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

MAR 3 1 2006

WBN-TS-05-07

10 CFR 50.90

U. S. Nuclear Regulatory Commission
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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]) - REQUEST FOR ADDITIONAL INFORMATION (TAC NO. MC9239)

The purpose of this letter is to provide TVA's response to NRC's request for additional information dated January 24, 2006, concerning the subject proposed technical specification request. The proposed amendment that was submitted on December 14, 2005 revises Technical Specification 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.

The Enclosure provides TVA's response to NRC's concerns. TVA has determined this additional information and the revision to the calculation, do not significantly change the conclusions in the December 14, 2005 submittal for the No Significant Hazards Considerations associated with the proposed change, consequently, the No Significant Hazards Consideration is not being revised.

A017

MAR 3 1 2006

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 31st day of March 2006.

Sincerely,

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

Enclosure
cc (Enclosure):
NRC Resident Inspector
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ENCLOSURE

WATTS BAR NUCLEAR PLANT (WBN) UNIT 1
TECHNICAL SPECIFICATION REQUEST WBN-TS-05-07
ONE-TIME FREQUENCY EXTENSION FOR TYPE A TEST
REQUEST FOR ADDITIONAL INFORMATION

NRC QUESTION 1

The risk assessment methodology used to support the integrated leakage rate test (ILRT) interval extension for Watts Bar is based on a methodology developed by the Electric Power Research Institute (EPRI) in 1994. A revision to this methodology, developed for the Nuclear Energy Institute (NEI) by EPRI in 2001, corrected/improved the original methodology in several areas. Based on a Nuclear Regulatory Commission (NRC) staff assessment, the revised methodology (referred to as the NEI interim guidance) would indicate larger risk impacts (e.g., Δ large early release frequency (LERF)) for the ILRT interval extension than the original. In view of the nonconservative nature of the original EPRI methodology, please provide a reassessment of the risk impacts of the requested change for Watts Bar based on the NEI interim guidance. In reporting risk results (for Δ person-rem, Δ LERF, and Δ conditional containment failure probability), include results corresponding to a change in test frequency from three tests in 10 years to one test in 15 years

TVA RESPONSE

TVA's attached calculation, CN-NUC-WBN-MEB-MDN00199920050099, was revised such that it is now based on the 2001 NEI interim guidance. The calculation includes results corresponding to a change in test frequency from both three tests in 10 years to one test in 15 years and one test in 10 years to one test in 15 years. Based on this revision to the calculation, a revised Risk Assessment from pages E1-6 and E1-7 of TVA's December 14, 2005, request is also attached as Attachment 1 for your convenience.

NRC QUESTION 2

In Enclosure 4, the population dose for each release class is obtained based on information in Table 6, together with an assumption that the 50-mile population dose for an intact containment (1 La) is equal to the average conditional population dose (2.76E+5 person-rem per core damage event). The resulting population dose for each release class is substantially higher than estimated in the Tennessee Valley Authority's evaluation of severe accident mitigation alternatives (SAMDAs) performed in 1994 (Reference 10 in Enclosure 4). For example, the population dose assigned to the intact containment release class is 2.76E+5 person-rem per event in the ILRT amendment request, versus approximately 200 person-rem per event in the SAMDA evaluation;

the population dose assigned to the largest release class is 2.76E+7 person-rem per event in the ILRT amendment request versus approximately 4E+5 person-rem per event in the SAMDA evaluation. Furthermore, use of a very large population dose for the intact containment release class in the ILRT evaluation (both in absolute terms and relative to the largest release class) leads to an over-estimate of the impact of the ILRT extension on population dose. Please reconcile the population dose values with those in the SAMDA analysis, and provide a reassessment of the impact of the ILRT interval extension on population dose based on appropriate population dose values.

TVA RESPONSE

TVA's attached calculation was revised as requested, to use the population dose values documented in the SAMDA analysis. NRC approval of those values is documented in Reference 16 of the calculation.

NRC QUESTION 3

Inspections of some reinforced and steel containments (e.g., North Anna, Brunswick, D. C. Cook, and Oyster Creek) have indicated degradation from the uninspectable (embedded) side of the steel shell and liner of primary containments. Please describe the uninspectable areas of the Watts Bar containment, and the programs used to monitor their condition. Provide a quantitative assessment of the impact on LERF due to age-related degradation in these areas, in support of the requested ILRT interval extension to 15 years. This could be based on methods such as those utilized in the Browns Ferry ILRT extension request.

TVA RESPONSE

3a. **Watts Bar Steel Containment Vessel (SCV) inaccessible Areas**

The inaccessible surface areas for the WBN Unit 1 SCV are identified as areas of the exterior SCV surface with insulation and the shielding area around the fuel transfer penetration. The area below the floor of the embedded metal liner and concrete base slab is also inaccessible for the inspection.

The inaccessible surface areas due to insulation are identified in TVA's WBN Engineering Specification entitled "*Installation,* *Modification and Maintenance of Heat and Anti-Sweat Insulation."* These areas are from 54 degrees to 126 degrees between Elevation 716.0 feet and 747.0 feet.

A walkdown of the SCV also identified additional insulation locations from 50 degrees to 126 degrees between Elevation 713.0 feet and 716.0 feet and from 50 degrees to 54 degrees between Elevation 716.0 feet and 733.0 feet.

The total inaccessible area for inspection, including the shielding area around the fuel transfer penetration, is estimated as 2800 square feet. The area below the floor of the embedded metal liner and concrete base slab is not included in the 2800 square feet.

3b. Inspection Programs

General visual examination of accessible exterior surfaces of WBN Unit 1 SCV was performed during Unit 1 Cycle 6 Refueling Outage for any evidence of structural deterioration that may affect either the containment structural integrity or leak-tightness. It is noted that the conditions identified during Unit 1 Cycle 3 Refueling Outage general visual examination do not appear to have changed significantly.

Currently, there are no monitoring programs established for the WBN Unit 1 inaccessible areas of the exterior SCV surface with insulation and the area around the fuel transfer penetration. The inaccessible surface areas due to insulation are provided with a moisture barrier prior to installing the insulation. However, general visual inspection of the SCV during the Unit 1 Cycle 6 Refueling Outage did not identify any moisture present at the edges of the moisture barrier.

During Unit 1 Cycle 6 Refueling Outage, general visual inspection of SCV per Surveillance Instruction entitled, *General Visual Inspection of Steel Containment Vessel*, identified several areas on the annulus side exhibiting light rust at the floor-to-SCV interface (Elevation 702 Annulus Floor Elevation), tee joint for ice condenser seal with heavy rust (Elevation 743 Azimuth 300) and medium rust on SCV (Elevation 757, Azimuth 292-301) in the containment side. The condition of the containment structure is covered under 10 CFR 50.65, *Maintenance Rule* and the results of the Unit 1 Cycle 6 Refueling Outage examination were reviewed to include the structure condition for maintenance rule reporting. The conditions of WBN Unit 1 SCV were evaluated by WBN

Engineering with support from Non-destruction examination (NDE) Specialists. The rust was considered to be superficial and there was no flaking, pitting, or visible loss of base metal. It was determined that there was no impact on the leak tightness and structural integrity of the SCV.

3c. <u>Quantitative Assessment of impact on LERF due to age-related degradation</u>

TVA's attached calculation was revised to include a quantitative assessment of the impact on LERF due to a degradation mechanism as found at North Anna, Brunswick, D. C. Cook, and Oyster Creek.

NRC QUESTION 4

In Enclosure 4, it is assumed that the LERF associated with both internal and external events can be estimated by doubling the LERF associated with only internal events. This simplified approach has been accepted by the NRC if sufficient justification is provided that the core damage frequency (CDF) from external events, including seismic and fire events, is approximately equal to or less than that for internal events. Although fire risk is discussed briefly in Section 9.0, the contribution from seismic events was dismissed on the basis that the seismic margin assessment did not calculate seismic CDF or LERF. Provide additional justification that the contribution from seismic events is small. This could be based on simplified methods such as those utilized in the Browns Ferry ILRT extension request.

TVA RESPONSE

TVA's attached calculation was revised to include a quantitative assessment of the impact of seismic events based upon a simplified method used in the BFN ILRT extension request.

ENCLOSURE
ATTACHMENT 1

WATTS BAR NUCLEAR PLANT (WBN) UNIT 1
TECHNICAL SPECIFICATION REQUEST WBN-TS-05-07
ONE-TIME FREQUENCY EXTENSION FOR TYPE A TEST
REQUEST FOR ADDITIONAL INFORMATION

REVISED TVA RISK ASSESSMENT

ENCLOSURE
ATTACHMENT 1

WATTS BAR NUCLEAR PLANT (WBN) UNIT 1
TECHNICAL SPECIFICATION REQUEST WBN-TS-05-07
ONE-TIME FREQUENCY EXTENSION FOR TYPE A TEST

REVISED TVA RISK ASSESSMENT

The risk assessment below is a revision to the Risk Assessment in TVA's letter to NRC dated December 12, 2005 on Page E1-6 and E1-7). This revised assessment is being provided for your convenience.

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 containment integrity leak rate testing (CILRT) frequency. 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 in December 14, 2005 request] quantifies the increase in large early release frequency (LERF), population dose, and conditional containment failure probability as a result of a decrease in the frequency of performing a CILRT (see Enclosure 4 in December 14, 2005 request).

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 other containment failure mechanisms.

The risk assessment showed the increase in LERF to be 1.58E-07/reactor years (ry) when the frequency of a Type A test was decreased from one in 10 years to one in 15 years. The risk assessment showed the increase in LERF to be 3.72E-07/ry when the frequency of a Type A test was decreased from three in 10 years to one in 15 years. The total LERF was calculated to be 2.78E-06. 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 in December 14, 2005 request] defines small changes in LERF as increases in LERF less than 1.0E-6/ry but greater than 1.0E-07/ry. A small change in

LERF is acceptable provided the total LERF is less than 1.0E-05/ry. Therefore, the proposed change in Type A test frequency is acceptable.

The risk assessment also showed the increase in population dose to be 5.93E-03 person-rem/ry when the frequency of a Type A test was decreased from one in 10 years to one in 15 years. The risk assessment showed the increase in population dose to be 1.40E-02 person-rem/ry when the frequency of a Type A test was decreased from three in 10 years to one in 15 years.

The risk assessment showed the increase in conditional containment failure probability to be 0.52 percent when the frequency of a Type A test was decreased from one in 10 years to one in 15 years. The risk assessment showed the increase in conditional containment failure probability to be 1.23 percent when the frequency of a Type A test was decreased from three in 10 years to one in 15 years.

ENCLOSURE
ATTACHMENT 2

WATTS BAR NUCLEAR PLANT (WBN) UNIT 1
TECHNICAL SPECIFICATION REQUEST WBN-TS-05-07
ONE-TIME FREQUENCY EXTENSION FOR TYPE A TEST
RISK ASSESSMENT

CALCULATION CN-NUC-WBN-MEB-MDN00199920050099

TVAN CALCULATION COVERSHEET/CCRIS UPDATE

REV 0 EDMS/RIMS NO.	EDMS TYPE:	EDMS ACCESSION NO (N/A for REV. 0)
T71051101801	calculations(nuclear)	T71 060323 817

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	REVISION APPLICABILITY
CURRENT	CN	NUC	WBN	MEB	MDN00199920050099	1	2	Entire calc. ☒
NEW								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
XBPYZJO24	(423) 751-4124	MEB	N/A	☐ Yes ☒ No

PREPARER SIGNATURE	DATE	CHECKER SIGNATURE	DATE
Calvin A. McCullough *Calvin A. McCullough*	2/23/2006	*William Murf*	3-14-06
VERIFIER SIGNATURE	**DATE**	**APPROVAL SIGNATURE**	**DATE**
N/A		*for* *[signature]*	3/23/06

STATEMENT OF PROBLEM/ABSTRACT

This calculation determines the effect on Large Early Release Frequency (LERF), Conditional Containment Failure Probability (CCFP), and population dose as a result of a proposed 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 fission product releases to the environment which correlates to a calculated increase in population dose. Revision 3 of the PSA is used for this calculation.

The numerical results of this calculation are provided in Section 7 for ILRT frequencies of between 3/10 years and 1/20 years.

The increase in LERF is "small" per RG 1.174, and is acceptable since the total LERF is less than 1E-05. Increases in the CCFP 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
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TVAN CALCULATION COVERSHEET/CCRIS UPDATE

CALC ID	TYPE	PLANT	BRANCH	NUMBER	REV
	CN	SQN	MEB	MDN00199920050099	2

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 = 31	
2	Revised to address NRC RAI questions (reference 17) 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: all Total Pages = 43	

TVAN COMPUTER INPUT FILE STORAGE INFORMATION SHEET

Document MDN001-999-2005-0099	Rev. 2	Plant: WBN	

Subject:

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

☐ 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.)

The spreadsheet (Microsoft Excel), this document (Microsoft Word), and Riskman files are stored by Filekeeper.

File Name: mdn-001-999-2005-099-r2.zip

Reference ID: 308158

☐ Microfiche/Fiche

TVAN CALCULATION TABLE OF CONTENTS

Calculation Identifier: MDN001-999-2005-0099	Revision: 2

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TVAN CALCULATION TABLE OF CONTENTS

Calculation Identifier: MDN001-999-2005-0099	Revision: 2	

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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 fission product releases to the environment which correlates to an increase in calculated population dose. This calculation quantifies the increase in large early release frequency, conditional containment failure probability, 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 3.
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. Reserved.
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, (SAMDA) (T25 940630 838).
11. Watts Bar Nuclear Plant Probabilistic Safety Assessment, Revision 3.
12. Reserved.

13. Reserved.

14. Reserved.

15. "Interim Guidance for Performing Risk Impact Assessments In Support of One-Time Extensions for Containment Integrated Leakage Rate Test Surveillance Intervals." Developed for NEI by EPRI, November, 2001.

16. NUREG-0498, Supplement 1. Final Environmental Statement related to the operation of Watts Bar Nuclear Plant, Units 1 and 2. April 1995.

17. Letter from NRC to TVA, WATTS BAR NUCLEAR PLANT, UNIT 1 — REQUEST FOR ADDITIONAL INFORMATION REGARDING EXTENSION OF THE INTEGRATED LEAKAGE RATE TEST INTERVAL (TAC NO. MC9239), 1/24/2006. (L44 060130 066).

18. "One-time extensions of containment integrated leak rate test interval – additional information." NEI letter from Anthony R. Pietrangelo to NEI Administrative Points of Contact. 11/30/2001.

19. Kennedy, Robert P. "Overview of Methods for Seismic PRA and Margin Analysis Including Recent Innovations." Proceedings of the OECD/NEA Workshop on Seismic Risk. Committee on the Safety of Nuclear Installations PWG3 and PWG5. Hosted by the Japan Atomic Energy Research Institute under the Sponsership of the Science and Technology Agency. 10-12 August, 1999, Tokyo, Japan. (B44060310001)

20. TVA Calculation CN-NUC-BFN-NTB-NDN0-064-2004-0005, R0. Risk Assessment for Integrated Leak Rate Test (ILRT) Extension.

21. EPRI NP-6395 D, Probabilistic Seismic Hazard Evaluations at Nuclear Plant Sites in the Central and Eastern United States: Resolution of the Charleston Earthquake Issue. April, 1989.

22. Marks' Standard Handbook for Mechanical Engineers. Eighth Edition.

23. Letter from Calvert Cliffs Nuclear Power Plant to NRC. Response to Request for Additional Information Concerning the License Amendment Request for a one-time Integrated Leakage Rate Test Extension. March 27, 2002. (B44060310001)

24. Letter from NRC to Duke Energy Corporation. McGuire Nuclear Station, Units 1 and 2 re: issuance of amendments [including SER]. March 12, 2003. (B44060310001)

25. Letter from Duke Energy Corporation to the NRC. Catawba Nuclear Station and McGuire Nuclear Station Proposed TS Amendments ... One-Time Extension of Integrated Leak Rate testing (ILRT) Interval. January 8, 2003. (B44060310001)

26. Letter from Indiana Michigan Power Company to the NRC. Donald C. Cook Nuclear Plant Response to NRC Request for Additional Information Regarding License Amendment Request for One-Time Extension of Containment Integrated Leakage Rate Test Interval. November 11, 2002.

27. NUREG 1350, Volume 16. Information Digest, 2004-2005 Edition.

28. Information Notice 86-39: Degradation of Steel Containments [Oyster Creek event]. 12/8/1986.

29. Information Notice 86-99 Supplement 1: Degradation of Steel Containments. 2/14/1991.

30. Information Notice 2004-09: Corrosion of Steel Containment and Containment Liner. 4/27/2004.

3.0 Design Input Data

Appendix A contains the only design input data specific to this calculation.

4.0 Assumptions

Assumptions and associated justification are documented in the relevant text paragraphs and tables.

5.0 Requirements/Limiting Conditions

None.

6.0 Computations and Analyses

6.1 Core Damage Frequency

6.1.1 Internal Events

Core damage due to internal events, including internal flooding, was calculated using revision 3 of the WBN PSA (reference 11). The PSA exists as a Riskman model. The level I model of record, denoted WR3ES, was executed using master frequency file

(MFF) WBNREV3 with a universal truncation frequency of 1E-11. The output from this model included unconditional frequencies of the plant damage states (PDSs), provided in Table 1. Physical characteristics of the PDSs are described in section 4.3 of reference 1.

Table 1
Plant Damage State Frequencies for Internal Events

PDS	Frequency	PDS	Frequency	PDS	Frequency
FCI	8.11E-06	HCS	3.54E-09	GGB	2.31E-10
ENI	1.57E-06	LCS	3.17E-09	FRL	2.10E-10
HCI	8.59E-07	AGI	2.39E-09	DCS	2.01E-10
LCI	7.58E-07	GGI	2.20E-09	HII	1.70E-10
GNI	6.59E-07	KNI	1.94E-09	CNS	1.42E-10
ECB	4.29E-07	LEI	1.77E-09	FTL	1.34E-10
BCI	3.78E-07	BCS	1.72E-09	FEB	1.31E-10
FGI	2.19E-07	LNI	1.45E-09	KNS	1.16E-10
FNI	1.68E-07	FII	1.22E-09	BNI	1.07E-10
ENS	1.25E-07	LGI	9.23E-10	HNS	1.02E-10
ENB	7.36E-08	FGS	9.22E-10	CNI	7.83E-11
FCB	7.16E-08	GTL	9.00E-10	DNI	7.22E-11
EGI	6.78E-08	BEI	8.63E-10	ERL	7.13E-11
ATV	5.36E-08	GNB	8.46E-10	HEB	6.61E-11
GNS	4.90E-08	HPL	8.09E-10	DPL	4.15E-11
DCI	4.58E-08	LPL	7.75E-10	FIB	3.72E-11
FCS	3.82E-08	BGI	7.66E-10	HGS	2.41E-11
HCB	3.75E-08	FNS	7.58E-10	HIB	2.08E-11
HNI	2.22E-08	LII	5.96E-10	HNB	1.85E-11
EIB	1.32E-08	HEI	4.97E-10	CTL	1.76E-11
FPL	1.03E-08	BPL	4.64E-10	ANS	1.44E-11
ETL	1.03E-08	FNB	4.11E-10	HTL	1.38E-11
HGI	9.94E-09	GCB	3.00E-10		
EEB	8.34E-09	BII	2.93E-10	Total PDS	1.38E-05
KGI	5.62E-09	EGS	2.79E-10		
EGB	4.70E-09	DGI	2.56E-10		
FEI	3.64E-09	FGB	2.44E-10		

The PDSs were condensed into key plant damage states (KPDSs) in accordance with section 4.6 of reference 1. The frequency of each KPDS is calculated using an Excel spreadsheet. Results of that summation are provided in Table 1a.

Note that KPDSs ENIYA, ENIYB, ENIYN, LNIYA, and LNIYC are actually a subset of PDSs ENI and LNI, respectively. These two KPDSs were subdivided according tc Table 1b. 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 1c.

Table 1a
Key Plant Damage State Frequencies for Internal Events

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	Refer to Tables 1b and 1c
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, and 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	Refer to Tables 1b and 1c
LNIYC	7.24E-10	
Total KPDS	1.33E-05	

Table 1b

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

Table 1c
PDS ENI, LNI Mapping to
KPDS

PDS	Frequency	Relative Frequency	Description	KPDS
ENI	1.5330E-06	33.34%	Model WR3ES with IC=S, AR=S and HH=S	ENIYA
ENI	1.5314E-06	33.31%	Model WR3ES with IC=S, AR=S and HH=F	ENIYB
ENI	1.5330E-06	33.34%	Model WR3ES with IC=S, AR=F and HH=F	ENIYN
	4.5974E-06			
LNI	1.1186E-09	50.00%	Model WR3ES with IC=S, AR=S and HH=S	LNIYA
LNI	1.1186E-09	50.00%	Model WR3ES with IC=S, AR=F and HH=S	LNIYC
	2.2372E-09			

Note 1: The frequency for KPDS ENIYA, for example, is calculated as the product of the frequency of PDS ENI (from the Level 1 model) by the appropriate relative frequency (from this table).

The Sequoyah Level II model (reference 3) was used to transform KPDSs to key release categories (KRCs). The Sequoyah Level II model was recently updated and represents the state-of-the-art analysis. This treatment is acceptable for this WBN calculation due to the physical similarity of WBN to SQN.

The level II portion of the PSA reports the frequency of KRCs which have a frequency greater than 1E-11. Columns 2 and 3 of Table 3 (refer to section 6.1.4), labeled "internal events," present the results of this Riskman quantification. Physical characteristics of the KRCs are described in section 4.9 of reference 1.

6.1.2 Fire Events

The WBN PRA does not include a model for fire events. The Individual Plant Examination of External Events (IPEEE, reference 6) documents a screening analysis referred to as a Fire-Induced Vulnerability Evaluation (FIVE). The IPEEE does not calculate a CDF due to fire events. However, quoting from the NRC Staff Evaluation Report (SER, reference 8), "A quantification for fire events, that utilized the FIVE methodology, indicated that the contribution to plant CDF from fire was about 7E-6 per reactor-year (RY)." This value is consistent with a summation of fire-related CDF for areas screened at the second level, listed in Table 5.2 of reference 6.

This value was inserted into Table 3 as the total CDF for fire events. It is assumed that the total fire event CDF can be allocated across release categories using the relative ranking of KRCs from internal events. For example, the frequency of KRC R21 due to fire events is calculated as

$$7E\text{-}06 \times 72.56\% = 5.08E\text{-}06.$$

This allocation model is based on the analysis finding that fire does not pose a significant risk to containment integrity, as evaluated by the FIVE of the reactor building, documented in section 3.3 of reference 6.

6.1.3 Seismic Events

Reference 6, the WBN IPEEE, documents completion of a Seismic Margins Assessment (SMA). The SMA is a deterministic process which does not calculate risk values.

Reference 19 provides a simplified methodology (Simple Hybrid Method) for estimating the seismic risk based on a SMA analysis. This approach was used for the BFN ILRT frequency reduction calculation (reference 20), and was recommended by the NRC via reference 17.

The approach consists of 4 steps.

1. Determine the High Confidence Low Probability of Failure (HCLPF) seismic capacity from the SMA analysis.

2. Estimate the 10% conditional failure probability capacity. The following equations are from reference 19, section 6.2. β is the variability of the plant damage fragility. Reference 19, section 6.3, recommends a value of 0.3 for β.

$$C_{10\%} = F_\beta * C_{HCLPF}$$

$$F_\beta = \exp(1.044 * \beta)$$

3. Determine the hazard exceedance frequency ($H_{10\%}$) that corresponds to $C_{10\%}$ from the hazard curves.

4. Determine the seismic risk (which is set equal to the seismic CDF).

$$CDF = 0.5 * H_{10\%}$$

From reference 6, C_{HCLPF} is greater than 0.30g peak ground acceleration (pga).

Table 2 provides the annual probability of exceedance for pga at WBN, from reference 21, Table 3-107. This same information is presented graphically in Figure 1 (reference 21 Figure 3-319).

The conversion factor for acceleration is 1g to 980.665 cm/sec^2 (reference 22, page 1-36).

$$F_B = \exp(1.044 * 0.3) = 1.37$$

$$C_{10\%} = (1.37)*(0.30\ g)(980.665\ cm/sec^2 g) = 403\ cm/sec^2$$

From Figure 1, the annual probability of exceedance for $C_{10\%}$ is approximately 2E-5.

$$CDF = (0.5)*(2E-5) = 1E-5 \text{ per year.}$$

This value was inserted into Table 3 as the total CDF for seismic events. It is assumed that the total seismic event CDF can be allocated across release categories using the relative ranking of KRCs from internal events. This treatment is justified because "there were no vulnerabilities noted in the containment walkdown and review that would lead to an early release due to the IPEEE RLE" (section 3.1.5 of reference 6).

Table 2

Annual Probability of Exceedance

Acceleration (cm/sec^2)	Probability of Exceedance
5	1.90E-02
50	1.80E-03
100	5.70E-04
250	7.70E-05
500	9.90E-06
700	3.00E-06
1000	7.50E-07

Note 1: Probabilities are mean values.
Note 2: Reference 21, Table 3-107.

Figure 1
Probability of exceedance of 0.30g pga

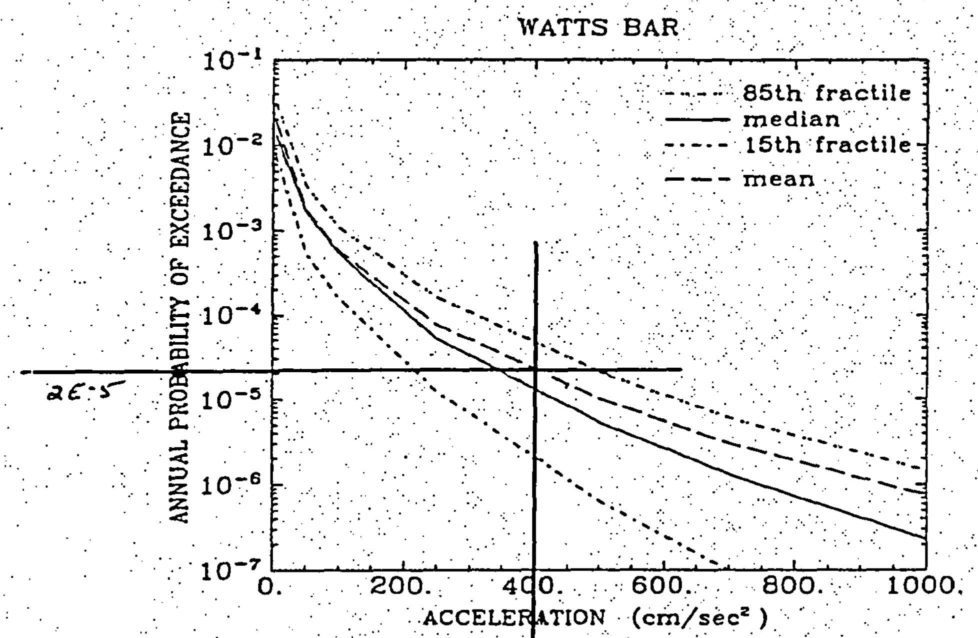

Figure 3-319. Annual probability of exceedance of peak ground acceleration: Watts Bar site.

REFERENCE 21

6.1.4 Total CDF

The total frequency for each KRC is just the sum of the frequencies due to internal events, fire events, and seismic events. The sum of all KRC frequencies is the total CDF. These results are presented in Table 3. Note that the total CDF is approximately 2.3 times the internal events CDF.

Table 3
Key Release Categories for Internal and External Events

	Internal Events (Note 1)		Fire (Notes 2, 4)		Seismic (Notes 3, 4)		Total	
KRC	Frequency	Percentage	Frequency	Percentage	Frequency	Percentage	Frequency	Percentage
R21	9.65E-06	72.56%	5.08E-06	72.56%	7.26E-06	72.56%	2.20E-05	72.56%
R17L	8.54E-07	6.42%	4.50E-07	6.42%	6.42E-07	6.42%	1.95E-06	6.42%
R11I	5.18E-07	3.90%	2.73E-07	3.90%	3.90E-07	3.90%	1.18E-06	3.90%
R11IF	5.17E-07	3.89%	2.72E-07	3.89%	3.89E-07	3.89%	1.18E-06	3.89%
R17LU	4.94E-07	3.71%	2.60E-07	3.71%	3.71E-07	3.71%	1.13E-06	3.71%
R20	3.93E-07	2.96%	2.07E-07	2.96%	2.96E-07	2.96%	8.95E-07	2.96%
R17U	3.23E-07	2.43%	1.70E-07	2.43%	2.43E-07	2.43%	7.36E-07	2.43%
R01DI	2.19E-07	1.65%	1.15E-07	1.65%	1.65E-07	1.65%	4.99E-07	1.65%
R22	9.64E-08	0.72%	5.07E-08	0.72%	7.25E-08	0.72%	2.20E-07	0.72%
R04IF	8.84E-08	0.66%	4.65E-08	0.66%	6.65E-08	0.66%	2.01E-07	0.66%
R19	5.35E-08	0.40%	2.82E-08	0.40%	4.03E-08	0.40%	1.22E-07	0.40%
R01IF	3.51E-08	0.26%	1.85E-08	0.26%	2.64E-08	0.26%	8.00E-08	0.26%
R02IF	2.18E-08	0.16%	1.15E-08	0.16%	1.64E-08	0.16%	4.97E-08	0.16%
R03IF	1.35E-08	0.10%	7.09E-09	0.10%	1.01E-08	0.10%	3.07E-08	0.10%
R04	6.87E-09	0.05%	3.62E-09	0.05%	5.17E-09	0.05%	1.57E-08	0.05%
R18	5.63E-09	0.04%	2.96E-09	0.04%	4.23E-09	0.04%	1.28E-08	0.04%
R03I	4.20E-09	0.03%	2.21E-09	0.03%	3.16E-09	0.03%	9.57E-09	0.03%
R04UIF	2.25E-09	0.02%	1.18E-09	0.02%	1.69E-09	0.02%	5.13E-09	0.02%
R01UIF	9.63E-10	0.01%	5.07E-10	0.01%	7.24E-10	0.01%	2.19E-09	0.01%
R05LIF	6.78E-10	0.01%	3.57E-10	0.01%	5.10E-10	0.01%	1.54E-09	0.01%
R03UIF	5.88E-10	0.00%	3.10E-10	0.00%	4.42E-10	0.00%	1.34E-09	0.00%
R06IF	4.33E-10	0.00%	2.28E-10	0.00%	3.26E-10	0.00%	9.87E-10	0.00%
R05IF	7.99E-11	0.00%	4.21E-11	0.00%	6.01E-11	0.00%	1.82E-10	0.00%
R03	1.97E-11	0.00%	1.04E-11	0.00%	1.48E-11	0.00%	4.50E-11	0.00%
R01I	1.06E-11	0.00%	5.56E-12	0.00%	7.94E-12	0.00%	2.41E-11	0.00%
R06LIF	1.03E-11	0.00%	5.44E-12	0.00%	7.77E-12	0.00%	2.35E-11	0.00%
Total	1.33E-05	100.00%	7.00E-06	100.00%	1.00E-05	100.00%	3.03E-05	100.00%

3.03E-05	CDF

Note 1: KRC Frequencies for internal events from the PSA level II model.
Note 2: Fire-related total KRC frequency from reference 8.
Note 3: Seismic-induced total KRC frequency calculated in section 6.1.3.
Note 4: Individual KRC frequencies for fire and seismic are calculated as the product of the total and the internal event KRC percentages.

6.2 Population Dose

KRCs are mapped to NUREG-1150 Accident Progression Bins (APBs) in Table C-1 of reference 10. In turn, each APB has an associated representative WBN population dose (person-rem), provided in Table C-4 of reference 10. The WBN population doses are based on equivalent doses for the Sequoyah Nuclear Plant (SQN), adjusted for meteorology and population distributional differences between SQN and WBN. Reference 16 states that "the staff concludes that the conversion of the WBN Plant release categories into the SQN Plant APBs appears to have been performed properly and is, therefore, acceptable."

Note that the doses documented in reference 10 are for the 1980 population surrounding WBN, and must be adjusted to the current population. Rather than evaluate the population based upon available census data, the scaling factor of 1.41 from reference 16, representing the projected WBN 50-mile population in 2035, will be used.

The mapping of KRCs to APBs is detailed in Table 4. Table 4a provides a description of each APB.

Table 4
KRCs mapped to APBs

KRC	Frequency	Percentage	NUREG-1150 Accident Progression Bins (Note 1)									
			1	2	3	4	5	6	7	8	9	10
R21	2.20E-05	72.56%								2.20E-05		
R17L	1.95E-06	6.42%					1.95E-06					
R11I	1.18E-06	3.90%					1.18E-06					
R11IF	1.18E-06	3.89%					1.18E-06					
R17LU	1.13E-06	3.71%					1.13E-06					
R20	8.95E-07	2.96%							8.95E-07			
R17U	7.36E-07	2.43%						7.36E-07				
R01DI	4.99E-07	1.65%			4.99E-07							
R22	2.20E-07	0.72%										2.20E-07
R04IF	2.01E-07	0.66%				2.01E-07						
R19	1.22E-07	0.40%							1.22E-07			
R01IF	8.00E-08	0.26%			8.00E-08							
R02IF	4.97E-08	0.16%			4.97E-08							
R03IF	3.07E-08	0.10%		3.07E-08								
R04	1.57E-08	0.05%		1.57E-08								
R18	1.28E-08	0.04%							1.28E-08			
R03I	9.57E-09	0.03%		9.57E-09								
R04UIF	5.13E-09	0.02%				5.13E-09						
R01UIF	2.19E-09	0.01%			2.19E-09							
R05LIF	1.54E-09	0.01%			1.54E-09							
R03UIF	1.34E-09	0.00%				1.34E-09						
R06IF	9.87E-10	0.00%			9.87E-10							
R05IF	1.82E-10	0.00%			1.82E-10							

Table 4
KRCs mapped to APBs

KRC	Frequency	Percentage	NUREG-1150 Accident Progression Bins (Note 1)									
			1	2	3	4	5	6	7	8	9	10
R03	4.50E-11	0.00%				4.50E-11						
R01I	2.41E-11	0.00%			2.41E-11							
R06LIF	2.35E-11	0.00%			2.35E-11							
Total	3.03E-05	1.00E+00	0.00E+00	5.59E-08	6.34E-07	2.08E-07	5.43E-06	7.36E-07	1.03E-06	2.20E-05	0.00E+00	2.20E-07

Representative WBN Population Dose person-rem (Note 2)	3.90E+05	1.85E+05	3.18E+05	3.41E+05	6.86E+04	2.14E+04	4.07E+05	1.98E+02	2.41E+05	1.43E+02

Representative WBN Population Dose person-rem (Note 3)	5.50E+05	2.61E+05	4.48E+05	4.81E+05	9.67E+04	3.02E+04	5.74E+05	2.79E+02	3.40E+05	2.02E+02

Note 1: from Table C-1, Reference 10, except KRCs R05IF, R05LIF, R06IF, R06LIF
KRCs R05IF, R05LIF, R06IF, R06LIF from Section 4.9, Reference 1

Note 2: from Table C-4, Reference 10.

Note 3: Corrected for estimated year 2035 population (factor of 1.41) from reference 16.

Table 4a
NUREG-1150 Accident Progression Bins (Note 1)

Number	Description
1	VB, early CF (during CD)
2	VB, alpha, early CF (at VB)
3	VB > 200 psi, early CF (at VB)
4	VB < 200 psi, early CF (at VB)
5	VB, late CF
6	VB, BMT, very late CF
7	Bypass
8	VB, no CF
9	No VB, early CF (during CD)
10	no VB

Note 1: from Table C-1, Reference 10.
Note 2: Vessel Breach (VB), Containment Failure (CF)
Core Damage (CD), Basemat Melt-Through (BMT)

To follow the guidance of reference 15, WBN population doses must be allocated to each EPRI accident class (classes are defined in Table 4c), except classes 3a and 3b. Doses associated with each KRC is taken from Table 4. Each KRC is allocated to an EPRI accident class using the definition of each KRC contained in section 4.9 of reference 1. If multiple KRCs are allocated to a single EPRI accident class, the resultant dose is a frequency-weighted average across the applicable KRCs. Doses for classes 3a and 3b are as defined in reference 15. Results of this allocation are detailed in Table 4b. EPRI accident classes are described in Table 4c.

Note that the reference 15 method maps class 1 (containment intact) accident sequences into class 3a and 3b. The method conservatively ignores the fact that some accident sequences will lead to a large early release regardless of the existence of a pre-existing leak.

Table 4d provides a summary of the dose calculations.

Table 4b
KRCs mapped to EPRI Class

KRC	Frequency	Percentage	EPRI Accident Class (Note 1)								
			1	2	3a	3b	4	5	6	7	8
R21	2.20E-05	72.56%	2.20E-05								
R17L	1.95E-06	6.42%								1.95E-06	
R11I	1.18E-06	3.90%								1.18E-06	
R11IF	1.18E-06	3.89%								1.18E-06	
R17LU	1.13E-06	3.71%								1.13E-06	
R20	8.95E-07	2.96%									8.95E-07
R17U	7.36E-07	2.43%								7.36E-07	
R01DI	4.99E-07	1.65%								4.99E-07	
R22	2.20E-07	0.72%	2.20E-07								
R04IF	2.01E-07	0.66%								2.01E-07	
R19	1.22E-07	0.40%									1.22E-07
R01IF	8.00E-08	0.26%								8.00E-08	
R02IF	4.97E-08	0.16%								4.97E-08	
R03IF	3.07E-08	0.10%								3.07E-08	
R04	1.57E-08	0.05%								1.57E-08	
R18	1.28E-08	0.04%									1.28E-08
R03I	9.57E-09	0.03%								9.57E-09	
R04UIF	5.13E-09	0.02%								5.13E-09	
R01UIF	2.19E-09	0.01%								2.19E-09	
R05LIF	1.54E-09	0.01%								1.54E-09	
R03UIF	1.34E-09	0.00%								1.34E-09	
R06IF	9.87E-10	0.00%								9.87E-10	
R05IF	1.82E-10	0.00%								1.82E-10	
R03	4.50E-11	0.00%								4.50E-11	
R01I	2.41E-11	0.00%								2.41E-11	
R06LIF	2.35E-11	0.00%								2.35E-11	
Total	3.03E-05	100.00%	2.22E-05	0.00E+00						7.06E-06	1.03E-06

Note 1: Class designation from Table 1, Reference 15

Table 4c
EPRI Accident Classes (Note 1)

Number	Description
1	Containment intact, accident sequences do not lead to failure, not affected by changes to ILRT leak testing frequencies
2	Failure of isolation system to operate from common cause or power failure; not affected by changes to ILRT leak testing frequencies.
3a	Small pre-existing leak in containment structure or liner, identifiable by ILRT; affected by ILRT testing frequency
3b	Large pre-existing leak in containment structure or liner, identifiable by ILRT; affected by ILRT testing frequency
4	Type B tested components fail to seal, not affected by ILRT leak testing frequencies
5	Type C tested components fail to seal, not affected by ILRT leak testing frequencies
6	Failure to isolate due to valves failing to stroke closed, not affected by ILRT testing frequency, low probability
7	Failure induced by severe accident phenomena, not affected by ILRT testing frequency.
8	Containment Bypass, not affected by ILRT testing frequency.

Note 1: from Table 1, Reference 15.

Table 4d
Baseline Dose
mapped to EPRI Class

class	Description	Baseline Dose (person-rem/event)	Basis
1	Containment Intact	2.78E+02	weighted average of APBs 8 and 10 (Note 1)
2	Isolation Failures, common cause		no populated release categories mapped to this EPRI Accident Class
3a	Small pre-existing Leak	2.78E+03	10 La per reference 15
3b	Large pre-existing Leak	9.74E+03	35 La per reference 15
4	Type B tested components		N/A per reference 15
5	Type C tested components		N/A per reference 15
6	Isolation Failures, redundant valves		N/A per reference 15
7	Severe Accident Phenomena	1.34E+05	weighted average of APBs 1-6, 9
8	Bypass	5.74E+05	APB 7

Note 1: This baseline dose is set equal to
1 La.

6.3 Containment Failure Probability

6.3.1 Pre-existing

The probability of containment failure such that the failure would be detectable by an ILRT but not LLRTs is developed in reference 15 for EPRI accident classes 3a and 3b, and will not be repeated here. Those values are entered into Table 5d as source data.

6.3.2 Corrosion

NRC review of previous ILRT test extension requests has set the precedent that failures described above as "pre-existing" exclude containment failures due to corrosion.

NEI guidance, provided in reference 15, does not address a separate corrosion failure mechanism.

A corrosion model acceptable to the NRC was developed in reference 23. This model, as applied to WBN, has the following features:

1. Corrosion failures are represented as a failure rate that increases with time. The rate doubling time was assumed to be 5 years. The base failure rate was assumed to be the average failure rate for years 6 through 10, inclusive.
2. Industry events can be classified as either "small" (class 3a) or "large" (class 3b) and are recorded by existing reporting mechanisms, such as LERs and Inspection Notices.
3. Success data begins in September 1996 when 10 CFR 50.55a started requiring visual inspections.
4. Recorded industry events are not screened for applicability to WBN. This is a very conservative treatment since most events have been associated with construction errors at a concrete-liner interface, and the WBN SCV is freestanding.
5. Recorded industry events are not screened for applicability to this corrosion model. This is a conservative treatment because most events have been associated with construction errors at a concrete-liner interface and are therefore more representative of an infant mortality mechanism than a wear-out mechanism.
6. For "large" failures in which no industry events have been recorded, 0.5 failures are assumed for the purpose of generating a failure rate.
7. The probability that a visual inspection fails to identify a flaw for inspectable areas is 5%.
8. The probability that a visual inspection fails to identify a flaw for un-inspectable areas is 100%.

9. The exposure time for corrosion failures is assumed to be T/2, where T is the ILRT interval. This is a conservative treatment for accelerating failure mechanisms.
10. Flaws in the SCV underneath the basemat are assumed not to be detectable via ILRT (reference 25).

Table 5 summarizes the input data for the corrosion model. Appendix B contains a list of industry events involving primary containment corrosion. Those assessed as a failure are so indicated.

Table 5a and 5b document the calculation of the base failure rates for class 3a and 3b events.

Table 5c documents the time-dependent failure rates for small and large events. Note that the average failure rate for years 6 to 10 is set to the base rate. The rate for the previous period is just ½ of the base rate. Rates for succeeding periods are accelerated at 2x per five years.

Table 5d summarizes the pre-existing, corrosion, and total failure probabilities as a function of the ILRT frequency. The total failure probability is used in subsequent calculations.

Table 5
Corrosion Model Parameters

Value	Units	Parameter	Basis
5		Number of small corrosion events in industry	Appendix B (those events classified as failures)
0		Number of large corrosion events in industry	Reference 25
104		Number of operating nuclear units	Reference 27, page 38
31-Jan-2002		Date of Calvert Cliffs ILRT Extension Request	Reference 23
4	years	Calendar years from Calvert Cliffs ILRT Extension Request to present	
5.5	years	Data reporting period for corrosion events	Reference 23
20857	square feet	WBN Containment surface area -- dome	Appendix A
40443	square feet	WBN Containment surface area -- cylinder	Appendix A
61300	square feet	WBN Containment surface area -- total except basemat	sum of dome + cylinder, 1 side only
2800	square feet	WBN Containment surface area that is not inspectable (not including basemat)	Appendix A
no		Basemat liner failures detectable by ILRT?	Reference 25
5	years	Containment failure probability due to corrosion doubling time	Reference 23
6 - 10	year interval	Containment failure base rate anchor point	Reference 23
5	percent	Failure probability for visually detecting containment corrosion damage to inspectable area	Reference 23
100	percent	Failure probability for visually detecting containment corrosion damage to un-inspectable area	Reference 23

Table 5a
Containment Failure Frequency due to Corrosion
Base Rate for Small Through-Wall Holes

5	Number of events at all US nuclear units
988	Exposure time for all US nuclear units (years)
5.06E-03	Frequency of small holes (events/year)
4.57E-02	Conditional Probability that hole occurs in un-inspectable location
9.54E-01	Conditional Probability that hole occurs in inspectable location
0.05	Conditional Probability that hole in inspectable location is not detected visually
9.34E-02	Conditional Probability that hole is not visually detected
4.73E-04	Frequency of small holes that are not visually detected (events/year)

Note 1: conditional probability for hole location based upon the ratio of surface areas
Note 2: conditional probability that hole is not visually detected is based on 100% of un-inspectable area plus 5% of inspectable area

Table 5b
Containment Failure Frequency due to Corrosion
Base Rate for Large Through-Wall Holes

0	Number of events at all US nuclear units
988	Exposure time for all US nuclear units (years)
5.06E-04	Frequency of large holes (events/year)
4.57E-02	Conditional Probability that hole occurs in un-inspectable location
9.54E-01	Conditional Probability that hole occurs in inspectable location
0.05	Conditional Probability that hole in inspectable location is not detected visually
9.34E-02	Conditional Probability that hole is not visually detected
4.73E-05	Frequency of large holes that are not visually detected (events/year)

Note 1: Frequency of large events based upon 0.5 failures.

Table 5c
Containment Failure Frequency due to Corrosion
Containment Failure Frequency as a Function of Time

Year	Small	Large
1	1.57E-04	1.57E-05
2	1.85E-04	1.85E-05
3	2.17E-04	2.17E-05
4	2.55E-04	2.55E-05
5	2.99E-04	2.99E-05
6	3.51E-04	3.51E-05
7	4.04E-04	4.04E-05
8	4.64E-04	4.64E-05
9	5.33E-04	5.33E-05
10	6.12E-04	6.12E-05
11	7.03E-04	7.03E-05
12	8.07E-04	8.07E-05
13	9.27E-04	9.27E-05
14	1.07E-03	1.07E-04
15	1.22E-03	1.22E-04
16	1.41E-03	1.41E-04
17	1.61E-03	1.61E-04
18	1.85E-03	1.85E-04
19	2.13E-03	2.13E-04
20	2.45E-03	2.45E-04

r = yearly rate of return to double in 5 years
(1 + r)^5 = 2.0
r = (2.0)^0.20 - 1
1.49E-01

average frequency of small failures for years 6 - 10
4.73E-04
average frequency = (year 6 + year 7 + year 8 + year 9 + year 10)/5
x = year 6 frequency of small failures
average = ((x) + x*(1+r) + x*(1+r)^2 + x*(1+r)^3 + x*(1+r)^4)/5
x = 5*(average)/(1 + (1+r) + (1+r)^2 + (1+r)^3 + (1+r)^4)
3.51E-04

average frequency of large failures for years 6 - 10
4.73E-05
average frequency = (year 6 + year 7 + year 8 + year 9 + year 10)/5
x = year 6 frequency of large failures
average = ((x) + x*(1+r) + x*(1+r)^2 + x*(1+r)^3 + x*(1+r)^4)/5
x = 5*(average)/(1 + (1+r) + (1+r)^2 + (1+r)^3 + (1+r)^4)
3.51E-05

Table 5d
Containment Failure Probability as a Function of ILRT Frequency

		Corrosion		Pre-existing		Total	
ILRT Frequency	T/2 (months)	Small	Large	Small	Large	Small	Large
3/10 years	18.00	5.21E-04	5.21E-05	0.0270	0.0027	0.0275	0.0028
1/10 years	60.00	1.74E-03	1.74E-04	0.0900	0.0090	0.0917	0.0092
1/15 years	90.00	4.10E-03	4.10E-04	0.1350	0.0135	0.1391	0.0139
1/20 years	120.00	8.83E-03	8.83E-04	0.1800	0.0180	0.1888	0.0189

Note 1: corrosion probabilities are calculated as (average frequency)*(1/12)*(T/2)
Note 2: Small pre-existing failure probability calculated as (0.027)*(1/18)*(T/2)
Note 3: Large pre-existing failure probability calculated as (0.0027)*(1/18)*(T/2)
Note 4: T/2 is used as the exposure time for corrosion failures. This is a conservative treatment.
Note 5: the probabilities for small and large pre-existing failures for the 3/10 year ILRT frequency are from reference 15.

6.4 Accident Class Information as a Function of ILRT Frequency

EPRI accident class population doses (person-rem/reactor year) are calculated in Tables 6 through 6c for ILRT frequencies of 3/10 years, 1/10 years, 1/15 years, and 1/20 years, respectively. The dose per event is from Table 4d. The frequencies for classes 2, 7, and 8 are from Table 4b. The frequency for class 1 is per the guidance of reference 15. The frequencies for classes 3a and 3b are from Table 5d.

Table 6
Accident Class Information for ILRT Frequency of 3/10 years

class	Description	Dose (person-rem/event)	Frequency (events per reactor year)	Population Dose (person-rem/reactor year)
1	Containment Intact	2.78E+02	2.13E-05	5.93E-03
2	Isolation Failures, common cause		0.00E+00	0.00E+00
3a	Small pre-existing Leak	2.78E+03	8.34E-07	2.32E-03
3b	Large pre-existing Leak	9.74E+03	8.34E-08	8.13E-04
4	Type B tested components			
5	Type C tested components			
6	Isolation Failures, redundant valves			
7	Severe Accident Phenomena	1.34E+05	7.06E-06	9.46E-01
8	Bypass	5.74E+05	1.03E-06	5.91E-01
	Total		3.03E-05	1.55E+00

Note 1: Frequency for Class 1 set equal to Frequency of Class 1 from Table 4b less Frequency of Classes 3a and 3b. This maintains the correct CDF.
Note 2: Frequency for Class 3a and 3b from Table 5d.
Note 3: This case is refered to as the baseline case in Reference 15.

Table 6a
Accident Class Information for ILRT Frequency of 1/10 years

class	Description	Dose (person-rem/event)	Frequency (events per reactor year)	Population Dose (person-rem/reactor year)
1	Containment Intact	2.78E+02	1.91E-05	5.33E-03
2	Isolation Failures, common cause		0.00E+00	0.00E+00
3a	Small pre-existing Leak	2.78E+03	2.78E-06	7.74E-03
3b	Large pre-existing Leak	9.74E+03	2.78E-07	2.71E-03
4	Type B tested components			
5	Type C tested components			
6	Isolation Failures, redundant valves			
7	Severe Accident Phenomena	1.34E+05	7.06E-06	9.46E-01
8	Bypass	5.74E+05	1.03E-06	5.91E-01
	Total		3.03E-05	1.55E+00

Note 1: Frequency for Class 1 set equal to Frequency of Class 1 from Table 4b less Frequency of Classes 3a and 3b. This maintains the correct CDF.
Note 2: Frequency for Class 3a and 3b from Table 5d.

Table 6b
Accident Class Information for ILRT Frequency of 1/15 years

class	Description	Dose (person-rem/event)	Frequency (events per reactor year)	Population Dose (person-rem/reactor year)
1	Containment Intact	2.78E+02	1.76E-05	4.89E-03
2	Isolation Failures, common cause		0.00E+00	0.00E+00
3a	Small pre-existing Leak	2.78E+03	4.21E-06	1.17E-02
3b	Large pre-existing Leak	9.74E+03	4.21E-07	4.11E-03
4	Type B tested components			
5	Type C tested components			
6	Isolation Failures, redundant valves			
7	Severe Accident Phenomena	1.34E+05	7.06E-06	9.46E-01
8	Bypass	5.74E+05	1.03E-06	5.91E-01
	Total		3.03E-05	1.56E+00

Note 1: Frequency for Class 1 set equal to Frequency of Class 1 from Table 4b less Frequency of Classes 3a and 3b. This maintains the correct CDF.
Note 2: Frequency for Class 3a and 3b from Table 5d.

Table 6c
Accident Class Information for ILRT Frequency of 1/20 years

class	Description	Dose (person-rem/event)	Frequency (events per reactor year)	Population Dose (person-rem/reactor year)
1	Containment Intact	2.78E+02	1.59E-05	4.43E-03
2	Isolation Failures, common cause		0.00E+00	0.00E+00
3a	Small pre-existing Leak	2.78E+03	5.72E-06	1.59E-02
3b	Large pre-existing Leak	9.74E+03	5.72E-07	5.57E-03
4	Type B tested components			
5	Type C tested components			
6	Isolation Failures, redundant valves			
7	Severe Accident Phenomena	1.34E+05	7.06E-06	9.46E-01
8	Bypass	5.74E+05	1.03E-06	5.91E-01
	Total		3.03E-05	1.56E+00

Note 1: Frequency for Class 1 set equal to Frequency of Class 1 from Table 4b less Frequency of Classes 3a and 3b. This maintains the correct CDF.
Note 2: Frequency for Class 3a and 3b from Table 5d.

6.5 Population Dose as a Function of ILRT Frequency

Changes in population dose as a function of ILRT frequency, expressed both in absolute terms (person-rem/reactor year) and as a percentage, are documented in Table 7.

Table 7
Class 3a + 3b Population Dose as a function of ILRT Frequency

ILRT Frequency	Dose for Class 3a and 3b (person-rem/reactor year)	Dose for Class 3a and 3b (percent of total)	Delta Class 3a and 3b dose from baseline case (person-rem/reactor year)	Delta Class 3a and 3b dose from baseline case (percent of total)
3/10 years	3.13E-03	0.20%	0.00E+00	0.00%
1/10 years	1.04E-02	0.67%	7.31E-03	0.47%
1/15 years	1.58E-02	1.02%	1.27E-02	0.81%
1/20 years	2.15E-02	1.38%	1.84E-02	1.17%

6.6 LERF as a Function of ILRT Frequency

Changes in LERF as a function of ILRT frequency, expressed in absolute terms (events/reactor year), are documented in Table 8. For the purposes of this calculation, LERF is set equal to the frequency of EPRI class 3b. Note that LERF so calculated represents only large early release accident sequences that are affected by the ILRT frequency.

Because the delta LERF values are greater than 1E-7, a calculation of total LERF is provided in Table 8a. This value of LERF represents all accident sequences.

Table 8
LERF as a function of ILRT Frequency

ILRT Frequency	Frequency of Class 3b (events per reactor year)	LERF (events per reactor year)	Delta LERF from baseline case (events per reactor year)
3/10 years	8.34E-08	8.34E-08	0.00E+00
1/10 years	2.78E-07	2.78E-07	1.95E-07
1/15 years	4.21E-07	4.21E-07	3.38E-07
1/20 years	5.72E-07	5.72E-07	4.89E-07

Table 8a
Total LERF

KRC	Total KRC Frequency	LERF
R21	2.20E-05	
R17L	1.95E-06	
R11I	1.18E-06	
R11IF	1.18E-06	1.18E-06
R17LU	1.13E-06	
R20	8.95E-07	
R17U	7.36E-07	
R01DI	4.99E-07	4.99E-07
R22	2.20E-07	
R04IF	2.01E-07	2.01E-07
R19	1.22E-07	1.22E-07
R01IF	8.00E-08	8.00E-08
R02IF	4.97E-08	4.97E-08
R03IF	3.07E-08	3.07E-08
R04	1.57E-08	1.57E-08
R18	1.28E-08	1.28E-08
R03I	9.57E-09	9.57E-09
R04UIF	5.13E-09	5.13E-09
R01UIF	2.19E-09	2.19E-09
R05LIF	1.54E-09	
R03UIF	1.34E-09	1.34E-09
R06IF	9.87E-10	
R05IF	1.82E-10	
R03	4.50E-11	4.50E-11
R01I	2.41E-11	2.41E-11
R06LIF	2.35E-11	

Total all KRCs	3.03E-05	2.21E-06

Total LERF	2.78E-06

Note 1: Total KRC Frequency from Table 3.
Note 2: Allocation of KRCs to LERF per section 4.9 of reference 1.
Note 3: Total LERF is the sum of LERF for the total all KRCs plus LERF from Table 8 for ILRT frequency of 1/20 years.

6.7 CCFP as a Function of ILRT Frequency

CCFP and percentage changes in the CCFP as a function of ILRT frequency are documented in Table 9.

Table 9
CCFP as a function of ILRT Frequency

ILRT Frequency	Frequency of Class 1 (events per reactor year)	Frequency of Class 3a (events per reactor year)	CCFP (percent)	Delta CCFP (percent)
3/10 years	2.13E-05	8.34E-07	26.99%	0.00%
1/10 years	1.91E-05	2.78E-06	27.63%	0.64%
1/15 years	1.76E-05	4.21E-06	28.10%	1.12%
1/20 years	1.59E-05	5.72E-06	28.60%	1.61%

CDF	3.03E-05

Note 1: CCFP (percent) = 1- (frequency of class 1 + frequency of class 3a)/CDF

7.0 Summary of Results

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

Table 10
Summary of Results

	Delta LERF per reactor year	Delta CCFP percent	Delta Population Dose person-rem	Delta Population Dose percent
ILRT Change from 3/10 years to 1/10 years	1.95E-07	0.64%	7.31E-03	0.47%
ILRT Change from 3/10 years to 1/15 years	3.38E-07	1.12%	1.27E-02	0.81%
ILRT Change from 3/10 years to 1/20 years	4.89E-07	1.61%	1.84E-02	1.17%
ILRT Change from 1/10 years to 1/15 years	1.44E-07	0.47%	5.39E-03	0.34%

Table 10a
Summary of Results w/ 10% safety factor

	Delta LERF	Delta CCFP	Delta Population Dose	Delta Population Dose
	per reactor year	percent	person-rem	percent
ILRT Change from 3/10 years to 1/10 years	2.14E-07	0.71%	8.04E-03	0.52%
ILRT Change from 3/10 years to 1/15 years	3.72E-07	1.23%	1.40E-02	0.90%
ILRT Change from 3/10 years to 1/20 years	5.38E-07	1.77%	2.02E-02	1.29%
ILRT Change from 1/10 years to 1/15 years	1.58E-07	0.52%	5.93E-03	0.38%

8.0 Supporting Graphics

Figure 1 is provided in section 6.1.3.

9.0 Conclusions

The following conservative treatments are used within this calculation. The numerical results are therefore deemed conservative.

- The EPRI interim methodology assumes that all events classified as 3a or 3b are mapped from class 1 (containment intact). This ignores events in which a pre-existing containment leak is masked by other containment failure modes.
- The frequency of fire-induced CDF is based on the FIVE screening analysis performed for the IPEEE.
- The 50 mile surrounding population is assumed to be equal to the projected value at year 2035.
- The corrosion model does not screen industry events for applicability to the containment design or to a degradation mechanism. All industry events are assumed to be applicable to WBN.
- The numerical results include a 10% margin.

The increase in LERF when the frequency of an ILRT is decreased from 1/10 years to 1/15 years is 1.58E-07. This value is a "small" increase in LERF (less than 1E-6 and greater than 1E-7) per Regulatory Guide 1.174 (reference 5). A small increase in LERF is acceptable if the total LERF is shown to be below 1E-5 per reactor year. Table 8a documents the total LERF. The proposed ILRT extension is acceptable with respect to ΔLERF.

The change in the calculated CCFP is small, indicative of a proposed change that does not significantly challenge the principle of defense in depth.

The change in the calculated population dose is very small, and indicative of a proposed change that does not significantly increase risk to the public.

Based on these risk measures, the proposed ILRT frequency change is acceptable.

Appendix A
SCV Inspection Area

Watts Bar Steel Containment Vessel (SCV) Inspection Area

The Watts Bar Nuclear Plant (WBN) Unit 1 SCV surface area is estimated using Chicago Bridge & Iron Company (CBI) (Contract No. 75320) drawings (See References). The exterior surface areas of the dome (20857 sq.ft.) and cylinder (40443 sq.ft.) are added to determine the estimated exterior surface area for the Unit 1 SCV. The total Unit 1 SCV surface area is estimated as 61300 sq.ft.

Inaccessible Area for Inspection

The WBN Unit 1 SCV exterior insulation types and locations are identified in WBN Engineering Specification "N3M-936" Revision 4, "For Installation, Modification and Maintenance of Heat and Anti-Sweat Insulation", Section 4.10.2.3. These areas are from 54° to 126° between elevation 716'-0" and 747'-0".

Walkdown of the SCV also identified additional insulation locations from 50° to 126° between elevation 713'-0" and 716'-0" and from 50° to 54° between elevation 716'-0" and 733'-0".

The inaccessible surface areas for the WBN Unit 1 SCV are identified as areas of the exterior SCV surface with insulation and the shielding area around the fuel transfer penetration. The area below the floor of the embedded metal liner and concrete base slab is also inaccessible for the inspection.

The total inaccessible area for inspection including the shielding area around the fuel transfer penetration is estimated as 2800 sq.ft. The area below the floor of the embedded metal liner and concrete base slab is not included in the 2800 sq.ft.

Reference Drawings:

1. 1, 2-48N401 Rev. 1, "Structural Steel Containment Vessel Anchor Bolt Plan and Base Dets SH 1"
2. Chicago Bridge & Iron Company (CBI) (Contract No. 75320) Drawing No. 400 Rev 6 "Roof Plan View"
3. Chicago Bridge & Iron Company (CBI) (Contract No. 75320) Drawing No. 34 Rev 8 "Shell Ring No. 1, Assy. 34-A"
4. ISI-0503-C-03 Rev 1, "Watts Bar Nuclear Plant Unit 1 Metal Containment Penetrations & Elevations"
5. ISI-0503-C-04 Rev 1, "Watts Bar Nuclear Plant Unit 1 Metal Containment Penetrations & Elevations"
6. ISI-0503-C-05 Rev 1, "Watts Bar Nuclear Plant Unit 1 Metal Containment Penetrations & Elevations"

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Appendix B
Industry Primary Containment Failures due to Corrosion

Plant:	Davis Besse
Date:	July, 2002
Extent:	corrosion of free-standing shell where it meets floor. As-found containment thickness above minimum.
Cause:	no moisture barrier.
ID Method:	visual inspection
Reference:	Reference 30.
Failure:	no

Plant:	Sequoyah 2
Date:	May, 2002
Extent:	degraded coating and surface rust
Cause:	clogged floor drain
ID Method:	visual inspection
Reference:	Reference 30.
Failure:	no

Plant:	Dresden 2
Date:	November, 2001
Extent:	missing coating, corrosion area 2-4 inches wide, encircling drywell near floor. Degraded area within corrosion allowance.
Cause:	not reported
ID Method:	visual inspection
Reference:	Reference 30.
Failure:	no

Plant:	D. C. Cook 2
Date:	March, 2001
Extent:	Through-wall hole. Corrosion outside liner, near hole.
Cause:	Hole caused by construction error.

	Corrosion near hole caused by wood embedded in concrete.
ID Method:	visual during weld repair inspection
Reference:	Reference 30.
Failure:	yes
Size:	small

Plant:	D. C. Cook 1
Date:	February, 1988
Extent:	Corrosion, not through-wall. More than 60 pits in which as-found wall thickness less than minimum.
Cause:	Moisture barrier failure.
ID Method:	not reported.
Reference:	Reference 30.
Failure:	yes
Size:	small (size based upon reference 26)

Plant:	Surry 2
Date:	Fall, 2003
Extent:	degraded coating and rust at the junction of the metal liner and interior concrete floor. Not through-wall.
Cause:	failed moisture barrier.
ID Method:	visual inspection
Reference:	Reference 30
Failure:	no

Plant:	Palisades
Date:	October, 1999
Extent:	minor corrosion at floor-to-liner crevice.
Cause:	moisture barrier not installed.
ID Method:	not reported.
Reference:	Reference 30
Failure:	no

Plant:	North Anna 2
Date:	9/22/1999
Extent:	1 through-wall hole. LLRT passed.

Cause: Lumber embedded in concrete.
ID Method: visual inspection
Reference: Reference 23.
Failure: yes
Size: small

Plant: Brunswick 2
Date: May, 1999
Extent: 3 through-wall holes
Cause: Glove and wood embedded in concrete.
ID Method: not reported.
Reference: References 23, 30
Failure: yes
Size: small

Plant: Robinson 2
Date: December, 1996
Extent: degraded caulk, insulation, coating. Some corrosion of liner. As-found thickness greater than minimum.
Cause: degraded caulk
ID Method: visual inspection
Reference: Reference 30
Failure: no

Plant: Oyster Creek
Date: 12/8/1986
Extent: wall thinning, no through-holes reported
Cause: Contact with wet sand. Moisture from failed seals used during refueling.
ID Method: visual detection of water from drains
Reference: References 17, 28, 29.
Failure: yes, based upon reference 17.
Size: small