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W3F1-2007-0031

June 13, 2007

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

Subject: Technical Specification Bases Update to the NRC for the Period July 2, 2006 through June 1, 2007
Waterford Steam Electric Station, Unit 3
Docket No. 50-382
License No. NPF-38

Dear Sir or Madam:

Pursuant to Waterford Steam Electric Station Unit 3 Technical Specification 6.16, Entergy Operations, Inc. (EOI) hereby submits an update of all changes made to Waterford 3 Technical Specification Bases since the last submittal per letter W3F1-2006-0037, dated July 18, 2006. This TS Bases update satisfies the requirement listed in 10 CFR 50.71(e).

There are no commitments associated with this submittal. Should you have any questions or comments concerning this submittal, please contact Ron Williams at (504) 739-6255.

Very truly yours,

RJM/RLW/dpg

Attachment: Waterford 3 Technical Specification Bases Revised Pages

A001

NRR

cc: (w/Attachments)

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ATTACHMENT 1
To W3F1-2007-0031

Waterford 3 Technical Specification Bases Revised Pages

T.S. Bases Change No.	Implementation Date	Affected TS Bases Pages	Topic of Change
46	9/26/2006	Page B 3/4 1-1a	Change No. 46 to TS Bases section 3/4.1.1.4 was implemented by ER-W3-2006-0243. TS Bases Section 3/4.1.1.4 was changed to reflect License Amendment 205, which increased minimum temperature for criticality from 520°F to 533°F.
47	10/26/2006	Page B 3/4 1-1a	Change No. 47 to TS Bases section 3/4.1.1.3 was implemented by ER-W3-2006-0260. TS Bases Section 3/4.1.1.3 was changed to reflect License Amendment 206, which changed TS 3.1.1.3, moderator temperature coefficient to permit use of the Startup Test Activity Reduction Program (STAR).
48	10/26/2006	Index Page VI Index Page XI Index Page XII Index Page XVI Index Page XVII Index Page XXI Page B 3/4 4-2 Page B 3/4 4-3 Page B 3/4 4-3a (new) Page B 3/4 4-3b (new) Page B 3/4 4-3c (new) Page B 3/4 4-3d (new) Page B 3/4 4-4e	Change No. 48 to TS Bases section 3/4.4.4 was implemented by ER-W3-2006-0225. TS Bases 3/4 4.4, Steam Generators (SG), was changed to reflect the changes to the primary-to-secondary leakage requirements, SG tube surveillance integrity program, and SG surveillance requirements contained within TS TS 3/4.4.4, and two new specifications, TS 6.5.9, "Steam Generator (SG) Program," and TS 6.9.1.5, "Steam Generator Tube Inspection Report." The TS changes implemented the guidance for the industry initiative in Nuclear Energy Institute 97-06, Steam Generator Program Guidelines." The changes were also consistent with TS Task Force (TSTF) Change TSTF-449, Revision 4, "Steam Generator Tube Integrity."
49	11/9/2006	Page B 3/4 4-3a Page B 3/4 4-3d	Change No. 49 to TS Bases section 3/4 4.4 was implemented by ER-W3-2006-0228. TS Bases Section 3/4.4.4 was administratively changed to reflect the extent of SG tube inspections in the hot-leg side of the tubesheet delineated in TS 6.5.9 and the addition of a Westinghouse analysis reference that was used to support the TS change.

50	2/27/2007	Page B 3/4 7-4b	Change No. 50 to TS Bases section 3/4.7.6.5 was implemented by ER-W3-2007-0046. TS Bases Section 3/4.7.6.5 was changed to eliminate the potential for misinterpretation of the TS 3.7.6.5 ACTION 'a.' TS Bases 3/4.7.6.5, "Control Room Isolation and Pressurization" lists only two of the six HVC isolation valves that are applicable to Action 'a.' of Tech Spec 3.7.6.5. The existing Bases statement only included valves HVC-102 & HVC-101, while it should have also included valves HVC-306, -307, -313, and -314 since these latter valves are also part of the Control Room envelope pressure boundary isolation. The Bases change made in association with License Amendment 115 should have included the omitted valves. This omission was captured in condition report CR-WF3-2006-02190.
51	3/12/2007	Page B 3/4 7-3e	Change No. 51 to TS Bases section 3/4.7.1.6 was implemented by ER-W3-2005-1650. TS Bases Section 3/4. 7.1.6 was changed to replace the numerical static stroke time discussion with a qualitative discussion. The numerical static stroke time that corresponds to the 6 second TS requirement is subject to change due to multiple design factors that are capable of being changed without impacting any design functions or methodology.
52	5/1/2007	Page B 3/4 4-4 Page B 3/4 4-4a Page B 3/4 4-4b Page B 3/4 4-4c Page B 3/4 4-4d Page B 3/4 4-4e (deleted)	Change No. 52 to TS Bases section 3/4. 3.4.5.1 was implemented by ER-W3-2007-0038. TS Bases Section 3/4. 4.5.1 was changed to reflect implementation of Amendment 212, which deleted reference to the Containment Fan Cooler condensate flow switch.

TECHNICAL SPECIFICATION BASES
CHANGE NO. 46 REPLACEMENT PAGE
(1 page)

Replace the following page of the Waterford 3 Technical Specification Bases with the attached page. The revised page is identified by Change Number 46 and contains the appropriate DRN number and a vertical line indicating the areas of change.

Remove	**Insert**
B 3/4 1-1a	B 3/4 1-1a

3/4.1.1.3 MODERATOR TEMPERATURE COEFFICIENT

The limitations on moderator temperature coefficient (MTC) are provided to ensure that the assumptions used in the accident and transient analysis remain valid through each fuel cycle. The Surveillance Requirements consisting of beginning of cycle measurements, plant parameter monitoring, and end of cycle MTC predictions ensures that the MTC remains within acceptable values. The confirmation that the measured values are within a tolerance of $\pm 0.16 \times 10^{-4}$ delta k/k/°F from the corresponding design values prior to 5% power and 40 EFPD provides assurances that the MTC will be maintained within acceptable values throughout each fuel cycle. CE NPSD 911 and CE NPSD 911 Amendment 1, "Analysis of Moderator Temperature Coefficients in Support of a Change in the Technical Specifications End of Cycle Negative MTC Limit", provide the analysis that established the design margin of $\pm 0.16 \times 10^{-4}$ delta k/k/°F.

3/4.1.1.4 MINIMUM TEMPERATURE FOR CRITICALITY

→(DRN 05-896, Ch. 41; 06-790, Ch. 46)

This specification ensures that the reactor will not be made critical with the indicated Reactor Coolant System cold leg temperature less than 533°F. This limitation is required to ensure (1) the moderator temperature coefficient is within its analyzed temperature range, (2) the protective instrumentation is within its normal operating range, (3) the pressurizer is capable of being in an OPERABLE status with a steam bubble, and (4) the reactor pressure vessel is above its minimum RT_{NDT} temperature.

←(DRN 05-896, Ch. 41; 06-790, Ch. 46)

TECHNICAL SPECIFICATION BASES
CHANGE NO. 47 REPLACEMENT PAGE
(1 page)

Replace the following page of the Waterford 3 Technical Specification Bases with the attached page. The revised page is identified by Change Number 47 and contains the appropriate DRN number and a vertical line indicating the areas of change.

Remove	**Insert**
B 3/4 1-1a	B 3/4 1-1a

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→(DRN 06-814, Ch. 47)

For fuel cycles that meet the applicability requirements of WCAP-16011-P-A, Revision 0, “Startup Test Activity Reduction Program,” SR 4.1.1.3.2.a may be met prior to exceeding 5% of RATED THERMAL POWER after each fuel loading by confirmation that the predicted MTC, when adjusted for the measured RCS boron concentration, is within the MTC limits. WCAP-16011-P-A also provides the basis for using only the near 40 EFPD surveillance test result to justify elimination of the near two-thirds of expected core burnup surveillance when applicability requirements are met. Performance of only one measurement at power is justified based on the WCAP-16011-P-A conclusion that ITC startup test data between different operating conditions is poolable.

The applicability requirements in WCAP-16011-P-A ensure core designs are not significantly different than those used to benchmark predictions and require that the measured RCS boron concentration meets specific test criteria. This provides assurance that the MTC obtained from the adjusted predicted MTC is accurate.

For fuel cycles that do not meet the applicability requirements in WCAP-16011-P-A, the verification of MTC required prior to entering MODE 1 after each fuel loading is performed by measurement of the isothermal temperature coefficient.

←(DRN 06-814, Ch. 47)

3/4.1.1.4 MINIMUM TEMPERATURE FOR CRITICALITY

→(DRN 05-896, Ch. 41; 06-790, Ch. 46)

This specification ensures that the reactor will not be made critical with the indicated Reactor Coolant System cold leg temperature less than 533°F. This limitation is required to ensure (1) the moderator temperature coefficient is within its analyzed temperature range, (2) the protective instrumentation is within its normal operating range, (3) the pressurizer is capable of being in an OPERABLE status with a steam bubble, and (4) the reactor pressure vessel is above its minimum RT_{NDT} temperature.

←(DRN 05-896, Ch. 41; 06-790, Ch. 46)

TECHNICAL SPECIFICATION INDEX AND BASES
CHANGE NO. 48 REPLACEMENT PAGES
(13 pages)

Replace the following pages of the Waterford 3 Technical Specification Index and Bases with the attached pages. The revised pages are identified by Change Number 48 and contain the appropriate DRN number and a vertical line indicating the area of changes.

Remove	**Insert**
Index VI	Index VI
Index XI	Index XI
Index XII	Index XII
Index XVI	Index XVI
Index XVII	Index XVII
Index XXI	Index XXI
B 3/4 4-2	B 3/4 4-2
B 3/4 4-3	B 3/4 4-3
-----	B 3/4 4-3a
-----	B 3/4 4-3b
-----	B 3/4 4-3c
-----	B 3/4 4-3d
B 3/4 4-4e	B 3/4 4-4e

INDEX

LIMITING CONDITION FOR OPERATION AND SURVEILLANCE REQUIREMENTS

←(DRN 05-747, Ch. 40)

→(DRN 05-747, Ch. 40)

INDEX

BASES

←(DRN 05-747, Ch. 40; 06-916, Ch. 48)

BASES

SECTION PAGE

3/4.4 REACTOR COOLANT SYSTEM (Continued)

3/4.5 EMERGENCY CORE COOLING SYSTEMS (ECCS)

3/4.6 CONTAINMENT SYSTEMS

3/4.7 PLANT SYSTEMS

←(DRN 05-747, Ch. 40)

→(DRN 05-747, Ch. 40)

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ADMINISTRATIVE CONTROLS

INDEX

ADMINISTRATIVE CONTROLS

←(DRN 05-747, Ch. 40)

LIST OF TABLES (Continued)

←(DRN 05-747, Ch. 40)

SAFETY VALVES (Continued)

valves are OPERABLE, an operating shutdown cooling loop, connected to the RCS, provides overpressure relief capability and will prevent RCS overpressurization. In addition, the overpressure protection system provides a diverse means of protection against RCS overpressurization at low temperatures.

During operation, all pressurizer code safety valves must be OPERABLE to prevent the RCS from being pressurized above its safety limit of 2750 psia. The combined relief capacity of these valves is sufficient to limit the system pressure to within its Safety Limit of 2750 psia following a complete loss of turbine generator load while operating at RATED THERMAL POWER and assuming no reactor trip until the first Reactor Protective System trip setpoint (Pressurizer Pressure-High) is reached and also assuming no operation of the steam dump valves.

Demonstration of the safety valves' lift settings will occur only during reactor shutdown and will be performed in accordance with the provisions of Section XI of the ASME Boiler and Pressure Vessel Code.

3/4.4.3 PRESSURIZER

An OPERABLE pressurizer provides pressure control for the Reactor Coolant System during operations with both forced reactor coolant flow and with natural circulation flow. The minimum water level in the pressurizer assures the pressurizer heaters, which are required to achieve and maintain pressure control, remain covered with water to prevent failure, which could occur if the heaters were energized while uncovered. The maximum water level in the pressurizer ensures that this parameter is maintained within the envelope of operation assumed in the safety analysis. The maximum water level also ensures that the RCS is not a hydraulically solid system and that a steam bubble will be provided to accommodate pressure surges during operation. The steam bubble also protects the pressurizer code safety valves against water relief. The requirement to verify that on an SIAS test signal the pressurizer heaters are automatically shed from the emergency power sources is to ensure that the non-Class 1E heaters do not reduce the reliability of or overload the emergency power source. The requirement that a minimum number of pressurizer heaters be OPERABLE enhances the capability to control Reactor Coolant System pressure and establish and maintain natural circulation.

The auxiliary pressurizer spray is used to depressurize the RCS by cooling the pressurizer steam space. The auxiliary pressurizer spray is used during those periods when normal pressurizer spray is not available, such as the later stages of a normal RCS cooldown. The auxiliary pressurizer spray also distributes boron to the pressurizer when normal pressurizer spray is not available.

The auxiliary pressurizer spray is used, in conjunction with the throttling of the HPSI pumps, during the recovery from a steam generator tube rupture accident. The auxiliary pressurizer spray is also used during a natural circulation cooldown as a safety related means of RCS depressurization to achieve shutdown cooling system initiation conditions and subsequent COLD SHUTDOWN per the requirements of Branch Technical Position (RSB) 5-1.
→(DRN 06-916, Ch. 48)
←(DRN 06-916, Ch. 48)

→(DRN 06-916, Ch. 48)

3/4.4.4 STEAM GENERATOR TUBE INTEGRITY

Background

Steam generator (SG) tubes are small diameter, thin walled tubes that carry primary coolant through the primary to secondary heat exchangers. Steam generator tubes are an integral part of the reactor coolant pressure boundary (RCPB) and, as such, are relied on to maintain the primary system's pressure and inventory. The SG tubes isolate the radioactive fission products in the primary coolant from the secondary system. In addition, as part of the RCPB, the SG tubes are unique in that they act as the heat transfer surface between the primary and secondary systems to remove heat from the primary system. This Specification addresses only the RCPB integrity function of the SG. The SG heat removal function is addressed by LCO 3.4.1.1, "RCS Loops - MODES 1 and 2," LCO 3.4.1.2, "RCS Loops - MODE 3**," LCO 3.4.1.3, RCS Loops - MODE 4," and LCO 3.4.1.4, "RCS Loops - MODE 5 with reactor coolant loops filled**."

SG tube integrity means that the tubes are capable of performing their intended RCPB safety function consistent with the licensing basis, including applicable regulatory requirements. Steam generator tubing is subject to a variety of degradation mechanisms. Steam generator tubes may experience tube degradation related to corrosion phenomena, such as wastage, pitting, intergranular attack, and stress corrosion cracking, along with other mechanically induced phenomena such as denting and wear. These degradation mechanisms can impair tube integrity if they are not managed effectively. The SG performance criteria are used to manage SG tube degradation.

Specification 6.5.9, *Steam Generator Program*, requires that a program be established and implemented to ensure that SG tube integrity is maintained. Pursuant to Specification 6.5.9, tube integrity is maintained when the SG performance criteria are met. There are three SG performance criteria: structural integrity, accident induced leakage, and operational leakage. The SG performance criteria are described in Specification 6.5.9. Meeting the SG performance criteria provides reasonable assurance of maintaining tube integrity at normal and accident conditions. The processes used to meet the SG performance criteria are defined by NEI 97-06, *Steam Generator Program Guidelines* (Reference 1).

Safety Analysis

The Steam Generator Tube Rupture (SGTR) accident is the limiting design basis event for SG tubes and avoiding a SGTR is the basis for this Specification. The analysis of a SGTR event is based on the leakage rate associated with a double-ended rupture of a single tube. The accident analysis for a SGTR assumes a Loss of Offsite Power with subsequent releases to the atmosphere via Main Steam Safety Valves and Atmospheric Dump Valves.

The analysis for design basis accidents and transients other than a SGTR assume the SG tubes retain their structural integrity (i.e., they are assumed not to rupture). For those analyzed events that do not result in faulted steam generators, greater than or equal to 75 gpd primary to secondary leakage per steam generator is assumed in the analysis. For those analyzed events that result in a faulted steam generator (e.g., MSLB), 540 gpd primary to secondary leakage is assumed through the faulted steam generator while greater than or equal to 75 gpd primary to secondary leakage is assumed through the intact steam generator.

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

The LCO requires that SG tube integrity be maintained. The LCO also requires that all SG tubes that satisfy the repair criteria be plugged in accordance with the *Steam Generator Program*. During a SG inspection, any inspected tube that satisfies the *Steam Generator Program* repair criteria is removed from service by plugging. If a tube was determined to satisfy the repair 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.

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.

<u>Actions</u>

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.

ACTION "a" applies if it is discovered that one or more SG tubes examined in an inservice inspection satisfy the tube repair 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 repair 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.

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.

The *Steam Generator Program* determines the scope of the inspection and the methods used to determine whether the tubes contain flaws satisfying the tube repair 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.

As required by SR 4.4.4.2, any inspected tube that satisfies the *Steam Generator Program* repair criteria is removed from service by plugging. The tube repair 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 repair 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.

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

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 04-1223, Ch. 33)

REACTOR COOLANT SYSTEM.

BASES (continued)

Monitoring Containment Sump In-Leakage Flow

During automatic operation of the containment sump pumps (after a containment sump pump has operated), the flow calculation performed by the plant monitoring computer based on a level change will no longer be accurate since the level in the sump will be lowering. A 20 minute time period has been conservatively determined based on engineering calculations for this equipment operation. In addition, upon reboot of the plant monitoring computer, a period of 10 minutes is required for the leak rate calculation to become available. It has been determined these time periods (independent or combined) of calculation sump in-leakage flow inaccuracies, the instrumentation remains adequate to detect a leakage rate, or its equivalent, of one gpm in less than one hour; therefore, the containment sump level instrumentation and the corresponding flow calculation is considered to remain operable.

References

2. 10 CFR 50, Appendix A, Section IV, GDC 30.
3. Regulatory Guide 1.45, Revision 0, dated May 1973.
4. UFSAR, Sections 5.2.5 and 12.3.

← (DRN 04-1223, Ch. 33)

3/4.4.5.2 OPERATIONAL LEAKAGE

Industry experience has shown that while a limited amount of leakage is expected from RCS, the unidentified portion of this leakage can be reduced to a threshold value of less than 1 gpm. This threshold value is sufficiently low to ensure early detection of additional leakage.

The 10 gpm IDENTIFIED LEAKAGE limitation provides allowances for a limited amount of leakage from known sources whose presence will not interfere with the detection of UNIDENTIFIED LEAKAGE by the leakage detection systems.

The Surveillance Requirements for RCS pressure isolation valves provide added assurance of valve integrity thereby reducing the probability of gross valve failure and consequent intersystem LOCA. Leakage from the RCS pressure isolation valves is IDENTIFIED LEAKAGE and will be considered as a portion of the allowable limit.

→(DRN 04-1243, Ch. 38; 06-916, Ch. 48)

The primary to secondary leakage limit of 75 gallons per day through any one SG is based on the operational leakage performance criterion in NEI 97-06. The Steam Generator Program operational leakage performance criterion in NEI 97-06 states, "The RCS operational primary to secondary leakage through any one SG shall be limited to 150 gallons per day." The NEI 97-06 limit is based on operating experience with SG tube degradation mechanisms that result in tube leakage. The operational leakage rate criterion (since it is less than 150 gpd through any one SG) in conjunction with the implementation of the Steam Generator Program is an effective measure for minimizing the frequency of steam generator tube ruptures.

←(DRN 04-1243, Ch. 38; 06-916, Ch. 48)

TECHNICAL SPECIFICATION BASES
CHANGE NO. 49 REPLACEMENT PAGE(S)
(2 pages)

Replace the following pages of the Waterford 3 Technical Specification Bases with the attached pages. The revised pages are identified by Change Number 49 and contain the appropriate DRN number and a vertical line indicating the areas of change.

Remove	**Insert**
B 3/4 4-3a	B 3/4 4-3a
B 3/4 4-3d	B 3/4 4-3d

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)

The LCO requires that SG tube integrity be maintained. The LCO also requires that all SG tubes that satisfy the repair criteria be plugged in accordance with the *Steam Generator Program*. During a SG inspection, any inspected tube that satisfies the *Steam Generator Program* repair criteria is removed from service by plugging. If a tube was determined to satisfy the repair 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, from 10.6 inches below the bottom of the hot leg expansion transition or top of the hot leg tubesheet, whichever is lower, completely around the U-bend to 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)

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

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

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)

7. Westinghouse WCAP-16208-P, Revision 1, "NDE Inspection Length for CE Steam Generator Tubesheet Region Explosive Expansions," May 2005

TECHNICAL SPECIFICATION BASES
CHANGE NO. 50 REPLACEMENT PAGE
(1 page)

Replace the following page of the Waterford 3 Technical Specification Bases with the attached page. The revised page is identified by Change Number 50 and contains the appropriate DRN number and a vertical line indicating the areas of change.

Remove	**Insert**
B 3/4 7-4b	B 3/4 7-4b

3/4.7.6.3 and 3/4.7.6.4 CONTROL ROOM AIR TEMPERATURE

Maintaining the control room air temperature less than or equal to 80°F ensures that (1) the ambient air temperature does not exceed the allowable air temperature for continuous duty rating for the equipment and instrumentation in the control room, and (2) the control room will remain habitable for operations personnel during plant operation.

The Air Conditioning System is designed to cool the outlet air to approximately 55°F. Then, non-safety-related near-room heaters add enough heat to the air stream to keep the rooms between 70 and 75°F. Although 70 to 75°F is the normal control band, it would be too restrictive as an LCO. Control Room equipment was specified for a more general temperature range to 45 to 120°F. A provision for the CPC microcomputers, which might be more sensitive to heat, is not required here. Since maximum outside air make-up flow in the normal ventilation mode comprises less than ten percent of the air flow from an AH-12 unit, outside air temperature has little affect on the AH-12s cooling coil heat load. Therefore, the ability of an AH-12 unit to maintain control room temperature in the normal mode gives adequate assurance of its capability for emergency situations.

The ACTION to suspend all operations involving movement of irradiated fuel assemblies shall not preclude completion of movement to a safe conservative position.

3/4.7.6.5 CONTROL ROOM ISOLATION AND PRESSURIZATION

This specification provides the operability requirements for the control room envelope isolation and pressurization boundaries. The Limiting Condition for Operation (LCO) specifies specific ACTION STATEMENTS for inoperable components of the control room ventilation systems, separate from the S-8 and AH-12 units. The operability of the remaining parts of the system affect the ability of the control room envelope to pressurize.

→(DRN 07-112, Ch. 50)

ACTION STATEMENTS a and b focus on maintaining isolation characteristics. The valves in the flow path referred to in ACTION a are HVC-101, HVC-102, HVC-306, HVC-307, HVC-313 and HVC-314. The Outside Air Intake (OAI) "series isolation valves" of ACTION b and c are as follows:

←(DRN 07-112, Ch. 50)

NORTH OAI -	HVC-202B & HVC-201A
	HVC-202A & HVC-201B
SOUTH OAI -	HVC-204B & HVC-203A
	HVC-204A & HVC-203B

TECHNICAL SPECIFICATION BASES
CHANGE NO. 51 REPLACEMENT PAGE(S)
(1 page)

Replace the following page of the Waterford 3 Technical Specification Bases with the attached page. The revised page is identified by Change Number 51 and contains the appropriate DRN number and a vertical line indicating the areas of change.

Remove	**Insert**
B 3/4 7-3e	B 3/4 7-3e

3/4.7.1.6 MAIN FEEDWATER ISOLATION VALVES (con't)

The TS is annotated with a 3.0.4 exemption, allowing entry into the applicable MODES to be made with an inoperable MFIV closed or isolated as required by the ACTIONS. The ACTIONS allow separate condition entry for each valve by using "With one or more MFIV...". This prevents immediate entry into TS 3.0.3 if both MFIVs are declared inoperable.

→(DRN 03-1807, Ch. 30; 04-1243, Ch. 38, 05-1650)

The Surveillance Requirement to verify isolation in less than or equal to 6 seconds is based on the time assumed in the accident and containment analyses. The design basis correlates a static test utilizing one accumulator to demonstrate the ability of the MFIVs to close in less than or equal to 6 seconds under design basis accident conditions with two accumulators. The static stroke time test that utilizes one accumulator is allowed to exceed the 6 second Surveillance Requirement since both accumulators are credited in the design basis Accidents in order to isolate within the 6 second Surveillance Requirement. The 6 second required closure time includes a 1 second allowance for instrument response time.

← (DRN 05-1650)

The MFIVs should not be tested at power since even a partial stroke exercise increases the risk of a valve closure with the plant generating power and would create added cyclic stresses. The Surveillance to verify each MFIV can close on an actual or simulated actuation signal is normally performed when the plant is returning to operation following a refueling outage. Verification of valve closure on an actuation signal is not required until entry into Mode 3 consistent with TS 3.3.2. The 18 month frequency is based on the refueling cycle. Verification of closure time is performed per the Inservice Testing Program. This frequency is acceptable from a reliability standpoint and is in accordance with the Inservice Testing Program.

→(DRN 03-1807, Ch. 30)

←(DRN 02-1684, Ch. 15; 04-1243, Ch. 38)

Credited Non-Safety Related Support Systems for MFIV Operability

Reactor Trip Override (RTO) and the Auxiliary Feedwater (AFW) Pump High Discharge Pressure Trip (HDPT) are credited for rapid closure of the Main Feedwater Isolation Valves (MFIVs) during main steam and feedwater line breaks. Crediting of these non-safety features was submitted to the NRC as a USQ and approved. (Reference letter dated September 5, 2000 from the NRC to Charles M. Dugger, "Waterford 3 Steam Electric Station, Unit 3 - Issuance of Amendment RE: Addition of Main Feedwater Ioslation Valves to Technical Specifications and Request for NRC Staff Review of an Unreviewed Safety Question.")

The feature of RTO that is credited for MFIV closure is the rapid SGFP speed reduction upon reactor trip initiation. This feature reduces the differential pressure across the valve disc at closure, thus allowing rapid valve closure. Therefore, the RTO feature must be able to decrease SGFP speed to minimum on a reactor trip during SGFP operation for OPERABILITY of the MFIVs.

The AFW Pump HDPT reduces the differential pressure across the valve disc at closure during AFW Pump operation. Therefore, this feature must be functional during AFW Pump operation for OPERABILITY of the MFIVs. When the AFW pump is not running, this trip is not required.

In MODES 1, 2, 3, and 4, the MFIVs are required to be OPERABLE. Because the MFIVs are required to be OPERABLE in MODES 1, 2, 3, and 4, RTO must be able to decrease SGFP

←(DRN 02-1684, Ch. 15)

TECHNICAL SPECIFICATION BASES
CHANGE NO. 52 REPLACEMENT PAGE(S)
(5 pages)

Replace the following pages of the Waterford 3 Technical Specification Bases with the attached pages. The revised pages are identified by Change Number 52 and contain the appropriate DRN number and a vertical line indicating the areas of change.

Remove	Insert
B 3/4 4-4	B 3/4 4-4
B 3/4 4-4a	B 3/4 4-4a
B 3/4 4-4b	B 3/4 4-4b
B 3/4 4-4c	B 3/4 4-4c
B 3/4 4-4d	B 3/4 4-4d
B 3/4 4-4e	

3/4.4.5 REACTOR COOLANT SYSTEM LEAKAGE

3/4.4.5.1 LEAKAGE DETECTION SYSTEMS

→ (DRN 04-1223, Ch. 33)

Background

GDC 30 of Appendix A 10 CFR 50 (Ref. 1) requires means for detecting and, to the extent practical, identifying the location of the source of RCS leakage. Regulatory Guide 1.45 (Ref. 2) describes acceptable methods for selecting leakage detection systems.

Leakage detection systems must have the capability to detect significant reactor coolant pressure boundary(RCPB) degradation as soon after occurrence as practical to minimize the potential for propagation to a gross failure. Thus an early indication or warning signal is necessary to permit proper evaluation of all UNIDENTIFIED LEAKAGE.

→(DRN 07-203, Ch. 52)

Industry practice has shown that water flow changes of 0.5 gpm to 1.0 gpm can readily be detected in contained volumes by monitoring changes in water level or in flow rate. The containment sump used to collect UNIDENTIFIED LEAKAGE is instrumented to alarm for increases of 0.5 gpm to 1.0 gpm in the normal flow rates. This sensitivity is acceptable for detecting increases in UNIDENTIFIED LEAKAGE.

←(DRN 07-203, Ch. 52)

The reactor coolant contains radioactivity that, when released to the containment, can be detected by radiation monitoring instrumentation. Reactor coolant radioactivity levels will be low during initial reactor startup and for a few weeks thereafter until activated corrosion products have been formed and fission products appear from fuel element cladding contamination or cladding defects. Radioactivity detection systems are included for monitoring particulate activities, because of their sensitivities and rapid responses to RCS leakage.

Air temperature and pressure monitoring methods may also be used to infer UNIDENTIFIED LEAKAGE to the containment. Containment temperature and pressure fluctuate slightly during plant operation, but a rise above the normally indicated range of values may indicate RCS leakage into the containment. The relevance of temperature and pressure measurements is affected by containment free volume and, for temperature, detector location. Alarm signals from these instruments can be valuable in recognizing rapid and sizable leakage to the containment. Temperature and pressure monitors are not required by this LCO.

Applicable Safety Analyses

The need to evaluate the severity of an alarm or an indication is important to the operators, and the ability to compare and verify with indications from other systems is necessary. The system response times and sensitivities are described in the UFSAR (Ref. 3). Multiple instrument locations are utilized, if needed, to ensure the transport delay time of the leakage from its source to an instrument location yields an acceptable overall response time.

The safety significance of RCS leakage varies widely depending on its source, rate, and duration. Therefore, detecting and monitoring RCS leakage into the containment area are necessary. Quickly separating the IDENTIFIED LEAKAGE from the UNIDENTIFIED LEAKAGE provides quantitative information to the operators, allowing them to take corrective action should leakage occur detrimental to the safety of the facility and the public. RCS leakage detection instrumentation satisfies Criterion 1 of 10 CFR 50.36(c)(2)(ii).

← (DRN 04-1223, Ch. 33)

Limiting Condition for Operation

One method of protecting against large RCS leakage derives from the ability of instruments to rapidly detect extremely small leaks. This LCO requires instruments of diverse monitoring principles to be OPERABLE to provide a high degree of confidence that small leaks are detected in time to allow actions to place the plant in a safe condition when RCS leakage indicates possible RCPB degradation.

→(DRN 07-203, Ch. 52)

The LCO is satisfied when monitors of diverse measurement means are available. Thus, the containment sump monitors (either the containment sump level instrumentation/time rate of change or the containment flow instrumentation (weir)), in combination with a particulate radioactivity monitor, provide acceptable monitoring capability for leakage detection.

←(DRN 07-203, Ch. 52)

→(DRN 07-203, Ch. 52)

←(DRN 07-203, Ch. 52)

Applicability

Because of elevated RCS temperature and pressure in MODES 1, 2, 3, and 4, RCS leakage detection instrumentation is required to be OPERABLE.

In MODE 5 or 6, the temperature is ≤200°F and pressure is maintained low or at atmospheric pressure. Since the temperatures and pressures are far lower than those for MODES 1, 2, 3, and 4, the likelihood of leakage and crack propagation is much smaller. Therefore, the requirements of this LCO are not applicable in MODES 5 and 6.

Actions

The Actions are modified by a Note that indicates the provisions of TS 3.0.4 are not applicable. This allowance is provided because other instrumentation is available to monitor RCS leakage.

Action a

→(DRN 07-203, Ch. 52)

With the containment atmosphere particulate radioactivity monitoring instrumentation inoperable, alternative action is required. Either grab samples of the containment atmosphere must be taken and analyzed, or water inventory balances, in accordance with SR 4.4.5.2.1, must be performed to provide alternate periodic information. With a sample obtained and analyzed or an inventory balance performed every 24 hours, the reactor may be operated for up to 30 days to allow restoration of the radioactivity monitor.

←(DRN 07-203, Ch. 52)

The 24 hour interval provides periodic information that is adequate to detect leakage. A Note is added allowing that SR 4.4.5.2.1 is not required to be performed until 12 hours after establishing steady state operation (stable temperature, power level, pressurizer and makeup tank levels, makeup and letdown). The 12 hour allowance provides sufficient time to establish stable plant conditions. The 30 day allowed outage time recognizes at least one other form or leakage detection is available.

← (DRN 04-1223, Ch. 33)

If ACTION 'a' cannot be met within the allowed outage time, the plant must be brought to a MODE in which the LCO does not apply. To achieve this status, the plant must be brought to at least MODE 3 within 6 hours and to MODE 5 within the following 30 hours. The allowed outage times are reasonable, based on operating experience, to reach the required plant conditions from full power conditions in an orderly manner and without challenging plant systems.

Action b

→(DRN 07-203, Ch. 52)

If the containment sump monitor is inoperable, (i.e., both the containment level instrumentation/time rate of change and the containment flow instrumentation (weir)), no other form of sampling can provide the equivalent information.

←(DRN 07-203, Ch. 52)

→(DRN 07-203, Ch. 52)

However, the containment atmosphere radioactivity monitor will provide indication of changes in leakage. Together with the atmosphere monitor, the periodic surveillance for RCS water inventory balance, SR 4.4.5.2.1, must be performed at an increased frequency of 24 hours to provide information that is adequate to detect leakage. A Note is added allowing that SR 4.4.5.2.1 is not required to be performed until 12 hours after establishing steady state operation (stable temperature, power level, pressurizer and makeup tank levels, makeup and letdown). The 12 hour allowance provides sufficient time to establish stable plant conditions.

←(DRN 07-203, Ch. 52)

Restoration of the required sump monitor to OPERABLE status is necessary to regain the function in an allowed outage time of 30 days after the monitor's failure. This time is acceptable considering the remaining OPERABLE leakage detection instrumentation and the frequency and adequacy of the RCS water inventory balance required by the ACTION.

If ACTION 'b' cannot be met within the allowed outage time, the plant must be brought to a MODE in which the LCO does not apply. To achieve this status, the plant must be brought to at least MODE 3 within 6 hours and to MODE 5 within the following 30 hours. The allowed outage times are reasonable, based on operating experience, to reach the required plant conditions from full power conditions in an orderly manner and without challenging plant systems.

← (DRN 04-1223, Ch. 33)

→(DRN 07-203, Ch. 52)

←(DRN 07-203, Ch. 52)

→(DRN 07-203, Ch. 52)
Action c
←(DRN 07-203, Ch. 52)

If all required monitors are inoperable, no automatic means of monitoring leakage are available and immediate plant shutdown is required. ACTION must be initiated within 1 hour to be in MODE 3 within the next 6 hours and MODE 5 in the following 30 hours. These times are consistent with TS 3.0.3.

Surveillance Requirements

SR 4.4.5.1.a, 4.4.5.1.b - Channel Check
→(DRN 07-203, Ch. 52)

SR 4.4.5.1.a requires the performance of a CHANNEL CHECK of the required containment atmosphere particulate radioactivity monitor. SR 4.4.5.1.b requires the performance of a CHANNEL CHECK on the required containment sump level monitor/time rate of change. The CHANNEL CHECK is not required to be performed on the containment sump flow monitor (weir). The check gives reasonable confidence the channel is operating properly. The frequency of 12 hours is based on instrument reliability and is reasonable for detecting off normal conditions.
←(DRN 07-203, Ch. 52)

→(DRN 05-1333, Ch. 44)
SR 4.4.5.1.a, - Channel Functional Test
←(DRN 05-1333, Ch. 44)

SR 4.4.5.1.a requires the performance of a CHANNEL FUNCTIONAL TEST of the required containment atmosphere particulate radioactivity monitor. The test ensures that the monitor can perform its function in the desired manner. The test verifies the alarm setpoint and relative accuracy of the instrument string. *A successful test of the required contacts of a channel relay may be performed by the verification of the change of state of a single contact of* the relay. This clarifies what is an acceptable CHANNEL FUNCTIONAL TEST of a relay. This is acceptable because all of the other required contacts of the relay are verified by other Technical Specifications and non-Technical Specifications tests at least once per refueling interval with applicable extensions. The frequency of 92 days considers instrument reliability, and operating experience has shown it proper for detecting degradation.

→(DRN 05-1333, Ch. 44; DRN 07-203, Ch. 52)

←(DRN 07-203, Ch. 52)

SR 4.4.5.1.a, SR 4.4.5.1.b - Channel Calibration
←(DRN 05-1333, Ch. 44)

These SRs require the performance of a CHANNEL CALIBRATION for each of the RCS leakage detection instrumentation channels. The calibration verifies the accuracy of the instrument string, including the instruments located inside containment. The frequency of 18 months is a typical refueling cycle and considers channel reliability. Operating experience has shown this frequency is acceptable.
←(DRN 04-1223, Ch. 33)

Monitoring Containment Sump In-Leakage Flow

During automatic operation of the containment sump pumps (after a containment sump pump has operated), the flow calculation performed by the plant monitoring computer based on a level change will no longer be accurate since the level in the sump will be lowering. A 20 minute time period has been conservatively determined based on engineering calculations for this equipment operation. In addition, upon reboot of the plant monitoring computer, a period of 10 minutes is required for the leak rate calculation to become available. It has been determined these time periods (independent or combined) of calculation sump in-leakage flow inaccuracies, the instrumentation remains adequate to detect a leakage rate, or its equivalent, of one gpm in less than one hour; therefore, the containment sump level instrumentation and the corresponding flow calculation is considered to remain operable.

References

3. 10 CFR 50, Appendix A, Section IV, GDC 30.
4. Regulatory Guide 1.45, Revision 0, dated May 1973.
5. UFSAR, Sections 5.2.5 and 12.3.

← (DRN 04-1223, Ch. 33)

3/4.4.5.2 OPERATIONAL LEAKAGE

Industry experience has shown that while a limited amount of leakage is expected from RCS, the unidentified portion of this leakage can be reduced to a threshold value of less than 1 gpm. This threshold value is sufficiently low to ensure early detection of additional leakage.

The 10 gpm IDENTIFIED LEAKAGE limitation provides allowances for a limited amount of leakage from known sources whose presence will not interfere with the detection of UNIDENTIFIED LEAKAGE by the leakage detection systems.

The Surveillance Requirements for RCS pressure isolation valves provide added assurance of valve integrity thereby reducing the probability of gross valve failure and consequent intersystem LOCA. Leakage from the RCS pressure isolation valves is IDENTIFIED LEAKAGE and will be considered as a portion of the allowable limit.

→(DRN 04-1243, Ch. 38; 06-916, Ch. 48)

The primary to secondary leakage limit of 75 gallons per day through any one SG is based on the operational leakage performance criterion in NEI 97-06. The Steam Generator Program operational leakage performance criterion in NEI 97-06 states, "The RCS operational primary to secondary leakage through any one SG shall be limited to 150 gallons per day." The NEI 97-06 limit is based on operating experience with SG tube degradation mechanisms that result in tube leakage. The operational leakage rate criterion (since it is less than 150 gpd through any one SG) in conjunction with the implementation of the Steam Generator Program is an effective measure for minimizing the frequency of steam generator tube ruptures.

←(DRN 04-1243, Ch. 38; 06-916, Ch. 48)