC in the vault causing the HIC to rupture. Resins in the HIC are bulk dewatered solids; therefore, there will be no airborne contamination. The 2 foot thick concrete walls of the vault provide shielding, thus, a member of the public would receive a very small fraction of 10CFR100 dose limits.

Crane Malfunction During Load Handling: The crane is equipped with an automatic hoist brake and a manual load release. Any dose the public receives from skyshine is bounded by the HIC being suspended in the air if the crane malfunctions during load handling. Calculations demonstrate that a 44 rem/hr HIC would only render 0.1 mrem/hr at the nearest site boundary (approximately 980 ft. Away). It is evident that the HIC would be properly dispositioned from the stuck position before a member of the public receives the allowable dose limit of 2.5 rem (10% of the 10CFR100 limits).

Transportation Accident During Transfer of Wastes: Transport of containers will remain entirely on the owner controlled area. Rupture of the HIC is not postulated as a credible accident since the HIC can fall no more than 6 or 7 feet off of the delivery vehicle (HICs are proven to retain structural integrity through drops of up to 25 feet).

Other events, fire, tornado and flooding were also evaluated for the LLRWSF. Administrative controls will be used to ensure that no ignition sources are introduced to the facility. Calculations demonstrate that the whole body dose to the general public will be well below the 10% of the 10CFR100 limit using the conservative assumption that a fire totally engulfs all the DAW (Dry Active Waste) containers. Fire protection water used would be contained within the facility. The HIC's are protected by the 2 fo. thick concrete vault, and therefore are not engulfed in the fire. Design of the vault walls are enveloped by the design of the Reactor Auxiliary Building exterior walls which were designed to withstand tornado loads in combination with dead, live, and equipment loads. Maximum elevation flood waters are expected to rise, based on the worst floods postulated at Waterford 3 is Elevation 27.5 feet. The HIC vaults will remain dry (top is at El. 34 ft.) but the DAW containers would be submerged. Conservative calculations demonstrate that the whole body dose to the general public at the closest site boundary will be significantly less than the allowed 10CFR100 dose limit.

The facility has been designed so that an individual only receives a small fraction (10%) of the dose limits stipulated in 10CFR20 for normal operation and 10CFR100 for accident conditions. A member of the public will receive a maximum dose of less than 0.05 rem/yr during normal operations which is 10% of the allowable. During accident conditions a member of the public would not receive any thyroid dose and less than 2.5 rem whole body which is 10% of the allowable. Therefore there is no reduction in the margins of safety as described by the Technical Specifications.

2. DRN-C-9500371, Relocate RM-11B Console

DESCRIPTION OF CHANGE

This DRN documents the relocation of the RM-11B Console from the Health Physics (HP) Count Room to the HP office. Both areas are on the -4 elevation of the Reactor Auxiliary Building (RAB). FSAR Figure 1.2-10 is revised to indicate this relocation.

REASON FOR CHANGE

The DRN has no impact on the function of the console, which is to provide the operator with the status of each RM-80 in the loop.

SAFETY EVALUATION

According to the safety evaluation there is no unreviewed safety question associated with the DRN. The console will continue to perform its function. The safety evaluation notes that the RM-11B is not required to function in any accident scenario and that no new system interfaces or interconnections are introduced. The activity does not reduce the margin of safety as defined in the basis for any Technical Specification.

3. DRN-C-9600047, Resolution of DBD-028 Open Items (FSAR Section 3.8.4.1.1 and FSAR Tables 3.5-10 and 3.8-39

DESCRIPTION OF CHANGE

The DRN involves revision to the design information in the FSAR to reflect the required design load combinations for the Shield Building as used in the design/analysis. The changes do not in any way affect the structural integrity and function of the Reactor Shield Building.

REASON FOR CHANGE

Waterford 3 Design Basis Document (DBD) -028 identified discrepancies between the Reactor Shield Building Design Calculation, 6W12-RB-001 (Q), and the FSAR as related to the load parameters and load factors used in the design load combinations.

SAFETY EVALUATION

According to the safety evaluation the DRN is specific and addresses only the alteration of the Design Loads in the FSAR for the Shield Building, it does not affect the structural integrity of the shield structure. The shield structure is not part of the initiation of any of the accidents listed in the FSAR and does not increase the probability of occurrence of an accident previously evaluated in the FSAR.

The revised design loading does not affect the components important to safety, since there is no load path between the shield structure and the components. The DRN does not involve any new system interactions or connections with the shield building, also, all pipe penetrating the shield building is connected by expansion bellows which isolate the piping loads from the structure. The deleted loads never actually existed and cannot be rationally postulated. Thus, no credible failure modes are created or deleted.

The DRN does not reduce the margin of safety as defined in the bases for any technical specification or the appropriate safety analysis.

4. DRN-E-9401129, Installation Specification for Fire, Air, Water, and Radiation Penetration and Conduit Seals

DESCRIPTION OF CHANGE

The DRN lessens the internal conduit sealing requirements and revises the specification to indicate the new requirements.

REASON FOR CHANGE

Criteria established as a result of a fire test program sponsored in part by EPRI is less restrictive than that to which Waterford 3 is currently designed. This DRN revises the Seal Installation Specification to reflect these less restrictive requirements.

SAFETY EVALUATION

According to the safety evaluation the effects of fires have been evaluated in the FSAR for each of the plant's fire areas. Engineering evaluation and fire testing have shown that conduit seals are not necessarily required in all conduits penetrating fire barriers. Unsealed conduits meeting defined specifications for diameter, fill, and configuration have been shown to have no adverse effect on a fire barrier's ability to prevent the passage of flame, smoke, and hot gases from one fire area to another. As a result, none of the fires postulated in the FSAR are affected by this DRN. The evaluation concludes that there is no change to the required response to a plant fire as the result of this DRN.

5. DRN-E-9401550, Installation Details/Fire Protection & Raceway Separation Details

DESCRIPTION OF CHANGE

This DRN allows exceptions to the 1 inch separation criteria for instrumentation/control circuits in enclosed raceways for safety related equipment.

REASON FOR CHANGE

Separation is a design feature for maintaining the independence of one redundant service from that which it is redundant. Due to conduit congestion in local areas, minimum separation can not always be maintained.

SAFETY EVALUATION

According to the safety evaluation there is no accident in the SAR affected by this DRN. The SAR states an analysis may be performed in lieu of flame retardant material to provide minimum separation distance. Analysis NCR-W3-7621 provides an electrical separation criteria analysis stating that when instrumentation/control circuits in one enclosed raceway which are within one inch (including touching) of another raceway containing instrumentation/control circuits, a modification to the design is not required since there is neither sufficient energy to result in a fault in a redundant or safety related system circuit.

The DRN does not impact any protective boundary, any margin of safety, or previously analyzed acceptance limits. The DRN allows exceptions for instrumentation/control circuits separation criteria because there is insignificant energy level to impact redundant safety raceways.

6. DRN-I-9400095, Sampling System Flow Diagram

DESCRIPTION OF CHANGE

The DRN updates the Sampling System Flow Diagram to reflect the plant configuration.

REASON FOR CHANGE

The system flow diagram indicates that two pressure reducing valves are installed, however. the valves shown are only used in high pressure applications, this is a low pressure application. The DRN will revise the flow diagram to reflect that the two valves are not installed.

SAFETY EVALUATION

According to the safety evaluation this is a "paper change" and the configuration of the plant is not affected. Because there is no change to the plant there is no reduction in any margin of safety associated with this DRN.

7. DRN-I-9401389, I-9401432, and I-9401441, Emergency Diesel Generator Oil Schematic Drawings

DESCRIPTION OF CHANGE

The DRNs update Emergency Diesel Generator (EDG) Oil (EGL) Schematic drawings to reflect as-built conditions.

REASON FOR CHANGE

Changes to the drawings will show the proper location of a EGL pressure transmitter, correct routing for the tubing for the transmitter and deletion of a valve from the tubing.

SAFETY EVALUATION

According to the safety evaluation the DRNs will update the drawing to reflect as-built conditions. The function, operation, and integrity of the EDG Lube Oil System is unchanged by this drawing change. There are no plant changes associated with the DRNs.

8. DRN-M-9203391, Emergency Feedwater Drawings

DESCRIPTION OF CHANGE

Correct drawings to reflect as built plant conditions.

REASON FOR CHANGE

Drain valves were not installed by the A/E but the system drawings reflected that they were. System walkdown verified that the valves were not installed as indicated in system drawings.

SAFETY EVALUATION

The safety evaluation states that the drain valves were never installed and that removing them from system drawings will not affect any SAR accidents. The valves do not affect system operation. There is no unreviewed safety question associated with this DRN.

9. DRN-M-9203392, M-9203398, M9401791, Extraction Steam System Drawing Update

DESCRIPTION OF CHANGE

These DRNs update FSAR Figures 10.2-4, Sheets 3 & 4 and 10.4-5, Sheet 1 to reflect the as-built location of vents and drains associated with the Extraction Steam (ES) System. No physical changes are made to the plant.

REASON FOR CHANGE

The DRNs document installed vents and drains, to the drawings only, that were installed during construction. Architect/Engineer procedures were used during construction to install vents and drains to aid in system filling, venting, and testing.

SAFETY EVALUATION

The ES system is non-safety related and this activity has no effect on FSAR evaluated accidents or radiological consequences. According to the safety evaluation the vents and drains added are not required for operation of the ES system. The evaluation states that no protective boundaries are affected by this DRN and there are no margins of safety affected.

10. DRN-M-9203393, M-9203400, M-9301815, and M-9301816, Feedwater Heater Vent (FHV) System

DESCRIPTION OF CHANGE

The listed DRNs update system drawings to reflect vents that were installed during construction to aid in filling, testing, or draining the system.

REASON FOR CHANGE

During plant construction vents were added at piping high points to aid in filling, testing, or draining the system. These vents were installed in accordance with the A/E procedures. These DRNs update system drawings to reflect the vents, verified by system walkdowns.

SAFETY EVALUATION

According to the safety evaluation this activity only documents vents installed during construction, there are no changes to the system as a result of these DRNs. The system is non-safety related and the vents being documented are not required or used for operation of the FHV system. There are no protective boundaries affected by this activity and there are no new system interactions or connections caused by the DRNs.

11. DRN-M-9203394, Circulating Water System Drawing Update

DESCRIPTION OF CHANGE

This DRN updates FSAR Figure 10.4-5, Sheet 1 to reflect the as-built location of vents and drains associated with the Circulating Water (CW) System. No physical changes are made to the plant.

REASON FOR CHANGE

The DRN deletes a vent, from the drawing only, that was not installed during construction. Architect/Engineer procedures were used during construction to install vents and drains to aid in system filling, venting, and testing. This vent was inadvertently added to the system drawing.

SAFETY EVALUATION

The CW system is non-safety related and this activity has no effect on FSAR evaluated accidents or radiological consequences. According to the safety evaluation the vent is not required for operation of the CW system. The evaluation states that no protective boundaries are affected by this DRN and there are no margins of safety affected.

12. DRN-M-9400990, Fire Protection Flow Diagram

DESCRIPTION OF CHANGE

The DRN revises the Fire Protection flow diagram to reflect the as-built configuration.

REASON FOR CHANGE

The DRN deletes a vent valve that was originally specified for air removal in a section of piping that is no longer used (service pump was previously deleted resulting in the piping being non-functional). Instrumentation changes identified in the DRN reflect the as-built configuration of the instrumentation and the flow diagram will be in agreement with the instrumentation details drawing.

SAFETY EVALUATION

The safety evaluation states that the DRN has no impact on the function or operation of the fire protection system. Instrumentation affected by the DRN is flow instrumentation used only for periodically flow testing the fire pumps and is not analyzed in any SAR accident scenarios.

13. DRN-M-9500331, LTOP Relief Valve Capacity Revision

DESCRIPTION OF CHANGE

The DRN revises the required and full accumulation capacities contained in the FSAR for the Low Temperature Over Pressure (LTOP) relief valves from 3089 GPM and 3505 GPM to 3102 GPM and 3345 GPM respectively.

REASON FOR CHANGE

New values for required capacity of the LTOP valves were obtained from calculation EC-M94-002 and the full accumulation capacity was provided on the NV-1 form contained in the purchase order for the valves.

SAFETY EVALUATION

According to the evaluation the LTOP relief valves are not included in any accidents previously evaluated in the FSAR. The change does not physically affect the valves, it only documents the required and full accumulation capacities of the LTOPs. The full accumulation capacity is greater than the required capacity, therefore, the valve is capable of performing its safety function. The function of the LTOP relief valves is to provide overpressure protection to the Reactor Coolant System under low temperature operation. Alignment of the LTOPs to the RCS is administratively controlled by aligning shutdown cooling to the RCS. This DRN does not affect a protective boundary.

14. DRN-M-9502113, M-9502198, and M-9502199, Circulating Water System

DESCRIPTION OF CHANGE

The DRNs update various drawings associated with the Circulating Water System to reflect the physical arrangement of the plant.

REASON FOR CHANGE

Correct various Circulating Water System drawing and FSAR drawing to reflect as-built condition of the plant.

SAFETY EVALUATION

According to the safety evaluation the DRNs do not cause any physical changes to the plant. No accidents evaluated in the FSAR will be affected by the DRNs. No physical changes to plant systems are caused by the DRNs.

E. LICENSE DOCUMENT CHANGE REQUESTS (LDCR)

1. LDCR-94-0204, Remove Reference to a Blowdown/Waste Pond Discharge Radiation Monitor

DESCRIPTION OF CHANGE

This LDCR removes a reference to a radiation monitor located upstream of the Circulating Water radiation monitor (PRM-IRE-1900) which does not exist.

REASON FOR CHANGE

Section 10.4.8.2 of the FSAR states "In addition, a second radiation monitor is provided upstream of the Circulating Water System monitor which will automatically isolate the Circulating Water/waste pond discharge upon indication of high radioactivity." However, FSAR Figure 10.4-5, sheet 2 of 2 indicates this monitor to be a "Future" monitor. There is a connection available but the monitor is not installed. This LDCR will correct FSAR section 10.4.8.2.

SAFETY EVALUATION

According to the safety evaluation there are no new system interactions, because the monitor does not exist. The Steam Generator Blowdown discharge, to either the Circulating Water System or the waste pond, was analyzed in the context of the limiting Steam Generator Tube Rupture (SGTR) event previously analyzed in the FSAR. It was determined that superimposing the additional pathways on the SGTR event produced a negligible increase in the total off-site dose consequence when compared to the 10CFR100 limits. Thus it is concluded that the LDCR will not create the possibility of an accident of a new or different type than any previously evaluated in the FSAR.

2. LDCR-94-0206, Revises Stroke Time of SI-602 A & B

DESCRIPTION OF CHANGE

This LDCR increases the maximum allowable stroke time of valves SI-602 A&B (Safety Injection System Sump Valves) from 25 seconds to 35 seconds for surveillance testing.

REASON FOR CHANGE

The 25 second stroke time used to procure the motor operator did not account for surveillance testing in accordance with plant procedure UNT-006-021, "Pump and Valve Inservice Testing" (i.e., Technical Specification 4.0.5). Additional margin (35 second stroke time) is needed to avoid unnecessary increases in the surveillance testing frequency that may be required by UNT-006-021.

SAFETY EVALUATION

According to the safety evaluation the Loss of Coolant Accident (LOCA) is the only accident that may be affected by this LDCR. Valves SI-602 A&B are used only to mitigate the consequences of a Design Basis Accident, they cannot initiate or cause any accident to occur. The function of the valves remains the same, thus changing the stroke time of the valves does not increase the probability of an accident. Engineering calculations were reviewed for the change in stroke time and the change has no impact on design basis radiological calculations.

3. LDCR-94-0207, Update to Chapters 7 and 8 and Table 8.3-12

DESCRIPTION OF CHANGE

This is a change to the FSAR regarding power and control equipment identification. The change will alter the color code requirements. Additionally, the change deleted the "colored dots" used for Associated Circuits and updates FSAR Table 8.3-12.

REASON FOR CHANGE

A Waterford 3 Corrective Action document (CR-93-153) concluded that the power and control equipment color coding is not in compliance with the FSAR (which is committed to R.G. 1.70). This change provides a consistent color coding scheme in compliance with the FSAR and R.G. 1.70.

SAFETY EVALUATION

The safety evaluation concluded that the new color coding scheme in the LDCR will enhance personnel identification of the correct equipment during plant operation and maintenance. Personnel are trained to identify equipment in accordance with procedures and component numbers. A color coded border will be used for equipment labels in lieu of a color coded background. Safety Trains "B" and "AB" and Associated Circuits "B" and "AB" are changed. In all other cases the old background label colors will match the new color border.

Colored dots currently used for Associated Circuits identification will be replaced with "Class Designators" (PA PB, etc.), "Class Designators" will also be added to Class 1E equipment.

The evaluation states that there is no impact on affected equipment, no degradation of the safety-related function of any equipment or system, and no unreviewed safety questions are created by the LDCR.

4. LDCR-94-0213, Revise Load Factors for Concrete Design

DESCRIPTION OF CHANGE

The LDCR will allow the use of ACI 318-71 load factors for the normal operating loads flexure analysis of beams and slabs when checking existing structures such as the Shield Building, Reactor Auxiliary Building, Fuel Handling Building, and Component Cooling Water system Structure for increased loads. The LDCR will still require that ACI 318-63 be used for the actual structural design of the concrete. The current load factors for any accident conditions will still be retained.

REASON FOR CHANGE

The LDCR will allow the load factors to be lowered slightly when performing flexure analysis of beams and slabs. This will result in existing concrete beams and slabs having a slightly greater design capacity than was allowed during the original design of the plant. No physical changes will be made to the plant as a direct result of this LDCR. The load factors being changed only affect the design of reinforced concrete structures covered by section 3.8.4 of the FSAR. The only portion of the design being affected is the load factor that is used for dead load and live load for flexure analysis of beams and slabs. The load factors for buoyancy, Loss of Coolant Accident, earth loads, OBE, SSE, wind loads, and internal negative and positive pressures will not be affected by this change.

SAFETY EVALUATION

The safety evaluation states that revising the load factors used in flexure analysis of reinforced concrete beams and slabs will not have any impact on the probability of occurrence or the consequences of an accident. Load factors for use during accident conditions will not be revised. The lowered load factors will be used only for reviewing existing concrete structures and are for normal operating loads only. The evaluation concluded that the margin of safety as determined in the bases for any technical specification or the appropriate safety analysis will not be reduced.

5. LDCR-94-0216, FSAR Chapter 15 Revisions

DESCRIPTION OF CHANGE

FSAR Table 15.0-3, "Reactor Protective System Trips Used in the Safety Analysis," is footnoted to document that the analysis for CEA Withdrawal from Subcritical conditions uses a different High Logarithmic Power Level Trip assumption (2.6%) than the value listed in Table 15.0-3 (2.0%). Minor changes are also made to Section 15.4.1.1 to document the 2.6% setpoint.

The Cycle 2 Loss of Condenser Vacuum (LOCV) analysis, Appendix 15.D.1, is being removed from the FSAR and will be documented in the Waterford 3 Safety Analysis Design Basis Document (SADBD).

Section 15.3.3.2 is revised to remove the analysis of a Reactor Coolant Pump (RCP) shaft seizure with a concurrent failure of a Main Steam Safety Valve (MSSV). This information is being transferred to the SADBD.

REASON FOR CHANGE

Waterford 3 Condition Report (CR), CR-93-102, documented a discrepancy between the CEA Withdrawal from Subcritical Conditions analysis and the equipment setpoint values for the High Logarithmic Power Level trip. This discrepancy prompted the re-analysis of the CEA Withdrawal event, there is no change in the 0.257% equipment setpoint documented in Technical Specifications Section 2.

FSAR Appendix 15.D.1 documented a bounded analysis for the LOCV, the Licensing basis LOCV analyses are in Section 15.2.1.3 and 15.2.2.3. Information from Appendix 15.D.1 will be documented in the SADBD.

The RCP shaft seizure with a concurrent MSSV failure analysis (Section 15.3.3.2) was performed in response to NRC FSAR Question 211.46. This analysis is not required by the Standard Review Plan and was included in the FSAR for information. The analysis does not assume worse case primary-to-secondary leakage. The licensing basis analysis for the RCP seized/sheared shaft event is documented in Section 15.3.3.1. Information from Section 15.3.3.2 will be transferred to the SADBD.

SAFETY EVALUATION

According to the safety evaluation there is no unreviewed safety question associated with the LDCR. The LDCR does not reduce the margin of safety defined in the bases for any Technical Specification. Re-analysis of the Subcritical CEA Withdrawal event with a 2.6% analysis setpoint demonstrates that the applicable acceptance criteria are met for that event (i.e., fuel centerline temperature remains below 4900 degrees F. and DNBR remains equal to or greater than 1.26).

Removal of the non-licensing basis analyses from the FSAR does not impact any margin of safety. These non-licensing basis analyses will be included in the Waterford 3 SADBD.

6. LDCR-94-0232, Pressurizer Safety Valves (PSV) and Main Steam Safety Valves (MSSV) Setpoint Tolerance Increase

DESCRIPTION OF CHANGE

This LDCR revises the allowable opening tolerances on the PSV and MSSV from +/-1% to +/-3%. The as-left setting, however, would always be adjusted within +/-1% of the specified lift setting.

REASON FOR CHANGE

Chapters 6 and 15 of the FSAR were affected by this LDCR. Currently the allowed opening setpoint tolerance for the PSV and MSSV is +/-1% per Technical Specifications. However, the allowed tolerance has been occasionally exceeded during past surveillance testing. Increasing the setpoint tolerance will allow more flexibility during plant operation and reduce the number of LERs that may result when valve setpoints are outside tolerance. ABB-CE performed the necessary evaluations to support this change.

SAFETY EVALUATION

According to the safety evaluation the LDCR does not involve any change to the physical characteristics of the PSVs and MSSVs and will have no impact on the as-left settings for these valves. The change only allows for a higher as-found setpoint tolerance. The analyses of the impacted events demonstrated that the events are either bounded by the present FSAR analyses or have results that are within the Standard Review Plan acceptance criteria. Thus the change has no impact on the probability of occurrence or the consequences of events previously evaluated in the FSAR.

The evaluation concluded that the change will not impact the possibility or consequences of a malfunction of equipment important to safety of a different type than any previously evaluated in the FSAR.

7. LDCR-94-0234, FSAR Figure 9.2-8, Sheet 1 of 3

DESCRIPTION OF CHANGE

This LDCR revises FSAR Figure 9.2-8 to better illustrate the instrumentation for the Chill Water System (CHW) expansion tanks. There are no physical changes made to the plant.

REASON FOR CHANGE

The LDCR clarifies the CHW expansion tanks instrumentation.

SAFETY EVALUATION

The safety evaluation states that this is "paper change" only and does not involve any field changes. There are no accidents affected by this LDCR. The drawing change will show the instrumentation as designed and installed more correctly than the current drawing. According to the evaluation the LDCR does not affect a protective boundary and does not reduce any margin of safety for the following reasons:

The LDCR changes a general arrangement drawing only,

The LDCR does not affect the design or operation of the expansion tanks, and,

The expansion tank level instrumentation is designed and installed in accordance with plant drawings.

8. LDCR-95-0012, Air Products Hydrogen Pipeline Project

DESCRIPTION OF CHANGE

LDCR-95-0012 incorporates the evaluation of a new hydrogen pipeline which crosses Louisiana Power & Light Company property in the vicinity of Waterford 3. The pipeline is evaluated as an external event due to the potential for pipeline break and explosion.

REASON FOR CHANGE

In the fall of 1994 construction will start on a new 12.75 diameter pipeline which will transport hydrogen. The pipeline will enter LP&L property to the south-west of Waterford 3 and exit the company property to the south-east of Waterford 3. The pipeline will be buried to a minimum depth of five feet on LP&L property. The closest above ground section of the pipeline will be 3200 feet from the eastern boundary of LP&L property.

SAFETY EVALUATION

The safety evaluation notes that the probability of a pipeline break and explosion will increase slightly (1.944E-4/year to 2.025E-4/year). However, the explosive effects of a break in the new pipeline are bounded by the present FSAR analyses and have no effect on the Waterford 3 safety related structures. The design basis explosive event probability (LPG tank truck explosion on Louisiana Highway 18) remains unchanged. Similarly, since the explosive effects of the new pipeline are bounded by the explosive effects of the Evangeline Pipeline (See Item 69 of "Report of Facility Changes, Tests and Experiments, Waterford 3 letter W3F2-92-0033, dated December 10, 1992) the probability of the design basis natural gas pipeline explosion as presented in FSAR section 2.2.3.1.3.1.2 remains unchanged. There is therefore no increase in the probability of a design basis accident due to the new pipeline.

The new pipeline does not reduce any margin of safety because the maximum overpressure from the break and explosion of the new pipeline (less than one psi) is less than the design basis overpressure (three psi) for the Waterford 3 safety related structures.

9. LDCR-95-0049, Revision of Figures in FSAR Chapters 4, 5, 6, 9, 10, and 11

DESCRIPTION OF CHANGE

This LDCR adds a note to various FSAR figures to indicate that valve positions are for reference only and may be different than that shown. The actual positions and locking requirements are controlled by Operations Department procedures.

REASON FOR CHANGE

This LDCR was generated to update applicable FSAR figures to reflect, in part, the corrective action associated with Waterford 3 Corrective Action Document CR-94-158.

SAFETY EVALUATION

The safety evaluation indicates that this LDCR does not alter any system or require any operating procedure to change. The LDCR adds a note to flow diagrams to clarify that valve positions and locking requirements are controlled by plant procedures. Addition of the note will prevent any discrepancies between plant procedures and the drawings. The evaluation states that there is no affect on any protective boundaries or the accident response of any plant systems because of this LDCR.

10. LDCR-95-0055, Revises FSAR 9.3.2.2.2, 9.2.6.3, 10.4.1.2, and Table 11.2-4

DESCRIPTION OF CHANGE

This LDCR clarifies information in the FSAR related to the Sample Recovery Tank discharge path. And it also clarifies information related to the Condensate Storage Tanks (CST) to indicate that the CST may contain contaminated water during periods of steam generator tube leakage and receipt of condenser hotwell water.

REASON FOR CHANGE

Current information in the FSAR indicates that the Sample Recovery Tank can only be pumped to the main condenser, it can also be pumped to the Industrial Waste Sump. The FSAR currently indicates that the CST never contains radioactivity contaminated water, however, it may contain contaminated water during periods of steam generator tube leakage and receipt of condenser hotwell water.

SAFETY EVALUATION

The safety evaluation indicates that the only accident that would be affected by this LDCR is the design basis liquid radioactive waste system accident. However, because the LDCR does not affect the physical part of the Boron Waste Management System or the Liquid Waste Management System (LWMS) the probability of the design basis accident in the FSAR is not increased.

According to the evaluation total activity discharged (in the form of radionucleides) is not changed, except during periods of Steam Generator tube leakage, and the quantity and concentration of radionuclides released will be controlled by the Off-site Dose Calculation Manual (ODCM). Additionally the discharge will be stopped by a radiation monitor. The ODCM does not require revision as a result of this LDCR.

The safety evaluation concludes that the contaminated water stored in the CST is expected to be used in the secondary system and that off-site dose consequences of a CST failure are limited because of application of TS 3.11.1.4 limits to the CST.

The evaluation notes that protective boundaries are unaffected by this LDCR. The evaluation also included a Radioactive Waste Systems Additional Safety Evaluation and a Environmental Impact Evaluation.

11. LDCR-95-0066, Extension of Turbine Steam Valve Test Interval

DESCRIPTION OF CHANGE

This LDCR changes the Turbine Steam Valve Test Interval from monthly to quarterly and adds a summary of the manufacturer's analysis to FSAR Chapter 3.

REASON FOR CHANGE

Less frequent testing of the valves means less wear-related demand failures, reduces the probability of a sudden loss of load on the generator and the reliability of the valves has improved since Waterford 3 started operation.

SAFETY EVALUATION

According to the safety evaluation the probability of wear-related valve failure and the probability of loss of load events are decreased by this change and the valve failure rate has shown great improvement since the original FSAR analysis. The turbine manufacturer (Westinghouse) performed a study in October 1994, "Turbine Valve Test Frequency for Entergy Operations at Waterford Station," that shows the probability of missile generation has improved from 1E-4 (original FSAR analysis) to 1.05E-5 even with a three month test interval. This change does not alter the effect that the malfunction of the turbine steam valves would have on any accidents evaluated in the FSAR, it only changes the valve test frequency.

12. LDCR-95-0069, Revises FSAR 5.4.15.2.3.b

DESCRIPTION OF CHANGE

Revises FSAR 5.4.15.2.3.b to delete reference to and the specific method of leak testing the Reactor Gas Venting System (RCGVS) valves.

REASON FOR CHANGE

Valves in the RCGVS are tested to ASME Section XI requirements for Category B valves per FSAR 5.4.15.1.i. Category B valves do not require leakrate testing, inservice testing adequately verifies the operability of these valves.

SAFETY EVALUATION

According to the safety evaluation there is no unreviewed safety question associated with this LDCR. The evaluation states that the valves will continue to be tested as required by NUREG 0737 as Category B valves. ASME Section XI testing of the RCGVS will continue to ensure that valve degradation is identified. The valves are monitored as part of overall Reactor Coolant System leakage monitoring requirements. Any leakage that could occur would be confined to Containment and would be restricted by upstream orifices which limit leakage to within the capacity of two charging pumps. Based on limits imposed by Technical Specification 3.4.5.2, there are no accidents affected by this LDCR. Since the system operation and testing has not changed, there will be no change in the margin of safety defined in the bases for Technical Specification or any safety analysis.

13. LDCR-95-0071, Revises Table 6.2-32, Concerning EFW Isolation Valves

DESCRIPTION OF CHANGE

The LDCR adds a footnote to the Containment Isolation Valve Table (Table 6.2-32) to address the operability requirements associated with the Emergency Feedwater (EFW) System control and isolation valves.

REASON FOR CHANGE

The LDCR will allow either EFW series isolation valve to fulfill the requirements of Technical Specification 3.6.3. Currently, with either the control or the isolation valve inoperable, the penetration must be isolated within 4 hours to comply with TS 3.6.3. The LDCR makes it clear that containment isolation, including single active failure, is accommodated with a single valve and a closed system inside containment consistent with General Design Criteria (GDC) 57.

SAFETY EVALUATION

According to the safety evaluation the LDCR does not involve an unreviewed safety question, will not deviate from the current licensing basis, and promotes plant safety. The change will impact the administrative implementation of TS 3.6.3, "Containment Isolation," by allowing either EFW series isolation valve to fulfill the TS operability requirement. The LDCR has no impact on the containment isolation function or the EFW functions.

The LDCR clarifies the two isolation barriers required for containment isolation by crediting the Secondary System as a closed system inside containment pursuant to GDC 57 and allowing either of the EFW series isolation and/or control valves to provide the second isolation barrier. The Secondary System meets the design requirements of Standard Review Plan 6.2.4, that are specified when crediting a closed system inside containment as an isolation barrier.

14. LDCR-95-0104, FSAR Tables 3.9-9 and 3.9C-1

DESCRIPTION OF CHANGE

The LDCR removes Nitrogen System (NG) valves NG-161A(B) and NG-162A(B) from the Non-NSSS Supplied Active Valve List in FSAR Tables 3.9-9 and 3.9C-1.

REASON FOR CHANGE

Removal of the valves will accurately reflect plant design because these valves do not perform an active safety function.

SAFETY EVALUATION

The safety evaluation states that removal of NG-161A(B) and NG-162A(B) from the FSAR tables does not alter the operation or function of the valves or any system and can not cause or affect any accidents described in the FSAR. The valves supply and isolate Nitrogen to the Safety Injection Tanks (SIT) and provide a Class break, Safety Class 2 to Non-Nuclear Safety. FSAR Tables 3.9-9 and 3.9C-1 incorrectly states that these valves must open during an accident.

15. LDCR-95-0135, Removal of Valves CAP-102, CAP-205, EFW-223A&B, And EFW-224A&B from FSAR Table 7.5-3 and Clarification in Table 6.2-32

DESCRIPTION OF CHANGE

This change removes Containment Atmospheric Purge (CAP) valves CAP-102 and CAP-205 and Emergency Feedwater (EFW) valves EFW-223A&B and EFW-224A&B from FSAR Table 7.5-3. The change corrects several administrative and typographical errors in Table 7.5-3. The changes also adds a reference note to the CAP valves in FSAR Table 6.2-32

REASON FOR CHANGE

The valves listed above are not considered to be Regulatory Guide 1.97 Category 1 for position indication. Thus this LDCR removes those valves from Table 7.5-3. The added note to Table 6.2-32 clarifies that CAP-102 and CAP-205 are not credited for containment isolation.

SAFETY EVALUATION

According to the safety evaluation the LDCR clarifies Waterford 3's position with regard to containment isolation valve control room position indication for the valves listed the "Description of Change" above. The LDCR has no affect on the function or operation of the valves. The Waterford 3 CAP system has three associated isolation valves in series. CAP-102 and CAP-205 are the third valve in series (from inside containment to the outside of containment) and as such are not credited for meeting containment isolation requirements.

EFW-223A&B and EFW-224A&B are currently listed as RG 1.97 Category 1 containment isolation valves and the Table indicates that the valves have position indication in the Control Room. However, the Control Room indication is for controller output to the valves, not the valve position. Other parameters are available to the operators to determine valve position. These valves are only credited for containment isolation if one or more of the associated EFW isolation valves (EFW-228A&B and EFW-229A&B) are inoperable. If this were to occur the TS EFW 72 hour Action Statement would be entered. This Action is much more restrictive than the 7-day Action for the Accident Monitoring Instrumentation TS.

According to the evaluation the requirements and margin of safety for containment isolation and accident monitoring instrumentation are preserved. The LDCR will not reduce the margin of safety as defined in the bases for any technical specification or the appropriate safety analysis.

16. LDCR-95-0150, FSAR Section 9.5.1.3.1.D.3(d)

DESCRIPTION OF CHANGE

The LDCR clarifies Waterford 3 use of fire barrier penetration seal qualification standards to allow the use of engineering evaluations in justifying acceptability. The use of engineering evaluations is permitted under Generic Letter (GL) 86-10.

REASON FOR CHANGE

The LDCR documents the fact that untested seals are evaluated by a qualified fire protection engineer, it does not impact the function of fire barrier penetration seals.

SAFETY EVALUATION

The safety evaluation states that this is a paper change only that clarifies the acceptance criteria of the fire barrier penetrations located in required fire barriers. Appendix A fire barrier penetrations have been qualified by an acceptable fire test or an engineering evaluation has been performed per GL 86-10 to evaluate the variances from the fire test. The GL 86-10 evaluation ensures that the fire barrier penetrations are acceptable and will, in the event of a fire, perform as a component of the fire barrier.

F. MISCELLANEOUS EVALUATIONS

1. Portal Monitor Source Check Frequency, Change to Commitment P-20781

DESCRIPTION OF CHANGE

This change to Waterford 3 commitment P-20781 changes the source check frequency for the Portal Monitors (PM-7) from every shift during outages to that frequency stated in FSAR 12.5.2.2.3.

REASON FOR CHANGE

Licensee Event Report 91-006, submitted to the NRC June 10, 1991 documented an event where a personnel monitor failed to detect a hot particle which was close to the detection limit of the equipment. In addition to other corrective actions contained in the LER Waterford 3 also increased the periodicity of source checks on the portal monitors during outages from daily to every shift. Since installation in the spring of 1991, several improvements have been implemented for the monitors, the four years of operational experience has only resulted in the replacement of four detectors, two power supplies and two logic boards. The implementation of the shift checks could be deemed as an unnecessary extra conservative action to prevent recurrence of the LER event. Because of the equipment demonstrated reliability and enhancements the committed practice of shift source checks is not necessary.

SAFETY EVALUATION

According to the safety evaluation changing the source check periodicity of the PM-7s has no effect on any accident listed in the SAR. The frequency specified in the appropriate Health Physics procedure will not exceed FSAR 12.5.2.2.3 which states that source checks of personnel survey instrumentation will be performed weekly when in use. PM-7 personnel monitors are not equipment important to safety nor do they affect equipment important to safety. Based on operational performance of these PM-7 monitors, this change will not decrease its performance. The change will not increase the instrument's current drift level that affects the monitor's sensitivity. Thus, the alarming level will be within the appropriate margin.

2. Cycle 8 Core Reload

DESCRIPTION OF CHANGE

The Cycle 8 core will utilize a "very low leakage" fuel management scheme, with an estimated reactivity life of 521 effective full power days (EFPD) at 30 ppm boron concentration at a Cycle 7 end point of 500 EFPD. The Cycle 8 Reload Analysis supports an operation range of 512-549 EFPD for the long Cycle 7 end point (516 EFPD) and an operation range of 529-565 EFPD at the short Cycle 7 end point (486 EFPD).

The Cycle 8 core consists of 96 new Batch K assemblies and 121 previously irradiated assemblies (all 92 Batch J and 29 Batch H assemblies). Reload batch K will consist of 20 type K0 assemblies (4 burnable poison rods (0.028 gm B10/in) per assembly), 4 type K1 assemblies (8 burnable poison rods (0.020 gm B10/in) per assembly), 56 type K2 assemblies (16 burnable poison rods (0.028 gm B10/in) per assembly), and 16 type K3 assemblies (12 burnable poison rods (0.020 gm B10/in) per assembly). The design cycle length for the Cycle 8 core is slightly longer than it was for Cycle 7 (521 EFPD vs. 510 EFPD at 30 ppm soluble boron).

The Cycle 8 core potentially has a more positive moderator temperature coefficient (MTC) at beginning of cycle hot zero power conditions than does the previous cycle. However, the best estimate value ($+.221 \times 10^{-4}\ \Delta\rho/°F$,) is less positive than the maximum used in the safety analysis input ($+.5 \times 10\text{-}4\ \Delta\rho/°F$). The MTC at full power is sufficiently negative, as required by the Technical Specifications and stated in the COLR.

As a demonstration that shutdown margin has not eroded from the Reference Cycle, shutdown margin (assuming the most reactive control rod stuck out) was appropriately calculated for the end of cycle full power (most limiting) main steam line break transient. The net available scram worth (7.81 % $\Delta\rho$) remains above the required (5.15 % $\Delta\rho$).

The Cycle 8 maximum integrated fuel rod burnup at the safety analysis upper burnup limit is 59,827 MWD/T. The burnup is higher than that of Cycle 7 but remains below the 60,000 MWD/T limit imposed by the ABB-CE topical report.

Technical Specifications Amendment 108 (dated June 14, 1995) increased the maximum enrichment for the spent fuel pool and the temporary storage racks from 4.1 to 4.9 weight percent U-235 when the fuel assemblies contain fixed poisons. Analysis demonstrated that lattices containing higher enrichment combined with poison shims can be designed such that the calculated k_{eff} is less than 0.95 under all conditions. The new fuel rack will not be used for Cycle 8. The maximum reactivity of the fresh zoned fuel (K0 assemblies) was shown to be less than that of the base bundle design used to support Amendment No. 108. The as-built stack height densities and enrichments were found to be within the tolerances assumed in the safety analyses used to support the enrichment increase. Therefore, the Cycle 8 fresh fuel can be safely stored in the spent fuel rack and the temporary fuel rack. Irradiated fuel has considerably lower reactivity than the base bundle. Thus, the fuel from Batches H and J which will be discharged from Cycle 8 can also be safely stored in the spent fuel pool with no restrictions.

The mechanical design of the Batch K fuel assemblies are identical to those of the Batches H and J fuel used in Cycle 7, with the following exceptions:

The previous standard Inconel bottom spacer grid assembly is replaced with a redesigned Inconel bottom spacer grid assembly, called the Guardian Spacer Grid, in the Batch K fuel assemblies . The design features employed in the Guardian grid improves its ability to filter and entrap debris.

In conjunction with the Guardian grid assembly, the fuel rod assembly was redesigned to include:

- increased nominal active fuel length from 149.61" to 150.0"
- a redesigned low-volume plenum spring
- the removal of the upper alumna spacer disc, and
- a redesigned lower-end-cap

The poison rod assembly was also redesigned to include:

- increased poison column length from 135" to 136"
- a redesigned low-volume plenum spring
- a redesigned lower-end-cap, and
- the use of 0.5" long burnable poison pellets (vs. 1.0")

As a result of these design changes, the nominal weight of the Batch K fuel bundle decreases approximately 0.9% (approximately 13 pounds).

It should be noted that the overall envelopes of the fuel rod assembly, the poison rod assembly and the fuel bundle assembly all remain unchanged from the previous reloads. The HID-1L Zircaloy spacer grids are the same for Batch K as was used for Batches H and J. The fuel bundle assembly widths and shoulder gaps are unchanged. The CEA guide tube assembly is also unchanged.

REASON FOR CHANGE

The evaluation was performed to support the fuel reload for operational Cycle 8.

SAFETY EVALUATION

Evaluations of the impact of the design changes implemented for the Batch K reload fuel assemblies with the Guardian spacer grid, including the increased nominal active fuel and poison column lengths, the redesigned fuel/poison rod low-volume plenum springs, the redesigned fuel/poison rod lower endcaps, the removal of the upper alumna spacer disc in fuel rods, and the use of 0.5" long burnable poison pellets, have shown that none of the fuel assembly mechanical structural design criteria (FSAR 4.2.1) for the normal operating and upset conditions, emergency conditions, and faulted conditions are violated.

The probability of fuel failure due to mechanical or flow induced vibration and fretting with the spacer grids (FSAR 4.2.1.2.1.g, 4.2.3.1.1, 4.2.3.1.3, 4.2.3.2.1 and 4.2.3.2.4) will not be increased. The Guardian grid fuel assembly has essentially the same structural cage as the previous reloads. Its fuel rods and poison rods have the same external dimensions, materials, clad thickness, and approximate mass as the Cycle 7 rods. The increased rod span length between the lower most Zircaloy spacer grid and the Guardian grid will not introduce any vibratory concerns; the (decreased) natural frequency of the fuel and poison rods is further away from the RCS pump blade frequency than they were in the previous batches. As such, the Guardian grid fuel assembly is dynamically the same as the non-Guardian grid assemblies.

The probability of a fuel handling accident (FSAR 4.2.1.1., 4.2.3.1.5, and 15.7.3.4) will not be increased. These assemblies have essentially the same structural cage as that previously used at Waterford 3 and will be capable of withstanding the expected handling loads. These assemblies will continue to be compatible with the fuel handling equipment. The mass of these new assemblies is reduced by approximately 10 pounds compared to the previous batch. Hence, the probability of a fuel handling accident is not increased.

The probability of CEA mis-operation [FSAR 15.4.1.4] is not increased. The dimensions and positions of the CEA guide tube assemblies are unchanged compared to the assemblies used in the previous cycles. Also, any dimensional changes due to irradiation, such as assembly bow, will not be altered since no changes in the guide tubes material have occurred.

No changes to the plant equipment or operating procedures are required for Cycle 8. No impact to any accident initiator occurs due to the Cycle 8 fuel. Therefore, the probability of an accident previously evaluated in the FSAR will not be increased due to the Guardian grid fuel assembly design.

As documented in the Cycle 8 Reload Analysis Report, ABB CE has reviewed all the accident analyses to determine whether these events are bounded by the Reference Analysis or need further evaluation and/or reanalysis. In most cases, comparison of key input parameters between Cycle 8 and the previous cycle determined that Cycle 8 inputs were bounded by the previous cycle inputs, thus no reanalysis was required. However, ABB-CE performed specific analyses for all events for which comparison of key input parameters could not demonstrate that the Cycle 8 results would be bounded by the Reference Analyses results. These events include:

- Increased Main Steam Flow with Loss of Offsite Power (Excess Load)
- Main Steam Line Break (Pre-Trip Power Excursion)

- Single Reactor Coolant Pump Shaft Seizure/Sheared Shaft

Increased Main Steam Flow With Loss of Offsite Power (Excess Load) was analyzed using HERMITE computer code, as was done for Cycle 7. The analysis of this event for Cycle 8, resulted in 2.4% of the fuel pins to experience DNB. This result remains bounded by the 3% value reported in the FSAR for the Reference Analysis (Cycle 6).

Main Steam Line Break (MSLB) (Pre-Trip Power Excursion) analysis used the criterion that all fuel rods for which the CE-1 DNBR falls below the 1.26 SAFDL experience cladding failure. The results of the inside and outside containment cases for Cycle 8 indicate that 3.36% (Reference Analysis 4.5%) of the fuel pins fail for the inside containment MSLB and 0.76% (Reference Analysis 3.0%) of the fuel pins fail for the outside containment MSLB. Since the predicted fuel failure for Cycle 8 is less than that of the Reference Analysis, the Reference Analysis offsite doses remain valid.

Single Reactor Coolant Pump Shaft Seizure/Sheared Shaft with Loss of Offsite AC (LOAC) power was reanalyzed due to a change in Cycle 8 fuel failure pin census. Both the RCP shaft seizure and sheared shaft were evaluated for Cycle 8. LOAC was evaluated for each and the sheared shaft was determined to be more limiting than the RCP seized rotor event

The analysis showed that the sheared shaft with LOAC results in a fuel failure of 4.89% which is less than the 8.5% (Reference Analysis) reported in the FSAR. The resultant offsite doses are less than 10% of the 10CFR100 limits (30 rem thyroid and 2.5 rem whole body). Thus, this event is bounded by the Reference analysis.

Loss of Condenser Vacuum was reanalyzed due to a 3 °F reduction in the minimum cold leg temperature from 544 °F to 541 °F. The analysis resulted in an RCS peak pressure of 2728 psia which is greater than the current analysis of record peak pressure of 2719 psia. The maximum pressure of 2728 psia, however, does not violate the 110% of the pressure vessel design limit (2750 psia).

An ECCS performance analysis of the limiting break size was performed for Cycle 8 to support an increase in the Waterford 3 safety injection tank (SIT) level and pressure Technical Specification ranges (indicated level: from 78%-83.8% to 40%-83.8%; indicated pressure: from 600-625 psig to 600-670 psig).

The following table provides the NRC acceptance limits, Reference Analysis and Cycle 8 (with revised SIT level and pressure) results for the ECCS analysis for the limiting large break LOCA.

	NRC Acceptance Limit	Ref. Analysis	Cycle 8
Peak Clad Temperature, °F	2200	2173	2177
Maximum Local Oxidation	17%	8.4%	8.55%
Core Wide Oxidation	1%	<0.805%	<0.805%

The above results show that the Cycle 8 results are slightly more limiting than the Reference Analysis. However, the Cycle 8 results meet the NRC acceptance limit.

The FATES3B fuel performance analysis has demonstrated that the Guardian grid fuel assemblies in the Cycle 8 core are less limiting, with lower peak pin pressure, lower fuel centerline temperature, lower fuel average temperature, and higher power-to-melt ratio, than the non-Guardian grid fuel assemblies. Therefore, the Cycle 8 fuel rod internal pressure, which will continue to be limited to below system pressure, is expected to be lower than the Cycle 7 fuel rods.

All accidents have been shown to have results within the appropriate NRC acceptance limits. The LOCV and LOCA results were more adverse than analysis of record, but within the acceptable limits. Therefore, there is no reduction in any margin of safety.

The fuel performance of both the Guardian grid and non-Guardian grid fuel designs at higher Cycle 8 burnups has been evaluated using NRC approved codes (FATES3B) and all design criteria were confirmed to be met. The maximum fuel rod burnup projected for Waterford 3 Cycle 8 is 59,827 MWD/MtU, and is less than the 60,000 MWD/MtU licensed limit. The fuel rod internal pressure remains below system pressure for the projected Cycle 8 maximum burnup. The Cycle 8 burnup will be well within the industry experience base. The FATES3B fuel performance analysis has demonstrated that the Guardian grid fuel assemblies in the Cycle 8 core are less limiting, with lower peak pin pressure, lower fuel centerline temperature, lower fuel average temperature, and higher power-to-melt ratio, than the non-Guardian grid fuel assemblies.

Therefore, the margin to safety will not be reduced for the Cycle 8 core due to fuel management changes.

3. Cycle 8 CPC/CEAC Addressable Constants

DESCRIPTION OF CHANGE

Implements the initial Cycle 8 Core Protection Calculator/Control Element Assembly Calculator (CPC/CEAC) Addressable Constants in accordance with ABB-CE Letter L-95-025, dated September 22, 1995.

REASON FOR CHANGE

CPC/CEACs provide reactor protection by tripping the plant on low DNBR and high LPD. This change ensures that the CPC/CEACs will provide the necessary trip functions as analyzed by the Cycle 8 Reload Analysis Report.

SAFETY EVALUATION

See Cycle 8 Reload Analysis Report, Item I.F.2 of this report.

4. CPC Constants From 68% Power Plateau

DESCRIPTION OF CHANGE

Implements the CPC constants derived from power ascension testing. These constants are developed using Plant Operating Committee reviewed procedures and are in accordance with the methodology and setpoints assumed in the Cycle 8 Reload Analysis Report (RAR). The Cycle 8 RAR assumes these changes are made.

REASON FOR CHANGE

The change ensures the CPC/CEACs will provide the necessary trip functions as analyzed by the Cycle 8 RAR.

SAFETY EVALUATION

See Item I.F.2 of this report, Cycle 8 Reload Analysis Report.

5. COLR Revisions for Cycle 8

DESCRIPTION OF CHANGE

Several changes are made to the Waterford 3 Cycle 7 Core Operating Limits Report (COLR) for Cycle 8.

REASON FOR CHANGE

Waterford 3 originated COLR during Cycle 7 and obtained NRC approval for implementing COLR on March 1, 1995. Cycle specific variables included in the COLR include:

- Shutdown Margin
- Moderator Temperature Coefficient (MTC)
- Boron Dilution
- Movable Control Assemblies - CEA Position
- CEA Insertion Limits
- Linear Heat Rate
- Azimuthal Power Tilt
- DNBR Margin
- Axial Shape Index
- Boron Concentration

The Cycle 8 revision includes the following; 1) specific values for MTC are replaced with a curve, 2) DNBR limit line for COLSS Out of Service and CEAC operable is replaced with two limit lines, one for all power levels and one for power levels equal to or greater than 90% respectively, 3) DNBR limit line for COLSS Out of Service and CEACs inoperable is replaced with two limit lines, one for all power levels and one for power levels equal to or greater than 90% respectively, and 3) editorial changes to convert the Cycle 7 COLR to Cycle 8 COLR.

SAFETY EVALUATION

According to the safety evaluation the changes are consistent with the assumptions used in the Cycle 8 reload safety analyses (see Item I.F.2 of this report). All accidents evaluated for Cycle 8 have been shown to have results within the appropriate NRC acceptance limits. Therefore the margin of safety as defined for any Waterford 3 Technical Specification will not be reduced.

6. SETPOINT CHANGE (SPC) 94-003, Plant Protection System (PPS) Setpoints (Revision 0)

DESCRIPTION OF CHANGE

The setpoint change revises setpoints and allowable values for High Linear Power, High Pressurizer Pressure, and High Steam Generator Differential Pressure. Setpoint changes addressed by this SPC are in the conservative direction.

REASON FOR CHANGE

The setpoint changes are all conservative with respect to the present analytical limits which also serve as the basis for the revised setpoint. Waterford 3 has developed a Setpoint/Uncertainty methodology and implemented a program to evaluate site setpoints and loop uncertainties. The new setpoints are the result of a new calculation EC-I92-019 Revision A. The revised calculation values are a result of additional uncertainty terms that are now considered due to a more current calculation methodology than was used to perform the existing PPS Setpoint Analysis.

SAFETY EVALUATION

The safety evaluation states that the revised setpoints are bounded by the existing setpoints because, 1) all of the changes are in the conservative direction, and 2), the revised setpoints are based on the same analysis setpoints (analytical limits) as the existing setpoints. All values used to calculate the revised trip/actuation setpoints are in accordance with the existing accident analyses. The SPC does not reduce the margin of safety as defined in the bases for any Technical Specification or the appropriate safety analysis. (See Waterford 3 submittal, Technical Specification Change Request NPF-38-152, letter W3F1-94-0121, dated June 21, 1994.)

7. SPC 94-004, Pressurizer Pressure HI/LO Alarm Annunciator H0501, RC-IPAC-0100-X and RC-IPAC-0100-Y (Revision 0)

DESCRIPTION OF CHANGE

SPC 94-004 will lower the Pressurizer High Pressure Alarm setpoint to 2270 psia.

REASON FOR CHANGE

SPC 94-003, Item I.F.6 of this report, changed the Pressurizer Pressure High pressure trip setpoint to 2350 psia. This setpoint change (SPC 94-004) lowers the "pre-trip" annunciation setpoint, this will alert the operator to take necessary actions to avoid a trip on High Pressurizer Pressure.

SAFETY EVALUATION

The safety evaluation states that the change is in the conservative direction and will have no adverse affect on the accident analysis of the SAR The change lowers the Hi Pressurizer Pressure Alarm Annunciation (window H0501) from 2350 psia (current trip setpoint) to 2270 psia. This conservative change will alert the operator when Pressurizer pressure is close to initiating a reactor trip and will have no impact on any accident analysis or margin of safety.

8. SPC 94-014, Fuel Pool Heat Exchanger Tube Side Outlet Temperature

DESCRIPTION OF CHANGE

The SPC will lower the Fuel Pool Heat Exchanger (FPHX) tube side (Fuel Pool water) outlet temperature to a variable setpoint of 98 degrees F. to 105 degrees F. The Fuel Pool outlet temperature will not exceed 120 degrees F.

REASON FOR CHANGE

The current setpoint of 115 degrees F. is too high for the light heat load in the Fuel Pool. Flow control valve CC-620 must be gagged in a single position to prevent the valve from oscillating. CC-622 is throttled to control flow. The variable setpoint will allow for operation of CC-620 without oscillation.

SAFETY EVALUATION

According to the SE there is no unreviewed safety question associated with the SPC. The SE notes that the FPHX is isolated in the event of an accident and is not required for the safe shutdown of the plant or to mitigate the consequences of an accident. The SPC will have no negative impact on any accident analysis or margin of safety.

The minimum temperature requirement (55 degrees F.) for the Fuel Pool will continue to be met and the maximum FPHX tube side outlet (pool water) will remain below 120 degrees F.

8. SPC 94-014, Fuel Pool Heat Exchanger Tube Side Outlet Temperature

DESCRIPTION OF CHANGE

The SPC will lower the Fuel Pool Heat Exchanger (FPHX) tube side (Fuel Pool water) outlet temperature to a variable setpoint of 98 degrees F. to 105 degrees F. The Fuel Pool outlet temperature will not exceed 120 degrees F.

REASON FOR CHANGE

The current setpoint of 115 degrees F. is too high for the light heat load in the Fuel Pool. Flow control valve CC-620 must be gagged in a single position to prevent the valve from oscillating. CC-622 is throttled to control flow. The variable setpoint will allow for operation of CC-620 without oscillation.

SAFETY EVALUATION

According to the SE there is no unreviewed safety question associated with the SPC. The SE notes that the FPHX is isolated in the event of an accident and is not required for the safe shutdown of the plant or to mitigate the consequences of an accident. The SPC will have no negative impact on any accident analysis or margin of safety.

The minimum temperature requirement (55 degrees F.) for the Fuel Pool will continue to be met and the maximum FPHX tube side outlet (pool water) will remain below 120 degrees F.

9. SPEER 9301117, Control Element Drive Mechanism (CEDM) Cooling Fan Motors (Revision 0)

DESCRIPTION OF CHANGE

This SPEER evaluates the use of Westinghouse Shop Order 78B4769, Westinghouse Shop Order NO08196, and General Electric Model 5KS449SS208C motors as CEDM Cooling Fan Motors. The SPEER ensures that all documentation is updated to reflect the present configuration utilizing these motors. The SPEER also provides instruction for the implementing group regarding document updates required when these motors are refurbished and re-installed in the plant.

REASON FOR CHANGE

The SPEER is required because the original motors are obsolete and no longer available.

SAFETY EVALUATION

The safety evaluation determined that an unreviewed safety question does not exist because of this SPEER. According to the evaluation the motor is associated with Non-safety related systems and therefore will not affect any accident listed in the FSAR. The motor is classified as Seismic Category I. The SPEER evaluated the change as an equivalent or better replacement.

10. SPEER 9401216, Main Steam Atmospheric Dump Valve (ADV) Positioner and Local Control System (Revision 0)

DESCRIPTION OF CHANGE

The SPEER evaluates the substitution of a Moore 74G pneumatic positioner, a Conoflow GT25 current to pneumatic (I/P) transducer and two Fisher 67AFR-206 air filter-regulators for a Bailey AP5 electropneumatic positioner.

The local control system will be replaced with a system that utilizes a pneumatic signal and pneumatic positioner to actuate the ADVs.

REASON FOR CHANGE

The original positioner is obsolete and no longer manufactured. The pneumatic positioner and I/P transducer are recommended by the original equipment manufacturer as replacements for the Bailey AP5 positioner.

SAFETY EVALUATION

The safety evaluation determined that an unreviewed safety question does not exist. The evaluation identifies two accidents, listed in the FSAR, that are caused by the ADVs, Inadvertent Opening of a Steam Generator ADV and Inadvertent Opening of a Steam Generator ADV with a Concurrent Single Failure of an Active Component. The SPEER replaces the positioner and much of the tubing associated with operating the ADVs. The ADVs fail shut in the absence of air pressure, failure of tubing downstream of the positioner or upstream of the new filter regulator may cause the ADV to shut but will not cause the ADV to open inadvertently. The evaluation notes that the radiological consequences of accidents are not affected by the changes to the ADVs, requirements for ADV operation are not changed, reliability of the ADVs is improved by the SPEER.

The evaluation states that the SPEER will result in improved operation and reliability of the ADVs and that the margins of safety associated with the ADVs will not be adversely affected.

11. SPEER 9401228, Evaluation of Replacement Spring Spacer Ring in CS-125A & B (Revision 2)

DESCRIPTION OF CHANGE

The SPEER addresses a material and thickness change for the spring spacer ring on the operator of the Containment Spray (CS) Header Isolation Valves. The spacers are to be fabricated from ASTM A516 Grade 70, as opposed to ASTM A519 Grade 1020 which was the original steel for this part. The thickness of the ring will be decreased from 1 1/2" to 1 1/4".

REASON FOR CHANGE

The change of thickness of the spacer ring will allow the valve to have a longer stroke. The longer stroke length will allow the valve to seat better than originally, decreasing the possibility of valve leakage. The thickness change will affect the opening force and the stroke time of the valve.

SAFETY EVALUATION

The safety evaluation determined that no unreviewed safety question exists with the SPEER. The evaluation notes that there are no accidents listed in the FSAR that would be affected by the SPEER. The thinner spacer ring will result in a decrease in opening force of 1% and stroke length is increased by 1/4". These changes will only result in an estimated 0.2 second increase in stroke time assuming conservatively that the original stroke time was 10 seconds. (Acceptance testing for DC-3409 (Item I.A.34 of this report) will ensure that stroke time requirements are verified and acceptable.) Containment peak pressure is not affected by the SPEER because the valve will open on Containment Spray Actuation Signal as designed. There are no other boundary performance parameters affected by the SPEER. There are no margins of safety affected, as the function, operation, and reliability of the valves have not been adversely affected.

12. SPEER 9401249, Replacement Valves for ACC-114A&B (Revision 0)

DESCRIPTION OF CHANGE

Replaces the existing safety related ACC-114A&B "soft" seated butterfly valves in the Auxiliary Component Cooling Water (ACCW) system with metal seated butterfly valves.

REASON FOR CHANGE

Existing valves leak due to seat wear caused by abrasives in the system. The metal seated valves can handle the abrasives in the ACCW water and provide positive shutoff.

SAFETY EVALUATION

According to the safety evaluation the ACCW system can not cause an accident. It is used to aid in the mitigation of an accident. The replacement valves will have the same fit and function as the existing valves. The replacement valves are manually operated, as are the existing valves, and they have approximately the same flow characteristics as the existing valves. Flow characteristics of the new valves are greater than that of the original check valves which were replaced by DC-3293 (Item 18, Report of Facility Changes, Tests, and Experiments - 1994, W3F2-94-0051, dated October 20, 1994). The replacement valves are procured as safety related and Seismic Class 1 in order to match the quality of the existing valves.

13. SPEER 9401268, Replacement of Underwater Fuel Pool Fixtures (Revision 0)

DESCRIPTION OF CHANGE

Replaces dual incandescent underwater light fixtures in the Refueling Cavity (32 fixtures), Spent Fuel Pool (12 fixtures), Refueling Canal (4 fixtures), and Spent Fuel Cask Storage Pool (2 fixtures) with single 1000 watt High Pressure Sodium (HPS) fixtures (total of 17 fixtures).

REASON FOR CHANGE

Condition Report 94-237 documented the occurrence of an erroneous placement of a fuel assembly (placed in Spent Fuel Pool location B-27 vice B-26) during Refuel 6. Visual impairment was identified as a programmatic weakness. Results of this SPEER are expected to be: increased total illumination, reduced power consumption, reduced maintenance downtime due to expected longer life of the HPS bulb, and reduced radiation exposure due to fewer and easier bulb replacements.

SAFETY EVALUATION

The safety evaluation notes that the SPEER will minimize the erroneous placement of a fuel assembly due to visual impairment because of the increased illumination (2,380,000 lumens vs. 850,000 lumens) and the beam of the HPS fixtures provides broad coverage with minimal upward scattering to obscure visibility. The underwater fixtures, ballast and poles are non-safety, non-seismic and non-EQ. Seismic II over I concerns were evaluated and determined to be acceptable. Ballast units associated with the fixture contains small amounts of aluminum and zinc, however, the quantity is negligible to affect the hydrogen recombiner operation.

A small amount of mercury is contained in the HPS fixture arc tube. This was evaluated and it was determined that contamination due to lamp breakage could be tolerated without significant concern.

14. SPEER 9401300, RCP Motor Oil Pump Suction Filter Replacement (Revision 0)

DESCRIPTION OF CHANGE

The existing Reactor Coolant Pump (RCS) Motor Oil Filter is being replaced with an updated model. The replacement filter assembly is being changed to a standardized aluminum filter cover.

REASON FOR CHANGE

This SPEER is required because the original part is no longer available. The change is the result of a design upgrade and the vendor has determined that the replacement part is interchangeable and will perform the same function.

SAFETY EVALUATION

The safety evaluation states that the filter assemblies and the associated RCP motors are not required to perform any safety related function during an accident. The motors are considered non-safety related and the replacement filter assembly performs the same function as the original. The evaluation documents that the addition of aluminum (less than one pound for each of the eight covers) in containment is insignificant compared to the amount currently in containment. The filter only provides a pressure boundary for the RCP motor lube oil and this pressure boundary is not a safety related function.

15. SPEER 9401321, Equivalency Evaluation for Replacement of the Boric Acid Makeup Header Check Valve - BAM-146 (Revision 0)

DESCRIPTION OF CHANGE

SPEER 9401321 provides for the replacement of the one inch Velan Piston Check Valve, BAM-146, Boric Acid Makeup Header Check Valve. The replacement valve is a one inch Anchor Darling Model 1878 Swing Check Valve.

REASON FOR CHANGE

Replacement is required because the original part is obsolete.

SAFETY EVALUATION

The safety evaluation determined that replacement of BAM-146 will not reduce the margin of safety as defined in the basis of any Technical Specification or safety analysis, and no unreviewed safety questions are created. The evaluation states that the replacement valve performs the same function as the original valve. The replacement valve is a higher pressure rated valve, and has a slightly higher flow coefficient. The replacement valve is constructed to the Class 1 of the ASME Design Code.

16. SPEER 9501376, Replacement of Reactor Hot Leg Injection Drain Valves SI-301 and SI-302

DESCRIPTION OF CHANGE

The SPEER replaces the original two inch, air operated globe valves with one inch, air operated double disc gate valves from a different manufacturer. The SPEER states that to address the gate valve pressure locking issue, NRC Generic Letter 95-07, the manufacturer drilled a hole through one disc of the valve. This will allow a vent path that will prevent the bonnet from becoming pressurized. The issue of thermal binding is not applicable.

REASON FOR CHANGE

The original Velan Model Bo8-4074X-13MN globe valves are obsolete and no longer manufactured. These valves are currently leaking and have a history of frequent leakage past the seat, requiring high maintenance. The leakage also causes frequent filling of the Safety Injection Tanks (SIT).

SAFETY EVALUATION

The accidents considered by the safety evaluation are hot leg injection on Loss of Coolant Accident (LOCA) and interfacing system LOCA (IS LOCA). These valves are normally closed, fail closed and receive a signal to close on a Safety Injection Actuation Signal (SIAS) The replacement valves are designed to close with differential pressures up to 2735 psi and will continue to be able to close against full RCS pressure in the event of an IS LOCA.

The replacement valves are equivalent to the original valves and the function of the valves remains the same. There is no change in the protective boundary by this SPEER. The SPEER does not reduce the margin of safety as defined by Technical Specifications or the FSAR.

17. SPEER 9501377, Replacement of the Safety Injection Tank Leakage Drain Valves -- SI-303A&B and SI-304A&B

DESCRIPTION OF CHANGE

The SPEER replaces the existing Safety Injection Tank (SIT) Leakage Drain valves, one inch Fisher globe valves, with one inch Anchor Darling double disc gate valves. The SPEER states that to address the gate valve pressure locking issue, NRC Generic Letter 95-07, the manufacturer drilled a hole through one disc of the valve. This will allow a vent path that will prevent the bonnet from becoming pressurized. The issue of thermal binding is not applicable.

REASON FOR CHANGE

The existing valves will not maintain proper seating for a full cycle. The valves are currently leaking and have a history of frequent leakage past the seat. Replacing the globe valves with gate valves will provide better isolation, less wear, and allow more flow.

SAFETY EVALUATION

The accidents considered by the safety evaluation are High Pressure Safety Injection (HPSI) Injection on Loss of Coolant Accident (LOCA) and interfacing system LOCA (IS LOCA). These valves are required to close to prevent diversion of safety injection flow from the Reactor Coolant System (RCS). The replacement valves have a higher pressure rating than the original valves and will continue to be able to close against full RCS pressure in the event of an IS LOCA.

The replacement valves are equivalent to the original valves and the function of the valves remains the same. There is no change in the protective boundary by this SPEER. The SPEER does not reduce the margin of safety as defined by Technical Specifications or the FSAR.

18. SPEER 9501378, Replacement of the Safety Injection Tank Drain Header to Containment Isolation Valve -- SI-343

DESCRIPTION OF CHANGE

The SPEER replaces the existing two inch Masoneilan globe valve with a one inch Anchor Darling double disc gate valve. The SPEER states that to address the gate valve pressure locking issue, NRC Generic Letter 95-07, the manufacturer drilled a hole through one disc of the valve. This will allow a vent path that will prevent the bonnet from becoming pressurized. The issue of thermal binding is not applicable.

REASON FOR CHANGE

The existing valve, a Masoneilan two inch globe valve, was installed during Refuel Outage 5. The valve is presently leaking and failed to pass the LLRT.

SAFETY EVALUATION

The safety evaluation states that the replacement valve is equivalent to the original. The replacement valve performs the same function as the original valve. There is no change in the protective boundary by this valve replacement. The component is normally closed and is fail closed. The replacement valve will close in 10 seconds as required by technical specifications. The replacement valve is designed to a higher pressure class than the original, it will close against a differential pressure up to 2735 psi. This is greater than RCS pressure or HPSI pump pressure. Thus, this change does not reduce the margin of safety as defined by any technical specification or the FSAR.

19. SPEER 9501463, Replace the 5kV MCM Aluminum Calvert Bus Duct Cable with 5kV 750 MCM Copper Cable (Revision 0)

DESCRIPTION OF CHANGE

This SPEER substitutes a 5kV, 750 MCM, shielded copper cable for a 5kV, 1250 MCM, shielded aluminum cable. This cable is located in the 5kV Calvert bus duct between Unit Auxiliary Transformer "A" and switchgear 3A2 cubicle No. 1.

REASON FOR CHANGE

The original cable was damaged in a fire on June 10, 1995. The original cable is not available within the required time frame.

SAFETY EVALUATION

According to the safety evaluation the SPEER alters information contained in the FSAR but it does not change a procedure, nor does it have the potential to alter, a procedure as described, outlined or summarized in the FSAR. The classification of the 5kV Calvert bus duct between Unit Auxiliary Transformer (UAT) "A" and switchgear 3A2 cubicle no. 1 is non-safety and non-seismic. This change is a substitution of parts which can not affect the preferred power source (offsite power via the Startup Transformer). In addition no credit is taken for the 5kV Calvert bus duct between UAT "A" and switchgear 3A2 cubicle no. 1 in the SAR. Thus, the SPEER does not increase the probability of occurrence of an accident previously evaluated in the SAR. The characteristics of the replacement and existing cables were evaluated in the SPEER and were determined to be equivalent. Therefore there is no change to any protective boundary.

20. Technical Requirements Manual Change Request 001

DESCRIPTION OF CHANGE

Change 001 to the Technical Requirements Manual (TRM) deletes the Limiting Condition for Operation (LCO) and the Surveillance Requirement (SR) for the computer room Halon System.

REASON FOR CHANGE

DC-3374 installed a new Plant Monitoring Computer which resulted in the Halon System being removed from service. The TRM requirements for the system are no longer required.

SAFETY EVALUATION

See DC-3374, Item I.A.17 of this report.

II. PLANT PROCEDURES

1. CE-002-001, Maintaining Steam Generator Chemistry (Revision 10)

DESCRIPTION OF CHANGE

Revision 10 of CE-002-001 updates the procedure to reflect the guidance contained in the EPRI PWR Secondary Water Chemistry Guidelines, Revision 3, May 1993. Significant change is the pH upper limit of 9.6 is revised to 9.8.

REASON FOR CHANGE

EPRI PWR Secondary Water Chemistry Guidelines recommend operation at higher hydrazine concentrations to increase the resistance of the Steam Generator tubes to Intergranular Attack/Stress Corrosion Cracking (IGA/SCC). This revision to CE-002-001 incorporates those guidelines.

SAFETY EVALUATION

According to the safety evaluation the Steam Generator Tube Rupture (SGTR) and Feedwater System Pipe Break are two events that may be affected by this revision. The evaluation determined that the overall effect of having a higher upper limit on Steam Generator pH will be to reduce the probability of SGTR resulting from IGA/SCC. The evaluation states that the procedure revision does not affect any protective boundary.

2. CE-002-002, Maintaining Condensate and Feedwater Chemistry (Change 1, Revision 7)

DESCRIPTION OF CHANGE

Change 1 to Revision 7 of CE-002-002 provides instructions for the addition of carbohydrazide as a supplemental oxygen scavenger to hydrazine in the secondary steam cycle.

REASON FOR CHANGE

Carbohydrazide is an oxygen scavenger which will provide more efficient removal of oxygen at temperatures less than 275 degrees F. Addition of carbohydrazide is expected to increase system integrity and reliability by reducing condensate oxygen which will minimize corrosion and reduce the amount of metal transport to the steam generators.

SAFETY EVALUATION

According to the safety evaluation there are no accidents affected by this procedure change. Carbohydrazide will supplement hydrazine in dissolved oxygen removal. Carbohydrazide is a volatile substance, therefore, its use will maintain an all volatile treatment as described in FSAR Section 10.3.5. Chemical addition into the secondary steam cycle is via the gland seal leakoff tank and is a provision of CE-002-001 (Item II.A.1 of this report) which has been previously evaluated and approved. This change only adds a new chemical to the procedure.

3. Design Engineering Procedures (NOECPs)

DESCRIPTION OF CHANGE

This is a generic 10CFR50.59 Safety Evaluation applicable to various Design Engineering procedures.

REASON FOR CHANGE

This generic safety evaluation was developed to address future procedure changes bounded by the safety evaluation, procedures affected by the evaluation address administrative processes controlled by Design Engineering. Changes to the affected procedures maintain commitments made in the Waterford 3 Quality Assurance Manual.

SAFETY EVALUATION

According to the safety evaluation procedures affected by the evaluation do not directly change plant equipment or the content of any operating procedure. Thus, no accident consequences are affected. Equipment or operating procedure changes would be addressed by a specific safety evaluation if required.

4. <u>EP-002-050, Offsite Dose Assessment "Manual" (Revision 14)</u>
<u>EP-002-060, Radiological Field Monitoring, (Revision 18)</u>
<u>EP-003-040, Emergency Equipment Inventory (Revision 16)</u>

DESCRIPTION OF CHANGE

Revisions to the above procedures address the replacement of the Ludlum 2218 survey instrument with the Ludlum 12 survey instrument. Additionally, the revision to EP-002-060 updates the "Vehicle Location Map" to more accurately describe the parking location for Emergency Vehicles.

REASON FOR CHANGE

Radiological Field Monitoring Team personnel are more familiar with the Ludlum 12 instrument, it is easier to use than the Ludlum 2218, and replacement equipment is available without delay (this is not true for the Ludlum 2218).

SAFETY EVALUATION

The safety evaluation states that use of the Ludlum 12 instrument has no effect on any accident or its consequences. The instrument is used to detect Iodine activity in the field during a Radiological Accident and this capability will not be changed by the use of this instrument. The Ludlum 2218 (instrument being replaced) has a high rate of malfunction. Replacing it will reduce the rate of malfunction without changing the safety and quality of Field Team monitoring activities. The new equipment will result in quicker results from the Field Monitoring Teams for Dose Assessment activities.

5. HP-100-109, Dosimetry Administration (Revision 13)

DESCRIPTION OF CHANGE

Revision 13 adds the definition of passive whole body counting and removes whole body counting as a part of in-processing. Also revises the procedure to allow visitors to enter the RCA without the use of thermoluminescent dosimeters (TLDs).

REASON FOR CHANGE

FSAR 12.5.2.2.3 states that personnel entering the Radiation Controlled Area (RCA) will be issued TLD badges. Revision 13 of HP-100-109 eliminates this requirement, it provides for visitors who do not enter a high radiation area, contaminated area, or airborne radioactivity area to be issued self reading dosimeters (SRD), not TLDs.

The revision adopts passive whole body counting, eliminating personnel whole body counts for in-processing and exiting the site. This will allow for less frequent use of the whole body counter and expedite personnel access to the RCA.

SAFETY EVALUATION

The safety evaluation states that this is an administrative procedure describing the Dosimetry Program, it does involve any plant systems and cannot increase the consequence of any accident previously evaluated in the SAR. The evaluation notes that the Dosimetry Program as described in the SAR will be revised to reflect this revision.

6. ME-001-011, Temporary Power from Temporary Diesel for 3A2 and 3B2 4kV Buses (Revision 0)

DESCRIPTION OF CHANGE

This procedure provides instructions for installation of a temporary diesel to the 3A2 and 3B2 4kV buses during refueling outages. This is a source of temporary power for these buses in the event of a loss of off-site power and of the safety-related diesel during the outage.

REASON FOR CHANGE

To provide an alternate source of power to the 3A2 and 3B2 4kV buses in the event of loss of offsite power and the safety-related diesel or bus failure. The alternate source of power will only be available during refueling outages and only during modes 5 and 6.

SAFETY EVALUATION

According to the safety evaluation the temporary power source will be connected to the load side of the respective Heater Drain Pump Motor breakers on buses 3A2 and 3B2. The motor leads will be disconnected but this will not affect the function of the Heater Drain Pumps because they are not required during modes 5 and 6 (the only modes that this procedure will be implemented). To prevent cross connecting the two buses administrative controls will be in effect to prevent closing both Heater Drain Pump Motor breakers at the same time. The evaluation states that there is no reduction in margin of safety as defined in Technical Specifications and no protective boundaries are affected.

7. MI-003-345, Calibration of Triaxial Response Spectrum Recorders Units SM IYR6040, 41, 42, 45 (L, T, V) (Revision 3)

DESCRIPTION OF CHANGE

Revision 3 removes calibration of the Peak Shock Annunciator Control Unit from the procedure. (This activity will be performed utilizing another maintenance procedure (MI-003-346).

REASON FOR CHANGE

MI-003-345 provides instruction for calibration of the Engdahl RSR1600-H/V(A) Response Spectrum Recorders and it is also used for trouble shooting suspected instrument malfunctions. Testing of the Peak Shock Annunciator Control Unit will be accomplished in accordance with maintenance procedure MI-003-346, Peak Shock Annunciator Control Unit Channel Calibration and Functional Test SM IYZ6045.

SAFETY EVALUATION

According to the safety evaluation there are no accidents affected by this procedure revision. It does not increase the probability of an accident nor the consequences of any accident. Testing of the Peak Shock Annunciator Control Unit will now be performed utilizing MI-003-346.

8. MM-003-043, Containment Spray Isolation Valve Inspection and Testing (Change 1 to Revision 1)

DESCRIPTION OF CHANGE

Change 1 to Revision 0 of MM-003-043 changes the setting of the relief valve used for testing, added Modes 1 - 3 as modes that the test may be performed in and added instructions to ensure the system would not spray down containment if the valve leaked by.

REASON FOR CHANGE

This test is normally accomplished during refueling outages, the change will allow the test to be done during operation if required. The differential pressure (dp) test demonstrates that the valve (CS-125A/B) still meets its design basis.

SAFETY EVALUATION

The safety evaluation states the hydrostatic pump used to raise the Containment Spray (CS) riser pressure for the purpose of the dp test is a low flow pump and would require more than 15 minutes to increase riser level tc the spray ring header level. Administrative controls and local instrument monitoring near the test rig will ensure that the riser level stays below the ring header. Thus the inadvertent CS accident is not anymore likely to occur than is usual during Section XI testing of the CS system.

The relief valve setting used for this test, 410 psig +25, -0, was evaluated and determined to be acceptable (EC-M95-002).

9. MM-008-002, Containment Penetration Modification for Refueling (Revision 0)

DESCRIPTION OF CHANGE

This is a new maintenance procedure which provides instructions for the temporary modification of containment penetrations #13 and #63 for refueling outage activities.

REASON FOR CHANGE

Containment penetrations #13 and #63 are utilized to provide temporary services to containment during refueling outages. Penetration #13 will be utilized for temporary cables and penetration #63 will be utilized to provide temporary service air to the containment.

SAFETY EVALUATION

According to the safety evaluation the procedure will not be implemented until the plant is in mode 5 or 6. Containment integrity will be maintained for penetration #13 by the use of a temporary seal plate which utilizes threaded pipe caps and will be filled with silicone RTV foam that will form a fire/air seal between the containment building and containment annulus. Penetration #63 will have administrative controls in place to isolate the penetration in the event the line loses air pressure, during loss of shutdown cooling or during fuel movement activities. The evaluation determined that there is no unreviewed safety question associated with the procedure.

10. NE-002-060, Isothermal Temperature Coefficient Measurement (Change 1, Revision 3)

DESCRIPTION OF CHANGE

The procedure measures the all rods out and bank B inserted Isothermal Temperature Coefficients and measures the Moderator Temperature Coefficient (MTC) during low power physics testing to allow comparison to the limits of Technical Specification 3.1.1.3.

REASON FOR CHANGE

The change will allow direct comparison of the MTC measured during startup physics testing extrapolated to 70% power to the limits of Technical Specification 3.1.1.b.

SAFETY EVALUATION

The safety evaluation determined that there are no unreviewed safety questions associated with the procedure change. There are no accidents listed in the SAR concerning the testing performed to determine the MTC during low power physics testing, additionally, the method for measuring the data is unchanged. Comparison to the 70% power Technical Specification limit is performed with the data measured by this procedure.

11. NOECP-001, Development, Revision, and Deletion of Procedures, Standards and Guides (Revision 3)

DESCRIPTION OF CHANGE

Revises NOECP-001 to allow for Design Engineering to perform the Quality Review for all NOECPs except for NOECP-001 and NOECP-310, "Design Verification," an engineering procedure that does not authorize work to be performed.

REASON FOR CHANGE

The revision transfers QA review requirements to Design Engineering to be consistent with changes in the QA Program Manual. Requirements of the review are not changed by this revision.

SAFETY EVALUATION

According to the safety evaluation the procedure revision does not directly change plant equipment. Any change to the facility would require a separate application of the 10CFR50.59 process. This is a revision to an administrative procedure that provides instruction for processing Design Engineering procedures.

12. NOECP-252, Steam Generator EDDY Current Inservice Testing (Revision 2)

DESCRIPTION OF CHANGE

NOECP-252 implements the administrative guidelines for steam generator inservice testing of tubing and tubesheet plugging. This revision affects a commitment (Letter W3A89-0196, dated October 23, 1989) concerning the process used to identify and mark tubes to be plugged. This revision utilizes remote robotics indexing and two independent verification steps.

REASON FOR CHANGE

In the commitment referenced above Waterford 3 committed to use the ECT Probe Method to accurately identify and locate tubes in the opposite plenum. Design Engineering has evaluated both tube location methods and has determined that the 3-step verification process is adequate for meeting the intent of the commitment. The 3-step verification process reduces radiation exposure and increases work efficiency by eliminating the steam generator platform worker who would otherwise be exposed to manway shine in the application of the paint sponge to the ECT probe poly cable for tubesheet marking.

SAFETY EVALUATION

The safety evaluation notes that the accident affected by this revision is "Steam Generator Tube Rupture." The evaluation determined that Steam Generator tube integrity is not affected by this alternate method in locating target tubes because the accuracy of locating tubes to be plugged is the same as used previously by NOECP-252.

According to the evaluation the method of locating tubes to be plugged does not affect the protective boundary. The actual plugging evaluation is covered by the Cycle Reload Analysis. The plug (once installed) becomes the new pressure boundary which has been evaluated under the Cycle Reload Analysis for ECCS.

13. NOECP-303, Design Change (Revision 7)

DESCRIPTION OF CHANGE

Revision 7 of NOECP-303 incorporates clarifying information in several sections of the procedure. The procedure provides instructions only for the preparation, review, approval, and revision of design changes.

REASON FOR CHANGE

Procedure revised was required to update the procedure to reflect signature requires for Plant Change cover sheets, correct editorial errors, and provide clarification for the Design Change Package Minor Revision process.

SAFETY EVALUATION

See Item II.A.3 of this report, "Design Engineering Procedures."

14. NOECP-402, NPIS Common Foundation Basemat Integrity Check (Revision 0)

DESCRIPTION OF CHANGE

NOECP-402 replaces procedure PE-005-033, Revision 5. NOECP-402 provides the same methods for verifying the Nuclear Plant Island Structure (NPIS) common foundation basemat integrity as stated in PE-005-033.

REASON FOR CHANGE

All responsibilities associated with the Basemat Monitoring Program have been assigned to the Design Engineering-Civil organization from the Plant Engineering organization. Design Engineering procedures (NOECP) will now be used to verify the basemat integrity.

SAFETY EVALUATION

The safety evaluation determined that there is no unreviewed safety question associated with NOECP-402. The procedure provides the methods to monitor the settlement of the basemat and changes in crack width of specified cracks. The procedure also states action limits associated with these measurements to ensure the conditions within the basemat do not change significantly. The action limits provided in letter W3P87-1123, which the licensing commitments are based on, will be maintained by NOECP-402. Therefore the margin of safety is maintained and this procedure will not reduce the margin of safety as defined in the bases for any technical specification or the appropriate safety analysis.

15. NOECP-403, Replacement/Repair of Safety and Non-safety Piping Components Due to Erosion/Corrosion (Revision 1)

DESCRIPTION OF CHANGE

Revision 1 of this Design Engineering procedure addresses replacement/repair of safety related piping. The revision also incorporates minor changes that were identified in Change 1 to Revision 0.

REASON FOR CHANGE

Revision 0 of NOECP-403 addressed the replacement/repair of non-safety related piping components, Revision 1 includes safety related piping.

SAFETY EVALUATION

The safety evaluation determined that there is no potential safety issue or unreviewed safety question associated with Revision 1 of NOECP-403. The revision does not increase the probability of moderate, infrequent or limiting accidents included in FSAR Table 15.0-2. Replacement of the safety and non-safety carbon steel piping components with similar corrosion resistant components of chrome-moly/stainless steel/clad piping have no significant impact on the overall structural integrity of the piping. It has been widely demonstrated that material containing chromium are resistant to FAC damage. Replacing carbon steel piping with chrome-moly (P22) or stainless steel (Type 403 or equivalent) should in most cases alleviate FAC damge for the life of the plant. Similar benefit is achieved by using clad pipe with a high chrome or stainless steel inner layer surrounded by carbon steel outer layer. Performing weld repair/build-up on safety related piping per ASME Section XI and non-safety piping per standard industry practice does not impact the structural integrity of the piping pressure boundary. Since the strength of the base material remains unaltered, the probability of a steam line break does not increase.

Modifications will only be approved after careful review, by Design Engineering, of the impact on the piping system. The revision therefore, does not affect the accidents evaluated in the FSAR and does not result in a challenge to the safety of the system or cause the system to be operated outside its design/test limits.

16. <u>OP-001-003, Reactor Coolant System Drain Down (Change 1, Revision 15)</u>

See Item II.A.35, RF-003-002, of this report.

17. OP-003-016, Instrument Air (Change 2, Revision 6)

DESCRIPTION OF CHANGE

Change 2 of Revision 6 to OP-003-016 addressed changes to the Standby Valve Lineup: 1) deleted valves that are abandoned in place, 2) changed the name of IA-3029 to indicate that it now supplies Instrument Air (IA) to DW-1641, 3) changed the Standby Lineup position of IA-8031 to Locked Close, and, 4) changed Standby Lineup position of upstream isolation valves for Containment Instrument Cabinets from open to closed. Attachments were added to position these valve during outages.

REASON FOR CHANGE

Items 1, 2, and 3 above were addressed by DC-3386 and/or LDCR-93-0190 (both were reported by the Report of Facility Changes, Tests and Experiments, W3F2-94-0051, dated October 20, 1994; items #44 and #107 respectively). Item 4, addressing the change of position for the upstream isolation valves for Containment Instrument Cabinets eliminates the leakage risk associated with these sections of seldom used Containment Instrument Air piping.

SAFETY EVALUATION

(NOTE: The safety evaluation is written only for item 4, change of position for the Containment Instrument Cabinets isolation valves, other items were covered as indicated in the above "Reason for Change" section.)

As stated in the safety evaluation this change will isolate that portion of IA piping between the normally closed isolation valve at the Containment Instrument Cabinets and the IA Header. No other equipment or components are affected by closing these valves. The portion of the IA Header in Containment that will be depressurized is only that portion that supplies the Containment Instrument Cabinets. The only function of the IA at the cabinets is to supply a motive force when calibrating instruments that have been removed from service. The cabinets will only have IA aligned to them when needed.

18. OP-005-001, Auxiliary Boiler (Change 2, Revision 8)

DESCRIPTION OF CHANGE

This change to OP-005-001 provides guidance for operation of a portable auxiliary boiler in the event the main Auxiliary Boiler is out of service.

REASON FOR CHANGE

OP-005-001 provides instructions necessary for fill and vent, startup, operation, and shutdown of the Auxiliary Boiler. The procedure did not provide sufficient guidance regarding hookups and station equipment lineups for operating a Portable Auxiliary Boiler.

SAFETY EVALUATION

According to the safety evaluation the Auxiliary Boiler system is not important to safety. With the exception of fuel source the Auxiliary Boiler system is designed to facilitate interconnection of a Portable Auxiliary Boiler. System function and operation will be unaffected by this change and the potential for a new accident not previously evaluated in the SAR is not affected by this change. The change does not affect a protective boundary, margin of safety, or accident response.

19. OP-007-005, Resin Waste Management (Change 1, Revision 8)

DESCRIPTION OF CHANGE

Change 1 to Revision 8 of OP-007-005 provides direction to fill the line leading to the Portable Solidification Unit with water prior to placing Spent Resin Transfer (SRT) on Short Cycle Recirculation. The change also adds instruction for flushing Spent Resin piping after Short Cycle Recirculation and to flush the SRT inlet strainer.

REASON FOR CHANGE

Change 1 was developed to address ALARA concerns and to enhance system operation. Amount of waste water produced during flushing operations is reduced and dose received by plant staff is reduced.

SAFETY EVALUATION

According to the safety evaluation no unreviewed safety questions were identified. A Radioactive Waste Systems Additional Safety Evaluation was also completed for this change. The safety evaluation notes that the change enhances the Radioactive Waste Management procedure and does not affect the seismic qualifications, the tank contents, nor the design operational modes which were assumed for accident analysis, thus the results of the analysis are unchanged.

20. OP-009-008, Safety Injection System (Change A, Revision 11)

DESCRIPTION

This deviation to OP-009-008 deletes steps which involve sampling the High Pressure Safety Injection (HPSI) loop, on long cycle recirculation, prior to filling the Safety Injection Tanks (SIT).

REASON FOR THE CHANGE

The deviation is of limited duration and allows filling the SITs while repairs are being affected to Safety Injection (SI) components.

SAFETY EVALUATION

According to the safety evaluation there is no unreviewed safety question associated with this procedure deviation. The safety evaluation notes that the deviation is of limited duration and is intended to allow filling SITs while repairs are being affected to SI components. This will ensure SIT levels are maintained within limits prescribed by the Technical Specifications. Safety evaluation calculations demonstrate the ability of the system to maintain boron concentrations within Technical Specification limits. If Technical Specification limits cannot be maintained the procedure deviation will not prevent the completion of the applicable action statement.

21. OP-100-009, Control of Valves and Breakers (Change 3 and Change 5 to Revision 12 and Revision 13)

DESCRIPTION OF CHANGE

Change 3 to Revision 12 of OP-100-009 changes the required position of FP-3363, adds new locked valves located in the Maintenance Support Building (MSB) and deletes valves for the previously existing trailers. The change also adds instructions pertaining to closing and backseating valves.

Change 5 identifies CAP-1021 and CAP-2041 as "locked open" instead of "normally open."

Revision 13 of OP-100-009 incorporates outstanding changes to Revision 12, it also updates the procedure to more accurately reflect plant conditions.

REASON FOR CHANGE

The procedure is enhanced by updating the Locked Valve and Locked Breaker Lists to reflect actual plant operating conditions. Guidance on closing and backseating of valves was not previously provided in the procedure. DC-3394, Installation of Equipment and Tie-ins for the MSB (Item I.A.27 of this report) resulted in the removal of several trailers from the protected area. FP-3362 is the trailer sprinkler system isolation valve and is now locked closed.

CAP-1021 and CAP-2041 are currently normally open valves and the valve position will remain unchanged. Designation as locked open provides a stricter means of maintaining the valves in the open position.

SAFETY EVALUATION

There were no unreviewed safety questions as a result of these changes. According to the safety evaluation there are no SAR related accidents affected by the procedure changes. The changes are to an administrative procedure and updates the procedure to current actual plant operating conditions.

The safety evaluation for Revision 13 of OP-100-009 notes that the procedure maintains FSAR commitments regarding containment isolation valves, Safety Injection Tank isolation valves, demineralized water system valves, ECCS valves needed for SDC, condensate polisher bypass valves, and atmospheric dump valves. The evaluation states that adding a manual valve to either the "Locked Valve List" or the "Inaccessible Locked Valve List" can neither affect current accident analyses, create new accidents, nor reduce margin of safety. Valves removed from OP-100-009 meet several rigorous conditions, i.e., they are non-safety related, they are irrelevant to actions needed to either add water to or control level in a steam generator, they do not prevent inadvertent off-site releases, and they cannot initiate a reactor trip. Thus, deleting valves from OP-100-009 as described in Revision 13 will not reduce the margin of safety defined in the Technical Specification Bases.

(See Item I.E.13, LDCR-95-0071, of this report)

22. <u>OP-100-014, Technical Specification Compliance (Revision 2, Change A and Change 1 to Revision 3)</u>

<u>DESCRIPTION OF CHANGE</u>

Revision 2 added steps to the procedure to refer to the Technical Specifications when an Emergency Diesel Generator is declared inoperable and to provide guidance on actions to be taken when declaring an Emergency Feedwater Isolation or Flow Control Valve inoperable. The revision also adds a new attachment to the procedure, "Nitrogen Safety Accumulators," to provide guidance when an accumulator is declared out of service.

Change A to Revision 3 of OP-100-014 allows operating the CCW system with one Dry Cooling Tower (DCT) tube bundle isolated from service for cleaning purposes.

Change 2 to Revision 3 adds instructions to OP-100-014 for compliance with TS Table 3.7-3. The instructions increase allowance for fouling in the Ultimate Heat Sink.

<u>REASON FOR CHANGE</u>

This revision of the procedure will assist operators in determining the operable status of various safety related equipment by providing better guidance on TS requirements. It also provides a course of action for various items which previously had no guidance or where the guidance was vague.

Cleaning of the DCT tube bundles will result in improved flow and performance of the DCT.

<u>SAFETY EVALUATION</u>

According to the safety evaluation the "leave-as-is" steps of Revision 2 of OP-100-014 do not affect the likelihood of either of the following accidents:

Main Steam Line Break
Feedwater Line Break
Loss of Normal Feedwater
Steam Generator Tube Rupture
any accident in which the Steam Bypass Control Valves are unavailable.

That is, neither the probability of pipe ruptures, nor Loss of Offsite Power, nor Main Feedwater system failures are a function of valve position upon becoming out-of-service.

The safety evaluation for Change A to Revision 3 of OP-100-014 states that currently Technical Specification (TS) 3.7.4 allows one DCT tube bundle to be isolated. However, no guidance or bases exist for operating in this configuration. Evaluations have been performed, taking into account current DCT conditions, to determine the effect of isolating one bundle of the DCT has on the CCW system. Although results of these evaluations indicate that CCW flow is reduced below design values, it was also determined that the DCT and associated CCW system could perform its safety related function TS 3.7.4 requires that during a tornado watch, all 6 DCT tube bundles and associated fans be restored to operable status in one hour, or be in hot standby in six hours. It was determined that during a tornado event, 5 operable DCT tube bundles under the missile shielding, are adequate to maintain plant operability for the first 24 hours or until shutdown cooling is initiated. Therefore no unreviewed safety question exist.

Effects of isolating a DCT tube bundle, in the current flow restricted condition, have been evaluated and are documented on engineering input sheets for CI-300039 for DCT "A" and CI-300040 for DCT "B." Evaluations were also performed in response to Corrective Action Program documents CR-94-417 and CR-95-0955.

23. OP-500-003, Annunciator Response Control Room Cabinet "C" (Change A, Revision 7)

DESCRIPTION OF CHANGE

The deviation to OP-500-003 deleted reference to annunciators referring to Unit Auxiliary Transformer (UAT) "A", UAT "A" A2/A1 bus feeders and breakers.

REASON FOR CHANGE

The deviation was written to comply with TAR-95-006.

SAFETY EVALUATION

See TAR-95-006, Item I.C.6 of this report.

24. OP-500-011, Control Room Cabinet M (Change 3, Revision 8)
OP-500-012, Annunciator Response Procedure Control Room Cabinet M (Change 2, Revision 10)

DESCRIPTION OF CHANGE

Setpoint for the Wet Cooling Tower Level Alarm was changed to reflect the implementation of DC-3426.

See DC-3426, Item I.A.40 of this report.

25. OP-901-230, Condenser Tube Leakage (Revision 1)

DESCRIPTION OF CHANGE

Revision 1 of off-normal procedure OP-901-230 provides guidance to isolate the waterbox before performing leak detection. The revision also adds new directions which are the result of changing the chemistry action level setpoints and actions.

REASON FOR CHANGE

OP-901-230 provides guidance for operator actions in the event of a condenser tube leak. The revision provides consistency with current Chemistry procedures.

SAFETY EVALUATION

The safety evaluation states that the evaluation is performed because the off-normal procedure number listed in the FSAR is different from the current number.

21. WA-01076104, Vibration Data Acquisition on the CDM Cooling Fans and Motors on Missile Shield (Revision 0)

DESCRIPTION OF CHANGE

This Special Test Procedure (STP) is to be used to acquire vibration data on the four Control Element Drive Mechanism (CEDM) Cooling fans and motors on the missile shield in the Reactor Containment Building (RCB).

REASON FOR CHANGE

Collection of vibration data on the CEDM fans and motors on the missile shield requires that each of the fans be run individually and in various combinations for approximately five minutes each. Historically these fans have experienced high vibration amplitudes with associated mechanical failures, i.e., bearings, etc.

SAFETY EVALUATION

The safety evaluation states that the STP will not increase the probability of occurrence of any of the accidents evaluated in the SAR. The plant will be in Mode 4 (Hot Shutdown) when all four fans are running. The fans are part of a non-safety related system. Only two fans are needed to maintain a negative pressure inside the CEDM cooling shroud, if an equipment malfunction occurs the data acquisition program will be terminated and at least two fans will be available for operation.

26. <u>OP-901-231, Abnormal Condensate or Feedwater Chemistry (Revision 1)</u>

See Item II.A.25, OP-901-230

27. OP-901-501, PMC or Core Operating Limit Supervisory System Inoperable (Change A, Revision 1)

DESCRIPTION OF CHANGE

Change A is a temporary deviation to OP-901-501 which provides an administrative control to use a more restrictive (than the current Technical Specification) Core Protection Calculator (CPC) DNBR limit when COLSS is out of service.

REASON FOR CHANGE

The change is a result of new Cycle 7 core parameters and is more restrictive than the existing Technical Specification 3.2.4c and complying with the new limit line is required to remain bounded by the Cycle 7 safety analyses. (Technical Specification Change Request NPF-38-144, letter W3F1-93-0309, was issued to Waterford 3 as Amendment 93.)

SAFETY EVALUATION

The safety evaluation states that the accidents that are potentially affected are several Anticipated Operation Occurrence (AOO) that are analyzed to determine the thermal margin requirement with COLSS out of service so that specified acceptable fuel design limits (DNBR) are not exceeded. Analyses performed by the reactor vendor (Combustion Engineering), using the more restrictive CPC DNBR limit line demonstrated that acceptable consequences are obtained. That is, the limiting AOO does not result in DNB with the revised limit line. The new CPC DNBR limit line preserves sufficient thermal margin so that fuel failure does not occur for events. Thus, this change does not reduce any margin of safety.

28. OP-903-077, Fire Protection System Valve Cycling Check (Change 1, Revision 5)

DESCRIPTION OF CHANGE

Change 1 to OP-903-077 removes Fire Protection system (FP) valve FP-3362 from the list of valves to be cycled.

REASON FOR CHANGE

FP-3362 was removed as a result of the implementation of DC-3463, Service Building Extension.

SAFETY EVALUATION

See DC-3463, Item I.A.59 of this report.

29. OP-903-115, Train "A" Integrated Emergency Diesel Generator/Engineering Safety Features Test (Revision 3)
OP-903-116, Train "B" Integrated Emergency Diesel Generator/Engineering Safety Features Test (Revision 3)

DESCRIPTION OF CHANGE

The surveillance procedures provide instructions for performing the Integrated Emergency Diesel Generator (EDG)/Engineering Safety Features Test. The surveillance satisfies Technical Specification Surveillance Requirements 4.8.1.1.2.d.1 through 4.8.1.1.2.d.12, and 4.8.1.1.2.b.

The revision adds a new section to satisfy the Technical Specification requirement to test the EDG ability to handle the rejection of the single largest load.

REASON FOR CHANGE

The revision corrects sections of the procedures which were found to be inadequate in their testing. The revision does not change the purpose of the procedure.

SAFETY EVALUATION

According to the safety evaluation there are no configuration changes to the systems caused by this procedure revision. The revision is to the method of testing of the EDG. The testing is performed in modes 5 or 6 and the systems to be tested are not required to be operable. The train not being tested will remain operable and will not be impacted by the test.

The evaluation identifies a load of >498 kW to satisfy the testing requirements for the single largest load and this load will be obtained by operating the Essential Chiller compressor and varying loads on MCC-315A to achieve the desired load (>498 kW). Dry and Wet Cooling Tower fans will be cycled to achieve the load. The test will not require the Cooling Tower fans to be operated in an abnormal manner.

30. <u>OP-TEM-005, Temporary Emergency Diesel Generator (Revision 0)</u>

<u>DESCRIPTION OF CHANGE</u>

This temporary operating procedure provides instruction for the operation of the temporary emergency diesel generator installed for the Refuel 7 outage.

See Item II.A.6, ME-001-011, of this report.

31. PE-004-016, High Pressure Safety Injection (HPSI) Control Valve Stem Test (Revision 2)

DESCRIPTION OF CHANGE

Revision 2 of PE-004-016 updates the list of references and adds procedural steps to allow performance of the test when the plant is in Mode 1.

REASON FOR CHANGE

References are updated to incorporate the latest guidance on allowable valve free stem-disc motion. The Engineering Procedure measures the stem to disc connection free play on the High Pressure Safety Injection (HPSI) flow control valves, SI-225A&B, SI-226A&B, SI-227A&B, and SI-228A&B which are equipped with motor operators. The procedure is also revised to allow testing of the valves while tne plant is in Mode 1.

SAFETY EVALUATION

According to the safety evaluation the following accidents may be affected by the procedure revision: Main Steam Line Rupture, CEA Ejection, Inadvertent Safety Injection Actuation Signal (SIAS) at Power, Loss of Coolant (LOCA), Steam Generator Tube Rupture (SGTR), Inadvertent Opening of Pressurizer Safety, and Letdown/Primary Instrument Line Break. The evaluation states that the procedure starts and stops the HPSI pumps and operates (manually) the HPSI flow control valves which are designed to open automatically upon receipt of an SIAS. During power operation a Core Protection Calculator (CPC) low pressure trip ensures that the Reactor Coolant System (RCS) pressure is >1785 psia which exceeds cutoff head of the SI pumps. Therefore, opening the valves at power during the test will not cause injection of emergency cooling fluid in the RCS. This is the same reasoning used in the SAR to conclude that an "Inadvertent SIAS at Power" would not inject emergency cooling fluid in the RCS, thus resulting in no adverse affect on the plant. Safety Injection outlet check valves ensure flow in not injected in the Safety Injection Tanks.

The evaluation also notes that manual operation of the valves will not compromise the ability of the valves to perform their safety function because the valve design is such that receipt of an SIAS at power will disengage the valve handwheel and the valve will be positioned electrically to the desired position.

As noted by the safety evaluation the seismic qualification of the HPSI flow control valves will not be altered by the measuring device to be used for the test. Dead load stress due to the additional weight (<10 pounds) of the measuring device is considered negligible and non-consequential. The HPSI flow control valves are not considered part of the RCS pressure boundary, the FSAR concludes that the RCS pressure boundary for the HPSI cold leg injection lines does not extend to the HPSI flow control valves.

32. PE-004-018, Temporary Air Conditioning (Revision 2)

DESCRIPTION OF CHANGE

PE-004-018 provides instructions for installation, operation, and removal of temporary air conditioning for the Reactor Containment Building (RCB) and the Fuel Handling Building (FHB). Revision 2 adds a new section addressing slow speed operation of the Containment Fan Coolers during modes 5 and 6 without ESFAS actuation. The revision also adds provisions for construction of leakage containment of Component Cooling Water (CCW). (CCW activity was discussed in Items 110 and 111 of the Waterford 3 "Report of Facility Changes, Tests and Experiments," submitted by letter W3F2-94-0051, dated October 20, 1994.)

REASON FOR CHANGE

To reduce noise levels in the RCB during Modes 5 and 6 the procedure revision allows the CFCs to be operated at slow speed during these modes. Because of the activity of the CCW water (referenced above) provisions have been added to the procedure to contain any spillage of CCW water to allow sampling and proper disposal of the water.

SAFETY EVALUATION

Both the safety evaluation and the Radioactive Waste Systems Additional (RWSA) Safety Evaluation concluded that no unreviewed safety questions existed because of Revision 2 of PE-004-018. The RWSA evaluation was performed because of the activity of the CCW (referenced above).

According to the RWSA evaluation the measures established by PE-004-018, Revision 2, construction of the spillage containment, Health Physics (HP) surveys, and pressure testing of the temporary piping will preclude exceeding any 10CFR20 limits. HP calculation HP-CALC-93-014 determined the concentrations of radionuclides expected to be present within the CFC portion of the temporary air conditioning system.

The safety evaluation determined that operation of the CFCs on slow speed has no affect on accidents listed in the SAR. CFCs are designed to operate during normal operations and under Loss of Coolant Accidents (LOCA) or Main Steam Line Break (MSLB) conditions. These conditions are analyzed for modes 1 - 4 and the procedure revision only allows performance of this section during modes 5 and 6.

33. P'.G-009-014, Conduct of Planned Outages (Revision 2, Change 1)

DESCRIPTION OF CHANGE

Change 1 to Revision 2 of PLG-009-014 revises "suggested protected train posting" and "suggested operable train" postings to better reflect actual postings, lowers amount of volume to maintain in the Reactor Water Storage Pool (RWSP) during mid-loop operations and adds a definition to clarify application of "protected train" designation to electrical components. It also changes requirements on Component Cooling (CC) pump to "available" from "operable' with regards to Shutdown Cooling (SDC) operability.

REASON FOR CHANGE

PLG-009-014 is an administrative procedure that provides; 1) guidelines for the planning and conduct of planned outages, 2) guidance for establishing and maintaining safe plant conditions during the shutdown and, 3) establish how to review planned outage schedules and reporting of the review results. The procedure does not control plant configuration or direct equipment operation.

As a part of this procedure activity FSAR Section 9.2.2.2.1 was determined to require revision which resulted in the completion of the safety evaluation.

SAFETY EVALUATION

The safety evaluation states that accidents as described in the FSAR will continue at the same frequency as currently described, the revision to PLG-009-014 does not affect those initiator frequencies. The procedure does not control plant configuration or direct equipment operation.

The analysis related to reduced inventory operation appears in FSAR 9.6.3.4. The analysis determines the time to core uncovery by assuming that the only water available is in the RCS already. Although the time to core uncovery calculated does not depend on injecting additional water, this analysis does require operators to put the plant in a safe shutdown condition sometime before core uncovery occurs. Under many circumstances one CC loop is sufficient to keep safety-related loads properly supplied. One loop can be put into a planned outage when these conditions exist and only when the plant is in mode 5 or 6. A pump failure in the operating loop can be compensated for by starting the third pump, or during periods in the outage by relying on the heat capacity of a full reactor cavity and a full fuel storage pool.

34. <u>RF-003-002, Steam Generator Primary Side Services (Revision 4, Change 1 to Revision 5)</u>

DESCRIPTION OF CHANGE

Revision 4 of RF-003-002 incorporates the results of Calculation EC-S93-011. Calculation EC-S93-011 demonstrates that Steam Generator nozzle dams can be installed at seven (7) days or later after shutdown. Change 1 to Revision 5 changed the time for nozzle dam installation from 5 days to 4 days after shutdown.

REASON FOR CHANGE

Revision 4 of RF-003-002 provides for earlier installation of all steam generator nozzle dams during refueling outages.

SAFETY EVALUATION

According to the safety evaluation reactor decay heat levels will be slightly higher with the 7 day criteria than for the previous 9 day requirement. Engineering calculation EC-M88-012 demonstrated that decay heat at 9 days is 89.6% of the decay at 7 days. The difference in decay heat level, and associated plant heat loads, does not measurably impact the probability of occurrence of an accident provided the plant is operated in accordance with Technical Specifications and design requirements.

Limiting restrictions which controls when all nozzle dams can be installed is that the time before core uncovery must be equal to or greater than 60 minutes when worst case Reactor Coolant System (RCS) pressurization is considered. Revision 4 of RF-003-002 meets this criteria with the additional assumption that an operator in containment is assumed to release the water in one Safety Injection Tank (SIT) to the RCS in no more than 15 minutes after Shutdown Cooling (SDC) is lost. EC-S93-011 demonstrated the times required for the RCS to heat to bulk boiling conditions, with a midloop initial level and without accounting for adding the SIT inventory, at 7 days after shutdown as:

RCS initial temperature (indicated) 135 degrees F.	15.3 minutes
RCS initial temperature (indicated) 130 degrees F.	16.3 minutes
RCS initial temperature (indicated) 125 degrees F.	17.4 minutes
RCS initial temperature (indicated) 120 degrees F.	18.4 minutes

These times demonstrate the critical nature of the less than 15 minute assumption for operator action to release the SIT inventory to the RCS. With one SIT added to the RCS, EC-S93-011 predicts a core uncovery time of 184 minutes, easily exceeding the 60 minute requirement. The safety evaluation concluded that there is no unreviewed safety question associated with this procedure revision and that margin of safety is maintained.

(Additional information may be found in Waterford 3 letters W3C1-93-0039, dated October 1993 and W3C1-94-008, dated February 17, 1994.)

The safety evaluation for Change 1 to Revision 5 states that the change does not reduce the margin of safety. Since containment closure time of 1.5 hours is not reduced and the existing criteria of 1 hour to core uncovery is maintained for the day earlier nozzle dam installation, the margin of safety remains unchanged. Installation of nozzle dams at an earlier date was analyzed with the reduced Reactor Coolant System temperature of 110 degrees F. and will not result in core uncovery.

35. UNT-005-013, Fire Protection Program (Revision 4), (Revision 5)

DESCRIPTION OF CHANGE

Revision 4 of UNT-005-013 incorporates fire detection instrumentation installed as part of DC-3268 (Item I.A.10 of this report)

Revision 5 of UNT-005-013 deletes operability, surveillance, and compensatory action requirements related to fire protection systems and components.

REASON FOR CHANGE

UNT-005-013 specifies the administrative controls and equipment operability and surveillance requirements associated with the Fire Protection Program. Revision 4 incorporates the necessary surveillance and operability requirements associated with DC-3268, Fire Detection System Upgrade. Revision 5 removes the surveillance and operability requirements from the UNT because the requirements have been relocated to the Waterford 3 Technical Requirements Manual (TRM).

SAFETY EVALUATION

The safety evaluation states that the operation of the fire protection system and its associated response to a fire event are unchanged by this revision. Equipment necessary to respond to this form of accident remains the same and continues to be in compliance with applicable code and commitment. In that the regulatory requirements and code compliance requirements as previously presented and approved in Section 9.5.1 of the FSAR remain in effect or are enhanced by this revision, the margin of safely is maintained. Accident response to a fire both from a programmatic and systems perspective remain consistent with the previously approved program.

The safety evaluation for Revision 5 states that the approved Fire Protection Program is maintained, information deleted from the procedure by Revision 5 is relocated in the Waterford 3 TRM.

36. UNT-TEM-008, Release of Condensate to Low Volume Waste Ponds (Revision 0)

DESCRIPTION OF CHANGE

This temporary procedure provides controls and requirements for release of condensate containing low level tritium to the Low Volume Waste Ponds for subsequent release to the Mississippi River.

REASON FOR CHANGE

This temporary procedure is for periods of maintenance with the reactor shutdown and allows for release of excess condensate from the Condenser Hotwells with low levels of tritium contamination. Current procedures do not contain this provision.

SAFETY EVALUATION

According to the safety evaluation the only effect from performing this procedure is a reduction of the available doses (10CFR100) for each accident scenario. Since the calculated doses from this evolution are extremely low (approximately 1.9E-8 mrem), there is a negligible effect to all FSAR accident scenarios.

A "Radioactive Waste Systems Additional Safety Evaluation" was also performed for this temporary procedure. This evaluation notes that the condensate system is not normally a contaminated system but has a potential for contamination. Also stated in the evaluation is that UNT-TEM-008 provides controls for an effluent release from the condensate system containing low level tritium activity only. Calculations and estimated doses for this evolution are contained in Health Physics calculation HP-CALC-94-001. The calculated tritium activity in the Low Volume Waste Ponds will be well below the MPC of 3.0E-3 micro Curies/ml.

37. W2.301, Identification, Evaluation, and Reporting Process for 10CFR21 Compliance (Revision 2)

DESCRIPTION OF CHANGE

Revision 2 of W2.301 incorporates the assignment of Part 21 responsibilities to the Waterford 3 Licensing Department, this was originally assigned to the Shift Technical Advisors (STA). The revision also updates the procedure to include the current Condition Report process and other editorial enhancements.

REASON FOR CHANGE

Site Directive W2.301 implements the requirements for performance of 10CFR21 evaluations. This responsibility has been re-assigned to the Site Licensing Department. The revision also updates the procedure to delete redundant steps and delete duplication of work.

SAFETY EVALUATION

The safety evaluation determined that the procedure revision does not affect any accidents listed in the FSAR and that no equipment is affected by the revision. Implementation of the requirements of Part 21 remain unchanged by the revision.

B. SPECIAL TEST PROCEDURES (STP)

1. STP-01115211, Dynamic Test of Emergency Boration Valves (Revision 0)

DESCRIPTION OF CHANGE

The STP performs differential pressure stroke tests of emergency boration valves, BAM-113 A&B, BAM-133, and CVC-507, to demonstrate that the valves will operate under design basis conditions.

REASON FOR CHANGE

The STP will demonstrate that the emergency boration valves will operate under design basis conditions per NRC Generic Letter 89-10 requirements.

SAFETY EVALUATION

The safety evaluation indicates that there are no accidents affected by this STP, the test will be performed during Modes 5 or 6, during a Boric Acid Management (BAM) system outage. Boration of the Reactor Coolant System (RCS), if required, can be accomplished by the Refueling Water Storage Pool via the Safety Injection (SI) System. This SI path is totally different from that via the Charging System used in this test. Equipment affected by the STP are the valves being stroke tested and they will be operated within their design basis. The STP will demonstrate that the system will perform as designed in the worst case scenario. It will confirm, rather than reduce, the margin of safety of the system.

2. STP-01117251, HPSI Injection Line Normal Operating Pressure Test (Revision 0)

DESCRIPTION OF CHANGE

The STP will perform the required pressure test of the lines and components specified in the Waterford 3 "10 Year Inservice Inspection Program." The STP performs the pressure test of the High Pressure Safety Injection (HPSI) System.

REASON FOR CHANGE

The STP performs a nominal pressure test of the HPSI header in lieu of a hydrostatic test as allowed by Code Case N-498 of the ASME Boiler and Pressure Vessel Code. Per Code Case N-498, Normal Operating Pressure (NOP) testing can replace hydrostatic testing for Class 2 systems.

SAFETY EVALUATION

The STP operates the HPSI system in an abnormal lineup, however, the safety evaluation determined there was no unreviewed safety question. The evaluation identifies a Loss of Coolant Accident (LOCA), specifically an Inter System LOCA (ISLOCA) that may be affected by this STP. The evaluation determined that the STP lineup is similar to that used to verify operability of the HPSI pumps by recirculation to the Reactor Water Storage Pool (RWSP) through the hot leg injection with the exception that the "A" side Cold Leg Injection Flow Control Valves will also be open. HPSI operability is conducted for up to 1 hour at least 3 times each quarter. As such, placing the Safety Injection (SI) system in this configuration for an additional 6 hours once every 10 years does not represent a discernible increase in the time spent in this configuration. If an ISLOCA occurred, SI-301, HPSI Hot Leg Injection Drain, and/or SI-343, SI Tank Drain to RWSP Containment Isolation, would be shut to isolate the leak path to the RWSP. SI-301 will auto close on a Safety Injection Actuation Signal (SIAS) and SI-343 will auto close on a Containment Isolation Actuation Signal (CIAS). A LOCA from the Cold Leg Injection path would require failure of two in-line check valves. A double failure is not credible. In the FSAR analysis for ECCS ability to meet it's functional requirements following a single failure, check valve failures are not considered credible failures.

The evaluation identifies HPSI Pump "A", and valves SI-241, SI-242, SI-243, SI-244, SI-301, SI-343, and SI-510 as equipment important to safety that could be affected by the STP. However, the evaluation concludes that the equipment will be operated in the manner it has been designed for and at an acceptable frequency. Therefore, the likelihood of an equipment malfunction is not increased.

The evaluation also addresses the interaction of significant HPSI flow with the minimum flow recirculation path of Low Pressure Safety Injection (LPSI) Pump "A" and

Containment Spray (CS) Pump "A." The minimum flow recirculation path of CS Pump "A" does not perform a safety function, and is only required for testing. Thus, the interaction of HPSI flow with the CS minimum flow recirculation path does not create the possibility of a different type of CS failure. PEIR 86000 evaluated the LPSI interaction and determined that the LPSI pump minimum flow recirculation path would still perform its safety function despite the new HPSI flow path interaction.

Containment isolation valves SI-343 and SI-344 are opened during the performance of the STP. Should the containment safety function be required during performance of the STP, a CIAS will be generated which will close SI-343. SI-301 will also close, due to SIAS, to further isolate inside containment. Neither the connected piping nor SI-301 are tested as containment isolations, although the components are designed for 1750 psia. This condition exists when lined up to reduce pressure in the safety injection hot leg flow path, and has previously been determined to be acceptable in the FSAR. Therefore, the containment will adequately perform its safety function, and the margin of safety associated with the 10CFR100 limits for off-site exposure remain unchanged.

3. STP-WA-001126854, Component Cooling Water System Bleed to Waste Condensate Tank

DESCRIPTION OF CHANGE

The STP provides instructions for the bleed of Component Cooling Water (CCW) to a Waste Condensate Tank (WCT) through a temporary hose connection into the Liquid Waste Management System (LWM).

REASON FOR CHANGE

The STP provides instructions to bleed CCW to a WCT for the purpose of reducing chloride levels in the CCW system.

SAFETY EVALUATION

According to the safety evaluation there is no unreviewed safety question associated with this STP. A Radioactive Waste Systems Additional Safety Evaluation was also developed for this STP. The evaluation notes that the CCW system will utilize a fitting on a vendor portable demineralizer skid to connect the CCW and LWM systems. The STP does not alter the WCT, LWM nor any pressure retaining components.

As stated in the evaluation the bleed rate of CCW to the WCT will be the capacity of the pump used for the bleed activity, 15 gpm. CCW Make-up pumps have a design capacity of 600 gpm, assuring that the volume of the CCW system will be maintained. Any CCW system water collected in the WCT will be discharged in accordance with plant procedures currently used for discharge of the WCT contents.

4. STP-01127424, Test of Boric Acid Make-up Flow

DESCRIPTION OF CHANGE

Utilizing M&TE qualified test instrumentation, the STP will be used to obtain flow data while adding boric acid and water, in accordance with existing Operations procedure OP-002-005, to the Volume Control Tank (VCT).

REASON FOR CHANGE

The STP is to determine the reason for reduced boric acid flow through BAM-146 and associated piping which may occur during blended fills of the VCT.

SAFETY EVALUATION

The safety evaluation did not identify any unreviewed safety questions associated with the STP. The Boric Acid Make-up (BAM) system will be operated in accordance with existing Operations procedure, OP-002-005, to add boric acid and water to the VCT.

The evaluation identifies the Boron Dilution Accident, Uncontrolled CEA Withdrawal at Power and Control Element Assembly (CEA) Ejection as accidents that may be affected by the STP. The evaluation acknowledges the potential for dilution of the VCT during this STP but notes that the operators are aware of this potential and directed to take appropriate actions to prevent boron dilution through reactor power reduction and/or direct boration to the Reactor Coolant System, using existing Operations procedures. Because the STP does not involve changes to systems or procedures used to mitigate such an accident there is no increase in the consequences for such an accident. To avoid violating assumptions and conclusions in the safety analysis for Uncontrolled CEA Withdrawal at power, (i.e., single failure in the CEDMCS), the CEDMCS will not be enabled during performance of the STP. To ensure that the conclusions of FSAR Chapter 15 remain valid as related to the CEA ejection event reactor power will be at or below 100% measured during this evolution.

5. STP-01127968A, Load Shedding and Automatic Starting of 4KV 3AB3-S Components (Revision 0)

DESCRIPTION OF CHANGE

The STP is performed to make Component Cooling Water (CCW) Pump AB operable and partially test the operability of High Pressure Safety Injection (HPSI) Pump AB, as allowed in Amendment No. 98 , dated September 16, 1994, to the Facility Operating License, NPF-38. The STP has two parts. First the 4KV emergency bus 3AB3-S will be de-energized to verify that HPSI Pump AB, CCW Pump AB and Essential Chiller Pump AB will trip on a loss of voltage. Next the CCW Pump AB will be reloaded onto 3AB3-S by actuating its Safety Injection Actuation Signal (SIAS) relay and the "B" Sequencer in test.

REASON FOR CHANGE

Technical Specifications 4.8.1.1.2 d(3 and 5) requires, in part, (a.) verifying deenergization of the emergency busses and load shedding from the emergency busses, and (b.) "...verifying that the Emergency Diesel Generator (EDG) energizes the auto-connected shutdown loads through the load Sequencer ...". Condition Report 94-854 was initiated on September 7, 1994, to identify that the 4KV and 480 volt AB components are apparently not included in the 18 month EDG integrated tests used to satisfy the surveillance requirements of Technical Specifications 4.8.1.1.2.d.

The STP will:

1. Verify that CCW Pump AB, HPSI Pump AB and Essential Chiller AB load shed from the 3AB3-S 4KV Bus during a simulated loss-of-off-site power (LOOP) event. This includes support components for CCW Pump AB and HPSI Pump AB. Essential Chiller AB support components will be tested in another STP.

The test will also verify that subject breakers trip during a LOOP with SIAS condition. The 4KV AB undervoltage relays that generate the LOOP trip signal will function the same regardless of the SIAS.

2. Verify that the CCW Pump AB 4KV breaker and support components automatically close at load block 3 (7 seconds) of the EDG "B" Sequencer while simulating a LOOP in conjunction with a SIAS.

The same step of the STP will also verify that the CCW Pump AB breaker will close during a LOOP without SIAS condition. The relays that generate the LOOP with SIAS also provide the LOOP signal without SIAS and were previously tested per OP-903-116, Train "B" Integrated Emergency Diesel Generator/Engineering Safety Features Test.

Successful completion of the STP will satisfy the surveillance requirements of Technical Specifications 4.8.1.1.2.d (3 and 5) for CCW Pump AB per Amendment 98. And partially satisfy the requirements for HPSI Pump AB and Essential Chiller AB by verifying that they shed from emergency bus 3AB3-S.

SAFETY EVALUATION

The safety evaluation indicates that there is no unreviewed safety question. The normal Train "A" and "B" components will act in their normal capacity as the mitigating systems when bus 3AB3-S is de-energized. Undervoltage on 3AB3-S will not directly initiate any transient in the plant

The accident in the SAR that may have radiological release consequences altered by the proposed test is the Loss of Coolant Accident (LOCA) with a LOOP. This is due to the abnormal condition of CCW during the test. The CCW Pump AB will be in the test position and CCW Pump "B" will have its emergency start disabled. A LOCA with LOOP along with an assumed single failure on CCW Train "A" would result in a loss of all CCW. The STP procedure cautions that if an accident occurs which trips CCW Pump B, the pump must be restored to service via the AB assignment switch prior to Sequencer load block 3 (the "7 second" load block). This will prevent a total loss of CCW and therefore prevent the radiological consequences of a LOCA from being altered.

During the STP the fuses in the 480 Volt 3B31-S to 3AB31-S tie breaker trip circuit will be removed. Removing these fuses will not alter the effect of an electrical bus fault on 3AB31-S because the trip circuit is not part of the tie breaker overcurrent scheme. Overcurrent protection is provided by an Electronic Current Sensor in the tie breaker which mechanically opens the tie breaker independently of the trip circuit. Therefore an electrical fault on 3AB31-S will not affect 3B31-S.

The consequences of a malfunction of CCW "A" or EDG "A" could potentially be increased since CCW "B" and CCW AB will be inoperable during 60 minutes of the test. However, if a LOOP occurs, CCW "B" will be restored to service via the AB assignment switch prior to Sequencer load block 3. Therefore it will be sequenced onto EDG "B" as designed and the consequences of a malfunction of CCW "A" or EDG "A" are unchanged.

The new type of accident that must be considered is a LOCA with LOOP and a Single Failure of EDG "A" or CCW Train "A." This will result in a complete loss of CCW since, for 60 minutes, the test will disable the other two trains. This is actually not a new type of accident since CCW Pump "B" will be restored to service via the AB assignment switch prior to Sequencer load block 3 and will be loaded onto EDG "B" as designed.

The time during which the CCW Pump AB will be aligned to replace CCW Pump "B," disabling CCW Pump B's emergency start will be kept as brief as possible and less than 60 minutes. The probability of a LOCA with LOOP occurring within this 60 minutes is very low -- approximately 4x10E-9.

During the STP HPSI Pump AB and Essential Chiller Pump AB breakers will be in the test position which will make the pumps physically unable to start during an accident. These pumps are currently not selected and they are not required during an accident since the HPSI Pumps "A" and "B" and Essential Chiller Pumps "A" and "B" are operable and selected. CCW Pump AB breaker will also be in test making CCW Pump AB inoperable. At the same time, CCW Pump AB will be selected to replace CCW Pump "B" which will disable the emergency start of CCW Pump "B." This leaves only one CCW train able to respond to an accident, CCW "A," and a Single Failure must be assumed on EDG "A" or CCW train "A." The STP provides administrative controls which negate the possible increased consequences of an accident. CCW "B" will be running during the test and the STP cautions the operator to restore CCW Pump "B" to service via the AB assignment switch prior to Sequencer load block 3 if a LOOP occurs during the test.

The margins of safety to be considered are fuel design limits, containment peak pressure, RCS maximum pressure, and off site dose. The STP will not effect cladding chemically. Peak clad temperature and other fuel design limits are not sensitive to CCW availability during early phases of Main Steam Line Break (MSLB) and LOCA. The margins all remain the same as long as the administrative controls contained in the STP are followed. These controls require that CCW Pump "B" be restored to service via the AB assignment switch prior to Sequencer load block 3 if an accident were to occur during the STP.

6. STP-01127968B, Load Shedding and Automatic Starting of 4KV and 480V AB Components (Revision 0)

DESCRIPTION OF CHANGE

The STP is performed to make the following equipment operable, as allowed in Amendment No. 98 , dated September 16, 1994, to the Facility Operating License, NPF-38: High Pressure Safety Injection (HPSI) Pump AB, Essential Chiller AB, Charging Pump AB and the Plant Monitoring Computer (PMC) Static Uninterruptable Power Supply (SUPS) bypass feeder. The STP demonstrates that both HPSI Pump AB and Essential Chiller AB will reload automatically onto 4KV bus 3AB3-S upon Loss of Off-site Power (LOOP) and Safety Injection Actuation Signal (SIAS), by actuating their SIAS relays and sequencing the "B" Sequencer in test. The ability of Charging Pump AB and PMC SUPS bypass to automatically trip is demonstrated by actuation of the undervoltage relays in their trip circuits. The STP will also demonstrate the ability of Charging Pump AB to reload onto 480V 3AB31-S by: (a) actuating its undervoltage relays and (b) actuating its SIAS relay and concurrently sequencing the "B" Sequencer in test.

REASON FOR CHANGE

Technical Specifications 4.8.1.1.2.d(3 and 5) requires, in part, (a.) verifying deenergization of the emergency busses and load shedding from the emergency busses, and (b.) "...verifying that the Emergency Diesel Generator (EDG) energizes the auto-connected shutdown loads through the load Sequencer ...". Condition Report 94-854 was initiated on September 7, 1994, to identify that the 4KV and 480 volt AB components are apparently not included in the 18 month EDG integrated tests used to satisfy the surveillance requirements of Technical Specifications 4.8.1.1.2.d. The STP will only prove acceptability of the "AB" components supplied from switchgear 3AB3-S and 3AB31-S provided these buses are connected to switchgear 3B3-S and 3B31-S, respectively.

The STP will verify that:

1. Charging Pump AB and the Computer Secondary Feeder load shed from the 3AB31-S 480V bus during a simulated Loss of Off-site Power (LOOP) event, including support components for Charging Pump AB.

   Also that subject breakers and support equipment trip during a LOOP with Safety Injection Actuation Signal (SIAS) condition. The undervoltage relays that generate the LOOP trip signal will function the same regardless of whether an SIAS is present.

2. The following AB components automatically start while simulating a LOOP in conjunction with an SIAS. Other Design Basis Accidents and features verified are as follows:

4KV AB Components

HPSI Pump AB and support equipment automatically start when the 1.5 second indicating light of the EDG "B" Sequencer illuminates.

HPSI Pump B will not start when the 1.5 second indicating light of the EDG "B" Sequencer illuminates while HPSI Pump AB is substituted.

HPSI Pump AB will not start with a LOOP without SIAS.

Essential Chiller AB and support equipment automatically starts after the 168 second indicating light of the EDG "B" Sequencer illuminates.

Essential Chiller B will not start after the 168 second indicating light of the EDG "B" Sequencer illuminates while Essential Chiller AB is substituted.

LOOP in conjunction with an SIAS will also verify that Essential Chiller AB and support equipment will start during a LOOP without SIAS. The undervoltage relays that generate the LOOP signal will function the same regardless of whether an SIAS is present.

480V AB Components

Charging Pump AB and support equipment automatically starts when the 17 second indicating light of the EDG "B" Sequencer illuminates.

Charging Pump B does not start and receives a lock-out signal when the 17 second indicating light of the EDG "B" Sequencer illuminates while Charging Pump AB is substituted.

Charging Pump AB and support equipment will start during a LOOP without SIAS condition.

Completion of the STP will satisfy the surveillance requirements of technical specifications 4.8.1.1.2.d.(3 and 5) for HPSI Pump AB, Essential Chiller AB, Charging Pump AB, and the Computer Secondary Feeder as per Amendment No. 98, dated September 16, 1994, to Facility Operating License, NPF-38.

SAFETY EVALUATION

According to the safety evaluation there is no unreviewed safety question associated with this STP.

The safety evaluation notes that the proposed test has four parts:

I. The 480V breakers for Charging Pump AB and PMC SUPS bypass feeder will be racked into the test position and closed. An undervoltage condition will be inserted to demonstrate that the breakers will trip on undervoltage.

II. HPSI Pump AB will be aligned to replace HPSI Pump "B" (HPSI Pump AB will supply HPSI Train "B"). The HPSI "B" SIAS relay will be actuated and the "B" Sequencer will be sequenced in test to verify that HPSI Pump AB will reload onto bus 3AB3-S following an accident.

III. Essential Chiller AB will be aligned to replace Essential Chiller "B." The Essential Chiller "B" SIAS relay will be actuated and the "B" Sequencer will be sequenced in test to verify that Essential Chiller AB will reload onto bus 3AB3-S following an accident.

IV. Charging Pump AB, with its breaker racked into the test position, will be assigned to replace Charging Pump "B." The Charging Pump "B" SIAS relay will be actuated and the "B" Sequencer will be sequenced in test to verify that Charging Pump AB will reload onto bus 3AB31-S following an accident.

The safety evaluation states that the above operations will not increase the probability of initiating any of the accidents evaluated in the SAR.

The accidents in the SAR that must be considered for a possible change in consequences are Loss Of Off-site Power (LOOP), Loss of Feedwater, SG Tube Rupture (SGTR), Loss of Coolant Accident (LOCA) and Steam Line Break. During those parts of the test where the AB components are selected to replace the "B" components the emergency start of the "B" components will be disabled. Since the AB components are currently inoperable, only the "A" component is available for an emergency start and it must be assumed to experience a single failure

In actuality, HPSI "B" and Essential Chiller "B" will be available for an emergency start until the instant that the AB assignment switch is used to replace the "B" component. Immediately after the assignment, the HPSI Pump AB or Essential Chiller AB will be available for an emergency start since the test will be performed with the breaker racked into the operating position. Although the HPSI Pump AB and Essential Chiller AB have been declared inoperable, they have not failed and are expected to perform properly under emergency conditions. Combustion Engineering Owners Group Revised Standard Technical Specifications LCO 3.0.5 establishes allowance for restoring equipment to service under administrative controls when it has been declared inoperable, for the purpose of demonstrating the operability of the equipment being returned to service. The test procedure cautions that if an accident occurs during the test which sequences the Sequencer, Charging Pump "B" must be restored to service via the AB assignment switch and possibly the Charging Pump "B" control switch prior to the Sequencer "17 second" load block.

Because there will be two trains of HPSI, Essential Chillers, and Charging available throughout the test, no radiological release consequences will be altered.

The consequences of a malfunction of Charging Pump "A" or EDG "A" could potentially be increased since Charging Pumps AB and "B" will be inoperable during 120 minutes of the test. However, if a LOOP or actual SIAS occurs, Charging Pump "B" will be restored to service prior to the Sequencer "17 second" load block. Therefore it will be sequenced onto EDG "B" as designed and the consequences of a malfunction of Charging Pump "A" or EDG "A" are unchanged.

The safety evaluation discusses a new accident to considered is a LOCA, Steam Line Break or LOOP and a Single Failure of EDG "A" or Charging Pump "A." This will result in a complete loss of charging since, for 120 minutes, the test will disable the other two pumps. This is actually not a new type of accident since Charging Pump "B" will be restored to service prior to the Sequencer "17 second" load block and will be loaded onto EDG "B" as designed.

The time during which Charging Pump AB will be aligned to replace Charging Pump "B" will be kept as brief as possible and less than 120 minutes. The probability of a LOCA, Steam Line Break or LOOP occurring within this 120 minutes is very low, approximately 9X10E-6.

The margins of safety to be considered are fuel design limits, containment peak pressure, RCS maximum pressure, and off-site dose. The margins all remain the same as long as the administrative controls contained in the test procedure are followed. These controls require that Charging Pump "B" be restored to service prior to the Sequencer "17 second" load block if an accident were to occur during the test.

7. STP-01128262, Biocide Addition to the Fire Protection System (Revision 1)

DESCRIPTION OF CHANGE

The STP provides for a biological flush of the Fire Protection (FP) System

REASON FOR CHANGE

Intent of the STP is to enhance the FP System operation through the elimination of corrosion caused by Microbiological Induced Corrosion (MIC).

SAFETY EVALUATION

The safety evaluation states that the chemical treatment of the water does not affect the FP System's ability to respond to a fire event. It also notes that the consequences of a fire event remain unchanged from that previously analyzed in the FSAR. According to the safety evaluation the margin of safety and the plant's ability to achieve and maintain safe shutdown following a fire event are maintained consistent to that previously analyzed in the FSAR and supporting documentation related to the FP Program.

8. STP-01130277, CCW Make Up Pump Flow Test (Revision 0)

DESCRIPTION OF CHANGE

The Special Test Procedure will collect performance data on the Component Cooling Water (CCW) Make-up Pumps. Flow rate will be measured while discharging the Condensate Storage Pool (CSP) to Circulating Water to determine pump capacity.

REASON FOR CHANGE

Testing of the CCW Make-up pumps requires that the CCW M/U system be operated in a configuration not identified in the FSAR.

SAFETY EVALUATION

Leaks in the Emergency Diesel Generator Jacket Water Systems, Component Cooling Water System, and Chilled Water System are occurrences that may be affected by this STP. However, leaks in any of these systems are not expected to be common to both trains and are bounded by accident analyses which consider any failure such as leakage as a general failure of an entire train or system. CCW M/U pumps are not considered to have an active safety function, but are considered to be "available" to provide make-up when need. The lack of availability of CCW M/U pumps for a brief period of time does not cause any other interaction except to deny the capability of make-up from the CCW M/U pumps to the above systems.

9. STP-01135284, Carbohydrazide Addition to Condenser Hotwell C2

DESCRIPTION OF CHANGE

This STP work authorization (WA) provides for the injection of carbohydrazide at Condenser Hotwell C2 through a test connection at the northeast corner of the hotwell.

REASON FOR CHANGE

The STP is an attempt to enhance chemistry of the condensate by reducing dissolved oxygen. Chemistry control of the secondary cycle will be maintained by existing procedures. (See CE-002-001, Item II.A.1 and CE-002-002, Item II.A.2 of this report.)

SAFETY EVALUATION

According to the safety evaluation the Loss of Condenser Vacuum event will not be affected by this STP because of the use of a check valve at the test connection. Atmospheric pressure will seat the check valve in the event of a complete failure of the feed tubing.

Equipment important to safety is not affected by the STP, the Hotwell is not a safety related component. The use of carbohydrazide and its effect on the secondary steam cycle from a chemistry standpoint has been satisfactorily evaluated by a safety evaluation for CE-002-002, Change 1, Revision 7. (Item II.A.2 of this report.)

10. STP-WA-01136748, Pressure Test of SI Hot Leg 2 Class 1 Piping (Revision 0)
STP-WA-01136749, Pressure Test of SI Hot Leg 1 Class 1 Piping (Revision 0)

DESCRIPTION OF CHANGE

The STPs satisfy ASME Section XI ten year hydrostatic testing criteria by application of ASME Code Case N-498. This code case allows Class 1 lines to be examined at reactor coolant system nominal operating pressure rather than hydrostatic test pressure.

REASON FOR CHANGE

The testing will be performed during Mode 3 with the Reactor Coolant System (RCS) at 2250 psia, by using a hydrostatic test pump connected to the Class 2 portion of the Safety Injection (SI) Hot Leg Line in the -35 RCA Wing Area. Using Primary Make-up (PMU) as a test medium, the pressure in the SI Hot Leg will be elevated by the test pump to approximate the RCS nominal operating pressure of 2250 psia in the Class 1 portion of the hot leg. A VT-2 visual examination will then be performed.

SAFETY EVALUATION

According to the safety evaluation the STP creates a previously unanalyzed dilution mechanism by the addition of PMU by a test pump through the SI hot leg injection line. The addition of PMU to the RCS in itself is not an unusual evolution. An Inadvertent Boron Dilution is evaluated in the FSAR with the assumptions of a charging pump discharging 44 gpm PMU with the core shutdown margin at minimum value. Under these conditions the boron dilution alarm is expected to give the operators 15 minutes warning time prior to a loss of shutdown margin.

Prior to the STP the RCS boron concentration will be determined and if necessary the boron concentration will be raised such that the operation of the test pump at full flow (approximately 7 gpm) for one hour will not cause a RCS boron concentration to fall below shutdown margin. During the STP the boron dilution monitors will be in operation and an operator, stationed at the test rig, will be in continual contact with the control room. Thus, the likelihood of an inadvertent RCS dilution due to performance of this STP is much less likely than the accident analyzed for in the FSAR.

Following completion of the STP the SI Hot Leg would be depressurized and potentially filled with PMU, approximately 48 gallons. If this amount of PMU were injected into the RCS at one time the boron concentration change would be approximately 1.5 ppm at BOL conditions. Thus, the small amount of PMU that may be left in the SI Hot Leg will not be a significant safety concern.

11. STP-01139656, Chemical Control of Zebra Mussels in Circulating Water Systems (Revision 0)

DESCRIPTION OF CHANGE

This STP involves the injection of a chemical (Betz Clam-Trol CT-2) into the Circulating Water System (CWS) to eradicate Zebra mussel infestation. The STP also provides instructions for the use of bentonite clay for the deactivation of the chemical. System tie-ins necessary for the chemical treatment are also included in the STP.

REASON FOR CHANGE

The purpose of this STP is to control the growth of Zebra mussels in the CWS and to eradicate current Zebra mussel infestation in the system. The STP implements the addition of a biocide to the CWS system at the intake structure and deactivates on the outlet side of waterboxes A2 and B2. No radioactive system will be manipulated by the STP.

SAFETY EVALUATION

According to the safety evaluation there no accidents affected by the STP. CWS flow rates or temperatures will not be altered and overall system performance and reliability is maintained. Potential for pluggage of tubes exist as mussels die off. The vast number of condenser tubes in relation to the amount of Zebra mussel infestation observed to date demonstrates that the probability of completely plugging condenser tubes such that condenser vacuum is lost is not increased. Even in the worst case scenario whereby pluggage of condenser tubes with Zebra mussels results in loss of condenser vacuum, safety and/or atmospheric dump valves can be used to remove residual heat from the Reactor Coolant System. Injection rates (approximately 150 gpm) of CT-2 and bentonite clay are negligible in relation to the overall CWS flow. Injection may be suspended at any time plant conditions warrant. There is no safety related equipment involved in the performance of this STP.

12. STP-01140557, CCW System Flow Balance Test

DESCRIPTION OF CHANGE

The STP will align the Component Cooling Water (CCW) system to its accident lineup to determine that each safety related component receives the proper flow during accident conditions.

REASON FOR CHANGE

The STP will verify that the components cooled by CCW during an accident receive the proper design flow. Equipment affected by the STP is as follows:

a. Containment Fan Coolers (CFC)
b. Emergency Diesel Generators (EDG)
c. Shutdown Heat Exchangers (SDHX)
d. CCW Pump A, B, and AB suction and discharge cross connect valves
e. Essential Chillers
f. High Pressure Safety Injection Pumps (HPSI)
g. Containment Spray Pumps (CS)
h. Low Pressure Safety Injection Pumps (LPSI)
i. CCW nonessential loads

SAFETY EVALUATION

The safety evaluation states that the CCW system, as part of the Ultimate Heat Sink, is required to mitigate the consequences of a Loss of Coolant Accident (LOCA), a Main Steam Line Break (MSLB), or a Main Feedwater Line Break (MFLB) in containment by rejecting heat from containment. CCW is also required to supply cooling water to t he Essential Services loop (CFC, EDG, SDHX, Essential Chillers, HPSI and LPSI pumps and CS pumps). The STP aligns the CCW system to its accident (Safety Injection Actuation Signal (SIAS) or Containment Spray Action Signal (CSAS)) lineup. The STP will be conducted in Modes 5 or 6 and procedural administrative controls will prevent operating the system outside of its design limits. No new interconnections to other systems will be created by the STP.

Fuel Pool temperature will be monitored frequently during the performance of the STP to ensure that design limits are not exceeded. Reactor Coolant Pumps, CEDMs and Letdown will not be in service and the nonessential loop will be isolated (condition assumed in an accident situation).

The STP tests each CCW train separately, should any of the equipment fail, the opposite train will be available to perform the safety function. The STP does not change any protective boundaries and the CCW system will be operated within its design limits.

13. STP-01142286, Pressure Closing of HPSI Header Check Valve SI-243

DESCRIPTION OF CHANGE

This STP establishes a differential pressure across High Pressure Safety Injection (HPSI) cold leg 2A check valve SI-243 to seat the valve.

REASON FOR CHANGE

Safety Injection Tank (SIT) 2A leakage will be reduced by tightly seating SI-243 (normal position). This will be accomplished by aligning HPSI pump discharge pressure via the SIT drain header to the downstream side of the valve and venting the upstream side of the valve.

SAFETY EVALUATION

According to the safety evaluation the Loss of Coolant Accident and Main Steam Line Break accident could be affected by the STP. However, the evaluation states that neither accident will be affected because all the equipment used in the test will be operated within its design limits, Reactor Coolant System pressure will remain greater than the Safety Injection System pressure during the test. Air operated drain valves operated by the STP will automatically close on a Safety Injection Actuation Signal and the manual drain valve required to be opened for the STP will be closed by an operator who will be in close proximity to the valve during performance of the STP.

The evaluation notes that there are no unreviewed safety questions associated with the STP and there is no reduction in the margin of safety because of the STP.

14. STP-99003337, Special Test Procedure for Testing Drain Traps Installed by DC-3337

DESCRIPTION OF CHANGE

This STP provides the accpetance test for drain valves which were installed in the Fire Protection system by DC-3337 (Item I.A.12 of this report).

REASON FOR CHANGE

The STP will verify the functionability of the drain valves.

SAFETY EVALUATION

The safety evaluation states that no new failure modes or system operations or interactions are introduced by the STP. No direct effect is posed to equipment important to safety during a fire. The STP will ensure that system maintenance is better ensured by the installation of the drain valves.

15. STP-99003354, Acceptance Test of DC-3354

DESCRIPTION OF CHANGE

The STP provides acceptance testing for the modifications performed by DC-3354 (Item I.A.13 of this report). The DC installed a modified liquid level transmitter and relocated the High Pressure (HP) leg tap for the Spent Resin Tank (SRT), it also provided a new short cycle recirculation line and a clean water flush line.

REASON FOR CHANGE

The STP will confirm that the changes installed by the DC results in an operable system.

SAFETY EVALUATION

The safety evaluation states that no new accident possibilities are created by the STP and there are no new system interconnections or interactions created. The STP does not affect any boundaries or margins of safety.

16. STP-99003379, Corrosion Product Transport Monitoring (DC-3379) Acceptance Test.

DESCRIPTION OF CHANGE

The STP performs acceptance testing of the Secondary Metal Transport Monitors installed by DC-3379 (Item I.A. 20 of this report).

REASON FOR CHANGE

The STP provides instructions for testing flow rates, pressures, and temperatures at the transport equipment. No other facility changes are required to implement the STP. Testing of the corrosion monitoring equipment is accomplished by establishing Feedwater, Blowdown, Main Steam, Component Cooling Water, and Turbine Closed Cooling Water flows to the equipment.

SAFETY EVALUATION

Loss of Normal Feedwater is the accident identified in the safety evaluation that may be affected by the STP. However, the evaluation states that Feedwater flow thru the monitoring equipment will 1400 to 1500 ml/min and this flowrate is neglibile when compared to the total feedwater flowrate. Corrosion monitoring equipment will be isolated and the STP terminated if Blowdown or Main Steam radiation monitors go into alarm during the test.

The evaluation states that there are no protective boundaries affected by implementing the STP. The physical connections of the corrosion monitoring equipment to its respective system were made under DC-3379. The STP does not make any new sytem interactions. Flow rates to the equipment are insufficient to degrade system functions and integrity of the sytems is maintained.

17. STP-99003389, Alternate Chemical Addition for Secondary System Acceptance Test (DC-3389) (Revision 0)

DESCRIPTION OF CHANGE

This Special Test Procedure provides for acceptance testing of the chemical addition skid installed by DC-3389 (Item I.A.26 of this report).

REASON FOR CHANGE

The STP will verify proper operation of the chemical addition pumps, mixing tanks, calibration columns, and pressure gauges associated with DC-3389. It will also verify minimum flow rate capability of the chemical addition pumps.

SAFETY EVALUATION

The safety evaluation determined that there is no unreviewed safety question associated with the STP. No postulated accidents are affected by the testing of the alternate chemical addition skids. Testing will be performed using demineralized water (no chemicals) thus, there will be no chemical excursions of the secondary cycle. The highest flow rate of the pumps on the addition skid is 5.5 gph. Each pump will be individually tested, therefore, this additional flow rate to the secondary cycle will be negligible. The STP does not involve a protective boundary.

18. STP-99003426, Acceptance Test for DC-3426

DESCRIPTION OF CHANGE

DC-3426 (Item I.A.40 of this report) installed chemical addition and filtration capabilities for the Wet Cooling Towers (WCT). This STP verifies that the DC-3426 installation functions as designed.

REASON FOR CHANGE

To verify that the WCT Basin Filtration Pumps will trip on low suction pressure, low discharge pressure, high discharge pressure and low basin water level. Filtration pump flow rate and vibration will be verified to be acceptable and flow through each corrosion monitoring coupon rack will be verified. The test will also verify that chemicals can be transferred to the chemical addition tanks and injected into the basins.

SAFETY EVALUATION

The safety evaluation states that the Auxiliary Component Cooling Water (ACCW) system is required to mitigate the effects of a Loss of Coolant Accident (LOCA) or a Main Steam Line Break (MSLB). The STP will verify that the WCT Filtration system functions as designed. Therefore the test will not increase the likelihood of a LOCA or MSLB. The non-safety filtration system is seismically designed and includes siphon breakers to prevent draining the basin below Technical Specification limits. During testing activities personnel will be available to secure the filtration pump should a leak exist and the siphon breakers do not function as planned.

19. STP-99003430, Acceptance Test for DC-3430 (Plant Change) Replacement Valve Assembly for Letdown Heat Exchanger Temperature Control Valve (CC-636)

DESCRIPTION OF CHANGE

This acceptance test will determine if CC-636, Letdown Heat Exchanger Temperature Control Valve, functions properly as described in the FSAR.

REASON FOR CHANGE

Acceptance test is required because the valve and actuator were replaced by DC-3430, Item I.A.42 of this report.

SAFETY EVALUATION

According to the safety evaluation the section of the test performed in Mode 1 will be in accordance with Waterford 3 Operating Procedure OP-002-005. Valve manipulations in accordance with the STP will be performed only in Modes 5 or 6 with the Letdown system secured. Testing the ability of CC-636 to close on loss of Instrument Air or closure of CVC-103 and CVC-109 is performed in Modes 5 or 6 to eliminate any possible adverse effects on equipment important to safety. Letdown is not required for safe shutdown and the letdown heat exchanger has no specific requirement to function for post-accident operation.

There are no new system interactions or connections created by this STP. The STP will not result in a change to a protective boundary and no margins of safety are reduced.

20. STP-99003459, Acceptance Test For DC-3459, Loss of Remote Shutdown Capability During Control Room Fire (Revision 0)

DESCRIPTION OF CHANGE

The STP will demonstrate the acceptability of DC-3459, Item I.A.56 of this report.

REASON FOR CHANGE

The STP will operate SI-407A&B with deadman switches and lifted leads and the test is not described in the FSAR. The STP will be performed while the equipment is out of service.

SAFETY EVALUATION

According to the safety evaluation the test will be performed during Modes 5 or 6, with the equipment out of service and there are no accidents affected by the test. The test confirms that DC-3459 prevents a control room fire from compromising the integrity of SI-407A&B.