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

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

Gentlemen:

Enclosed is the Report of Facility Changes, Tests and Experiments for Waterford 3, which is submitted pursuant to 10CFR50.59. This report covers the period from June 1, 1999 through May 31, 2000. This letter does not contain commitments.

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

Very truly yours,

Everett P. Perkins Jr

E.P. Perkins, Jr.
Director
Nuclear Safety Assurance

EPP/LBB/rtk
Enclosure: 50.59 Summary Report

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

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

# REPORT OF FACILITY CHANGES, TESTS, AND EXPERIMENTS

# PER 10CFR50.59

# JUNE 1, 1999 THROUGH MAY 31, 2000

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

JUNE 1, 1999 THROUGH MAY 31, 2000

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## SUMMARY

This report provides the Waterford 3 Facility Changes made pursuant to 10CFR50.59(a)(1). The report covers the period from June 1, 1999, through May 31, 2000. None of the items in the report were found to involve an unreviewed safety question.

Section I identifies acronyms used in the Report.

Section II of the report identifies 91 Facility Changes which consist of: 6 Design Changes (DCs), 1 Temporary Alteration Request (TAR), 14 FSAR Changes, 5 Miscellaneous Evaluations, 41 Engineering Requests (ERs) and 24 Commitment Changes.

Section III of the report identifies 5 Procedure Changes which consist of: 5 Plant Procedures.

## I. LIST OF ACRONYMS

| ACRONYM | DEFINITION |
|---|---|
| AB | Auxiliary Boiler |
| ACCW | Auxiliary Component Cooling Water |
| ALARA | As Low As Reasonably Achievable |
| ANSI | American National Standards Institute |
| ARM | Area Radiation Monitor |
| ASME | American Society of Mechanical Engineers |
| ATS | Anticipated Transient System |
| ATWS | Anticipated Transient Without Scram |
| BD | Blowdown |
| BM | Boron Management |
| CA&A | Corrective Action and Assessment |
| CARB | Corrective Action Review Board |
| CBC | Critical Boron Concentration |
| CCW | Component Cooling Water |
| CD | Condensate |
| CE | Combustion Engineering |
| CHW | Chilled Water |
| CIAS | Containment Isolation Actuation Signal |
| CIV | Close Intercept Valve |
| COLR | Core Operating Limits Report |
| COLSS | Core Operating Limits Supervisory System |
| CMS | Commitment Management System |
| CMU | Condensate Makeup and Storage |
| CR | Condition Report |
| CRG | Condition Review Group |
| CROAI | Control Room Outside Air Intake |
| CS | Containment Spray |
| CSAS | Containment Spray Actuation Signal |
| CSP | Condensate Storage Pool |
| CST | Condensate Storage Tank |

| ACRONYM | DEFINITION |
|---|---|
| CVAS | Controlled Ventilation Area System |
| CVC | Chemical and Volume Control |
| CW | Circulating Water |
| DBA | Design Basis Accident |
| DBD | Design Basis Document |
| DCP | Design Change Package |
| DCT | Dry Cooling Tower |
| DE | Design Engineering |
| DEAM | Design Engineering Administrative Manual |
| DEFAS | Diverse Emergency Feedwater Actuation System |
| DEH | Digital Electro-Hydraulic Control System |
| DP | Differential Pressure |
| DRN | Document Revision Notice |
| DRTS | Diverse Reactor Trip System |
| DW | Demineralized Water |
| DWST | Demineralized Water Storage Tank |
| ECCS | Emergency Core Cooling System |
| EDG | Emergency Diesel Generator |
| EFAS | Emergency Feedwater Actuation Signal |
| EFW | Emergency Feedwater |
| EOI | Entergy Operations Inc. |
| EOP | Emergency Operating Procedures |
| EPRI | Electric Power Research Institute |
| EQ | Environmental Qualification |
| ER | Engineering Request |
| ERC | Engineering Request Change |
| ERCN | Engineering Request Change Notice |
| ERD | Engineering Request Database |
| ESF | Engineered Safety Features |
| ESFAS | Engineered Safeguards Features Actuation System |
| FHA | Fire Hazards Analysis |
| FHB | Fuel Handling Building |

| ACRONYM | DEFINITION |
|---|---|
| FMEA | Failure Modes and Effects Analysis |
| FOST | Fuel Oil Storage Tank |
| FSAR | Final Safety Analysis Report |
| FWIV | Feedwater Isolation Valve |
| FWPT | Feedwater Pump Trip |
| GDT | Gas Decay Tank |
| GL | Generic Letter |
| GMPO | General Manager Plant Operations |
| GV | Governor Valve |
| GWM | Gaseous Waste Management |
| HPSI | High Pressure Safety Injection |
| HVAC | Heating, Ventilation and Air Conditioning |
| HVC | Control Room Heating, Ventilation & Air Conditioning |
| I&C | Instrumentation and Control |
| IA | Instrument Air |
| IEEE | Institute of Electrical and Electronic Engineers |
| INI | Incore Nuclear Instrumentation |
| ISEG | Independent Safety Engineering Group |
| ISI / IST | Inservice Inspection / Inservice Testing |
| ITR | Independent Technical Review |
| IV | Intercept Valve |
| IWS | Industrial Waste Sump |
| LBD | Licensing Basis Document |
| LCO | Limiting Condition for Operation |
| LCP | Local Control Panel |
| LDA | Load Drop Anticipation |
| LDCR | Licensing Document Change Request |
| LER | Licensee Event Report |
| LOCA | Loss of Coolant Accident |
| LOOP | Loss of Offsite Power |
| LP&L | Louisiana Power & Light |
| LPDES | Louisiana Pollutant Discharge Elimination System |

| ACRONYM | DEFINITION |
|---|---|
| LPSI | Low Pressure Safety Injection |
| LPZ | Low Population Zone |
| LWM | Liquid Waste Management |
| LTC | Long Term Cooling |
| LTOP | Low Temperature Overpressure Protection |
| M/U | Make Up |
| MCC | Motor Control Center |
| MFIV | Main Feedwater Isolation Valve |
| MMIS | Material Management Information System |
| MNSA | Mechanical Nozzle Seal Assemblies |
| MOV | Motor Operated Valve |
| MPM | Mandatory Preventive Maintenance |
| MR | Maintenance Rule |
| MSIS | Main Steam Isolation Signal |
| MSIV | Main Steam Isolation Valve |
| MSL | Mean Sea Level |
| MSLB | Main Steam Line Break |
| MTC | Moderator Temperature Coefficient |
| NEI | Nuclear Energy Institute |
| NFPA | National Fire Protection Association |
| NPDES | National Pollutant Discharge Elimination System |
| NPSH | Net Positive Suction Head |
| NRA | Nuclear RTD Amplifier |
| NSA | Nuclear Safety Assurance |
| OEM | Original Equipment Manufacturer |
| OPC | Overspeed Protection Control |
| PCRS | Paperless Condition Reporting System |
| PDMS (CCL) | Cable and Conduit Listing |
| PM | Preventive Maintenance |
| PMC | Plant Monitoring Computer |
| PMP | Probable Maximum Precipitation |
| PPPM | Performance Prediction Program Methodology |

| ACRONYM | DEFINITION |
|---|---|
| PQD | Parts Quality Determination |
| PWST | Potable Water Storage Tank |
| QA | Quality Assurance |
| QAPM | Quality Assurance Program Manual |
| QR | Quality Reviewer |
| RAB | Reactor Auxiliary Building |
| RCA | Root Cause Analysis |
| RCP | Reactor Coolant Pump |
| RCS | Reactor Coolant System |
| RF | Refueling |
| RG | Regulatory Guide |
| RHSV | Reheat Stop Valve |
| RP | Radiation Protection |
| RPCS | Reactor Power Cutback System |
| RPS | Reactor Protection System |
| RPV | Reactor Pressure Vessel |
| RT | Repetitive Task |
| RTD | Resistance Thermal Detector |
| RWSP | Refueling Water Storage Pool |
| SAR | Safety Analysis Report |
| SBLOCA | Small Break Loss of Coolant Accident |
| Scfm | Standard cubic feet per minute |
| SDC | Shut Down Cooling |
| SDCHX | Shut Down Cooling Heat Exchanger |
| SER | Safety Evaluation Report |
| SG | Steam Generator |
| SGBDS | Steam Generator Blowdown System |
| SI | Safety Injection |
| SIAS | Safety Injection Actuation Signal |
| SIS | Safety Injection System |
| SPEER | Spare Part Equivalency Evaluation Report |
| SPS | Standard Project Storm |

| ACRONYM | DEFINITION |
|---|---|
| SRV | Safety Relief Valve |
| SSC | Structure, System, or Component |
| SSE | Safe Shutdown Earthquake |
| SUPS | Static Uninterruptible Power Supply |
| SUT | Startup Transformer |
| SVD | Secondary Makeup Vacuum Degasifier |
| TAR | Temporary Alteration Request |
| TRM | Technical Requirements Manual |
| TS | Technical Specifications |
| TSC | Technical Support Center |
| TSS | Main Transformers and Switching Station |
| TV | Throttle Valve |
| UHS | Ultimate Heat Sink |
| USQ | Unreviewed Safety Question |
| WCT | Wet Cooling Tower |
| WMS | Work Management System |
| Y2K | Year 2000 |

## II. FACILITY CHANGES

### A. DESIGN CHANGES

#### 1. DC-3521, Route DCT Sumps Discharge to Circulating Water System, Revisions 1 and 4

DESCRIPTION OF CHANGE

Revision 1 – Design Change Package DCP 3521 Revision 0 installed an 8" valve as a tie-point to the Circulating Water (CW) system. This connection required a system outage and was installed during RF8. DCP 3521 Revision 1 reroutes the discharge piping from Dry Cooling Tower (DCT) Sump A and connects it to the discharge piping from DCT Sump B. This change allows both sump pumps to discharge to the 40 Arpent Canal via a drainage ditch on the east side of the plant. In addition, DCP 3521 Revision 1 installs new piping and isolation valves, connected to the new branch connection to the CW system added by DCP 3521 Revision 0, to later allow both sump pumps to discharge to the CW system. DCP 3521 Revision 1 does not authorize opening of valve CW-421 or alignment of the DCT sump pumps to the CW system except for testing activities. The final phase of the design change will be issued following completion of a new ponding analysis which will allow alignment of both sumps to the CW system as the primary discharge path.

The single connection to the 40 Arpent Canal will be maintained to allow for maximum capacity discharge during a Probable Maximum Precipitation (PMP) event, or during periods of a Circulating Water System outage. Currently, the sump pumps discharge can be directed to the Liquid Waste Management (LWM) waste tanks if high radiation is detected and this aspect of the system design will not be changed. DCP 3521 Revision 1 installs approximately 400' of 6" piping, approximately 150' of 8" piping, and three pipe penetrations through the exterior floodwall of the Dry Cooling Tower area. This modification adds a discharge point to NPDES Permit LA0007374, thus requiring an Environmental Impact Evaluation.

Revision 4 - As a result of a new ponding analysis by calculation EC-M99-010, Revision 4 to the DC is issued to revise the licensing basis to reflect that: 1) a total pumping capacity of 600 gpm is needed in each DCT area during the PMP rainfall event to prevent partial submergence of safety related equipment; 2) a diesel powered sump pump with a minimum capacity of 300 gpm is provided in each DCT area to supplement the motor driven sump pumps and provide a total pumping capacity in excess of 600 gpm during the PMP rainfall event; 3) the diesel powered sump pumps must be started during the first 3 hours of a PMP event; 4) a pumping capacity of 300 gpm is needed in each DCT area during the Standard Project Storm (SPS) event to prevent partial submergence of safety related equipment; 5) the diesel powered sump pumps are credited during the SPS event and must be started within the first 3 hours of the event; 6) all operating restrictions related to the use of a new alternate discharge path from the DCT area sumps pumps to the CW system have been removed; 7) the single 100 gpm diesel powered sump pump is no longer credited for the SPS rainfall event and will be removed from the licensing basis; and

8) operation of the motor-driven sump pumps is not credited following a seismic event and will be removed from the licensing basis.

REASON FOR CHANGE

Revision 1 - Tube leaks in the Letdown Heat Exchanger and normal testing of the ACCW/CCW interface resulted in low levels of radioactive material in the Wet Cooling Tower basins, where over-spray and overflow resulted in discharge to the 40 Arpent Canal. This discharge path could create an ingestion pathway to the public. Current dose reporting requirements allow little credit for dilution, therefore requiring the setpoints of the radiation monitors to be lowered to enable detection. Upon lowering of the radiation monitor setpoints, alarm activations increased due to the natural occurring radionuclides from the large concrete surface area of the Dry Cooling Towers. The new discharge path to the Circulating Water system will allow credit for dilution and may allow the setpoints on the radiation monitors to be raised to eliminate nuisance alarms.

Revision 4 - A new discharge path to the CW system was to be created to allow significantly greater credit for dilution and therefore allow the setpoints on the radiation monitors to be raised to eliminate nuisance alarms. During implementation of DC-3521, calculation EC-M99-010 was prepared which reflected that the pumping capacity provided in the DCT areas was inadequate to prevent ponding rainwater from partially submerging safety-related equipment during the PMP and SPS events. In addition, calculation EC-M97-029 reflected that the capacity of the DCT area sump pump motor-driven sump pumps would be reduced from 325 gpm to approximately 300 gpm if the discharge were aligned to an inoperable CW system (most restrictive alignment). Therefore, use of the new discharge path could not be used until the new ponding analysis provided justification for the reduced capacity during CW system outages.

50.59 EVALUATION

Revision 1 - This 50.59 Safety Evaluation is for changes associated with DCP-3521 Revision 1, which authorizes physical changes to the plant necessary to combine the discharge path from both DCT sump pumps to a single discharge into the site drainage system. This evaluation also authorizes installation of piping and valves which will later be used to divert DCT sump pump flow to the CW system. A separate 50.59 Safety Evaluation will later evaluate the changes associated with discharging to the CW system.

Key issues evaluated related to the implementation of the design include: 1) provisions to maintain or quickly restore the integrity of the flood wall should a levee failure or tornado occur when the flood wall integrity is degraded during installation of three new pipe penetrations; 2) provisions to ensure that internal flooding does not occur while the sump pumps are inoperable during modification activities; 3) scheduling and sequencing of critical work activities to ensure that both trains of the dry cooling towers are affected for the shortest period of time while making final system tie-ins.

Because the function, capacity and operation of the DCT sump pumps will not be changed by the proposed modifications, this evaluation reflects that the proposed changes by DCP-3521 Revision 1 will not reduce the margin of safety as defined in the basis of any TS or safety analysis and no USQ is created.

Revision 4 – This evaluation focused on the results of calculation EC-M99-010 Revision 1, which established a new design basis for the portions of the Sump Pump system which protects safety related equipment in the DCT areas from ponding rainwater. This calculation documented the maximum potential depth of ponding rainwater for the PMP and SPS rainfall events, utilizing: 1) Regulatory Guide 1.59 requirements to establish rainfall intensities and duration, 2) current design drawings for establishing open areas, contributing areas or overflow from adjoining roofs, and available ponding areas, 3) plant walkdowns to verify design information and identify temporary materials stored in the DCT areas which reduce available critical dimensions, 5) previously allowed operator action to start motor driven pumps within 30 minutes of a Loss of Offsite Power (LOOP) event coincident with the PMP, 6) previous licensing basis assumptions that two of the four motor driven sump pumps are unavailable during the PMP, 7) previous licensing basis assumption that only a single diesel powered sump pump is available (in each DCT area) following a seismic event, and 8) input from calculation EC-M97-029 regarding pump performance when aligned to either an operable of inoperable CW system.

This 50.59 evaluation reflects that the existing motor-driven sump pumps, with a minimum capacity of 300 gpm, and supplemented with a diesel-powered sump pump with a minimum capacity of 300 gpm in each train, are capable of protecting safety-related equipment in the DCT areas during either the PMP or SPS rainfall events. The new design basis minimum pumping requirements assume that two of the three pumps are available in each train for the PMP, and that one of the three pumps is available in each train during the SPS. Therefore, no USQ is created and the licensing basis can be revised to reflect the new design basis and the results of the new ponding analysis.

2. DC-3526, EFW Heat Trace Reliability Improvements, Revision 2

DESCRIPTION OF CHANGE

DC-3526 was initiated to replace all of the EFW heat trace which was found to be in a degraded state. This DC was installed during RFO8. Revision 1 incorporates changes to prepare for package closure. Revision 2 removes circuit 1-8C from the temperature monitoring panel and replaces it with the temperature indication for the piping associated with circuit 1-8D.

REASON FOR CHANGE

The proposed change will eliminate the occasional low temperature conditions experienced for the steam supply piping associated with circuit 1-8C from initiating the local and control room alarms.

50.59 EVALUATION

The EFW Heat Trace Temperature Maintenance system is not specifically credited in any FSAR accident analysis but is required to prevent any adverse affects on the EFW Terry Turbine. Replacing the monitoring of circuit 1-8C with 1-8D will not affect operation of the turbine. The piping associated with circuit 1-8C has a slight natural slope from the trip and throttle valve back down to the inlet steam line drip pot. Thus this section is not capable of holding any substantial condensate. Both sections of pipe are short and have only a minor effect on the condensate loading during system startup. All of the existing heat trace cables currently in service on the various sections of the Terry Turbine steam supply piping will remain in service. There will be no adverse affect on previously evaluated accident or equipment probability or consequences. No new system interconnections are required and no new accident or equipment malfunction will be created. No margin of safety or protective boundary will be affected by this change and no USQ exists.

3. DC-3529, Remote Manual Operating Capability for CVC-209, Revisions 0 and 1

DESCRIPTION OF CHANGE

DC-3529 will upgrade the existing CVC-209 valve to provide for Train A/B Class 1E power and control circuit. The existing electrical and pneumatic components will be replaced on the valve and control board with qualified Class1E components. EQ seals will be added to the limit switches to enhance reliability. All cables from the valve to the control room will be replaced with Class 1E qualified cable and routed as to conform to RG 1.75 separation criteria. Provisions are also provided for backup air connection to valve CVC-209 from the existing Instrument Air (IA) accumulators and high pressure essential air bottles installed in the switchgear room on +21 elevation to enhance reliability. IA connections from the valve to the IA header will be relocated on the same IA header to improve constructability and ensure availability of air to the valve actuator should the non-safety IA be unavailable.

REASON FOR CHANGE

CVC-209 is designated as an Essential system not requiring automatic containment isolation. Per SER Section 6.2.4, the NRC review of the essential lines found this to be acceptable with provisions in the design for remote manual operation. The closure function of CVC-209 is credited in FSAR Section 6.2.4.1.2 and in the response to FSAR Question No 480.43. Should the CVC system malfunction or if containment isolation of the penetration is required, the valve should be capable of remote closure from the control room. This closure feature is presently supplied with air from the non-safety IA system and electrical power from a non-Class 1E power source. This valve is not equipped with air accumulators. The non-safety IA system and/or the non-safety electrical power may not be available to close CVC-209 post-accident. This change will enhance the reliability of the remote manual closure function for CVC-209 to ensure isolation capability for a 30-day post-accident period. Additionally, RG 1.97 classifies position indication for containment isolation valves as Category 'B1' and requires full qualification for the circuits and devices used for position indication. Waterford 3 committed to the NRC to upgrade the position indication.

50.59 EVALUATION

The Charging portion of CVCS, which includes valve CVC-209, is an Essential system credited in mitigating the effects of the Small Break Loss of Coolant Accident (SBLOCA) described in FSAR Section 15.6.3. Valve CVC-209 is designed to fail open and is normally locked open to ensure its availability during all modes of operation. TS requires verification of valve position every 31 days. The upgrades proposed in DC-3529 are designed to enhance the reliability and ensure the closure capability of the valve once Charging is secured post-accident. The change will provide additional assurance of CVC-209 valve functionality as described in FSAR Section 6.2.4.1.2 and in response to FSAR Question No. 480.43. Additionally,

position indication will be upgraded and qualified in accordance with RG 1.97. DC-3529, which upgrades the CVC-209 valve electrically and provides a backup Essential Air source for motive power and does not change the accident or failure analyses accounted for in the licensing basis. The changes add a minor load (< 0.5 amp) on Battery 3AB-S, however, the total battery load as stated in FSAR Table 8.3-5 remains the same due to inherent margin established when calculating the battery load profile. The function of valve CVC-209 is not changed from what is currently described in the plant's licensing basis. Therefore, the modification does not reduce the margin of safety and does not result in a USQ.

4. DC-3531, Dry Cooling Tower Pressure Equalization Bypass, Revision 1

DESCRIPTION OF CHANGE

The proposed change will install a manual bypass around check valves CC-181A and B. These bypass lines will be used to realign the Dry Cooling Towers (DCTs) after being isolated and will allow the DCT to equalize in pressure prior to opening the DCT inlet isolation valves, CC-135A(B). The bypass lines and valves are 2", exposed to the atmosphere, have no flow, and therefore are required to be protected from freezing. Freeze protection will be applied from panel FP2-1, circuits 6 and 7.

REASON FOR CHANGE

When the DCT is isolated, the water in the tubes will cool until it reaches ambient temperature. When the DCT is bypassed, and the water in the DCT is allowed to cool, a void can develop in the DCT. This could cause a water hammer to occur when the DCT is placed back in service. Freeze protection to the bypass valve and line will enhance the availability of the CCW system by preventing freezing and rupture during extreme cold weather conditions.

50.59 EVALUATION

The CCW system does not initiate any accidents; therefore, these changes will not increase the probability of any accident described in the FSAR. There are no accidents in the FSAR which postulate radioactive release consequences from the CCW system. To prevent the possibility of valve misalignment and leakage back to the DCT, valves CC-181A(B) will be administratively controlled in the locked position. The new bypass line is also designed to ASME Section III, Class 3, Seismic I requirements. Therefore, no equipment malfunction probability or consequences will be increased. All of the changes will be limited to the CCW system and will not create either a new accident or equipment malfunction. No protective boundary or margin of safety will be affected by this change. This change does not represent an unreviewed safety question.

5. DC-3537, CCW Train 'B' Radiation Monitor Piping, Revision 0

DESCRIPTION OF CHANGE

The proposed change reroutes the Train 'B' CCW process radiation monitor cooling water line from its current connection between CC-200A and CC-200B to a location upstream of CC-200B.

REASON FOR CHANGE

With the current configuration, in the event of a Safety Injection Actuation Signal (SIAS) without a Containment Spray Actuation Signal (CSAS), CCW Train 'A' would flow through line 3CC1-328B into CCW Train 'B', thus breaching separation. Additionally, for a CSAS with SIAS, all cooling water would be lost to the Train 'B' CCW process radiation monitor resulting in possible loss of function.

50.59 EVALUATION

There are no accidents previously evaluated in the FSAR that can be initiated by the CCW system. CCW does provide a cooling support function for other SSCs required in the mitigation of accidents. The proposed change will increase the reliability of the CCW system by maintaining train independence and integrity. Therefore, no accident or equipment malfunction probability or consequences will be increased. Rerouting the CCW process radiation monitor cooling flow does not create any new system interactions, but rather eliminates the potential for cross connecting the two CCW trains following a SIAS with no CSAS. No margin of safety for CCW is reduced by this change. This change does not represent an unreviewed safety question.

6. DC-3552, Add Desiccant Filler/Breathers to MFIV Actuator Hydraulic Reservoirs, Revision 0

DESCRIPTION OF CHANGE

This change replaces the existing breather/filler cap with a desiccant filler/breather on the atmospheric vent on each of the Feedwater Isolation Valve (FWIV) actuator hydraulic reservoirs. The new desiccant filler/breather will be non-safety and seismically mounted. The new desiccant filler/breather will be connected to the reservoir with a new steel adapter flange similar to the existing one. Easy access exists for the operators/maintenance to monitor and change out the desiccant bag filter when it becomes dirty/saturated (desiccant changes color when dirty). Anchor/Darling (original OEM) will supply the desiccant filler/breather and adapter as an actuator sub-component qualified to the original valve specification requirements. A qualification test was performed by Anchor/Darling and concluded that the new desiccant breather will not impact the safety or failure mode closure time of the FWIVs by creating a backpressure within the hydraulic reservoir greater than that of the original equipment reservoir filler cap.

REASON FOR CHANGE

The presence of water in the hydraulic fluid in the FWIV reservoir can contribute to the formation of gel in the fluid. If enough of this gel forms, the viscosity of the hydraulic fluid will increase and have an adverse affect on the stroke time of the FWIVs. The hydraulic reservoir for the FWIVs is vented to the atmosphere, which allows moisture to enter the tank through the breather cap.

50.59 EVALUATION

Replacing the existing breather/filler cap with a desiccant filler/breather on the atmospheric vent on each of the FWIVs actuator hydraulic reservoirs will not constitute a USQ. The change will increase the reliability of the FWIVs by reducing the potential for gel formation due to moisture entering the hydraulic reservoir and contaminating the hydraulic fluid. The hydraulic reservoirs for the Feedwater Isolation Valves (FWIV) are not initiators of any limiting accidents. Therefore, this modification will not increase the probability of occurrence of an accident previously evaluated in the SAR. Table 3.3-5 of the TRM, Engineered Safety Features Response Times, requires main feedwater isolation to occur in less than or equal to 6.0 seconds. This time limit exists to limit the mass and energy released to the containment during a postulated MSLB incident. Assuming a 1.0 second signal processing time, valve closure must be achieved in 5.0 seconds of receipt of the isolation signal (FSAR section 10.4.7). The OEM performed a qualification test, assuming worst case conditions, on the new desiccant filler/breather (a sub-component of the FWIV actuators). The test report confirmed that the backpressure caused by the addition of the desiccant breather will not affect the closing time of the FWIVs. Since the FWIVs will continue to close within 5.0 seconds, this change is bounded by the current accident analysis in the FSAR. Therefore, radiological consequences of the analyzed incidents remain unchanged. The safety function of

the FWIVs is to close within 5.0 seconds on a MSIS. The hydraulic reservoir is not needed for this function since the motive force comes from the stored hydraulic pressure in the safety related accumulators. Vendor qualification testing was satisfactorily performed verifying that the FWIVs closing time is not affected due to the backpressure induced by a worst case flow condition and a dirty desiccant breather. Further the qualification verified that during the worst case backpressure scenario in the hydraulic reservoir, integrity of the reservoir would be maintained. The vendor documented in the Seismic qualification Certificate of Conformance that the added weight of the desiccant breather, even if amplified by the added flexibility, is deemed insignificant compared to the total actuator weight and the desiccant breather would remain intact which is its only function during a seismic event. The new desiccant filler/breather is qualified for normal and accident conditions (temperature, humidity, and radiation field) expected in the RAB, Zones "B"-Temp and "DD"-Rad., over the 40 year life of the plant. The vendor documented that the desiccant material is compatible with the hydraulic fluid and the potential for the desiccant material in the bag filter getting out of the breather is minimal. Further, if the desiccant material were to get out of the desiccant breather a strainer exists in the reservoir vent to prevent the desiccant material from entering the reservoir. This change will enhance the FWIV actuators by reducing the moisture entering the hydraulic reservoir and contaminating the hydraulic fluid; and will not inhibit valve closure. Therefore, this activity does not increase the probability or consequences of a malfunction of equipment important to safety previously evaluated in the FSAR. Adequate barriers are in place to prevent the desiccant material from the new breather from getting into the reservoir. Further the desiccant breather vendor has documented that interaction between the desiccant material and the hydraulic fluid will have no adverse affect on the hydraulic fluid. Further, the hydraulic reservoirs for the FWIVs are not initiators of any limiting accidents. Therefore, this activity could not create the possibility of an accident of a different type than previously evaluated in the FSAR. Since the FWIVs will continue to close within 5.0 seconds, this change is bounded by the current accident analysis in the SAR. Since this activity will enhance the FWIV actuators by reducing the moisture entering the hydraulic reservoir and contaminating the hydraulic fluid, the margin of safety will not be reduced.

B. TEMPORARY ALTERATION REQUEST (TAR)

1. TAR-99-005, Reactor Coolant System Loop 2 Hot Leg Temperature Indication

DESCRIPTION OF CHANGE

This TAR will restore the Reactor Coolant System (RCS) Hot Leg temperature indication to Remote Shutdown Panel LCP-43. Currently, RCS Hot Leg temperature to LCP-43 is configured such that RCS Hot Leg 1 is on Channel B, and RCS Hot Leg 2 is on Channel A. Following installation of this TAR, RCS Hot Leg 1 will be on Channel A, and RCS Hot Leg 2 will be on channel B. This change is necessary to maintain the electrical separation of the remote shutdown panel, and provide for both RCS Hot Leg indications. The NRA Card for loop RC IT0122 HA2 will be pulled from the cabinet. This will "zero" the input to the Saturation (Subcooled) Margin Monitor.

REASON FOR CHANGE

RTD RC ITE0122 HA2 has failed. This RTD feeds a signal to LCP-43, which is the Remote Shutdown Panel, and the Saturation (Subcooled) Margin Monitor. LCP-43 is required to have indication from the RCS Hot Leg according to TS 3.3.3.5.

50.59 EVALUATION

The reassignment of Reactor Coolant hot leg temperature for LCP-43 has no affect on the safe shutdown and accident monitoring of the plant, nor does this TAR have an effect on the margin of safety of the plant as defined in the TS and LBDs. LCP-43 will be supplied with redundant indication of hot leg temperature that meets the separation requirements of RG1.75. This change impacts LCP-43 and components RC ITE01 22 HA2/HB2, RC ITE01 12 HA2/HB2. None of these SSC's can initiate an accident previously evaluated in the SAR. FSAR table 15.0-2 Initiated Events remains valid and unaffected. The proposed change will provide the operator with a redundant and independent indication of hot leg temperature as required by TS 3.3.3.5. This assures sufficient instrumentation is provided outside of the main control room to achieve prompt hot shutdown, maintain the unit in a safe condition during a hot shutdown, and achieve cold shutdown. As all required remote shutdown indications are provided, the consequences of an accident are unchanged. Per Table 3.2-1 and Chapter 7 of the FSAR, equipment important to safety affected by this configuration change includes the hot leg temperature indication on LCP-43 for both hot legs. The temperature indication on LCP-43 for the hot legs will be provided by safety class, seismic 1, environmentally qualified temperature elements. The re-assigning of the input signals to a different loop does not increase the probability of occurrence of a malfunction of equipment important to safety. The inputs will be provided by qualified instrumentation that has been designed to the same standards as the original installation. The instrumentation will meet the design criteria of FSAR Section 7.4. The hot leg instrumentation is not a

protective system as defined by IEEE-279-1971, however many of the criteria of IEEE-279-1971 have been incorporated in the design. The hot leg instrumentation on the auxiliary control panel conforms to IEEE-308-1971. No credible single failure will prevent safe shutdown, even in the event of a loss of offsite power. Channel independence is maintained by electrical and physical separation between redundant channels per RG1.75. The TAR will use alternate equipment already installed in the plant that will be rewired to connect the instrumentation to the appropriate indicators. The instrument's seismic classification is unaffected. Single failures include electrical faults and physical events such as missiles and fires. Hot leg instruments will be supplied by CP-61 and CP-62, which have redundant and independent power supplies. The TAR will not prevent the safe shutdown of the plant with a single failure of a component. This TAR has created no new system interactions that did not previously exist; thus, the possibility of a different accident is not created. The loss of indication to the plant site has been previously evaluated in the SAR and is bounded by a LOOP. The instrumentation required for the safe shutdown of the plant are designed and arranged so that no single failure can prevent a safe shutdown. This is achieved by electrical and physical separation of the circuits. No new credible failure modes are provided. No new system interactions are created. This TAR will not create the possibility of a malfunction of equipment important to safety of a different type than any previously evaluated in the SAR. This TAR will not affect the protective boundary or margins of safety of the plant as defined in the TS, or in any LBD.

## C. FSAR CHANGES

### 1. FSAR Section 10.4.9.2

DESCRIPTION OF CHANGE

Section 10.4.9.2 of the FSAR is clarified to indicate that events in addition to a Design Basis Tornado may also require Emergency Feedwater (EFW) inventory from the Wet Cooling Tower (WCT) basin.

REASON FOR CHANGE

To be consistent with other FSAR sections and design basis calculations. In addition, the EOPs credit the availability of the WCT basins.

50.59 EVALUATION

The EFW inventory credited for FSAR Section 6.3.3.4, Post-LOCA Long Term Cooling, FSAR Question 211.94, Branch Technical Position RSB 5-1, and FSAR Section 9.2.5.3.3, Design Basis Tornado events exceeds the amount stored in the Condensate Storage Pool (CSP). The SER accepted that the EFW inventory is provided by the CSP with backup supply available from the WCT basins. This clarification does not change any plant configuration, procedure, or analyses results. There is no reduction in margin of safety and no USQ is created.

## 2. FSAR Sections 9.3.3.2.2.5, 10.4.2.2 and 11.5.2.4.2.3

### DESCRIPTION OF CHANGE

FSAR Sections 9.3.3.2.2.5, Industrial Waste System (Turbine Building); 10.4.2.2, Main Condenser Evacuation System, System Description and 11.5.2.4.2.3, Industrial Wastes Sump Turbine Building Radiation Monitors are reworded to state that when the Industrial Waste Sump radiation monitor alarms, industrial waste sump discharge flow is stopped, not automatically sent to the LWM waste tanks. Flow is stopped because a manual valve in the line to the waste tanks is normally closed. Stopping flow allows the plant to determine corrective actions in response to the alarm.

### REASON FOR CHANGE

This change is a corrective action for CR-98-1340. The CR states that the FSAR has the Industrial Waste Sump discharge flow automatically diverting to the Waste Tanks, while the operations procedure stops flow on a radiation monitor alarm. Stopping flow prevents the introduction of oil into the LWM processing system

### 50.59 EVALUATION

There are no accidents listed in the FSAR that are affected or initiated by either the Industrial Waste Sumps (IWS) or the IWS radiation monitor, therefore, there are no changes in the consequences or probability of occurrence of any accident as a result of this change. There is no equipment important to safety involved in this change. The only equipment affected is the industrial waste sumps, the sump pumps, the IWS radiation monitor, the oil separator and the LWM waste tanks. Since no equipment important to safety is affected by this change, there are no changes to the consequences of any malfunction of equipment important to safety as a result of this change. There are no new system interactions or connections as a result of this change. This change makes the FSAR consistent with operations procedures. These procedures allow time to take corrective actions to prevent the discharge of radioactive water and to prevent the introduction of oil into the LWM processing system. Oil in the LWM processing resin will prevent the resin from removing activity from the water. Any contaminated water will still be processed in accordance with existing procedures. This action is similar to the action taken in response to a dry cooling tower radiation monitor alarm. Since this change does not involve any equipment important to safety, there is no possibility of a different type of malfunction of any equipment important to safety. The only possible equipment malfunction would involve the industrial waste sump pumps, which are non-safety, non-seismic components. If operations did not promptly secure these pumps, running at shut-off head can damage the pumps. No protective boundaries are altered and the margins of safety are unaffected by this change to the FSAR.

3. FSAR Chapter 13

DESCRIPTION OF CHANGE

The FSAR is being revised to show the new reporting relationship of the Plant Engineering Group. The Plant Engineering Group, previously identified as Systems Engineering, no longer reports to the General Manager Plant Operations but instead, now reports to the Director, Design Engineering.

REASON FOR CHANGE

Site reorganization

50.59 EVALUATION

No USQ is created by this change. This is an administrative change which moves the Plant Engineering Group from the responsibility of the General Manager Plant Operations to the Director, Design Engineering. This change does not affect any accident or equipment malfunction previously evaluated in the FSAR, does not create a new accident or equipment malfunction, and does not reduce the margin of safety as defined in the basis of any TS or safety analysis.

4. FSAR Chapter 1

DESCRIPTION OF CHANGE

The Independent Safety Engineering Group (ISEG) organization and administration are being changed to eliminate potential duplication of work and facilitate burden reduction. All essential ISEG functions previously performed by individuals in an independent group will be integrated into and distributed among existing groups within Nuclear Safety Assurance (NSA), Engineering, or Corporate Assessments. NSA, Engineering, and Corporate Assessment personnel do not report organizationally to the General Manager - Plant Operations and thus satisfy the independence criteria of NUREG-0737. To facilitate this change, FSAR Section 1.9.7 will be revised to more adequately reflect the manner in which the ISEG functions will be accomplished at Waterford 3. The major focus of these changes is to delete the requirement for 5 full-time ISEG engineers and to replace it with requirements for maintaining the ISEG function. The major goal of the change is to continue to carry out the function of ISEG but add the flexibility for the W3 organization to be able to fulfill that function organizationally in a manner that is more effective and efficient. Chapter 1, Appendix 1A, also requires revision to add the acronym ITR (Independent Technical Review).

REASON FOR CHANGE

ISEG is no longer a specific designated function. The commitment to NUREG-0737 for plant oversight and reduction of human errors is now included in the functions of Quality Assurance, Corrective Action and Assessment, Licensing, Engineering and Corporate Assessments. Human performance improvement is a management expectation for all Waterford 3 departments.

50.59 EVALUATION

This is an administrative change to FSAR Section 1.9.7 to delete the reference to ISEG and to Appendix 1A to add the acronym ITR. No physical changes are required to the plant or to any SSC that could affect either the probability or consequences of an accident or equipment malfunction. This administrative change has no affect on any margin of safety as defined in the basis for any TS or any fission product barrier. Therefore, there is no USQ associated with this change.

## 5. FSAR Chapter 13.1

### DESCRIPTION OF CHANGE

This change reflects reorganization of Waterford 3 onsite staff and offsite support organizations. This change includes management title changes, relocation of some reporting requirements and functional responsibilities. In addition, editorial changes were made such as renumbering lists and sections, and removing blank pages and previously deleted sections of the USAR.

### REASON FOR CHANGE

The Waterford 3 site organization has changed due to Nuclear Renewal and standardization efforts.

### 50.59 EVALUATION

The proposed changes reflect changes made to the Waterford 3 site organization due to Nuclear Renewal. These changes do not affect any accidents or important-to-safety equipment described in the FSAR and they do not increase the probability or consequences of either. There are no physical changes being made that would create either a new accident or equipment malfunction than one previously evaluated. While some organizations/positions have been deleted, other site or corporate organizations have absorbed their functions. The functions necessary for TS and regulatory requirements are maintained, even if the position responsible for the function has changed. No USQ is created by these changes.

6. FSAR Section 6.4.4.2.f

DESCRIPTION OF CHANGE

The proposed change revises FSAR Section 6.4.4.2.f for emergency air supply system capacity from 50,000 scf at 2000 psig to 45,000 scf at 1800 psig.

REASON FOR CHANGE

The Control Room Emergency Breathing Air system is required to maintain a minimum of 1800 psig of air storage, not 2000 psig. Although the system is designed for 2000 psig of air storage, it is required to operate with a minimum pressure of 1800 psig which is more than adequate to meet licensing requirements.

50.59 EVALUATION

According to the safety evaluation, the change is a clarification of the technical information in the FSAR concerning the capacity of the Control Room Emergency Breathing Air system. There is no change to the design, operation, or configuration of the system. The requirement for the system to operate in the event of a toxic chemical event to ensure the control room remains habitable for at least 6 hours is not affected. No accidents or important-to-safety equipment are affected by this change, no new system interactions are required, and no margin of safety is reduced.

7. FSAR Sections 5.4.12.2, 6.3.3.8, 12.1.2.i and Figures 9.3-2, 6.3-1

DESCRIPTION OF CHANGE

Revise FSAR Sections 5.4.12.2, 6.3.3.8, and 12.1.2.i to remove statements that suggest that all valves within the scope of the paragraphs have leakoff lines and/or double packing. Add a statement that valve packing glands have provisions to adjust packing compression to eliminate leakage. Also revise FSAR Figures 9.3-2, 6.3-1, and 9.3-6 to properly depict existing leakoff line configuration for valves that are currently shown incorrectly.

REASON FOR CHANGE

Design change SMP-1628 cut and capped valve leakoff lines for a number of valves and repacked the valves using only one set of packing consisting of 5 to 7 rings. Other design changes deactivated some valve leakoff lines. For some valves, leakoff lines are still active. The FSAR Sections and Figures being changed imply that all valves within the scope of those paragraphs and figures have leakoff lines and/or double packing.

50.59 EVALUATION

Some valves within the scope of this change could initiate an accident or are credited for accident mitigation for an accident previously evaluated in the FSAR. For instance, a valve failing closed and interrupting letdown flow is classified as a CVCS malfunction. A number of valves are in the Safety Injection System, which is designed to provide core cooling in the unlikely event of a LOCA. However, the proposed changes reduce the potential for valve failure. The new packing reduces friction so less force is required by the valve actuator to stroke the valve. Thus there is no increase in the probability or consequences of a previously evaluated accident. The reduction in stem friction also reduces the likelihood of occurrence of an equipment malfunction. The proposed change does not introduce any new failure modes and does not require any new system interconnections. The TS discuss allowable limits on RCS "identified" and "unidentified" leakage. No valves in this scope have leakage directed to either the Quench Tank or Reactor Drain Tank; therefore, identified leakage is not affected. Any leakage after the change would still be considered unidentified. However, the new packing is designed not to leak and any leakage that does occur is expected to be insignificant. No margin of safety is reduced by this change and no unreviewed safety question is created.

8. FSAR Sections 2.2A.1.3 and 6.4.4.2

DESCRIPTION OF CHANGE

The proposed changes adds a description of administrative controls in place that will ensure the Control Room envelope carbon dioxide concentration does not exceed 0.1%. Procedures will be revised to monitor carbon dioxide levels when the Control Room is in isolate mode during normal operations.

REASON FOR CHANGE

Current procedures require limiting access to the Control Room envelope to 16 people any time the Control Room is in Isolate mode. This limitation may require use of the backup Technical Support Center (TSC) as opposed to the normal TSC if the Emergency Plan is activated. These limitations are burdensome to the Control Room staff and the plant.

50.59 EVALUATION

The proposed change does not represent a USQ. The affected FSAR sections are revised to describe administrative controls to limit Control Room envelope carbon dioxide levels when the Control Room is in isolate during normal operations. No accidents described in the FSAR have either their probability or consequences affected by this change. No greater reliance is placed on any important-to-safety equipment nor will it be required to perform in any different manner. No protective boundary or margin of safety is affected by the change.

9. FSAR Figure 9.3-1, Sheets 1 through 6

## DESCRIPTION OF CHANGE

This change is made to revise Instrument Air flow diagrams G-152, Sheets 1 through 6, to reflect as-built conditions. The as-built changes entail valve position and tag number revisions which are for reference only and are not part of the plant design basis. The drawings are also revised to reflect removal of temporary Instrument Air connections, as directed by drawing notes.

## REASON FOR CHANGE

This is an administrative change only as the drawings are being changed to reflect the current plant as-built condition. No physical changes are being made to the plant.

## 50.59 EVALUATION

According to the safety evaluation, the change is purely administrative. No physical change to the plant is proposed. Therefore, the change will have no impact on any accident or important to safety equipment identified in the FSAR. The change does not create any new accident and does not reduce the margin of safety.

## 10. FSAR Tables 6.2-6 and 9.3-17

### DESCRIPTION OF CHANGE

The proposed change revises the affected tables to show results of new computer modeling of the Shutdown Cooling Heat Exchanger (SDCHx).

### REASON FOR CHANGE

The SDCHx thermal performance was analyzed using computer modeling with more detailed construction information from the manufacturer. The results of the analysis require revision of the FSAR heat exchanger data.

### 50.59 EVALUATION

The results of the evaluation show that no USQ exists. The SDCHx are required to mitigate consequences of a LOCA or MSLB but are not considered the initiators of any accident. The proposed change will not affect the function of the heat exchangers. A worse case heat transfer coefficient was calculated in MN(Q)-9-1. The existing heat removal analysis assumes a heat transfer coefficient which is less than that in the calculation. Therefore, the existing containment heat removal analysis is conservative. A maximum heat transfer coefficient was calculated in MN(Q)-9-1. The existing UHS analysis assumes a heat transfer coefficient which is greater than calculated; therefore, the existing UHS analysis is conservative. The evaluation demonstrated that the following important to safety equipment is not affected by this change: SDC system and heat exchangers, containment system, CCW and components cooled by CCW, UHS and equipment in the UHS. No physical changes are required to any SSC as a result of this change and no new system connections are required. The TS requirement to ensure that sufficient cooling capacity is available to remove decay heat and maintain water in the reactor pressure vessel below 140 deg. F has been maintained; therefore, no margin of safety is reduced.

11. FSAR Tables 12.5-1, 12.5-2, 12.5-3 and 12.5-4

DESCRIPTION OF CHANGE

The proposed change updates the instrument and radiation protection equipment tables in Chapter 12 of the FSAR:
Table 12.5-1 – Added clarification note to provide additional information about detector sensitivity requirements, relocated portable instrumentation to Table 12.5-2
Tables 12.5-2, 3, 4 – Specified instrumentation and equipment designated for E-Plan, updated tables to reflect current instrumentation used at Waterford3 and added statements for clarity.

REASON FOR CHANGE

These changes are being made as a result of corrective action for a condition report. These changes are to reflect the current instrumentation used at the site (quantity and range) and to clarify information.

50.59 EVALUATION

The updates and editorial changes do not affect the operation of any plant SSC. There are no accidents or important-to-safety equipment affected by these changes and no margin of safety reduced. This change does not represent an unreviewed safety question.

12. FSAR Tables 1.9-2, 2.2-4 and 9.4-3 and Sections 3.1.15, 6.4.2.4, 6.4.4.2.a and 6.4.5

DESCRIPTION OF CHANGE

1. Table 1.9-2 and section 6.4.4.2.a: Chlorine is no longer stored onsite. The reference to such storage is removed.
2. Table 2.2-4: Correct a typographical error – the Waterford3 control room design is a Type II (not Type IV) design as described in Reg. Guide 1.95. FSAR sections 2.2.3.3.3 and 6.4.4.2 document the W3 control room as satisfying the requirements of a Type II control room.
3. Section 3.1.15: There is no containment purge isolation signal in the control room isolation logic. An extensive review of plant documentation produced no evidence that this isolation signal was ever installed. There is no regulatory requirement to have this specific isolation signal. The response to Criterion 19, which included reference to this isolation signal, was written in error. The incorrect reference to this isolation signal is removed.
4. Section 6.4.2.4: The computer Halon system was deleted by DC-3374. The description of the system in this section is removed.
5. Section 6.4.5: The commitment statement is revised to directly state that the control room HVAC "safety" functions are tested to the requirements of the plant Technical Specifications.
6. Table 9.4-3:
   - Sheet 1 – The first component identification entry is changed to specify the "Normal" outside air intake. This detail is needed to differentiate the entry from the two redundant emergency air intakes.
   - Sheet 2 – The first Method of Detection entry is revised to correctly identify the instrumentation as a DP indicating switch.
   - Sheet 2 – The "Fails to Close" failure mode and effects of inlet dampers D-41 is added. This adds additional information about residual flow through a shutdown emergency filtration unit that was not previously included.
   - Sheet 2 – The emergency filter unit Remarks section for the "Filter clogs" failure mode is revised to match the system design and operating procedure.
   - Sheet 2 – The last Remarks entry is revised to state that a redundant flow measuring device is available. This better describes the redundant equipment pertinent to the failure of an emergency outside air flow measurement device.
   - Sheet 1 – The Effect on System, Method of Detection, and Remarks columns for outside air intake dampers D-40 (SA & SB) are being revised from "Fan will not start" to "No outside air flow"; "Class 1E fan status indicating light" to "Damper position indicator"; and "100 percent capacity... will start automatically" to "Control Room operator starts redundant air handling unit".
   - Sheet 1 – The "Fails to Start" failure mode and effects of the zone reheat coils are being added. This adds additional information about the control room supply air temperature that was not previously included.

- Sheet 3 (new) – the "Fails Open" failure mode and effects of the D-18 (SA & SB) and D-19 (SA & SB) toilet/kitchen/TSC isolation dampers to the air handlers is being added. This adds information about the failure modes of these dampers and the affects on the control room HVAC air balance.

REASON FOR CHANGE

The changes bring the FSAR into conformance with actual plant conditions.

50.59 EVALUATION

The changes revise the FSAR text to agree with existing plant conditions to provide correction, clarification and addition of information. The changes do not result in the physical modification of any plant equipment. Control room and computer room system operating parameters and monitoring equipment are unaffected. The changes do not rely on any instrumentation, do not result in any new system interfaces, do not involve addition of any electrical/electronic components, and will not degrade any existing systems or components that are important to safety. The changes do not impact overall control room system performance and will not impact the performance of control room personnel in a way that could lead to an accident or increase the probability of occurrence of an accident previously evaluated. No physical changes are being made to plant equipment or configuration that would impact either the probability or consequences of an accident or malfunction of equipment important-to-safety. In addition, these changes do not affect any protective boundaries or margin of safety. These changes do not represent an unreviewed safety question.

## 13. FSAR Section 11.2.2.2.1, Tables 11.2-1 and 11.2-2

### DESCRIPTION OF CHANGE

The FSAR is being revised to remove the specific resin type details for the Waste Condensate Ion Exchanger, Boric Acid Preconcentrator Ion Exchanger, and Boric Acid Condensate Ion Exchanger. This change also removes the filter size and type details for the Liquid Waste Management (LWM) Waste Filter, LWM Laundry Filter, and Boron Management (BM) Boric Acid Preconcentrator Filter to allow flexibility when processing liquid waste.

### REASON FOR CHANGE

These changes are being made to allow flexibility when processing liquid waste and to enhance operation of the system by using system components not normally used at this time.

### 50.59 EVALUATION

The only accident in which the affected systems must operate is a complete failure of all non-safety and non-seismic Category 1 equipment in the LWM and BM systems, which occurs as a result of a Safe Shutdown Earthquake. This failure is assumed to result in the simultaneous release of all liquids in the system tanks to the Reactor Auxiliary Building. No physical change is being made to the plant that will affect either the probability or consequences of this accident. Equipment affected by this change is not important-to-safety and is not used for accident mitigation or safe shutdown. No new system interconnections or interactions are required, no protective boundary is affected, and no margin of safety is reduced. This change does not represent an unreviewed safety question.

14. FSAR Section 9.5.1

DESCRIPTION OF CHANGE

Combustible Loading calculation EC-F91-025 indicates a combustible load and associated fire severity of 37 minutes which is greater than the 30 minute severity indicated in FSAR Section 9.5.1 for Fire Area RAB-23. The value used in the FSAR was the basis for an exemption to the requirements of 10CFR50, Appendix R. The exemption identified a 1.5-hour fire damper installed in a fire barrier rather than one meeting the 3-hour requirement for the barrier. The fire damper installed separates fire area RAB-23 and RAB-31. As part of the basis for the exemption, the request indicated there existed a fire severity of less than 30 minutes for fire areas on both sides of this damper. The exemption request basis for this same damper provided for fire area RAB-31 indicated a basis of "fire severity of adjacent areas is less than the rating of the damper." This FSAR change is to update this information and to incorporate a consistent fire severity limit for the fire area of "less than the rating of the damper".

REASON FOR CHANGE

Corrective action for CR-WF3-2000-0065.

50.59 EVALUATION

GL 86-10 states, "The licensee may make changes to the approved fire program without prior approval of the Commission only if those changes would not adversely affect the ability to achieve and maintain safe shutdown in the event of a fire." This change revises the basis for exemption from Appendix R to 10CFR50 for fire area RAB-23. Combustible Loading Calculation EC-F91-025 identifies a 37-minute fire severity for area RAB-23. This FSAR revision would alter the bases of the exemption request from "less than 30 minutes" to "less than the rating of the fire damper". This basis is consistent with exemptions granted for other plant areas with 1.5-hour dampers. This change introduces no new ignition sources, nor modifies any plant equipment. Thus the ignition frequency and probability of event initiation remain unchanged. This change would maintain a fire barrier fully capable of containing a fire which consumed all combustibles within the area. Because of this and based on the 1.5 hour damper maintaining a fire resistance greater than the available combustibles, it is concluded there is no impact on either the probability of occurrence or consequences of an accident or malfunction of equipment important to safety, nor is there a decrease in any margin of safety. It is concluded that this change does not represent an Unreviewed Safety Question.

## D. MISCELLANEOUS EVALUATIONS

### 1. Cycle 10 Core Operating Limits Report (COLR), Revision 2

DESCRIPTION OF CHANGE

The proposed change modifies the COLR 3.1.1.3 Restriction and adds COLR Figures 2A, 2B, and 2C to allow plant operation beyond a Critical Boron Concentration (CBC) of 200 ppm. The proposed change also makes editorial changes for clarity and consistency. COLR Section 3.2.7, Core Operating Limits Supervisory System (COLSS) Out of Service Power Level, is changed to be consistent with the TS.

REASON FOR CHANGE

COLR Revision 1 restricted operation to a CBC of 200 ppm. This change provides restrictions on the Limiting Conditions for Operation (LCOs) used to maintain the Moderator Temperature Coefficient (MTC) within its COLR requirements. The LCO restrictions are on cold leg temperatures versus power levels for ranges of Critical Boron Concentrations. The COLR MTC restrictions are for power levels greater than 70% and CBC less than or equal to 153 ppm. The COLR changes allow continued operation for the entire cycle and ensure the plant meets the Technical Specification and Safety Analysis MTC criteria for the entire cycle.

50.59 EVALUATION

The plant LCOs and consequently, any restrictions thereof (COLR changes) are not specific initiators of any accident or equipment malfunction previously evaluated in the FSAR. The COLR changes ensure the limiting inputs or assumptions used in the previously evaluated accidents remain bounded and consequently, the results remain bounding. For the safety analysis inputs that fall within the Technical Specification LCOs, the COLR changes ensure that the plant operation will be maintained within the Technical Specification LCOs. The COLR changes ensure that the safety analysis regulatory and design limits are met for the entire operating cycle. The COLR change also ensures that plant operation will be maintained within the Technical Specification LCOs. This COLR change has no affect on the design basis accident consequences with respect to design and regulatory acceptance criteria. There is no increase in probability or consequences of any DBA. No SSC is affected and there is no reduction in margin of safety.

2. Calculation EC-S97-016, Revision 1

DESCRIPTION OF CHANGE

The proposed change increases the TRM Table 3.3-5, FSAR Table 7.3-13 and FSAR Section 10.4.9.3.6 Emergency Feedwater (EFW) Pump response times from 54.0 seconds with Loss of Offsite Power and 42.0 seconds with Offsite Power available to 60.0 seconds and 50.0 seconds respectively, based on the results of calculation EC-S97-016, Revision 1.

This change will also clarify TRM Table 3.3-5 and UFSAR Table 7.3-13 such that both the EFW Block and Control valves response times are 25 seconds.

The following TRM Table 3.3-5 and FSAR Table 7.3-13 note is removed – "NOTE: Response time for all Motor-Driven and Steam-Driven Emergency Feedwater Pumps on all ESF Signal starts ≤ 54.0".

Notes "*" and "***" in TRM Table 3.3-5 and FSAR Table 7.3-13 are revised to clarify that the response times listed include EDG starting delays and sequence loading delays.

A note is added to FSAR Section 15.0.2 to describe the difference between the EFW pump response times and the FSAR Chapter 15 transient simulations.

FSAR Table 7.3-13 is changed to update the function and response time for the "Refueling Water Storage Pool – Low" to be consistent with TRM Table 3.3-5.

REASON FOR CHANGE

The EFW pump response time is intended to provide a larger margin between the equipment operation and surveillance requirements. The removal of the TRM Table 3.3-5 note is to enhance the clarity of the TRM. The note adds no value in terms of analysis assumptions or surveillance requirements. The EFW pump response times are governed by the TRM SG level low response times. The addition of the EFW Block Valve response clarifies the requirements for the valve. The surveillance requirements and analysis assumptions require a valve response time of less than or equal to 25 seconds. Specifically outlining the time reinforces this requirement. The TRM and FSAR Table notes '*' and '**' are changed as a result of corrective action for condition reports. TRM Table 3.3-5 had been previously updated concerning "Refueling Water Storage Pool – Low" but the corresponding FSAR Table was not changed. This change will make the FSAR Table consistent with the TRM.

50.59 EVALUATION

The increased EFW pump response time does not change any structures, systems or components of the EFW system. This increase also does not change the

intended function of the EFW system. In the short term, only the FSAR Chapter 15.1 and 15.2 events have the potential to initiate an EFAS actuation. The FSAR Chapter 15.1 and 15.2 events result in the most rapid depletion of steam generator inventory, and are thus limiting with respect to the EFW control system. Other non-limiting events where steam generator inventory is lost more slowly provide a greater capability for heat removal from the RCS making the time of EFW initiation less critical than the Chapter 15.1 and 15.2 events. The FSAR Chapter 15.1 events are for the increase in heat removal by the secondary system. For these accidents, the initiating event produces a cooldown of the RCS. The FSAR Chapter 15.2 events are for the decrease in heat removal by the secondary system. For these accidents, the initiating event produces a heatup of the RCS. The intent of providing EFW flow to the SG(s) is to provide inventory for primary heat removal. The heatup events (Chapter 15.2) require the largest amount of sensible heat removal post accident, thus the heatup events will require more inventory and thus bound FSAR Chapter 15.1 in terms of EFW requirements. Based on the results of EC-S97-016 Rev. 01, Chapter 15.2 event consequences with respect to design or regulatory criteria occur prior to EFW flow initiation or EFW flow reaches the SGs prior to SG dryout. This means that the increase in time before EFW flow is achieved does not affect the results of any of the analyses. For the long term cooling concerns, FSAR Chapter 15 and Non Chapter 15 events will be in a tripped condition and the plant will be in a period of stabilization and recovery. During this time, the increase in timing of EFW flow initiation will have a negligible impact because the EFW flow will occur prior to SG dryout. Thus, the proposed change has no effect on the DBA results with respect to design and regulatory criteria. The proposed change does not change any SSCs of the EFW system and does not change the intended function of the EFW system. Therefore, the increase in EFW pump response time does not reduce the margin of safety as defined in the licensing bases. The proposed change also does not increase the probability or consequences of any equipment malfunction or DBA, nor does it create the possibility of a different type of malfunction or accident.

3. TRM Change 99-009, Table 3.7-2

DESCRIPTION OF CHANGE

Remove Sprinkler System FPM-16 from TRM Table 3.7-2.

REASON FOR CHANGE

It was identified that sprinkler system FPM-16 did not protect safety-related Structures, Systems or Components. As such, this system is not required to be listed in TRM Table 3.7-2. TRM Table 3.7-2 is to include only those systems that protect safety-related equipment.

50.59 EVALUATION

The TRM bases for section 3.7.10 identifies applicability of the TRM to those systems that "...confine and extinguish fires occurring in any portion of the facility where safety-related equipment is located." This change corrects the erroneous inclusion of sprinkler system FPM-16 in TRM Table 3.7-2 and brings the facility into accurate compliance with the TRM. Sprinkler system FPM-16 serves the Fire Water Pump House, which contains no safety-related equipment. Section 9.5.1 of the FSAR (Fire Hazards Analysis) does not identify this plant area as containing or exposing safety-related SSCs. Removal of the sprinkler system from TRM Table 3.7-2 is administrative in nature and does not affect SSCs operability nor reduce function of the sprinkler system. The system does not provide protection for safety-related SSCs nor SSCs credited in the FSAR for DBA response or mitigation. No physical change to the plant is required and there is no impact on any safety-related equipment. Therefore, this change does not reduce the margin of safety as defined in the licensing bases, nor does it increase the probability or consequences of any equipment malfunction or DBA, or create the possibility of a different type of malfunction or accident.

4. Vendor Manual 460000048, Aerofin OPTIM Coil Selection Software

DESCRIPTION OF CHANGE

A vendor technical manual was developed to document the verification, validation, control, and use of the Aerofin OPTIM Coil Performance Software. The software is to be controlled as Class II software in accordance with the applicable site procedure. No plant change is required. Any changes to the facility due to future analyses using the OPTIM software will be supported by calculations and implemented using the appropriate configuration change process.

REASON FOR CHANGE

The OPTIM software supports design of the essential chilled water system and the HVAC systems supported by essential chilled water. The software more accurately predicts chilled water coil performance and will enable engineering to establish/verify the baseline design performance for the essential chilled water coils and evaluate coil performance for various conditions. The software will also provide a means of conducting future evaluations of the safety significance/past operability for nonconforming conditions, such as degraded chilled water or air flow.

50.59 EVALUATION

The OPTIM software will adequately predict the performance of essential chilled water coils in the essential chilled water system. As an analytical tool, the OPTIM software has the potential to affect the performance of the essential chilled water system and the HVAC systems supported by essential chilled water including control room air conditioning and cable vault and switchgear air conditioning. Essential chilled water serves the safety related room coolers in the following pump and heat exchanger rooms: CCW pump rooms, CCW heat exchanger rooms, charging pump rooms, emergency feedwater pump rooms, safeguards pump rooms and the shutdown cooling heat exchanger rooms. These systems provide environmental support in the mitigation of previously evaluated accidents including MSLB/LOCA. In order that OPTIM accurately predict essential chilled water coil performance, and thus not potentially adversely affect the performance of essential chilled water and the HVAC systems supported by essential chilled water, the software is controlled as Class II Software in accordance with W4.203. The software has been verified and validated by the manufacturer by comparing coil performance tested to ARI 410 with coil performance predictions. The software and test data are in close agreement. An additional verification and validation was conducted comparing manual coil performance calculations prepared per the methodology of ARI 410 with coil performance predicted by OPTIM. The manual computations and OPTIM computations are in close agreement. Therefore, there is reasonable assurance that the OPTIM software will correctly predict the performance of the essential chilled water coils and thus will not adversely impact the performance of the essential chilled water system or the HVAC systems supported by essential chilled water. No physical changes to the plant are required and use of this software will not result in a USQ. No physical change to the plant is being made and there is no impact on

any safety-related equipment. Therefore, use of OPTIM software for essential chilled water coil performance will not reduce the margin of safety as defined in the licensing bases, nor will it increase the probability or consequences of any equipment malfunction or accident, or create the possibility of a different type of malfunction or accident.

5. TRM Change 99-005, Incore Instrumentation

DESCRIPTION OF CHANGE

Revise the TRM to include the requirement for operability of at least one incore detector in each quadrant at each level.

REASON FOR CHANGE

The change is needed so that the FSAR, Topical Report EMEAD-02-NP, and TRM requirements will be in agreement for incore nuclear instrumentation operability. Inconsistencies were generated when the Incore Nuclear Instrumentation (INI) LCOs were removed from the Technical Specifications and incorporated into the TRM and FSAR Section 7.7.1.7. Specifically, the INI TS requirements were simply relocated from the TS to the TRM with no change in their content or wording. Also, an additional requirement was added to the FSAR which required operability of "...at least one incore detector in each quadrant at each level...". The FSAR additional requirement was added during implementation of NRC approved License amendment No. 107. Subsequently, a Central Design Engineering generated topical report (EMEAD-02-NP) was approved by the NRC. The approved topical report methodology used the more restrictive INI requirements. As such, it is desired to add the additional requirement that is stated in the FSAR to the TRM

50.59 EVALUATION

The incore detectors primary function is to provide inputs to the Core Operating Limits Supervisory System (COLSS). The incore detectors and COLSS are not safety related and COLSS is independent of the plant protection system. The proposed change is not a physical or procedural change to the plant. The TRM will continue to define the LCOs required to ensure that reactor core conditions during operations remain within the initial conditions assumed in the FSAR. The change does not alter the equipment credited in the mitigation of the design basis accidents or alter the way in which the plant operates. The proposed change does not create a USQ or a significant hazard. The health and safety of the public will not be affected by the proposed change. This action will not alter the impact of the station on the environment.

## E. ENGINEERING REQUESTS

### 1. ER-W3-99-0895-00-00, Inactivation of Secondary Vacuum Degassifier in Place

DESCRIPTION OF CHANGE

The proposed change will indicate that portions of the Secondary Makeup Vacuum Degasifier (SVD) skid are no longer in use and are "inactive".

REASON FOR CHANGE

The SVD is currently inoperable and there are no plans to restore it to service. Previous design changes, such as DC-3322, evaluated the change in philosophy to utilize contractor demineralizer equipment for site demineralized water. These previous changes rendered the majority of the site water treatment equipment obsolete and unused. However, these previous design changes did not revise all the necessary site documentation to indicate that the SVD skid was no longer in use.

50.59 EVALUATION

The SVD is not an initiator of any accidents previously evaluated nor is it used to mitigate the consequences of any accidents. The SVD is not important to safety, however it does interface with important to safety systems such as instrument air and 480 VAC. Opening breakers and closing valves to isolate the portions of the SVD that are no longer used does not create any new interfaces with these systems, nor does it adversely affect the existing connections to these systems. This change only places SVD components in positions that are currently allowed by operating procedures. There are no margins of safety associated with the SVD. The safety related water inventories are not contained in the tanks originally intended to be serviced by the SVD. The 50.59 Evaluation concludes that isolating portions of the SVD from other plant equipment will not result in a USQ or result in a reduction in the margin of safety of any TS. The SVD has no affect on any accidents or equipment important to safety.

## 2. ER-W3-99-0815-00-00, Fire Area RAB 30, Appendix R Compliance

### DESCRIPTION OF CHANGE

CR-WF3-1999-0790 identified that: (1) sprinkler system FPM21 (-4 RAB) is required to be listed in the TRM and (2) Appendix R separation requirements were not satisfied in the area. ER-W3-99-0815-00-00 provides a fire rated barrier to satisfy the separation requirements and updates the FSAR, FHA and TRM as required. The fire wrap material/design has not been used previously at W3.

### REASON FOR CHANGE

To provide compliance to 10CFR50 Appendix R.

### 50.59 EVALUATION

The licensing basis of the plant is compliance to the requirements of 10CFR50 Appendix R. Specifically, the separation of redundant safe shutdown equipment and cables (including associated cables). This change provides the level of protection as required in the licensing basis. Thus there is no increase in the probability or consequences of any accident introduced by these changes. In addition, the addition of sprinkler system FPM21 to the TRM is required since the area contains essential cables and the sprinkler system is in a safety-related fire area. The addition of fire wrap is not an initiator of an accident. The conduits do not support the weight of the fire wrap. The wall supports the wrap weight. The change corrects a condition adverse to the licensing basis and as such there is no negative impact on the probability of occurrence of malfunction. The addition of the wrap on the conduits has been evaluated and accepted in regards to ampacity derating, weight of the wrap on the conduits and seismic concerns. There is no increase in the consequences of a malfunction because the change brings the plant to the required condition. Fire accidents are addressed in the Fire Hazards Analysis that is contained in Section 9.5.1 of the FSAR. Compliance to Appendix R (and NRC approved exemptions) is the basis of fire accidents discussed in the SAR. This change provides compliance to Appendix R and as such no different type accidents are introduced. The addition of wrap typically impacts the ampacity of the cables being enclosed by the wrap due to the insulation provided by the wrap. Calculation EC-E99-003 verified that the required ampacity of the cables has not been impacted. This change does not introduce any new equipment malfunction scenarios. Fire wrap, fire sprinkler systems, etc., are not in the TS, but are listed in the TRM. The changes do not change any basis of the TS or TRM and therefore do not reduce any margin of safety. This change does not represent an unreviewed safety question.

3. ER-W3-99-0198-02-00, RCS Hot Leg Insulation Replacement

DESCRIPTION OF CHANGE

The Reactor Coolant System (RCS) provides the second pressure boundary which prevents the release of fission products from the reactor core to the environment, and it allows sufficient core cooling during normal plant evolutions and anticipated operational occurrences to prevent core damage. The "hot legs" are 42" (inside diameter) pipe assemblies between the reactor vessel and the steam generator inlet nozzles. Currently the hot legs are insulated with Transco brand reflective, encapsulated fiberglass insulation. This evaluation will allow portions of the hot leg insulation to be replaced with NUKON insulation blankets in order to facilitate the installation of the Mechanical Nozzle Seal Assemblies (MNSA). The hot leg sample/instrument nozzles and the MNSAs will not be insulated because the C-E stress report assumes a temperature gradient along the MNSA assembly (i.e. uninsulated). Lack of insulation on the MNSAs will leave an uninsulated area of approximately 24 square inches at the base of each nozzle.

REASON FOR CHANGE

Condition Report CR 99-0232 and 99-0234 documented inspections performed during RF-9 that found evidence of RCS leakage from three sample/instrument nozzle penetrations located on the RCS hot legs. ER-W3-99-0198-00-00 will authorize installation of three MNSAs from Combustion Engineering (C-E) in order to repair the three known leaking hot leg nozzles. The MNSA flange assemblies will be bolted to the outside wall of the hot leg piping which will require the hot leg insulation to be modified. The hot legs are currently insulated with a reflective, encapsulated fiberglass that is very difficult to modify.

50.59 EVALUATION

The NUKON blanket insulation is a quilted, light-density, semi-rigid fibrous glass insulation (a pad). NUKON is attached to the piping with Velcro quick release straps and covered with a stainless steel protective jacketing. NUKON has been evaluated and approved several times for replacing the existing reflective, encapsulated fiberglass insulation within containment. NUKON has been qualified and installed on the reactor head, pressurizer, reactor coolant pumps, and steam generators. In December 1978, the NRC "Final Staff Evaluation of Topical Report OCF-1, Nuclear Containment Insulation System" found NUKON insulation acceptable: "Based on the results of the quantitative and qualitative tests performed by or for Owens-Corning Fiberglass, the staff concludes that the Owens-Corning Fiberglass Corporation's nuclear containment insulation system (NUKON) is capable of retarding heat loss from piping and equipment in containment areas, and that the overall integrity of the blankets will not be adversely affected by the conditions found during the lifetime of the plant. We also conclude that during a loss of coolant accident, the Owens-Corning Fiberglass insulation system is not expected to interfere with the operation of the emergency recirculation cooling system." Performance Contracting Inc. (PCI) Test Report ESD-TR-IOF dated May 1991 reflects that the transport velocity (i.e.

water velocity to carry insulation in the water stream) of shredded fiberglass is between 0.17 and 0.20 feet per second. Entergy calculation MN(Q)-6-35 revision 1, reflects that the velocity of the water flowing through the Safety Injection Sump screen, at the maximum design flow rate, is 0.136 feet per second. This velocity is less than the minimum transport velocity of the fiberglass as determined by testing performed by PCI. In addition, because the insulation will be located on the hot legs which are approximately 50 feet from the Safety Injection Sump screens, the water velocity under the hot legs would be essentially zero. Therefore, the insulation would not be transferred to the Safety Injection Sump screens if it were to be damaged and fall to the floor of the containment building. This evaluation reflects that changes proposed by this ER will not reduce the margin of safety as defined in the basis of any Technical Specifications or safety analysis, and there are no unreviewed safety questions.

## 4. ER-W3-99-0682-01-00, Y2K Satellite Phone Installation

### DESCRIPTION OF CHANGE

Install a Y2K compatible satellite telephone system consisting of a mobile terminal/power supply and Control Room handsets. This new equipment is stand-alone with the exception of 120-vac power from a non-safety lighting panel. The handsets will be located in the Operations Administrative area in the Control Room envelope. The terminal/power supply will be located in a box mounted in the RAB +58 stairwell (below the Broad Range Gas Monitors). The antenna will not be permanently mounted (it will be placed on the RAB roof near the elevator machine room when needed).

### REASON FOR CHANGE

Required by NRC GL 98-01 to ensure continuous offsite communication capability.

### 50.59 EVALUATION

The communication system added by this change is not an accident initiator. It will typically be used post accident to communicate with offsite agencies in the event that the normal system fails. The new equipment does not interface with or support safety-related plant equipment. The installation of this communication system does not affect the operation of the Waterford 3 facility and does not represent a USQ.

5. ER-W3-99-0083-00-01, EFW Quantity for Chapter 15 Events

DESCRIPTION OF CHANGE

ER-W3-99-0083-00-00 provides the evaluation for determining the Emergency Feedwater (EFW) inventory required for the following conditions: design basis for the Condensate Storage Pool (CSP) TS value of 170,000 gallons, EFW required for the Long Term Cooling (LTC) analysis, EFW required for FSAR Chapter 15 events, and the EFW required to meet decay heat for 24 hours for the design basis tornado event.

REASON FOR CHANGE

These changes are prepared based on a design bases review open item to address the EFW and Ultimate Heat Sink (UHS) water consumption.

50.59 EVALUATION

The Auxiliary Component Cooling Water (ACCW) system is not an initiator of any design basis event. The EFW system can potentially be an initiator due to inadvertent activation causing an increase in feedwater flow. However, this ER does not make any physical change or add a new activity to any plant system and there is no impact to the existing plant operation. Therefore the proposed changes do not increase the possibility of occurrence of an accident previously evaluated in the FSAR. The design basis accident consequences are not increased provided that the design and regulatory limits are met for each accident (i.e. RCS peak pressure, fuel performance, dose consequences, etc.). The proposed change demonstrates that sufficient water exists to cope with any design basis accident. Thus, the accident consequences with respect to the design and regulatory limits remain unchanged. The accident analysis water consumption remains within the Waterford 3 licensing bases. The proposed change demonstrates that a conservative inventory margin is present for both EFW and UHS; thus, the consequences of an accident have not increased. The proposed change does not alter the operation or function of the ACCW or EFW systems or alter any component or structure of these systems. The change evaluates the ACCW and EFW inventory required to meet the plant's design basis events and no new activity or physical change is proposed. Thus, these changes do not increase the probability of a malfunction of equipment previously evaluated in the FSAR. The failure modes associated with the ACCW and EFW systems remain unchanged. Any postulated malfunction will result in the same consequences as are currently evaluated. Therefore the consequences of a malfunction of equipment important to safety previously evaluated in the FSAR are not affected. There are no physical changes as a result of this change and therefore it does not create the possibility of an accident or malfunction of different type than previously evaluated in the FSAR. The proposed change demonstrates that a conservative inventory margin is present for both EFW and UHS, thus there is no reduction in margin of safety for any TS or safety analysis.

6. ER-W3-99-0894-00-00, Inactivation of Auxiliary Boiler in Place

DESCRIPTION OF CHANGE

The proposed change will indicate that portions of the Auxiliary Boiler (AB) are no longer in use and are made "inactive".

REASON FOR CHANGE

Failure of a level control valve resulted in the failure of a number of boiler tubes. The age of the AB and the cost of the repairs resulted in the decision to no longer use the boiler.

50.59 EVALUATION

The auxiliary boiler (AB) and accessories are used primarily for plant start-up or when main steam is not available. The AB is an operational convenience only, to facilitate plant start-up. The auxiliary boiler is not the initiator of any accidents nor is it credited for the mitigation of any accidents. The Emergency Diesel Generators (EDGs) are credited in the mitigation of accidents in the FSAR. The EDG onsite fuel oil supply shall be sufficient to operate the EDGs for at least 7 days plus 10% (as required by ANSI N195). The 10% fuel oil margin is not available in the EDG fuel oil storage tanks. This 10% margin is currently maintained in the Auxiliary Boiler fuel oil storage tanks (FOST) as per TRM 3/4.8.1. The auxiliary boiler FOST will not be adversely affected by isolating the portions of the auxiliary boiler fuel oil system which are no longer used. The compensatory measures in place to transfer fuel oil to the EDG fuel oil storage tanks use temporary pumps, hoses and tanker trucks. The compensatory measures do not require interface with any part of the auxiliary boiler fuel oil piping, pumps or components. Isolation of fuel oil lines to/from the AB FOST will be accomplished by closing existing isolation valves. This configuration is no different than the current configuration when the auxiliary boiler is not in use. There are no new system interconnections created by the proposed change.
The 50.59 Evaluation concludes that isolating the AB from the plant will not result in a USQ or in a reduction in the margin of safety of any TS. The AB is an operational convenience only, to facilitate plant start up. It has no affect on any accidents or equipment important to safety.

## 7. ER-W3-99-0919-00-00, Relocate RCP Vibration Monitoring Location

### DESCRIPTION OF CHANGE

The purpose of this proposed temporary alteration is to move the monitoring location of the spared keyphasor B circuit from the Reactor Coolant Pump (RCP) motor shaft to near the pump discharge. The sensor on the RCP case will be an Endevco Model 2273AMI/AM20 accelerometer. The existing spared Bently-Nevada preamplifier will remain in place but will be replaced by a Endevco Model 2771 charge converter in the circuitry. The 2771 charge converter will be mounted in the same box with the existing Bently-Nevada preamplifier. This change will occur on RCP 2B and on RCP 1B, which will be used as a basis for comparison.

### REASON FOR CHANGE

RCP 2B has experienced recent failures of the pump seal baffle. Monitoring the pump vibration near the pump discharge on the pump case will provide input into determining a root cause for the baffle failures. This will also be performed on RCP 1B, which is unaffected by seal baffle failures, to provide a basis for comparison.

### 50.59 EVALUATION

The proposed change is not a test or experiment and does not change a procedure as described in the SAR. This is a change to the physical plant, but no change to any system function is involved. The use of the spared and deactivated keyphasor B circuit on RCP 2B and 1B poses no risk or adverse impact to plant safety. The use of the spared circuits for this purpose does not result in the loss or sacrifice of plant functions that were presently in use. The circuit after modification will perform a monitoring function only with no electrical or physical interconnections with other plant control circuits. The monitoring function of the revised circuits will not have the capability to adversely impact other plant components or functions and therefore will not result in an adverse impact to plant safety. Failure of the activated key-phasor is not a creditable initiator of any accident evaluated in the FSAR. The accelerometer will provide a monitoring function only and will not be electrically or physically interconnected with any plant control circuits, therefore the possibility of a circuit malfunction increasing the probability of an accident is non-existent. Consequently, the use of the spare inactivated key-phasor circuit will not remove any existing plant protective functions and will have no adverse impact on plant safety. The proposed change will not increase the consequences of an accident as evaluated in the FSAR. The keyphasor circuits are used for monitoring and trending purposes only and have no accident mitigating function. Only the spare key-phasor circuit will be activated; therefore, no accident mitigating or equipment required for safe shutdown will be removed from service to provide the additional vibration monitoring functions on the two RCPs. The seismic qualifications of RCP 1B & 2B were considered, but there is no impact because the additional mass (<1 lb.) is insignificant. Seismic II/I concerns were also considered and were determined to have no impact due to the insignificant mass of the accelerometer, mounting plate, and epoxy. Failure of the epoxy material was also considered, but the insulation blankets would retain the

instrument, and the operation and function of the RCPs would not be affected. Transport to the Safety Injection Sump (SIS) screens was also considered but the accelerometer and mounting plate would not float and therefore would not be transported to the SIS during a LOCA. Therefore, there is no increase in probability of an equipment malfunction. The proposed change to install an accelerometer on the case of RCPs 1B and 2B will not increase the consequences of a malfunction of equipment important to safety previously evaluated in the FSAR because the RCS system will continue to function as described in the FSAR. It will not create the possibility of an accident or equipment malfunction of a different type than any previously evaluated in the SAR because there will be no new system interactions and the integrity and function of the existing safety related systems and components will not be affected. The active or spared key-phasor circuits have no TS requirements and are not identified in the TS, and have no limits described in the FSAR. The activated spare key-phasor circuit on RCP 2B and 1B will perform vibration monitoring functions on their respective RCPs and will not be electrically or physically interconnected with any plant control circuits. Consequently the activated spare key-phasor circuit is not capable of adversely impacting the margin of safety in the technical specification basis.

## 8. ER-W3-99-0995-00-00, Evaluation Compaction Results for CR-97-1311

### DESCRIPTION OF CHANGE

CR-97-1311 documented that two compaction tests on Class A backfill were less than the required values given in specification LOU-1564-482 and in FSAR Appendix 2.5C.

### REASON FOR CHANGE

Evaluate the acceptability of the two compaction test results.

### 50.59 EVALUATION

The Class A backfill that had compaction tests less than the required minimum is located less than 5 to 6 feet from the surface in the security isolation zone just to the north of the Fuel Handling Building. There is no equipment important to safety that relies on any kind of support from this backfill material. The only equipment that could possibly be impacted by the compaction of this backfill would be some of the security equipment in the isolation zone. However, by observing the area in question, there has been no settlement in the area since the fill material was placed in 1997. The backfill is only used to fill an area so as to provide a relatively flat surface for access, and for security reasons. The compaction of this backfill will not have any impact on how any equipment is operated, or on how any systems are configured. The fill material is located less than 5 to 6 feet from the surface. The most critical locations for the backfill compaction are at the lower elevations where the soil has the highest strains due to the overlying soils. Improperly placed backfill at the lower elevations could allow the backfill to shear, thus placing a greater load on the nuclear island walls. However, the backfill in question here is very near the surface and does not have any significant loads of any kind placed on it. Also, with the backfill placed in the security isolation zone, there would have to be a major project if significant loads were placed on it in the future. Any major project would require some foundation preparation, which would cause the backfill to be replaced at that time. Liquefaction of this material is not a concern in the event of an earthquake since there are no structures located on top of this portion of the backfill. This backfill has no affect on the probability of an accident or on any important to safety equipment. While there is potential to affect the consequences of an earthquake, the area covered is only in the security isolation zone north of the FHB and is located less than 6 feet from the surface. The 50.59 Evaluation concludes that the lower compaction values for the limited area affected are acceptable and do not result in a USQ.

9. ER-W3-98-0743-00-00, Evaluate Four DCT Motors

DESCRIPTION OF CHANGE

This change involves the evaluation of replacement Dry Cooling Tower (DCT) fan motors. The new motors purchased from the original equipment manufacturer (OEM) using the same specification as the original are physically and electrically different. The differences are minor, such as changes in footing, additional space heater conduit box, locked rotor current, full load amps, and speed. The motors are safety class 1E and Seismic 1.

REASON FOR CHANGE

The original motors are obsolete.

50.59 EVALUATION

The replacement fan motors were purchased from the OEM as safety related motors. The OEM had made a few changes to the motor such as footing, additional conduit box for the space heater, minor changes in the full load current and locked rotor current. These changes were evaluated and are within the design basis of the Emergency Diesel Generators (EDGs) and the buses supplying the load. The overall system performance of the Component Cooling Water (CCW) and the EDG are not affected. The motors will be operating within their design limits. The motor functions to drive the dry cooling tower fans. The replacement motors will not adversely affect system performance (both original and replacement motor have the same horsepower and voltage ratings). The mechanical changes of the dimensions and the weight have also been evaluated and found to be acceptable. This change does not affect the accident analysis or create the possibility of an accident or malfunction of a different type than previously evaluated in the FSAR. This change also does not affect the facility and procedures as described in the FSAR. FSAR tables reflecting the motors and plant configuration documentation have been updated as required. Therefore, this change does not create a USQ or a change to any margin of safety.

10. ER-W3-99-0849-00-00, 4.16 KV Undervoltage Relays

DESCRIPTION OF CHANGE

Based on the conclusion of calculation EC-E91-053, the reset values of 4.16 KV undervoltage relays 27-1, 27-2, 27-3/A(B) are revised to reflect more realistic values (as-found field data). Consequently, the surveillance procedure and FSAR statements regarding the reset values of these undervoltage relays are revised.

REASON FOR CHANGE

The reset value of the 4.16 KV undervoltage relays 27-1, 27-2, 27-3/A(B) as stated in the FSAR is greater than the TS minimum voltage requirement for the EDGs. ER-W3-99-0849-00-00 evaluated the actual reset values of these undervoltage relays and concludes that the FSAR statements regarding the reset values of 4.16 KV undervoltage relays are very conservative and do not represent the actual reset values.

50.59 EVALUATION

No USQ exists as a result of this change. This change does not impact the equipment function or surveillance steps/procedures. A change to the FSAR is required only to ensure consistency between the FSAR and the design basis calculation. The original conclusion does not change and the relays will continue to perform their safety functions. There is no increased probability of an accident or malfunction of equipment due to this change.

11. ER-W3-98-0667-00-00, Drawing G-286 Does Not Show MCC 3A315 and 3B315 as a Tripped Load

DESCRIPTION OF CHANGE

This change corrects drawing G-286 to indicate certain breakers are tripped and to remove breaker programmer settings from drawing G-286 and FSAR Figure 8.1-7.

REASON FOR CHANGE

Breaker programmer settings are being removed from G-286 because the information is incorporated in other design documents. The elimination of this information from G-286 eliminates redundant information, reduces the potential for error, and saves resources when design documents are updated.

50.59 EVALUATION

This change does not impact the operation or physical configuration of the plant, the actual function or settings of breakers, plant safety, or the regulatory basis of Waterford 3. Therefore, there is no USQ associated with this change.

12. ER-W3-97-0566-00-00, As Build RAB +7 on G-149, Revision 23

DESCRIPTION OF CHANGE

The titles of many of the administration rooms on the +7 level of the Reactor Auxiliary Building (RAB) as shown on several drawings no longer match the functions of the rooms. This area was once used as the HP offices, dressing room, locker room, counting room, key issue area, and records and storage areas. These rooms are now used as offices and work areas. The titles of the rooms on the +7 elevation that have not changed are the HVAC Equipment Room, the vestibule, the corridor, the I&C Room, the Communication Room, and the Multiplexor Room. The locations of the rooms' walls have remained unchanged. Only the function of the rooms has changed. Also, the general arrangement drawings show many lockers that are no longer located on the +7 floor. This ER will change the titles of the rooms and decontrol the location of the lockers on the +7 drawings. The room titles have been changed to "office/work area."

REASON FOR CHANGE

To show the as-built function of the rooms.

50.59 EVALUATION

There is no physical work involved with this change. This ER will issue DRNs to change room titles on several drawings, FSAR figures, and sections of the FSAR. There is no USQ associated with this change.

13. ER-W3-99-0964-00-00, Revise FSAR Tables 6.2-25 and 6.2-26

DESCRIPTION OF CHANGE

FSAR Tables 6.2-25 and 6.2-26 are revised to indicate that the Reactor Coolant Pumps (RCPs), pressurizer, Steam Generators (SGs), and various piping have NUKON insulation. This change also corrects the FSAR statement that NUKON insulation replaced the metal reflective insulation on the Reactor Pressure Vessel (RPV) head (NUKON replacement was approved but not installed).

REASON FOR CHANGE

RG 1.70 requires the FSAR to discuss the types of insulation used inside containment and to identify where and in what quantities each type is used.

50.59 EVALUATION

The proposed changes have been reviewed and it has been determined that this ER does not authorize any physical changes to the plant, no changes to any operating procedure or test procedure, and no changes to any input document or calculation. It does allow revision to the FSAR for changes already approved by previous design changes, SPEERs, and ERs. The proposed changes do not affect the operation or function of any SSC important to safety or accident mitigation. The evaluation has concluded there is no USQ.

14. ER-W3-99-0437-00-00, DRTS Status Computer Input

DESCRIPTION OF CHANGE

This ER adds the capability of monitoring the Diverse Reactor Trip System (DRTS) switch status to the Plant Monitoring Computer (PMC).

REASON FOR CHANGE

Currently, there are two indications of Diverse Emergency Feedwater Actuation System (DEFAS) switch status but none for DRTS.

50.59 EVALUATION

Anticipated Transient System (ATS) is a backup system that will be actuated in the event a common mode failure prevents the Reactor Protection System (RPS) from performing its design safety function. DRTS and DEFAS are secondary systems which do not contribute to any initiating events for the accidents analyzed in the UFSAR. The operation of DRTS and DEFAS will not be altered by this design change. This change only affects the PMC indication of selector switch position for these systems. Therefore the consequences of an accident previously evaluated in the UFSAR will not be increased. Section 15.8 of the UFSAR states that DEFAS and DRTS are "diverse, secondary alternates to existing systems based on common mode failure in the systems not being assumed by the ATWS Rule, and a single failure will not cause the Diverse Emergency Feedwater Actuation System to adversely impact FSAR Chapters 6 and 15 events". This change, which only affects the indication of switch position on the PMC, will not increase the consequences or probability of occurrence of a malfunction of any equipment in these systems. The indication of switch status is provided to the PMC for monitoring and/or troubleshooting, Maintenance Rule trending, and (optional) Sequence of Events determination. As previously stated the actual operation of DRTS and DEFAS will not be altered by this change. These systems function to mitigate the consequences of an accident - they are not accident initiators. Therefore, the possibility of an accident of a different type than any previously evaluated in the UFSAR will not be created by this change. This change does not alter the original evaluation in the UFSAR that was established when the DRTS and DEFAS systems were added by DCP-3080. Therefore, this change does not create the possibility of a malfunction of equipment important to safety of a different type than any previously evaluated in the UFSAR. TS 3.3.3.5 and 3.3.3.6 govern some of the instruments that provide inputs to the DRTS and DEFAS systems. However, these instruments are not affected by this change. Therefore, there is no impact on any margins of safety resulting from this change.

15. ER-W3-98-0919-00-00, Calculation EC-M94-002, LTOP Safety Relief Valve Design Basis Calculation

DESCRIPTION OF CHANGE

Calculation EC-M94-002 provides the design basis for the required set pressure and relieving capacity of the Shut Down Cooling (SDC) suction line Low Temperature Overpressure Protection (LTOP) Safety Relief Valves (SRVs). The LTOP SRVs provide protection from an overpressure transient which could result in exceeding the pressure-temperature operating limits for the Reactor Coolant System (RCS) at low temperatures. Engineering Request ER-W3-98-0919-00-00 revises the FSAR and other documents to be consistent with the calculation results. The Engineering Request also returns the FSAR LTOP SRV capacity "Assumption" to the original value used in the FSAR analysis. No physical changes to the plant are proposed.

REASON FOR CHANGE

The calculation revision incorporates information from the original design basis calculation contained in CE Calculation C-PEC-117 and also revises the methodology used to determine the required set pressure and relieving capacity of the LTOP SRVs. The calculation revision is part of the Design Basis Reconstitution program. The Engineering Request revises the FSAR and other documents to be consistent with the calculation results. Calculation EC-M94-002 does not change the assumed LTOP capacity value used in the SAR analysis. The assumed LTOP capacity was previously changed from the value used in the FSAR analysis to depict the results of Revision 0 of the calculation. No physical changes to the plant are proposed.

50.59 EVALUATION

This calculation establishes the required set pressure and relieving capacity of the LTOP SRVS. This revision determined that the bounding case for the required SRV relief capacity is the sum total flow from two High Pressure Safety Injection (HPSI) pumps, three charging pumps, and all pressurizer heaters energized. The actual HPSI pump performance curves will be used for HPSI flow instead of "preliminary" delivery curves provided in the original calculation. Additionally, the HPSI flow injection valves are assumed to be fully open for maximum flow to the RCS. This calculation conservatively assumes all heat generated from all of the pressurizer heaters enters the RCS, with no loss to the containment atmosphere through the pressurizer walls. This calculation also considers the flow from only two of the HPSI pumps plus all three charging pumps instead of three HPSI pumps and three charging pumps as assumed in C-PEC-117. While considering only two HPSI pumps may seem to be a non-conservative action, the consideration of only two HPSI pumps for LTOP relief capacity is consistent with FSAR Figure 5.2B-3, ABB/CE Calculation C-PEC-252, and Waterford 3 operating procedures. The starting of only two HPSI pumps on a Safety Injection Actuation Signal (SIAS) is also acknowledged by the Waterford 3 SER in the discussion of the LTOP relief capacity. This calculation revision determined a new maximum relieving capacity requirement

of 2553 gpm for the SRVs, which is well below the rated capacity of 3345 GPM for each SRV. However, the calculation determined that the LTOP SRV lift setpoints should remain at their present setpoint of 415 psig. The difference between the rated and required capacities of the SRVs does not represent a margin of safety as defined by TS. Based on the above, this calculation revision does not represent a USQ.

16. ER-W3-99-0492-00-00, Replacement of Gaseous Waste Management (GWM) Discharge Flow Meter

DESCRIPTION OF CHANGE

This minor modification will replace the existing instrument flow transmitter with one that is not obsolete and improve the loop accuracy (mass flow is not affected by changes in pressure, temperature, or density of the process fluid). This physical change requires FSAR Figures 11.2-2 and 11.3-1 to be revised. In addition, procedures CE-003-513 and CE-003-515 will be revised to address new administrative limits for Gas Decay Tank (GDT) releases.

REASON FOR CHANGE

As documented in CR 98-1498, the GWM Discharge Instrument Flow Transmitter cannot be reliably calibrated. The currently installed flow instrument is a rotameter that is dependent upon the density of the flowing gas. Analysis of GDT releases have shown that the flowing gas composition between releases is never constant and significant density changes result in inaccurate flow readings. In addition, since the instrument is obsolete, replacement is required for maintenance.

50.59 EVALUATION

The proposed change simply replaces an existing rotameter with a mass flow meter. The meter has no safety function and is not used to limit radiological releases during an accident. The meter is used in conjunction with administrative controls to limit off-site discharges to ensure compliance with 10CFR20. The current administrative limit of 50 scfm will be revised to 100 scfm to address instrument uncertainties and GDT density changes between releases. The proposed increased administrative limit to 100 scfm is negligible with respect to normal plant stack flow, and will not impact effluent concentrations specified in TRM 3/4.11.2 or 10CFR20. Section 15.7.3.1 of the FSAR discusses a Radiological Waste Gas System Leak or Failure. Section 15.7.3.1.6 of the FSAR concludes "In the unlikely event of an accidental release of the contents of a waste gas decay tank resulting in a release of the maximum stored gaseous activity from one Reactor Coolant System volume, the doses at the exclusion area boundary and the outer boundary of the LPZ are less than the 10CFR100 doses". This minor modification does not involve the GDTs or their isolation valves, which are Seismic Class 1, but involves an instrument flow transmitter downstream of the isolation valve and on the low-pressure side of the GWM system. This change will replace the existing instrument flow transmitter (a rotameter with attached transmitters) with a Coriolis Mass Flow and Density Sensor/Transmitter. The existing non-safety, non-seismic instrument measures the discharge rate of the GDTs when released. The instrument is used for indication purposes only. The downstream radiation monitor is the component that ensures we do not exceed 10CFR 20 limits by terminating the release. This instrument is not credited with limiting the radiological release consequences of any accident. It is concluded that replacing GWMIFIT0648 will not reduce the margin of safety as defined in the basis for any TS that no USQs are created.

17. ER-W3-99-0763-00-00, Installation, Operation, and Maintenance of DCT Temporary Diesel Driven Sump Pumps

DESCRIPTION OF CHANGE

This Temporary Alteration installs a Diesel driven, self-priming pump in each Dry Cooling Tower (DCT) area. These pumps supplement the installed DCT Area Sump Pumps with an additional capacity of 200 gpm to meet the revised flow requirements established in CR-99-0789 for the Probable Maximum Precipitation (PMP) and Standard Project Storm (SPS) events.

REASON FOR CHANGE

This Temporary Alteration is needed to provide adequate flow from the DCT sumps during design basis rainfall events to prevent flooding of safety related electrical equipment located in the DCT areas.

50.59 EVALUATION

This evaluation reflects that the equipment installed for this Temporary Alteration will not have any adverse effects on any equipment important to safety and that there are no Unreviewed Safety Questions. This Temporary Alteration will ensure equipment important to safety is protected from ponding rainwater in the event of any design basis rainfall event. There are no FSAR evaluated accidents that credit the DCT area sump pumps, so probability of occurrence of an accident previously evaluated cannot be affected. There are no accidents analyzed in the SAR that credit the DCT sump pumps for consequence mitigation, thus the consequences of an accident previously evaluated cannot be increased. The DCT area sump pumps are considered "equipment important to safety" because they protect safety related MCCs located in the cooling tower areas from potential damage due to ponding of rainwater resulting from the PMP event. The new diesel driven sump pumps are completely independent from the existing electrically powered sump pumps and they do not share any common equipment except for the sump and drainage system, which collect the rainwater. Care will be taken when positioning the suction hose to ensure that it does not interfere with the level switches or pump suction strainer on the existing sump pumps. Procedures direct that the hose be secured approximately three feet above the bottom of the sump and away from the level switches to prevent any possible interference. Both the new suction hose and the installed sump pump are each provided with suction screens to prevent foreign materials from entering the pump suction. A test will be performed when the Temporary Alteration is installed to ensure that the flow capacity is greater than the required 200 gpm. The discharge hoses will be rolled and stored when not in use and they will be restrained or secured as appropriate. The suction hoses will be connected to the pump suction and properly positioned and restrained in the sump. Prepositioning of the suction hoses will eliminate the possibility of inadvertent damage to the float switches for the electric pumps during hose insertion into the sump. Following insertion, proper operation of the installed sump pumps will be confirmed through routine operation during normal rainfall events. When needed,

the discharge hoses will be connected and routed over the cooling tower exterior floodwall with no interface with the installed sump pump system piping. Because the integrity of the existing sump pumps will be maintained there is no increase in the probability of a malfunction of equipment important to safety previously evaluated in the FSAR. The equipment important to safety previously evaluated in the SAR includes the MCCs, transformers, and existing motor driven sump pumps that protect the DCT areas. Placing new sump pumps in the DCT areas will not affect the consequences of malfunction of any of the equipment in the DCT areas. The proposed change to install two new temporary sump pumps in the DCT areas cannot create the possibility of an accident different than any previously evaluated because the safety related equipment in the DCT area will be protected from flooding due to ponding rainwater. All seismic concerns associated with the storage of the temporary pumps are evaluated in Designated Storage Area Permits 99-0051 and 99-0052. The installation and testing requirements specified will ensure that the new pumps are capable of performing when needed and that they do not interfere with the operation of the existing motor driven pumps or create the possibility of a malfunction of equipment important to safety of a different type than previously evaluated in the FSAR. The operation or capabilities of the DCT area sump pumps are not contained in the basis for any TS; therefore, no margin of safety is reduced.

18. ER-W3-99-0713-01-00, Removal of Condensate Gland Seal Condenser Bypass Restrictive Orifice Sensing Line

## DESCRIPTION OF CHANGE

This minor modification will remove the high pressure and low pressure instrument sensing lines for the mechanical restrictive orifice CD MFE 0624 and install permanent pipe plugs in their place. The mechanical restrictive orifice is installed in the Gland Steam Condenser bypass line. There is no instrumentation installed on these lines and the proposed change will not eliminate the ability to add instrumentation in the future.

## REASON FOR CHANGE

The current configuration is leaking condensate at the threaded connection of the raised face flange connection. A leak repair has been installed at this location. However, any future significant leakage may require a plant shutdown to repair. Therefore, this change is proposed to provide a permanent fix.

## 50.59 EVALUATION

A 50.59 Evaluation was performed for this proposed change due to revision of FSAR Figure 10.4-2 Sheet 5. The conclusion reflects that removing the subject instrument sensing lines and installing pipe plugs in their place will not reduce the margin of safety as defined in the basis for any TS or Safety Analysis and that no USQ is created.
There are no accidents in the FSAR that may be affected by the proposed change. Significant leakage past the existing leak repair or new pipe plug may require a plant shutdown to repair. However, this would not be considered an accident initiator. The Condensate system and the subject instrument sensing lines are not credited for limiting radiological consequences. In addition, they are non-safety, non-seismic, non-quality, and not connected to any plant instrumentation or other systems. Removal of the instrument sensing lines and associated isolation valves does not affect the operation of the condensate system or any safety related system. The subject instrument sensing lines are not related to any accident analysis as described within the FSAR. This minor modification does not create any new system interactions that could result in the possibility of an accident of a different type. The only plausible failure would be external leakage, which is the current case now and is no different than other CD leakage (i.e. packing leakage). Installation of this minor modification will not result in any new interactions nor introduce any new methods of failure of equipment important to safety. No new system interactions were created as a result of the proposed minor modification and there is no reduction in the margin of safety as defined within the TS.

## 19. ER-W3-98-0642-01-00, Component Cooling Water (CCW) Makeup

### DESCRIPTION OF CHANGE

This ER will replace the spring return handswitches on CP-8 and CP-43 with handswitches that will be spring return from the start position only and have a maintain stop position. The legend plates for the new switches will be changed to show no spring return from the stop position. Some minor wiring changes will be necessary in Auxiliary Panel 1 and Auxiliary Panel 2.

### REASON FOR CHANGE

The manual switches for the CCW make-up pumps on CP-8 and CP-43 are three position handswitches with spring return to the center position. The positions are Start, Stop and spring return to the center position. With the handswitch in the center position, the CCW makeup pump will start on an auto start signal from either the CCW surge tank, the Emergency Diesel Generator (EDG) jacket water standpipe, or the Chilled Water (CHW) expansion tank. CR 97-2551 documented that a failure of a CCW makeup valve, in the open position, would result in continuous makeup to the CCW surge tank, EDG standpipes, or the CHW expansion tanks. In addition, the CCW makeup valves to the CCW Surge Tank (CMU-538A&B) are fail-open, air operated valves. Therefore, upon a loss of IA, continuous makeup to the CCW Surge Tank would occur if either of the makeup pumps were in operation and no operator action was taken to secure the pump. Since both CCW makeup pumps take suction from the Condensate Storage Pool (CSP), continuous makeup to any of the locations could result in overflow of the tanks, flooding in the RAB, and a loss of CSP inventory which is credited for Emergency Feedwater (EFW) usage in response to design bases accidents. This change will give the operator the ability to stop the pump if an auto start signal is present. This will help to prevent the depletion of the CSP with a single failure of a level switch in the surge tank, standpipe, or expansion tank.

### 50.59 EVALUATION

Although this change will modify the handswitch of the CCW makeup pump, the ability of the pump to support the safety systems will not be degraded. The new switch meets the same design material and construction standards applicable to the original design. No system interface is changed. Further the CCW makeup pump is not a credible initiator of an accident. The new switch will not change, degrade or prevent actions described or assumed in the SAR. The new switch does not alter any assumptions previously made in evaluating the radiological consequences of an accident described in the FSAR. The CCW makeup pump is important to safety and could be affected by the new switch. The most limiting failure associated with this configuration change occurs with the handswitch left in the stop position. With the handswitch in the Stop position, the ability of the CCW makeup pump to start on an automatic signal is disabled. The normal operator control board walkdowns will be used to ensure that the CCW makeup pumps switches are in the correct position and the automatic start function of the pumps is enabled. In such case that the

handswitch is mispositioned, disallowing the CCW makeup pump to respond when signaled, annunciation will warn the control room of insufficient level in the CCW surge tank, the EDG standpipe or the chilled water expansion tank, with ample time to correct the mispositioning as the CCW makeup is only anticipated to make up for minor leakage and normal system losses. There is no further modification to the operation of this circuit. Thus, the most limiting mode of failure associated with this change is a failure that disables the automatic start capability (i.e., the handswitch stuck in the stop position). The failure of the handswitch or the loss of a CCW make up pump is bounded by a failure of a single CCW pump, a failure of a single EDG, or a failure of a single essential Chilled Water pump, each of which is already analyzed by the FSAR. Both the old and new switches are in the Control Room and remain accessible during a DBA. This evaluation concludes that changes proposed by this ER will not reduce the margin of safety as defined in the basis of any Technical Specifications or safety analysis, and there are no unreviewed safety questions.

20. ER-W3-98-0629-00-00, CST, DWST, and PWST Loop Seal Manual Fill Line Addition

DESCRIPTION OF CHANGE

The proposed change installs a permanent connection between valves CMU-208 and DW-170. This physical change requires FSAR Figure 9.2-2 to be revised.

REASON FOR CHANGE

To eliminate the need to install a temporary hose between CMU-208 and DW-170 whenever performing the infrequent operation of manual filling the Condensate Storage Tank (CST), Demineralized Water Storage Tank (DWST), or Potable Water Storage Tank (PWST) loop seals from the CST pump discharge. This action is required whenever loop seal fill is needed and the vendor-supplied demineralizer is out of service.

50.59 EVALUATION

There are no accidents listed in the FSAR that may be affected by the proposed change, nor is the Condensate Makeup (CMU) system an accident initiator. This change is adding one inch diameter stainless steel tubing between CMU-208 and DW-170. Current plant conditions require Operations personnel to install a temporary hose between these two valves when performing "Infrequent Operations" Section 8.2, Filling Storage Tank Loop Seals through CST Pumps, of OP-003-007, System Operating Procedure Demineralized Water System. The proposed change involves the physical connection between CMU-208 and DW-170. In addition, the CMU system is not credited with limiting radiological release consequences. Both of these valves are non-safety and non-seismic and are not considered in the accident analysis of this facility. CMU-208 is a drain valve off the recirculation line of the Condensate Storage Tank pumps and DW-170 is a drain valve off of the fill header for the Condensate Storage Tank. Neither the components directly affected by the proposed change nor the supporting equipment is important to safety. The DW and CMU Systems are already connected but the proposed change creates a new permanent connection between CMU-208 and DW-170. Currently, a rubber hose is utilized by Operations under OP-003-007. This new connection is more reliable than the current hose and provides a permanent method for Operations to refill the Storage Tanks Loop Seals. This new interaction cannot create a new accident due to the fact that both systems are nonsafety and non-seismic and are not credited with mitigating any accidents. The proposed modification does not involve any protective boundaries. Thus, installing a permanent connection between CMU-208 and DW-170 will not reduce the margin of safety as defined in the basis for any TS or Safety Analysis and no USQ is created.

## 21. ER-W3-99-0152-00-00, DCT Sump Pumps

### DESCRIPTION OF CHANGE

This change is a minor modification that modifies the operating circuit of the Dry Cooling Tower (DCT) sump pumps. A manual switch will be added to the control circuitry of each pump. When this switch is placed in the "BYPASS" position, the Hi-Hi alarm contacts of the radiation monitor will be bypassed.

### REASON FOR CHANGE

During a Loss Of Offsite Power (LOOP), the DCT sump pumps, as well as their associated radiation monitors are de-energized. Provisions, both physically and procedurally, were made to reenergize the DCT sump pumps from the EDG'S. This was done in response to a LOOP coincident with a Probable Maximum Precipitation (PMP) event. When a radiation monitor is de-energized, it fails in the conservative direction and provides a Hi-Hi radiation signal to the circuits of the respective pumps. This prevents the sump pumps from pumping to their normal flow-path (currently the storm drain system to the 40 Arpent Canal). Operations must then either manually lineup the pumps to their secondary path, the LWM Waste Tanks, or restore power to the radiation monitors. Both of these actions are considered to be Operations Workarounds. This change will eliminate this burden from the Operations department.

### 50.59 EVALUATION

This safety evaluation concludes that there is no USQ due to the implementation of ER-W3-99-0152-00-00. The non-safety DCT Sump Pumps are not an accident initiator but they do provide a supporting function during implementation of Emergency Operating Procedures (EOP). It has been postulated in the FSAR, that a LOOP coincident with a Probable Maximum Precipitation (PMP) event could flood the DCT area, subsequently flooding the Ultimate Heat Sink Switchgear and Transformer. Based on this postulation, provisions were provided in the EOP's to reenergize the DCT Sump Pumps from the EDGs via a manual switch. However, this action does not energize the Radiation Monitors that are required for discharge. During a PMP event, storm water is collected and routed to two DCT area drain sumps. The normal discharge flow path of this storm water is into the gravity drainage system for offsite disposal through a radiation monitor. High radiation will automatically stop the pumps. The operator may then manually transfer the flow to the Waste Tanks. When a LOOP is concurrent with this condition, the DCT Sump Pumps and radiation monitors are stripped from the safety busses. The pumps can be loaded manually onto the EDGs after one half-hour. However, for the normal discharge path to be used without radiological monitoring, the requirements of TRM 3.3.3.10.b must still be followed including prompt restoration of the radiation monitors and "grab sample" analysis. The pumps will be capable of being operated even with the monitors disabled - an alarm makes the Control Room aware of this situation. This change does not alter the reliance on the DCT Sump Pumps to keep the MCCs/transformer dry. These components are necessary to provide power the

DCT fans. A postulated failure of a bypass switch has the same consequence (loss of safety related fans) as the failure of a sump pump/circuit breaker. The radiation monitors will still detect activity in the discharge of the DCT Sump Pumps and initiate a trip as designed. This ER will allow the trip to be bypassed by adding a switch contact in parallel with the existing trip contact. This bypass feature will only be used when the trip is the result of a de-energized radiation monitor. This is a conservative action to prevent flooding of the safety related MCCs/transformer. There are no new methods of failure as a result of this ER (this failure mode is bounded by failure of the DCT Sump Pumps or breakers). A Control Room alarm will alert the operators anytime the new switches are placed in the BYPASS position. There is no other affect on the pump controls. This change does not directly affect a protective boundary. The switchgear and transformers required for the DCT fans can not be adversely affected by the addition of these switches. This change does not represent an unreviewed safety question and there are no margins of safety affected by this ER.

22. ER-W3-98-0936-00-00, Evaluation of Discrepancy in FSAR Section 8.3.1.1.2.1.3f and Table 8.3-1

DESCRIPTION OF CHANGE

ER-W3-98-0936-00-00 evaluates the impact of non-safety equipment, which is sequenced on the engineered safety featured (ESF) bus after a loss of offsite power (LOOP). These loads include non safety SUPS (Plant Computer, Plant Security & AB), Emergency Diesel Generators A(B) (Compressors 1 & 2), Emergency Lighting Panels and non safety Power Distribution Panels.

REASON FOR CHANGE

CR-98-0763 and NRC URI 50-382/98-201-14 identified a condition that non-safety loads are sequenced on the ESF bus after a LOOP, noting that this was not in agreement with FSAR section 8.3.1.2.15(e) 7. The identified section states that reconnection of non-essential loads can only be done manually under administrative control. However, FSAR Table 8.3-1 and associated design documents identify non-safety loads sequenced onto the ESF buses. The FSAR contains information on separation criteria for non-safety loads that are sequenced automatically on the EDG. The ER was written to resolve the FSAR discrepancies and to resolve the issue of the identified non-safety load sequenced on the ESF buses.

50.59 EVALUATION

The EDG is required to mitigate the consequences of an event coupled with LOOP. There are no accidents that could be caused by the present plant configuration associated with the sequencing of nonsafety loads on the ESF buses after a loss of offsite power. The loading of the non-safety loads on the emergency diesel generators has been incorporated into calculation EC-E90-006 for fuel oil consumption and steady state loading of the EDG. Proper isolation exists between the non-safety loads and the class 1E buses. The present plant configuration allows for coordinated plant protective devices to prevent damage to the ESF bus. The computer SUPS and bypass supply have been evaluated for seismic and environmental related failure modes. The evaluation concludes that auto sequencing of these loads will not compromise the ability of the EDG to supply emergency power to safety related loads and therefore will not affect radiological release consequences. The non-safety loads identified in this evaluation receive power from class 1E equipment. However, these loads have double protection such that failure of one protective device will not have a negative impact on the class 1E power source. This design feature is in the licensing bases of the plant. It is concluded that the change to the FSAR to clarify the original design configuration of the plant will not change the probability of occurrence or consequences of accidents or malfunction of equipment. This evaluation does not create any new system interaction but resolves a discrepancy between sections of the FSAR. There are no physical changes being made to the plant. This evaluation provides the basis for the present plant configuration which allows non-safety loads to be sequenced on the ESF bus and has no affect on any margins of safety.

23. ER-W3-98-1178-00-00, EC-M98-027 Safety Injection System – LPSI Flow Rate Calculation

DESCRIPTION OF CHANGE

Calculation EC-M98-027 addresses the single active failure of a Low Pressure Safety Injection (LPSI) flow control valve to the full open position on receipt of a Safety Injection Actuation Signal (SIAS). The calculation determines the maximum total (including minirecirculation) LPSI pump flow rate, as well as the maximum LPSI injection flow rate to the cold legs. The calculation concludes that the single failure of a LPSI flow control valve in the full open position will not cause the LPSI injection flow rate to exceed safety analysis assumptions, and the total LPSI pump flow rate will not cause the LPSI pump to runout.

Engineering Request ER-W3-98-1178-00-00 adds information to Section 6.3 and Table 6.3-1 (FMEA Table) of the SAR. The single failure of a HPSI or LPSI flow control valve failing to the full open position on receipt of a SIAS is added to Table 6.3-1. Remarks are also added to the table to reflect the calculation conclusion that the failure of a flow control valve in the full open position will not cause the LPSI injection flow rate to exceed the maximum LPSI flow rate assumed in the safety analysis. Section 6.3 is also revised to reflect the results of calculation EC-M98-027. The ER also makes an editorial correction to SAR Figures 6.3-2a and 6.3-2b (LPSI pump curves) to be consistent with the certified manufacturer pump curves. Table 6.3-1 is also revised to add the UNID numbers to the HPSI and LPSI flow control valve entries.

REASON FOR CHANGE

The single failure of a HPSI or LPSI flow control valve in the full open position is not listed in the FMEA (Table 6.3-1). The safety analyses for the large break LOCA assumes a maximum LPSI injection flow rate of 5650 gpm from each LPSI pump. As stated in SAR Section 6.3.3.2.1, only half the flow generated by one LPSI pump is required to maintain a full reactor downcomer. Any LPSI flow in excess of this will spill out the break, reducing containment pressure and increasing the blowdown rate through the break. The FSAR is revised to document this single failure, as well as the calculation EC-M98-027 conclusion that a single active failure of a LPSI flow control valve will not affect the safety analyses. The single active failure of a HPSI flow control valve in the full open position is also added to the SAR for completeness, including the compensating provision of having a redundant HPSI subsystem. However, maximum HPSI injection flow is not a safety analysis parameter.

50.59 EVALUATION

The large break LOCA evaluation model (as discussed in FSAR Sections 6.3.2.2.2.1, 6.3.2.9.8, and 6.3.3.2.1) assumes the maximum safety injection flow rate is pumped to the reactor coolant cold legs by the two HPSI pumps and the two LPSI pumps. The maximum possible flow rate is the most limiting condition for a large

break LOCA because it produces the maximum spillage from the reactor coolant line break and a lower reflood rate to the reactor vessel. The maximum flow rate assumed for each of the LPSI trains in the large break LOCA model is 5650 gpm. The maximum flow rate calculated in EC-M98-027 for each LPSI System trains, assuming the throttle valves in the LPSI train go to the full open position, is approximately 5370 gpm. The calculated flow rate of 5370 gpm is less than the assumed flow rate used in the large break LOCA evaluation model. The maximum total (including mini-recirculation) LPSI pump discharge flow rate is 5450 gpm. This flow rate is less than the pump run out flow of 5650 gpm as noted in SAR Table 6.3-2.H. Therefore, this change does not represent a USQ.

24. ER-W3-99-0973-00-00, Inactivation of Steam Generator Blowdown Demineralizer Regeneration and Electro-Magnetic Filter Components

DESCRIPTION OF CHANGE

This change will isolate Steam Generator (SG) equipment that is no longer used. This isolation will be performed by closing identified boundary valves and removing power to various components to the maximum extent possible by opening identified circuit breakers. Affected equipment will be identified as inactive in the Component Database. Control Room drawings and Licensing Bases documents will also be revised accordingly.

REASON FOR CHANGE

Waterford 3 steam generator blowdown demineralizers are currently using non-regenerable resin, which upon exhaustion is sluiced from the vessels, de-watered, and processed for off-site disposal per Departmental Procedures. New resin is then added to the blowdown demineralizer vessels for continued system operation. The original system was designed to include the capability to regenerate resin within the Blowdown Demineralizers. As this capability is no longer used the supporting chemical feed subsystems including acid/caustic tanks, pumps, piping and controls are no longer required.

The Waterford 3 electromagnetic blowdown filters have been proven to be ineffective and are no longer used. The electromagnetic blowdown filters function of removal of suspended iron oxide particulates from the steam generator blowdown effluent is adequately performed by the blowdown demineralizers and condensate polishers. As a result, operation of the supporting subsystems including the filter flush tank, pump, piping, accumulator and controls are no longer required.

50.59 EVALUATION

The proposed change is within the existing licensing basis of Waterford 3. This 50.59 Evaluation documents the fact that the proposed change does not result in an USQ. In addition, it does not put the plant operation in an unanalyzed region. The changes herein are bounded by the analysis in the TS, the TRM and the SAR. The proposed change does not downgrade the performance of any structure, system, or component as defined in the SAR, the TRM or the TS.

Because the changes described above will meet or exceed the requirements of the original design (component integrity, capacity, functionality, etc.) and existing analyses, they will not degrade any important to safety systems, components, or structures nor will they degrade or prevent actions described in the SAR accident analysis. The changes do not increase the probability of occurrence or increase the consequences of malfunction of equipment important to safety or of a different type than previously evaluated in the SAR. The TS and the TRM are not affected, and the margin of safety remains unchanged. Therefore, this change does not constitute a USQ.

25. ER-W3-97-0066-00-01, Remove Radiation Protection Partitions and Associated Structures in the RAB

DESCRIPTION OF CHANGE

Remove radiation protection partitions and associated structures surrounding Reactor Shutdown Cooling Suction Lines A & B in RAB -4.0 wing area.

REASON FOR CHANGE

The partitions are no longer needed as controlled access area because they are seldom used. In the wing area, there are other shutdown cooling lines without partitions that also have the potential to change radiological status to the general area dose rate. When this develops, RP will control access to the entire wing area instead of individual locations. Also, the partitions interfere with proposed Safety Injection venting service platforms.

50.59 EVALUATION

These partitions were provided to aid the ALARA program, have no safety classification, and are non-seismic but are designed not to fall and interact with safety-related components during an SSE. There are no accidents whose probability or consequences will be increased by this change. No new system interconnections or failure modes are created by this change. There is no margin of safety associated with the partitions or associated structures.

26. ER-W3-98-0948-00-00, Replace Door 150

DESCRIPTION OF CHANGE

The proposed change replaces right-hand Door 150 with a right-hand reverse door.

REASON FOR CHANGE

To perform properly as a fire door, Door 150 must shut and latch after use. A closer is installed on the door to make sure this happens. However the ventilation DP across the door tends to hold it open. Changing the configuration of the door will allow the DP across it to force it shut.

50.59 EVALUATION

Door 150 is a rated fire door and is used for fire protection purposes. The direction of opening will not change its fire rating or change its function as a fire door. No new failure is introduced by this change as this change will make it identical to existing Door 151 which is a right hand reverse door located within a few feet of and in the same room as door 150. No accident or equipment important-to-safety are affected by this change and no USQ is created.

27. ER-W3-98-0590-00-00, Potential to Void in ACCW System to Essential Chillers

DESCRIPTION OF CHANGE

The proposed change adds a design feature to the Auxiliary Component Cooling Water (ACCW) system to eliminate the potential of a void and hydraulic transient when swapping the Essential Chiller condenser cooling from the Dry Cooling Tower (DCT) to the Wet Cooling Tower (WCT). This change will create an interlock between the Essential Chiller supply and return valve circuitry. A limit switch associated with supply valve ACC-112 A(B) installed in return valve ACC-139A(B) circuit will be wired to prevent the opening of the return valve prior to the supply valve.

REASON FOR CHANGE

There is a potential of the ACCW system to void when swapping the Essential Chiller condenser cooling from the DCT to the WCT if valve ACC-139A(B) opens before valve ACC-112A(B).

50.59 EVALUATION

There are no accidents for which the ACCW system is considered an initiator and no accidents whose consequences are adversely affected by this change. Addition of the limit switch will eliminate the possibility of a common mode failure due to a hydraulic transient which could render both trains of ACCW inoperable. However, the safety function of ACCW will not be adversely affected. Because the switch uses the same standards and construction as existing limit switches, no new accident or failure mode will be created. No protective boundaries are affected and no margin of safety reduced by this change.

## 28. ER-W3-98-1088-00-00, Replacement of Main Transformer 'B'

### DESCRIPTION OF CHANGE

The proposed change replaces Main Transformer 'B' with an ABB Power T&D model. The replacement was evaluated under procurement specification Qual-E-018.

### REASON FOR CHANGE

The trend of oil samples taken from the installed Main Transformer 'B' led to a recommendation to replace the equipment before a failure occurred.

### 50.59 EVALUATION

The 50.59 evaluation has concluded that replacing the existing Main Transformer 'B' with the ABB Power T&D model does not represent a USQ. The Main Transformers do not initiate any accidents and there are no accidents previously evaluated in the FSAR which would have an increase in consequences as a result of this change. Two parameters which are different for the new transformer are total weight and impedance. The change in weight is negligible. The slightly lower impedance of the replacement transformer causes it to be slightly more loaded (~1%) than the other transformer. This is within the ratings of the transformer and does not affect any grid analysis. In addition, neither of these conditions will cause an increase in the probability or consequences of an equipment malfunction. There are no new accidents or equipment malfunctions created and no margin of safety is reduced by this change.

29. ER-W3-98-0277-00-00, Design Documentation Update/Revision for ABB Refurbished Startup Transformer

DESCRIPTION OF CHANGE

This ER evaluates the refurbished Westinghouse transformer, (now ABB transformer serial number XLN8125), in place of the existing Startup Transformer (SUT) B. This change will affect the Main Transformers and Switching Station (TSS) system only. Because of the different impedance values associated with the refurbished transformer, UFSAR table 8.2-1 is revised by this ER. The change will not affect the function or capability of any plant structure, system, or component.

REASON FOR CHANGE

The refurbished Westinghouse transformer matches the specifications of the original transformer with two exceptions: the impedance and the weight. This change evaluates these differences to verify that they will have no adverse effects to plant operation or safety.

50.59 EVALUATION

The installation of the refurbished Westinghouse transformer will not affect the design criteria of the startup transformer, which includes the inability of the transformer to cause an accident upon its failure. The plant response to a Loss of Offsite Power (LOOP) concurrent with an accident will be unchanged, as analyzed by the FSAR. The transformer has been tested and evaluated for operability, reliability, and compatibility. The results show that the transformer is a suitable component to be utilized as the plant startup transformer with no modes of operation or consequences of failure different than what is already analyzed. No new consequences of failure or probability of failure of other plant equipment is introduced by this change.

30. ER-W3-98-0883-00-00, Disable Close Intercept Valve (CIV) from DEH Computer

DESCRIPTION OF CHANGE

Digital Electro-Hydraulic Control System (DEH) performs two main functions: control of main turbine speed and control of main turbine load. This is accomplished by controlling steam flow to the turbine by positioning Reheat Stop Valves (RHSVs), Intercept Valves (IVs), Throttle Valves (TVs), and Governor Valves (GVs). A sub-system of DEH is the Overspeed Protection Control (OPC) system. OPC consists of three parts: Close Intercept Valves (CIVs), Load Drop Anticipation (LDA), and overspeed control. CIV, or fast valving, provides an improved margin of stability during a partial load loss and is based on a mismatch between turbine mechanical power and electrical load. Closing the intercept valves provides a momentary reduction in generator output and aids in maintaining power system stability. This change will disable the CIV function by installing an electrical jumper in the DEH control panel which results in a CIV inhibit signal being generated. Also, the cables that would be used to close the intercept valves if a CIV actuation signal were to occur will be disconnected.

REASON FOR CHANGE

The CIV function is designed primarily for units in operation with a limited or no grid system, where the potential exists for an on-line overspeed event to occur. On-line overspeed is not considered a potential failure mechanism with the system connected to a large grid. The amount of hardware utilized to perform the CIV function and its likelihood of failure provide several "single point failure" paths which could lead to a reactor power cutback or a reactor trip. Westinghouse has recommended that the CIV function be disabled to improve plant reliability.

50.59 EVALUATION

The proposed change disables the CIV function from the non-safety DEH system. No new system interfaces are created by this proposed change. Disabling the CIV function will eliminate the potential for equipment damage and reactor power cutbacks or trips due to inappropriate actuation of this function. All other electrical and mechanical overspeed protection devices remain unchanged. Therefore, it is concluded that this modification will not affect the safety or environmental aspects of any licensing basis documents, will not reduce the margin of safety as defined the basis for any TS, and no USQ is created.

31. ER-W3-98-0626-00-00, Treated Effluent Pump Discharge to Industrial Waste Sump

DESCRIPTION OF CHANGE

This change will delete the original Condensate Polisher Treated Effluent pump discharge to the Metal Waste Pond and route the discharge directly to the industrial waste sump.

REASON FOR CHANGE

The original piping design provides a discharge path from the Treated Effluent Pumps to the Metal Waste ponds. This discharge path is no longer used due to the tritium present in the Condensate system. However, there is the potential to accidentally discharge treated effluent to the Metal Waste Ponds. This change will provide permanent piping so that a temporary hose will not be required to pump treated effluent to the Industrial Waste Sump.

50.59 EVALUATION

The functions of the Condensate Polisher Building Sumps Pumps and the Industrial Waste Sump are not affected by this change. In addition, the portions of the Condensate Polisher and Industrial Waste Sump system directly affected by this change are not credited for limiting the radiological consequences of an accident. All new components used to replace the temporary hose meet the original system design criteria. No margin of safety is reduced and no USQ created by this change.

32. ER-W3-98-0889-00-00, Add Switch to Place HVC System in Recirculation for Testing

DESCRIPTION OF CHANGE

This modification will install two key-controlled, single pole, double-throw switches to the North side outside air intake radiation monitors which, when placed in the "RECIRC" position, will deenergize Area Radiation Monitors (ARM) relays which in turn will place the Control Room Ventilation (HVC) system in the recirculation mode of operations.

REASON FOR CHANGE

TS surveillance requirements require the HVC system to be placed in the recirculation mode of operation with the ability to pressurize the control room envelope. To perform this task, the hi-radiation setpoint to the Control Room Outside Air Intake (CROAI) Radiation Monitors must be lowered to initiate a false signal. This places the HVC in the recirculation mode of operation. The manipulation of the setpoint requires several procedural steps to complete. Due to the numerous procedural steps required to be performed by Operations personnel, and the potential for human error in restoring the radiation monitor setpoints, this task has been classified as an Operations Workaround.

50.59 EVALUATION

The purpose for the Control Room Ventilation (HVC) system being placed in the "Recirculation" mode of operation with the ability to pressurize the control room envelope is to minimize the radiation dosage to Operations personnel within the control room envelope during and following accident conditions. A Safety Injection Actuation Signal (SIAS) or a High Radiation signal initiates the "Recirculation" mode of operation during accident conditions. The radiation monitoring system detects actual radiation levels at two different intakes. This allows operations personnel to select the intake path that is admitting air with the lowest concentration of radioactivity. The proposed change does not affect the ability of the radiation monitoring system from either providing the "Recirculation" signal or providing radiation concentrations at the selected intake paths. In addition, the Safety Injection Actuation Signal is not prevented from operation due to this proposed change. The addition of two keylock switches have the same technical and quality requirements of the radiation monitors (class 1E, seismic category 1). In addition, the proposed change is wired to ensure the system fails safe. If the switches were to fail in the closed position (their normal position), the radiation monitors will still monitor the incoming air. If a valid actuation has occurred, the radiation monitoring system will place the Control Room Ventilation system in its fail-safe position. If the subject switches were to fail in the open position, the control room isolation airborne radiation monitor relays will be deenergized and place the HVC system in its fail safe position. It is concluded that installing the proposed modification will not reduce the margin of safety as defined in the basis for any TS and that no USQ is created.

33. ER-W3-98-0869-00-00, Upgrade HBC Unit to Meet EPRI PPPM for Valves SI-602A and B

DESCRIPTION OF CHANGE

This ER will perform the following on the Safety Injection (SI) sump isolation valves, SI-602A and B: (1) replace the Limitorque H1BC gear head and associated mounting bracket with a Limitorque H2BC gear head and new mounting bracket, and (2) incorporate the EPRI Motor Operated Valve Performance Prediction Program Methodology (MOV PPPM) into the design requirements for the SI sump isolation valves, SI-602A and B, to ensure compliance with the W3 GL 89-10 MOV program and associated NRC commitments, and (3) replace the actuator spring pack.

REASON FOR CHANGE

During the closure of the NRC review of the Waterford 3 GL 89-10 program, W3 committed in NRC letter ILN 94-0281 to use the EPRI MOV PPPM to more precisely determine the performance requirements of the SI sump isolation valves, SI-602A and B. The basis for the use of the EPRI MOV PPPM was to provide a design verified analytical methodology in lieu of performing a dynamic Differential Pressure (DP) test on valves SI-602A and B. Since design basis conditions cannot be achieved for valves SI-602A and B, in-situ dynamic DP testing was deemed impracticable to perform under the W3 GL 89-10 program.

The torque requirements calculated using the EPRI PPPM indicate the existing design is not adequate to achieve the calculated torque values without exceeding the Limitorque H1BC unit general design rating. Based on the Waterford 3 commitment to continuously ensure adequate design basis MOV capability is maintained, valves SI-602A and B will be modified with larger capacity HBC units and spring packs in order to implement the NRC commitment and meet the EPRI MOV PPPM torque requirements. Modifying these valves to allow greater torque output provides additional assurance that the valves are capable of performing their safety related functions. The associated mounting bracket and bolting for the HBC unit will also be replaced to accommodate installation of the larger unit.

50.59 EVALUATION

This modification will enhance the design of valves SI-602A and B thus providing added assurance that the valves are capable of performing their safety function. The torque requirements calculated using the EPRI PPPM are more conservative than the existing vendor requirements. Valves SI-602A and B will be modified with larger capacity HBC units and spring packs in order to meet the EPRI MOV PPPM torque requirements. Modifying these valves will allow greater torque output without exceeding the HBC design rating and will in turn provide additional design margin between the minimum torque requirements and the actuator capability. The new Limitorque H2BC unit is very similar in design to the existing H1BC unit and the basic operation of the two actuators is identical. The stroke time for valves SI-602A and B will remain unchanged because the new HBC unit will have the same gear

ratio (70:1) as the existing unit. Therefore, the stroke times of 25 seconds nominal and 35 seconds maximum documented in the FSAR as well as the imposed stroke time of 50 seconds utilized in calculation EC-M98-008 will be preserved. The affected piping and pipe supports remain within allowable stress levels per the applicable codes and standards. In summary, the safety evaluation determined that no USQ exists as a result of this modification.

34. ER-W3-98-1380-00-00, Appendix R Problem for Static Uninterruptible Power Supply (SUPS) 3B-S

## DESCRIPTION OF CHANGE

Twenty-four non-essential unprotected circuits connected from SUPS 3B-S are routed through the cable vault. For a fire in the control room/cable vault area, these non-essential circuits may have the potential to impact the ability to safely shutdown the plant. It is required that these circuits be manually isolated via their associated circuit breakers located in PDP-391 to maintain plant safe shutdown capability during a control room fire event. The analysis assumes a fire would cause a series of cable faults one after another. Simultaneous faults are not postulated since the fault only lasts for a very short duration of approximately 10 to 45 milliseconds which is the time it takes for the PDP391 circuit breaker (Heineman Curve 2) to isolate the fault.

## REASON FOR CHANGE

A fire in the control room or cable vault could impact plant safe shutdown capability. The scenario assumes that the unprotected non-essential cables would fault and draw high current from the SUPS 3B-S. The SUPS would limit current when the resulting combination of fault current and load current exceed the SUPS capacity to supply power. While the SUPS is in the current limiting mode, the output voltage would degrade to an unacceptable voltage level and could potentially have an adverse impact on the operations of the essential equipment required for safe shutdown. The safe shutdown equipment for each fire area is required to be isolated from the non-essential circuits in the fire area so that hot shorts, open circuits, or line-to-line fault in the non-essential circuits will not prevent operation of the safe shutdown equipment. Therefore, to maintain plant safe shutdown capability during a control room fire event, the identified unprotected, non-essential circuits are required to be isolated.

## 50.59 EVALUATION

The proposed activity isolates those non-essential circuits that may have the potential to mitigate the ability to safely shutdown the plant during a control room fire event. This activity does not increase the probability or consequences of a malfunction of the SUPS 3B-S or its connected loads in any way. OP-901-502 Revision 5 incorporates the stripping of these circuits. A concurrent accident is not postulated during a control room fire event and therefore, the probability and consequences of an accident do not increase and the margin of safety is not reduced. During the control room fire event, the design objective is to achieve and maintain safe shutdown of the plant from remote control panel LCP-43, as directed by the applicable plant procedure. The equipment necessary to achieve and maintain safe shutdown is available and safe shutdown can be achieved and maintained. Thus, no USQ is created by this change.

35. ER-W3-97-0706-00-00, Blowdown Demineralizer Annunciator Deficiencies

DESCRIPTION OF CHANGE

This change permanently disables 18 annunciator points which are associated with the regenerative portion of the Steam Generator Blowdown System (SGBDS) that is no longer in service.

REASON FOR CHANGE

The Annunciator and Instrumentation Audit directs that for each deficiency over 12 months old, an ER be initiated for resolution and FSAR compliance.

50.59 EVALUATION

The proposed change permanently disables annunciator points associated with the regenerative portion of the SGBDS that is no longer in service. The SGBDS annunciator is not the initiator for any accident previously evaluated in the FSAR and does not affect the consequences of any accident previously evaluated in the FSAR. Deactivation of the local alarms affects only the operation of the SGBDS, which is not shown in the FSAR and performs no safety function. Therefore, no malfunction of equipment important-to-safety is affected by this change. No new system interconnections are required and no new failure modes are created; therefore, no new accidents or equipment malfunctions are created. No protective boundaries or margins of safety are affected by this change.

## 36. ER-W3-00-0180-00-00, Emergency Core Cooling System Pump NPSH

### DESCRIPTION OF CHANGE

Revise FSAR Sections 6.2.2.2.2.1 (Safety Injection System Sump Design), 6.2.2.3.2.1 (NPSH Calculations), 6.3.2.2.2.3 (Net Positive Suction Head), and Table 6.2-22 (Design Data for Containment Spray System Components). Revise W3-DBD-013, Containment Spray based on revisions to calculations. The calculation revisions result in the following FSAR changes:

- Safety Injection System (SIS) sump minimum outlet submergence is changed from 8 ft. to 7.5 ft.
- Containment Spray (CS) NPSHa is changed from 24.62 ft. to 24.1 ft. including associated margins
- High Pressure Safety Injection (HPSI) NPSHa is changed from 21.77 ft. to 21.3 ft. including associated margins
- Containment water level elevation (MSL) changes from –5.47 ft. to –5.95 ft.

### REASON FOR CHANGE

The FSAR and DBD require revision based on revisions to calculations MN(Q)-6-4, "Water Levels Inside Containment," and MN(Q)-6-27, "NPSH Calculation for HPSI and CS Pumps During Recirculation." Calculation MN(Q)-6-4 was revised to incorporate the conservative assumption that there is no free communication of water between the Reactor Cavity and the Containment Floor during a Loss of Coolant Accident. The calculation revision determined that the reduced water levels in the sump do not alter the sump model test conclusion that vortexing will not occur. Calculation MN(Q)-6-27 was revised to use the new minimum water level in containment in the determination of new static heads at the HPSI and CS pump suctions. The calculation conservatively used HPSI and CS pump runout flows as well as a more accurate as-built configuration of the HPSI and CS pump suction piping in the determination of friction losses in the pipes. The calculation methodology used in the determination of the new available Net Positive Suction Head values exceed the expectations of Reg. Guide 1.1 by using a saturated sump model which assumes that the water vapor pressure is equal to the containment pressure. While this calculation conservatively determined NPSHa values smaller than what was previously calculated, the calculation shows that sufficient NPSHa exists at the HPSI and CS pump suctions to allow proper pump operation even at runout flows.

### 50.59 EVALUATION

The proposed FSAR and DBD changes are within the existing licensing basis of Waterford 3. This 50.59 Evaluation documents the fact that the proposed changes do not result in a USQ. In addition:

1) It does not put the plant operation in an unanalyzed region. The changes herein are bounded by the analysis in the TS, the TRM and the FSAR.

2) The proposed change does not downgrade the performance of any structure, system, or component as defined in the TS, the TRM or the FSAR.

Because the changes described above will meet or exceed the requirements of the original design (component integrity, capacity, functionality, etc.) and existing analyses, they will not degrade any important to safety systems, components, or structures nor will they degrade or prevent actions described in the FSAR accident analyses. The changes do not increase the probability of occurrence or increase the consequences of malfunction of equipment important to safety or of a different type than previously evaluated in the FSAR. The TS and the TRM are not affected, and the margin of safety remains unchanged. Therefore, this change does not constitute a USQ.

37. ER-W3-99-0838-00-00, Flooding Analysis Outside Containment

DESCRIPTION OF CHANGE

FSAR Section 3.6A.6.4 is revised to show the results of calculation MN(Q) 3-5, Revision 3. The assumptions and design criteria used to determine the effects of flooding in areas outside the containment due to high energy pipe break, moderate energy pipe through wall crack, or actuation of a fire protection suppression system remain as previously discussed in the FSAR. The changes to the FSAR relate to the specific acceptance criteria used for each area to determine and verify that the plant operation, systems, component, or structures are not adversely affected when flooding is considered. There are no physical, operational, or procedural changes to the plant proposed or required by the calculation revision or FSAR changes.

REASON FOR CHANGE

The W3 design basis review program identified open items and upgrade recommendations for calculation MN(Q) 3-5.

50.59 EVALUATION

This change updates the FSAR to reflect the results of calculation MN(Q) 3-5, Revision 3. The calculation demonstrates that the safety-related components located in all areas (except the FHB, because equipment important for safe shutdown of the plant is not located in the FHB) outside the containment required to safely shut down, and maintain the reactor in cold shutdown are adequately protected from area flooding due to a postulated high energy pipe break, moderate energy through wall crack, and/or a postulated actuation of fire protection sprinklers. There is no USQ as a result of these changes to the FSAR.

38. ER-W3-99-0623-00-00, Revision to Fire Sprinkler System Specifications

DESCRIPTION OF CHANGE

ER-W3-99-0623-00-00 evaluates and re-establishes the design basis for plant fire sprinkler systems. The ER supports the revision of sprinkler system design densities for several systems. EBASCO Project Specifications LOU-1564-124D and 124D-A provided design densities for the plant sprinkler systems. Some of the sprinkler systems are specified to have a design density excessively conservative when compared to NFPA 13 and NFPA 15. NFPA 13 and NFPA 15 are the code of record documents as indicated in FSAR 9.5.1 (Section 9.5.1.1.4) and NRC SSER 3.

REASON FOR CHANGE

The design basis of the sprinkler systems is based on the code of records (NFPA 13 and NFPA 15). The change to the specified sprinkler densities eliminates overly conservative sprinkler discharge density requirements. The revised densities are consistent to that as would be expected for the hazards of the area and as typically selected when applying the design curves of the codes.

50.59 EVALUATION

The required design of the plant sprinkler systems is compliance to NFPA 13 and NFPA 15. A change that eliminates the overly conservative discharge design density maintains full compliance to the applicable codes while at the same time presents no impact or concerns related to the safety of the plant. There is no USQ associated with these changes.

39. ER-W3-97-0457-00-00, Evaluation of the Overfilled Plan Points that may Cause Cable Degradation

DESCRIPTION OF CHANGE

The changes covered by this ER are 1) revise FSAR section 3.10 to reflect that cable tray supports are considered rigid at frequencies less than 33 Hz, 2) correct a typographical error in FSAR section 3.10, 3) issue several calculation changes/revisions to show increases in weight on cable tray supports due to overfilled cable trays, and 4) revise PDMS (CCL) to reflect the cable tray plan points that are allowed to be increased above 60% fill.

REASON FOR CHANGE

Just prior to issuing the operating license for Waterford 3, the NRC performed a construction audit of the plant. Audit finding item 1.3 stated that additional loads had been added to cable tray supports without any further analyses being performed. Prior to this audit, the cable tray supports were designed as rigid supports (frequency > 33 Hz.). To reanalyze these supports and close the audit finding, Waterford changed this requirement to be that the supports were considered to be rigid if the natural frequency was 16 Hz or greater for supports attached to concrete and 20 Hz or greater for supports attached to flexible steel floor framing. This method of analyzing the supports was reviewed and accepted by the NRC in closing out the audit findings. However, FSAR section 3.10 was never updated to reflect this design criteria. Additionally, over the life of Waterford 3, modifications issued by DE-Electrical have increased the amount of cable fill in some cable trays so that they exceed the 60% fill allowable as defined in the PDMS (Cable and Conduit Listing). Calculations for the trays were based on a 60% fill. Some of the cable trays were actually filled to as much as 85%.

50.59 EVALUATION

This change revises the FSAR to reflect the actual design method used for cable tray supports and revises numerous calculations using this method to show as-built status. This FSAR change does not result in any physical changes to the plant. The stresses due to the revised cable tray loads remain within code allowables. The main impact of this change is on the design of the cable tray support itself. Since the cable trays at certain plan points already have a fill greater than 60%, this change does not affect any electrical concerns with the cable trays themselves. The evaluation has concluded that the trays that have up to 85% fill are adequately supported to withstand postulated events. Thus the trays that have exceeded the design requirement of 60% fill will not fail during a seismic event. Changing the description of how the cable tray supports are analyzed does not change the design criteria for W3. The changes do not increase the probability of occurrence or consequences of accidents or malfunctions of equipment and there is no affect on any margins of safety.

40. ER-W3-98-1220-00-00, Limitorque Technical Update 98-01 Modifications

DESCRIPTION OF CHANGE

Changes are being made to the following valves:

- MS-119B, the Main Steam Isolation Valve (MSIV) upstream emergency drain valve, is a normally closed motor-operated globe valve that opens to allow excess condensate to flow from drip pots upstream of the MSIVs. On a high level the valve will open, bypassing the steam trap and the normal drain valve MS-120B. The valve performs a containment isolation function and will close automatically on a Containment Isolation Actuation Signal (CIAS). This change will replace the gear set in the SMB-00 actuator resulting in an increase in the stroke time from approximately 4 to 7 seconds.

- MS-120B, the MSIV upstream normal drain valve is a normally open motor-operated globe valve that allows condensate to flow from drip pots upstream of the MSIV's through a steam trap to the condenser during normal operations. The valve performs a containment isolation function and will close automatically on a CIAS. This change will replace the gear set in the SMB-00 actuator resulting in an increase in the stroke time from approximately 4 to 7 seconds.

- SI-120A(B) and SI-121A(B), the Safety Injection (SI) recirculation header to RWSP isolation valves are normally open motor-operated gate valves. These valves are required in the minimum recirculation line to isolate the Refueling Water Storage Pool (RWSP) from the SI and Containment Spray (CS) pump discharge headers after a Recirculation Actuation Signal is initiated. The minimum recirculation lines are potential back leakage paths of highly contaminated water from the SI sump to the atmospherically vented RWSP in post accident conditions. This change will replace the existing 10 ft-lb motors with 15 ft-lb motors. The thermal overload protection for the new motors will also be replaced. The replacement of the motor will also require a change to FSAR table 8.1 -1. There is no impact on stroke time for these valves.

- SI-135A(B), the SI Shut Down Cooling (SDC) warm up line isolation valves, are normally closed motor-operated gate valves. These valves allow Low Pressure Safety Injection pump discharge to warm the SDC connection to the Hot Legs. Warming this line reduces the thermal shock to the piping when SDC starts to draw water from the Reactor Coolant System. These valves have no automatic actuation. This change will replace the gear set in the SMB-00 actuator resulting in an increase in stroke time from approximately 40 to 90 seconds which will not affect any FSAR or TS stroke time limits. There is no impact on the overall weight of the motor-actuator. Replacement of the existing spring pack to a stiffer spring pack is also required.

- SI-219A & B, the High Pressure Safety Injection orifice bypass valves are normally locked open motor operated gate valves. During the injection mode of operation, the orifice bypass valves are open allowing 100% of the HPSI pump discharge to flow to the Cold Leg injection header. When Hot Leg injection is

required, the bypass valves are remote manually closed and the orifice equalizes flow between the Hot and Cold Leg injection paths. The valves are normally locked open and are designed to fail as-is. For SI-219A, this change will replace the existing 7.5 ft-lb motor with a 10 ft-lb motor and the gear set in the SMB-00 actuator resulting in an increase in stroke time from approximately18 to 45 seconds which will not affect any FSAR or TS stroke time limits. The replacement of the motor will also require a change to FSAR table 8.1-1. For SI-219B, this change will replace the gear set in the SMB-00 actuator resulting in an increase in stroke time from approximately 18 to 30 seconds which will not affect any FSAR or TS stroke time limits.

## REASON FOR CHANGE

Limitorque technical update (LTU) 98-01 provided a more conservative methodology for determining the motor torque capability under reduced voltage conditions for MOVs. The Waterford 3 methodology for calculating torque capability for safety related gate and globe valves is not in agreement with the new Limitorque LTU 98-01 guidance. CR 98-0988, using the new guidance of LTU 98-01, determined that these safety related MOVs require changes in order to provide sufficient torque margins.

## 50.59 EVALUATION

The proposed changes will increase the output torque capability of the subject valve actuators, thus providing increased torque margin above the minimum design basis requirements and add assurance that the valves are capable of performing their safety functions. All of the proposed changes are outside of any system pressure boundary, and there are no new system interfaces created. Changes to MS-119B and MS-120B will increase their stroke times up to 7 seconds, which is within the required 10 seconds as described in FSAR Table 6.2-32. All other valves either do not have required stroke time limits associated with them or their stroke time is not affected by the change. All new motors have been seismically analyzed, and the new thermal overloads have been sized per calculation EC-E95401 to provide the required protection for the larger actuator motors. There will be no changes to the basic mechanical or electrical operation of the subject valve actuators due to implementation of this design change. The changes do not increase the probability of occurrence or consequences of accidents or malfunctions of equipment important to safety evaluated in the FSAR. There is no affect on any margins of safety as described in the basis for any Technical Specification. This change does not result in a USQ.

41. ER-W3-99-0661-00-00, Design Basis Temperature Increase for CCW Pump Rooms, CCW Heat Exchanger B, EFW Pump Rooms, Shutdown Heat Exchanger Rooms and Charging Pump Room A

DESCRIPTION OF CHANGE

Calculation 5-W determines the space temperatures in the various safety-related areas served by the Essential Chilled Water System (using the worst case 52 °F chilled water outlet temperature to the room coolers, which would occur during a design basis tornado). The calculation concludes that for a short duration after a tornado event, six sets of rooms will rise above their design basis temperature of 104 °F (75 °F for the Control Room). These rooms and their respective temperatures are: CCW Pump Rooms, 108 °F; CCW Heat Exchanger Rooms, 111 °F; Emergency Feedwater Pump Rooms, 115 °F; Shutdown Heat Exchanger Rooms, 113 °F; Control Room, 85 °F; and the Charging Pump Room A, 111 °F. The temperatures above the Design Ambient Temperature of 104 °F (75 °F for the Control Room) are not included and have never been included in the Design Basis or the Temperature Zone Maps.

REASON FOR CHANGE

Calculation 5-W was targeted in the Design Basis Review as a calculation that needed to be revised using the most current heat load inputs from other calculations. When 5-W was corrected, some of the room temperatures increased above the temperatures calculated in the previous revision. It should be noted, however, that many of the room temperatures calculated in Rev. 1 of Calculation 5-W were already above their design ambient temperature limit. ER-W3-99-0661-00-00 was written to address the elevated room temperatures of Calculation 5-W, and it used OPTIM, a verified software package. OPTIM is the software package that was provided by the manufacturer of the original cooling coils, and it is used to determine more accurately the room temperatures following the tornado event. The OPTIM runs show that the room temperatures will actually be lower than the temperatures calculated in 5W, Rev. 0, Change 1. (The calculation uses a more conservative Log Mean Temperature Difference method to calculate the room temperatures). ER-W3-99-0661-00-00 also addressed the Charging Pump B Room and the Control Room, which were not previously included in Calculation 5-W.

50.59 EVALUATION

The Ultimate Heat Sink Testing (Confirmatory Issue 2.4.5), which simulated the conditions during a tornado event, has previously been evaluated and addressed in Waterford 3 letter to the NRC, Letter #W3I82-0146. This letter was reviewed and accepted by the NRC per NUREG-0787, Supplement No. 5, Section 2.4.5, before Waterford 3 was licensed. The elevated room temperatures, which occur during a tornado event, have previously been evaluated to only exceed the 104 °F design ambient temperature limit for a short period of time. This short excursion has been analyzed to not have an effect on the equipment in the rooms or the ability of the

equipment to perform their safety function. The conclusion of the letter to the NRC and the NRC's acceptance response was that the event is not significant, is not a Significant Construction Deficiency per 10CFR50.55(e) and does not adversely impact the safety or the Environmental Qualification Program at Waterford 3. This was based on the fact that none of the rooms contain extreme temperature dependant equipment, and the temporary transient would not adversely affect the equipment. The rooms evaluated in ER-W3-99-0661-00-00 which were not included in this letter were the Control Room and the Charging Pump Room B. The elevated temperature of 85 °F in the Control Room is acceptable because the Control Room equipment was specified for a temperature range of 45 °F to 120 °F (per Tech Spec Bases 3/4.7.6.3) and is bounded by the short 120 °F temperature transient during the Station Black Out event. The elevated temperature in Charging Pump Room B will fall under the same justification as in the letter for Charging Pump Room A. The Charging Pump Room B, like the rooms evaluated in the letter, does not contain extreme temperature dependant equipment, so the temporary temperature transient will not adversely affect the equipment.

The temporary elevated temperatures will also not adversely affect the Controlled Ventilation Area System (CVAS) filter system. The CVAS is designed to provide filtration and iodine adsorption for air exhausted from the CVAS following a design basis accident, and it limits the post accident radiological releases below the guidelines of 10CFR100. The Shutdown Cooling Heat Exchangers rooms are the only areas of the identified rooms in the ER that flow into the CVAS filtration system. During a safe shutdown tornado, elevated doses are not expected as with other design basis accidents (e.g., LOCA). However, the iodine impregnated charcoal filters in the CVAS system are designed to remove iodines when operating at 70% relative humidity and 150 °F. The slightly elevated temperatures in the SDCHX rooms are below the 150 °F operating temperature of the charcoal, so the CVAS filter units will not be adversely affected.

Technical Specification Bases 3/4.7.6.3 addresses the Control Room Air Temperature. This Tech Spec states that even though 70 °F – 75 °F is the normal ambient temperature control band for the Control Room, it is too restrictive to be an LCO. The equipment in the Control Room was specified for a more general temperature range of 45 °F to 120 °F. The control Room temporary ambient temperature of 85 °F during the tornado event is well within the limits of 45 °F – 120 °F, so there is no reduction in margin of safety for the Tech Spec. None of the other areas with potential elevated temperatures have related Technical Specifications for ambient temperature, so there is no reduction in margin of safety for any of the rooms addressed. This change does not result in a USQ.

## F. COMMITMENT CHANGES

### 1. Response to Fire Protection Engineering Performance of Quarterly Fire Watch Evaluation

SUMMARY

This commitment is being closed. The fire protection program is non-safety, quality related and does not directly affect the operation of plant structures, systems or components. In this case the commitment was an enhancement of the administrative methods employed to ensure compensatory actions were carried out in accordance with the TRM. Plant personnel performance has demonstrated that enhanced observations are no longer necessary and that performance is adequate to ensure adherence with the approved fire protection program.

### 2. Dedication of Commercial Grade Items for Safety Related Applications

SUMMARY

Commitment text is being revised due to changes in processes. Parts Quality Determinations (PQDs) are no longer utilized to perform item safety classifications. Procedure UNT-007-021, "Spare Parts Equivalency/Parts Quality Level Determination" has been deleted and was replaced by NOECP-153, "Commercial Grade Item Dedication Evaluation." The applicable requirements from NOECP-153 for commercial grade items have been incorporated into DEAM procedure number PE-P-002-00, "Commercial Grade Item Evaluation." UNT-005-015 "Work Authorization Preparation and Implementation", identifies that for safety related applications, if the item is quality class L2 and the intended application is not listed in the MMIS S07 screen with an End Use authorization or if the item's quality class is L3 or L4, then a safety classification/commercial grade evaluation shall be performed. The Engineering Request Process in conjunction with DEAM procedure number PE-P-002-00 are in place to ensure that the correct commercial grade item per design is installed and received an engineering evaluation, if required, to ensure adequate dedication criteria was established for replacement items.

### 3. Spare Parts Equivalency Evaluations (SPEERS)

SUMMARY

The commitment text is being revised due to changes in processes. Inception of the Engineering Request Process replaces the SPEER process. SPEERS are now performed using the ER process. Existing plant procedures require initiation of an ER if adequate information is not available to specify material technical and quality requirements, including changes affecting design of permanent plant equipment.

4. Procedures for Procurement of Spare Parts for Safety Related Applications

SUMMARY

The Commitment text is being revised due to changes in processes. Procedures exist to ensure in-stock items with a quality level less than the quality classification of the major component are evaluated for suitability of use. For commercial grade items, the evaluation determines appropriate dedication criteria based on the item's safety function.

5. Controlling Field Changes for Design Change Packages

SUMMARY

These commitments are being closed. Document Revision Notices (DRNs) are no longer stand-alone documents and the requirements for the commitment no longer apply. DRNs are attachments to Design Changes, and are tracked, revised and approved by the Engineering Request Change (ERC) process. All approvals and rush changes are performed under the ERC process. The ERC process is currently addressed in procedures W4.104 and W4.105, or by the new Engineering Request Change Notice (ERCN) process addressed in DC-115, and DC-116. These mechanisms ensure changes to Design Packages are properly approved and issued in a timely manner.

6. RPCS to Remain Out of Service

SUMMARY

The cause of the problem was that a single relay failed in the trip position in one of the Feedwater Pump Trip (FWPT) AUX Cabinets. Reactor Power Cutback System (RPCS) requires two separate channels of a trip signal from each FWPT to generate reactor power cutback. A single channel trip from either FWPT causes a single trip alarm. The failed relay had both channels of inputs to RPCS. When the single relay failed in the trip position, it initiated a RPCS. PC-3447 separated the two channels of trip signal to two relays, TT-4 and TT-5. Now a single relay failure will only cause a trouble alarm. This condition no longer exists due to the implementation of PC-3447; therefore this commitment can be closed.

7. Minimizing Cooling Tower Overspill Design Change to be implemented during Refuel 10

SUMMARY

The Dry Cooling Tower (DCT) sumps were included as outfalls in the new LPDES permit issued on 2/1/99. The overspray from the Auxiliary Component Cooling Water (ACCW) Wet Cooling Towers (WCT) is included in the outfall description and characterization for Outfall 701 (DCTS #1) and Outfall 801 (DCTS #2). Therefore, the overspray from the WCTS is permitted and allowed in LPDES Permit LA0007374. There are currently no environmental requirements for performing the modification. WCT overspray is now allowed in the LPDES Permit LA0007374. There are no environmental requirements or reasons requiring this modification, therefore this commitment is deleted. Note that another commitment remains in effect and addresses implementing the modification to reduce/contain WCT overspray for the purposes of personnel safety and protection of electrical cabinets and other equipment.

8. Corrective Actions During Operations Phase

SUMMARY

This commitment is being closed based on the fact that Nuclear Facilities are required by 10CFR50 Appendix B to establish and maintain a corrective action program. The program is presently implemented by W2.501; however, this procedure will be replaced by EOI procedure LI-102. The program is audited regularly by the NRC and the site QA program requires an audit to be performed every two years. Therefore, this passive commitment is no longer required.

9. Reporting, Tracking, Correcting and Reinspecting Findings of Management Audits

SUMMARY

This commitment is being closed based on the fact that Nuclear Facilities are required by 10CFR50 Appendix B to establish and maintain a Corrective Action Program. The program is presently implemented by W2.501, however this procedure will be replaced by EOI procedure LI-102. The program is audited periodically by the NRC and the site QA program requires an audit to be performed every two years. The program requires that all personnel working at EOI facilities identify adverse conditions. Adverse conditions range from near misses to plant trips. Therefore, audit findings identified by outside or independent organizations would be reported, tracked, corrected and reinspected per the Corrective Action Program.

10. Deficiency Tracking via Audit Finding Report

SUMMARY

This commitment is being closed based on the fact that Waterford 3 is required by 10CFR50 Appendix B, applicable committed ANSI standards and EOI QAPM to establish and maintain a Corrective Action Program. The program is presently implemented by W2.501, however, this procedure will be replaced by EOI Corporate procedure LI-102, "Corrective Action Process". Audit Findings are tracked and controlled via the corrective action program. The code requirements are more than adequate commitments to ensure that audit findings are properly controlled; therefore an additional CMS commitment is not required.

11. Established Mechanisms which maintain plant procedures current – corrective action program

SUMMARY

The commitment text is being revised to incorporate new practices. It is the responsibility of all Waterford 3 personnel to identify and document conditions adverse to quality, industrial safety, and plant reliability. The Corrective Action Program is implemented by the site's Condition Report (CR) process. CRs are categorized by the Condition Review Group (CRG) as significant, non-significant or below the scope according to criteria established in the procedure. Significant CRs receive a root cause evaluation and Non-Significant CRs an apparent cause. Procedure content and compliance are part of the analysis to determine a root cause. Should inadequate procedures be identified, they are promptly changed or revised.

12. Condition Review Group (CRG)

SUMMARY

Number 2 of the commitment text has been revised. 2) The CRG is the management group responsible for review, classification, categorization and assignment of responsibilities for CRs. At the sites, the group is chaired by the station's General Manager Plant Operations (GMPO) or, if unavailable, a designee. The CRG chairman ensures that adequate representation is in attendance at meetings.

13. UNT-006-011 Revised to include prompts for identification of necessary interim actions

SUMMARY

This commitment is closed. It is our belief that the prompts for interim actions are no longer necessary. The sensitivity to ensure that appropriate immediate/interim corrective actions on new conditions are established is sustained by the CRG. They regularly demonstrate a good questioning attitude related to correcting a condition and its generic implications. Additionally, Corporate Procedure LI-102, "Corrective Action Process", requires that the individual identifying an adverse condition take and document in PCRS appropriate immediate actions.

14. QA will continue to verify Corrective Action Program effectiveness during the audit process

SUMMARY

Commitment text is being revised to read as follows: Quality Assurance will continue to verify Corrective Action Program effectiveness during the audit process. In addition, the function of performing effectiveness reviews will be as specified in the Root Cause Analysis (RCA) for significant CRs classified as Category A and B. The following provides justification for the change. The QA audit organization verifies the effectiveness of CR corrective action during the audit process per QAP-024. The Entergy Root Cause Desk Guide, which is used when performing root cause analysis, requires that corrective action plans provide actions(s) to measure effectiveness. Waterford 3 specifically requires the RCA for Category A and B CRs to include a section delineating effectiveness.

15. Annual Training for site root cause investigators to reinforce expectations on root cause evaluations

SUMMARY

This commitment is being closed. Site Corrective Action procedure W2.501, (to be replaced by Corporate procedure LI-102, Corrective Action Process), requires that root cause investigations be performed or reviewed by a qualified evaluator. The Corrective Action & Assessment group as required in the FSAR and these procedures maintains a list of these evaluators. Presently WF3 is the only EOI nuclear site that requires annual requalification /refresher training. The goal of EOI is to standardize processes as exemplified by the issuance of LI-102. An EOI Natural Work Team exists for the root cause process and the topic of Computer Based Training to conduct refresher training is still available and may be scheduled as deemed necessary by the CRG/CARB or CA&A group. The emphasis for providing and maintaining quality root cause evaluations is not lessened by the closure of this commitment.

16. Design Change Implementation and Closeout procedure revised to include verification of repetitive tasks having been implemented as specified in the design change.

SUMMARY

The commitment text is being revised to reflect the new process for verifying Repetitive Task (RT) implementation. Procedure W4.105 requires the departments responsible for RTs to be notified of the ER implementation completion and accept responsibility for processing updates to the affected RT in a time frame that supports the plant needs. Procedure DC-115 which will replace W4.104 and W4.105 when implemented, will require activities associated with a modification such as PM task development be identified, entered into the Engineering Request Database (ERD) and accepted by the responsible department prior to ER approval. The ER can not be closed in ERD until all associated activities have been completed.

17. Completion of EQ Data Record Form

SUMMARY

The commitment specifies that "these procedures require that the EQ data record form from procedure MD-001-020 be completed when any maintenance is performed on EQ equipment. The EQ data record forms identify the EQ requirements for each piece of equipment."

Procedure MD-001-020 was deleted. The EQ Data Record Form and all maintenance controls are now located in UNT-001-015. Remove all references to MD-001-020, and replace accordingly with UNT-001-015.

The commitment is revised to read: "...these procedures require that EQ data record form from UNT-001-015 be completed when any maintenance which is necessary for maintaining EQ status is performed on EQ equipment. The EQ data record forms identify the EQ requirements for each piece of equipment, should requirements exist." All maintenance is controlled in accordance with UNT-001-015.

18. Results of ABB/CE Evaluation on safety significance of log power instrument event will be submitted in a supplement to LER 96-003

SUMMARY

LER 96-003-00 - The original decision to revise the LER was based on providing additional information that may have been derived from the Combustion Engineering (CE) evaluation (CE NPSD-1052-P) conclusions reached. A review of the evaluation revealed that the conclusion essentially restated the predicted outcome, which had already been reported in Rev. 0. At this time nothing would be gained by issuing the LER revision. This also did not necessitate a new LER.

19. Continuous running CCW Makeup Pump may affect CSP inventory – revise LER 97-026

SUMMARY

The original decision to revise the LER was based on providing additional information that may have been derived from the root cause analysis (CR 97-2551). A review of the RCA indicated that although not in specific detail, the root cause stated in the original LER did not change. Therefore it is not necessary to revise the LER at this time. The details are available onsite in the CR files. This also did not necessitate a new LER.

20. Perform periodic follow-up inspections within the feedpump control cabinet to verify sealed conduits have eliminated the condensation intrusion

SUMMARY

In accordance with RCA CR 98-0947, dated 8/27/98, the periodic inspections were to be performed following the approval of the RCA and completed by 10/31/98. The periodic inspections were in fact performed weekly and considered sufficient time to verify that the sealed conduits have eliminated the water intrusion concerns. The intent of this commitment was not to have an on-going inspection task, but to perform this inspection for a sufficient period of time to have a high level of confidence that the sealed conduits eliminated the water intrusion concerns. Commitment text is revised to read "I&C Maintenance shall perform periodic follow-up inspections within the feedpump control cabinet to verify sealed conduits have eliminated the condensation intrusion. These periodic inspections will be completed by 10/31/98, if no abnormal conditions are found.

21. Standing Instruction 95-13 which is associated with the Ultimate Heat Sink (UHS) minimum fan requirements

SUMMARY

Standing Instruction 95-13 was revised to ensure that TS 3.7.4 compliance will be maintained under all circumstances. The information in revised Standing Instruction 95-13 has subsequently been incorporated into procedure OP-100-014, "Technical Specification Compliance" and Standing Instruction 95-13 was cancelled. The change to OP-100-014 was an interim measure until TS 3.7.4 could be changed. This passive commitment can be closed due to the approval and issuance of Amendment No. 123 by the NRC which revised TS Section 3.7.4. These requirements are no longer needed as a standing instruction or in OP-100-014.

22. NOECP-102 will be revised to require notification of the lead maintenance planner each time the component database is changed.

SUMMARY

NOECP-102 will require notification to planning supervision whenever a change is made to the safety classification / Q-List fields of a component in the component database. A corresponding change will be reflected in UNT-005-012 to require a change to the maintenance database and outstanding work authorizations whenever the lead maintenance planner is notified of a component database change.

23. Project Files Storage and Maintenance of Uncontrolled Records

SUMMARY

Commitment text is being revised because new processes which control the record storage process, have been implemented. Administrative Services – Records Management (Records Center) / Document Control is the focal point for storage and maintenance of uncontrolled records and documents. The filing system used is a computerized document retrieval system. Completed records forwarded to records management are indexed on the computer then scanned as images and optically stored. These records are thus effectively filed under document number, record type, date, title, vendor equipment number, etc., allowing a user to retrieve documents in a timely manner.

## 24. Records Quality Review

SUMMARY

Commitment text was revised due to new processes, which control the record storage process, having been implemented. Quality related document design change packages are reviewed by the QA group before final closure and transmittal to records management. A Quality Reviewer (QR) completes a QA review checklist on the DCP to ensure that records establishing proper review and other necessary records are retained. The QR review scope ensures that documents required by the DCP index and controlling procedures are included, proper review and approval is indicated on the records, applicable codes and quality standards are identified, test and inspection requirements are documented and safety evaluation and design verification is performed.

## III. PROCEDURE CHANGES
## A. PLANT PROCEDURES

1. W2.302, 10CFR50.59 Review Program (Deletion) and Revision of FSAR Chapter 13

DESCRIPTION OF CHANGE

Site procedure W2.302 is being deleted and replaced by corporate management procedure LI-101, "10CFR50.59 Review Program" and FSAR Chapter 13 is being revised to address implementation of corporate procedures at the Waterford 3 site.

REASON FOR CHANGE

The corporate procedure was developed to implement one system-wide process for 50.59 reviews and to implement the guidance of NEI 96-07, "Guidelines for 10CFR50.59 Evaluations". The Waterford 3 FSAR does not currently recognize implementation of corporate procedures at the site. The language of FSAR Chapter 13, while it does use the term "corporate support entities", implies that procedures used at Waterford 3 are initiated, prepared, and controlled by cognizant supervisors. The W3 Quality Assurance Program Manual (QAPM) states that directors and managers are responsible for the development and approval of safety related procedures and instructions which affect activities within their area of responsibility and ensuring that W3 commitments and obligations are adequately addressed. Thus, the QAPM does not explicitly recognize corporate procedures. However, the EOI single QA program will replace the individual site QA programs and it does recognize corporate procedures. The new single QA program will be implemented before LI-101 is effective on 7/1/99.

50.59 EVALUATION

The deletion of site procedure W2.302 and the revision of FSAR section 13.5 do not create a USQ. The first is an administrative change that deletes one site procedure in favor of a corporate one - the process for 50.59 reviews will remain essentially the same. The FSAR change is being made to allow implementation of corporate procedures at the Waterford 3 site. No changes are being made to the plant that could adversely affect the probability of either an accident or a malfunction of equipment important-to-safety. No physical changes are being made that could create either a new accident or a new equipment failure mode. No protective boundary is affected by this change and no margin of safety as defined in the basis of any TS is reduced by this change.

2. UNT-005-013-8, Fire Protection Program

DESCRIPTION OF CHANGE

Revision of the Fire Protection Program to incorporate changes in organization structure, functional responsibilities, and titles as a result of Entergy Renewal. Incorporated updated procedure references and changes related to Work Management System (WMS) and the Engineering Request (ER) process.

REASON FOR CHANGE

Implementation of Entergy Renewal and procedure/process changes related to WMS.

50.59 EVALUATION

These are administrative changes to the Fire Protection Program. They have no affect on any SSC or on operation of the plant. They do not create a USQ or reduce the margin of safety.

3. CE-003-327, Operation of the Primary Sample Panel

DESCRIPTION OF CHANGE

This revision reformats the procedure to current site requirements. It also adds section 10.19 which provides a means of supplying temporary cooling water to the Reactor Coolant System (RCS) Hot Leg sample cooler.

REASON FOR CHANGE

The temporary cooling water supply is necessary to support RCS Hot Leg samples as requested during conditions when the normal non-safety related Component Cooling Water (CCW) system supply to the Primary Sample Panel is isolated due to plant conditions.

50.59 EVALUATION

The proposed change does not represent or create a USQ. The proposed change will provide a temporary cooling water supply to the RCS Hot Leg sample cooler. Implementation of the proposed change affects an isolated non-safety related component. The proposed change is intended to be used only when the normal CCW supply to the primary sample has been isolated due to an Engineered Safety Feature Actuation Signal (ESFAS) actuation and cannot be restored and an RCS sample has been requested.

## 4. UNT-005-012, Repetitive Task Program, Rev. 6

### DESCRIPTION OF CHANGE

UNT-005-012 and the FSAR are being changed to reference a new risk-informed process for designating a task as Mandatory Preventative Maintenance (MPM) or Preventative Maintenance (PM). This process will be used to re-baseline the current MPM and PM tasks for appropriate classification, and to designate new, future tasks. Risk significance will also be added as a consideration when deleting PM tasks, or extending their maintenance intervals. The technical principles that will be used for designating a task as a PM or MPM are as follows: 1) The MPM designation is required for tasks on systems that are in the Maintenance Rule (MR) scope and that impact risk significant functions. 2) The MPM designation may be used for tasks on SSCs whose failure could result in a reactor trip, personnel injury, or adverse economic impact. 3) The PM designation will be established for maintenance tasks on low risk significant MR systems or on systems not scoped in the MR. 4) Tasks need not automatically continue to be classified as MPMs or continue to be performed because there is a Commitments Management System (CMS) commitment or regulatory document like the FSAR or Licensee Event Report (LER) linked to the task. Engineering Request ER-W3-99-1103-00-00 "Study for Implementing Risk Informed Preventative Maintenance at Waterford 3" provides the regulatory justification for re-classifying a MPM task as a PM task, or deleting a PM task, provided the risk-informed criteria and process, and other technical factors are applied. For tasks that will be changed and are linked to a CMS commitment, CMS should be updated prior to the change to ensure that NRC legal obligations continue to be met. 5) Documentation on the justification for all tasks that are downgraded or deleted will be written in the Work Management System (WMS) task screen.

### REASON FOR CHANGE

The change will reduce the burden of performing and maintaining MPM tasks that do not have risk significance. This change will require mandatory preventative maintenance designations for structures, systems, and components that have risk significance, based on the Maintenance Rule criteria. During this process, a review of current PMs will be performed to upgrade any risk significant PMs to MPMs, based on the new risk informed process. The use of risk significance as a consideration for interval extension or task deletion will also ensure resources are focused on tasks that provide the greatest impact on safe operation.

### 50.59 EVALUATION

The implementation of the new risk-informed process is in accordance with and consistent with the regulatory criteria for the maintenance program at Waterford 3. The program will ensure that quality is commensurate with the design bases, that safety is not compromised, and that license limits are not violated. The program will ensure that maintenance is performed on risk significant systems, components, and structures. The program will not affect nor reconstitute the actual maintenance performed for systems or components. The program will continue to be an integral

part of and work in tandem with other maintenance elements, such as predictive, diagnostic, or corrective maintenance. The program will not affect the maintenance required by Technical Specifications, ISI/IST program, or 10CFR50.49. The safe operation of the plant is not affected by the application of the new risk informed process and criteria. This change does not entail a physical change to structures, systems, or components; an analytical change; or a change to the operation of the plant. This change does not involve an unreviewed safety question.

## 5. PE-005-040, Diagnostic Pressure Testing of Motor Operated Valves

### DESCRIPTION OF CHANGE

The proposed change provides a means of testing valve SI-120A during Plant Modes 1, 2 or 3. The change utilizes as its basis the previously approved High Pressure Safety Injection (HPSI) pump In-Service Testing (IST) alignment. This allows the valve's testing to be conducted in conjunction with (or separately from) the conduct of the normally scheduled HPSI IST testing. The changes also reformat the procedure to comply with W2.110 Procedure Format Standards.

### REASON FOR CHANGE

Inclusion of the ability to test valve SI-120A during the higher operational modes (Modes 1, 2 or 3) allows greater flexibility for the planning and scheduling of the test, and removes the restriction of performing the test exclusively during plant outages. Performance of the test in conjunction with the HPSI IST will simplify the test, and does not require the rigorous test parameters associated with injection into the Reactor Coolant system. This facilitates the plant's compliance with the Joint Owner's Group (JOG) commitment to the Periodic Verification Program and Generic Letter 96-05.

### 50.59 EVALUATION

The proposed change is based upon the currently approved HPSI pump in-service test, and differs in that valve SI-120A is stroked to collect differential pressure test data. The affected plant equipment utilized in the conduct of the test is functionally used to mitigate the consequences of many of the accidents in the FSAR. Operation of the equipment as described in the procedure will not be outside the equipment's design capacity or in a manner inconsistent with the system's intended function. No change to plant operation, design basis or physical change to any plant components is being made. The equipment alignment as proposed in the change is supported by plant design and safety analysis, and is not inconsistent with those procedures already in use for previously approved tests. No unreviewed safety questions exist.