Tennessee Valley Authority, Post Office Box 2000, Spring City, Tennessee 37381-2000

DEC 14 2005

WBN-TS-05-07

10 CFR 50.90

U. S. Nuclear Regulatory Commission
Mail Stop: OFWN P1-35
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Washington, D.C. 20555-0001

Gentlemen:

In the Matter of) Docket No. 50-390
Tennessee Valley authority)

WATTS BAR NUCLEAR PLANT (WBN) UNIT 1 - PROPOSED LICENSE AMENDMENT REQUEST NO. WBN-TS-05-07, ONE-TIME FREQUENCY EXTENSION FOR TYPE A TEST (CONTAINMENT INTEGRATED LEAK RATE TEST [CILRT])

Pursuant to 10 CFR 50.90, TVA is submitting a request for a Technical Specification change (WBN-TS-05-07) to License NPF-90. The proposed TS change will revise TS Section 5.7.2.19, "Containment Leakage Rate Testing Program," to allow a one time, 5-year extension to the current 10 year test interval for the performance-based leakage rate test program for 10 CFR 50, Appendix J, Type A test.

The proposed change is submitted on a risk informed basis as described in Regulatory Guide 1.174, *"An Approach for Using Probabilistic Risk Assessment in Risk-Informed Decisions On Plant-Specific Changes to the Licensing Basis,"* and makes use of Revision 3 of the WBN Probabilistic Safety Assessment (PSA). TVA has determined that the resultant increase in Large Early Release Frequency (LERF) for the proposed change is "very small" (i.e., less than 1.0E-07/reactor year) and satisfies the Regulatory Guide 1.174 criteria.

TVA's American Society of Mechanical Engineers (ASME) Subsection IWE program performs containment inspections in

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order to detect evidence of degradation that may affect either the containment structural integrity or leak tightness.

TVA's application represents a cost beneficial licensing change. Performance of a Type A test imposes a significant expense to TVA ($225,000) while the safety benefit of performing a test within 10 years versus 15 years is minimal.

It should be noted that in Fall 2006, WBN will replace steam generators on Unit 1. This project will include cutting the containment structure for removal of the original steam generators. TVA has evaluated requirements associated with post-modification testing (PMT) of the steel containment vessel following steam generator replacement. Based on the evaluation of PMT test requirements, TVA is proposing to perform an ASME code pressure test and local leak rate test of the affected areas in lieu of a full CILRT. This is a technically sound post-modification test that has been performed on similar containment modifications at Sequoyah, Turkey Point, Fitzpatrick, Vermont Yankee and St. Lucie Nuclear Stations. The approach complies with 10 CFR 50, Appendix J, Nuclear Energy Institute (NEI) 94-01, American Nuclear Standard (ANS) 56.8, 1994, and the ASME Section XI Code, 1992 Edition, 1992 Addenda, Subsection IWE.

TVA has determined that there are no significant hazards considerations associated with the proposed change and that the TS change qualifies for categorical exclusion from environmental review pursuant to the provisions of 10 CFR 51.22(c)(9). Additionally, in accordance with 10 CFR 50.91(b)(1), TVA is sending a copy of this letter to the Tennessee State Department of Public Health.

Enclosure 1 to this letter provides description and evaluation of the proposed change. This includes TVA's determination that the proposed change does not involve a significant hazards consideration, and is exempt from environmental review. Enclosure 2 contains a copy of the appropriate TS page, marked-up to show the proposed changes. Enclosure 3 contains the revised TS page, which incorporate the proposed change. Enclosure 4 contains the TVA evaluation of risk significance.

TVA requests approval of this TS change by August 1, 2006, to allow final planning, scheduling, and preparation for the WBN

Unit 1 Cycle 7 refueling outage (scheduled to begin in Fall 2006). In addition, TVA requests implementation of the revised TS be within 45 days of NRC approval.

There are no regulatory commitments associated with this submittal. If you have any questions concerning this matter, please call me at (423) 365-1824.

I declare under penalty of perjury that the foregoing is true and correct. Executed on this 14th day of December, 2005.

Sincerely,

P. L. Pace
Manager, Site Licensing
and Industry Affairs

Enclosures:
1. TVA Evaluation of Proposed Change
2. Proposed Technical Specification Changes (mark-up)
3. Proposed Technical Specification Changes (re-typed)
4. TVA Calculation MDN001-999-2005-0099, Revision 1 - Risk Evaluation

cc: See Page 4

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Enclosures
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ENCLOSURE 1

TENNESSEE VALLEY AUTHORITY
WATTS BAR NUCLEAR PLANT (WBN) UNIT 1
DOCKET NO. 50-390

PROPOSED LICENSE AMENDMENTJ REQUEST WBN-TS-05-07
DESCRIPTION AND EVALUATION OF THE PROPOSED CHANGE

1.0 DESCRIPTION

This letter is a request to amend the Operating License No. NPF-90 for WBN Unit 1.

The proposed change would revise the Operating License to add a one-time, 5-year deferral of the Containment Integrated Leak Rate Test (CILRT), also referred to as the 10 CFR 50, Appendix J, Type A test to Technical Specification (TS) Section 5.7.2.19, "*Containment Leakage Rate Testing Program.*" Section 5.7.2.19 contains the general 10 CFR 50, Appendix J test and leakage requirements for the WBN steel containment structure. The Containment Leakage Rate Testing Program refers to requirements contained in 10 CFR 50, Appendix J, Option B and NRC Regulatory Guide 1.163, "*Performance-Based Containment Leak Test Program,*" dated September 1995 [Reference 1]. The regulatory guide endorses Nuclear Energy Institute (NEI) 94-01, Revision 0, entitled "*Industry Guideline for Implementing Performance Based Option of 10 CFR 50, Appendix J,*" [Reference 2] which requires that Type A tests be performed "at least once per 10-years based on acceptable performance history."

2.0 PROPOSED CHANGE

TVA's proposed change requests, on a one-time basis, an extension to the current 10-year Type A test interval to allow a 15-year test interval.

Section 5.7.2.19, "Containment Leakage Rate Testing Program," is revised to add the following provision for Unit 1:

> "The Fall 2007 end date for conducting the 10 year interval containment integrated leakage rate (Type A) test may be deferred up to 5 years but no later than Fall 2012."

In summary, the proposed above change to TS 5.7.2.19 will revise the Containment Leakage Rate Test Program requirements to allow a one-time, 5-year Type A test frequency extension. Following, the one-time 5-year extension, TVA will revert back to the 10-year interval.

3.0 BACKGROUND

General Description of the Project

The WBN primary containment structure for Unit 1 consists of a freestanding steel vessel with an ice condenser and a separate secondary containment that is a reinforced concrete Shield Building. The primary containment vessel consists of a cylindrical wall, a hemispherical dome, and a bottom liner plate encased in concrete. WBN's Updated Final Safety Analysis Report (UFSAR) Figure 3.8.2-1 shows the outline and configuration of the steel containment vessel. Section 6.2.1 of the WBN FSAR describes WBN's containment design features.

The four steam generators for WBN Unit 1 will be replaced during the Fall 2006 refueling outage. To support the replacement of the original steam generators with the replacement steam generators, access openings will be created in the roof of the concrete Shield Building and the roof of the steel containment vessel (SCV).

The Shield Building concrete dome will have one opening (oval in shape and approximately 22 foot by 45 foot) cut in the dome over each pair of steam generators (two openings total). Each opening will be sized to allow the removal and replacement of the steam generators as well as the cut portion of the SCV beneath it. The cut lines are positioned to avoid the cutting of surface mounted plates and to minimize the cutting of the circumferential stiffeners. The cut sections of the containment vessel will be welded back to their original configuration using full penetration welds. The concrete dome will be restored by removing concrete to expose sufficient rebar, splicing new rebar to the existing rebar, and pouring new concrete. (See WBN License Amendment Request WBN-TS-04-18 dated December 4, 2004 concerning Bar-Lock Splices currently in NRC review.)

Non-destructive examination of the SCV welds will consist of magnetic particle testing (MT) of the back-gouged area of the root pass prior to welding from the secondary side of the weld. After successful completion of the root pass MT, the weld will be completed and 100 percent radiograph tested (RT). If any weld defects are found in the MT or RT, appropriate repairs and retesting will be performed.

Steel Containment Vessel Test

The WBN TS (Section 5.7.2.19) establishes the requirements for implementing a program to perform containment leakage rate testing in accordance with 10 CFR 50.54(o) and 10 CFR 50, Appendix J, Option B, as modified by approved exemptions. The types of containment leakage tests include Type A (CILRT), Type B (local leak rate testing [LLRT]) for containment penetrations, hatches, personnel air locks, electrical penetrations, etc.), and Type C (LLRT for containment isolation valves). WBN's maximum allowable containment leakage rate is 1.0 L_a which is defined as 0.25 percent of the containment free air volume per day at an accident pressure of 15.0 pounds per square inch (psi).

The Type A test interval for WBN is based on Type A test history and performance and is currently once every 10 years. The test interval for Type A testing is based on NEI 94–01, that states: *"Type A testing shall be performed during a period of reactor shutdown at a frequency of at least once per 10 years based on acceptable performance history. Acceptable performance history is defined as completion of two consecutive periodic Type A tests where the calculated performance leakage rate was less than 1.0 La."* Also included within NEI 94–01, Section 11.3, is consideration of Plant–Specific Testing Program.

TVA plans to perform an Appendix J, Type B test by pressurizing containment to the required test pressure of at least P_a (peak calculated accident pressure of 15.0 pounds per square inch gauge [psig]) and performing a bubble test of the repair welds after a hold time of at least 10 minutes. The acceptance criterion is zero detectable leakage. The test pressure will be held above 15.0 psig during the bubble test. The Appendix J, Type B test is to satisfy the steam generator replacement Post Maintenance Test (PMT) requirements.

ASME Section XI, Subsection IWE–5221 states in part:

"Except as noted in IWE–5222, repairs/replacement activities performed on the pressure retaining boundary of Class MC or Class CC components shall be subjected to a pneumatic leakage test in accordance with the provisions of Title 10, Part 50 of the Code of Federal Regulations, Appendix J, Paragraph IV.A." [Paragraph IV.A states in part that *any major modification, replacement of a component which is part of the primary reactor containment boundary, or resealing a seal welded door, performed after the preoperational leakage rate test shall be followed by a either a Type A, Type B, or Type C test as applicable for the area affected by the modification."]*

10 CFR 50 Appendix J, Option B, Paragraph III.B defines Type A and B test in part:

"Type A tests to measure the containment system overall integrated leakage rate . . ." and "Type B pneumatic tests to detect and measure local leakage rates across pressure retaining, leakage—limiting boundaries. . ."

This test complies with the definition above for Type B in that it will locally detects leakage across the pressure retaining leakage limiting boundary. An alternative Type A test would involve a variation of the Type A testing prescribed in Appendix J. TVA is not proposing any variation to this type of test. The only element of the proposed Type B test which is similar to a Type A test is the pressurization of the entire containment vessel. The test, which WBN plans to perform, will pressurize the repair welds to the required pressure in the direction of applied stress which would be seen during accident conditions. The acceptance criterion of zero detectable leakage is more stringent than the Type A test acceptance criteria for the repair area. Accordingly, the repair welds will be tested in accordance with ASME Section XI, Subsection IWE, Paragraph IWE-5221, and 10 CFR 50, Appendix J. Since the acceptance criterion is zero leakage under accident pressure conditions, there is no adverse affect upon the overall containment leakage rate and the repair welds are shown to have complete integrity. Therefore, the prescribed testing is in accordance with ASME Section XI and 10 CFR 50, Appendix J.

TVA has performed two Type A tests on WBN Unit 1 containment. The last test was conducted in September 1997 during the Unit 1 Cycle 1 Refueling Outage. In accordance with the current WBN TS requirements, Unit 1 is required to perform the next 10-year CILRT prior to Fall 2007. In order to perform the test prior to that date, the CILRT would have to be performed during the upcoming Unit 1 Cycle 7 Refueling Outage in the Fall of 2006.

The cost to TVA for performing a CILRT is substantial (conservative estimated cost is $225,000). Additional replacement power costs include 20 hours of critical path time. A conservative estimate of radiological cost to perform a Type A test is 500 millirem of dose. Accordingly, TVA is proposing a change to TS Section 5.7.2.19 to defer conducting a Type A test to save critical path time during the upcoming Cycle 7 Refueling Outage in which the steam generators are being replaced. TVA realizes that the cost and dose associated with the performance of the CILRT will only be delayed to a subsequent outage. However, due to a costly steam generator replacement in an already extended outage, performance of a CILRT would add an addition cost and burden to TVA without increasing the health and safety of the public.

NRC has approved a similar request for TVA's Sequoyah Nuclear Plants (SQN) Unit 1 and 2, to allow a one-time five year extension of the CILRT [Reference 3].

4.0. TECHNICAL ANALYSIS

The testing requirements of 10 CFR 50, Appendix J, provide assurance that leakage through the containment, including systems and components that penetrate the containment, does not exceed the allowable leakage value specified in the WBN TSs, (L_a). The limitation of containment leakage provides assurance that the containment would perform its design function following a design basis accident.

The 10 CFR 50, Appendix J rule was revised in 1995, to allow licensees to choose containment leakage testing under Option A, "*Prescriptive Requirements*" or Option B, "*Performance-Based Requirements.*" TVA requested a license amendment for WBN to allow implementation of Option B and was granted approval by the NRC in License Amendment 5 [Reference 4]. The WBN TS was revised by Amendment 5 to include Option B. The revision included a reference to Regulatory Guide 1.163 [Reference 1] for performing Type A, B, and C testing. Regulatory Guide 1.163 specifies a method acceptable to the NRC for complying with Option B by endorsing the use of NEI 94-01 [Reference 2] and ANSI/ANS 56.8-1994, "*Containment System Leakage Testing Requirements,*" subject to specific regulatory positions in the regulatory guide.

Exceptions to the requirements of Regulatory Guide 1.163 are allowed by 10 CFR 50, Appendix J, Option B, Section V.B, "*Implementation,*" which states:

> *The Regulatory Guide or other implementing document used by a licensee, or applicant for an operating license, to develop a performance-based leakage-testing program must be included, by general reference, in the plant technical specifications. The submittal for technical specification revisions must contain justification, including supporting analyses, if the licensee chooses to deviate from methods approved by the Commission and endorsed in a regulatory guide.*

The adoption of the Option B performance-based containment leakage rate testing program did not alter the basic method by which Appendix J leakage rate testing is performed, but did alter the frequency of measuring primary containment leakage in Type A, B and C tests. Frequency is based upon an evaluation which looks at the "as found" leakage history to determine the frequency for leakage testing that provides assurance that leakage limits will be maintained. The changes to the Type A test frequency do not directly result in an increase in containment leakage. Similarly, the proposed change to the WBN Type A test frequency will not directly result in an increase in containment leakage.

The allowed frequency for testing was based upon a generic evaluation documented in NUREG-1493, "*Performance-Based Containment Leakage-Test Program.*" Section 10.1.2 of this NUREG provided the following observations with regard to the Type A test frequency:

> *Reducing the frequency of Type A tests (ILRTs) from the current three per 10 years to one per 20 years was found to lead to an imperceptible increase in risk. The estimated increase in risk is very small because ILRTs identify only a few potential containment leakage paths that cannot be identified by Type B and C testing, and the leaks that have been found by Type A tests have been only marginally above the existing requirements.*
>
> *Given the insensitivity of risk to containment leakage rate (Chapter 5) and the small fraction of leakage paths detected solely by Type A testing, increasing the interval between integrated leakage-rate tests is possible with minimal impact on public risk…*
>
> *The findings to date strongly support earlier indications that Type B and C testing can detect a very large fraction of containment leaks. The fraction of leaks that can be detected only by integrated containment leakage test is small on the order of a few percents.*

TVA Risk Assessment

A risk assessment for this one-time frequency extension on WBN Unit 1 was performed to determine the risk significance of a decrease in CILRT frequency from 1/10 year to 1/15 year. The effect of a decrease in the frequency of performing a CILRT is that the exposure time of a pre-existing leak in the containment shell increases. The resulting increase in the calculated frequency of both large and small fission product releases to the environment correlates to an increase in calculated population dose. This calculation [Reference 5] quantifies the increase in release frequency and population dose as a result of a decrease in the frequency of performing a CILRT (see Enclosure 4).

The fault tree for small and large containment isolation failures used in the Probablistic Safety Analysis (PSA) accounts for the following failures to isolate containment:

a failure of instrumentation to generate a containment isolation signal along with failure of the operator to manually initiate this action,

a containment penetration failing to isolate as the result of the failure of the inboard and outboard isolation valves to close, and

the existence of a preexisting leak in the containment.

The first two containment isolation failures listed above are identified by engineered safety features actuation system (ESFAS) testing or stroke testing containment isolation valves, respectively. The existence of a leak in a containment penetration is identified by either a LLRT or a CILRT. The existence of a leak in the containment shell is identified by a CILRT. The decrease in the frequency of conducting CILRTs increases the calculated probability of a preexisting leak in containment, but does not affect the probability of the other containment isolation failure mechanisms listed above.

The risk assessment showed the increase in the Large Early Release Frequency (LERF) to be 3.26E-08/ry when the frequency of a Type A test was decreased from 1/10 year to 1/15 year. This LERF value meets the NRC Regulatory Guide 1.174, *"An Approach for Using Probabilistic Risk Assessment In Risk-Informed Decisions On Plant-Specific Changes to the Licensing Basis"*, [Reference 6] that defines very small changes as increases in LERF less than 1.0E-7/ry.

The risk assessment also showed the increase in population dose is 1.1 person-rem when the frequency of a Type A test was decreased from 1/10 year to 1/15 years.

TVA Deterministic Evaluation

In addition to TVA's risk assessment, TVA's proposed TS change is based on performance history from previous Type A tests and WBN's ASME Section XI, Subsection IWE examination and inspection program. A description of Type A test history, inspection results, and future examinations are provided as follows:

Test History Information

The previous WBN Type A test results have shown leakage to be below the 1.0 L_a leakage limit. The performance leak rates of the last two consecutive tests were 0.0143 percent/day = 0.0572 L_a (June 1994) and 0.0444 percent/day =

0.1776 L_a (September 1997). Margins to date from previous tests indicate at least 80 percent margin (worst case).

The risk is further minimized by continued 10 CFR 50, Appendix J Type B and Type C testing. WBN's in-service inspection (ISI) program provides additional confidence in containment structural integrity and leak tightness. Accordingly, the proposed extension of the Type A test represents minimal risk for increased leakage.

Containment Penetrations with Mechanical Bellows

The WBN containment penetration mechanical bellows are within the scope of containment inspection and Appendix J Type A, B, or C leak testing and are two-ply laminated testable bellows. Each bellow is local leak rate tested (Type B) by pressurizing between the two plies. These bellows incorporate a screen mesh between the inner and outer plies to ensure separation is maintained. This design prevents a "pinch" from occurring at the folds and ensures that the entire space between the plies is pressurized and leak tested during Type B testing. If the bellows test fails the Appendix J, Type B test, the bellows' sheet metal cover is removed, the bellows are pressurized to test pressure, and visually inspected for leakage using a bubble solution (snoop), lights, mirrors, etc. The bellows are repaired or replaced as necessary if the bellows are found to be leaking.

Option B of 10 CFR 50, Appendix J allows extended test intervals up to 120 months for Type B components, based on acceptable performance. Due to industry concerns, WBN has limited extended test intervals for bellows to 60 months. Additionally, penetrations with bellows are tested on a staggered basis such that a portion are tested each refueling outage.

A review of TVA records since start up in May 1996 has revealed no test failures of these bellows for WBN Unit 1. The one time 5-year (from 10 to 15 years) extension of the CILRT frequency has no effect on this testing since the frequency of inspection and testing of these bellows is limited to 60 months.

ASME Code Examination and Inspection (Subsection IWE)

TVA performs inspection activities on the containment structure that also support performance of the required Type A test. WBN performs containment inspections in accordance with the ASME Section XI Subsection IWE ISI program. The IWE program will continue to perform inspection activities on WBN Unit 1 containment through the proposed Appendix J test extension interval.

TVA's IWE program is based on the applicable portions of Subsections IWA and IWE of the 1992 Edition, Winter 1992 Addenda, of ASME Section XI. The first inspection interval for the containment ISI program began September 9, 1998 and ended September 8, 2001. The second inspection period ended September 8, 2005 and the third inspection period will end September 8, 2008, in accordance with ASME Section XI. The second inspection interval for containment will begin September 9, 2008.

Visual examinations of the Unit 1 SCV have been performed in accordance with the IWE program. To date, no indications of containment degradation have been found. These periodic IWE examinations provide assurance that degradation of the containment structure will be detected and corrected before it can affect structural integrity or leak tightness.

A general visual examination was performed on the Unit 1 SCV during the Cycle 3 and 6 Refueling Outages. These examinations were performed to meet the ASME Section XI, Subsection IWE, Table IWE-2500-1, Examination Category E-A, Item Number E1.11 requirements and the WBN TS 3.6.1.1 requirements. A general visual examination is required to be performed once per inspection period on the accessible exterior surface areas of the SCV per 10 CFR 50.55a (b)(2)(ix)(E). The TS general visual examination is performed on the accessible interior and exterior surface areas of the SCV prior to each 10 CFR 50, Appendix J, Type A, CILRT and during one other refueling outage if the Type A test has been extended to 10 years. There were no conditions identified during these general visual examinations that affected the leak tightness or structural adequacy of the SCV.

WBN issued an initial evaluation of potential areas for augmented containment inservice inspection (CISI) examination (areas likely to experience accelerated aging and degradation). This evaluation was updated based on completion of the WBN Unit 1, Cycle 3, general visual examination. This evaluation determined that there were no areas of WBN SCV which should be considered as requiring augmented examinations in accordance with IWE-1240 and IWE program.

Future Code Inspections

A VT-3 examination to meet ASME Section XI (i.e., Subsection IWE, Table IWE-2500-1, Examination Category E-A, Item Number E1.12) requirement to examine the accessible surface areas at the end of the interval from one side of the SCV is scheduled during the Unit 1 Cycle 8 Refueling Outage (third period of the first containment ISI interval).

The total estimated area of the SCV from the base concrete floor slab to the top of the SCV on the exterior side is approximately 61,300 square feet. The inaccessible exterior surface area is estimated to be approximately 2800 square feet due to insulation on the exterior SCV surface and the area around the fuel transfer penetration. It is estimated that 95 percent of the SCV exterior side is assessable for general visual and VT-3 visual examination in accordance with ASME Section XI, Examination Category E-A.

The area below the floor is not included in the area for examination because the embedded metal liner and concrete base slab are exempt from examination in accordance with IWE–1220(b) and IWL-1220(b) of Subsections IWE and IWL of ASME Section XI.

Additional Inspections

TVA is not proposing any additional IWE examination or non-destructive examinations of the WBN SCV based on the following:

- WBN has no areas identified for augmented examinations in accordance with IWE-1240.
- The WBN SCV general visual examinations performed in accordance with the ASME Section XI code did not identify any conditions that affected the leak tightness or structural adequacy of the SCV.

The WBN and SQN primary containment structures consist of a freestanding steel vessel with an ice condenser and a separate secondary containment that is a reinforced concrete Shield Building. The maximum internal pressure for WBN and SQN is 15.0 and 12.0 pounds per square inch, respectively. Review of the WBN and SQN SCV design drawings indicates that the WBN SCV is designed with greater wall thickness than the SQN SCV. This additional wall thickness for WBN SCV enhances the leak tightness or structural adequacy of the SCV.

Based on the comparison of WBN and SQN SCV, TVA has no plans to perform any additional inspections in inaccessible areas to validate integrity of the steel containment vessel.

Related Containment Relief Requests

- TVA's Request for Relief CISI-01 was approved by the NRC for Examination Category E-D, seals and gaskets [Reference 7]. TVA's CISI-01 included alternative requirements for ensuring leak tightness of seals and gaskets. Alternative leak testing is performed in accordance with 10 CFR Part 50, Appendix J (Type B testing). A Type B test is performed at least once each ISI interval as required by 10 CFR Part 50, Appendix J, and during each disassembly and re-assembly sequence. As identified in TVA's request for relief, there are no examinations of seals and gaskets which will be performed in accordance with Subsection IWE. The relief request CISI-01 allows Appendix J testing to be performed in lieu of code examinations. The extension of the CILRT interval from 10 to 15 years will not affect the frequency at which the seals and gaskets are tested for Appendix J. The provisions of Option B (10 CFR 50, Appendix J) allow extended test intervals up to 120 months for Type B components, based on acceptable performance. At WBN seals and gaskets are tested in accordance with 10 CFR Part 50, Appendix J, Option B and are Type B tested during a 60-month period for the full population. The seals and gaskets are tested on a staggered basis such that a portion is tested each refueling outage. Since Option B was first implemented at WBN (in Spring 1997 for Unit 1), seals and gaskets have been tested at least once and are undergoing their third round of testing on a staggered basis. In addition to the 60-month tests, testing is performed prior to and following disassembly of a containment penetration. Testing of seals and gaskets will also be performed as part of the CILRT (Type A test) at the end of the 5-year extended interval.

- TVA's Request for Relief CISI-02 was approved by the NRC [Reference 7] for a visual examination, VT-2, following repair, modification, or replacement of containments per Paragraph IWE-5240 of the 1992 Edition, 1992 Addenda, of ASME Section XI. VT-2 visual examinations are conducted to detect evidence of leakage from pressure retaining components during the conduct of a system pressure test.

  These VT-2 visual examinations are principally performed to locate water or steam leaks from pressure retaining components. VT-2 examination for evidence of air leakage does not provide effective detection of leakage. Table IWE-2500-1, Examination Category E-P, requires 10 CFR 50, Appendix J testing for all containment pressure retaining components. Appendix J provides requirements for testing as well as acceptable leakage criteria. Additionally, 10 CFR 50.55a(b)(2)(ix)(E) requires a general visual examination of the containment, as required by Subsection IWE, be performed each period. This examination would identify structural degradation that may contribute to leakage. Repairs and replacements, including modifications, must be performed in accordance with Article IWA-4000, which provides additional assurance of structural integrity of the containment. Performance of a VT-2 visual examination in addition to these requirements would not provide additional assurance of detection of containment pressure boundary leakage.

- TVA's Request for Relief CISI-03 was approved by the NRC [Reference 7]. Paragraphs IWE-2420(b) and IWE-2420(c) of the 1992 Edition, 1992 Addenda, of ASME Section XI require that when component examination results require evaluation of flaws, evaluation of areas of degradation, or repairs in accordance with Article IWE-3000, and the component is found to be acceptable for continued service, the areas containing such flaws, degradation, or repairs shall be reexamined during the next inspection period listed in the schedule of the inspection program of Paragraph IWE-2411 or Paragraph IWE-2412, in accordance with Table IWE-2500-1, Examination Category E-C.

  The purpose of a repair is to restore the component to an acceptable condition for continued service. Furthermore, if the repair area is subject to accelerated degradation, the area would require augmented examination in accordance with Table IWE-2500-1, Examination Category E-C. The successive examination of repaired areas in accordance with Paragraphs IWE-2420(b) and IWE-2420(c) is a burden without a compensating increase in safety or quality. This requirement has been removed in the 1988 Edition of ASME Section XI.

  Paragraphs IWB-2420(b), IWC-2420(b), and IWD-2420(b) for Class 1, 2, and 3 components, respectively, do not require a repaired component be subject to successive examination requirements. Additionally, the requirement to perform successive examinations of repaired areas in accordance with Paragraphs IWE-2420(b) and IWE-2420(c) has been removed in the 1998 Edition of ASME Section XI.

- TVA's Request for Relief CISI-04 was approved by the NRC [Reference 7] for Examination Category E—G, bolting. TVA's CISI-04 pertained to bolt torque and tension tests (Item No. E8.20). CISI—04 was approved to waive performance of bolt torque and tension tests for bolted connections that have not been disassembled and reassembled during the inspection interval. The VT-1 visual examinations required by Item No. E8.10 of Examination Category E-G, will continue to be performed. Examinations required by Item No. E8.10 were not deferred during the first period.

- TVA's Request for Relief CISI-05 was approved by the NRC [Reference 7] for the augmented examinations of Table IWE-2500-1, Examination Category E-C, which are to be performed on containment surface areas likely to experience accelerated degradation and aging. Subarticle IWE-2500(c)(1) and (2) requires areas subject to augmented examinations be VT-1 visually examined from both sides or be examined for wall thinning using an Ultrasonic (UT) thickness measurement method. Subarticle IWE-2500(c)(3) requires augmented UT thickness measurements be taken in one foot square grids. Subarticle IWE-2500(c)(4) requires the minimum wall thickness within each grid be determined and marked such that periodic reexamination of that location can be performed. This provides for monitoring of a point that may not be the most susceptible to accelerated degradation and requires taking numerous ultrasonic thickness readings within a grid which may not have exhibited degradation.

Code Case N-605, "*Alternative to the Requirements of IWE-2500(c) for Augmented Examination of Surface Areas,*" provides for UT thickness measurements to be taken at grid line intersections. Code Case N-605 also permits variations in grid line spacing, provides a sampling plan for performing the UT thickness measurements, and provides a plan for sample expansion for areas exhibiting degradation. Code Case N-605 incorporates the requirements of Sub article IWE-2500(c)(1) for performance of VT-1 visual examinations.

Taking numerous UT thickness readings within a grid which had not exhibited degradation, results in hardship or unusual difficulty without a compensating increase in the level of quality and safety. The use of Code Case N-605 to determine examination requirements for VT-1 visual examinations and UT thickness measurements of areas requiring augmented examinations provides for an acceptable level of quality and safety. The requirements of Code Case N-605 have been incorporated in the 1998 Edition of ASME Section XI.

5.0 REGULATORY SAFETY ANALYSIS

TVA's proposed revision to the technical specifications (TSs) for Watts Bar Nuclear Plant (WBN) Unit 1, adds a notation to Section 5.7.2.19, "Containment Leakage Rate Testing Program," to allow a 1–time, 5–year extension to the current 10-year interval for 10 CFR 50, Appendix J, Type A testing.

5.1 No Significant Hazards Consideration

TVA has evaluated whether or not a significant hazard consideration is involved with the proposed amendment by focusing on the three standards set forth in 10 CFR 50.92, "*Issuance of Amendment,*" as discussed below:

1. **Does the proposed change involve a significant increase in the probability or consequences of an accident previously evaluated?**

Response: No.

The proposed change for extending Type A test frequency does not significantly increase the probability of an accident previously evaluated since the change is not a modification to plant systems, nor a change to plant operation that could initiate an accident.

TVA performed an evaluation of the risk significance for the proposed increase to the WBN Unit 1 Type A test frequency. The results of the TVA risk evaluation indicates that the increase in Large Early Release Frequency (LERF) remains below the level of risk significance defined in the NRC Regulatory Guide 1.174, "*An Approach for Using Probabilistic Risk Assessment In Risk-Informed Decisions On Plant-Specific Changes to the Licensing Basis.*" TVA's evaluation indicates that the calculated increase in frequency for all releases (small, large, early and late) and the increase in radiation dose to the population are also non-risk significant.

The proposed test interval extension does not involve a significant increase in the consequences of an accident. Research documented in NUREG-1493, "*Performanced-Based Containment Leakage-Test Program,*" determined that generically, very few potential containment leakage paths fail to be identified by Type A tests. An analysis of 144 Type A test results, including 23 failures, found that no failures were due to containment liner breach. The NUREG concluded that reducing the Type A test frequency to once per 20 years would lead to an imperceptible increase in risk. Furthermore, the NUREG concluded that Type B and C testing provides assurance that containment leakage from penetration leak paths (i.e., valves, flanges, containment air-locks) identify any leakage that would otherwise be detected by the Type A tests.

In addition to the NUREG conclusions, TVA's American Society of Mechanical Engineers (ASME) IWE program performs containment inspections in order to detect evidence of degradation that may affect either the containment structural integrity or leak tightness.

Therefore, the proposed extension of the Type A test interval does not involve a significant increase in the probability or consequences of an accident previously evaluated.

2. **Does the proposed change create the possibility of a new or different kind of accident from any accident previously evaluated?**

Response: No.

The proposed change to extend the Type A test interval does not create the possibility of a new or different type of accident because there are no physical changes made to the plant or plant equipment governing normal plant operation. There are no changes to the operation of the plant that would introduce a new failure mode creating the possibility of a new or different kind of accident. Therefore, the proposed extension does not create the possibility of a new or different kind of accident from any previously evaluated.

3. **Does the proposed change involve a significant reduction in a margin of safety?**

Response: No.

The proposed change to extend the Type A test interval will not significantly reduce the margin of safety. A generic study documented in NUREG-1493 indicates that extending the Type A leak test interval to 20 years would result in an imperceptible increase in risk to the public. The NUREG also found that, generically, the containment leakage rate contributes a very small amount to the individual risk and that the decrease in the Type A test frequency would have a minimal affect on risk because most potential leakage paths are detected by Type C testing.

Previous Type A leakage tests conducted on WBN Unit 1 indicate that leakage from containment have been less than the 10 CFR 50, Appendix J leakage limit of 1.0 L_a. A review of the previous Type A test results indicate a stable trend with an increase of less than 15 percent of L_a, well below the 1.0 L_a leakage limit.

Therefore, these test results, in conjunction with the research findings from NUREG-1493, provide assurance that the proposed extension to the Type A test interval does not involve a significant reduction in a margin of safety.

Based on the above, TVA concludes that the proposed amendment presents no significant hazards consideration under the standards set forth in 10 CFR 50.92(c), and accordingly, a finding of "no significant hazards consideration" is justified.

5.2 Applicable Regulatory Requirements/Criteria

Title 10 of the Code of Federal Regulations, Part 50 (10 CFR 50), Section 50.54(o) and 10 CFR 50, Appendix J, contain primary reactor containment leakage test requirements for water-cooled power reactors. The 10 CFR 50, Appendix J requirements are divided into Option A (prescriptive requirements) and Option B (performance-based requirements) The Option B rulemaking in 1995 provided licensees with an alternative approach to determine test intervals for containment leakage rate testing. The Option B approach was based on system and component performance in lieu of compliance with prescriptive requirements of the NRC REGULATORY GUIDE 1.163, [Reference 1] was developed as a method acceptable to the NRC staff for implementing Option B. This regulatory guide endorses, with certain exceptions, NEI 94-01, Revision 0, [Reference 2]. A Type A test is an overall (integrated) leak rate test of the containment structure. NEI 94—01 specifies an initial test interval of 48 months, but allows an extended interval of 10 years, based upon two consecutive successful tests. TVA submitted a license amendment to request use of Option B at WBN that was subsequently, approved by the NRC letter dated May 27, 1997 [Reference 4].

The WBN TSs (Section 5.7.2.19) currently contain program requirements for implementing leak rate testing of containment as required by 10 CFR 50.54(o) and 10 CFR 50, Appendix J, Option B, as modified by approved exemptions. Section 5.7.2.19 further states that this program shall be in accordance with the guidelines contained in REGULATORY GUIDE 1.163. Accordingly, the WBN Type A test interval is once per 10 years.

The proposed change is submitted on a risk informed basis as described in Regulatory Guide 1.174 [Reference 6]. TVA has determined that the resultant increase in LERF for the proposed change is "very small" and satisfies the Regulatory Guide 1.174 criteria.

TVA's ASME IWE program performs containment inspections in order to detect evidence of degradation that may affect either the containment structural integrity or the leak tightness. In conclusion, based on the considerations discussed above: (1) there is reasonable assurance that the health and safety of the public will not be endangered by operation in the proposed manner, (2) such activities will be conducted in compliance with the Commission's regulations, and (3) the issuance of the amendment will not be inimical to the common defense and security or to the health and safety of the public.

6.0 ENVIRONMENTAL CONSIDERATION

A review has determined that the proposed amendment would change a requirement with respect to installation or use of a facility component located within the restricted area, as defined in 10 CFR 20, or would change an inspection or surveillance requirement. However, the proposed amendment does not involve (i) a significant hazards consideration, (ii) a significant change in the types or significant increase in the amounts of any effluent that may be released offsite, or (iii) a significant increase in individual or cumulative occupational radiation exposure. Accordingly, the proposed amendment meets the eligibility criterion for categorical exclusion set forth in 10 CFR 51.22(c)(9). Therefore, pursuant to 10 CFR 51.22(b), no environmental impact statement or environmental assessment need be prepared in connection with the proposed amendment.

7.0 REFERENCES

1. Regulatory Guide 1.163 "Performance-Based Containment Leak Test Program," dated September 1995.

2. Nuclear Energy Institutes (NEI) 94-01, "Industry Guideline for Implementing Performance Based Option of 10 CFR 50, Appendix J," Revision 0, dated July 26, 1995.

3. NRC letter to TVA dated May 29, 2003, "Sequoyah Nuclear Plant, Units 1 and 2 - Issuance of Amendments Regarding Risk Informed Integrated Leak Rate Testing Extension (TAC Nos. MB 6987 and MB 6988).

4. NRC letter to TVA dated May 27, 1997, Watts Bar Nuclear Plant Unit 1 - Issuance of Amendment - Implementation of 10 CFR 50, Appendix J, Option B, Performance-Based Containment Leakage Testing, (TAC No. M97698).

5. TVA Calculation "MDN001-999-2005-0099, Revision 1, "Evaluation of the Risk Significance of Decreased Containment Integrated Leak Rate Test frequency," (Enclosure 4).

6. Regulatory Guide 1.174, "An Approach for Using Probabilistic Risk Assessment in Risk-Informed Decisions On Plant-Specific Changes to the Licensing Basis.

7. NRC letter to TVA dated November 24, 1999, "Relief from ASME Code Requirements for Containment Related Examinations and Tests at Watts Bar Nuclear Plant (TAC No. MA6069).

ENCLOSURE 2

WATTS BAR PLANT (WBN) UNIT 1
PROPOSED TECHNICAL SPECIFICATION (TS) CHANGE
MARKED PAGES

I. AFFECTED PAGE LIST
5.0-28

II. MARKED PAGES
See attached.

5.7 Procedures, Programs, and Manuals

5.7.2.18 Safety Function Determination Program (SFDP) (continued)

A loss of safety function exists when, assuming no concurrent single failure, a safety function assumed in the accident analysis cannot be performed. For the purpose of this program, a loss of safety function may exist when a support system is inoperable, and:

a. A required system redundant to the system(s) supported by the inoperable support system is also inoperable; or

b. A required system redundant to the system(s) in turn supported by the inoperable supported system is also inoperable; or

c. A required system redundant to the support system(s) for the supported systems (a) and (b) above is also inoperable.

The SFDP identifies where a loss of safety function exists. If a loss of safety function is determined to exist by this program, the appropriate Conditions and Required Actions of the LCO in which the loss of safety function exists are required to be entered.

5.7.2.19 Containment Leakage Rate Testing Program

A program shall be established to implement the leakage rate testing of the containment as required by 10 CFR 50.54(o) and 10 CFR 50 Appendix J, Option B, as modified by approved exemptions. This program shall be in accordance with the guidelines contained in Regulatory Guide (RG) 1.163, "Performance-Based Containment Leak-Test Program," dated September 1995.

The peak calculated containment internal pressure for the design basis loss of coolant accident, P_a, is 15.0 psig.

The maximum allowable containment leakage rate, L_a at P_a, is 0.25% of the primary containment air weight per day.

"The Fall 2007 end date for conducting the 10 year interval containment integrated leakage rate (Type A) test may be deferred up to 5 years but no later than Fall 2012."

(continued)

ENCLOSURE 3

WATTS BAR PLANT (WBN) UNIT 1
PROPOSED TECHNICAL SPECIFICATION (TS) CHANGE
RETYPED PAGES

I. **AFFECTED PAGE LIST**
5.0-28

II. **MARKED PAGES**
See attached.

5.7 Procedures, Programs, and Manuals

5.7.2.18 Safety Function Determination Program (SFDP) (continued)

A loss of safety function exists when, assuming no concurrent single failure, a safety function assumed in the accident analysis cannot be performed. For the purpose of this program, a loss of safety function may exist when a support system is inoperable, and:

a. A required system redundant to the system(s) supported by the inoperable support system is also inoperable; or

b. A required system redundant to the system(s) in turn supported by the inoperable supported system is also inoperable; or

c. A required system redundant to the support system(s) for the supported systems (a) and (b) above is also inoperable.

The SFDP identifies where a loss of safety function exists. If a loss of safety function is determined to exist by this program, the appropriate Conditions and Required Actions of the LCO in which the loss of safety function exists are required to be entered.

5.7.2.19 Containment Leakage Rate Testing Program

A program shall be established to implement the leakage rate testing of the containment as required by 10 CFR 50.54(o) and 10 CFR 50 Appendix J, Option B, as modified by approved exemptions. This program shall be in accordance with the guidelines contained in Regulatory Guide (RG) 1.163, "Performance-Based Containment Leak-Test Program," dated September 1995.

The Fall 2007 end date for conducting the 10 year interval containment integrated leakage rate (Type A) test may be deferred up to 5 years but no later than Fall 2012.

The peak calculated containment internal pressure for the design basis loss of coolant accident, P_a, is 15.0 psig.

The maximum allowable containment leakage rate, L_a at P_a, is 0.25% of the primary containment air weight per day.

(continued)

ENCLOSURE 4

WATTS BAR PLANT (WBN) UNIT 1
EVALUATION OF THE RISK SIGNIFICANCE OF DECREASED CONTAINMENT
INTEGRATED LEAK RATE TEST FREQUENCY
MDN001-999-2005-0099, REVISION 1

TVAN CALCULATION COVERSHEET/CCRIS UPDATE

REV 0 EDMS/RIMS NO.	EDMS TYPE:	EDMS ACCESSION NO (N/A for REV. 0)
T71051101801	calculations(nuclear)	T69 051121 054

Calc Title: EVALUATION OF THE RISK SIGNIFICANCE OF DECREASED CONTAINMENT INTEGRATED LEAK RATE TEST FREQUENCY

CALC ID	TYPE	ORG	PLANT	BRANCH	NUMBER	CUR REV	NEW REV	
CURRENT	CN	NUC	WBN	MEB	MDN00199920050099	0	1	REVISION APPLICABILITY
NEW								Entire calc ☒ Selected pages ☐

ACTION	NEW ☐ REVISION ☒	DELETE ☐ RENAME ☐	SUPERSEDE ☐ DUPLICATE ☐	CCRIS UPDATE ONLY ☐ (Verifier Approval Signatures Not Required)	No CCRIS Changes ☐ (For calc revision, CCRIS been reviewed and no CCRIS changes required)

UNITS	SYSTEMS	UNIDS
1	999	N/A

DCN,EDC,N/A	APPLICABLE DESIGN DOCUMENT(S)	CLASSIFICATION
N/A	N/A	D

QUALITY RELATED?	SAFETY RELATED? (If yes, QR = yes)	UNVERIFIED ASSUMPTION	SPECIAL REQUIREMENTS AND/OR LIMITING CONDITIONS?	DESIGN OUTPUT ATTACHMENT?	SAR/TS and/or ISFSI SAR/CoC AFFECTED
Yes ☐ No ☒	Yes ☐ No ☒	Yes ☐ No ☒	Yes ☐ No ☒	Yes ☐ No ☒	Yes ☒ No ☐

PREPARER ID	PREPARER PHONE NO	PREPARING ORG (BRANCH)	VERIFICATION METHOD	NEW METHOD OF ANALYSIS
XB?YZJ024	(423) 751-4124	MEB	N/A	☐ Yes ☒ No

PREPARER SIGNATURE	DATE	CHECKER SIGNATURE	DATE
Calvin A. McCullough [signature]	11/21/05	William Z. Mims [signature]	11-21-05
VERIFIER SIGNATURE	**DATE**	**APPROVAL SIGNATURE**	**DATE**
N/A		[signature] for CRA	11-21-05

STATEMENT OF PROBLEM/ABSTRACT

This calculation determines the effect on release frequency and population dose as a result of a decrease in the frequency of performing containment integrated leak rate testing (ILRT).

The effect of a decrease in the frequency of performing an ILRT is that the exposure time to a pre-existing leak in the containment shell increases. This results in an increase in the calculated frequency of both large and small fission product releases to the environment which correlates to a calculated increase in population dose. Revision 3 of the PSA is used to determine the increase in a) the frequency of large releases, b) the frequency of all (large and small) releases, c) the conditional containment failure probability, and d) the increase in population dose due to decreased ILRT frequencies of between 3/10 years and 1/20 years.

The numerical results of this calculation are provided in Section 7.

The increase in LERF is very small per RG 1.174, and therefore acceptable. In addition, the increase in all releases, the conditional containment failure probability, and the population dose are not risk significant.

MICROFICHE/EFICHE Yes ☐ No ☐ FICHE NUMBER(S)

☐ LOAD INTO EDMS AND DESTROY
☒ LOAD INTO EDMS AND RETURN CALCULATION TO CALCULATION LIBRARY. ADDRESS. EQB 1A-WBN
☐ LOAD INTO EDMS AND RETURN CALCULATION TO:

TOTAL P.01

TVAN CALCULATION COVERSHEET/CCRIS UPDATE

CALC ID	TYPE	PLANT	BRANCH	NUMBER	REV
	CN	SQN	MEB	MDN00199920050099	1

BLDG	ROOM	ELEV	COORD/AZIM	FIRM	Print Report Yes ☐
N/A	N/A	N/A	N/A	TVA	
CATEGORIES N/A					

KEY NOUNS (A-add, D-delete)

ACTION (A/D)	KEY NOUN	A/D	KEY NOUN

CROSS-REFERENCES (A-add, C-change, D-delete)

ACTION (A/C/D)	XREF CODE	XREF TYPE	XREF PLANT	XREF BRANCH	XREF NUMBER	XREF REV

CCRIS ONLY UPDATES:
Following are required only when making keyword/cross reference CCRIS updates and page 1 of form NEDP-2-1 is not included:

PREPARER SIGNATURE	DATE	CHECKER SIGNATURE	DATE
N/A	N/A	N/A	N/A
PREPARER PHONE NO. N/A		EDMS ACCESSION NO. N/A	

TVAN CALCULATION RECORD OF REVISION	
CALCULATION IDENTIFIER: MDN001-999-2005-0099	
Title	EVALUATION OF THE RISK SIGNIFICANCE OF DECREASED CONTAINMENT INTEGRATED LEAK RATE TEST FREQUENCY
Revision No.	DESCRIPTION OF REVISION
0	Initial Issue The Living SAR has been reviewed by Calvin A. McCullough and this revision of the calculation does not affect the SAR. This calculation supports a proposed change to Technical Specification paragraph 5.7.2.19, "Containment Leakage Rate Testing Program." Total Pages = 31
1	Revised to incorporate WBN Site Comments. The Living SAR has been reviewed by Calvin A. McCullough and this revision of the calculation does not affect the SAR. This calculation supports a proposed change to Technical Specification paragraph 5.7.2.19, "Containment Leakage Rate Testing Program." Pages revised: 1, 2, 3, 6, 7, 25. Total Pages = 32

TVAN COMPUTER INPUT FILE STORAGE INFORMATION SHEET

Document MDN001-999-2005-0099	Rev. 1	Plant: WBN	

Subject:

EVALUATION OF THE RISK SIGNIFICANCE OF DECREASED CONTAINMENT INTEGRATED LEAK RATE TEST FREQUENCY

☐ Electronic storage of the input files for this calculation is not required. Comments:

☒ Input files for this calculation have been stored electronically and sufficient identifying information is provided below for each input file. (Any retrieved file requires re-verification of its contents before use.)

Spreadsheet (Microsoft Excel) and Riskman files are stored by Filekeeper.

File Name: mdn-001-999-2005-099.zip

Reference ID: 307560

☐ Microfiche/Fiche

TVAN CALCULATION TABLE OF CONTENTS

Calculation Identifier: MDN001-999-2005-0099	Revision: 1

TABLE OF CONTENTS

1.0 Purpose

The purpose of this calculation is to determine the risk significance of a decrease in ILRT frequency. The effect of a decrease in the frequency of performing an ILRT is that the exposure time of a pre-existing leak in the containment shell increases. This results in an increase in the calculated frequency of both large and small fission product releases to the environment which correlates to an increase in calculated population dose. This calculation quantifies the increase in release frequency and population dose as a result of a decrease in the frequency of performing an ILRT.

2.0 References

1. *Sequoyah Nuclear Plant Probabilistic Safety Assessment*, Revision 1 Report, (B38 960806800).

2. NUREG-1493, *Performance-Based Containment Leak-Test Program*, September, 1995.

3. SQNP Probabilistic Risk Assessment, Revision 3, Level II Model SQNR3L2.

4. NUREG/CR-4551, Volume 5, Revision 1, Part 1, *Evaluation of Severe Accident Risks: Sequoyah, Unit 1*, December, 1990.

5. Regulatory Guide 1.174, *An Approach for Using Probabilistic Risk Assessment In Risk-Informed Decisions On Plant-Specific Changes to the Licensing Basis.*

6. Watts Nuclear Plant - Generic Letter 88-20 Supplements 4 and 5, *Individual Plant Examinations of External Events (IPEEE) for Severe Accident Vulnerabilities* (T04 980217 539).

7. Regulatory Guide 1.177, *An Approach for Plant-Specific, Risk-Informed Decisionmaking: Technical Specifications.*

8. *Staff Evaluation Report of the Individual Plant Examinations of External Events (IPEEE) Submittal on Watts Bar Nuclear Plant, Unit 1* (L44 000530 002).

9. TVA Calculation CN-NUC-SQN-NTB-SQS20211, R1, Evaluation of the Risk Significance of Decreased Containment Integrated Leak Rate Test Frequency.

10. ERIN letter and report, *Value Impact Analysis of Potential Plant Enhancements for Watts Bar Nuclear Plant* (T25 940630 838).

11. WBNP Probabilistic Risk Assessment, Revision 3.

12. SQN Technical Specifications.

13. WBNP Safety Analysis Report.

14. SQNP Safety Analysis Report.

3.0 Design Input Data

Refer to Table 7.

4.0 Assumptions

Revision 3 of the WBN PSA is assumed to be of sufficient quality to support the analysis and conclusions of this calculation.
Justification – Previous revisions of the WBN PSA have been successfully used for risk-informed applications. Revision 3 is an improvement over revision 2. Draft Revision 3 was peer-reviewed by the Westinghouse Owners Group (WOG).

Reference 3, the SQN Containment Event Tree (CET) model was used to map KPDSs to KRCs.
Justification – The design of SQN and WBN are very similar; the containments are expected to have the same failure modes and failure probabilities.

It is assumed that the Large Early Release Frequency (LERF) associated with both internal and external initiating events can be estimated by doubling the LERF associated with only internal events.
Justification – This is a general rule of thumb used for many risk-informed PSA applications (e.g., Severe Accident Mitigation Alternatives Analysis such as reference 10) and has been adopted by the NRC in lieu of a plant-specific external events PSA.

5.0 Requirements/Limiting Conditions

None.

6.0 Computations and Analyses

6.1 Effect of ILRT Frequency on the Probability of a Preexisting Containment Leak

The effect of a decrease in the frequency of performing an ILRT is that the exposure time to a preexisting leak in the containment shell increases, resulting in an increase in the calculated probability of containment failure. The fault tree for small and large containment isolation failures used in the PSA (from reference 11) accounts for the following failures to isolate containment:

- a failure of instrumentation to generate a containment isolation signal along with failure of the operator to manually initiate this action,
- a containment penetration failing to isolate as the result of the failure of the inboard and outboard isolation valves to close or
- the existence of a preexisting leak in the containment.

The first two containment isolation failures listed above are identified by ESFAS testing or stroke testing containment isolation valves, respectively. The existence of a leak in a containment penetration is identified by either a local leak rate test (LLRT) or an integrated leak rate test (ILRT). The existence of a leak in the containment shell is identified by an ILRT. The decrease in the frequency of conducting ILRTs increases the calculated probability of a preexisting leak in containment, but does not affect the probability of the other containment isolation failure mechanisms listed above.

For a component that does not change state, the failure probability of the component (Q) is given by:

$$Q = \lambda * (T/2 + TM)$$

where,

- λ = the failure rate
- T = the test interval and
- TM = the PSA mission time

Since T >>TM, the failure probability for a preexisting containment leak is approximately:

$$Q = \lambda T/2$$

As discussed above, the existence of a leak in a containment penetration is identified by either a LLRT or an ILRT. The probability of a preexisting leak in a containment penetration can be rewritten as:

$$Q_p = \lambda_p\{P(LLRT)T_{LLRT} + P(ILRT/\overline{LLRT})T_{ILRT}\}/2$$

where,

Q_p = the probability of a preexisting leak in a containment penetration,

λ_p = the rate of occurrence of a containment penetration leak,
P(LLRT) = the probability of detecting a preexisting containment penetration leak with a LLRT,
T_{LLRT} = the test interval for the LLRT,
$P(ILRT/\overline{LLRT})$ = the probability of detecting a preexisting leak with an ILRT given it was not detected with a LLRT and
T_{ILRT} = the test interval for the ILRT

As described in reference 2, LLRTs are performed prior to the ILRT so detecting a preexisting containment penetration leak during an ILRT is contingent upon not detecting it during a LLRT. Since all preexisting containment penetration leaks are detected by either a LLRT or a subsequent ILRT it follows that:

$$P(LLRT) + P(ILRT/\overline{LLRT}) = 1.0$$

Reference 2 determined that: $P(ILRT/\overline{LLRT}) = 0.03$; and therefore, P(LLRT) = 0.97 so the probability of a preexisting containment penetration leak is given by:

$$Q_p = \lambda_p\{0.97T_{LLRT} + 0.03T_{ILRT}\}/2$$

As discussed above, the existence of a leak in the containment shell is only identified by an ILRT. The probability of a preexisting leak in the containment shell can be written as:

$$Q_I = \lambda_I T_{ILRT}/2$$

where,

Q_I = the probability of a preexisting leak in the containment shell and
λ_I = the rate of occurrence of a containment shell leak

Therefore, the probability of a preexisting containment leak, Q, is equal to:

$$Q = \lambda_p\{0.97T_{LLRT} + 0.03T_{ILRT}\}/2 + \lambda_I T_{ILRT}/2$$

The remaining parameters in the above equation are the λ's - the failure rates. The failure rate of the containment shell is expected to be comparable to the failure rate of a storage tank rupture. The storage tank rupture failure rate distribution used in the PSA (reference 11) has a mean failure rate of 2.66E-8/hr (ZTTK1B) or 1.94E-05/month.

Previous analyses (reference 9) have used values of Q of 0.064 for preexisting small containment leaks and 0.021 for preexisting large containment leaks. These failure probabilities correspond to a 3-in-10 year ILRT test frequency and all containment penetrations being subjected to a LLRT once per refueling cycle. Using a LLRT interval of 18 months and a ILRT test interval of 40 months (3-in-10 years), the values λ_p are calculated from the below equations:

small: $0.064 = \lambda_p\{0.97T_{LLRT} + 0.03T_{ILRT}\}/2 + (1 - 0.021/0.064)(1.94E\text{-}05)T_{ILRT}/2 \Rightarrow \lambda_p = 6.83E\text{-}03$

large: $0.021 = \lambda_p\{0.97T_{ILRT} + 0.03T_{ILRT}\}/2 + (0.021/0.064)(1.94E\text{-}05)T_{ILRT}/2 \Rightarrow \lambda_p = 2.24E\text{-}03$

The increase in the probability of a preexisting small and large containment leak is given in Table 1 as the ILRT test interval is varied from 40 months to 20 years:

Table 1

ILRT Test Interval (months)	Relative Probability of a pre-existing small leak (basis 0.064)	Relative Probability of a pre-existing large leak (basis 0.021)
40	1.00	1.00
120	1.14	1.14
180	1.24	1.24
240	1.34	1.35

6.2 Effect of a Preexisting Containment Leak on Releases

The level I portion of the PSA (reference 11) determines the frequency of accident scenarios or sequences which result in damage to the core. In addition, the level I portion of the PSA determines the state or condition of the plant for the sequences which result in core damage (CD). Key information about the state of the plant determined for each CD sequence is:

- RCS pressure,
- the availability of secondary heat removal,
- if the RWST has been injected into containment,
- the availability of containment sprays and
- if the containment is isolated or bypassed.

The above described key information results in various combinations of plant conditions (referred to as plant damage states, PDSs). Every CD sequence which has a frequency greater than a selected frequency is assigned to a PDS. The sum of the frequencies of all CD sequences assigned to a given PDS yields the frequency of the PDS. The level I portion of the PSA reports the frequency of PDSs which have a frequency greater than 1.0E-11.

The key information from a PDS for determining the effect of the increased probability of a containment leak is the state of containment. The PDSs characterize the containment as being either intact, having a small or large isolation failure (hole) or as being bypassed by a small or large leak (SGTR or ISLOCA). This information is used to characterize the fission product release from containment and is presented in Table 2.

The PDSs can be thought of as initiating events to the level II portion of the PSA. Rather than analyzing all 76 PDSs in the level II portion of the PSA, the PDSs are combined into 17 Key Plant Damage States (KPDSs) as summarized in Table 3.

Table 2

PDS	Frequency	Intact (Note 1)		small bypass (Note 2)		large bypass (Note 3)		small isolation failures (Note 4)		large isolation failures (Note 5)	
		analyzed in Level II (Note 6)	not analyzed in Level II (Note 7)	analyzed in Level II (Note 6)	not analyzed in Level II (Note 7)	analyzed in Level II (Note 6)	not analyzed in Level II (Note 7)	analyzed in Level II (Note 6)	not analyzed in Level II (Note 7)	analyzed in Level II (Note 6)	not analyzed in Level II (Note 7)
FCI	8.11E-06	8.11E-06									
ENI	1.57E-06	1.57E-06									
HCI	8.59E-07	8.59E-07									
LCI	7.58E-07	7.58E-07									
GNI	6.59E-07	6.59E-07									
ECB	4.29E-07				4.29E-07						
BCI	3.78E-07	3.78E-07									
FGI	2.19E-07	2.19E-07									
FNI	1.68E-07	1.68E-07									
ENS	1.25E-07							1.25E-07			
ENB	7.36E-08			7.36E-08							
FCB	7.16E-08			7.16E-08							
EGI	6.78E-08	6.78E-08									
ATV	5.36E-08					5.36E-08					
GNS	4.90E-08							4.90E-08			
DCI	4.58E-08	4.58E-08									
FCS	3.82E-08							3.82E-08			
HCB	3.75E-08				3.75E-08						
HNI	2.22E-08	2.22E-08									
EIB	1.32E-08			1.32E-08							
FPL	1.03E-08									1.03E-08	
ETL	1.03E-08									1.03E-08	

Table 2

PDS	Frequency	Intact (Note 1)		small bypass (Note 2)		large bypass (Note 3)		small isolation failures (Note 4)		large isolation failures (Note 5)	
		analyzed in Level II (Note 6)	not analyzed in Level II (Note 7)	analyzed in Level II (Note 6)	not analyzed in Level II (Note 7)	analyzed in Level II (Note 6)	not analyzed in Level II (Note 7)	analyzed in Level II (Note 6)	not analyzed in Level II (Note 7)	analyzed in Level II (Note 6)	not analyzed in Level II (Note 7)
HGI	9.94E-09		9.94E-09								
EEB	8.34E-09				8.34E-09						
KGI	5.62E-09		5.62E-09								
EGB	4.70E-09				4.70E-09						
FEI	3.64E-09		3.64E-09								
HCS	3.54E-09								3.54E-09		
LCS	3.17E-09								3.17E-09		
AGI	2.39E-09		2.39E-09								
GGI	2.20E-09		2.20E-09								
KNI	1.94E-09		1.94E-09								
LEI	1.77E-09		1.77E-09								
BCS	1.72E-09								1.72E-09		
LNI	1.45E-09	1.45E-09									
FII	1.22E-09		1.22E-09								
LGI	9.23E-10		9.23E-10								
FGS	9.22E-10								9.22E-10		
GTL	9.00E-10									9.00E-10	
BEI	8.63E-10		8.63E-10								
GNB	8.46E-10				8.46E-10						
HPL	8.09E-10										8.09E-10
LPL	7.75E-10										7.75E-10
BGI	7.66E-10		7.66E-10								
FNS	7.58E-10							7.58E-10			

Table 2

PDS	Frequency	Intact (Note 1)		small bypass (Note 2)		large bypass (Note 3)		small isolation failures (Note 4)		large isolation failures (Note 5)	
		analyzed in Level II (Note 6)	not analyzed in Level II (Note 7)	analyzed in Level II (Note 6)	not analyzed in Level II (Note 7)	analyzed in Level II (Note 6)	not analyzed in Level II (Note 7)	analyzed in Level II (Note 6)	not analyzed in Level II (Note 7)	analyzed in Level II (Note 6)	not analyzed in Level II (Note 7)
LII	5.96E-10		5.96E-10								
HEI	4.97E-10		4.97E-10								
BPL	4.64E-10										4.64E-10
FNB	4.11E-10				4.11E-10						
GCB	3.00E-10				3.00E-10						
BII	2.93E-10		2.93E-10								
EGS	2.79E-10								2.79E-10		
DGI	2.56E-10		2.56E-10								
FGB	2.44E-10				2.44E-10						
GGB	2.31E-10				2.31E-10						
FRL	2.10E-10										2.10E-10
DCS	2.01E-10								2.01E-10		
HII	1.7E-10		1.70E-10								
CNS	1.42E-10								1.42E-10		
FTL	1.34E-10										1.34E-10
FEB	1.31E-10				1.31E-10						
KNS	1.16E-10								1.16E-10		
BNI	1.07E-10		1.07E-10								
HNS	1.02E-10								1.02E-10		
CNI	7.83E-11		7.83E-11								
DNI	7.22E-11		7.22E-11								
ERL	7.13E-11										7.13E-11
HEB	6.61E-11				6.61E-11						

Table 2

PDS	Frequency	Intact (Note 1)		small bypass (Note 2)		large bypass (Note 3)		small isolation failures (Note 4)		large isolation failures (Note 5)	
		analyzed in Level II (Note 6)	not analyzed in Level II (Note 7)	analyzed in Level II (Note 6)	not analyzed in Level II (Note 7)	analyzed in Level II (Note 6)	not analyzed in Level II (Note 7)	analyzed in Level II (Note 6)	not analyzed in Level II (Note 7)	analyzed in Level II (Note 6)	not analyzed in Level II (Note 7)
DPL	4.15E-11										4.15E-11
FIB	3.72E-11				3.72E-11						
HGS	2.41E-11								2.41E-11		
HIB	2.08E-11				2.08E-11						
HNB	1.85E-11				1.85E-11						
CTL	1.76E-11										1.76E-11
ANS	1.44E-11								1.44E-11		
HTL	1.38E-11									1.38E-11	
sums	1.38E-05	1.29E-05	3.33E-08	1.58E-07	4.82E-07	5.36E-08	0.00E+00	2.13E-07	1.02E-08	2.15E-08	2.52E-09
CDF	1.38E-05	Total Intact (Note 8)	1.29E-05	Total small bypass (Note 8)	6.40E-07	Total large bypass (Note 8)	5.36E-08	Total small Isolation Failure & small preexistin g leaks (Note 8)	2.23E-07	Total large Isolation failure & large preexisting leaks (Note 8)	2.40E-08
								small isolation failures (Note 9)	1.71E-07	large isolation failures (Note 10)	4.07E-09

Table 2

PDS	Frequency	Intact (Note 1)		small bypass (Note 2)		large bypass (Note 3)		small isolation failures (Note 4)		large isolation failures (Note 5)	
		analyzed in Level II (Note 6)	not analyzed in Level II (Note 7)	analyzed in Level II (Note 6)	not analyzed in Level II (Note 7)	analyzed in Level II (Note 6)	not analyzed in Level II (Note 7)	analyzed in Level II (Note 6)	not analyzed in Level II (Note 7)	analyzed in Level II (Note 6)	not analyzed in Level II (Note 7)
								small preexisting leaks (Note 11)	8.85E-07	large preexisting leaks (Note 12)	2.90E-07

Note 1: PDSs which end in I.
Note 2: PDSs which end in B.
Note 3: PDSs which end in V.
Note 4: PDSs which end in S.
Note 5: PDSs which end in L.
Note 6: These are the PDSs which are evaluated in the Level II portion of the PSA.
Note 7: These PDSs are not evaluated in the Level II portion of the PSA, but are used in this analysis for characterizing releases from containment.
Note 8: Sum of the PDSs analyzed in Level II and not analyzed in Level II
Note 9: The probability of a small preexisting leak used in the PSA is 3.80E-03 (basic event CNTLK1_PREEXISTS).
The frequency of small containment penetration isolation failures is calculated as
[total small isolation failures & preexisting leaks] - [CDF] x [3.80E-03]
Note 10: The probability of a large preexisting leak used in the PSA is 1.44E-03 (basic event CNTLK1_PREEXISTL).
The frequency of large containment penetration isolation failures is calculated as
[total large isolation failures & preexisting leaks] - [CDF] x [1.44E-03]
Note 11: Since the PSA used a smaller value for the frequency of small preexisting containment leaks than determined in Section 6.1, the frequency for small preexisting containment leaks is calculated as the product of the probability of a small preexisting containment leak and the CDF. From Section 6.1, the probability of a small preexisting containment leak for the 3 in 10 year ILRT is 0.064.
Note 12: Since the PSA used a smaller value for the frequency of large preexisting containment leaks than determined in Section 6.1, the frequency for large preexisting containment leaks is calculated as the product of the probability of a large preexisting containment leak and the CDF. From Section 6.1, the probability of a large preexisting containment leak for the 3 in 10 year ILRT is 0.021.

Table 3

KPDS	Frequency	Description
FCI	8.11E-06	
EIB	1.44E-07	Sum of PDSs EIB, FCS, FCB, ETL, GTL, HTL, and FPL
ENIYA	5.22E-07	(Note 1)
ENIYB	5.21E-07	
ENIYN	5.22E-07	
FNI	1.68E-07	
BCI	3.78E-07	
ENB	2.49E-07	Sum of PDSs ENB, GNS, ENS, FNS
FGI	2.19E-07	
LCI	8.04E-07	Sum of PDSs LCI and DCI
GNI	6.59E-07	
HCI	8.59E-07	
ATV	5.36E-08	
HNI	2.22E-08	
EGI	6.78E-08	
LNIYA	7.24E-10	(Note 2)
LNIYC	7.24E-10	
Total KPDS	1.33E-05	

Note 1: Calculated as 0.33 x ENI
Note 2: Calculated as 0.5 x LNI

In addition to the previously discussed causes of containment isolation failure, there are additional containment failures that result from the progression of the accident. These failures are identified in the level II portion of the PSA. The level II portion of the PSA (reference 1) determines the frequency of accident scenarios or sequences which result in containment failures. In addition, the level II portion of the PSA determines the plant/conditions in containment for the sequences which result in containment failure (CF). Key information about the state of the plant determined for each CF sequence is:

- RCS pressure at the time of vessel failure,
- time, size, and location of the containment failure or bypass,
- containment spray operation and ice condenser function and
- ex-vessel debris cooling.

For KPDSs ENI and LNI, the functionality of the Air Return Fans and Hydrogen Igniters are also key information. These two KPDSs were subdivided according to Table 3a. Sensitivity runs were used to determine the split fractions. In addition, the "Y" designator indicates that the ice beds are available. Details of the sensitivity runs are provided in Table 3b.

Table 3a

Hydrogen Control Designators		
Air Return	Ignitors	
Fans	Yes	No
Available	A	B
Not avail.	C	N

Table 3b

KPDS	Frequency	Relative Frequency	Description
ENI	1.53E-06	33%	Model WR3ES with IC=S, AR=S and HH=S
ENI	1.53E-06	33%	Model WR3ES with IC=S, AR=S and HH=F
ENI	1.53E-06	33%	Model WR3ES with IC=S, AR=F and HH=F
	4.60E-06		
LNI	1.12E-09	50%	Model WR3ES with IC=S, AR=S and HH=S
LNI	1.12E-09	50%	Model WR3ES with IC=S, AR=F and HH=S
	2.2372E-09		

The Sequoyah Level II model was used to transform KPDSs to key release categories (KRCs), which represent key source term characteristics. Every CF sequence which has a frequency greater than a selected frequency is assigned to a KRC. The sum of the frequencies of all CF sequences assigned to a given KRC yields the frequency of the release category. The level II portion of the PSA reports the frequency of KRCs which have a frequency greater than 1.0E-11.

The KRCs characterize the releases from containment as being either early or late and as being either small or large. The KRCs for small and large early releases are due to either the containment isolation failures, preexisting leaks or bypasses previously identified in the level I portion of the PSA or due to severe accident progression (e.g. a large containment failure due to a hydrogen burn). The KRCs for late releases (either small or large) are due solely to severe accident progression. The frequency of the KRCs and their characterization are presented in Table 4. The characterization of the KRCs is consistent with Table 4.9-3 of reference 1 and Table 2-13 of reference 10.

Table 4

KRC	Frequency	Long Term Intact Containment	Small Early CF and Bypass	Large Early CF and Bypass	Late and Long Term Release (Note 1)	Percent of Total
R21	9.65E-06	9.65E-06				72.56%
R17L	8.54E-07				8.54E-07	6.42%
R11I	5.18E-07				5.18E-07	3.90%
R11IF	5.17E-07				5.17E-07	3.89%
R17LU	4.94E-07				4.94E-07	3.71%
R20	3.93E-07		3.93E-07			2.96%
R17U	3.23E-07				3.23E-07	2.43%
R01DI	2.19E-07			2.19E-07		1.65%
R22	9.64E-08	9.64E-08				0.72%
R04IF	8.84E-08			8.84E-08		0.66%
R19	5.36E-08			5.36E-08		0.40%
R01IF	3.51E-08			3.51E-08		0.26%
R02IF	2.18E-08			2.18E-08		0.16%
R03IF	1.35E-08			1.35E-08		0.10%
R04	6.87E-09			6.87E-09		0.05%
R18	5.63E-09			5.63E-09		0.04%
R03I	4.20E-09			4.20E-09		0.03%
R04UIF	2.25E-09			2.25E-09		0.02%
R01UIF	9.63E-10			9.63E-10		0.01%
R05LIF	6.78E-10		6.78E-10			0.01%
R03UIF	5.88E-10			5.88E-10		0.00%
R06IF	4.33E-10		4.33E-10			0.00%
R05IF	7.99E-11		7.99E-11			0.00%
R03	1.97E-11			1.97E-11		0.00%
R01I	1.06E-11			1.06E-11		0.00%
R06LIF	1.03E-11		1.03E-11			0.00%
Total	1.33E-05	9.75E-06	3.94E-07	4.52E-07	2.71E-06	100.00%

Small early releases due to severe accident progression (Note 2)	2.26E-08
Large early releases due to severe accident progression (Note 3)	3.77E-07

Note 1: Large and small source terms are not distinguished.

Note 2: This is calculated as the total of [small early CF & Bypass] minus [small bypass -- analyzed in level II] minus [small isolation failures -- analyzed in level II]

Note 3: This is calculated as the total of [large early CF & Bypass] minus [large bypass -- analyzed in level II] minus [large isolation failures -- analyzed in level II]

Using the results from Tables-1, 2, and 4 the effect of increasing the ILRT test frequency is summarized in Table 5.

Table 5

Class	Description	ILRT Frequency			
		1/40 mo.	1/120 mo.	1/180 mo.	1/240 mo.
1	Containment Intact (Note 1)	8.68E-06	8.52E-06	8.40E-06	8.28E-06
2	Small Dependent Containment Penetration Isolation Failures (Note 3)	1.71E-07			
3	Large Dependent Containment Penetration Isolation Failures (Note 3)	4.05E-09			
4	Small Early Containment Failures Due Severe Accident Progression (Note 3)	2.26E-08			
5	Large Early Containment Failures Due to Severe Accident Progression (Note 3)	3.77E-07			
6	Late Containment Failures (Small & Large) Due to Severe Accident Progression (Note 3)	2.71E-06			
7	Small Containment Bypasses (Note 3)	6.40E-07			
8	Large Containment Bypasses (Note 3)	5.36E-08			
9	Small Preexisting Leaks (Note 4)	8.85E-07	1.01E-06	1.10E-06	1.19E-06
10	Large Preexisting Leaks (Note 4)	2.90E-07	3.30E-07	3.60E-07	3.89E-07
	CDF	1.38E-05			
	LERF (sum of classes 3, 5, 8 & 10)	7.25E-07	7.64E-07	7.94E-07	8.24E-07
	Change in LERF (based on a 1/120 month ILRT frequency)	-3.96E-08	0.00E+00	2.97E-08	5.93E-08
	All Releases (sum of classes 2 through 10)	5.15E-06	5.31E-06	5.43E-06	5.55E-06
	Change in All Releases (based on a 1/120 month ILRT frequency)	-1.60E-07	0.00E+00	1.20E-07	2.40E-07
	Change in All Releases (% based on a 1/120 month ILRT frequency)	-3.02%	0.00%	2.26%	4.52%
	Conditional Containment Failure Probability (CCFP) (Note 5)	0.3601	0.3840	0.3927	0.4014
	Change in CCFP (based on a 1/120 month ILRT frequency)	-0.0239	0.0000	0.0087	0.0174
	Change in CCFP (% based on a 1/120 month ILRT frequency)	-6.23%	0.00%	2.26%	4.52%

Note 1: Calculated as CDF - sum (class 2 through class 10).
Note 2: Reserved.
Note 3: Invariant to changes in ILRT frequency.
Note 4: See Table 1 for the multiplier. See Table 2 for the baseline frequency of these leaks.
Note 5: Calculated as (1 - class 1)/CDF.

6.3 Effect of a Preexisting Containment Leak on Population Dose

The release classes determined in Section 6.2 are assigned a leakage rate in Table 6, consistent with reference 9.

Table 6

class	Description	Maximum Leak Rate (in L_a) (Note 1)
1	Containment Intact	2
2	Small Containment Penetration Isolation Failures	35
3	Large Containment Penetration Isolation Failures	35
4	Small Early Containment Failures Due Severe Accident Progression	100
5	Large Early Containment Failures Due to Severe Accident Progression	100
6	Late Containment Failures (Small & Large) Due to Severe Accident Progression	100
7	Small Containment Bypasses (Note 2)	0
8	Large Containment Bypasses (Note 2)	0
9	Small Preexisting Leaks	10
10	Large Preexisting Leaks	100

Note 1: L_a is 0.25% per day

Note 2: These sequences involve containment bypasses so their leak rate is not quantified in terms of L_a. These sequences are not affected by changes in ILRT frequency.

The dose to the surrounding population from severe accidents was documented for SQN in reference 4. Pertinent characteristics of SQN and WBN are summarized in Table 7. The dose to the surrounding population from severe accidents was determined for WBN based upon parametric conversion from the SQN value, detailed in Table 7a.

Table 7

Value	Units	Parameter (Note 1)	Reference (Note 2)
12	person-rem/year	SQN population dose within 50 miles (mean)	Reference 4, page 5.9
3411	MWth	SQN power level	Reference 4, page S-1
3459	MWth	WBN NSSS power level	WBN FSAR, page 1.2-3
736,270	persons	SQN population within 50 miles (1980)	SQN FSAR, Table 2.1.3-8
935,795	persons	WBN population within 50 miles (2000)	WBN FSAR, table 2.1-13
5.60E-05	per year	SQN CDF	Reference 4, page 5.9
1.38E-05	per year	WBN CDF	Reference 11

Note 1: Sequoyah values provided are for the plant as analyzed in Reference 4.
Note 2: copies of pertinent reference sheets are provided in Appendix A.

Table 7a

Ratio	Parameter
1.01	Power Level (Note 1)
1.27	Population within 50 miles (Note 1)
0.25	CDF (Note 1)
0.32	Ratio Product (Note 3)
3.82	WBN population dose within 50 miles (person-rem per reactor year) (Note 4)
2.76E+05	WBN population dose within 50 miles (person rem per core damage event) (Note 2)

Note 1: WBN value divided by SQN value
Note 2: WBN population dose divided by WBN CDF. This value is the estimated WBN offsite dose and is set equal to 1 La for the Table 8 dose calculations.
Note 3: product of power level, population, and CDF ratios.
Note 4: ratio product times SQN population dose.

The 50 mile population dose was set equal to the dose from a leak rate of 1 L_a. This is a very conservative treatment since the 50 mile population dose is integrated over all accident classes, not just the containment intact class. Consistent with reference 9, the population dose is increased linearly with L_a to determine the population dose for a given class of containment releases. The effect on population dose as ILRT frequency is decreased is calculated in Table 8.

Table 8

Class	Description	Population dose as a function of ILRT Frequency (Note 1)			
		1/40 mo.	1/120 mo.	1/180 mo.	1/240 mo.
1	Containment Intact	4.79E+00	4.70E+00	4.64E+00	4.57E+00
2	Small Dependent Containment Penetration Isolation Failures	1.65E+00			
3	Large Dependent Containment Penetration Isolation Failures	3.92E-02			
4	Small Early Containment Failures Due to Severe Accident Progression	6.25E-01			
5	Large Early Containment Failures Due to Severe Accident Progression	1.04E+01			
6	Late Containment Failures (Small & Large) Due to Severe Accident Progression	7.47E+01			
7	Small Containment Bypasses	Not Quantified			
8	Large Containment Bypasses	Not Quantified			
9	Small Preexisting Leaks	2.44E+00	2.78E+00	3.03E+00	3.28E+00
10	Large Preexisting Leaks	8.02E+00	9.11E+00	9.93E+00	1.07E+01
	Total Dose (person-rem) (Note 2)	1.03E+02	1.04E+02	1.05E+02	1.06E+02
	Change in population dose (Note 3)	-1.34E+00	0.00E+00	1.00E+00	2.01E+00
	Change in population dose (Note 4)	0.00E+00	1.34E+00	2.34E+00	3.34E+00

Note 1: This is calculated as the product of the frequency of the described sequences from Table 5, the magnitude of the release from Table 6, and conditional dose from Table 7.

Note 2: Sum of classes 1 through 10

Note 3: Change based upon baseline case, 1 ILRT per 120 months.

Note 4: Change based upon 1 ILRT per 40 months.

7.0 Summary of Results

Table 9 provides a summary of the results. Table 9a provides the same figures of merit as Table 9, but includes 10% margin. Table 9a should be referenced for the proposed ILRT frequency Technical Specification change.

Table 9

	Delta LERF	Delta All Releases	Delta CCFP	Delta Population Dose
	per reactor year	per reactor year	percent	person-rem
ILRT Change from 1/10 years to 1/15 years	2.97E-08	1.20E-07	2.26%	1.00E+00
ILRT Change from 3/10 years to 1/15 years	6.92E-08	2.80E-07	8.50%	2.34E+00

Table 9a

	Delta LERF	Delta All Releases	Delta CCFP	Delta Population Dose
	per reactor year	per reactor year	percent	person-rem
ILRT Change from 1/10 years to 1/15 years	3.26E-08	1.32E-07	2.49%	1.10E+00
ILRT Change from 3/10 years to 1/15 years	7.61E-08	3.08E-07	9.35%	2.57E+00

8.0 Supporting Graphics

None.

9.0 Conclusions

The increase in LERF when the frequency of an ILRT is decreased from 1/10 years to 1/15 years is 3.26E-08/ry (reference Table 9a) which is a very small increase in LERF per Regulatory Guide 1.174 (reference 5). A very small increase in LERF is acceptable regardless of whether total LERF (due to both internal and external initiating events) has been quantified.

An Individual Plant Examination for External Events (IPEEE) was performed in accordance with Generic Letter 88-20 to identify severe accident vulnerabilities from external events (reference 6). The seismic evaluation was performed using the Seismic Margin Assessment methodology and therefore did not include a calculation of seismic CDF or LERF. The risk from fire was evaluated using the Fire Induced Vulnerability Examination (FIVE) methodology, producing a conservative estimate for CDF of 7E-06/ry. The IPEEE confirmed that relative to containment failure no containment failure modes were introduced in the fire evaluation that differ significantly from those seen in the IPE. The CDF for internal events, for comparison, is 1.38E-05/ry. High winds, flooding, and Other (HFO) contributors to risk were dismissed as insignificant. The IPEEE methodology was approved by the NRC (reference 8).

A general rule of thumb used for many risk-informed PRA calculations (e.g., Severe Accident Mitigation Alternatives Analysis) and approved by the NRC is to double the internal events CDF and LERF to estimate total CDF and LERF.

The internal events LERF with a 1/10 years ILRT is 7.64E-07/ry. The estimated LERF from both internal and external events, for an ILRT frequency of 1/10 years is

2 x 7.64E-07/ry = 1.53E-06/ry.

The internal events LERF with a 1/15 years ILRT is 7.94 E-07/ry. The estimated LERF from both internal and external events is

2 x 7.94 E-07/ry = 1.59E-06/ry.

The estimated change from 1/10 to 1/15 from both internal and external events is

1.59E-06 – 1.53E-06 = 6 E-08

which a very small increase in LERF per RG 1.174.

10.0 Appendices and Attachments

A copy of pertinent input data, summarized in Table 7, is provided in Attachment A.

ATTACHMENT A

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ATTACHMENT A
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Reducing the RCS pressure at VB, of course, reduces the loads placed on the containment at VB, and thus reduces the probability of CF.

The LCFs are generally associated with the population that does not evacuate. Thus, this risk measure is not particularly sensitive to the timing of CF, but rather to whether the containment fails. Furthermore, because there is no threshold effect for LCFs, this consequence measure is not as sensitive to the magnitude of the release as is the EF risk. LCF risk is primarily dependent on frequency of CF. Unlike EF risk, late CFs as well as EFs of the containment are important to the latent cancers.

There are several features of the Sequoyah plant that reduce the magnitude of the source term. In the majority of the accidents analyzed, the in-vessel releases experience decontamination by the ice condenser (IC). Many times if VB is predicted to occur, the CCI is either inhibited because a coolable debris bed is formed and the cavity water is replenished, or the release from the CCI is scrubbed by an overlying water pool. Operation of the containment spray system (CSS) also helps to mitigate the source term.

Table 5.1-1
Distributions for Annual Risk at Sequoyah Due to Internal Initiators
(All values per reactor-yr; population doses in person-rem)

Risk Measure	5th%tile	Median	Mean	95th%tile
Core Damage	1.5E-5	3.9E-5	5.6E-5	1.5E-4
EFs	4.7E-8	2.4E-6	2.6E-5	1.2E-4
LCFs	5.6E-4	4.8E-3	1.4E-2	5.3E-2
Population Dose 50 mi	8.7E-1	5.0E+0	1.2E+1	4.6E+1
Population Dose Entire Region	3.5E+0	2.9E+1	8.1E+1	3.1E+2
Ind. EF Risk, 0 - 1 mile	4.6E-11	1.5E-9	1.1E-8	4.3E-8
Ind. LCF Risk 0 - 10 miles	3.9E-10	3.2E-9	1.0E-8	3.5E-8

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VOLUME 5, REVISION 1, PART 1
EVALUATION OF SEVERE ACCIDENT RISKS: SEQUOYAH UNIT 1

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SUMMARY

S.1 Introduction

The United States Nuclear Regulatory Commission (NRC) has recently completed a major study to provide a current characterization of severe accident risks from light water reactors (LWRs). This characterization is derived from integrated risk analyses of five plants. The summary of this study, NUREG-1150,[1] has been issued as a second draft for comment.

The risk assessments on which NUREG-1150 is based can generally be characterized as consisting of four analysis steps, an integration step, and an uncertainty analysis step:

1. Accident frequency analysis: the determination of the likelihood and nature of accidents that result in the onset of core damage.

2. Accident progression analysis: an investigation of the core damage process, both within the reactor vessel before it fails and in the containment afterwards, and the resultant impact on the containment.

3. Source term analysis: an estimation of the radionuclide transport within the reactor coolant system (RCS) and the containment, and the magnitude of the subsequent releases to the environment.

4. Consequence analysis: the calculation of the offsite consequences, primarily in terms of health effects in the general population.

5. Risk integration: the assembly of the outputs of the previous tasks into an overall expression of risk.

6. Uncertainty analysis: the propagation of the uncertainties in the initiating events, failure events, accident progression branching ratios and parameters, and source term parameters through the first three analyses above, and the determination of which of these uncertainties contributes the most to the uncertainty in risk.

This volume presents the details of the last five of the six steps listed above for the Sequoyah Nuclear Station, Unit 1. The first step is described in NUREG/CR-4550.[2]

S.2 Overview of Sequoyah Nuclear Station, Unit 1

The Sequoyah Power Station, Unit 1 is operated by the Tennessee Valley Authority (TVA) and is located on the west shore of the Chickamauga Lake in southeastern Tennessee, about 10 miles northeast of Chattanooga, Tennessee. There are two units located on the site; Unit 2 is essentially identical to Unit 1.

The nuclear reactor of Sequoyah Unit 1 is a 1148 MWe (3411 MWt) pressurized water reactor (PWR) designed and built by Westinghouse. The reactor coolant system (RCS) has four U-tube steam generators (SGs) and four reactor

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The inherent design of the pressurized water, closed-cycle reactor minimizes the quantities of fission products released to the atmosphere. Three barriers exist between the fission product accumulation and the environment. These are the fuel cladding, the reactor vessel and coolant loops, and the reactor containment. The consequences of a breach of the fuel cladding are greatly reduced by the ability of the uranium dioxide lattice to retain fission products. Escape of fission products through fuel cladding defects would be contained within the pressure vessel, loops, and auxiliary systems. Breach of these systems or equipment would release the fission products to the reactor containment where they would be retained. The reactor containment is designed to adequately retain these fission products under the most severe accident conditions, as analyzed in Chapters 6 and 15.

The license application NSSS power level is 3,475 MWt which includes 16 MWt from the reactor coolant pumps. Reactivity coefficients and other design parameters, which are supported by analysis and experience with other similar plants, provide the basis for concluding that this reactor can be operated safely at the power levels of the application rating. The initial core load has a negative moderator temperature coefficient of reactivity at operating temperature at all times throughout core life.

The reactor core, with its related control and protection system, is designed to function throughout its design lifetime without exceeding the acceptable fuel damage limits. The core design, together with process and residual heat removal (RHR) systems, provides for this capability under expected conditions of normal operations with appropriate margins for uncertainties and anticipated transient situations, including, as examples, the effects of the loss of reactor coolant flow, turbine trips due to steam and power conversion system malfunctions, and loss of external electrical load. Acceptable fuel damage limits can be found in Section 4.2.

The reactor core is a multi-region cycled core. The fuel rods are cold worked Zircaloy/ZIRLO™ tubes containing slightly enriched uranium oxide fuel. The fuel assembly is a canless type with the basic assembly consisting of the guide thimbles mechanically fastened to the grids, top, and bottom nozzles. The fuel rods are held in the grids by spring clips. The internals, consisting of the upper and lower core support structures, are designed to support, align, and guide the core components, direct the coolant flow and guide the in-core instrumentation. Dissolved boric acid is used as a reactivity control device to minimize the use of burnable absorbers.

Rod cluster control assemblies (RCCAs) and burnable absorber rods are inserted into the guide thimbles of the fuel assemblies. The absorber sections of the RCCAs are fabricated of boron carbide pellets with silver-indium-cadmium alloy slugs sealed in stainless steel tubes. The absorber material in the burnable absorber rods is in the form of borosilicate glass sealed in stainless steel tubes. The control rod drive mechanisms for the RCCAs are of the magnetic jack type. The latches are controlled by three magnetic coils. They are so designed that upon a loss of power to the coils, the RCCA is released and falls into the core by gravity to shut down the reactor.

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TABLE 2.1.3-8

1980 POPULATION DISTRIBUTION WITHIN FIFTY MILES OF SITE

Direction	Total	0-10	10-20	20-30	30-40	40-50
		Miles from Site				
N	15,605	730	3,560	2,030	2,535	6,750
NNE	20,805	440	6,485	4,120	4,705	5,055
NE	23,270	315	1,230	2,860	7,615	11,250
ENE	46,035	555	3,900	6,200	24,740	10,640
E	21,920	505	11,930	3,380	2,005	4,100
ESE	51,760	1,195	34,815	3,350	1,075	11,325
SE	15,040	900	6,835	3,140	1,795	2,370
SSE	56,420	1,045	6,840	9,005	36,080	3,450
S	51,060	1,275	9,565	9,895	22,290	8,035
SSW	156,825	2,785	90,575	42,330	14,695	6,440
SW	162,260	2,860	115,955	29,725	8,655	5,065
WSW	54,975	6,785	23,310	4,595	11,440	8,845
W	17,480	3,845	1,470	4,820	3,705	3,640
WNW	14,875	3,385	2,645	3,160	3,835	1,850
NW	17,880	4,930	1,050	1,460	765	9,675
NNW	10,060	1,160	510	2,725	1,555	4,110
Total	736,270	32,710	320,675	132,795	147,490	102,600

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TABLE 2.1-13

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YEAR 2000 POPULATION DISTRIBUTION
WITHIN 50 MILES OF THE SITE

	DISTANCE FROM SITE (MILES)					
DIR	0-10	10-20	20-30	30-40	40-50	TOTAL
N	1,107	1,807	1,908	3,112	3,723	11,657
NNE	884	7.551	16,836	8,671	1,117	35,058
NE	1,247	2,970	17,041	26,873	47,179	95,310
ENE	414	2,186	8,744	34,361	118,713	164,419
E	536	8,589	7,835	9,442	15,203	41,605
ESE	639	4,066	10,621	3,020	300	18,646
SE	535	16,066	9,724	3,537	3,381	33,242
SSE	734	3,620	7,289	3,557	5,866	21,066
S	1,616	1,004	29,684	34,923	10,069	77,296
SSW	783	5,146	15,097	22,305	102,188	145,518
SW	482	5,898	8,182	59,551	108,848	182,961
WSW	1,274	9,406	1,860	6,401	5,862	24,803
W	900	879	4,739	2,619	1,852	10,988
WNW	501	1,298	2,580	3,243	15,839	23,461
NW	2,634	290	6,352	6,127	8,634	24,037
NNW	2,114	787	15,955	4,176	2,697	25,729
TOTAL	16,400	71,560	164,446	231,919	451,470	935,795

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