ENTERGY

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Waterford 3

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U.S. Nuclear Regulatory Commission
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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 19, 1992 through May 31, 1994.

If you have any questions regarding this report, please contact G.E. Wilson, at (504)739-6657.

Very truly yours,

R.F. Burski

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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 - 1994

ENTERGY OPERATIONS, INC.
WATERFORD 3
10CFR50.59 REPORT
JUNE 19, 1992 THROUGH MAY 31, 1994

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I. FACILITY CHANGES

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Waterford 3 SES

1994 Report of Facility Changes, Tests and Experiments

SUMMARY

This report provides the Waterford 3 Facility Changes made pursuant to 10CFR50.59(a)(1). The report covers the period from June 19, 1992 through May 31, 1994. None of the items in this report represent an unreviewed safety question. No experiments not described in the FSAR were conducted during the report period.

The report identifies 124 Facility Changes (49 Design Changes, 14 Condition Identification/Work Authorizations, 15 Temporary Alterations, 6 Document Revision Notices, 25 License Document Change Request, and 15 Miscellaneous Evaluations) and 41 Procedure evaluations (28 Plant procedures and 13 Special Test Procedures).

A. Design Changes (DCs)

1. DC-3026, Gaseous Waste Management System (GWMS) Compressor Replacement

Description of Change

This change reports the installation of two new waste gas compressors. All piping and conduit was to be reconfigured in the compressor cubicles. New moisture removal equipment was installed upstream of the compressors to prevent water damage to the compressors.

Reason for Change

The change was made to improve equipment maintenance and reliability.

Safety Evaluation

According to the safety evaluation, all equipment is designed in accordance with Regulatory Guide 1.143. The GWMS is not required for safe shutdown of the plant or to mitigate an accident. No change in the way the GWMS is operated is made by the design change. FSAR Section 15.7.3.1, "Radwaste Waste Gas System Leak or Failure," remains unchanged, with the failure of the Gas Decay Tank as the limiting fault.

2. DC-3031 Spent Resin System Enhancement (Revision 0)

Description of Change

The DC upgrades the existing manually operated plug valve to a pneumatically actuated pinch valve with an auto/manual controller and a leak detection device. It also adds the capability to monitor the differential pressure across the Spent Resin Transfer Pump.

Reason for Change

Operational problems due to inadequate pump pressure indication and clogging of the existing discharge throttle valve.

Safety Evaluation

According to the safety evaluation no unreviewed safety question exists. This change is non-safety and enhances the operation of the Spent Resin Transfer System by upgrading certain existing components. No changes to original configuration or operations are incorporated. No Technical Specification requirements exist.

3. DC-3033 Installation Of Mass Flow Probes ON 11 PIG Monitors

Desc ption of Change

DC-3033 describes the replacement of vacuum switches on the Particulate Iodine Gas (PIG) radiation monitors with Kurz mass flow meters. Two of the affected monitors are safety related (Plant Stack PIG Monitors).

Reason for Change

The vacuum switches previously used on these monitors for flow control were difficult to maintain, and prone to drift out of calibration. This replacement increased the accuracy and sensitivity of the flow control.

Safety Evaluation

There o change in the design function or operation of the PIGs associated with this change, nor are there any new system interactions created. Therefore, the safety evaluation concludes that there is no unreviewed safety question.

Note: The "Miscellaneous" Safety Evaluation for "Firmware Change Package FCP-91-01," reported in the 1991 Annual 50.59 Report, W3F2-91-0039, is related to this item.

4. DC-3055, Safety Bus for Chemistry Lab

Description of Change

DC-3055 details providing a new electrical feeder to supply LP-3003-NA from MCC-3AB311-S. MCC-3AB311-S can be fed from either Emergency Diesel Generator (EDG).

Reason for Change

During a loss of offsite power (LOOP), the chemistry department could not perform the chemical analyses required to support plant operations.

Safety Evaluation

The design change provided a new feed for the chemistry lab using two breakers for double isolation from the emergency bus in accordance with FSAR Section 8.3.1.2.18 and Regulatory Guide 1.75-1975. The calculation for the EDG loading sequence shows that with the addition of this load, the EDG loading remains below 100% of its continuous rating. Because the FSAR and Regulatory Guide design requirements are met, there are no equipment malfunction concerns. Similarly, the margin of safety as defined in TS Sections 4.4.6 and 3/4.8 will not be reduced.

5. DC-3082 (Revision 0), Boric Acid Make-up Tanks Sample Lines

Description of Change

The DC re-routes the sample lines for the Boric Acid Make-up (BAM) Tanks, "A" and "B," to an existing sample sink located outside the tank rooms. The change also includes a flush line from nearby condensate system piping to the sample sink. This line will be used to flush the BAM tank fluid drained from the line prior to sampling, into the radwaste system through the sink drain.

Reason for Change

Present BAM Tanks "A" and "B" sample lines are very inaccessible and difficult to use. Routing sample lines outside the room will reduce the time required for collecting samples and also the radiation exposures for the personnel collecting samples.

Safety Evaluation

The safety evaluation concluded that no unreviewed safety question exists. The DC does not change the procedures as described in the FSAR or the function of the BAM tank sampling system. The equipment added also meets the original design requirements (ASME III, Class 3, and Seismic Category I).

6. DC-3126, Main Turbine Generator Monitor / Protection Instrumentation (Revision 0)

Description of Change

The purpose of this design change is to provide on-line monitoring of vibration at the main generator stator end turn windings (twelve locations) and to provide an active shaft grounding system that protects the turbine generator shaft, bearings and seals from electrical damage. Also, the generator condition monitor (GCM) is relocated from the feedwater pump turbine A and B cabinet to a new generator monitoring/protection cabinet.

Reason for Change

The new instrumentation was added to improve the monitoring and protective capability of the turbine generator. The system of generator stator end turn winding vibration monitoring is needed to predict high vibration, which could result in possible damage due to winding and support structure failure. Pitting of the generator rotor bearings was revealed during refueling outages, which indicates signs of electrolysis. An active shaft grounding system will aid in the elimination of shaft voltages that might cause undue wear on the bearing surfaces. The GCM is being relocated because the present location does not permit ease of accessibility for calibration and maintenance.

Safety Evaluation

The affected instrumentation is non-safety-related and is associated with the turbine generator. The equipment is not required to function in the event of an accident. Therefore, the probability or consequence of an accident previously evaluated in the FSAR will not be increased.

The possibility of an accident which is different than any already evaluated in the FSAR will not be created because the instrumentation will not initiate any new plant perturbations or introduce new failure modes to safety-related equipment.

7. DC-3127, NPDES Discharge Point Modification (Revision 0)

Description of Change

The DC will allow adequate sampling of the effluent discharged from the Yard Oil/Water Separator through a new recirculation line and butterfly valves. Sampling can be conducted while ensuring unacceptable effluent is not discharged. In addition a new, more efficient weir will be installed to prevent discharge of oily waste.

Reason for Change

Recirculation of the Yard Oil/Water Separator sump is required to ensure that oily waste and sludge is not discharged.

Safety Evaluation

The safety evaluation concludes that the DC affects only a non-safety related system, located outside of the Turbine Building and that it does not affect any protective boundary or margin of safety.

8. DC-3150, Corrosion Rate Monitoring of TCCW Heat Exchanger (Revision 0)

Description of Change

The DC installs a Corrosion Deposit Test Unit (CDTU) and a Corrosion Test Loop Assembly (CTLA) in the Turbine Cooling Closed Water (TCCW) System. The CDTU will be used to monitor corrosion rates in the tube side (river water) of the TCCW Heat Exchanger. The CTLA will be used to monitor corrosion rates in the shell side of the TCCW Heat Exchanger.

Reason for Change

The Corrosion Rate Monitoring system will be used to determine if significant corrosion exists on the tube and shell side of the TCCW Heat Exhangers. Information obtained may be used to evaluate the corrosive attack on engineering materials and to develop recommendations to the inhibitor treatment program to maintain the TCCW System integrity.

Safety Evaluation

As indicated in the safety evaluation the DC does not affect any safety related equipment, will not affect the quality or operability of any equipment important to safety and does not affect or reduce any margins of safety.

9. DC-3155, Vibration & Loose Parts Monitoring System (V&LPMS) Replacement/Enhancement (Revision 2)

Description of Change

The DC replaces the internals of the existing V&LPMS cabinet with a system having enhanced capabilities which will meet the requirements of R.G. 1.133/NUREG-0737, eliminate the excessive false alarm occurrences and restore operator confidence in the system reliability. Associated field cabling, in-containment sensors and associated hardware remain in place. The functions of the system will be equal or better than the functions of the original system.

Reason for Change

Due to the occurrence of an excessive number of false alarms and the lack of spare parts, Waterford 3 has experienced considerable difficulty in maintaining the operational integrity of the current system.

Safety Evaluation

As stated in the safety evaluation, the V&LPMS has no safety function. No safety boundary is affected by the V&LPMS or this DC. No margin of safety is affected, therefore, mitigation of an accident or condition/prevention of radiological releases are not affected by this DC.

10. DC-3187, Enhancement to the Sewage Treatment System Inside the Protected Area (Revision 0)

Description of Change

This DC will convert the existing Sewage Treatment Unit into a new sewage lift station. All sanitary sewage generated inside the Protected Area will discharge to an existing Lift Station, #3, which discharges to the local Sanitary Waste System.

Reason for Change

The design capacity of the Sewage Treatment Unit is routinely exceeded, the solid wastes or "sludge" which collects in the bottom must be removed by a contractor approximately every six months. During refueling outages the load on the Sewage Treatment Unit increases to the point where the sludge must be removed virtually every week. The DC will eliminate these problems with the Sewage Treatment Unit.

Safety Evaluation

As indicated in the safety evaluation the Sanitary System is non-safety related and has no interfaces with safety related equipment. The modification does not reduce the margin of safety as defined in the basis for any technical specification or safety analysis. The DC does not introduce any new unreviewed safety questions.

11. DC-3191, Civil Defense Radio Relocation (Revision 0)

Description of Change

The DC will relocate the State Civil Defense Radio (SCDR) from its present location (+46 RAB) to LP&L's Labadieville District microwave tower site. Communications with the SCDR shall be established by the addition of a new microwave channel from the Emergency Offsite Facility to Labadieville.

Reason for Change

The SCDR provides a third alternate means of plant-to-offsite emergency communication capability during normal, fire, and accident conditions. The DC will provide for a more reliable and a higher degree of quality communication with the Civil Defense organizations.

Safety Evaluation

The safety evaluation results determined that the DC will not impact the operation or function of any equipment important to safety that is described in the safety analysis report and that there are no unreviewed safety questions. No Technical Specification changes are required.

12. DC-3195, Safety Injection Sump Outlet Valves SI-602A & B (Revision 3)

Description of Change

The DC changes valves SI-602A & B, from air operated valves to motor operated valves. The DC eliminates the present safety related air accumulators that provide a limited source of motive air for post accident valve operation. It also removes the manually connected nitrogen backup accumulators added during Refuel 3. The control circuit logic for the new motor operators allows the same quarterly ASME XI surveillance testing of the valves. The requirements for the valves to close on an SIAS and open with a RAS are included in the control circuit logic for the new valve operators.

Reason for Change

Waterford 3 Licensee Event Report (LER-89-007-00, May 1, 1989) reported concerns on the size of the Instrument Air (IA) accumulators for the Safety Injection Sump Outlet valves (SI-602A & B). It was determined that the original design criteria for sizing the accumlators did not consider the limiting accident, i.e., a small break LOCA coincident with a loss of IA. Manual operation of the valves was not considered an adequate backup due to potential high radiation levels at the valve location.

Safety Evaluation

The safety evaluation for DC-3195 notes that the new motor operators perform the same accident mitigating function as the original pneumatic actuators, that they will not change the radiological release consequences of a LOCA, and that there is no effect on system performance and assumptions credited in the accident analysis.

The additional EDG loading is acceptable from an EDG load and fuel consumption aspect for all accident scenarios as indicated in calculation EC-E90-006. The modified valve assemblies have been seismically requalified and a seismic re-evaluation of the Safety Class 2 lines were acceptable.

The safety evaluation also addressed the increase in stroke time for the new operators, 25 seconds for the motor operator and 5 seconds for the

air operated. The safety evaluation concluded that the increase is acceptable based on the runout flow of the CS and HPSI pumps and the capacity remaining in the RWSP is adequate to operate the pumps for greater than 7 minutes with motor operator valves.

Revision 3 noted that the criteria for ILRT leakage also provides an acceptable leak rate criteria for the valves (Calculation EC-S91-016)

The safety evaluation concluded that the DC does not reduce the margin of safety as defined in the bases for any technical specification or approriate safety analysis.

13. DC-3197, Fire Protection Penetration Seals (Revision 2)

Description of Change

The DC provides for field installation or modification of penetration seals, fire barriers and fire dampers. Revision 1 consisted of penetration seal and fire barrier rework. Revision 2 provides for seal rework and new damper installation. The DC derates the floor of the H&V Room at elevation +46.00 and the room will become a part of Fire Area RAB 24. The South wall of the H&V Room will be a 3-hour rated barrier, maintains separation between Fire Areas RAB 24 and RAB 2, and door D258 will be replaced with a fire-rated door assembly.

Reason for Change

The DC is to bring the Fire Protection System into compliance with licensing commitments. A 100% penetration seal inspection conducted during 1988 - 1989 identified numerous seal and fire damper deficiencies.

Safety Evaluation

The safety evaluation concluded that the Fire Protection program is not part of the Technical Specifications, that the relocation of the associated fire barrier does not reduce the margin of safety for any other Technical Specification and that there is no impact on safety related equipment nor the relationship between safety related equipement in Fire Areas RAB 2 and RAB 24.

14. DC-3212, Steam Generator Snubber Pressurized Fluid Reservoirs (Revision 1)

Description of Change

The DC involves replacing ten (5 per generator) gravity fed fluid reservoirs for the Steam Generator (SG) support snubbers with eight (4 per generator) pressurized reservoirs.

Reason for Change

SG snubbers have been found to have internal corrosion. The gravity fed fluid reservoirs are vented to atmosphere allowing moisture to be entrained in the snubber fluid. The entrained moisture in the snubber fluid is believed to be the cause of the internal corrosion.

Safety Evaluation

The results of the safety evaluation show that implementation of the DC will not have any effect on the safe operation of any safety related system nor is there the possibility of any type of radiological discharge into the environment. The DC does not result in a change to any Technical Specification or result in an unreviewed safety question.

15. DC-3255, Boric Acid Condensate Chemical Addition Tank (Revision 0)

Description of Change

DC-3255 installs a Chemical Addition Tank in the recirculation piping which is routed from the Boric Acid Condensate Pumps to the Boric Acid Condensate Tanks. The Chemical Addition Tank will provide a permanent means for adding neutralizing chemicals to the Boric Acid Condensate Tanks for adjusting the pH of the water prior to discharge.

Reason for Change

The Boron Management System (BMS) collects and processes radioactive waste from various plant systems for recycle or disposal. After collection the liquid is normally transferred to the Boric Acid Concentrators for processing. The clean distillate from the concentrator is stored in the Boric Acid Condensate Tanks prior to discharge to the Circulating Water System. The pH of the distillate is often lower than the NPDES discharge permit limit, this requires the addition of neutralization chemical prior to disposal. There are no permanent provisions for adding these chemicals which necessitates the use of temporary tubing and a pump. This presents a personnel safety hazard.

Safety Evaluation

The safety evaluation reflects that the DC does not result in an unreviewed safety question or an unreviewed environmental question and confirms that all appropriate criteria for modifications to radioactive waste systems have been met.

16. DC-3265, Emergency Diesel Generator Engine Control Cabinets Ventilation (Revision 0)

Description of Change

The DC adds an exhaust fan to the top of each EDG Engine Control Cabinet (2 cabinets - "A" & "B") with an inlet grill on each door (2 per cabinet). The modification is designated as seismic, non-safety related. The DC will provide cooling by forced convection so that cabinet internal temperatures will remain less than 120 degrees F., the design temperature.

Reason for Change

The EDG Engine Control Panels are located in the EDG Rooms, RAB +21. The EDG Rooms tend to stay warm, normally near 100 degrees F. The Engine Control Panels have internal heat loads but do not have cooling fans or ventilation grilles. The internal temperature normally exceeds room temperature and this results in the failure of numerous annunciator module cards.

Safety Evaluation

The safety evaluation indicates that the DC will have no impact on any protective boundaries and does not reduce the margin of safety as defined in the bases for any Technical Specification or safety analysis. The fans will be non-safety related but are seismically qualified - designed for II/I seismic concerns.

17. DC-3271, Reactor Cavity Cooling Fans Inlet Damper Deletion (Revision 0)

Description of Change

The DC removes the internal damper blades, operating linkage and limit switches from the Reactor Cavity Cooling Fans (RCCF), RCCMVAAA101A & B, inlet dampers.

Reason for Change

The RCCF inlet damper, RCCMVAAA101B, has not been functional since the operating linkage was disconnected by WA-01027201. These non-safety related dampers are not required for system operation and are not required for air flow balance, and continued maintenance is not necessary or justified.

Safety Evaluation

The safety evaluation indicates that the DC will not reduce the margin of safety as defined in the bases for any Technical Specification or safety analysis and no unreviewed safety questions are created.

18. DC-3293, Testing of Valves ACC 114 A&B and ACC 116 A&B (Revision 0)

Description of Change

DC-3293 replaces Auxillary Component Cooling Water (ACCW) System check valves ACC-114A & B with "Locked Closed" butterfly valves. This will allow testing of ACC-114A & B and ACC-116A & B as required by ASME Section XI.

Reason for Change

The DC will alleviate the possibility of ACCW intruding and polluting the Condensate Storage Pool during required In-Service Testing of ACC-114A & B and ACC-116A & B. (See LER-89-014-00)

Safety Evaluation

The safety evaluation concludes that the margin of safety is not reduced from that which is defined in the bases for any Technical Specification or appropriate safety analysis and that no unreviewed safety question exists. The new valves and associated piping are designed to the same codes and standards as the original valves.

19. DC-3295, Steam Generator No. 2 Blowdown System (Revision 2)

Description of Change

This modification adds a four inch gate valve with a three inch bypass valve in a four inch line to the non-safety related portion of the Steam Generator #2 Blowdown System (BD). The bypass around BD-1043B will allow a slow, controlled fill of the blowdown line to preclude water hammer.

The containment isolation valve platform is also extended to facilitate operation of the new valves.

Reason for Change

Waterford 3 has experienced severe water hammer resulting in pipe support damage when establishing Steam Generator #2 blowdown. All events occurred during startup of the BD System or cycling of the BD containment isolation valves.

Safety Evaluation

The safety evaluation indicates that containment penetration loads remain within analyzed allowable limits and stresses remain within ASME Section III code allowable stresses. The extension of the valve operation platform complies with FSAR Section 3.7.2 and does not affect any plant systems. No unreviewed safety question exists as a result of the DC.

20. DC-3296, MSIV Replacement/Enhancement Stem Improvement (Revision 0)

Description of Change

This DC replaces the existing Main Steam Isolation Valve (MSIV) stem with a newly designed stem. The new stem is made of 17-4PH material as was the old design, the new stem is expected to provide a longer operating life because of reduction in stress concentration at the stem-to-stem head juncture. The DC alters only the stem-to-stem head fillet size without changing the material or any other dimensions of the MSIV stem.

Reason for Change

Numerous design related problems have plague the W-K-M MSIVs, such as guide rail failures and stem failures. In LER-89-023 Waterford 3 discussed redesigning the MSIV stems for increased life expectency. New stems of the same 17-4PH material but with a new design for the elliptic fillets at the stem-to-stem head connection are expected to increase the service life of the MSIVs.

Safety Evaluation

The safety evaluation notes that the change of the MSIV stem will allow more reliable operation of the MSIV and does not reduce the margin of safety, the DC does not alter any operational functions of the MSIVs.

21. DC-3304, Nitrogen Supply Header Modification
(Revision 0)

Description of Change

The change re-routes the nitrogen gas line which supplies the make-up water storage tanks and the Turbine Generator Building (TGB). This line will be tied into the high pressure Nitrogen supply line in the RAB. A pressure reduction station will be installed downstream of the new connection in the TGB to reduce the header pressure to 60 psig.

Reason for Change

Nitrogen gas is supplied to both non-radioactive and radioactive systems from a common source. This design has allowed non-radioactive systems to become contaminated by a backflow of nitrogen from radioactive systems. The incident occurred when the pressure of the Gas Decay Tanks was allowed to equalize with the Nitrogen line pressure.

Safety Evaluation

The safety evaluation indicates that the DC does not affect the bases for any Technical Specification or safety analysis. No unreviewed safety questions exists.

22. DC-3305, Modification of Secondary System Instruments and Panels (Revision 0)

Description of Change

The -4 RAB secondary sampling laboratory will be modified to reduce overcrowding and replace obsolete hydrazine, silica, and dissolved oxygen analyzers. Also, existing chart recorders will be replaced by a Micromax 2 Data Aquisition Unit which will allow for a more comprehensive display of data. The DC will also add a Power Distribution Panel (PDP) to the -4 RAB area

Reason for Change

The DC will reduce overcrowding to provide a more suitable working environment for chemistry personnel. The new analyzers will enhance instrumentation reliablility and the Micromax 2 unit will allow for a more comprehensive presentation of data.

Safety Evaluation

The safety evaluation determined that an unreviewed safety question does not exist for the DC. The function of the secondary lab will not be altered by the DC; it will only be improved. The DC only affects non-quality, non-safety related equipment, the modification is independent of any safety related functions and therefore, margin of safety is not affected.

23. DC-3315, Heated Junction Thermocouple (HJTC) Cable Supports (Revision 0)

Description of Change

The DC installs two retractable cable support devices at Nozzles 95 and 100 (one for each HJTC cable bundle). The installation of the cable support devices shall restrain and laterally and vertically support the HJTC cables, restricting the lateral and vertical movement of the cables, thereby reducing fatigue damage to the connectors.

The cable support device is designed to be retracted during refueling so as not to restrict bullet nose installation/removal and/or access to the top of the reactor vessel during cable demating.

Reason for Change

The existing HJTC cables that run from the Reactor Vessel Head to the Refueling Angle Support do not have sufficient lateral or vertical support to offset the air flows created by the Control Element Drive Mechanism cooling fans. This unrestricted lateral and vertical movement results in fatigue related damage to the HJTC probe receptacles.

Safety Evaluation

The safety evaluation demonstrates that this modification does not adversely affect the safety analysis, does not create the possibility of any new accident or malfunction of equipment important to safety and does not reduce the margin of safety as defined in the bases of any Technical Specification. Therefore, the DC does not create an unreviewed safety question.

24. DC-3318, RCP Lower Oil Reservoir Remote Oil Fill Line (Revision 0)

Description of Change

The DC provides remote oil addition capability to the lower reservoir on the Reactor Coolant Pump (RCP) motors. This is accomplished by routing one inch diameter stainless steel tubing from the existing fill connections on the motors, up the inside of the "D-rings," through the feedwater piping penetrations in the shield wall and terminated adjacent to the existing hand pump which was installed by SMR-1353. (SMR-1353 was reported in the 1987 Report of Facility Changes, W3P87-1751, dated October 27, 1987.)

Reason for Change

To provide the capability for remotely adding oil, when needed, to the lower oil reservoir on the RCP motors from outside the D-Ring during power operation.

Safety Evaluation

The safety evaluation states that the modification does not reduce the margin of safety as defined in the bases for any Technical Specification or safety analysis. The RCP motor lubricating oil system is not required for safe shutdown and has no connection to any safety related system.

25. DC-3325, Station Air and Electrical Services to Safeguard and Shutdown Heat Exchanger Rooms (Revision 0)

Description of Change

The DC adds Station Air (SA) supply connections and 480V receptacles in the Safeguard and Shutdown Heat Exchanger Rooms on the -35 elevation of the RAB. Three SA connections, one in each of the Safeguard Rooms A & B and one in the Shutdown Heat Exchanger Room B. 480V electrical receptacles will be located in Safeguard Room B and in Shutdown Heat Exchanger Rooms A & B.

Reason for Change

Maintenance activities in the Safeguard and Shutdown Heat Exchanger Rooms require the use of SA and electrical receptacles. To allow passage of service air hoses and electrical extension cords requires propping open the air lock doors to the rooms. With the doors opened the Controlled Ventilation Area System (CVAS) can not perform its design function.

Safety Evaluation

The safety evaluation identifies that the SA piping and the electrical conduit will penetrate fire resistant wall which are the boundaries for the CVAS, a safety related system. any penetrations created by this DC will be sealed with fire resistant material to prevent any air leakage in the CVAS boundary, thus the penetrations made by the DC will not affect the CVAS. The SA piping and the electrical conduit are seismically supported to prevent breaking at the CVAS boundary.

The modification will not affect any accident described in the FSAR and will not increase the probability or occurrence of an accident previously evaluated by the FSAR. The safety evaluation noted that no unreviewed safety questions have been identified.

26. DC-3329, Hoist Addi[illegible]n to Reactor Containment Building and Q-Deck Area (Revision 1)

Description of Change

Work associated with this DC involves two different areas. The first area is in Cooling Tower "A" and includes relocating an existing stairway to increase the Q-Deck work area. New platforms will be added and an endless monorail and two 2-ton hoists are to be installed. New support steel will be added to the existing structure to support the monorail.

The second area is in the RCB. A 2-ton jib crane is to be mounted at EL +62.25' on the east "D" ring concrete wall. The removeable equipment hatch located at EL +46.00' will be modified to allow for the installation of an operating platform for this crane. The existing platform at EL +21.00' will be reinforced to provide a new area to be used for the inspection and cleaning of the Reactor Vessel head studs. A 2-ton jib crane and a monorail with a 2-ton hoist will be added at EL +36.08' to aid in this process.

Reason for Change

The current arrangement for handling materials and tools for maintenance activities inside the RCB, the majority of which passes through the main equipment hatch at the +21.00' elevation, is very time consuming and wastes essential Q-Deck work space. It is also difficult for the plant Health Physics personnel to inspect and survey material passing through the equipment hatch for possible contamination.

Safety Evaluation

The safety evaluation documents that the DC will not have any affect on the accidents evaluated in the FSAR. The change used the same criteria established in designing the existing structures. The newly designed hoists and jibs are classified as seismic Category II and are seismically restrained, during operations the hoists and jibs will be restrained in the parked position by a locking device.

The safety evaluation noted that no heavy load drop analysis is required for the DC, either because no safety equipment is located below a crane or hoist or because the weight of loads the equipment will hoist is

below that stated by NUREG 0612 as requiring heavy load drop analysis. The DC does not create an unreviewed safety question.

27. DC-3332, Reactor Coolant System Recorder Enhancements (Revision 0)

Description of Change

The DC brings Reactor Coolant System (RCS) recorders into compliance with RG 1.97. Enhancements are made to the Pressurizer Level recorder, RCS Hot Leg Temperature, RCS Cold Leg Temperature and Reactor Regulating System 1 & 2 T Ave/T Ref recorders.

RCS Hot Leg Temperature Recorder RC-ITR-0112 1/2: The circuitry from the Resistance Temperature Detector (RTD) to recorder, as well as the power supply will be Class 1E.

RCS Cold Leg Temperature Recorder RC-ITR-0115/0125: The circuitry from RTD to recorder, as well as the power supply will be Class 1E, instrumentation inputs will be isolated and the range will be changed from 0 - 600 Deg. F. to 50 - 750 Deg. F.

RRS 1&2 T AVE/T REF Recorder RC-ITR-0121: Location on panel CP-2 will be changed. This relocation will allow placing a barrier around recorders RC-ITR-0115/0125 and RC-ITR-0112 1/2 to maintain separation requirements.

Reason for Change

The DC results from a review of the Waterford 3 Regulatory Guide 1.97 instrumentation requirments. The review indidcated that wiring configuration and scale ranges of the recorders did not meet RG 1.97. The DC brings the recorders into compliance with RG 1.97.

Safety Evaluation

The safety evaluation concluded that the DC will provide the required accident monitoring instrumentation, that no unreviewed safety question is created and that no Technical Specifications changes are required. It further stated that the DC does not reduce the margin of safety as defined in the bases for any technical specification or the safety analysis.

28. DC-3335, Enhancement to the Missile Shield (Revision 0)

Description of Change

The DC provides platforms and ladders on the missile shield to allow access to Component Cooling Water (CCW) piping and valves and one platform to provide access to the Control Element Drive Mechanism (CEDM) duct disconnect point. It also adds vent and drain lines to CCW piping spool pieces to be used when the spool pieces are removed during refueling activities.

Reason for Change

During each refueling outage the Missile Shield must be repositioned to allow polar crane access to the reactor vessel head. To accomplish this requires the removal of various piping and duct work and the installation of temporary scaffolding, ladders and a manbasket suspended from the polar crane. The DC should result in a decreased use of the polar crane, and reduced risk and exposure to the workers.

Safety Evaluation

The safety evaluation reflects that the DC does not result in an unreviewed safety question or an unreviewed environmental question and all appropriate criteria for modifications to the missile shield and CCW systems have been met.

29. DC-3348, EFW Condensate Storage Pool Level and SG Wide Range Level Recorders (Revision 0)

Description of Change

The DC will modify control room panel CP-8 display instrumentation to comply with RG 1.97. The changes will: 1) Provide Class 1E recording of Condensate Storage Pool water level and Steam Generators 1 & 2 wide range water level and, 2) Change safety channel SA Containment Sump water temperature from a recorded display to a meter and recorder inputs to provide a human factor control panel layout.

Reason for Change

Provide Class 1E recording for Steam Generator wide range water level and Condensate Storage Pool water level to satisfy RG 1.97 recommendations.

Safety Evaluation

The results of the safety evaluation determined no unreviewed safety questions and no Technical Specifications changes were required. The evaluation also noted that Technical Specification requirements for accident monitoring instrumentation have been maintained and that the DC does not reduce the margin of safety as defined in the bases for any technical specification or the safety analysis.

30. DC-3349, High Pressure Turbine Crossunder Pipe Manway Addition (Revision 0)

Description of Change

The DC adds ten manways to the non-safety related High Pressure Turbine crossunder piping. The manways are located in areas that will enable inspection for erosion/corrosion to be performed.

Reason for Change

The crossunder pipes have experienced damage due to erosion/corrosion. Addition of the manways will allow access to the currently inaccessible areas of the piping, enabling inspection and repair.

Safety Evaluation

The safety evaluation notes that no protective boundaries are affected by this DC, and that the modification is to non-safety related piping only. There are no unreviewed safety questions associated with this DC.

31. DC-3351, Replacement/Repair of Non-Safety Secondary Piping Components (Located Outside of Containment) Due to Erosion/Corrosion (Revision 0 and Revision 1)

Description of Change

Wall thinning in carbon steel piping with two-phase flow such as extraction steam and turbine reheater piping has been a potential safety hazard for plant personnel as well as a significant issue in plant cost and availability. At Waterford 3 pipe wall thinning is regularly monitored through the erosion/corrosion program. As a part of this effort a large number of non-safety piping components, located outside of containment, have been selected for examination. The DC provides for the repair and or replacement of non-safety class piping components identified by the erosion/corrosion program. Revision 1 of the DC provides for the stainless steel weld repair of non-safety related components covered by the DC.

Reason for Change

Piping components affected by erosion/corrosion are to be repaired/replaced to retain the pressure boundary and to function within the code stipulated limits. Non-safety carbon steel piping components other than cross under piping that do not meet the minimum thickness requirements will be replaced with similar components of low alloy steel or repaired to restore nominal wall thickness. Degraded cross under piping should be repaired in cases when the reduction in thickness is more than 1/8".

Safety Evaluation

The safety evaluation shows that there is no potential safety issue or unreviewed safety question. The evaluation notes that reduction in margin of safety is generally related to boundary performance parameters addressed in the Technical Specifications and that the DC does not impact the boundary performance parameters, thus, the safety limits as defined in the bases for any Technical Specification or appropriate safety analysis do not change.

32. DC-3355, Shutdown Cooling Alarms (Revision 0)

Description of Change

The DC will add one alarm for monitoring Shutdown Cooling operations, with one input from low Shutdown Cooling flow and one input from High Core Exit Temperature (Hi CET). The Hi CET Temperature will only be available with the reactor head on. The alarm will be displayed on CP-8 Annunciator and printed out on the Plant Monitoring Computer alarm log printer.

Reason for Change

The need for the alarm was identified by the NRC and as an ongoing task of addressing Shutdown Cooling risks by the Outage Risk Assessment Task Force.

Safety Evaluation

The safety evaluation notes that no unreviewed safety question exists as a result of the DC. In addition, the evaluation concludes that the Lo Flow alarm will prompt the operator to re-establish flow and that the Hi CET alarm will warn the operator of core boiling.

33. DC-3357, Replacement Spares for Obsolete Gould/ITE Molded Case Circuit Breakes (Revision 0)

Description of Change

The DC will procure Westinghouse replacement spares for all obsolete Gould/ITE 100A, 150A and 225A Frame breakers currently in service. It will also install 15 replacement breakers in spare Motor Control Centers (MCC) cubicles.

Reason for Change

MCCs at Waterford 3 contain Gould/ITE molded case circuit breakers which provide overcurrent protection for various electrical loads. Siemens has derated these breakers thereby lowering their interrupting ratings.

Safety Evaluation

The safety evaluation concluded that no unreviewed safety question exists, that this is not a protective boundary change and that no margin of safety will be reduced.

34. DC-3363, Hydraulic Power Unit (HPU) for Fuel Handling system (Revision 0)

Description of Change

A new pair of Hydraulic Power Units (HPU) for the Fuel Handling System will be installed with larger pumps, hoses and motors. The HPU is a part of the transfer system that moves fuel assemblies into and out of the containment building and provides the motive force for raising and lowering the upender with the fuel carrier.

Reason for Change

Upending time is unnecessarily slow and should be reduced to save critical path refueling outage time. The transfer system components will be used to save critical path refueling outage time by reducing the "one way" time of the upender and to improve reliability.

Safety Evaluation

The safety evaluation notes that the upender does not perform a safety function and that this DC is not safety related. There are no unreviewed safety questions as a result of the DC.

35. DC-3365, Main Generator Transient Protection/Monitoring Enhancements (Revision 0)

Description of Change

The DC will perform the following:

1) Document the installation of new degraded frequency and generator out-of-step function relaying in the offsite switching station. (Relays will be installed by offsite organization.)

2) Replace the generator metering and relaying frequency transducer.

3) Replace the generator excitation system current transducer, and,

4) Provide a reliable SUPS power supply for the generator metering and relaying circuitry.

Reason for Change

1) The present relay trip scheme makes no provision to protect the main generator in the event of system disturbance.

2) The present plant relay trip scheme makes no provision to protect the plant in the event of a degraded frequency condition.

3) The plant power frequency metering requires power from an uninterruptible power source.

4) The present current transducer used in monitoring the Generator Excitation System has no input-to-output isolation. As a result, the plant computer points and the control room indicators are affected by noise from the permanent magnet generator.

Safety Evaluation

The safety evaluation concluded that there is no unreviewed safety question and no required Technical Specification changes. The DC does not impact any protective boundary, any margin of safety or previously analyzed acceptance limits. The changes will enhance the reliability of the main generator, the main generator exciter field monitoring circuitry and the main generator metering and relaying circuitry.

36. DC-3366, Dewatering Filter and PHP Filter Skids from IA to SA (Revision 0)

Description of Change

Change the use of Instrument Air (IA) to Service Air (SA) for some Backwash Treatment System components located in the Dewatering Filter Skid and the Pneumatic Hydro Pulse (PHP) Filter Skid, located in the Condensate Polisher Building. The change will reduce some of the burden on the IA System.

Reason for Change

The IA System is adversely affected by the unnecessary burden of the components located on the two skids mentioned above. The original specification called for the SA System to supply these loads; this DC will return the air supply to the original source.

Safety Evaluation

The safety evaluation determined that there are no unreviewed safety questions associated with the DC. The modification will not affect any equipment important to safety nor involve any interaction with a protective boundry. The DC does not reduce the margin of safety as defined in the basis for any technical specification or safety analysis.

37. DC-3368, Process Analog Control Cabinet Ventilation (Revision 0 and Revision 1)

Description of Change

The DC will reduce ambient temperatures inside the Process Analog Control (PAC) cabinets by installing supply and exhaust fans designed by Westinghouse in each of the 18 PAC cabinets. Revision 1 of the DC added Digital Electro-Hydraulic (DEH) Control System cabinet ECP-21. Fans installed in safety related cabinets will be powered from safety related power distribution panels and those installed in non-safety related cabinets will receive power from non-safety power distribution panels.

Reason for Change

The PAC consists of the electronic control loops for the majority of the plant systems. The design ambient temperature range of the cabinets is 45 to 120 Degrees F. with normal temperatures of 70 to 75 Degrees F. The natural convection method of cooling has proven inadequate with temperatures as high 160 Degrees F. recorded on individual cards. The high temperatures have resulted in premature failure of numerous cards. The DC will reduce the ambient temperatures inside the cabinets.

Safety Evaluation

The safety evaluation states that the DC does not reduce the margin of safety as defined in the bases for any technical specification or safety analysis and that no unreviewed safety questions are created. The reliability of the PAC and DEH Systems will be significantly improved.

38. DC-3371, PPS Matrix Relay Hold Pushbutton (Revision 0)

Description of Change

Replacement of all the Plant Protection System (PPS) cabinet Matrix Test Modules HOLD pushbutton switches (total of six) with three position selector switches and indications. The replacement switch positions are OFF, HOLD, and LOGIC TRIP, spring returned to OFF from HOLD position. Adds PPS cabinet matrix test power channelization. This change involves adding an eight position selector switch to the test power suppply panel. The channelized test power scheme allows only one channel to be in test at a time.

Reason for Change

PPS cabinet surveillance testing has caused unwarranted actuation of ESF at Waterford 3. While the circumstances of each case has been different, the root cause has been attributed to the following:

The present configuration of the matrix test circuits does not provide an ideal man/machine interface, and is susceptible to switch failures and contact degradation.

The existing series connections of the PPS cabinet test power is susceptible to excessive line losses and relies on the proper operation of all the circuit elements upstream of the component being tested.

Safety Evaluation

The safety evaluation demonstrates that this DC does not adversely affect the safety analysis, does not create the possiblity of any new accident or malfunction of equipment important to safety, and does not reduce the margin of safety as defined in the bases of any Technical Specification. Therefore this change does not create an unreviewed safety question. The modification does not necessitate a change to the Technical Specifications.

39. DC-3372, Alternate 120 VAC Power for PAC Cabinets, 29, 30, 31 and 62 (Revision 0)

Description of Change

The DC provides a second source of 120 VAC power to each of four Process Analog Control (PAC) cabinets. Additional power cables will be run to PAC cabinets CP-29, CP-30, CP-31, and LCP-62. These cables will supply AC to the backup DC power supplies in each cabinet. Current arrangement provides only a single AC source to both the primary and backup power supplies.

Reason for Change

Currently both the primary and backup power supplies in each PAC cabinet receive 120 VAC input power from the same source. If this single feed is lost then the entire PAC cabinet operation is lost. This occurrence can cause the plant to trip. Installing a separate feed to the backup power supply, which is sourced from a different panel on a different SUPS from those of the primary feed, a single failure will not cause a loss of DC voltage within the PAC cabinets.

Safety Evaluation

The safety evaluation determined that the DC will not degrade plant safety or the licensing basis and that an unresolved safety question does not exist. The DC does not reduce the margin of safety provided in any Technical Specification. The change increases reliablitiy of power to four PAC cabinets.

40. DC-3377, SI-407A/B Valve Motor Changeout (Revision 0)

Description of Change

The DC replaces motors on valves SI-407 A & B with larger motors to increase available thrust to operate the valves during design basis service conditions. The existing motors, by calculation, deliver the required torque with little available margin.

Reason for Change

Safety Injection isolation valves SI-407 A & B are part of the NRC Generic Letter (GL) 89-10 Motor-Operated Valve Program. A design basis review (DBR) calculation and a dynamic test were performed in accordance with GL-89-10. The DBR calculation recognized that the valves are equipped with marginally sized actuators, for the opening direction only, based on the motor capability. Additionally, the dynamic test revealed that the valves actually required a higher stem thrust in the opening direction than the calculation estimated. The DC was initiated to changeout the motors and increase the torque to provide the required thrust and assure the valve will open during design basis service conditions.

Safety Evaluation

The safety evaluation determined that none of the protective boundaries are affected by the DC. The valves are containment isolation valves which are key locked in the closed position so containment integrity is still maintained. The evaluation also noted that there are no unreviewed safety questions associated with this DC.

41. DC-3378, West side Access Facility (Revision 0)

Description of Change

Install a new building on the west side of the plant for processing of personnel into the Reactor Containment Building (RCB) during refueling outages.

Reason for Change

In the past a temporary access to the Radiation Controlled Area has been used during refuel outages to expedite the processing of personnel into the RCA. This DC will make this a permanent access to the RCA for refueling activities and continue the expedited access for personnel.

Safety Evaluation

The safety evaluation determined that the DC does not affect any procedures, equipment important to safety, or accidents evaluated in the FSAR. It does not create any new types of accidents and it does not affect the Technical Specifications. The DC does not result in an unreviewed safety question or change to the Technical Specifications.

42. DC-3380, EGF Transfer Pump Flow Measuring Instrumentation (Revision 0)

Description of Change

Install a flow tube along with a local differential pressure gauge in the Emergency Diesel Generator Fuel Oil (EGF) system transfer pump recirculation line back to the storage tank. The DC will provide direct and reliable flow rate information.

Reason for Change

The Safety Evaluation Report (SER) for the Inservice Testing Pump and Valve Plan, Revision 7, Change 1, requires that flow instrumentation be installed for the EGF transfer pumps. Currently there is no flow instrumentation and flow rate must be derived using the change in feed tank level over time. Installation of the DC will allow the surveillance to be performed with fewer operators, without installing jumpers, draining the feed tank, or entering Technical Specification action statements.

Safety Evaluation

Results of the safety evaluation found no unreviewed safety questions or need for Technical Specification changes. The DC does not reduce any margins of safety as defined in the bases for any Technical Specification or the appropriate safety analysis.

43. DC-3381, PMC-Emergency Response Data System (ERDS) Datalink (Revision 0)

Description of Change

The DC will install a personal computer with modem in the Computer Room, Reactor Auxiliary Building (RAB) +46. The computer will provide emergency response data to the NRC in accordance with 10 CFR 50, Appendix E.

The Emergency Response Data System (ERDS) computer will be provided with the same battery backed power as the Plant Monitoring Computer (PMC). Data from the PMC will be supplied to the ERDS computer via a peripheral switch and transmitted to the NRC via a telephone modem.

Reason for Change

10 CFR 50, Appendix E requirements.

Safety Evaluation

The safety evaluation notes that the DC will not result in any unreviewed safety questions. The PMC is not used for any accident or safety analysis to safely shutdown the plant. The ERDS computer will provide data to the NRC during an emergency and is a monitoring device only.

44. DC-3386, Circulating Water Pump Bearing Lube Water Emergency Make-Up (Revision 0)

Description of Change

The DC will abandon the existing automatic cross-connection between the Fire Protection System and the Treated Water System. Two hose connections will be added in its place, one on the parish water supply to the Clearwell Tank and one on the discharge of the Circulating Water Pump Bearing Lube Water Pumps. These connections will provide a means for maually connecting the fire water system for emergency makeup water supply upon loss of parish water or loss of both Circulating Water Pump Bearing Lube Water Pumps. Level control instrumentation on the Clearwell Tank will be replaced and reinstalled at a higher elevation to provide a larger reserve capacity.

Reason for Change

The automatic backup that supplies fire water to the equipment on the Intake and Discharge Structures has seldom if ever been used because it has proven to be unreliable. The location of components for the automatic backup are in locations that make routine inspection and maintenance difficult and exposes the components to ground or rain water which contributes to the unreliability.

Safety Evaluation

The safety evaluation confirms that the DC will not reduce the margin of safety as defined in the bases of any Technical Specification or safety analysis and no unreviewed safety questions are created. The systems affected serve no safety function since they are not required for operation during the safe shutdown of the plant following an accident or to mitigate the consequences of an accident.

45. DC-3399, EH System: Fluid Chemistry Enhancements (Revision 0)

Description of Change

The Electro-Hydraulic (EH) reservoir contains two clean out plates on the top of the reservoir. These plates will be removed and replaced with new plates, one containing a penetration for connection to a new desiccant-breather and the other containing two pipe penetrations designated for use as connection points for a portable filtration skid.

Reason for Change

It is expected that better chemistry control of the fluid in the EH system will result in prolonged life of EH components and help to ensure more consistent operability characteristics.

Safety Evaluation

The safety evaluation concludes that the DC will not reduce the margin of safety as defined in the bases for any Techncial Specifications or any other safety analysis for safety related equipment. All protective boundaries will stay intact. Accident response and analysis and margin of safety remain unchanged by this DC.

46. DC-3406, Reactor Coolant Pumps 90% and Zero Speed Indication (Revision 0)

Description of Change

The DC will abandon in place the existing Reactor Coolant Pump (RCP) motor Zero speed and 90% speed switches. The Lift Oil Pumps for the RCP will be started by a contact on the RCP Breaker.

Reason for Change

The current speed switches are difficult to maintain and maintaining them interferes with RCP motor work. This DC utilizes equipment outside the containment building, thus improving maintenance and ALARA concerns. Zero speed indication will be obtained from the Plant Monitoring Computer and a contact on the RCP Motor breaker will be used for starting the Lift Oil pumps to provide lubrication on a coast down.

Safety Evaluation

The results of the safety evaluation indicated that there were no unreviewed safety questions. The RCPs and associated lubrication system are not safety related components. Functions of the speed switches will be performed by the PMC, for Zero speed indication, and a contact in the RCP motor breaker for the 90% speed switch.

47. DC-3408, Replacement of Agastat 7000 Series Time Delay Relays for EDG "A" and "B" Sequencer Circuits (Revision 0)

Description of Change

The DC replaces the E7000 series Agastat electropneumatic time delay relays in the ESFAS sequencer circuits with Agastat mode EDSC solid state relays. The solid state timing relays have improved accuracy, reliability (i.e., limited drift) and qualified relay life.

Reason for Change

Information Notice (IN) 92-77 "Questionable Selection and Review to Determine Suitabliity of Electropneumatic Relays for Certain Applications" stated that calibration of sequencer relays could drift beyond that given in Technical Specification 4.8.1.1.2.d.11. The relay drift could affect the scheduled loading of safety related equipment on to the 4.16kV and 480 V busses under accident condition (i.e., LOCA with or without LOOP, MSLB with or without LOOP, and LOOP).

Waterford 3 evaluated the impact of IN 92-77 and found that some relays had drifted beyond the +/-10% range given in Technical Specification 4.8.1.1.2.d.11 within a short period of time (days) while other relays have not shown any drift for extended period of time. Informal testing for determining the reason for drift beyond the +/-10% technical specification range proved inconclusive. The results demonstrated that the Agastat electropneumatic relays are inaccurate devices with no trendable characteristics.

Safety Evaluation

The safety evaluation determined that the DC does not alter the control logic or the timing of the ESFAS sequencer. The ESFAS sequencer will function as originally designed with improved accuracy, reliabliity, and qualified relay life. An unreviewed safety question does not exist. The evaluation also notes that the DC is consistent with the existing design bases for safety analyses. The tighter tolerances for the new relays will ensure that all equipment required to protect boundary conditions will actuate more precisely according to design conditions.

48. DC-3420, MSIV Operator Packing Retainer Enhancement (Revision 0)

Description of Change

The DC provides for the installation of a Main Steam Isolation Valve (MSIV) Actuator Stem Seal Leakage Collector. This device is mounted directly below the actuator stem seal, and is designed to collect inadvertent actuator fluid leakage. The design also provides leak-off ports for attaching tubing used to route the leakage away from the valve stem and body. The MSIV Actuator Stem Seal Leakage Collector Assembly is classified Non-Safety, Quality Class 3. The DC does not affect portions of the MSIV that are considered safety related.

Reason for Change

Corrosion of the MSIV stem and body has occurred in the past when actuator fluid (Fyrquel 150) accumulated at the stem base and on top of the valve body. The corrosion is caused by hydrolyzation of this fluid into phosphoric acid over time, when exposed to heat above 150 Degrees F., and moisture. No leakage has occurred since changing the original stem packing to the polypak design at Refuel 5. This DC will provide added confidence that the new valve stems, installed during Refuel 6, will last their expected life span.

Safety Evaluation

The safety evaluation notes that the activity is considered non-safety and does not affect portions of the MSIV that are considered safety related. Also, the implementation of this activity does not create new accidents or incorporate new type interactions or interfaces that did not previously exist. Therefore, no unreviewed safety questions were found to exist. The evaluation also concludes that no protective boundary is affected and that no margins of safety as defined in the bases for any technical specification are affected by this DC.

49. DC-3422, Fuel Transfer Tube Flange Modification (Revision 0)

Description of Change

This DC reduces the number of stud bolts required for the Fuel Transfer Tube blind flange from thirty-two to a minimum of eight. Calculation EC-M94-003 reflects that a minimum of eight stud bolts are sufficient to meet the design requirements. The DC specifies testing to verify that the bolting is adequate.

Reason for Change

Removal and installation of the thirty-two stud bolts in the Fuel Transfer Tube blind flange is time consuming and results in significant radiation exposure.

Safety Evaluation

The safety evaluation confirmed that the DC will not result in reducing the margin of safety as defined in the basis of any Technical Specification or safety analysis and no unreviewed safety questions are created. The integrity of the RCB Penetration 25 will be preserved and calculation EC-M94-003 reflects that eight bolts are adequate to ensure the integrity of the penetration for the design temperature and pressure.

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

50. CI-270579/WA-01063490, Repositioning of Valve FS-307

Description of Change

The CI documents the inoperable status (half open and immovable in either the open or close direction) of the fuel pool purification pump middle suction inlet valve (FS-307). This is one of three suction inlet valves in the spent fuel pool purification sytem. The three valves are located at elevations +7'6", +22', and +36'6" respectively and facilitate fuel pool purification suction at different depths in the pool.

Normal system lineup requires all three purification suction valves to be open.

Reason for Change

Due to the inaccessibility of FS-307, twenty feet below the water surface, and it's proximity to the spent fuel bundles located in the pool, corrective maintenance of this valve would be extremely difficult. Corrective action for this non-conforming condition is to "Use-As-Is" and to document the valve position (half open) as it's normal position. A review of the design basis for the system has shown that with FS-307 half open flow through it is marginally restricted and total flow suction to the purification pump is not impacted. Isolation of the suction inlet line will still be accomplished by using the purification suction isolation valve, FS-309.

Safety Evaluation

The safety evaluation determined that no unreviewed safety question exists because of this CI. The CI does not reduce the margin of safety for any relevant technical specification. System operation remains unchanged, with only a slight modification to normal valve line-up. Use of the valve in this manner is consistent with system normal operation, and does not require any compensatory action. System design parameters and configuration will remain unaltered.

51. CI-277058/WA-01082760, Use-As-Is 3/8" Tubing from 1" Sample Line to Sample Sink in BMS

Description of Change

The CI documents a non-conforming condition related to sample lines connected to the Boron Management Hold-Up Tanks drain and Recirculation Pumps discharge line. Disposition of the CI is to "Use-As-Is" the installed sample line tubing.

Reason for Change

Documentation of the sample line tubing did not exist. The tubing is routed to a sample sink and allows obtaining a sample of the Hold-Up Tank during draining or recirculation. Obtaining the sample at the sample sink reduces the possiblity of a spill of the sample. Documentation of the tubing is accomplished by this CI.

Safety Evaluation

The safety evaluation noted that there are no unreviewed safety questions associated with this CI and that the non-conforming condition does not affect any Technical Specifications or safety analysis.

52. CI-278512, NCR Repair-Abandoning of Electric Heating Coil EHC-58 in Place

Description of Change

Electric Heating Coil EHC-58 is located in a branch of the Switchgear and Cable Vault Ventilation System (SVS) supply duct which services the south end of the auxiliary relay room, elevation +35, Reactor Auxiliary Building (RAB). This CI documents the "Repair" of the non-conforming condition created by DCN No. HV-134, in 1982, which re-routed the duct containing EHC-58 and the associated temperature controller. The controller and EHC-58 are to be removed from service and abandoned in place.

Reason for Change

Original design of the SVS had EHC-58 and its associated controller maintaining the temperature in the corridor of the +35, RAB. In an effort to reduce temperature in the relay room DCN-HV-134R1 was initiated to reroute the duct to the relay room to provide more cooling flow. The controller for EHC-58 was not relocated, thus it continued to control EHC-58 from the corridor.

Safety Evaluation

The safety evaluation determined that abandoning EHC-58 in place will have no adverse effects on relay room ventilation. Rather, eliminating the posibility for EHC-58 to become energized will enhance the cooling of the relay room. EHC-58 is not safety related and is not required during or after any accident listed in the FSAR.

53. CI-281397/WA-01099254, Update to COLSS Database & Power Distribution Algorithm

Description of Change

The CI describes several activities associated with the Core Operating Limits Supervisory System (COLSS). 1) Rework of the COLSS Detailed Report Tasks to increase the precision of the data presentation. 2) Rework of the ABB/CE provided test inputs for COLSS Test Cases to meet the format requirements of the Waterford 3 COLSS Test Executive. 3) Rework of the COLSS addressable and hardcoded constants per the ABB/CE provided Cycle 6 update to the Waterford 3 COLSS Database Document. 4) Rework the COLSS Power Distribution algorithm per the ABB/CE provided update to the Waterford 3 COLSS plant specific supplement.

Reason for Change

The CI reworks the Waterford 3 Plant Monitoring Computer (PMC) COLSS software and addressable constants to support Cycle 6 operation.

Safety Evaluation

The four items presented in the Description were each evaluated and determined that only the Power Distribution change required a safety evaluation. The evaluation noted that the power distribution algorithm change will provide a more accurate representation of core conditions, as determined by the ABB/CE core model for Waterford 3.

The evaluation determined that the PMC and COLSS are non-safety related and are assumed unavailable for purposes of Design Basis Accident Analysis. COLSS is quality related and has no interfaces to nor direct control of plant equipment important to safety.

The Core Protection Calculators provide the safety protection with respect to Departure from Nucleate Boiling (DNBR), Linear Power Distribution (LPD), and Axial Shape Index (ASI).

54. <u>CI-281951/WA-01099801, Hydrogen Excess Flow Valve</u>

Description of Change

This CI/WA changes out the excess flow valve in the Hydrogen Bulk Storage and Supply System, HG-122, with one that closes at a hydrogen flow rate of 1000 scfh.

Reason for Change

Excess flow valve, HG-122 closes at a flow rate of 1447 scfh. This flow rate has the potential to exceed the maxium hydrogen cncentration of 2% as per NUREG 0800. To correct this non-conforming condition the excess flow valve will be replaced with one that closes at a flow rate of 1000 scfh.

Safety Evaluation

The safety evaluation noted that this change does not involve an unreviewed safety question. The evaluation notes that NUREG 0800 stipulates a protective boundary of 2% maximum hydrogen in the affected area. With a shutoff flow of 1447 scfh, assuming a 75% mixing efficiency between circulated air and hydrogen, the concentration has the potential to reach 3%. Using a flow rate shutoff of 1000 scfh will prevent exceeding the 2% maximum concentration.

55. CI-283170/WA-01108203, PSL-108 and PSL-205 Welded Fittings

Description of Change

This "Use-As-Is" non-conformance CI documents two socket welded Swagelock fittings along with their coupling adapters as a permanent plant configuration. The lines are off the Pressurizer surge line sample and Reactor Coolant Primary sample lines. They ar, changed from 1/2" threaded nipple with a cap to a 1/2" welded coupling with a welded Swagelock fitting and a cap.

Reason for Change

TAR-92-020 (1992 Report of Facility Changes, Test and Experiements, W3F2-92-0033, dated December 10, 1992 - Item 43) installed tubing between PSL-108 and PSL-205 to provide a flow path for sampling of primary coolant form the Pressurizer Surge Line. This was due to RC-104 being out of service due to leakage.

Safety Evaluation

The safety evaluation notes that no unreviewed safety questions exists. The drain lines are not safety related and will not affect any safety related equipment or accidents evaluated in the FSAR.

56. CI-288022/WA-01115303, Tube Steel Interferences with the Yoke of Valve SI-125B

Description of Change

This CI/WA modifies the tube steel support for the reach rod associated with Valve SI-410B. A portion of the tube steel support interferes with the yoke of valve SI-125B, the modification will allow for thermal growth of the valve yoke (SI-125B) and associated piping.

Reason for Change

Design Engineering identified this non-conformance during a field walkdown. It was identified that the installation of the tube steel support did not provide adequate clearances for thermal growth of the piping, resulting in a displacement of the valve yoke.

Safety Evaluation

The safety evaluation concluded that the modification is an enhancement to the existing configuration and will not have any negative impact on the operability and that no unreviewed safety question exists. The modification eliminates the load imposed on the valve yoke (SI-125B) as a result of the interference. Hence this will not constitute a change to the protective boundary and will not affect the margin of safety related to the performance of the containment.

57. <u>CI-289755/WA-01121379, Gear Change on SI-120 A&B and SI-121 A&B Motor Operators</u>

<u>Description of Change</u>

This non-conforming CI will change the gears, spring pack, and limiter plate on the motor operators of SI-120 A&B and SI-121 A&B to improve the motor output torque at a reduced voltage and increase the torgue switch trip margin. The repair will increase the overall actuator ratio which will result in an increased available torque at degraded voltage to accommodate an increase in valve factor or stem coefficient of friction. The change increases the assumed closure time for these valves from 30 seconds to 45 seconds which impacts the Post LOCA Offsite Dose.

<u>Reason for Change</u>

The existing actuators for valves SI-120 A&B and SI-121 A&B, Safety Injection Recirculation isolation valves,are considered non-conforming due to less than adequate predicted torque switch trip margin in comparison to the acceptance criteria in Design Engineering Guide MSPE-I-002, Revision 1, Guideline for Review of Motor Operated Valve Test Data. However, the valves pose no operability concern.

<u>Safety Evaluation</u>

The safety evaluation concluded that there are no unreviewed safety questions or concerns associated with this CI. The maximum stroke time of the valves increased from 21 seconds to 39 seconds.

Calculations performed (EC-S92-001 Revision 1 and EC-S93-007) to support this CI confirmed no increase in the consequences of Offsite and Control Room dose. Values continue to be less than 10 CFR 100 limts of 30 Rem Whole Body and 300 Rem Thyroid. The values also remain less than those acceptable limits of the Safety Evaluation Report (NUREG-0787), (222 Rem), and those documented in the FSAR, (290 Rem).

The impact of a 45 second close time upon Control Room dose and radiation levels within the Reactor Auxiliary Building due to activity in the Reactor Water Storage Pool (RWSP) would be negligible. The TID dose to equipment around the RWSP would be significantly less than the 10E5 Rad threshold for a harsh radiation zone.

58. CIs 261581, 261605, 261606, and 261608, Embedded Conduits Perpendicular to Seal Depth

Description of Change

The listed CIs document the acceptability and "Use-As-Is" of four non-conforming installed fire barrier penetration seals. The seals lacked direct qualification of 3 hour fire resistance. The CIs justify, through engineering evaluation, the "as-built" status of fire seals IIIA0249, IIIA0250, IIIA0282, and IIIA0284.

Reason for Change

The listed fire barrier penetration seals were found, upon destructive examination, to be in an untested arrangement. Specifically, embedded conduits were found to run perpendicular to the seal depth (the conduits were "buried" in the seal material, and invisible to observation when the seal material was in place). As part of DC-3197 (Item 13 of this report), similar configurations were subjected to third party fire testing, but failed to achieve complete qualification for use in a 3 hour fire barrier (the seals were qualified, however, for 2 hours and 47 minutes). Although the subject assemblies do not possess direct fire test qualification in accordance with industry specifications, the seals have been evaluated by Design Engineering-Fire Protection in the manner allowed by both Generic Letter 86-10 and Information Notice 88-04.

Safety Evaluation

The safety evaluation documents that the engineering evaluation for the acceptability of the subject seals are performed under the requirements of Generic Letter 86-10 and consider 1) the existing fire hazards, 2) the existing fire protection "defense-in-depth" features, and 3) the lack of an adverse impact to safe plant shutdown as the result of seal failure.

The subject seals are located entirely external to the plant's Radiation Controlled Area and do not have the potential to alter or increase the consequences of a radiological release to the environment.

The evaluation further notes that the Fire Protection Program is not part of the Technical Specifications. The barrier in which the subject fire barrier penetration seals are located is a three hour rated barrier

which separates the cable vault from the Relay Room on Elevation +35. Fire tests have documented that the subject assemblies can withstand a fire exposure of 2 hours and 47 minutes. A cable vault fire has the potential to require the transfer of control activities to the remote shutdown rooom and LCP-43 on elevation +21. Engineering evaluations have shown that the operation of transfer switching, if required, can be accomplished within the 2 hour and 47 minute rating of the subject seals.

NOTE: Items 59, 60, 61, and 62 were all generated as a result of the CS-125A event which Waterford 3 reported in LER-93-004, "Containment Spray Isolation Valve, CS-125A Failure to Stroke Open," (W3F1-94-0139, dated August 29,1994, transmitted Revision 2 of LER-93-004).

59. CI-287461, Pressure Surge and Fluid Transient in the CS Train "A" System (Includes Change 1 to Revision 1)

Description of Change

Containment Spray System (CS) piping not currently designed for a pressure of 650 psig will be qualified to this pressure.

Reason for Change

During the performance of a special test, pressures as high as 469 psig were measured in line 2CS10-7A and a fluid transient was observed in line 2CS10-9A due to check valve CS-111A slam. Similar conditions may have occurred at other times, and it is conservatively hypothesized based on Control Room readings that pressure surges as high as 570 psig may have existed in the line.

Safety Evaluation

The safety evaluation determined that no unreviewed safety question exists. Also, the evaluation states that the CS system functions to mitigate the consequences of, 1) a Main Steam Line Break or, 2) a Loss of Coolant Accident and limits containment peak pressure below the 44 psig boundary. The observed pressure surge and fluid transient in the line do not constitute a change to this protective boundary and do not affect the margin of safety related to the performance of the containment.

The evaluation notes that the structural integrity of the CS system has not been adversely affected by the occurrence of pressure surge and fluid transient.

60. CI-287462, 2CS10-7A Vent Line Addition

Description of Change

This CI adds one inch vent line to Containment Spray Train "A" header line 2CS10-7A.

Reason for Change

The one inch vent line is similar in design to numerous other vent and drain lines on this header. It will provide for better venting of this portion of the CS Train "A" header.

Safety Evaluation

The safety evaluation determined that no unreviewed safety question exists. The evaluation notes that installation of an additional vent line does not constitute a change to this protective boundary and does not affect the margin of safety related to the performance of the containment.

61. CI-287492, CS 125-A Solenoid Valve Addition

Description of Change

This CI/WA adds an additional solenoid valve to the actuator of CS-125A.

Reason for Change

The addition of the solenoid valve to the actuator of CS-125A is to decrease the opening time of the valve by providing an additional vent path.

Safety Evaluation

The safety evaluation determined that no unreviewed safety question exists. The evaluation notes that the additional solenoid valve will be actuated by the same signal and circuitry as the currently installed valve.

The installation of the solenoid valve on the CS-125A actuator does not constitute a change to this protective boundary and does not affect the margin of safety related to the performance of the containment. It will provide added assurance of meeting accident analysis assumptions by ensuring that CS flow rate will be achieved by opening CS-125A faster.

62. <u>CI-287536, Pressure Surge in Excess of Line Design Pressure in Train "B" of the Containment Spray System</u>

This item is the same as the discussion of Item 59 for Train "A" of the Containment Spray System.

63. CI-282605/WA-01101194, NCR Repair of Broken Incore Instrument Thimble R02

Description of Change

This CI/WA repairs a non-conforming Incore Instrument (ICI) thimble at core location R02. The thimble will be cut a nominal 4.5 inches below the ICI plate. Any remaining ICI assembly will be removed and replaced with a dummy ICI Rod and Seal Plug assembly.

Reason for Change

During removal of the Upper Guide Structure an ICI thimble was observed extending down from the Fuel Alignment Plate. To successfully lower the ICI plate during reassembly most of the the thimble extending down from the plate requires removal.

Safety Evaluation

The safety evaluation determined there is no adverse impact on the safety of the Reactor Coolant System pressure boundary or the reactor internals. No impact on the ability to meet Technical Specification requirements was also noted.

C. TEMPORARY ALTERATION REQUEST

64. TAR-92-018, Replacing the CPC Channel D RCP-2B Speed Sensor with COLSS RCP-2B Speed Sensor

Description of Change

The TAR replaces the Core Protection Calculator (CPC) Channel D Reactor Coolant Pump (RCP) 2B Speed sensor with the Core Operating Limits Supevisory System (COLSS) RCP-2B Speed sensor.

Reason for Change

A failed speed sensor for CPC Channel D has resulted in that channel being placed in "bypass."

Each RCP has six speed sensor outputs, two to COLSS and four to the CPCs. The COLSS inputs are designated "primary" and "backup," all of the speed sensors are qualified for safety related applications. COLSS will continue to function using the backup speed sensor and the bypass will be removed from CPC Channel D.

Safety Evaluation

The safety evaluation determined that no unreviewed safety question exists. The evaluation states that the TAR does not affect the accident analysis. The TAR will not change any protective boundary and will not reduce the margin of safety as defined in the bases for any Technical Specification or the appropriate safety analysis.

65. TAR-92-021, Hydrazine Injection into Gland Steam Condenser

Description of Change

This TAR provides an alternate hydrazine injection point to the Condensate System. It will allow injection of hydrazine to the steam side of the Gland Seal Steam Condenser.

Reason for Change

The TAR is installed to optimize reduction of condensate dissolved oxygen.

Safety Evaluation

The safety evaluation notes that no safety related components are affected by the TAR, no new methods of failure are identified and that no FSAR postulated accidents are affected.

66. TAR-92-028, Setpoint change to the Spent Resin Tank Programmable Controller (Also includes Revision 1)

Description of Change

The TAR changes setpoints in the Resin Waste Allen Bradley Programable Logic Controller (PLC) to allow operation of the Spent Resin Transfer Pump (SRTP).

Reason for Change

The PLC will not allow operation of the SRTP because the Spent Resin Tank level transmitter has been removed for repair. Changing the PLC setpoints will allow operation of the SRTP during this period.

Safety Evaluation

The safety evaluation determined that no unreviewed safety questions exists.

67. TAR-92-032, Temporary Diesel for the 3A2 and 3B2 4KV Buses

Description of Change

This TAR provides an alternate source of power to the 3A2 and 3B2 4.16kV buses, during periods of reduced Reactor Coolant System (RCS) inventory, with a temporary diesel.

Reason for Change

This alternate source of power to the 4.16kV buses would enable borated water to the RCS in the event of loss of off-site power and of the Emergency Diesel Generator or safety bus. The 3A2 and 3B2 4.16kV buses provide non-safety plant auxiliary loads and they also provide the normal source of power to the safety related 3A3-S and 3B3-S 4.16kV buses.

Safety Evaluation

The safety evaluation determined that no unreviewed safety question exists. The evalution also states that there is no reduction in the margin of safety as defined in the Technical Specifications bases or safety analysis. The temporary diesel will provide power in the event of loss of off site-power and the safety diesel or bus failing. Therefore no accident response is affected.

68. TAR-92-034, Connect Additional Air Capacity to the IA System

Description of Change

The TAR provides an additional air supply to the Instrument Air (IA) System receiver via a connection to an auxiliary portable compressor.

Reason for Change

The additional IA compressor capacity may be necessary to maintain IA pressure while the Station Air System is out of service for maintenance.

Safety Evaluation

The safety evaluation notes that the TAR does not affect any protective boundaries and does not reduce the margin of safety as defined in the bases for any Technical Specification or safety analysis.

69. TAR-92-042, Jumper Out QSPDS Channel 1, HJTC Sensor #3

Description of Change

This TAR lifts the leads from the heater, the heated thermocouple, and the unheated thermocouple for Heated Junction Thermocouple (HFTC) sensor #3, QSPDS Channel 1. The thermocouple will be replaced by an electrical jumper and the heater will be replaced by a 25 ohm, 75 watt resistor.

Reason for Change

This TAR supercedes TAR-92-004 (Reported in the 1992 Report of Facility Changes, Tests and Experiments, W3F2-92-0033, dated December 10, 1992) which jumpered out Sensors #1 and #3. A cable replacement has allowed Sensor #1 to be returned to service, while Sensor #3 is failed in the reactor vessel. QSPDS Channel 1 will continue to function, with seven sensors instead of the normal eight.

Safety Evaluation

The safety evaluation concludes that the TAR does not affect the margin of safety since the QSPDS Channel remains operable and able to perform its function. Technical Specifications allow the Channel to be operable with one sensor in the reactor vessel head and three in the plenum. This TAR results in two operable sensors in the head and five operable sensors in the plenum.

70. TAR-92-043, QSPDS Channel 2,HJTC Heater Sensor #4 Disconnection

Description of Change

This TAR lifts the leads from the heater, the heated thermocouple, and the unheated thermocouple for Heated Junction Thermocouple (HFTC) sensor #4, QSPDS Channel 2. The thermocouple will be replaced by an electrical jumper and the heater will be replaced by a 25 ohm, 75 watt resistor.

Reason for Change

QSPDS Channel 2 will continue to function, with seven sensors instead of the normal eight.

Safety Evaluation

The safety evaluation concludes that the TAR does not affect the margin of safety since the Channel remains operable and able to perform its function. Technical Specifications allow the Channel to be operable with one sensor in the reactor vessel head and three in the plenum. This TAR results in three operable sensors in the head and four operable sensors in the plenum.

71. TAR-92-045, Isolating the Load Transfer CEDM Coil of CEA #38

Description of Change

The TAR lifts the power supply leads to the Load Transfer CEDM coil for CEA #38, this will cause the coil not to operate.

Reason for Change

The Load Transfer CEDM coil functions to assist in the insertion and withdrawal of the CEA. There is an intermitten problem with the cable/connector for the Load Transfer CEDM coil for CEA #38. This coil is being deenergized to prevent possible damage to the other CEDM coils for CEA #38.

Safety Evaluation

The safety evaluation determined that no unreviewed safety question exists. The evaluation recognizes if CEA #38 is exercised the CEDM coils will be operated in an abnormal manner, the evaluation assumes that CEA #38 is exercised.

According to the evaluation an occurrance which may be affected by this TAR is the dropped CEA, however, because one gripper is not released until the other gripper is engaged and the disabled CEDM Load Transfer coil will only result in a longer stepping time, the probability of a dropped CEA is not increased.

The evaluation notes that the safety function of the grippers is to release the CEA for gravity insertion in the core and the TAR will in no way affect that function.

72. TAR-92-046, Cut and Cap SI-211 Drain Line to Reduce SIT Leakage

Description of Change

The TAR will stop leakage past Safety Injection System (SI) drain valve SI-211 by cutting and capping the drain line downstream of the seismic support. The cut is in the non-safety part of the drain line.

Reason for Change

Intent of the TAR is to reduce leakage from the Safety Injection Tank.

Safety Evaluation

The safety evaluation notes that the TAR only affects non-safety, non-seismic SI system drain lines and that failure of the drain line will not cause or affect any accident.

73. TAR-92-048, CMU Water Supply to Condenser "B" Manway Cover Leak Repair Device

Description of Change

The TAR installs a flexible hose downstream of Condensate Makeup System (CMU) CMU-2364 to supply water to a leak repair device installed on the "B" condenser manway cover. A valve is provided at the leak repair device to provide a "trickle flow."

Reason for Change

The CMU system supplies makeup to the Stator Cooling Water System (SCW) which is a closed loop system and requires very little makeup. The leak repair device on condenser "B" manway requires makeup due to contineous depletion of the water inside the device. This supply is being obtained from the CMU makeup to the SCW system.

Safety Evaluation

The safety evaluation determined no unreviewed safety question exists. the evaluation also noted that the TAR does not affect any safety related systems or accidents adddressed in the FSAR.

74. TAR-93-001, CMU Water Supply to "A" Condensate Pump

Description of Change

The TAR provides for the attachment of a water source to the air space surrounding the "A" Condensate pump shell.

Reason for Change

Condensate Pump "A" developed an air leak below the mounting flange which can not be repaired during operation. This air leakage significantly raises the dissolved oxygen content in the condensate system. Water will be injected into the air space as neccessary to maintain the air inleakage as low as practical.

Safety Evaluation

Based on the safety evaluation no unreviewed safety question exists. The evalution states that the TAR will not affect any accidents addressed in the FSAR, no safety related equipment, and no protective boundaries or margins of safety.

75. TAR-93-002, Feedwater Regulating Valve FW-173A Solenoid Valve Repair

Description of Change

This TAR installs a tubing union and a tubing cap on the end of the vent port of solenoid valve FW-ISV-173A to allow isolation of the vent port. The isolation of the vent port removes the "fail-as-is" capability of valve FW-173A.

Reason for Change

The Main Feedwater regulating valves are equipped with a pressure switch which detects a loss of Instrument Air pressure and sends a signal to a solenoid valve on the actuator which fails the control valve "as-is." The solenoid valve has developed a leak which requires sealing to allow normal operation of the "A" Feedwater Regulating valve.

Safety Evaluation

The safety evaluation concludes that the TAR will not increase the probablility of any accident occurrence. The regulating valve will perform normally in all conditions except loss of air. During loss of air the valve would lose function regardless of the TAR. The valve provides a backup Feedwater isolation function to the Feedwater Isolation valve and this feature will be unaffected.

76. TAR-93-004, Replace RTD (RCITE0122HC) for CPC "C" with RCITE0121X

Description of Change

The TAR modifies temperature loop RCITE0122HC and RCITE0121X by replacing the input from failed temperature element RCITE0122HC1 with an input from element RCITE0121X.

Reason for Change

Temperature element RCITE0122HC has failed, this TAR temporarily replaces the measuring section of the failed element with the Loop 2 hotleg input (RCITE0121X). This will restore Core Protection Calculator Channel "C" to operability.

Safety Evaluation

The safety evaluation concludes that instrument inaccuracies will change but will remain within that required. Calibration to compensate for the additional cable will be necessary. The evaluation also states that no unreveiwed safety questions exists and that the margin of safety remains within acceptable limits since the replacement RTD is the same model as the failed RTD.

77. TAR-94-003, Temporary Access to Containment During Refueling Outage #6

See also Item 41 of this report.

Description of Change

The purpose of the TAR is to activate the two temporary card readers located in the Westside Access Building, the TAR also activates four normally de-activated doors to ensure positive accountability during Refuel 6.

Reason for Change

Use of this TAR will enhance Containment access during the refueling outage.

Safety Evaluation

The safety evaluation notes that information as described in the Physical Security Plan is affected by this TAR but that requirements of the plan continue to be met. The evaluation concludes that there are no unreviewed safety questions associated with the TAR, that there are no accidents listed in the FSAR which the Security System is required to operate and that there is no equipment important to safety affected.

78. TAR-94-010, Transfer Machine Vertical Proximity Switch Temporary Alteration

Description of Change

The purpose of the TAR is to jumper out the Transfer Machine Upender's Fuel Handling Building side vertical limit switch and replace it with a toggle switch that can be actuated after the operator verifies the upender is in the vertical position.

Reason for Change

Due to mechanical problems associated with the vertical limit switch and the impact on refueling operations, the limit switch was bypassed with a manual toggle switch.

Safety Evaluation

The safety evaluation notes that the TAR does not have any effect on the Design Basis Fuel Handling Accident. The evaluation addresses FSAR section 9.1.4.3.2, "Failure of any of these interlocks in the event of operator error will not result in damage to more than one fuel assembly...The results of the safety analysis (Chapter 15) demonstrate that applicable dose limits are not exceeded as a result of the design basis fuel handling accident."

D. DRAWING REVISION NOTICES (DRNs)

79. DRN-E9202402, Temporary Barriers in Battery Rooms A, B, and AB (Fire Protection Penetration Seals - DC-3197, Item 13 of this report)

Description of Change

This DRN documents a temporary barrier comprised of visquene and duct tape between the existing battery banks and the penetration seal work location. The barrier will also contain a piece of flexible duct installed from the HVAC register through the visquene and into the part of the room containing the station batteries. The flexible duct will enable the supply/exhaust cycle to continue uninterrupted throughout the required rework of the penetration seal.

Reason for Change

Fire penetration seal work in accordance with DC-3197, Item 13 of this report, requires the use of a temporary barrier if batteries are present in the room during the work activities. The temporary barrier will be installed in only one room at a time. Room temperature will be monitored before and after barrier installation to ensure that volume change-over rate is constant.

Safety Evaluation

The safety evaluation determined that no increase in probability or consequence of battery malfunction will occur because of installation of the temporary barrier.

80. <u>DRN-E9302090, Telephone Communication System Diagram</u>

<u>Description of Change</u>

This DRN documents a process by which field personnel can reroute/rework existing unnumbered conduits which contain communication cables.

<u>Reason for Change</u>

Communication circuits (telephones, network functions) are typically installed in unnumbered raceways. These circuits use cables to provide voice and data transmission within the plant. Information Technology (formerly Information Systems) has the responsibility for this equipment. Construction is responsible for the installation of raceways and hardware.

Frequently new devices (computer terminals, etc.) have to be added and telephones have to be relocated. Sometimes, it is necessary to reroute the cables supplying these components. Therefore, the conduits may require rework which should observe all plant design criteria, separation criteria, seismic mounting considerations, and penetration integrity, etc.

<u>Safety Evaluation</u>

The safety evaluation determined that the requirement to perform this work in accordance with approved Waterford 3 procedures that no adverse consequences will occur because of this DRN. The evaluation also noted that there are no Technical Specifications affected by the DRN and that all existing safety margins will be unchanged.

81. DRN-M9102263, Reactor Coolant Pumps Material of Construction (1564-4276) (Rev 5)

Description of Change

This DRN adds additional vendor supplied parts list information, revises information that was missed by previous approved change documents and corrects some typographical errors on the existing Materials of Construction in the Pump Assembly Parts List of the Byron Jackson Pump Procedure No. IT-5668 for the Reactor Coolant Pumps.

Reason for Change

This is documenting changes to the Materials of Construction of the Pump Assembly Parts List consisting of upgraded materials information from the vendor, revisions already approved or typographical corrections, there is no change to equipment function or purpose.

Safety Evaluation

The safety evaluation concludes that the DRN incorporates upgraded materials information for the Reactor Coolant Pumps and does not reduce any margin of safety.

82. DRN-M9102920, Safety Injection Flow Diagram

Description of Change

The DRN revises the Flow Diagram - Safety Injection System (FSAR figures) to reflect as-built component UNID numbers for the shutdown cooling system vacuum priming pump motors.

Reason for Change

The power source for the vacuum priming pumps is incorrectly identified on the flow diagram and the FSAR figures. Updating the drawings and figures to correctly identify the power source will reduce the chance of tagging out the wrong power source when performing maintenance.

Safety Evaluation

The safety evaluation concluded that the DRN does not alter or have the potential to alter the operation or function of the plant system. No abnormal system operation is involved in the DRN. The vacuum priming pumps are non-safety and non-seismic and this DRN will only update the drawings to reflect existing plant conditions and correctly identify the power source to the vacuum priming pump motor.

83. DRN-M9103066, Component Cooling Water System

Description of Change

This DRN revises a section of lines 3CC6-204A and 3CC6-204B and valves ACC-113A & B which currently have a design pressure of 75 psig, which are directly connected, with no isolation, to piping (lines 3CC6-6A and 3CC6-6B) which has a design pressure of 125 psig.

Reason for Change

The lines discussed in the description supply cooling water from the Auxiliary Component Cooling Water System to the Essential Coolers. The DRN will only allow a section of the lines, 3CC6-204A & B, and valves ACC113A & B to be hydrotested with the rest of the Component Cooling Water System. The DRN does not affect the function of any equipment or procedure.

Safety Evaluation

The safety evaluation determined that no unreviewed safety question exists. The evaluation states that the DRN revises the design conditions of piping and valves to be consistent with as built piping. The original stress analysis shows that this section of piping and valves was always considered to be at the higher design pressure, and the pipe stresses are within the allowables. Therefore the DRN will not reduce the margin of safety as defined in the bases for any Technical Specification or the appropriate safety analysis.

84. DRNs for the Secondary Sampling System

The following DRNs were developed for the Secondary Sampling System and were all addressed by the same Safety Evaluation:

DRN-M9200176	DRN-M9201611
DRN-M9200177	DRN-M9201612
DRN-M9200178	DRN-M9201613
DRN-M9200185	DRN-M9201614
DRN-M9200709	DRN-M9201680

Description of Change

The DRNs update the drawings to reflect as-built configuration of the Secondary Sampling System.

Reason for Change

Walkdowns of the Secondary Sampling System revealed discrepencies in the as-built configuration of the system and the drawings for the system. The DRNs listed above update the system drawings to reflect plant configuration, there are no physical changes involved.

Safety Evaluation

The safety evaluation notes that the Secondary Sampling System is non-safety, non-seismic and is not required for safe shutdown of the plant and does not affect the protective boundary or margin of safety as defined in the Technical Specifications. The evaluation also notes that there are no physical changes to the plant, only that the drawings are updated to reflect plant conditions.

E. LICENSE DOCUMENT CHANGE REQUESTS (LDCRs)

85. LDCR-92-0054, Removal of Personnel Resumes and Staffing Numbers

Description of Change

The LDCR deletes Chapter 13A and Table 13.1-1 from the FSAR. Chapter 13A consist of resumes of Waterford 3 personnel and Table 13.1-1 contains staffing numbers for the Waterford 3 organization.

Reason for Change

Resumes and staffing levels were required in the FSAR prior to receipt of the operating license. Deleting the resumes and staffing levels was discussed with the Waterford 3 NRC Project Manager before implementing this LDCR. The information deleted is available through the Human Resources Department.

Safety Evaluation

The safety evaluation concludes that the LDCR is an administrative change only and has no impact on the function of plant structures, systems or components and poses no affect on the nuclear safety of the operating plant.

86. LDCR-92-0333, Revises Figures 9.5-1 (Sheet 1 of 4) and 9.5-2 (Sheet 1 of 2)

Description of Change

This LDCR revise and updates FSAR Figures 9.5-1 (Sheets 1 - 4) and 9.5-2 (Sheets 1 - 2) to reflect as-built configuration of telephone/paging communication system.

Reason for Change

The typical designation for Figures 9.5-1 and 9.5-2 will be removed and the figures will be updated as required. This LDCR will update the figures to reflect as-built conditions.

Safety Evaluation

The safety evalution identifies that this LDCR does not impact any safety related equipment and only updates information in the FSAR. No changes are made to the equipment.

87. LDCR-92-0340, Clarify HALON Testing Frequency to be in Accordance with NFPA-12A

Description of Change

The LDCR revises Section 9.5.1 of the FSAR to clarify the frequency of HALON Testing to be in accordance with NFPA-12A as referenced in the NRC Branch Technical Position BTP-9.5-1.

Reason for Change

The change corrects the FSAR to reflect semi-annual testing of HALON tanks/cylinders as described in NFPA-12A and the current version of the approved Fire Protection Program.

Safety Evaluation

The safety evaluation concludes that there is no impact to nuclear safety. The Fire Protection Program is maintained as existing and in conformance with NFPA-12A

88. LDCR-92-0341, Table 7.5-3 Note 14 Addition

Description of Change

This LDCR revises FSAR Table 7.5-3 to identify Accident Monitoring Instrumentation that is de-energized due to Station Blackout coping strategy or, where one diesel is not available after a Loss of Offsite power, which requires de-energizing loads to conserve the batteries. The change adds NOTE 14 to the table and delineates the devices affected by the NOTE.

Reason for Change

The LDCR identifies the instruments which are load stripped to cope with the aforementioned cases, it is an information clarification only; the instrumentation will continue to meet the requirements of the RG 1.97.

Safety Evaluation

The safety evaluation concludes that the change only provides clarification for Table 7.5-3 information and does not change any design basis for the Accident Monitoring Instrumentation. Instrumentation is de-energized by the Operations staff to cope with a Station Blackout situation or the loss of one diesel, which requires conserving battery capacity.

For the case where one diesel is not available, the faulted bus will be isolated. This eliminates non-essential and redundant loads to assure adequate battery capacity for trouble-shooting and restart of the diesel. Since all Category 1 Accident Monitoring Instrumentation is redundant, the remaining channel of Accident Monitoring Instrumentation will be available in accordance with RG 1.97.

For the case of Station Blackout, the basis for this scenario is a double failure, which is beyond the design basis for the Accident Monitoring Instrumentation. The LDCR update identifies the instruments which are load stripped to cope with the aforementioned cases. The LDCR is an information clarification and no unreviewed safety question exists.

89. LDCR-92-0447, FSAR Revision: Chapter 6.2.2.2.2

Description of Change

FSAR Section 6.2.2.2.2 is being corrected by this LDCR to correctly state that the mini-flow recirculation valves for the ESF pumps, SI-120 A&B and SI-121 A&B, must be manually shut following a Recirculation Actuation Signal (RAS).

Reason for Change

The FSAR is incorrect in stating SI-120A&B and SI-121A&B close automatically on receipt of a RAS. The valves receive a Close Permissive signal but must be manually closed by the operators.

Safety Evaluation

The safety evaluation determined that no unreviewed safety question exists in connection with the LDCR. The evaluation also notes operator action to isolate the mini-flow recirculation lines within two minutes of RAS will effectively prevent any contribution to offsite doses from this release path by effectively limiting radioactivity transport to the Reactor Water Storage Pool to negligible amounts due to valve leakage. Calculation EC-S92-001 documents this. Thus, the proposed change does not involve any reduction in the margin of safety as defined in the bases for any Technical Specification.

90. LDCR-92-0453, Waterford 3 Organization Changes

Description of Change

This LDCR reflects organizational improvements made in the areas of Quality Assurance, assessments, operational experience, regulatory reporting and interface, licensing, and corporate support. Changes in the corporate organization are also reflected. The change is administrative in nature and does not affect nuclear safety.

Reason for Change

Changes in the organizational structure as described in the FSAR, Chapters 1 and 13, require the submittal of this LDCR. There are no changes to the facility associated with the LDCR. Functions of the Operations Support and Assessments Group were re-assigned to other groups in the Waterford 3 organization, such as Nuclear Safety. Environmental and Hazardous Material functions, for example, were re-assigned to the plant staff organization.

Safety Evaluation

The safety evaluation concludes that the LDCR is administrative in nature and does not impact plant equipment, accidents listed/analyzed in the FSAR or any margin of safety.

91. <u>LDCR-93-0001, Correction to FSAR Table 5.3-13</u>

<u>Description of Change</u>

The LDCR corrects information in Table 5.3-13 to be consistent with the correct information contained in Table 5.2-6.

<u>Reason for Change</u>

The affected table (5.3-13) provides summary information on nil-ductility reference temperatures (RTndt) for materials in the reactor vessel beltline. The information contained in Table 5.2-6 is more comprehensive and is correct. There is no affect on equipment or procedures associated with this LDCR.

<u>Safety Evaluation</u>

According to the safety evaluation this is an administrative change only and does not affect any equipment or accidents. Correct information for RTndt is provided in the Technical Specifications and Table 5.2-6. this LDCR will only make Table 5.3-13 consistent with that information.

92. LDCR-93-0003, Changes to FSAR Section 9.2.5.2 and Table 9.2-11

Description of Change

The LDCR corrects FSAR information in Section 9.2.5.2 and Table 9.2-11 concerning the automatic control of the Wet Cooling Tower (WCT) fans and instrumentation and alarms.

Reason for Change

During preparation of instrument loop calculations EC-191-005, 036, and 037 discrepancies were identified in the information presented for the WCT fans, instrumentation, and alarms. This LDCR corrects the following:

Section 9.2.5.2 clarifies and corrects information concerning the automatic starting and stopping of the WCT fans.

Table 9.2-11 is updated to indicate a WCT basin high temperature alarm, removes the indication that there is control room recording of the WCT basin water level and corrects control panel numbers to reflect actual plant design.

Safety Evaluation

The safety evaluation determined that there are no unreviewed safety questions or Technical Specification changes and that the LDCR will not affect the safety analysis, accident probabilities, consequences, equipment malfunctions or margin of safety. The LDCR only clarifies and corrects the description of the system in the FSAR.

93. LDCR-93-0004, Containment Purge Isolation Valves

Description of Change

The LDCR revises information contained in FSAR Section 6.2.4.3.2 and Table 6.2-32 regarding the Containment Purge Isolation valves.

Reason for Change

The LDCR revises the FSAR to address the existing control arrangement of the Containment Purge Isolation valves. Section 6.2.4.3.2, Single Failure Analysis was revised to better justify the existing control arrangement for these valves. The control arrangement was also addressed in NRC Inspection Reports 50-382/91-30 and 50-382/91-31. In IR 91-31 the item was reviewed and considered resolved.

Safety Evaluation

The safety evaluation states that the facility is not altered by this LDCR, only the information in the FSAR is revised to clarify the existing control arrangement. There are no system changes implemented thus, no accidents are affected and no new failure methods created.

94. LDCR-93-0024, Revise and Clarify Nitrogen Accumulator Operating Time for ADV and EFW Valves

Description of Change

FSAR section 9.3.6.3.3, 10.3.1, and 10.4.9.3.1 are revised to show the calculated ten hour operating time for the safety related accumulators that supply backup motive gas for the Atmospheric Dump Valve (ADV) and Emergency Feedwater (EFW) valve actuators. The FSAR previously indicated valve operating times of 36 hours and 24 hours, respectively for the ADV and EFW accumulators.

Reason for Change

The LDCR does make any change to any equipment. The ADV and EFW accumulator durations of 36 hours and 24 hours respectively are changed to 10 hours. In the limiting case the 10 hour supply of nitrogen is sufficient to bring the plant to Shutdown Cooling (SDC) entry conditions at equal to or less than 400 degrees F. This temperature equals the design temperature of SDC system components, but it is more than the normal 350 degrees F. SDC entry temperature, per FSAR 6.3.2.9.7 and Waterford 3 procedures. Manual closure of EFW valves for containment isolation service may be as early as 10 hours instead of 24 hours after loss of Instrument Air.

Safety Evaluation

The safety evaluation concluded that there are no unreviewed safety questions. The accumulators are sized to operate the ADV and EFW valves for 10 hours, which is consistent with the natural circulation cooldown analysis described in Section 9.3.6.3.3 of the FSAR. Ten hours is sufficient to reach SDC entry condition at equal to or less than 400 degrees F. per FSAR Figures 9.3-8A and 9.3-8B. Additionally, the valves may also be operated manually, if necessary.

The LDCR does not increase the probablilty of occurrence or consequences, of accidents of equipment important to safety, as previously evaluated. There is no change to any protective boundary and accident response and acceptance limits are not affected.

95. LDCR-93-0033, Revises Table 9.4-10 (Sheets 1 & 3 of 5)

Description of Change

Table 9.4-10, Sheet 1 of 5 is revised to show static pressure for Air Handling Unit (AH) AH-25 fan of 5.0 inches w.g and the AH-25 cooling coils capacity to be 2,096,000 BTU/hr for each coil. Sheet 3 of 5 is revised to show the correct tag numbers and static pressure for the Battery Rooms, A, B, and AB Exhaust Fans.

Reason for Change

The LDCR was generated to correct information in the FSAR to reflect as-built conditions. Values listed prior to this LDCR were conservative with the exception of the value given for the AH-25 cooling coil. The incorrect value was within 0.5% of the correct value.

Safety Evaluation

The safety evaluation determined that there is no unreviewed safety question associated with the LDCR. No changes are made to equipment or systems; no changes are made to the Technical Specifications, and no margin of safety is affected.

96. LDCR-93-0034, Revises Table 9.4-8 (Sheet 1 of 2)

Description of Change

The LDCR corrects FSAR Table 9.4-8, Sheet 1 to reflect the as-built, design basis capacity of the Diesel Generator Room ventilation system exhaust fan E-28 (3B-SB).

Reason for Change

No equipment changes are made, only information in the FSAR is corrected to show as-built, design basis information.

Safety Evaluation

The safety evaluation concluded that the LDCR only corrects FSAR information to reflect as-built, design basis information and as such will not affect the probability, occurrence, or consequences of any accident. There is no unreviewed safety question associated with the LDCR, and there is no affect on Technical Specification bases.

97. LDCR-93-0037, Changes to FSAR Section 9.2.2 and Table 9.2-2

Description of Change

The LDCR revises the accident response description in FSAR Table 9.2-2, Sheet 2 to correct the setpoint for automatic actions related to Component Cooling Water (CCW) and Auxiliary Component Cooling Water (ACCW) flow to the Essential Chillers. Section 9.2.2 is also revised to clarify the difference between the nominal setpoint for alignment of ACCW to the Essential Chillers, and the maximum CCW design temperature.

Reason for Change

Review of Control Wiring Diagrams, Plant Operating procedures, and Design Basis Documents concluded that FSAR Table 9.2-2 provided an inaccurate description of system response during accident conditions and Section 9.2.2 required clarification.

Safety Evaluation

The safety evaluation concluded that the LDCR changes only information in the FSAR and does not affect equipment operation or function. The revision brings the FSAR into agreement with existing plant conditions and design documents. Both CCW and ACCW are required to operate in accident conditions and this safety function will not change, thus, the margin of safety for the CCW system will not be affected.

98. LDCR-93-0058, Design Data for Compressed Air System Components

Description of Change

Table 9.3-2 is corrected to reflect correct operating and design pressure values for the Station Air (SA) and Instrument Air (IA) compressors and the correct capacity and design pressure for the SA and IA air receivers.

Reason for Change

The LDCR corrects a discrepancy between FSAR Table 9.3-2 and current specifications, drawings, setpoints and design documents. Incorrect information in the FSAR is corrected to agree with design information.

Safety Evaluation

The safety evaluation determined that no unreviewed safety question exists, there is no change to the facility and that no accidents in the FSAR are affected by the LDCR. Additionally, the evaluation concluded that no protective boundaries are affected and that the margin of safety as defined in the bases for any Technical Specification or safety analysis is not reduced.

99. LDCR-93-0062, IA and SA Compressor Capacity

Description of Change

The LDCR removes the 100% capacity portion of the statement describing the Instrument Air (IA) and Station Air (SA) compressors in Section 9.3.1.2(a) of the FSAR.

Reason for Change

The current wording of Section 9.3.1.2(a) implies that each compressor will meet normal system demands (100% capacity), this misrepresents the as-built capacity of the compressors, i.e., it is not correct to state that each compressor is 100% capacity.

Safety Evaluation

The safety evaluation states that none of the accidents in the FSAR can be caused or affected by this change, which only revises information in the FSAR. The evaluation notes that the compressors are non-safety/non-seismic and are not required for safe shutdown or accident mitigation. The change does not affect any protective boundary and does not reduce the margin of safety as defined in the bases for any Technical Specification or safety analysis.

100. LDCR-93-0091, Incorporates Calculation EC-E90-006A, Revision OC

Description of Change

The LDCR revises FSAR Section 9.5.4 and Table 8.3-1 to conform with the revised loading of the Emergency Diesel Generators (EDGs) and the design basis for the sizing of the Fuel Oil Storage Tanks (FOST) as indicated in calculation EC-E90-006A, Revision OC.

Reason for Change

There is no change to the function of the EDGs or the FOST. They are required to power the ESF loads during a Loss of Off-site Power and contain seven days of fuel, respectively. The loads have been revised to show revised panel loadings and use motor brake horsepowers. This has resulted in the final EDG electrical loading to be reduced from the values previously shown in the FSAR.

Safety Evaluation

The safety evaluation notes that the accidents of concern are Loss of Coolant with Loss of Off-site Power, Main Steam Line Break with Loss of Off-site Power and Loss of Off-site Power. The probablility of an accident is not affected by the load on the EDG, the size of the FOST or by the amount of fuel required for seven days of operation, therefore the probability of an accident previously evaluated is not increased.

Consequences of an accident are not affected by the loading on the EDG (when within its rating), the size of the FOST or by the amount of fuel required for seven days of operation (when sufficient fuel is stored on-site to power ESF loads for seven days).

The probability of occurrence of a malfunction of equipment important to safety (EDG) is not increased as long as the total load is less than its rating.

Calculation EC-E90-006A, Revision OC, revised the EDG electrical loading and the FOST sizing requirements, therefore the LDCR does not increase the consequences of a malfunction of equipment important to safety.

The LDCR does not cause any new system interactions or connections.

Calculation EC-E90-006A, Revision 0C, shows an overall decrease in the steady state loading of the EDG, due to the use of motor brake horsepowers in lieu of the nameplate horsepowers to calculate motor input kW. Technical Specification 3.8.1.1.b.2 contains the requirement for the quantity of fuel oil to be stored on-site in each FOST (38,760 gallons, 759.6 which is unusable, leaving 38,000.4 usable gallons in each FOST). Calculation EC-E90-006A, Revision 0C indicates that 37,832.1 gallons are required for seven days operation for a LOCA with LOOP which is the most limiting of the three accidents evaluated in the FSAR for fuel oil consumption. The change does not reduce the margin of safety as defined in the bases for any technical specification or the appropriate safety analysis. No unreviewed safety question is associated with this LDCR.

101. LDCR-93-0094, Revision to FSAR Section 8.3.1.2.4(c)

Description of Change

Revises FSAR Section 8.3.1.2.4(c) to allow the Emergency Diesel Generator (EDG) frequency to momentarily decrease to 56HZ (93.3% of 60HZ) resulting from a step load increase. This will envelope the worst case loading conditions postulated in Information Notice (IN) 92-53.

Reason for Change

The EDG is designed so that loads are sequenced or connected in a way that allows the engine and voltage regulator to recover nominal speed and voltage after each large motor load is started and before the next load is connected. Loads are sequenced on a predetermined schedule based on the required accident loading sequence and the EDG's loading capabilities. However, some Emergency Safety Features Actuation System (ESFAS) loads will not start until a signal from the load sequencer and a process signal are present. The process signal could change the scheduled loading of the EDG under certain conditions (i.e., during a small break LOCA, MSLB, etc.). Changing the load sequence due to process signals was the concern raised by IN 92-53, "Potential Failure of Emergency Diesel Generators due to Excessive Rate of Loading." Waterford 3 evaluated the impact of IN 92-53 and found that frequency could momentarily decrease below the 57HZ (95% of 60HZ) minimum required by RG 1.9-1971. FSAR Section 8.3.1.2.4(c) indicates Waterford 3 is committed to RG 1.9-1971.

Safety Evaluation

RG 1.9-1971 requires the EDG frequency to be a minimum of 57HZ (95% of 60HZ) upon application of a step load. The EDG Unit Dynamic Loading Study, dated November, 1992, computed a frequency deviation below the 57HZ minimum (i.e., 56), for the unusual conditions noted in IN 92-53. The momentary EDG frequency deviation below 57HZ is acceptable based on the EDG Dynamic Load Study. The study concluded that all started motors accelerate to full speed and all running motors will re-accelerate to full speed during and following the frequency deviation. Thus, an unreveiwed safety question does not exist.

Accidents of concern are Loss of Coolant with Loss of Off-site Power, (LOCA with LOOP) Main Steam Line Break with Loss of Off-site Power (MSLB with LOOP) and Loss of Off-site Power (LOOP).

The frequency deviation may result in marginally slower acceleration times of started motors, however, the probability of occurrence of an accident is not a function of EDG frequency deviation or slower acceleration times of started motors. Discussions with Safety and Engineering Analysis indicated that the slower acceleration times do not impact the accident analysis, the analysis program codes envelope the resulting slower acceleration times, since the codes do not go to that detailed level of time increments. Thus the proposed change does not increase the probability of occurrence or the consequences of an accident previously evaluated in the FSAR.

No running motors stalled as a result of the frequency deviation and the slower acceleration times of the started motors is negligible, thus, the LDCR does not increase the probability of occurence or the consequences of a malfunction of equipment important to safety previously evaluated in the FSAR.

The timing of events assumed in the FSAR accident analysis are not refined to the point of accounting for momentary frequency changes which correspond to equipment acceleration time changes. Thus, the LDCR does not reduce the margin of safety as defined in the bases for any technical specification or the appropriate safety analysis.

102. LDCR-93-0116, Delete Reactor Power Cutback System Reporting Commitment

Description of Change

The LDCR deletes the commitment in FSAR Section 7.7.1 requireing a Licensee Event Report (LER) submittal due to the inadvertent, or spurious operation or malfunction of the Reactor Power Cutback System which challanges the ESFs including reactor trip.

Reason for Change

FSAR Section 7.7.1, "Reactor Power Cutback System," states "For at least the first two fuel cycles of operation, a LER will be submitted upon inadvertent or spurious operation or malfunction (exclusive of testing) of the Reactor Power Cutback system which challenges the ESFs including reactor trip.

Safety Evaluation

The safety evaluation conludes that the reporting requirement is being deleted but the system will continue to operate the same. The system is not important to safety, however, if a reactor trip should result from the Reactor Power Cutback System, the reactor trip is already analyzed. There is no unreviewed safety question associated with the LDCR.

103. LDCR-93-0144, RAB Wing Area Post-Accident Radiation Levels

Description of Change

This LDCR corrects the post-LOCA doses calculated for the Reactor Auxiliary Building (RAB) Wing Area. The RAB Wing Area post-LOCA doses have been re-calculated to account for the presence of piping in the -4 elevation RAB Wing Area containing highly radioactive Safety Injection System (SIS) sump or Reactor Coolant System (RCS) fluid. The subject lines are part of the Containment Spray System (CS), SIS, or Shutdown Cooling (SDC) piping. The changes affect the -35, -4, and the +21 elevations of the RAB Wing Area.

Reason for Change

By Letter W3F1-92-0483, dated December 23, 1992, Waterford 3 submitted LER-92-015, "Design Error Results in Radiation Levels Higher Than Previously Assumed." This LER reported the discovery that the post-accident background radiation levels in the vicinity of Component Cooling Water (CCW) valve, CC-713, could preclude manual operation of the valve following a LOCA. The Safety and Engineering Analysis group calculated the dose rate in the area at 6 hours post-LOCA, and a review of the previous calculations indicated that the radiation dose rate in the RAB Wing Area was much higher that previously known. Calculated post-LOCA doses have significantly increased as a result of revising the calculations based upon actual Waterford 3 geometry (calculations EC-S93-003, EC-93-004, EC-93-006, and Ebasco Calculations 3-C-3-58 and 3-C-3-60) Additionally, the Small Break LOCA has been found to be the worst case condition for the RAB Wing Area due to the larger source volume and the higher source term mandated by NUREG-0737.

Safety Evaluation

The safety evaluation concluded that no unreviewed safety question exists. The impact upon the plant of the increased post-LOCA radiation levels, either for personnel access or EQ, is acceptable from a nuclear safety standpoint.

The revised radiation levels are changes to the plant response after an accident, thus have no impact upon the probability of an accident. The evaluation also notes that equipment qualification (EQ) is judged acceptable. Calculations demonstrated that the +21 RAB Wing Area and

Switchgear Room "B" will remain environmentally mild radiation areas, with a total integrated dose of only 1,400 Rad (neutron, beta, and gamma). Thus, there will be no malfunction of equipment important to safety of a different type than previously evaluated.

According to the evaluation operators will still have access to the -4 RAB Wing Area (for access to CC-641 and CC-713 after a large break LOCA for durations of at least 20 minutes. (Temporary shielding was installed as part of corrective actions for LER-92-015.) Thus, there will be no increase in the consequences of a malfunction of equpiment important to safety.

The LDCR documents correct post accident radiation levels and does not cause any new system interactions or connections. Because the LDCR still allows personnel access to required plant locations after an accident without violating the 5 Rem GDC 19 requirement, there will be no impact of the changed post-LOCA radioactivity levels upon any margin of safety per Technical Specifications or other safety analyses.

104. LDCR-93-0164, Update FSAR Section 3.11 to Reflect Current Waterford 3 Environmental Qualification Program

Description of Change

FSAR Section 3.11 is updated to reflect the current implementation of the Waterford 3 Environmental Qualification Program and clarification of temperature environmental effects, "normal" and "harsh."

Reason for Change

Currently, FSAR Section 3.11 states that documentation which identifies environmental conditions and design bases of safety related electrical equipment that must function post accident is contained in the NRC submittal "Waterford SES Unit No. 3 Response to NUREG-0588." Section 3.11 also states that this submittal is "maintained up-to-date." The Waterford 3 EQ Program has evolved and this LDCR documents the current program.

Safety Evaluation

The safety evaluation concluded that the LDCR will not impact the intended functions of the affected equipment and that no unreviewed safety question exists. Plant systems, components, or stuctures or not altered by the LDCR.

105. LDCR-93-0165, Update FSAR Section 3.11

Description of Change

FSAR Section 3.11 states that "qualification of mechanical equipment is accomplished by requiring Vendor certification of the required capability and supporting documentation of test results or of operating experience on similar equipment." This LDCR updates Section 3.11 to eliminate the statement.

Reason for Change

Aging of mechanical equipment is primarily related to operational wear and metallic corrosion and erosion, and secondarily related to non-metallic degradation. The Waterford 3 Station Information Management System (SIMS) indicates that 515 out of the total 520 mechanical EQ components have surveillance testing with frequencies significantly less than, if applicable, their respective EQ maintenance task. With this preventative maintenance program currently in place at Waterford 3, operability of these components is verified, or the aging mechanisms will be detected and corrective maintenance will be performed. The remaining 5 mechanical EQ components are all metallic in construction, therefore do not contain age sensitive materials. Based on this, eliminating the Mechanical EQ Program will have no impact on the function or the safe operation of the plant.

Safety Evaluation

Results of the safety evaluation indicated that the elimination of the Mechanical EQ Program will not impact the intended functions of the affected equipment and that no unreviewed safety question exists. The elimination of the Mechanical EQ Program will not alter any plant systems, components or structures. The Waterford 3 mechanical Environmental Qualification Program deletion does not decrease safety margin and can be eliminated based on the following:

a) All 520 Mechanical EQ components comply with 10CFR50, Appendix A, General Design Criterion 4. The GDC4 evaluations are documented in EQ Files, and the evaluations establish that the Mechanical EQ equipment is capable to withstand the normal operating and postulated accident conditions.

b) A preventative maintenance program is established that will verify operability on 515 of 520 Mechanical EQ components. The remaining 5 components are metallic in construction, therefore do not contain age sensitive materials.

c) 193 of 520 Mechanical EQ components are metallic in construction, therefore are considered unaffected from aging and radiation effects.

d) Procurement controls are established to ensure design requirements of equipment replacements or refurbishments are preserved.

e) A failure and trending program and an inservice inspection are utilized to enhance the preventative maintenance program, and a corrective action program is in place to prevent recurrences of adverse conditions.

f) NPRDS data is utilized to identify common failures in the industry. This information is supplied to key departments for analysis to eliminate possible future failures or resolve current problems.

106. LDCR-93-0189, Updates FSAR Chapters 11, 12, and 13

Description of Change

The LDCR revises FSAR Chapters 11, 12, and 13 to implement the new 10CFR20 requirements, makes editorial corrections and organizational title changes. The LDCR will make the FSAR consistent with the revised 10CFR20 requirements with one exception. The effluent program as described in the revised 10CFR20 will not be implemented until a Technical Specification change is granted.

Reason for Change

The LDCR is required because of new requirements associated with the revised 10CFR20, Radiation Protection job titles and responsibilties changed because of organizational changes at Waterford 3 and the current description of the Industrial Waste Sump Turbine Building (TBIWS) Radiation Monitors does not clearly describe automatic actions that result when the monitor detects a high radiation level.

Safety Evaluation

The safety evaluation concludes that the LDCR does not alter the function or operation of a radwaste system, nor does it affect the potential for an unplanned or unmonitored release from the TBIWS. The LDCR only clarifies the system description for the TBIWS.

107. LDCR-93-0190, Decontrol Service Building

Description of Change

The LDCR "de-controls" information in the FSAR related to the Instrument Air (IA) and Station Air (SA) systems located in the Service Building.

Reason for Change

Modifications to the Service Building resulted in a Quality Notice being generated (QN #QA-92-142). This QN recommeded that the Service Building portion of the IA and SA systems be de-controlled.

Safety Evaluation

The safety evaluation concluded that the IA and SA systems serve no safety functions, and are not required for safe shutdown of the plant or limiting radiological release. Therefore the change will not reduce the margin of safety as defined in the Safety Analysis Report or Technical Specification bases nor result in any unreviewed safety questions.

108. LDCR-93-0194, Deletes Requirement for Color Banding of Safety-Related Cables

Description of Change

The LDCR affects wording in the FSAR regarding color banding of safety related cables in accordance with their assigned train. In the FSAR, Waterford 3 agreed to color coding cables at points of entry or exit from cable trays, at each end of a conduit, and at the cable end points. The current commitment is a deviation from the requirements of R.G. 1.75. This LDCR deletes the requirement for color banding of safety related cables altogether.

Reason for Change

The benefits of color coding during plant construction, when large quantities of cables were pulled through raceways, is readily apparent -- quick visual inspection reveals any discrepancy in separation of safety related cables through color consistency. Beyond initial construction, however, when cables are installed by work package the benefits practically disappear. Waterford 3 uses a cable and conduit computer data base which checks cable routes for validity and flags invalid routes. Cables are identified by a label at each end and safety related cable installation is verified by Quality Control inspection.

Safety Evaluation

The safety evaluation states that there are no unreviewed safety questions associated with this LDCR. The evaluation references an attached analysis of the LDCR that addresses the methods of assuring that separation criteria are met. Methods such as using existing routes for replacement cables, the computer data base mentioned above, identification of safety related cables at each end of the cable, development of cable "pull cards" which specify the route new cable must take and the Quality Control verification of safety related cable installation. The evaluation states that the LDCR does not reduce the margin of safety as defined in the bases for any technical specification or safety analysis because, as the referenced analysis indicates, alternate methods of assuring that separation criteria are met are implemented at Wateford 3.

109. LDCR-94-0199, Hydrogen Analyzer Accuracy

Description of Change

The LDCR revises the Hydrogen Analyzer full scale accuracy from 2.0% to 2.5% full scale. The revision affects FSAR Section 6.2.5.2.1 and Table 6.2-33.

Reason for Change

The Hydrogen Analyzers were replaced under Station Modification Package 983, Revision 3. (Reported in the Report of Facility Changes, Tests and Experiments for 1989 per 10CFR50.59, W3P89-2161, dated December 18, 1989). This LDCR addresses only the change in accuracy as described above because it was not included in the FSAR update at that time.

Safety Evaluation

As stated in the safety evaluation the accuracy of the hydrogen analyzers (2.5%) installed by SMP-983 is sufficient to meet the requirements stated in R.G. 1.7, this is based on a review of FSAR Figure 6.2-54 and Engineering Calculation EC-191-047. Thus, the LDCR does not result in an unreviewed safety question.

As noted in the evaluation, the limit for combustible gas concentration in the containment is 4% as determined by R.G. 1.7. According to Waterford 3 Operating procedures the Hydrogen Recombiners will be started within 24 hour post-LOCA or when the hydrogen concentration in containment reaches 3% (whichever occurs first). The Containment Atmosphere Release System will be used if the hydrogen recombiners are not available. Per Calculation EC-191-047 the loop uncertainty during an accident is 0.51% hydrogen. This uncertainty was determined based on the accuracy of the analyzers being 2.5% full scale. Therefore, the setpoint of 3% hydrogen is appropriately chosen with sufficient margin of safety to accomodate the loop uncertainty.

F. MISCELLANEOUS EVALUATIONS

110. Component Cooling Water Activity <10E-4 microcuries/gm with Letdown HX Leakage < 0.30 GPM

Description of Change

The evaluation addresses the acceptability, from a nuclear safety analysis viewpoint, of low levels of radioactivity (equal to or less than 10E-4 microcuries/gram) in the Component Cooling Water (CCW) System and of low leakage rates in the Letdown Heat Exchanger. The acceptable Letdown HX leakage is conservatively established at 0.30 GPM; the acceptance limit is based upon the accident analysis for a Sheared/Seized Reactor Coolant Pump Shaft with Loss of Off-site Power (LOOP)

Reason for Change

The evaluation is required because of low level contamination of the CCW system which has occurred due to a leak in the Letdown HX. The leak rate was estimated to be less than 0.005 GPM.

Safety Evaluation

The evaluation concludes that the consequences fo all events remain acceptable for Letdown HX leaks of up to 0.30 GPM and for initial CCW activities of up to 10E-4 microcuries/gram, the Technical Specification alarm setpoint. This conservatively assumes all activity leaking to the CCW system is instantaneously released to the environment, instead of realistically considering mixing in and leakage from the CCW system. Thus, there is no unreviewed safety question.

111. Component Cooling Water Activity for Cycle 6

Description of Change

This evaluation supersedes Item #110. The changes which this evaluation addresse are the two criteria which must be applied to determine the appropriate limits for CCW activity:

The limit for the incremental contribution to off-site thryoid doses due to activity initially contained in the CCW system. The limiting event for off-site thyroid dose is Large Break LOCA, and

The limit upon steady-state CCW DEI-131 activity as a function of RCS activity. This relation is used as a measure of RCS to CCW leakage (assumed through the Letdown Heat Exchanger), to ensure it remains less than or equal to 1.0 GPM. The acceptance limit is based upon the accident analysis for a Sheared/Seized RCP Shaft with Loss of Off-site Power (LOOP), and is consistent with the Technical Specification limit on primary-to-secondary leakage through another heat exchanger, the Steam Generator.

The CCW activity limits are intended as temporary limits until the source of the RCS-to-CCW leakage can be repaired. The evaluation is not meant as a permanent change to design basis analyses.

Reason for Change

The evaluation is required because of low level contamination of the CCW system which has occurred due to a leak in the Letdown HX. The leak rate was estimated to be less than 0.005 GPM. The evaluation addresses the remainder of Cycle 6 operations.

Safety Evaluation

The conclusion of the evaluation is that the presence of activity in the CCW system of:

equal to or less than 4.7E-2 microcuries/gm DEI-131 (thyroid dose equivalent; limit based on release of initial CCW activity, with LOCA the limiting event, and

steady-state CCW DEI-131 activity as a function of RCS activity less than the limit in Figure 1 supplied as part of the evaluation (to preserve Letdown HX leakage of equal to or less than 1.0 GPM).

results in acceptable consequences. Figure 1 of the evaluation also incorporates the 4.7E-2 microcuries/gm DEI-131 limit; this limit will be the more restrictive of the two limits only for relatively high RCS activities, i.e., within about an order of magnitude of the 1.0 microcurie/gm DEI-131 Technical Specification limit.

Review of whole body doses indicates that CCW activity, in terms of DE Xe-133, would have to exceed the RCS activity of FSAR Table 11.1-2, based on 1% fuel failure, to impact the whole body dose acceptance criteria. Thus, the whole body dose limit for CCW activity would be so non-restrictive that it is larger than the maximum RCS DE Xe-133 activity during power operation. Thus, it is not necessary to impose a limit on DE Xe-133 activity because the limit would be so high that it has no practical meaning or implication.

Note that the Technical Specification limit on total primary-to-secondary leakage is 1.0 GPM. Because the letdown leak rate is monitored based upon steady-state CCW activity as a function of RCS activity and because the activity limit would be relatively unaffected by the small difference between 1.0 GPM and the nuclear safety limit of 1.10 GPM, this evaluation supports the equal to or less than 1.0 GPM Letdown HX Leak Rate Limit.

The CCW activities at these limits uses up almost all available margin for the 2 hour off-site dose consequences for LOCA and RCP Seized/Sheared Shaft. Additional margin to address future concerns would have to be obtained by removing conservatism from input assumptions for these analyses. Margin has also been removed from the analysis for Excess Main Steam Flow with LOOP in determining these limits.

Thus, there is no unreviewed safety question provided CCW activity is less than the limit of Figure 1 supplied with the evaluation. The steady-state CCW activity as a function of RCS activity preserves the equal to or less than 1.0 GPM limit on Letdown HX leak rate.

The evaluation applies to Cycle 6 only.

112. Cycle 7 Core Reload

Description of Change

The Cycle 7 core will utlize a "very low leakage" fuel management scheme, with an estimated reactivity life of 483 Effective Full Power Days (EFPD) at 30 ppm boron concentration. Cycle 6 utilized an "ultra-low leakage" fuel management scheme, with an estimated reactivity life of 450 to 480 EFPD.

The Cycle 7 core will consist of 92 Batch J fuel assemblies, 84 Batch H assemblies (once burned, initially inserted in Cycle 6), 40 Batch G assemblies (twice burned, initially inserted in Cycle 5), one twice burned Batch C assembly which was discharged at the end of Cycle 2. All 48 Batch F assemblies and 44 Batch G assemblies in the Cycle 6 core will be discharged to the spent fuel pool

Batch J will consist of 20 type J0 assemblies (no poison rods), 8 type J1 assemblies (8 burnable poison rods per assembly), 36 type J2 assemblies (16 burnable poison rods per assembly), and 28 type J3 assemblies (16 burnable poison rods per assembly). The core will be loaded with quarter core rotational symmetry.

Batch J includes the debris-resistant fuel design which was originally incorporated in the Batch G assemblies (the entire Cycle 7 core will consist of debris resistance assemblies except for one Batch C assembly.

The Cycle 7 core is planned to be a sourceless core, i.e., the neutron sources used in Cycle 1 through Cycle 6 will be removed. During progressive cycles these sources have provided less of the total neutron source during fuel movement and approach to criticality, and the neutrons from irradiated fuel have become the dominant source. Studies (W3C1-94-0007, dated February 10, 1994) during previous refuelings have shown that the requirements for neutron instrumentation are met without the sources and that they are no longer needed

The Cycle 7 fuel was reviewed for consistency with the fuel storage rack safety analysis assumptions and the review concluded that the fuel is equivalent to or less reactive than the fuel assumed in those analysis. This included a revised analysis of the spent fuel rack which evaluated the effects of degradation in the neutron absorbing material Boraflex

and specifically accounted for the reactivity effects of debris-resistant features.

Reason for Change

Evaluation was performed for the fuel load (core configuration) for operational Cycle 7.

Safety Evaluation

All Cycle 7 changes and design basis events were found to be bounded by the Reference Analyses and within NRC acceptance criteria.

Comparison of key input parameters between Cycle 7 and the previous Cycle determined that Cycle 7 was bounded by the previous Cycle input, thus no re-analysis was required. Specific analysis was performed for all events for which comparison of key input parameters could not demonstrate that the Cycle 7 results would be bounded. These events include:

Increased Main Steam Flow with Loss of Off-site Power (Excess Load)
Main Steam Line Break (MSLB) (Pre-Trip Power Excursion)
Single Reactor Coolant Pump Shaft Seizure/Sheared Shaft

The three events required analysis due to changes in pin census for Cycle 7. The analysis of the Increased Main Steam Flow in combination with Loss of Off-site Power resulted in 2.4% of the fuel pins to experience DNB. This result remains bounded by the 3% value reported in the FSAR.

To remain bounded by the Reference Analysis fuel failure, 118% Required Overpower Margin (ROPM) was credited for the MSLB event (Cycle 6 analysis assumed a 116% ROPM). This ROPM is ensured by COLSS constants input prior to Cycle 7 startup. The results of the inside and outside containment cases for Cycle 7 indicate that 3.09% (Reference Analysis 4.5%) of the fuel pins fail for the inside containment MSLB and 1.10% (Reference Analsis 3.0%) of the fuel pins fail for the outside containment MSLB. The Reference Analysis remains bounding.

The analysis showed that the RCP Shaft Seizure/Sheared Shaft event is bounded by the Reference Analysis. The sheared shaft with LOAC results

in a minimum calculated CE-1 DNBR of 0.86 assuming an initial ROPM of 117%, which is larger than Cycle 6, but less than the ROPM assumed for some other events. This results in a fuel failure of 7.7% which is greater than Cycle 6 (6.7%), but is less than the 8.5% (Reference Analysis) reported in the FSAR. The resultant off-site dose is less than 10% of the 10CFR100 limits (30REM thyroid and 2.5 REM whole body).

An ECCS performance analysis, which consisted of an evaluation of the differences between Cycle 7 and the Reference Analysis, for the limiting small break LOCA was also performed for Cycle 7. It was determined that all input data for Cycle 7 are bounded by the Reference Analysis data for the small break LOCA event.

All accidents have been shown to have consequences bounded by the Reference Analyses and below the appropriate NRC acceptance limits, therefore there is no reduction in any margin of safety.

113. Pump and Valve Inservice Test Plan (Change 2 - Revision 7)

Description of Change

Revises the pump test loops, for the High Pressure Safety Injection and the Low Pressure Safety Injection pumps, to allow the pumps to be tested at a higher flow rate.

Reason for Change

The revised test flow path for the pumps is expected to produce higher quality predictive information, which will be used to maintain the components at a higher state of readiness by improving the component reliability.

Safety Evaluation

The safety evaluation determined that this does not represent an increase in the probablility or consequences of accidents or malfunctions of equipment previously analysed in the FSAR, nor does it represent the creation of a new accident or malfunction not previously analyzed in the FSAR. The evaluation also notes that the margin of safety for fuel clad performance and the margin of safety associated with the 10CFR100 limits for offsite exposure remain unchanged.

114. SPC-93-001 (Revision 0), Instrument and Control System Changes as a Result of the T-Hot Reduction

Description of Change

The Setpoint Change (SPC) will revise coefficients and constants in the Steam Bypass Control System to match revised steam bypass pressure program, revise coefficients and constants in the Reactor Regulating System for rescaling inputs to the pressurizer level program, rescale the steam header transmitters, replace main control board meter scales for steam header pressure, rescale the Plant Monitoring Computer points to the new transmitter ranges, and revise annunciators for high T-hot and T-cold.

Reason for Change

The Reactor Coolant System (RCS) temperature program for Waterford 3 was revised for Cycle 6 as an action to postpone or eliminate the need for Steam Generator replacement. Previously, the nominal RCS cold leg temperature was 545 degrees F. at 0% power and rose linearly to 553 degrees F. at 100% power. Full power RCS temperature is being reduced from 553 degrees F. to 545 degrees F., to increase Steam Generator corrosion resistance. The SPC is needed to improve control system responses to a load rejection.

Safety Evaluation

The safety evaluation concluded that because the changes are within the temperature range allowed per Technical Specifications there are no unreviewed safety questions associated with the SPC. The function and capabilities of the RCS as described in the FSAR will be unaffected.

The Technical Specifications and the Cycle 6 Safety Analysis Groundrules state that operation with a cold leg temperature of 544 to 558 degrees F. is acceptable at greater than 30% of rated thermal power. The evaluation notes that safety analyses have confirmed that the consequences of all events will remain within established acceptance criteria throughout Cycle 6.

115. SPC-93-006 (Revision 0), Emergency Diesel Generator Temperature Differential in EGL EGC systems, EGLMVAAA209A & B; EGCMVAAA106A & B

Description of Change

The SPC will increase the current temperature differential between the jacket water and lube oil discharge temperatures. This increase will optimize the long term health of the Emergency Diesel Generator (EDG) and is recommended by the manufacturer.

Reason for Change

The manufacturer (Cooper-Bessemer) recommended in a letter dated July 24, 1992, and Bulletin #688 a minimum temperature differential of 5 degrees F. between the jacket water and lube oil discharge temperatures.

Safety Evaluation

The evaluation concludes that the SPC will not reduce the margin of safety as defined in the bases for any technical specification and that there are no unreviewed safety questions associated with the SPC.

116. SPC-93-011 (Revision 0), Boric Acid Make-up Tank Temperature Alarms and Heater Controller

Description of Change

The SPC lowers the Boric Acid Make-up Tank (BAMT) temperature alarms and heater controller setpoint to cycle around 100 degrees F.

Reason for Change

Lowering the setpoints for the temperature alarms and heater controller results in a revision of a footnote on Table 9.3-13 of the FSAR. This change to the information in the FSAR results in the need for a safety evaluation. Setpoints remain significantly greater than the Technical Specification requirement of 55 degrees F.

Safety Evaluation

The safety evaluation demonstrates that there are no unreviewed safety questions associated with the SPC. Technical Specifications limit the temperature of the BAMT to a minimum of 55 degrees F., this is the minimum temperature assumed in accident analysis for a small break LOCA and post-LOCA long term cooling, the SPC will result in the temperature being maintained at 100 degrees F. Thus, there is no impact on the margin of safety or safety analysis.

117. Construction of the Maintenance Support Building

Description of Change

The evaluation addresses the construction of a three story office building on the north side of the plant, west of the present Administration Building and outside of the Protected Area.

Reason for Change

Contruction of the Maintenance Support Building (MSB) will provide additional office space for the plant staff organization, resulting in the removal of several trailers which have been used for office space.

Safety Evaluation

The evaluation notes that the MSB is designed structurally to withstand 110 Miles per hour winds, equal to or exceeding other buildings, such as the Administration Building or Service Building.

The building is not designed as a Seismic Class I structure however, it is located 120 feet from the Nuclear Island Structure. It is 45 feet high and would not affect any safety related equipment if it were to collapse.

To minimize the effect of pile driving on plant systems such as circulating water pipes, all piles will be pre-drilled through the backfill area.

The probable maximum flood is postulated based on the Mississippi River levee failing with the resulting water surface at approximately 27 feet. The construction of the building will not effect this ponded water surface elevation.

The MSB is located approximately 100 feet from any safety related structures or components, this distance will be more than sufficient to prevent a fire at the construction site from affecting essential plant equipment.

The evaluation concludes that there are no postulated accidents that would be caused, affected, or have consequences altered by the construction of the MSB.

118. <u>Substitute Part Equivalency Evaluation Report (SPEER)-9201001, Equivalency Evaluation for Replacement of 1" Gate Valves with 1" Globe Valves in the Treated Water System (Revision 0) and SPEER-9201025, Equivalency Evaluation for Replacement of 1-1/2" Gate Valves with 1-1/2" Globe Valves (Revision 0)</u>

<u>Description of Change</u>

This SPEER evaluates the equivalency replacement of eight 1 inch gate valves in the Treated Water System with one inch globes valves. The modification requires the revision of FSAR Figure 10.4-1 to reflect the new configuration.

<u>Reason for Change</u>

Replacing the gate valves with globe valves, which are more suitable for throttling flow, will result in easier flow balancing of the system.

<u>Safety Evaluation</u>

The safety evaluation concluded that the modification 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.

119. SPEER-9201094, Evaluation of Replacement of SI-343 WKM Gate Valve with Masoneilan Control Valve Model 48-20761 (Revision 0)

Description of Change

The SPEER evaluates the replacement of a 2" gate valve, Safety Injection Valve, SI-343, having a pneumatic piston operator with a 2" globe valve with pneumatic diaphragm operator.

Reason for Change

The change is being made because the original valve has failed local leak-rate tests because of excessive seat leakage. SI-343 functions as the inside containment isolation valve in the 2" drain/test line from the Safety Injection Tanks to the Reactor Water Storage Pool.

Safety Evaluation

The safety evaluation determined that there is no effect on any accidents evaluated in the FSAR because the new valve will perform the containment isolation function as well as or better than the original valve. The valve is seismically and environmentally qualified and stroke time conforms to FSAR Table 6.2-32.

The normal and failure positions, closed, of the replacement valve are the same as the original valve; the consequences of a malfunction would be the same as if the original valve were to malfunction.

120. SPEER-9201099, Substitution of Allen-Bradley Model "P" Relays for Model "BX" Relays (Revision 0)

Description of Change

The SPEER evaluated the substition of an Allen-Bradley Model P relay for the obsolete Allen-Bradley Model BX relay use in the speed control circuits of specific plant cranes manufactured by KRANCO.

Reason for Change

The application of KRANCO cranes at Waterford 3 are: Service Building Machine Shop Crane, Intake Structure Bridge Crane, Radioactive Waste Cask Handling Crane and Steam Generator Feed Pump Crane. The Allen-Bradley Model BX relay is obsolete and no longer manufactured.

Safety Evaluation

The safety evaluation was required because the Radioactive Waste Cask Handling crane is identified in the FSAR, Chapter 11 and Table 3.2-1. The other three cranes are not identified in the FSAR.

The evaluation concluded that an unreviewed safety question does not exist associated with the SPEER. The Radwaste Cask Handling Crane does not interface with any system that could affect an accident described in the FSAR or any system important to safety.

121. SPEER-9201104, Replacement of Velan 2" Check Valve: CVC-194A, B, AB, and CVC-202 (Revision 0)

Description of Change

The SPEER evaluates the replacement of a 2" Velan stainless steel inclined piston check valve with an Anchor Darling 2" stainless steel horizontal piston check valve.

Reason for Change

The existing Velan check valve is obsolete as the result of Velan's valve standardization program. The valves in the plant have experienced galling between the bonnet and body and cannot be reworked. The SPEER determined the Anchor Darling valve to be equivalent to the Velan check valve and will have no impact on the function of the system or component. Pressure drop across the valve will increase by less than 1 psi, this will not affect the function or operation of any system.

Safety Evaluation

The safety evaluation results indicate that there are no unreviewed safety questions as the result of the SPEER. The evaluation states that the replacement valve operation and function is equivalent to that of the existing valve. Operation of the affected system will not be changed because of the replacement valve. The replacement valves are designed and manufactured in accordance with the ASME Boiler and Pressure Vessel Code, Section III, Subsection NB. No margin of safety will be reduced as a result of the replacement valve.

122. <u>SPEER-9301135, Evaluation for Replacement of the Control Room Carpet (Revision 0)</u>

<u>Description of Change</u>

The SPEER evaluated new carpeting for the Control Room. The new carpet will conform to design specification LOU 1564.742E. This specification is updated to reflect changes in the carpet industry, some of which include performing newer flammibility and smoke generation testing which are more applicable for flooring materials.

<u>Reason for Change</u>

Replacement carpeting is required because the original carpet is obsolete. The original carpet was purchased by a construction specification and the replacement carpet is being purchased by a performance specification.

<u>Safety Evaluation</u>

The safety evaluation determined that the change will not cause any unreviewed safety questions. Although the fire protection requirements have changed, the material is still considered to be a Class 1 material in accordance with NFPA 101. The change of carpeting will not affect any accident described in the FSAR.

123. SPEER-9301165, Evaluation of Replacement Disc for 10" WKM SAF-T-SEAL Gate Valves CS-125A & B

Description of Change

The SPEER affects the Continment Spray Header Isolation Valves and is to allow the use of a replacement part which does not contain an ASME Section III "N" Stamp or Code Data Sheet. A replacement part without an ASME Section III "N" Stamp is allowed if it is provided to meet the requirements of NRC Generic Letter 89-09.

Reason for Change

Provides for the use of replacement parts supplied under the guidelines of NRC Generic Letter 89-09. The parts are no longer obtainable with the ASME Section III "N" Stamp, they are identical to the original parts with that exception.

Safety Evaluation

The results of the safety evaluation show that the change will not reduce the margin of safety as defined in the bases of any Technical Specification or safety analysis, and that no unreviewed safety questions are created. The evaluation notes that Generic Letter 89-09 requires the parts to be manufactured to the original component construction codes and that Authorized Nuclear Insurer inspection of the fabrication is required. The FSAR is also revised to reflect that the parts are obtained per the guidance of Generic Letter 89-09.

124. SPEER-9401228, Evaluation of Replacement Spring Spacer Ring in CS-125A & B (Revision 1)

Description of Change

The SPEER was developed to address a material and thickness change for the spring spacer ring on the operator of the Containment Spray Header Isolation valves (CS-125 A&B). The spacers are to be fabricated from ASTM A516 Grade 70, as opposed to ASTM A519 Grade 1020 which was the original steel. The thickness of the spacer ring will change from 1-1/2" to 1-1/4".

Reason for Change

The change of thickness 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 change in thickness will affect the opening force and the stroke time of the valve. The changes are considered to be insignifcant.

Safety Evaluation

The safety evaluation indicates that the thinner spacer ring will result in a decrease in opening force of 1%. In addition, the stroke length is increased by 1/4". The change of opening force of the valve is insignificant. The evaluation concludes that no accidents are affected by the change because the valve will continue to function in the original manner.

As indicated in the evaluation the containment peak pressure is not affected by the change because the valve will open on containment spray actuator signal as it should. There are no other boundary performance parameters affected by the change. There are no margins of safety affected by the change, as the function, operation, and reliability of the valve has not been adversely affected.

II. PROCEDURES

A. PLANT PROCEDURES

125. CE-002-006 (Revision 8), Maintaining Reactor Coolant Chemistry

Description of Change

CE-002-006 is a technical procedure which provides instructions for maintaining Reactor Coolant System Chemistry. Revision 8 to the procedure incorporates a lithium control program that is in accordance with current industry standards as established by EPRI and endorsed by ABB-CE. The revision also updates the procedure references, corrects the procedure format and makes clarifying changes to various sections of the procedure.

Reason for Change

The procedure was revised to update the procedure to current EPRI and ABB-CE guidelines, provide enhancements for log keeping, and procedure clarification.

Safety Evaluation

The safety evaluation concludes that changes to the lithium control program will have no direct bearing or effects on any existing accident analysis. The lithium control program will provide an environment in the system that will minimize the potential for corrosive attack and failure of system components.

126. CE-002-002 (Revision 5, Revision 6, and Change 1 to Revision 6), Maintaining Condensate and Feedwater Chemistry

Description of Change

CE-002-002 is a technical procedure which provides instructions and specifications for maintaining chemistry control in the condensate and feedwater systems.

Revision 5 incorporates new secondary chemistry guidelines on corrosion products, reduces Action Level 2 limit on Condensate Polisher (CPD) dissolved oxygen, provides instructions for adding ammonium chloride for molar ratio control, and removes the upper limit on feedwater hydrazine.

Revision 6 updates procedure reference section, re-defines Action Levels 2 and 3, replaces "Recorders" with "Micromax," and changed the condensate and feedwater pH upper limt from 9.6 to 9.8.

Change 1 to Revision 6 increased the pH upper limit to 10.0.

Reason for Change

The revisions to the procedure are intended to minimize corrosion product transport and reduce molar ratio to minimize Steam Generator Intergranular Attack/Stress Corrosion Cracking. Changing the pH upper limit will provide for operating at higher hydrazine concentrations which will assist in minimizing corrosive affects on the condensate and feedwater systems.

Safety Evaluation

The safety evaluations noted that the revisions do not result in an unreviewed safety question and that there are no changes to any protective boundary. The changes are in accordance with a recommendation of the EPRI PWR Secondary Water Chemistry Guidelines, Revision 3, May 1993.

127. CE-002-010 (Revision 6), Maintaining Safety Injection Tank Chemistry

Description of Change

CE-002-010 is a technical procedure which provides instructions for maintaining chemistry in the Safety Injection Tanks (SIT). This revision provides for a title change from "Plant Chemist" to "Chemistry Superintendent," adds a precaution to state the Technical Specification requirement for sampling, and clarifies the sampling and analysis of SITs after filling.

Reason for Change

Revision 6 provides an acceptable, alternate method for verifying SIT boron concentration (in lieu of sampling) subsequent to a tank volume increase, due to deliberate filling, of greater than or equal to 1% volume. The revision will ensure the chemistry limits are maintained as prescribed by the Technical Specifications.

Safety Evaluation

The safety evaluation concluded that the procedure revision implementing an alternate method for verifying boron concentration is not in conflict with the current licensing basis and will continue to satisfy Technical Specification requirements.

128. CE-002-100 (Revision 7), Chemistry Technical Specifications Surveillance Performance Coordination

Description of Change

CE-002-100 is a surveillance procedure that identifies the individual Chemistry Procedures and actions necessary to perform Technical Specification surveillance administrative controls, and Limiting Conditions for Operation Actions that are the responsibility of the Chemistry Department.

Reason for Change

Provides clarification of sampling anlysis required for deliberate filling of the SITs to restore level.

Safety Evaluation

See Item 127. The same safety evaluation addressed both procedures.

129. NE-005-055 (Revision 0), Bypass of QSPDS HJTC Sensor

Description of Change

NE-005-055 is a maintenance procedure which provides a method to bypass a failed Heated Junction Thermocouple (HJTC) heater and jumper out the thermocouples to allow continued operation of the probe.

Reason for Change

The procedure will allow bypassing a failed sensor in accordance with the technical manual. By providing this procedure, use of Temporary Alteration Requests (TAR) in the plant can be minimized.

(Items 69 and 70 of this report address bypassing HJTCs utilizing a TAR.)

Safety Evaluation

The safety evaluation notes that performance of this procedure results in the maximum number of level indications being available to the operator. As long as the requirements of Technical Specifications 3.3.3.6, Table 3.3-10 are met, the channel is declared operable with one sensor in the head and three in the plenum. Therefore, this procedure does not affect the margin of safety since the channel remains operable and able to perform its function.

130. NOCP-002 (Revision 1), NOC Procedure Classification, Type, Content, Numbering Format and Use

Description of Change

NOCP-002 is an administrative procedure which provides specific requirements and methods that are to be used to prepare Nuclear Operations Construction (NOC) Procedures. Revision 1 reformats the procedure, adds new definitons and adds a new section on continuous and informational use of procedures.

Reason for Change

The revision provides administrative guidance in determining the level of procedure use and makes the procedure compliant with Site Directive W2.101, Procedure Compliance, and consistent with plant procedure UNT-001-002,Procedure Classification, Type, Content, Numbering and Format.

Safety Evaluation

The safety evaluation states that the revision is administrative in nature, is conservative,i.e., adds information or requirements, and in no way adversely impacts activities affecting nuclear safety.

131. NOCP-003 (Revision 1), NOC Procedure Initiation, Review and Approval; Change and Revision and Deletion

Description of Change

NOCP-003 is an administrative procedure that provides instructions for the initiation, review, and approval of new NOC procedures, changes and revisions to, and deletion of NOC procedures. Revision 1 clarifies procedure preparer responsibilities for obtaining and reviewing commitment reports for affected procedures and for submitting changes to Licensing. Responsibilities for UNT-005-027, Infrequently Performed Tests or Evolutions, review were also clarified.

Reason for Change

The revision will ensure that the commitment database is adequately maintained, makes procedure requirements consistent with UNT-005-027, and incorporates the guidance of Site Directive, W2.101, Procedure Compliance.

Safety Evaluation

The safety evaluation states that the revision is administrative in nature, is conservative, and in no way adversely impacts activities affecting nuclear safety.

132. NOCP-303 (Revision 3), Installation of Electrical Cables

Description of Change

This revision to NOCP-303 adds requirements for Fire Protection System impairments. Also adds requirements for temporary (between working shifts) replacement of separation materials during cable installation.

Reason for Change

The revision will enhance requirements for fire impairments and impose requirments for temporary replacement of separation materials during performance of the procedure.

Safety Evaluation

The safety evaluation states that the revision involves more stringent requirements during the performance of the procedure and that no accident would be caused by the revision. The chance of a fire in one train affecting the other train is lessened by the revision.

133. <u>NOECP-403 (Revision 0), Replacement and Repair of Non-Safety Secondary Piping Components Due to Erosion/Corrosion</u>

Item 31 of this report, DC-3351, discusses the replacement and repair of non-safety piping components due to erosion/corrosion.

134. OP-001-003 (Change 1 Revision 14), Reactor Coolant System Drain Down

Description of Change

OP-001-003 provides instructions for draining the Reactor Coolant System (RCS) and for maintaining RCS level while drained down. The procedure includes limitations on when the RCS is allowed to have all cold leg nozzle dams installed based on RCS vents which have been established and the number of days since the reactor has been shutdown.

The procedure change incorporates the requirements for placing Steam Generator nozzle dams in the hot and cold legs as calculated by EC-M88-012, Revision 4.

Reason for Change

Calculation EC-M88-012, Revision 4, was performed to allow the nozzle dams to be installed if the reactor has been shutdown greater than nine days, with (RCS) temperature equal to or less than 110 degrees F. The pressurizer manway and all ten reactor vessel closure head instrument penetrations must be open. Based upon information received since the calculation was last revised, excess conservatism was able to be removed, resulting in less restrictive limits on when nozzle dams could be installed. This could result in a decrease in outage length.

Safety Evaluation

The safety evaluation notes that limitations on when all cold leg nozzle dams (and hence, all nozzle dams) can be installed do not affect accident probabilities. Nozzle dams are installed when the reactor is shutdown. Possible shutdown events include reactivity-related events, losses of primary coolant while shutdown, and a loss of Shutdown Cooling. Nozzle dam installation and configuration will have no affect upon the probabilities of these events.

The revision to OP-001-003 will maintain the current requirement that the core uncovery time be 60 minutes or greater assuming worst case RCS pressurization. As stated in calculation EC-M88-012, Revision 4, this is a slightly more restrictive requirement than the containment closure requirement that the best estimate core uncovery time be 90 minutes or more. Because these requirements are met, the margin of safety is unchanged.

135. OP-002-005 (Change 3 and Change B to Revision 10), Chemical and Volume Control(CVCS)

Description of Change

Change 3 to OP-002-005 shows the required position of CVC-403, Reactor Coolant Pump (RCP) Controlled Bleedoff Outlet Header Isolation Valve, to be "open" vice "throttled."

Change B is a deviation to the procedure that provides provisions for operation of the CVCS during a short-term, interim system line-up while Letdown is isolated at power to inspect and repair the Letdown Heat Exchanger.

Reason for Change

CVC-403 is a manual header isolation valve and should be positioned in the open position to maintain RCP Seal at a minimum backpressure of 40-50 psi, as recommended by the seal vendor. Change 3 does not affect any operating features.

Change B provides provisions for operation of CVCS with Letdown isolated for inspection and repair of the Letdown Heat Exchange. Provisions are included for maintaining RCS inventory and reactivity control by manually controlling Pressurizer and Volume Control Tank levels within acceptable bands via cyclic operation of a selected charging pump.

Safety Evaluation

The safety evaluation for Change 3 notes that the change does not change the way the plant and system is currently operated, it only designates the required position for CVC-403 as "open" vice "throttled." (The valve is currently in the open position). This change does not impact the ability of CVCS to perform it's safety function.

Accidents in the FSAR that may be caused or affected by Change B to OP-002-005 as discussed in the safety evaluation are:

CVCS malfunction; boron dilution,increased inventory, increased inventory with single active failure,

Letdown Line Break,

LOCA, and

Increase in Heat Removal by the Secondary System

Change B only applies at power so the boron dilution accident is not applicable. The increased inventory and increased inventory w/single failure are identified as moderate frequency incidents and the safety evaluation concluded that increased operator awareness of the Pressurizer level and Charging pump operation should decrease the probability of this accident occuring during the short duration the plant will be in this configuration.

For the Letdown Line Break, the failure of the Letdown line outside containment is the event of concern. Letdown will be isolated by the procedure Change by closure of the Letdown Stop Valve, which is upstream of the Letdown Containment Isolation valves, the probability of occurrence of this limiting fault accident is decreased by the Change.

The change is expected to result in additional thermal cycles of the charging inlet nozzles, the total number of thermal cycles on the nozzles to date (28) plus the expected number (9 - 12) during the Letdown Heat Exchanger repair will not exceed the currently allowed (100). For the LOCA the likelihood of having the charging inlet nozzles fail as a result of excessive thermal cycles and cause a LOCA will not be increased.

There are no new system interactions or connections being created by the Change. No new failure modes will be created as a result. The CVCS and its interacting systems will be operated within originally designed limits.

The safety evaluation concludes that the margin of safety as defined in the FSAR is not impacted by the change. The change does not affect a protective boundary.

136. OP-003-018 (Change 1 Revision 11), Lube Oil Storage, Transfer and Purification

Description of Change

Change 1 to this operating procedure adds new sections to address Startup and Shutdown of alternate oil purification for the Steam Generator Feedwater Pump (SGFP) oil sumps.

Reason for Change

The new sections of the procedure are required in the event that the SGFP centrifuge becomes unavailable or is no longer adequate for removing contaminants. These sections detail the use of the Oilquip Vacuum Dehydrator as an alternate means of purification for the SGFP oil sump.

Safety Evaluation

The safety evaluation concludes that the equipment is not important to safety and as such any failure of the feedwater pump/turbine, the oil purifier and any adjacent equipment will have no impact on equipment important to safety. No unreviewed safety questions were found as a result of this Revision to the operating procedure.

137. OP-003-019 (Change 2 Revision 8), Nitrogen System

Description of Change

Change 2 to Revision 8 of this system operating procedures changes the position of valve NG-15211 from Open to Closed on the Standby Valve Lineup.

Reason for Change

This change isolates the Blowdown Filter Flush Accumulator which will affect the flushing operation of the Electromagnetic Filters (EMF).

Safety Evaluation

According to the safety evaluation implementing this change will not cause or affect any of the accidents listed in the FSAR. The Steam Generator Blowdown system will operate as designed with the exception of flow passing through the EMF. This does result in the loss of the filters to perform their function, however, the EMF are not currently being used, but no new accidents will occur as a result. The change does not reduce the margin of safety as defined in the bases for any technical specification or appropriate safety analysis.

138. OP-003-031 (Change 3 Revision 9), Operation of the Condensate Polisher (CDP) Unit Without Resin Pre-coat

Description of Change

Change 3 to operating procedure OP-003-031 adds the option to operate the Condensate Polisher without a resin precoat. Operation without resin precoat will be performed during steady state operations.

Reason for Change

Operation without resin precoat, utilizing the vessels only as a filter unit, is expected to reduce the cation to anion molar ratio in the Steam Generators. During steady operation the CDP does not provide a significant amount of ionic impurity removal. The filter elements alone will provide filtration removal of particulate metal oxides.

Safety Evaluation

The safety evaluation states that the change is an operation mode change only that does not require any equipment/system modification. The change will only diminish the ionic removal capabilities of the system. OP-003-031 still allows for precoating when the need arises, such as, startup or condenser tube leak. According to the safety evaluation no unreviewed safety questions are associated with this change.

139. <u>OP-010-001 (Change 2 Revision 15), General Plant Operations</u>

Description of Change

Change 2 to Revision 15 of this general operating procedure adds guidelines on ranges of T-cold for operating under the guidelines of the T-hot reduction program.

(Item #123 of this report also discusses changes related to the T-hot Reduction program.)

Reason for Change

The T-hot Reduction Evaluation Report discusses the benefits gained by reducing T-hot within the band allowed by the Technical Specifications. Reducing T-hot is expected to have a stong beneficial impact upon Steam Generator corrosion, thus eliminating or postponing the need for Steam Generator replacement.

Safety Evaluation

The safety evaluation concludes that since Waterford 3 will still be operating within the cold leg temperature range allowed per Technical Specification 3.2.6, the change remains consistent with the Waterford 3 Technical Specifications. As documented in the Safety Analysis Groundrules for Cycle 6, safety analyses assume an indicated T-cold range of 544 to 558 degrees F. at Hot Full Power. Waterford 3 authorized ABB/CE to assume a nominal RCS temperature program based upon a T-cold of 545 degrees F. at all power levels. thus, the change is consistent with Technical Bases. Since a nominal RCS temperature program based upon a 545 Degree F. T-cold at all power levels has been assumed for the Cycle 6 core, safety analyses have thereby confirmed that the consequences of all events will remain within established acceptance criteria throughout Cycle 6.

140. OP-903-030 (Revision 9), Safety Injection Pump Operability Verification

Revision 9 incorporates revised pump test loops to allow the pumps to be tested at a higher flow rate. Item 113 of this report discusses this information.

141. OP-903-119 (Revision 0), Secondary Auxiliaries Quarterly IST Valve Tests

Description of Change

This is a new surveillance procedure which separates the quarterly testing of secondary auxiliary systems valves from OP-903-032. The procedure also includes new tests for several valves. It implements the Inservice Testing described in the Waterford 3 Pump and Valve Inservice Test Plan, Revision 7, Change 1.

Reason for Change

The new surveillance procedure provides instructions to perform quarterly in-service testing of specific valves listed in the Waterford 3 Pump and Valve Inservice Test Plan. The procedure is also used to test valves following maintenance per the requirements of Technical Specification 4.6.3.1.

Safety Evaluation

The safety evaluation determined that the testing performed is consistent with the requirements of Section XI and the tests place the systems in a condition for which their safety function was designed to perform, there is no unreviewed safety questions.

142. OP-903-121 (Revision 0), Safety Systems Quarterly IST Valve Tests

Description of Change

This is a new surveillance procedure which separates the quarterly testing of safety systems valves from OP-903-032. The procedure also includes new tests for several valves. It implements the Inservice Testing described in the Waterford 3 Pump and Valve Inservice Test Plan, Revision 7, Change 1.

Reason for Change

The new surveillance procedure provides instructions to perform quarterly in-service testing of specific valves listed in the Waterford 3 Pump and Valve Inservice Test Plan.

Safety Evaluation

The safety evaluation determined that the testing performed is consistent with the requirements of Section XI and the tests place the systems in a condition for which their safety function was designed to perform, there is no unreviewed safety questions.

143. PE-004-019 (Revision 0), HECA Sampling

Description of Change

This new technical procedure provides instructions for obtaining a representative sample from HECA (High Efficiency charcoal Adsorber) vertical deep bed adsorbers of ESF and Non-ESF filtration units. Also,to refill sample cannisters and to provide laboratory samples when all sampling canisters have been depleted and previous analysis indicates the adsorbent material will pass future laboratory testing.

Reason for Change

Several ventilation system filters only have one sample remaining and some systems have none. Past history indicates the charcoal is still effective but current practice is to replace the charcoal once all samples are depleted. By utilizing this procedure, the plant will realize savings in manpower, materials and radioactive waste disposal.

The procedure will be utilized only when all HECA cannister samples are depleted and past sample data indicates that the adsorbent is still efficient and is expected to pass laboratory testing.

Safety Evaluation

The safety evaluation determined that there are no unreviewed safety questions associated with the new procedure. Utilizing the slotted tube sampling method, equivalent to installed sampler canisters, does not decrease the systems ability to function as designed. (ANSI N509-1989 endorses the use of the slotted tube method of sampling.)

144. QAP-012 (Revision 11.0), Quality Notice

Description of Change

QAP-012 is an administrative procedure of the Quality Assurance Department. This revision reflects a change in organizational responsibilites as a result of a revised Waterford 3 organization structure.

Reason for Change

Re-organization of the Waterford 3 organization resulted in the re-assignment of responsibilities for 10CFR21 reviews.

Safety Evaluation

The safety evaluation concluded that the revision does not create an unreviewed safety question or reduce the effectiveness of the Quality Assurance Program. The revision is to an administrative procedure and has no effect on accident probabilities or consequences.

145. <u>RF-004-001 (Revision 5), Reactor Vessel Head and Internals Removal</u>

<u>Description of Change</u>

This Refueling procedure is revised to reflect the use of a new Long Gripper Tool and a new CEA Extension Shaft Weighing Tool. Added a precaution to establish and verify communications between the Control Room and the refueling stations. A new step was added to verify at least 2 feet of clearance between the Reactor Vessel Head and alignment pins prior to any horizontal movement of the Reactor Vessel Head. And, a caution was added to clear all non-essential personnel from the +46 in containment prior to lifting the Upper Guide Structure.

<u>Reason for Change</u>

Use of the Long Gripper Tool and the CEA Extension Shaft Weighing Tool will allow for these activities to be accomplished from the Refueling Bridge. The Head clearance restriction prior to movement is intended to ensure all vessel internals are clear prior to horizontal movement. Clearing non-essential personnel from the +46 level will prevent unnecessary exposure to these personnel resulting in a lower overall dose.

<u>Safety Evaluation</u>

The safety evaluation reveals that use of the new tool only changes the steps required to couple and uncouple the CEAs and that the function of the tool remains the same. The other changes created by this revision do not raise any safety concerns or conflicts with information in the FSAR.

The design basis accident analyzes the dropping of a fuel assembly, the new tools addressed by the revision are used for coupling and uncoupling the CEA Extension shafts and or not involved in any lifting of the fuel assemblies.

146. RF-006-001 (Revision 5), Reactor Vessel Head and Internals Installation

Description of Change

Revision 5 of this Refueling procedures relocates information related to the Spent Fuel Pool Gates to another refueling procedure, adds information concerning the Long Gipper Operating tool and the CEA Weighing Tool. Steps were also added to allow the use of a Biach Stud Elongation Measurement System to measure stud elongation.

Reason for Change

Use of the Long Gripper Tool and the CEA Extension Shaft Weighing Tool will allow for these activities to be accomplished from the Refueling Bridge. Use of the new measurement tool will shorten the time required to measure stud elongation

Safety Evaluation

The safety evaluation reveals that use of the new tool only changes the steps required to couple and uncouple the CEAs and that the function of the tool remains the same. The other changes created by this revision do not raise any safety concerns or conflicts with information in the FSAR.

The design basis accident analyzes the dropping of a fuel assembly; the new tools addressed by the revision are used for coupling and uncoupling the CEA Extension shafts and or not involved in any lifting of the fuel assemblies.

147. RW-002-221 (Revision 0), Installation of Temporary Filter System to Remove Sludge From the Containment Sump

Description of Change

This is a new operating procedure which will change the normal flow path from the Containment Sump to the Liquid Waste Management (LWM) System waste tanks. The new flow path will bypass the in-line filter and use a remote filter system to transfer sump contents to the waste tanks. This procedure will only be implemented during plant outages.

Reason for Change

Bypassing the in-line filters, which do not allow for normal removal of waste in the Containment sump, and using a temporary filter unit will allow for sludge removal from the sump. The filtered water will be processed to the Waste Tank A. Use of this filter unit will only be implemented during outages.

Safety Evaluation

The safety evaluation determined that there is no unreviewed safety question. Two containment isolation valves are the only safety related components affected by the procedure. The valves are not impacted by the procedure because they are in the normal flow path from the containment sump to the LWM Waste Tank A. These two valves are the only protective boundary components affected; thus, the procedure does not reduce the margin of safety as described in the bases of the Technical Specifications.

148. <u>UNT-001-015 (Revision 3), Environmental Qualification Program</u>

See Items #104 and 105

149. UNT-005-013 (Revision 3), Fire Protection Program

Description of Change

This administrative procedure describes, delineates responsibilities, control and implementing requirements for the Fire Protection Program for Waterford 3. This revision provides clarification and simplification of controls associated with the Fire Protection Program.

Reason for Change

The revision is administrative only, incorporates editorial and human factors format revisions.

Safety Evaluation

The safety evaluation states that the level of fire protection previously approved by the Safety Evaluation Report and License is maintained consistent to achieve and maintain safe shutdown conditions following a fire.

150. UNT-005-028 (Revision 0), Maintaining Control Room Envelope Integrity During Maintenance Outages

Description of Change

This is a new plant procedure developed to enhance the administrative controls on all work activities and evolutions which have the potential to alter the Control Room HVAC system. The procedure also provides instructions for monitoring Control Room envelope integrity status and identifies the actions required to restore the Control Room to a sufficiently air-tight condition in the event of a Toxic Chemical Release.

Reason for Change

The procedure was developed to establish positive administrative controls on Control Room Envelope integrity during maintenance activities involving a breach of the Envelope. Intentional breaching of the Envelope is only allowed during Plant Modes 5 and 6 with all operations involving core alterations or positive reactivity changes suspended.

Safety Evaluation

The safety evaluation concluded that there are no unreviewed safety questions. The procedure does not affect any equipment important to safety and does not increase the probability of any accident evaluated in the FSAR.

151. UNT-006-010 (Revision 9), Event Notification and Reporting

Description of Change

This revision changes departmental responsibilities as a result of a revision to the Waterford 3 organization.

Reason for Change

The revision changes departmental responsibilities to match the revised organization structure of Waterford 3.

Safety Evaluation

The safety evaluation notes that this is a revision to an administrative procedure and has no effect on accident analyses.

152. UNT-007-052 (Revision 1). Outside Containment Temperature Trending Program

Description of Change

The revision to UNT-007-052 eliminates the 10CFR50.59 Safety Evaluation requirement to change the outside containment design basis temperatures. Actual operating temperatures determined in accordance with this procedure will be used as the basis of qualified life for equipment. (The accident design basis temperatures for outside containment (104 degrees F. for 120 days plus a 4 hour peak temperature if applicable) used to determine post-accident operability requirements on the affected equipment are not within the scope of this change and will not be affected.)

Reason for Change

The change will strengthen the EQ program by using actual operating temperatures as the basis of qualified life for equipment.

Safety Evaluation

At Waterford 3, the current outside containment EQ Baseline Temperature for normal operations is 104 degrees F. based on design calculations that were performed during plant construction. These design calculations included conservative estimates on outside air temperatures, heat sink temperatures, internal heat generation and the capability of the HVAC Systems. In accordance with this procedure temperature logger devices are strategically located to collect actual ambient temperatures EQ equipment is subject to during normal plant operations. Once a temperature profile is obtained and verified in the monitored area, the 104 degree F. EQ Baseline temperature will be revised and the qualified life of the affected EQ equipment will be updated in accordance with this procedure. The accident design basis temperatures for outside containment (104 degrees F. for 120 days plus a 4 hour peak temperature if applicable) used to determine post-accident operability requirements on the affected equipment are not within the scope of this change and will not be affected.

B. SPECIAL TEST PROCEDURES (STPs)

153. STP-01088375 (Revision 0), Air Side Seal Oil Filters

Description of Change

The STP will temporarily install instrumentation on the Main Generator Air Side Seal Oil filters to measure differential pressure and to obtain oil samples. The test will be performed with the filters in operation.

Reason for Change

The Main Generator Air Side Seal Oil filters have been clogging and giving a useful life of only about eight days before filter change out is required.

Safety Evaluation

The safety evaluation determined that the STP will not affect any accident scenarios currently analyzed in the SAR, no equipment important to safety will be affected by the test.

154. STP-01102098 (Revision 0) and STP-01108840, Special Test Procedure for SI-207A, B, & AB

Description of Change

The STP tests Safety Injection (SI) valves SI-207 A, B, & AB in the closed direction to satisfy ASME Boiler and Pressure Vessel Code, Section XI, Article IWV.

Reason for Change

A portion of the test requires that High Pressure Safety Injection (HPSI) Pump A and HPSI Pump AB to be cross connected through SE-212A and this represents a test which operates the system in an abnormal manner. the STP verifies that SI-207A, B, & AB will stroke closed when HPSI flow control valves are open in each train.

Safety Evaluation

The safety evaluation determined that no unreviewed safety question exists. The STPs implement the Inservice Test requirements of the Inservice Test Plan, Revision 7, Change 1. Inservice testing is required by Technical Specifications and the FSAR and is intended to improve the reliablility of components. The test requires that Technical Specification Limiting Condition for Operation 3.5.2, Action (a) be entered during the test of SI-201A due to the alignment of the AB electrical bus to the B train. However, this condition is allowed by Technical Specifictions for up to 72 hours and therefore does not represent an operation of the system in an abnormal manner.

155. STP-01102845 (Revision 0), Investigating Problems Associated With Containment Purge Isolation Radiation Monitor ARMIRE5027

Description of Change

The STP is developed for troubleshooting of Area Radiation Monitor (ARM) 5027 by swapping the detector and high voltage power supply cables with ARM 5026.

Reason for Change

The STP is to assist in determining the components that are causing the intermittent spiking of the Containment Purge Isolation Radiation Monitor, ARMIRE5027. It is expected that the problem is in the cabling between the detector and the RM-80. The signal and high voltage power supply cables will be swapped between ARMIRE5027 and ARMIRE5026.

Safety Evaluation

The safety evaluation notes that the equipment monitors radiation levels in areas of Containment, to isolate Contaiment Purge when the setpoint is exceeded. The two monitors affected by the STP will be out of service and no credit will be taken for them relating to Technical Specifications. There will still be operable Train A and B Channel Containment Purge Isolation Monitors (ARMIRE5024 and 5025), thus the Technical Specification requirements will continue to be met. Thus, there will be no reduction in any margin of safety as defined in the Technical Specifications.

The safety evaluation addressed the impact on separation criteria because of the switched cables and determined that it is acceptable. Switching the cables results in Channel B cables in a "A" Train penetration and Channel A cables in a "B" Train penetration. This configuration is only for a short duration and the cables affected are low energy instrumentation cables.

156. STP-01106975 (Revision 0), Installation/Removal of Spent Fuel Pool Suction Line Isolation Plug

Description of Change

The STP describes the installation, use, and removal of a pneumatic pipe plug in the suction line of the Spent Fuel Pool Cooling piping. This is a temporary means of isolation and will be removed immediately following valve repairs.

Reason for Change

The pneumatic plug will serve as a means of isolating the suction line of the pumps so that repairs can be made to the isolation valves (FS-101B and FS-106B) on Fuel Pool Cooling Pump "B."

Safety Evaluation

Use of the pneumatic plug requires a temporary configuration of the FS system that is not described in the SAR. The safety evaluation did not reveal any unreviewed safety questions concerning SAR evaluated accidents, malfunction of equipment important to safety or margin of safety.

A credible accident associated with this STP is the reduction of fuel pool level to elevation +40 feet. This accident is not listed with those as having been evaluated in the SAR. The accident has been evaluated, however, in the Failure Mode and Effects Analysis for the Fuel Pool System. The inherent compensating provision for this accident is stated as the short distance between the pool suction pipe and the water surface. This design limits draining to the bottom of the suction pipe.

The bases for Technical Specification 3/4.9.11 are based upon sufficient water depth to absorb the 10% iodine gap associated with a fuel pin rupture. The placement of the suction line of the FS cooling system precludes siphoning the pool level below elevation +40. Normal makeup to the pool would restore pool level. The accident scenario of the independent failure of both plug chambers, reduction of pool level, and the spontaneous failure of a fuel pin is not credible.

157. STP-01107285 (Revision 0), Isolation of Gland Seal Leakoff Tank

Description of Change

The STP will allow isolation of the Main Feedwater Pump Turbine Gland Seal Leakoff Tank from the Main Condenser in order to quantify the amount of dissolved oxygen (DO2) being injected into the Main Condenser from this tank.

Reason for Change

To determine the amount of DO2, the Gland Seal Leak-Off Tank will be isolated for approximately 15 minutes, the effluent from the tank will be discharged into the condensate pump pit.

Safety Evaluation

The safety evaluation states that Loss of Condenser Vacuum could occur because of the removal of a spool piece on the discharge of the Gland Seal Leakoff Tank Pump. However, to preclude this event, two valves downstream of the pump will be closed, isolating the condenser from the atmosphere.

The STP will not reduce the margin of safety as defined in the bases for technical specifications or any other safety analysis for safety related equipment. All protective boundaries will stay intact. Accident response and analysis of margin of safety remain unchanged by this STP.

158. STP-01113917 (Revision 0, Change 1), Test of "B" Containment Spray Header

Description of Change

Change 1 adds a flushing and venting section and also will stroke valve CS-125B at a higher pressure.

(See also Item #62 of this report)

Reason for Change

Addition of the flushing and venting section is to remove air from the header and pressure increase is to demonstrate that the valve will open with peak header pressure upstream of the valve.

Safety Evaluation

The safety evaluation notes that based on the results of the "Design Engineering Evaluation of Containment Spray Valve, CS-125B, Performance," the valve is capable of operating at upstream pressures as high as 500 psig with the Containment Spray riser level maintained in accordance with Technical Specification limits and the likelihood of failure of the valve will not be increased and will not increase the consequences of an accident.

The STP does not increase the likelihood of malfunction of the valve and the conditions will not increase the consequences of malfunction should it occur.

The evaluation determined that there are no unreviewed safety questions created.

159. STP-01114094 (Revision 0), Perform LLRT of SP-106

Description of Change

The STP allows local leak rate testing (LLRT) of SP-106 (Containment Sump Pump Discharge Header Outside Containment Isolation Valve) from the reverse direction through drain SP-107.

Reason for Change

Performance of the STP will allow LLRT retest of SP-106 to be completed at power to declare the valve operable and exit Technical Specification 3.6.3. Reverse testing of the valve has been determined to be equivalent to the normal direction of testing.

Safety Evaluation

The safety evaluation demonstrates that reverse pressure testing of SP-106 is acceptable. The 4 hour action statement of Technical Specification 3.6.3 provides adequate assurance the the likelihood of occurrence of a limiting fault accident during performance of the STP is acceptably small.

SP-106 is a bi-directional valve which can withstand differential pressure in either direction. Also, due to the seating mechanism of this diaphragm valve, a leak test in either direction will be representative of the leakage rate expected in both directions.

No margins of safety will be affected by this STP.

160. STP-01117875 (Revision 0), Component Cooling Water Discharge Check Valve Test (Also includes Revision 0, Change 2 and Revision 1)

Description of Change

This STP is to verify that the Component Cooling Water (CCW) pumps discharge check valves and the Dry Cooling Tower (DCT) outlet check valves are full open at the FSAR specified accident flow conditions. It also verifies that the CCW Pump "A" and "B" discharge check valves are full closed when the CCW Pump "AB" is aligned to the associated train.

Change 2 to Revision 0 addresses bypassing the Shutdown Cooling Heat Exchanger flow control valve to obtain the required 6554 gpm test flow.

Revision 1 of the STP verifies that CCW Pump "AB" discharge check valve is full open at the FSAR specified accident flow conditions, it also verifies that CCW Pump "B" discharge check valve is full closed when CCW Pump "AB" is aligned to the "B" Train.

Reason for Change

The inservice testing performed by this STP is required by the Waterford 3 Pump and Valve Inservice Test Plan program.

Safety Evaluation

The STP operates the CCW system in an abnormal manner. However, the safety evaluation determined that this STP does not represent an increase in the probability or consequences of accidents or malfunctions of equipment previously analyzed in the FSAR, nor does it represent the creation of a new accident or malfunction not previously analyzed in the FSAR. The evaluation determined that the margin of safety is maintained.

161. STP-255644-D (Revision 0, Change 1), Testing of Ambient Temperatures Inside Containment

Description of Change

The STP documents the results of in containment temperature monitoring during power operations. Containment temperature monitoring is to address the concerns of NRC Information Notices 87-65 and 89-30

Reason for Change

NRC Information Notices 87-65 and 89-30 addressed concerns pertaining to local ambient temperatures above the Design Basis temperatures identified in the FSAR. This STP will document the temperature monitoring accomplished to address those concerns.

Safety Evaluation

The safety evaluation determined that the containment temperature monitoring will not impact the intended functions of the monitored equipment. The primary coolant system pressure boundary or any other boundaries will not be breached in any fashion. RTDs used for monitoring are not electrically or physically interlocked with any other existing plant electrical or I&C circuits. They are positioned and secured not to impact the function or have a potential to adversely affect a safety function of any plant equipment.

162. STP-289682 (Revision 0), Instrument Air System Leakage Test

Description of Change

The STP will determine if Station Air System (SA) valves, SA-126 and SA-127 can be relied upon to isolate the Instrument Air (IA) system while SA is secured to implement a design change.

Reason for Change

SA supplies backup air to the IA system. The STP will allow the SA system to be secured and depressurized to determine the capability of the IA system to maintain normal pressure using SA-126 and SA-127 as isolation. This configuration will be used during implementation of a SA design change.

Safety Evaluation

The results of the safety evaluation were that no unreviewed safety question exists. The IA system is not safety related and is not required for safe shutdown of the plant or for limiting radiological releases. The IA system does provide air to safety related valves in the Component Cooling and Containment Vacuum Relief systems. These valves have accumulators to ensure that they can perform their safety functions on loss of air. The STP will be secured and the system returned to normal if the IA system pressure falls below 100 psig.

163. STP-99000376 (Revision 0, Change 1), Acceptance Test for DC-3031

Description of Change

The STP verifies system operation upon completion of DC-3031. See Item 2 of this report.

164. STP-99000415 (Revision 0), Dynamic Test of ECCS Recirculation Valves

Description of Change

The STP performs dynamic testing of Safety Injection valves SI-120A & B, and SI-121A & B.

Reason for Change

The STP is being performed to satisfy Entergy Operations' commitment to comply with Generic Letter 89-10.

Safety Evaluation

The safety evaluation notes that the testing will be performed during Mode 5 or 6, the High Pressure Safety Injection (HPSI) pumps cannot be at shutoff head during a LOCA. Therefore the recirculation lines do not have to perform their function, i.e., prevent pump overheating. However, the test line-up will not prevent this function. The amount of flow diverted by the recirculation lines is limited by the recirculation orifice, thus the number of open valves will have little or no effect on recirculation flow.

There is no reduction in the margin of safety because in Modes 5 and 6 the recirculation lines do not have to prevent overheating of the HPSI pumps and the test line-up will not prevent recirculation flow.

165. STP-99003380 (Revision 0), EDG "B" Fuel Oil Transfer Pump Flow Instrumentation and STP-99003380A (Revision 0), EDG "A" Fuel Oil Transfer Pump Flow Instrumentation

See Item #42 of this report for a discussion of DC-3380 which installed flow instrumentation for EDG "A" and "B."