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Waterford 3

W3F1-2014-0067

November 05, 2014

U.S. Nuclear Regulatory Commission
Attn: Document Control Desk
Washington, DC 20555-0001

Subject: Technical Specification Index and Bases Update to the NRC for the Period June 30, 2013 through November 3, 2014
Waterford Steam Electric Station, Unit 3 (Waterford 3)
Docket No. 50-382
License No. NPF-38

Dear Sir or Madam:

Pursuant to Waterford Steam Electric Station Unit 3 (Waterford 3) Technical Specification (TS) 6.16, Entergy Operations, Inc. (EOI) hereby submits an update of all changes made to the Waterford 3 Technical Specification Index and Bases since the last submittal per letter W3F1-2013-0040 (ADAMS Accession No. ML13190A391), dated July 9, 2013. This update satisfies the submittal frequency required by TS 6.16, which indicates that the submittal will be made at a frequency consistent with 10 CFR 50.71(e) and exemptions thereto.

There are no commitments associated with this submittal. Should you have any questions or comments concerning this submittal, please contact the Regulatory Assurance Manager, John Jarrell, at (504) 739-6685.

Sincerely,

JPJ/JRM

Attachments:

1. Waterford 3 Technical Specification Index and Bases Change List

2. Waterford 3 Technical Specification Index and Bases Revised Pages

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Attachment 1 to

W3F1-2014-0067

Waterford 3 Technical Specification Index and Bases Change List

Waterford 3 Technical Specification (TS) Index and Bases Change List

T.S. Bases Change No.	Implementation Date	Affected TS Bases or Index Pages	Topic of Change
77	3/31/14	B 3/4 4-3a B 3/4 4-3b B 3/4 4-3c B 3/4 4-3d	Change No. 77 to TS Bases sections 3/4.4.4.4 was implemented by Licensing Basis Document Change Request (LBDCR) 14-006 to reflect the steam generators replacement.
78	4/10/14	B 3/4 3-1 B 3/4 3-1a1 B 3/4 3-1a2	Change No. 78 to TS Bases section 3/4.3.1 and 3/4.3.2 was implemented by Licensing Basis Document Change Request (LBDCR) 14-003 to add wording to clarify what is meant by the note “limited plant cooldown or boron dilution is allowed provided the change is accounted for in the calculated SHUTDOWN MARGIN”
79	5/13/14	B 3/4 8-2	Change No. 78 to TS Bases section 3/4.8.1, 3/4.8.2, 3/4.8.3 was implemented by Licensing Basis Document Change Request (LBDCR) 14-010 to add Regulatory Guide 1.9 Revision 4 (2007) and IEEE Standard 387-1995.
80	6/4/14	B 3/4 8-1a B 3/4 8-1a1	Change No. 80 to the TS Bases section 3/4.8.1, 3/4.8.2, 3/4.8.3 was implemented by LBDCR 13-017 to provide explanation of purpose for Surveillance Requirement (SR) 4.8.1.1.2e.10, Emergency Diesel Generator fuel oil transfer pump cross connect surveillance.

Attachment 2 to

W3F1-2014-0067

Waterford 3 Technical Specification Index and Bases Revised Pages

(There are 10 unnumbered pages following this cover page)

For accidents that do not involve fuel damage, the primary coolant activity level is assumed to be equal to the LCO 3.4.7 *RCS Specific Activity* limits. For accidents that assume fuel damage, the primary coolant activity is a function of the amount of activity released from the damaged fuel. The dose consequences of these events are within the limits of GDC 19 (Reference 2) and 10 CFR 50.67 (Reference 3). Steam generator tube integrity satisfies Criterion 2 of 10 CFR 50.36(c)(2)(ii).

Limiting Condition for Operation

→(DRN 06-997, Ch. 49; LBDCR-14-006, Ch. 77)

The LCO requires that SG tube integrity be maintained. The LCO also requires that all SG tubes that satisfy the plugging criteria be plugged in accordance with the *Steam Generator Program*. During a SG inspection, any inspected tube that satisfies the *Steam Generator Program* plugging criteria is removed from service by plugging. If a tube was determined to satisfy the plugging criteria but was not plugged, the tube may still have tube integrity. In the context of this Specification, a SG tube is defined as the entire length of the tube, including the tube wall between the tube-to-tubesheet weld at the tube inlet and the tube-to-tubesheet weld at the tube outlet. The tube-to-tubesheet weld is not considered part of the tube.

←(DRN 06-997, Ch. 49; LBDCR-14-006, Ch. 77)

A SG tube has tube integrity when it satisfies the SG performance criteria. The SG performance criteria are defined in Specification 6.5.9, *Steam Generator Program*, and describe acceptable SG tube performance. The Steam Generator Program also provides the evaluation process for determining conformance with the SG performance criteria.

There are three SG performance criteria: structural integrity, accident induced leakage, and operational leakage. Failure to meet any one of these criteria is considered failure to meet the LCO.

- The structural integrity performance criterion provides a margin of safety against tube burst or collapse under normal and accident conditions, and ensures structural integrity of the SG tubes under all anticipated transients included in the design specification. Tube burst is defined as, “The gross structural failure of the tube wall. The condition typically corresponds to an unstable opening displacement (e.g., opening area increased in response to constant pressure) accompanied by ductile (plastic) tearing of the tube material at the ends of the degradation.” Tube collapse is defined as, “For the load displacement curve for a given structure, collapse occurs at the top of the load versus displacement curve where the slope of the curve becomes zero.” The structural integrity performance criterion provides guidance on assessing loads that significantly affect burst or collapse. In that context, the term “significantly” is defined as “An accident loading condition other than differential pressure is considered significant when the addition of such loads in the assessment of the structural integrity performance criterion could cause a lower structural limit or limiting burst/collapse condition to be established.” For tube integrity evaluations, except for circumferential degradation, axial thermal loads are classified as secondary loads. For circumferential degradation, the classification of axial thermal loads as primary or secondary loads will be evaluated on a case-by-case basis. The division between primary and secondary classifications will be based on detailed analysis and/or testing.

  Structural integrity requires that the primary membrane stress intensity in a tube not exceed the yield strength for all ASME Code, Section III, Service Level A (normal operating conditions) and Service Level B (upset or abnormal conditions) transients included in the design specification. This includes safety factors and applicable design basis loads based on ASME Code, Section III, Subsection NB (Reference 4) and Draft Regulatory Guide 1.121 (Reference 5).

- The accident induced leakage performance criterion ensures that the primary to secondary leakage caused by a design basis accident, other than a SGTR, is within the accident analysis assumptions. The accident analysis assumes that accident induced leakage does not exceed 540 gpd through any one SG. The accident induced leakage rate includes any primary to secondary leakage existing prior to the accident in addition to primary to secondary leakage induced during the accident.

- The operational leakage performance criterion provides an observable indication of SG tube conditions during plant operation. The limit on operational leakage is contained in LCO 3.4.5.2, *Reactor Coolant System Operational Leakage*, and limits primary to secondary leakage through any one SG to $\leq$ 75 gallons per day. This limit is based on assumptions in radiological analyses. This limit is less than the 150 gallons per day through any one SG limit of NEI 97-06, which assumes that a single crack leaking this amount would not propagate to a SGTR under the stress conditions of a LOCA or a Main Steam Line Break. If this amount of leakage is due to more than one crack, the cracks are very small, and the above assumption is conservative.

Actions

The ACTIONS are modified by a Note clarifying that the ACTIONS may be entered independently for each SG tube. This is acceptable because the ACTIONS provide appropriate compensatory actions for each affected SG tube. Complying with the ACTIONS may allow for continued operations, and subsequent affected SG tubes are governed by subsequent application of associated ACTIONS.

→(LBDCR-14-006, Ch. 77)
ACTION "a" applies if it is discovered that one or more SG tubes examined in an inservice inspection satisfy the tube plugging criteria but were not plugged in accordance with the *Steam Generator Program* as required by SR 4.4.4.2. An evaluation of SG tube integrity of the affected tube(s) must be made. Steam generator tube integrity is based on meeting the SG performance criteria described in the *Steam Generator Program*. The SG plugging criteria define limits on SG tube degradation that allow for flaw growth between inspections while still providing assurance that the SG performance criteria will continue to be met. In order to determine if a SG tube that should have been plugged has tube integrity, an evaluation must be completed that demonstrates that the SG performance criteria will continue to be met until the next refueling outage or SG tube inspection. The tube integrity determination is based on the estimated condition of the tube at the time the situation is discovered and the estimated growth of the degradation prior to the next SG tube inspection. If it is determined that tube integrity is not being maintained, ACTION "b" applies.
←(LBDCR-14-006, Ch. 77)

An allowed outage time of 7 days is sufficient to complete the evaluation while minimizing the risk of plant operation with a SG tube that may not have tube integrity. If the evaluation determines that the affected tube(s) have tube integrity, ACTION "a.2" allows plant operation to continue until the next refueling outage or SG inspection provided the inspection interval continues to be supported by an operational assessment that reflects the affected tubes. However, the affected tube(s) must be plugged prior to entering HOT SHUTDOWN following the next refueling outage or SG inspection. This time period is acceptable since operation until the next inspection is supported by the operational assessment.

ACTION "b" applies if the ACTIONS and associated allowed outage time of ACTION "a" are not met or if SG tube integrity is not being maintained, the reactor must be brought to HOT STANDBY within the next 6 hours and COLD SHUTDOWN within the following 30 hours. The allowed outage times are reasonable, based on operating experience, to reach the desired plant conditions from full power conditions in an orderly manner and without challenging plant systems.

Surveillance Requirements

During shutdown periods the SGs are inspected as required by SR 4.4.4.1 and the Steam Generator Program. NEI 97-06, *Steam Generator Program Guidelines* (Reference 1), and its referenced EPRI Guidelines, establish the content of the *Steam Generator Program*. Use of the *Steam Generator Program* ensures that the inspection is appropriate and consistent with accepted industry practices.

During SG inspections a condition monitoring assessment of the SG tubes is performed. The condition monitoring assessment determines the "as found" condition of the SG tubes. The purpose of the condition monitoring assessment is to ensure that the SG performance criteria have been met for the previous operating period.

→(LBDCR-14-006, Ch. 77)
The *Steam Generator Program* determines the scope of the inspection and the methods used to determine whether the tubes contain flaws satisfying the tube plugging criteria. Inspection scope (i.e., which tubes or areas of tubing within the SG are to be inspected) is a function of existing and potential degradation locations. The *Steam Generator Program* also specifies the inspection methods to be used to find potential degradation. Inspection methods are a function of degradation morphology, non-destructive examination (NDE) technique capabilities, and inspection locations.

The *Steam Generator Program* defines the frequency of SR 4.4.4.1. The frequency is determined by the operational assessment and other limits in the SG examination guidelines (Reference 6). The *Steam Generator Program* uses information on existing degradations and growth rates to determine an inspection frequency that provides reasonable assurance that the tubing will meet the SG performance criteria at the next scheduled inspection. In addition, Specification 6.5.9 contains prescriptive requirements concerning inspection intervals to provide added assurance that the SG performance criteria will be met between scheduled inspections. If crack indications are found in any SG tube, the maximum inspection interval for all affected and potentially affected SGs is restricted by Specification 6.5.9 until subsequent inspections support extending the inspection interval.

As required by SR 4.4.4.2, any inspected tube that satisfies the *Steam Generator Program* plugging criteria is removed from service by plugging. The tube plugging criteria delineated in Specification 6.5.9 are intended to ensure that tubes accepted for continued service satisfy the SG performance criteria with allowance for error in the flaw size measurement and for future flaw growth. In addition, the tube plugging criteria, in conjunction with other elements of the *Steam Generator Program*, ensure that the SG performance criteria will continue to be met until the next inspection of the subject tube(s). Reference 1 provides guidance for performing operational assessments to verify that the tubes remaining in service will continue to meet the SG performance criteria.
←(LBDCR-14-006, Ch. 77)

→(LBDCR-14-006, Ch. 77)

The frequency of prior to entering HOT SHUTDOWN following a SG inspection ensures that the Surveillance has been completed and all tubes meeting the plugging criteria are plugged prior to subjecting the SG tubes to significant primary to secondary pressure differential.

A structural integrity analysis for the Replacement Steam Generators (RSGs) was performed, in part, to confirm the adequacy of the tube repair criteria (plugging criteria). The guidance of NRC Regulatory Guide 1.121, "Bases for Plugging Degraded PWR Steam Generator Tubes," August 1976, was used in performing this analysis, which accounts for flaw growth between inspections and eddy current measurement uncertainty. In this analysis, a 15% through-wall allowance was used to account for flaw growth and measurement uncertainty (from 40% to 55%). To maintain this allowance of 15 percent between the plugging criteria (40%) and the tube's structural limit (55%) under all design basis conditions (normal, transient, and design basis accidents), maximum reactor coolant system (RCS) pressure during normal heatup and cooldown is 2000 psia below an RCS temperature of 450°F and 2250 psia between a temperature of 450°F and 470°F.

←(LBDCR-14-006, Ch. 77)

REFERENCES

1. NEI 97-06, *Steam Generator Program Guidelines*.

2. 10 CFR 50 Appendix A, GDC 19.

3. 10 CFR 50.67.

4. ASME Boiler and Pressure Vessel Code, Section III, Subsection NB.

5. Draft Regulatory Guide 1.121, *Basis for Plugging Degraded Steam Generator Tubes*, August 1976.

6. EPRI, *Pressurized Water Reactor Steam Generator Examination Guidelines*.

←(DRN 06-916, Ch. 48)

→(DRN 06-997, Ch. 49)

→(LBDCR-14-006, Ch. 77)

←(LBDCR-14-006, Ch. 77)

3/4.3.1 and 3/4.3.2 REACTOR PROTECTIVE AND ENGINEERED SAFETY FEATURES ACTUATION SYSTEMS INSTRUMENTATION

The OPERABILITY of the Reactor Protective and Engineered Safety Features Actuation Systems instrumentation and bypasses ensures that (1) the associated Engineered Safety Features Actuation action and/or reactor trip will be initiated when the parameter monitored by each channel or combination thereof reaches its setpoint, (2) the specified coincidence logic is maintained, (3) sufficient redundancy is maintained to permit a channel to be out of service for testing or maintenance, and (4) sufficient system functional capability is available from diverse parameters.

The OPERABILITY of these systems is required to provide the overall reliability, redundancy, and diversity assumed available in the facility design for the protection and mitigation of accident and transient conditions. The integrated operation of each of these systems is consistent with the assumptions used in the safety analyses.

The redundancy design of the Control Element Assembly Calculators (CEAC) provides reactor protection in the event one or both CEACs become inoperable. If one CEAC is in test or inoperable, verification of CEA position is performed at least every 4 hours. If the second CEAC fails, the CPCs will use DNBR and LPD penalty factors to restrict reactor operation to some maximum fraction of RATED THERMAL POWER. If this maximum fraction is exceeded, a reactor trip will occur.

➜(LBDCR-14-003 Ch.78)

Table 3.3-1 ACTION 4 requires the suspension of all operations involving positive reactivity changes with the number of channels OPERABLE one less than required by the Minimum Channels OPERABLE requirement. With one of the two required minimum operable channels inoperable, it may not be possible to perform a CHANNEL CHECK to verify the sole remaining required channel is OPERABLE. Therefore, with one or more required channels inoperable, the logarithmic power monitoring function cannot be reliably performed. Consequently, the Required Actions are the same for one required channel inoperable or more than one required channel inoperable.

The (*) for ACTION 4 was added to allow small positive reactivity additions (i.e. temperature or boron fluctuations) necessary to maintain plant conditions. These activities may result in addition to the RCS of water at a temperature different than that of the RCS, may result in slight RCS temperature changes, and may require inventory makeup from sources that are at boron concentrations less than RCS concentration. Depending on core loading and time in core life, raising temperature may add positive reactivity and should be minimized when possible. This allowance is intended to give Operations flexibility to perform actions required to maintain plant conditions but should not be utilized to significantly change plant conditions.

🡐(LBDCR-14-003 Ch.78)

3/4.3.1 and 3/4.3.2 REACTOR PROTECTIVE AND ENGINEERED SAFETY FEATURE SAFETY ACTUATION SYSTEMS INSTRUMENTATION (Continued)

→(EC-17731, Ch. 63)

Note 2 of Table 4.3-1 provides requirements for the periodic calibration of CPC power indications using calorimetric power as the calibration standard.

No calibration of CPC power indications are required at less than 15% RATED THERMAL POWER since inherent conservatisms in the CPC calculations at these power levels compensate for any potential decalibration. Significant differences between CPC power indications and calorimetric power observed during surveillances should always be investigated to determine the cause of the deviation. The most accurate calorimetric power indication available at the time of calibration should be used.

Between 15% and 80% power, if the daily surveillance finds that a CPC power indication is greater than the calorimetric power indication by more than 10% RTP, it should be adjusted to be within 8.0% and 10.0% RTP above the calorimetric. If the CPC power indications have been calibrated properly to the calorimetric power indication at high power (meaning 80% or above), then the most appropriate thing to do is not calibrate CPC powers below 80% power if they are conservative relative to calorimetric. In the extremely unlikely event that a CPC power indication is found to be more than 10.0% RTP higher than the calorimetric, it should be adjusted as little as possible to meet the requirements of the Technical Specifications. If this situation were to occur, it is likely that there is an anomaly in the calibration data or instrumentation. The safety and setpoint analysis does not explicitly address this situation because it is an unreasonable scenario without some other anomaly in the measurements, calibration or instrumentation. The

←(EC-17731, Ch. 63)

3/4.3.1 and 3/4.3.2 REACTOR PROTECTIVE AND ENGINEERED SAFETY FEATURE SAFETY ACTUATION SYSTEMS INSTRUMENTATION (Continued)

➔(EC-17731, Ch. 63)
probability of being greater than +10.0% from calibration following a power reduction from a calibrated condition, recalibrating to between +8.0% and +10.0% and then having a power increasing event which requires a CPC trip and having CPC be non-conservative at the point the trip is needed is too low to consider it as being within the CPC design basis.

At or above 80% RATED THERMAL POWER, the Note 2 phrase "as close as practical to calorimetric power" implies that the as-left difference between the affected CPC power indication and calorimetric power should be as near to 0% RATED THERMAL POWER as possible.

CPCs use the addressable constant PCALIB to determine power dependent biases for use in its calculations. Thus, when calibrations of CPC power indications are performed, it may be necessary to adjust the CPC constant PCALIB as described below:

- While operating below 80% RATED THERMAL POWER, whenever the calibration of either CPC neutron flux power or CPC △T power is adjusted, PCALIB must be set equal to the lower of the power level (in % RATED THERMAL POWER) of that adjustment and the power level (in % RATED THERMAL POWER) of the most recent calibration adjustment (or verification) of the other power indication (the one not being calibrated).

- PCALIB can be set to the current power level (in % RATED THERMAL POWER) whenever both CPC neutron flux power and CPC △T power are adjusted or verified to be within the Technical Specification requirements at that power level.

- PCALIB can be set to 100.0 whenever both CPC neutron flux power and CPC △T power have been adjusted or verified to be within the Technical Specification requirements at or above 80% RATED THERMAL POWER (plus uncertainty).

- PCALIB must be set to 20.0 prior to initi al power ascension following refueling.

🡐(EC-17731, Ch. 63)

Table 3.3-3 ACTION 19 allows for continued operation in the applicable MODE(S) with one of the Refueling Water Storage Pool (RWSP) - Low or Steam Generator △P Emergency Feedwater Actuation Signal (EFAS) channels inoperable provided the channel is placed in the bypass or tripped condition within 1 hour. If an inoperable channel of the RWSP - Low or Steam Generator △P EFAS channel is required to be placed in the tripped condition within one hour, then within 48 hours the channel must either be restored to OPERABLE status or be placed in the bypassed condition. The bypassed channel must be restored to OPERABLE status prior to entering the applicable MODE(S) following the next MODE 5 entry. With one of the RWSP - Low or Steam Generator △P (EFAS) channels inoperable and in bypass, and a failure occurs or repairs are necessary on one of the remaining channels, ACTION 20 must be entered.

A.C. SOURCES, D.C. SOURCES, AND ONSITE POWER DISTRIBUTION SYSTEMS (Continued)

→(EC 47119, Ch 79)

The Surveillance Requirements for demonstrating the OPERABILITY of the diesel generators are consistent with the recommendations of Regulatory Guides 1.9 "Selection of Diesel Generator Set Capacity for Standby Power Supplies," Revision 4, March 2007, and 1.137, "Fuel Oil Systems for Standby Diesel Generators," Revision 1, October 1979. Other provisions are derived from Generic Letter 93-05 "Line-Item Technical Specifications Improvements to Reduce Surveillance Requirements for Testing During Power Operation" 94-01 "Removal of Accelerated Testing and Special Reporting Requirements for Emergency Diesel Generators," and NUREG 1432 Standard Technical Specifications Combustion Engineering Plants.

←(EC 47119, Ch 79)

The minimum voltage and frequency stated in the Surveillance Requirement are those necessary to ensure the diesel generator can accept the Design Basis Accident loading while maintaining acceptable voltage and frequency levels. Stable operation at the nominal voltage and frequency values is also essential to establishing diesel generator OPERABILITY, but a time constraint is not imposed. This is because a typical diesel generator will experience a period of voltage and frequency oscillations prior to reaching steady state operation if these oscillations are not dampened out by load application. This period may extend beyond the 10 second acceptance criteria and could be a cause for failing the Surveillance Requirement. In lieu of a time constraint in the Surveillance Requirement, the actual time to reach steady state operation is monitored and trended. This is to ensure there is no voltage regulator or governor degradation which could cause a diesel generator to become inoperable. The 10 seconds in the Surveillance Requirement is met when the diesel generator first reaches the specified voltage and frequency, at which time the output breaker would close if an automatic actuation had occurred.

→(DRN 02-0607)
←(DRN 02-0607)

The maximum voltage limit in Surveillance test 4.8.1.1.2.e.2 was increased to 5023 volts in response to NRC Information Notice 91-13; Inadequate Testing of Emergency Diesel Generators. A maximum voltage limit is provided to ensure that components electrically connected to the diesel generator are not damaged as a result of the momentary voltage excursion experienced during this test.

The Surveillance Requirement for demonstrating the OPERABILITY of the station batteries are based on the recommendations of Regulatory Guide 1.129, "Maintenance Testing and Replacement of Large Lead Storage Batteries for Nuclear Power Plants," February 1978, and IEEE Std 450-1980, "IEEE Recommended Practice for Maintenance, Testing, and Replacement of Large Lead Storage Batteries for Generating Stations and Substations."

Verifying average electrolyte temperature above the minimum for which the battery was sized, total battery terminal voltage on float charge, connection resistance values and the performance of battery service and discharge tests

3/4.8.1, 3/4.8.2, and 3/4.8.3 A.C. SOURCES, AND ONSITE POWER DISTRIBUTION SYSTEMS (Continued)

→(DRN 03-375, Ch. 19)

The OPERABILITY of the minimum specified A.C. and D.C. power sources and associated distribution systems during shutdown and refueling ensures that(1) the facility can be maintained in the shutdown or refueling condition for extended time periods and (2) sufficient instrumentation and control capability is available for monitoring and maintaining the unit status. With the minimum AC and DC power sources and associated distribution systems inoperable the ACTION requires the immediate suspension of various activities including operations involving positive reactivity additions that could result in loss of required SHUTDOWN MARGIN (MODE 5) or boron concentration (MODE 6). Suspending positive reactivity additions that could result in failure to meet the minimum SHUTDOWN MARGIN or boron concentration limit is required to assure continued safe operation. Introduction of coolant inventory must be from sources that have a boron concentration greater than that what would be required in the RCS for minimum SHUTDOWN MARGIN or refueling concentration. This may result in an overall reduction in boron concentration, but provides acceptable margin to maintaining subcritical operation. Introduction of temperature changes, including increases when operating with a positive moderator temperature coefficient, must also be evaluated to ensure they do not result in a loss of required SHUTDOWN MARGIN. Suspension of these activities does not preclude completion of actions to establish a safe conservative condition.

←(DRN 03-375, Ch. 19)

→(EC-38571, Ch. 71)

The fuel handling accident (UFSAR Section 15.7.3.4) analysis assumes protection against load movements with or over irradiated fuel assemblies that could cause fuel assembly damage. Examples of load movements include movement of new fuel assemblies, irradiated fuel assemblies, and the dummy fuel assembly. The load movements do not include the movement over assemblies in a transfer cask using a single-failure-proof handling system. The load movements do not include the movement of the spent fuel machine or refuel machine without loads attached. It also does not include load movements in containment when the reactor vessel head or Upper Guide Structure is still installed. Load movements also exclude suspended loads weighing less than 1000 lbm (e.g. Westinghouse analysis CN-NFPE-09-57 describes no fuel failure for loads weighing less than 1000 lbm based upon the 2000 lbm analysis for drops distributed over two assemblies).

←(EC-38571, Ch. 71)

→(LBDCR 13-017, Ch. 80)

SR 4.8.1.1.2e.10

This SR provides verification that each EDG fuel oil transfer pump maintains capability to take suction from the opposite train fuel oil storage tank and transfer that fuel oil to its associated diesel fuel oil feed tank.

Demonstrating this capability supports text contained in Safety Evaluation from Amendment No. 157, Amendment for a previously Un-reviewed Safety Question regarding Emergency Diesel Generator Fuel Oil Storage and Transfer Systems Design Basis. Specifically, Waterford 3 does not fully align with Regulatory Guide 1.137 (RG 1.137), Revision 1, Fuel Oil Systems for Standby Diesel Generators, in that the RG 1.137 endorsed ANSI standard (N195-1976) requires a 7 day time-dependent load calculation plus 10% margin to mitigate design basis accidents. However, Waterford 3 engineering evaluation indicates that the minimum fuel oil required in each Fuel Oil Storage Tank (FOST) per the current Technical Specification is only sufficient to operate its associated EDG for 7 days plus approximately 1% margin.

The NRC evaluation section in Safety Evaluation of Amendment No. 157, for the EDG FOST not having 10% margin in fuel oil inventory, credited acceptability of the design based upon Waterford 3 having EDG Fuel Oil Storage and Transfer Systems cross connecting capabilities. With the ability to cross-tie the two EDG Fuel Oil Storage and Transfer Systems, one EDG will be able to operate continuously for a period of well over 7 days.

Per Safety Evaluation in Amendment 180, TS SR 4.8.1.1.2e verifies that each fuel oil transfer pump transfers fuel to its associated diesel oil feed tank by taking suction from the opposite train FOST via the installed cross connect. This test is performed by aligning the "A" fuel oil transfer pump suction to the "B" FOST, or the "B" fuel oil transfer pump suction to the "A" FOST. Only one train is tested at a time, and that train is considered inoperable during the test. The train that is being tested is considered inoperable. The test alignment requires the normal fuel transfer suction valve to be closed and two cross-connect valves to the opposite train to be opened. When an increase in volume is observed in the associated train's diesel oil feed tank, the fuel oil transfer pump is secured and valves realigned.

←(LBDCR 13-017, Ch. 80)

→(EC-10752, Ch. 56)

<u>LCO 3.8.1.3</u>

ACTION a

→(EC-15945, Ch. 61)

This ACTION ensures that each diesel generator fuel oil storage tank (FOST) contains fuel oil of a sufficient volume to operate each diesel generator for a period of 7 days. An administrative limit of greater than 40,033 gallons assures at least 39,300 usable gallons are stored in the tank accounting for volumetric shrink and instrumentation uncertainty. This useable volume is sufficient to operate the diesel generator for 7 days based on the time-dependent loads of the diesel generator following a loss of offsite power and a design bases accident and includes the capacity to power the engineered safety features in conformance with Regulatory

←(EC-10725, Ch. 56; EC-15945, Ch. 61)