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10 CFR 50.90

RS-03-013

January 29, 2003

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

Subject: Clinton Power Station, Unit 1
Facility Operating License No. NPF-62
NRC Docket No. 50-461

Request for Amendment to Technical Specifications 3.6.5.1, "Drywell" and 5.5.13, "Primary Containment Leakage Rate Testing Program"

In accordance with 10 CFR 50.90, AmerGen Energy Company (AmerGen), LLC, hereby requests the following amendment to Appendix A, Technical Specifications (TS), of Facility Operating License No. NPF-62 for Clinton Power Station (CPS), Unit 1. Specifically, the proposed change will revise TS 3.6.5.1, "Drywell," Surveillance Requirement (SR) 3.6.5.1.3 to delay the performance of the next drywell bypass leakage test to no later than November 23, 2008. This request will also revise TS 5.5.13, "Primary Containment Leakage Rate Testing Program," to remove an exception which is no longer applicable and to reflect a one-time deferral of the primary containment Type A test to no later than November 23, 2008.

SR 3.6.5.1.3 establishes the performance-based criteria for the periodic conduct of the drywell bypass leakage test. When the measured leakage is maintained below the applicable drywell leakage limit acceptance criterion, the maximum extension of the test frequency to 120 months is permitted.

TS 5.5.13 establishes the leakage rate testing of the primary containment for CPS, Unit 1, as required by 10 CFR 50.54, "Conditions of licenses." paragraph (o) and, 10 CFR 50, Appendix J, "Primary Reactor Containment Leakage Testing for Water-Cooled Power Reactors," Option B, as modified by approved exemptions. Additionally, the testing is performed in accordance with the guidance contained in Regulatory Guide 1.163, "Performance-Based Containment Leak-Test Program," dated September 1995 as modified by the exceptions in TS 5.5.13.

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AmerGen is requesting this amendment in anticipation of an expected rule change to 10 CFR 50 that will extend the Type A testing frequency to at least 15 years. The drywell bypass leakage test is currently conducted at the same frequency, and uses much of the same equipment and plant lineups as the Type A test. For these reasons, it is also included in this request. Approval of this proposed change will allow sufficient time for this rule change to be processed and the appropriate revisions to be made to the CPS TS.

The information supporting the proposed TS changes is subdivided as follows.

Attachment 1 is the notarized affidavit.
Attachment 2 provides our evaluation supporting the proposed changes.
Attachment 3 contains the copies of the marked up TS pages.
Attachment 4 provides the retyped TS pages and Bases pages for information only.
Attachment 5 provides the risk assessment supporting the proposed changes.

The proposed TS change has been reviewed by the CPS Plant Operations Review Committee (PORC) and approved by the Nuclear Safety Review Board (NSRB) in accordance with the Quality Assurance Program.

AmerGen is notifying the State of Illinois of this application for amendment by transmitting a copy of this letter and its attachments to the designated State Official.

We request approval of the proposed change by November 1, 2003. Approval by this date will support planning for the upcoming refueling outage, currently scheduled for February 2004.

Should you have any questions concerning this submittal, please contact Mr. Timothy A. Byam at (630) 657-2804.

Sincerely,

Keith R. Jury

Keith R. Jury
Director-Licensing and Regulatory Affairs
Mid-West Regional Operating Group
AmerGen Energy Company, LLC

Attachments:

Attachment 1 Affidavit
Attachment 2 Evaluation of Proposed Changes
Attachment 3 Markup of Proposed Technical Specification Page Changes
Attachment 4 Retyped Pages for Technical Specification Changes and Bases Changes (for information only)

Attachment 5 Clinton Power Station Risk Assessment to Support ILRT (Type A) Interval Extension Request

cc: Regional Administrator – NRC Region III
NRC Project Manager, NRR – Clinton Power Station
NRC Senior Resident Inspector – Clinton Power Station
Office of Nuclear Facility Safety – Illinois Department of Nuclear Safety

ATTACHMENT 1
Affidavit

STATE OF ILLINOIS	**)**	
COUNTY OF DUPAGE	**)**	
IN THE MATTER OF	**)**	
AMERGEN ENERGY COMPANY, LLC	**)**	**Docket Number**
CLINTON POWER STATION, UNIT 1	**)**	**50-461**

SUBJECT: **Request for Amendment to Technical Specifications 3.6.5.1, "Drywell" and 5.5.13, "Primary Containment Leakage Rate Testing Program"**

AFFIDAVIT

I affirm that the content of this transmittal is true and correct to the best of my knowledge, information and belief.

Keith R. Jury

Keith R. Jury
Director – Licensing and Regulatory Affairs
Mid-West Regional Operating Group
AmerGen Energy Company, LLC

Subscribed and sworn to before me, a Notary Public in and

for the State above named, this 29th day of

January, 2003.

Timothy Byam

Notary Public

"OFFICIAL SEAL"
TIMOTHY A. BYAM
NOTARY PUBLIC STATE OF ILLINOIS
COMMISSION EXPIRES 12/04/05

1.0 INTRODUCTION

In accordance with 10 CFR 50.90, AmerGen Energy Company (AmerGen), LLC, hereby requests the following amendment to Appendix A, Technical Specifications (TS), of Facility Operating License No. NPF-62 for Clinton Power Station (CPS), Unit 1. Specifically, the proposed change will revise TS 3.6.5.1, "Drywell," Surveillance Requirement (SR) 3.6.5.1.3 to delay the performance of the next drywell bypass leakage rate test (DBLRT) to no later than November 23, 2008. This request will also revise TS 5.5.13, "Primary Containment Leakage Rate Testing Program," to remove an exception which is no longer applicable and to reflect a one-time deferral of the primary containment Type A test to no later than November 23, 2008.

AmerGen is requesting this amendment in anticipation of an expected rule change to 10 CFR 50 that will extend the Type A testing frequency to at least 15 years. The DBLRT is currently conducted at the same frequency, and uses much of the same equipment and plant lineups as the Type A test. For these reasons, it is also included in this request. Approval of this proposed change will allow sufficient time for this rule change to be processed and the appropriate revisions to be made to the CPS TS.

2.0 DESCRIPTION OF PROPOSED AMENDMENT

The proposed change modifies the frequency of SR 3.6.5.1.3 to delay the next required performance of the DBLRT to no later than November 23, 2008. This request also deletes from TS 5.5.13 the expired exception that allowed deferral of the leakage rate testing of the primary containment penetration 1MC-042 until the seventh refueling outage. In addition, this proposed change adds one new exception to TS 5.5.13 that modifies the schedule for the next Type A test for CPS, Unit 1, to a 15-year interval. The proposed wording associated with these changes is identified below in bold type.

The note associated with the SR 3.6.5.1.3 120 month Frequency is modified as follows.

NOTES

1. **The next required performance of this SR may be delayed to November 23, 2008.**
2. SR 3.0.2 is not applicable for extensions > 12 months.

TS 5.5.1.3 is revised as follows.

5.5.13 <u>Primary Containment Leakage Rate Testing Program</u>

A program shall be established to implement the leakage rate testing of the primary 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 1.163, "Performance-Based Containment Leak-Test Program," dated September 1995 as modified by the following exceptions: (1) Bechtel Topical Report BN-TOP-1 is also an acceptable option for performance of Type A tests, and **(2) NEI 94-01 – 1995, Section 9.2.3: The first Type A test performed after November 23, 1993 shall be performed no later than November 23, 2008.**

3.0 BACKGROUND

CPS is a General Electric BWR/6 plant with a Mark III containment design. The Mark III containment design is a single-barrier pressure containment and a multi-barrier fission containment system consisting of the drywell and primary containment. The suppression pool is an annular pool of demineralized water between the drywell and the outer primary containment boundary. This pool covers the horizontal vent openings in the drywell to maintain a water seal between the drywell interior and the remainder of the containment volume. The primary containment is penetrated by access, piping and electrical penetrations.

The leaktightness of the drywell is periodically verified by performance of the DBLRT. This test ensures that the measured drywell bypass leakage is bounded by the safety analysis assumptions. The drywell integrity is further verified by a number of additional tests, including drywell airlock door seal leakage tests, overall drywell airlock leakage tests, drywell isolation valve tests and periodic visual inspections of exposed accessible interior and exterior drywell surfaces. Additional confidence that significant degradation in the drywell integrity has not developed is provided by the periodic qualitative assessment of drywell performance. This assessment was credited in the NRC's acceptance of the current performance-based surveillance frequency, approved with Amendment 106 for CPS (Reference 8).

The integrity of the CPS primary containment penetrations and isolation valves is verified through Type B and Type C local leak rate tests (LLRTs) and the overall leak-tight integrity of the primary containment is verified by a Type A integrated leak rate test (ILRT) as required by 10 CFR 50, Appendix J, "Primary Reactor Containment Leakage Testing for Water-Cooled Power Reactors," (Reference 1). These tests are performed to verify the essentially leak-tight characteristics of the primary containment at the design basis accident pressure.

Revisions to 10CFR50, Appendix J (i.e., Option B) allow individual plants to extend the Type A ILRT surveillance testing requirements from three-in-ten years to at least once per 10 years. The revised Type A frequency is based on an acceptable performance history defined as two consecutive periodic Type A tests at least 24 months apart in which the calculated performance leakage is less than the maximum allowable primary containment leakage rate, L_a, of 1.0 L_a. CPS proposed implementation of 10CFR50 Appendix J Option B in Reference 2. The NRC subsequently approved implementation at CPS in Amendment 105 (Reference 3).

The basis for the current 10-year test interval is provided in Section 11.0 of Nuclear Energy Institute (NEI) 94-01, Revision 0, "Industry Guideline for Implementing Performance-Based Option of 10 CFR 50, Appendix J," dated July 26, 1995 (Reference 4). This document was generated during development of the performance-based Option B to Appendix J. Section 11.0 of NEI 94-01 states that NUREG-1493, "Performance-

Based Containment Leak Test Program," dated September 1995, provides the technical basis to support rulemaking to revise leakage rate testing requirements contained in Option B to Appendix J. The basis consisted of qualitative and quantitative assessments of the risk impact (in terms of increased public dose) associated with a range of extended leakage rate test intervals. To supplement the NRC's rulemaking basis, NEI undertook a similar study. The results of that study are documented in Electric Power Research Institute (EPRI) Research Project Report TR-104285, "Risk Impact Assessment of Revised Containment Leak Rate Testing Intervals."

Option B, "Performance-Based Requirements," of Appendix J to 10 CFR 50 requires that a Type A test be conducted at a periodic interval based on historical performance of the overall primary containment system. CPS TS 5.5.13 requires that a program be established to comply with the primary containment leakage rate testing requirements of 10 CFR 50.54(o) and 10 CFR 50, Appendix J, Option B, as modified by exemptions. Additionally, this program is established in accordance with the guidelines contained in Regulatory Guide (RG) 1.163, "Performance-Based Containment Leak-Test Program," dated September 1995 (Reference 5). RG 1.163 endorses, with certain exceptions, NEI 94-01, Revision 0.

NEI 94-01 specifies for Type A tests, an initial test interval of 48 months and allows an extension of the interval to 10 years, based on two consecutive successful tests. CPS is currently on a 10-year testing interval.

The proposed change modifies the frequency of SR 3.6.5.1.3 and adds one exception to TS 5.5.13 to allow a one-time deferral from the guidelines contained in RG 1.163 and NEI 94-01 regarding the Type A test interval. The proposed change will extend the next drywell bypass leakage and Type A tests for CPS to a 15-year interval.

The last drywell bypass leakage and Type A tests for CPS were successfully performed on November 23, 1993. The proposed change will require the next drywell bypass leakage and Type A tests for CPS to be performed by November 23, 2008.

In addition to the above change, CPS is proposing to remove an exception from TS 5.5.13 that is no longer applicable. In Reference 6, CPS proposed to change TS 5.5.13 to allow the performance of the LLRT on the primary containment penetration for the reactor pressure vessel head spray piping (1MC-042) to be deferred until the seventh refueling outage (C1R07). This proposed change was requested as a result of the conditions that prevented the performance of the LLRT on this penetration during the sixth refueling outage. The NRC approved this proposed change as Amendment 121 (Reference 7). CPS has since successfully performed the required LLRT on containment penetration 1MC-042 during the seventh and eighth refueling outages. Therefore, the exception identified in TS 5.5.13 is no longer applicable.

4.0 REGULATORY REQUIREMENTS & GUIDANCE

10 CFR 50.36, "Technical specifications." provides the regulatory requirements for the content required in a licensee's TS.

10 CFR 50, Appendix J, Option B, Section V.B, "Implementation," specifies that the regulatory guide or other implementing documents used to develop a performance-based leakage testing program must be included, by general reference, in the plant TS. Additionally, deviations from guidelines endorsed in a regulatory guide are to be submitted as a revision to the plant TS.

5.0 TECHNICAL ANALYSIS

5.1 Primary Containment Pressure Suppression Testing

The function of the Mark III containment is to isolate and contain fission products released from the Reactor Coolant System (RCS) following a design basis Loss of Coolant Accident (LOCA) and to confine the postulated release of radioactive material to within limits. The Mark III containment consists of the drywell and the primary containment.

The drywell houses the reactor pressure vessel, the reactor coolant recirculating loops, and branch connections of the RCS, which have isolation valves at the primary containment boundary. The function of the drywell is to maintain a pressure boundary that channels steam from a LOCA to the suppression pool, where it is condensed. Air forced from the drywell is released into the primary containment through the suppression pool. The suppression pool is a concentric open container of water with a stainless steel liner that is located at the bottom of the primary containment. The suppression pool is designed to absorb the decay heat and sensible heat released during a reactor blowdown from safety/relief valve (SRV) discharges or from a LOCA. The SRVs are located on the main steam lines between the reactor vessel and the first isolation valve within the drywell. Each SRV discharges steam through a discharge line to a point below the minimum water level in the suppression pool.

The function of the primary containment is to isolate and contain fission products released from the RCS following a design basis accident (DBA) and to confine the postulated release of radioactive material to within limits. The primary containment consists of a steel lined, reinforced concrete vessel, which surrounds the RCS and provides an essentially leak-tight barrier against an uncontrolled release of radioactive material to the environment. Additionally, this structure provides shielding from the fission products that may be present in the primary containment atmosphere following accident conditions.

The drywell post-LOCA vacuum relief system consists of four vacuum relief lines that are located between the drywell and the containment. Each vacuum relief line contains two vacuum relief valves in series which are designed to start opening when the drywell pressure is approximately 0.2 psid less than the containment and will be fully opened when this differential pressure is 0.5 psid. As soon as the drywell pressure drops below the containment pressure, the drywell vacuum breakers will open and noncondensible gases from the containment will flow back into the drywell until the pressures in the two regions equalize. The drywell post-LOCA vacuum relief system must function in the event of a large break LOCA to control rapid weir wall overflow that could cause drag and impact loadings on essential equipment and systems in the drywell above the weir wall. In addition, the drywell vacuum relief subsystems are the means by which noncondensibles are transferred from the primary containment back to the drywell during operation of the hydrogen mixing compressors. The system is required to assist in hydrogen dilution but not to protect the structural integrity of the drywell following a large break LOCA.

During a LOCA, the drywell pressure increases rapidly due to the injection of the break flow. The peak drywell pressure occurs during the vent-clearing phase of the transient as suppression pool water is being cleared from the vents. Following vent clearing, the drywell pressure decreases as the break flow decreases. During the reactor pressure vessel depressurization phase, most of the noncondensible gases initially in the drywell are forced into the containment. However, following the depressurization the noncondensibles will redistribute between the drywell and containment via the vacuum breaker system. This redistribution takes place as steam in the drywell is condensed by the relatively cool Emergency Core Cooling System (ECCS) water that is beginning to cascade from the break causing the drywell pressure to decrease.

The concept of the pressure suppression reactor containment is that any steam released from the primary system will be condensed by the suppression pool and will not have an opportunity to produce a significant pressurization effect on the containment. This is accomplished by channeling the steam into the suppression pool through the vent system. Steam that enters the containment airspace directly from the drywell airspace will bypass the condensing capabilities of the suppression pool, thereby causing a higher containment pressure response. Therefore, drywell bypass leakage must be minimized to prevent overpressurization of the primary containment during the drywell pressurization phase of a LOCA. This requires periodic testing of the drywell bypass leakage (i.e., TS SR 3.6.5.1.3), confirmation that the drywell airlock is leak tight (i.e., TS SR 3.6.5.1.1 and SR 3.6.5.1.2), operability of the drywell isolation valves (i.e., TS SR 3.6.5.3.1, SR 3.6.5.3.2, and SR 3.6.5.3.3), and confirmation that the drywell vacuum relief valves are closed (i.e., TS SR 3.6.5.6.1). The DBLRT, TS SR 3.6.5.1.3, verifies that the total bypass leakage between the drywell airspace and containment airspace is consistent with accident assumptions.

In Amendment No. 106 for CPS (Reference 8), the NRC approved a revision to TS SR 3.6.5.1.3 that revised the scheduling of the DBLRT. The amendment requires that the DBLRT be conducted at least once every 10 years on a performance-based frequency. In the event that a test is performed with the bypass leakage greater than the bypass leakage limit the test frequency becomes once every 48 months. Following two consecutive tests with bypass leakage greater than the bypass leakage limit, the test frequency is every 24 months until two consecutive tests are less than or equal to the bypass leakage limit. The last DBLRT was successfully conducted in November 1993. The next performance of this surveillance test is required by November 2003.

The CPS DBLRT is performed in conjunction with the ILRT due to the commonality of test equipment and system lineups necessary to support both tests. As a result, the proposed changes to TS 5.5.13 will require modification of the SR 3.6.5.1.3 test frequency for the DBLRT as the next required performance of this test is not consistent with the proposed changes to the ILRT test interval.

The DBLRT experience to date at CPS has been good. CPS documented the DBLRT history in the amendment request (Reference 9) which resulted in Amendment 106. A total of six drywell leakage rate tests have been performed and, as documented in References 8 and 9, except for the initial low pressure drywell leakage test, the calculated drywell bypass leakage has been less than 1% of the allowable limit and 0.1% of the design limit.

As documented in Reference 8, CPS committed to perform a qualitative assessment of the drywell leak tightness at least once per operating cycle. This assessment provides added assurance that the drywell has not seriously degraded between performances of the bypass leakage rate tests. By checking for gross leakage, this assessment provides an indication of the ability of the drywell to perform its design function. CPS also makes a continuing, qualitative on-line assessment of drywell integrity. This is possible due to the existence of small instrument air system leaks and from normal operation of pneumatic controls and operators in the drywell that pressurize the drywell, creating a differential pressure between the drywell and primary containment. As documented in References 8 and 9, the drywell is being vented approximately once per day when pressure approaches the upper TS limit of 1.0 psid. Pressurization rates have remained consistent with those addressed in Reference 9. It has been concluded that as long as the drywell continues to pressurize, regardless of the rate, an unacceptable leakage path does not exist and drywell integrity is assured. Any significant change in the frequency of drywell venting would result in investigation and correction of the cause. CPS continues to perform these assessments and the proposed changes to SR 3.6.5.1.3 and TS 5.5.13 do not change these commitments. Additionally, the proposed changes do not modify the acceptance criteria for the DBLRT or any of the other tests performed to ensure the containment pressure suppression function is maintained.

Therefore, it has been determined that the proposed changes to SR 3.6.5.1.3 and TS 5.5.13 do not modify commitments made to qualitatively assess the drywell leak tightness or test acceptance criteria of the primary containment pressure suppression components and systems.

5.2 10CFR 50, Appendix J, Option B

The testing requirements of 10 CFR 50, Appendix J, provide assurance that leakage through the primary containment, including systems and components that penetrate the primary containment, does not exceed allowable leakage rate values specified in the TS and Bases. The allowable leakage rate is determined so that the leakage assumptions in the safety analyses are not exceeded. The limitation of primary containment leakage provides assurance that the primary containment would perform its design function following an accident, up to and including the design basis accident.

10 CFR 50, Appendix J, was revised effective October 26, 1995. The purpose of this revision was to allow licensees to choose primary containment leakage testing under Option A "Prescriptive Requirements" or Option B. CPS Amendment No. 105 (Reference 3) was issued to permit implementation of 10 CFR 50, Appendix J, Option B. TS 5.5.13 currently requires the establishment of a Primary Containment Leakage Rate Testing Program in accordance with 10 CFR 50.54(o) and 10 CFR 50, Appendix J, Option B, as modified by approved exemptions. This program implements the guidelines contained in RG 1.163 which specifies a method acceptable to the NRC for complying with Option B by approving the use of NEI 94-01, subject to several regulatory positions stated in the RG.

Exceptions to the requirements of RG 1.163 are permitted by 10CFR 50, Appendix J, Option B, as discussed in Section V.B, "Implementation." Therefore, this application does not require an exemption from 10CFR 50, Appendix J, Option B.

The adoption of Option B performance-based primary containment leakage rate testing program by CPS did not alter the basic method by which Appendix J leakage rate testing is performed or its acceptance criteria. Adoption of Option B did alter the test frequency of primary containment leakage in Type A, B, and C tests. The required test frequency is based upon an evaluation which uses the "as found" leakage history to determine the frequency for leakage testing which provides assurance that leakage limits will be maintained.

The allowed frequency for Type A testing is based, in part, upon a generic evaluation documented in NUREG-1493, "Performance-Based Leak-Test Program." NUREG-1493 made the following observations with regard to changing the test frequency.

- Reducing the Type A testing frequency to once per 20 years was found to lead to an imperceptible increase in risk. The estimated increase in risk is small because Type A tests identify only a few potential leakage paths that cannot be identified by Type B and C testing, and the leaks that have been found by Type A tests have only been marginally above the existing requirements. Given the insensitivity of risk to primary containment leakage rate, and the small fraction of leakage detected solely by Type A testing, increasing the interval between Type A testing has minimal impact on public risk.

- While Type B and C tests identify the vast majority (i.e., greater than 95%) of all potential leakage paths, performance-based alternatives are feasible without significant risk impacts. Since leakage contributes less than 0.1 percent of overall risk under existing requirements, the overall effect is very small.

The required surveillance frequency for Type A testing in NEI 94-01 is at least once per 10 years based on an acceptable performance history (i.e., two consecutive periodic Type A tests at least 24 months apart or refueling cycles where the calculated performance leakage rate was less than 1.0 L_a) and consideration of the performance factors in NEI 94-01, Section 11.3. The proposed changes do not impact the CPS leakage rate test program as developed in accordance with NUREG-1493.

5.3 CPS Integrated Leak Rate Testing History

Type A testing is performed to verify the integrity of the containment structure in its LOCA configuration. Industry test experience has demonstrated that Type B & C testing detect a large percentage of containment leakages and that the percentage of containment leakages detected only by integrated containment leakage testing is very small. Results of CPS's previous ILRTs below demonstrate the CPS containment structure remains essentially a leak-tight barrier and represents minimal risk to increased leakage. These plant specific results support the conclusions of NUREG-1493.

10 CFR 50 Appendix J, Option B Test Information

The performance leakage rate for the four Type A tests conducted at CPS are calculated as follows.

Performance leakage rate = ➊ + ➋ + ➌, where

➊ is the Type A upper confidence limit (UCL),
➋ is the As-left minimum pathway leakage rate for all Type B and C pathways not challenged during the Type A test, and
➌ is the As-left minimum pathway local leakage rate for leakage pathways that were isolated during the performance of the Type A test because of excessive leakage.

Preoperational Test Conducted 12/28/1985-01/02/1986

Total Time UCL	➊	0.2930%/day	$0.451L_a$.
	➋	+0.0436%/day	$0.067L_a$.
	➌	+0.0097%/day	$0.015L_a$.
Performance Leakage Rate:		0.3463%/day	$0.533L_a$.

Preoperational Test Conducted 11/03/1986-11/04/1986

Total Time UCL	➊	0.2875%/day	$0.442L_a$.
	➋	+0.0058%/day	$0.009L_a$.
	➌	+0.0000%/day	$0.000L_a$.
Performance Leakage Rate:		0.2933%/day	$0.451L_a$.

First Periodic Test Conducted 02/15/1991-02/16/1991

Total Time UCL	➊	0.2209%/day	$0.340L_a$.
	➋	+0.0082%/day	$0.013L_a$.
	➌	+0.0000%/day	$0.000L_a$.
Performance Leakage Rate:		0.2291%/day	$0.353L_a$.

Second Periodic Test Conducted 11/22/1993-11/23/1993

Total Time UCL	➊	0.2089%/day	$0.321L_a$.
	➋	+0.0115%/day	$0.018L_a$.
	➌	+0.0000%/day	$0.000L_a$.
Performance Leakage Rate:		0.2204%/day	$0.339L_a$.

Testing Interval Evaluation:

Acceptable Type A performance history is defined as completion of two consecutive periodic tests where the calculated performance leakage rate is less than 1.0 L_a (0.65%/day). Elapsed time between the first and last tests in a series of consecutive satisfactory tests used to determine performance shall be at least 24 months.

All four Type A test performance leakage rates have met the performance criteria of 1.0 L_a (0.65%/day). The two most recent periodic tests were separated by an elapsed time of 33 months.

5.4 Type B and C Testing

Type B and C testing assures containment penetrations such as flanges, sealing mechanisms and containment isolation valves are essentially leak tight. Type B and C tests identify the vast majority of all potential leakage paths.

The Type B and C testing requirements will not be changed as a result of the extended ILRT interval.

5.5 Containment Inspections

a) Appendix J Visual Inspections

The Appendix J Program requires visual inspections to be performed of accessible interior and exterior surfaces of the containment system for structural problems that may affect either the containment structural leakage integrity or performance of the Type A Test. These examinations are conducted prior to initiating a Type A test, and during two other refueling outages before the next Type A test based on a 10-year frequency (Reference 5).

These requirements will not be changed as a result of the extended ILRT interval.

b) Containment Inservice Inspection Program (CISI)

CPS performs a comprehensive primary containment inspection to the requirements of American Society of Mechanical Engineers (ASME) Section XI, "Inservice Inspection," Subsections IWE, "Requirements for Class MC and Metallic Liners of Class CC Components of Light-Water Cooled Power Plants," and Subsection IWL, "Requirements of Class CC Concrete Components of Light-Water Cooled Power Plants." The CPS Containment Inservice Inspection Program (CISI) began development in 1996 and the initial inspections were completed in September 2001. The components subject to Subsection IWE and IWL requirements are those, which make up the containment structure, its leak-tight barrier (including integral attachments) and those that contribute to its structural integrity. Specifically included are Class MC pressure retaining components, including metallic shell and penetration liners of Class CC pressure retaining components, and their integral attachments. The ASME Code Inspection Plan was developed in accordance with the requirements of the 1992 Edition with the 1992 Addenda of the ASME Boiler and Pressure Vessel Code, Section XI, Division 1, Subsections IWE and IWL, as modified by NRC final rulemaking to 10 CFR 50.55a published in the Federal Register on August 8, 1996.

The initial ISI inspections of the CPS Metal / Concrete Containment have been completed. Various indications were observed, documented, evaluated and determined to be acceptable. No areas of the containment liner surfaces require augmented examination. No loss of structural integrity of primary containment was observed.

There will be no change to the schedule for these inspections as a result of the extended ILRT interval.

c) Containment Coatings Inspections

A program to maintain containment coatings was developed to meet the requirements of Regulatory Guide 1.54, "Quality Assurance Requirements for Protective Coatings Applied to Water-Cooled Nuclear Power Plants," Revision 0. This program is implemented using CPS procedure 1080.01, "CPS Protective Coating Program." Each refueling outage, a preventive maintenance activity to inspect the protective coatings in the containment building is performed. The most-recent inspection, performed in April 2002, found the condition of the containment coatings in very good to excellent condition. The inspection requirements of the containment coatings program will not be changed as a result of the extended ILRT interval.

5.6 Information Notice 92-20

Information Notice 92-20, "Inadequate Local Leak Rate Testing," discussed the inadequate local leak rate testing of two-ply stainless steel bellows. A modification (i.e., FH-030) was installed on the Inclined Fuel Transfer System (IFTS) containment penetration in 1995 to eliminate the concern raised by Information Notice 92-20 and to allow bellows testing to be performed using Type B test methods. The testing assembly provides a means of applying a static test pressure to the bellows to ensure containment integrity will be maintained in accordance with 10 CFR 50, Appendix J.

5.7 Risk Information

The risk analysis performed to support this submittal is contained in Attachment 5. The CPS Level 1 and 2 PSA (Rev. 3) used as input to this analysis is characteristic of the as-built, as-operated plant.

The risk analysis determined that the proposed changes result in:

- Increasing the current 10-year ILRT and DBLRT interval to 15 years results in an insignificant increase in total population dose rate of 0.48 percentage points.

- The increase in the Large Early Release Frequency (LERF) risk measure is small, a 1.4E-7/yr increase. This LERF increase is categorized as right on the border between Region III and Region II per NRC Regulatory Guide 1.174, "An Approach for Using

Probabilistic Risk Assessment in Risk-Informed Decisions on Plant-Specific Changes to the Licensing Basis."

- Likewise, the conditional containment failure probability (CCFP) increases insignificantly by 0.5 percentage points.

Regulatory Guide 1.174 provides guidance for determining the risk impact of plant-specific changes to the licensing basis. Regulatory Guide 1.174 defines very small changes in risk as resulting in increases of CDF below 10^{-6}/yr and increases in LERF below 10^{-7}/yr. Since the ILRT and DBLRT do not impact CDF, the relevant criterion is LERF. The increase in LERF resulting from a change in the ILRT and DBLRT interval from 10 years to 15 years (using the change in the EPRI Category 3b frequency per the NEI Interim Guidance) is 1.4E-7/yr. Guidance in Regulatory Guide 1.174 defines small changes in LERF as above 10^{-7}/yr and less than 10^{-6}/yr. Therefore, increasing the Clinton ILRT and DBLRT interval from 10 to 15 years results in a small change in risk, and is an acceptable plant change from a risk perspective.

Per Regulatory Guide 1.174, when the calculated increase in LERF due to the proposed plant change is in the range of 1E-7 to 1E-6 per reactor year (Region II, "small change" in risk), the risk assessment must also reasonably show that the total LERF is less than 1E-5.

Per the Clinton internal events PSA (Rev. 3) documentation, the Clinton LERF due to internal event accidents is 2.63E-7/yr. Therefore, the total LERF for Clinton of 2.63E-7/yr is significantly less than the Regulatory Guide 1.174 acceptance guideline of 1E-5/yr.

The change in CCFP is also calculated as an additional risk measure to demonstrate the impact on defense-in-depth. The change in CCFP is found to be very small (0.5% increase) and represents a negligible change in the Clinton defense-in-depth.

The change in population dose rate is also reported consistent with previously approved ILRT interval extension requests. The change in population dose rate from the current once in 10 year ILRT and DBLRT frequency to a once in 15 year frequency is an insignificant 0.48% increase.

Based on the above, the proposed changes to SR 3.6.5.1.3 and TS 5.5.13 will continue to provide assurance that leakage through the CPS drywell and primary containment will not exceed allowable leakage rate values specified in the TS and Bases, and that the containment features will continue to perform their design function following an accident, up to and including the design basis accident.

6.0 REGULATORY ANALYSIS

10 CFR 50.36, "Technical specifications." provides the regulatory requirements for the content required in a licensee's TS. 10 CFR 50.36(c)(3) "Surveillance requirements." requires tests, calibrations or inspections to assure that the necessary quality of systems and components is maintained, that facility operation will be within safety limits, and that the limiting conditions for operation will be met. 10 CFR 50.36(c)(5), "Administrative controls." requires provisions relating to organization and management, procedures, recordkeeping, review and audit, and reporting necessary to assure operation of the facility in a safe manner will be included in a licensee's TS.

Additionally, 10 CFR 50, Appendix J, Section V.B, "Implementation," specifies that the regulatory guide or other implementing documents used to develop a performance-based leakage testing program must be included, by general reference, in the plant's TS. Additionally, deviations from guidelines endorsed in a regulatory guide are to be submitted as a revision to the plant's TS.

The proposed changes will revise TS SR 3.6.5.1.3 to defer the next performance of the DBLRT. These changes will also revise TS Section 5.5.13 to reflect a one-time deferral from the program requirements for the Type A test for CPS. This deferral represents an exception to the guidelines contained in Regulatory Guide 1.163 and NEI 94-01. Thus, the proposed changes are consistent with the requirements of 10 CFR 50.36(c)(3), 10 CFR 50.36(c)(5) and 10 CFR 50, Appendix J, Section V.B.

Additionally, in accordance with 10 CFR 50, Appendix J, Section V.B, the proposed changes to CPS TS do not require a supporting request for an exemption to Option B of Appendix J, in accordance with 10 CFR 50.12.

7.0 NO SIGNIFICANT HAZARDS CONSIDERATION

AmerGen Energy Company (AmerGen), LLC, has evaluated the proposed changes to the Technical Specifications (TS) for Clinton Power Station (CPS), Unit 1, and has determined that the proposed changes do not involve a significant hazards consideration and is providing the following information to support a finding of no significant hazards consideration.

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

Response: No

The proposed changes will revise TS 3.6.5.1, "Drywell," Surveillance Requirement SR 3.6.5.1.3 to delay the performance of the next drywell bypass leakage rate test (DBLRT) to no later than November 23, 2008. This request will also will revise CPS TS 5.5.13, "Primary Containment Leakage Rate Testing Program" to reflect a one-time deferral of the primary containment Type A test to no later than November 23, 2008. The current Type A test interval of 10 years, based on past performance, would be extended on a one-time basis to 15 years from the last Type A test. In addition, AmerGen is proposing to delete from TS 5.5.13 the expired exception that allowed deferral of the leakage rate testing of the primary containment penetration 1MC-042 until the seventh refueling outage.

The drywell houses the reactor pressure vessel, the reactor coolant recirculating loops, and branch connections of the Reactor Coolant System (RCS), which have isolation valves at the primary containment boundary. The function of the drywell is to maintain a pressure boundary that channels steam from a Loss of Coolant Accident (LOCA) to the suppression pool, where it is condensed. Air forced from the drywell is released into the primary containment through the suppression pool. The suppression pool is a concentric open container of water with a stainless steel liner that is located at the bottom of the primary containment. The suppression pool is designed to absorb the decay heat and sensible heat released during a reactor blowdown from safety/relief valve (SRV) discharges or from a LOCA.

The function of the Mark III containment is to isolate and contain fission products released from the RCS following a design basis LOCA and to confine the postulated release of radioactive material to within limits. The test interval associated with the drywell bypass leakage and Type A testing is not a precursor of any accident previously evaluated. Therefore, extending these test intervals on a one-time basis from 10 years to 15 years does not result in an increase in the probability of occurrence of an accident. The successful performance history of the drywell bypass leakage and Type A testing provides assurance that the CPS drywell and primary containment will not exceed allowable leakage rate values specified in the TS and will continue to perform its design function following an accident. The risk assessment of the proposed changes has concluded that there is an insignificant increase in total population dose rate and an insignificant increase in the conditional containment failure probability.

Therefore, the proposed changes do not involve a significant increase in the probability or consequences of an accident previously evaluated.

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

Response: No

The proposed changes for a one-time extension of the drywell bypass leakage and Type A tests and deletion of an expired local leak rate test exception for CPS, will not affect the control parameters governing unit operation or the response of plant equipment to transient and accident conditions. The proposed changes do not introduce any new equipment or modes of system operation. No installed equipment will be operated in a new or different manner. As such, no new failure mechanisms are introduced.

Therefore, the proposed changes do not create the possibility of a new or different kind of accident from any previously evaluated.

Does the change involve a significant reduction in a margin of safety?

Response: No

CPS is a General Electric BWR/6 plant with a Mark III containment system. The Mark III containment design is a single-barrier pressure containment and a multi-barrier fission containment system consisting of the drywell and primary containment. The drywell houses the reactor pressure vessel, the reactor coolant recirculating loops, and branch connections of the RCS, which have isolation valves at the primary containment boundary. The function of the drywell is to maintain a pressure boundary that channels steam from a LOCA to the suppression pool, where it is condensed. The suppression pool is an annular pool of demineralized water between the drywell and the outer primary containment boundary. This pool covers the horizontal vent openings in the drywell to maintain a water seal between the drywell interior and the remainder of the containment volume. The primary containment consists of a steel-lined, reinforced concrete vessel, which surrounds the RCS and provides an essentially leak-tight barrier against an uncontrolled release of radioactive material to the environment. Additionally, this structure provides shielding from the fission products that may be present in the primary containment atmosphere following accident conditions. The primary containment is penetrated by access, piping and electrical penetrations.

The integrity of the drywell is periodically verified by performance of the DBLRT. This test ensures that the measured drywell bypass leakage is bounded by the safety analysis assumptions. The drywell integrity is further verified by a number of additional tests, including drywell airlock door seal leakage tests, overall drywell airlock leakage tests and periodical visual inspections of exposed accessible interior and exterior drywell surfaces. Additional confidence that significant degradation in the drywell leaktightness has not developed is provided by the periodic qualitative assessment of drywell performance.

The integrity of the primary containment penetrations and isolation valves is verified through Type B and Type C local leak rate tests (LLRTs) and the overall leak-tight integrity of the primary containment is verified by a Type A integrated leak rate test (ILRT) as required by 10 CFR 50, Appendix J, "Primary Reactor Containment Leakage Testing for Water-Cooled Power Reactors." These tests are performed to verify the essentially leak-tight characteristics of the primary containment at the design basis accident pressure. The proposed changes for a one-time extension of the drywell bypass leakage and Type A tests and deletion of an expired local leak rate test exception for CPS, do not effect the method for drywell or containment testing or the test acceptance criteria.

Therefore, the proposed changes do not involve a significant reduction in a margin of safety.

Based upon the above, AmerGen 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.

8.0 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, "Standards for Protection Against Radiation," 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, "Criterion for categorical exclusion; identification of licensing and regulatory actions eligible for categorical exclusion or otherwise not requiring environmental review." Paragraph (c)(9). Therefore, pursuant to 10 CFR 51.22, Paragraph (b), no environmental impact statement or environmental assessment need be prepared in connection with the proposed amendment.

9.0 PRECEDENT

The proposed amendment incorporates into the CPS TS changes that are similar to changes approved by the NRC for Susquehanna Steam Electric Station on March 8, 2002.

10.0 REFERENCES

1. 10CFR50 Appendix J, "Primary Reactor Containment Leakage Testing for Water-Cooled Power Reactors"

2. Letter from W. Connell (Illinois Power Company) to U. S. NRC, "Clinton Power Station Proposed Amendment of Facility Operating License No. NPF-62 (LS-95-014)," dated May 1, 1996

3. Letter from U. S. NRC to M. W. Lyon (Illinois Power Company), "Issuance of Amendment No. 105 to Facility Operating License No. NPF-62 – Clinton Power Station, Unit 1 (TAC No. M95321)," dated June 21, 1996

4. Nuclear Energy Institute (NEI) 94-01, "Industry Guideline for Implementing Performance-Based Option of 10CFR50 Appendix J," Revision 0, dated July 26, 1995

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

6. Letter from W. G. MacFarland (Illinois Power Company) to U. S. NRC, "Clinton Power Station Proposed Amendment of Facility Operating License No. NPF-62 (LS-98-011)," dated October 5, 1998

7. Letter from U. S. NRC to J. V. Sipek (Illinois Power Company), "Issuance of Amendment No. 121 to Facility Operating License No. NPF-62 – Clinton Power Station, Unit 1 (TAC No. MA3754)," dated March 8, 1999

8. Letter from U. S. NRC to M. W. Lyon (Illinois Power Company), "Issuance of Amendment No. 106 to Facility Operating License No. NPF-62 – Clinton Power Station, Unit 1 (TAC No. M94889)," dated September 4, 1996

9. Letter from W. Connell to U. S. NRC, "Clinton Power Station Proposed Amendment of Facility Operating License No. NPF-62 (LS-96-001)," dated February 22, 1996

ATTACHMENT 3

MARKUP OF PROPOSED TECHNICAL SPECIFICATION PAGE CHANGES

<u>Revised TS Pages</u>
3.6-54b
5.0-16a

SURVEILLANCE REQUIREMENTS (continued)

SURVEILLANCE		FREQUENCY
SR 3.6.5.1.3	Verify bypass leakage is less than or equal to the bypass leakage limit. However, during the first unit startup following bypass leakage testing performed in accordance with this SR, the acceptance criterion is ≤ 10% of the drywell bypass leakage limit.	24 months following 2 consecutive tests with bypass leakage greater than the bypass leakage limit until 2 consecutive tests are less than or equal to the bypass leakage limit AND 48 months following a test with bypass leakage greater than the bypass leakage limit AND -----NOTES----- **1. The next required performance of this SR may be delayed to November 23, 2008.** **2. SR 3.0.2 is not applicable for extensions > 12 months.** ------------- 120 months

(continued)

5.5 Programs and Manuals (continued)

5.5.13 Primary Containment Leakage Rate Testing Program

A program shall be established to implement the leakage rate testing of the primary 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 1.163, "Performance-Based Containment Leak-Test Program," dated September 1995,as modified by the following exceptions: (1) Bechtel Topical Report BN-TOP-1 is also an acceptable option for performance of Type A tests, and (2) ~~the leakage rate testing of primary containment penetration 1MC-042 may be deferred until the seventh refueling outage.~~**NEI 94-01 - 1995, Section 9.2.3: The first Type A test performed after November 23, 1993 shall be performed no later than November 23, 2008.**

The peak calculated containment internal pressure for the design basis loss of coolant accident, P_a, is 9.0 psig.

The maximum allowable primary containment leakage rate L_a, at P_a, shall be 0.65% of primary containment air weight per day.

Leakage Rate acceptance criteria are:

a. Primary containment leakage rate acceptance criterion is $\leq 1.0\ L_a$. During the first unit startup following testing in accordance with this program, the leak rate acceptance criteria are $\leq 0.60\ L_a$ for the Type B and Type C tests and $\leq 0.75\ L_a$ for Type A tests;

b. Air lock testing acceptance criteria are:

   1) Overall air lock leakage rate is ≤ 5 scfh when tested at $\geq P_a$,

   2) For each door, leakage rate is ≤ 5 scfh when the gap between door seals is pressurized $\geq P_a$.

The provisions of SR 3.0.2 do not apply to the test frequencies specified in the Primary Containment Leakage Rate Testing Program.

The provisions of SR 3.0.3 are applicable to the Primary Containment Leakage Rate Testing Program.

(continued)

ATTACHMENT 4

RETYPED PAGES
FOR
TECHNICAL SPECIFICATION CHANGES
AND
BASES CHANGES (FOR INFORMATION ONLY)

<u>Retyped TS Pages</u>
3.6-54b
5.0-16a

<u>Retyped Bases Pages (for information only)</u>
B 3.6-105a

SURVEILLANCE REQUIREMENTS (continued)

SURVEILLANCE		FREQUENCY
SR 3.6.5.1.3	Verify bypass leakage is less than or equal to the bypass leakage limit. However, during the first unit startup following bypass leakage testing performed in accordance with this SR, the acceptance criterion is ≤ 10% of the drywell bypass leakage limit.	24 months following 2 consecutive tests with bypass leakage greater than the bypass leakage limit until 2 consecutive tests are less than or equal to the bypass leakage limit AND 48 months following a test with bypass leakage greater than the bypass leakage limit AND -----NOTES----- 1. The next required performance of this SR may be delayed to November 23, 2008. 2. SR 3.0.2 is not applicable for extensions > 12 months. -------------- 120 months

(continued)

5.5 Programs and Manuals (continued)

5.5.13 Primary Containment Leakage Rate Testing Program

A program shall be established to implement the leakage rate testing of the primary 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 1.163, "Performance-Based Containment Leak-Test Program," dated September 1995,as modified by the following exceptions: (1) Bechtel Topical Report BN-TOP-1 is also an acceptable option for performance of Type A tests, and (2) NEI 94-01 - 1995, Section 9.2.3: The first Type A test performed after November 23, 1993 shall be performed no later than November 23, 2008.

The peak calculated containment internal pressure for the design basis loss of coolant accident, P_a, is 9.0 psig.

The maximum allowable primary containment leakage rate L_a, at P_a, shall be 0.65% of primary containment air weight per day.

Leakage Rate acceptance criteria are:

a. Primary containment leakage rate acceptance criterion is $\leq 1.0\ L_a$. During the first unit startup following testing in accordance with this program, the leak rate acceptance criteria are $\leq 0.60\ L_a$ for the Type B and Type C tests and $\leq 0.75\ L_a$ for Type A tests;

b. Air lock testing acceptance criteria are:

   1) Overall air lock leakage rate is ≤ 5 scfh when tested at $\geq P_a$,

   2) For each door, leakage rate is ≤ 5 scfh when the gap between door seals is pressurized $\geq P_a$.

The provisions of SR 3.0.2 do not apply to the test frequencies specified in the Primary Containment Leakage Rate Testing Program.

The provisions of SR 3.0.3 are applicable to the Primary Containment Leakage Rate Testing Program.

(continued)

SURVEILLANCE REQUIREMENTS

SR 3.6.5.1.3

The analyses in Reference 1 are based on a maximum drywell bypass leakage. This Surveillance ensures that the actual drywell bypass leakage is less than or equal to the acceptable $A/\sqrt{k}$ design value of 1.0 ft^2 assumed in the safety analysis. As left drywell bypass leakage, prior to the first startup after performing a required drywell bypass leakage test, is required to be ≤ 10% of the drywell bypass leakage limit. At all other times between required drywell leakage rate tests, the acceptance criteria is based on the design $A/\sqrt{k}$. At the design $A/\sqrt{k}$ the containment temperature and pressurization response are bounded by the assumptions of the safety analysis. One drywell air lock door is left open during each drywell bypass leakage test such that each drywell air lock door is leak tested during at least every other drywell bypass leakage test. This ensures that the leakage through the drywell air lock is properly accounted for in the measured bypass leakage and that each air lock door is tested periodically.

This Surveillance is performed at least once every 10 years (120 months) on a performance based frequency. The Frequency is consistent with the difficulty of performing the test, risk of high radiation exposure, and the remote possibility that sufficient component failures will occur such that the drywell bypass leakage limit will be exceeded. **This Frequency is modified by a note that allows for a one-time deferral of this surveillance until November 23, 2008.** If during the performance of this required Surveillance the drywell bypass leakage is determined to be greater than the leakage limit, the Surveillance Frequency is increased to at least once every 48 months. If during the performance of the subsequent consecutive Surveillance the drywell bypass leakage is determined to be less than or equal to the drywell bypass leakage limit, the 10-year Frequency may be resumed. If during the performance of the subsequent consecutive Surveillance the drywell bypass leakage is determined to be greater than the drywell bypass leakage limit, the Surveillance Frequency is increased to at least once every 24 months. The 24-month Frequency must be maintained until the drywell bypass leakage is determined to

(continued)

ATTACHMENT 5

CLINTON POWER STATION RISK ASSESSMENT TO SUPPORT ILRT (TYPE A) INTERVAL EXTENSION REQUEST

ERIN *Engineering And Research, Inc*

CLINTON POWER STATION RISK ASSESSMENT TO SUPPORT ILRT (TYPE A) INTERVAL EXTENSION REQUEST

ERIN Report No. C46702024-4924

Principal Contributors

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Prepared for

Amergen
Clinton Power Station

DECEMBER 2002

CLINTON POWER STATION

RISK ASSESSMENT TO SUPPORT ILRT (TYPE A) INTERVAL EXTENSION REQUEST

ERIN Report No. C46702024-4924

Prepared by: E T Burns Date: 12/26/02

Reviewed by: S. C. Teagarden Date: 12/27/02

Approved by: [illegible] Date: 12/27/02

Accepted by: Anthony Haller Date: 1-14-03

Revisions:

Rev.	Description	Preparer/Date	Reviewer/Date	Approver/Date
1	EDITORIAL CHANGES to CLARIFY the CALCULATIONS	E T Burns 1/13/03	S. G. Teagarden 1/13/03	[illegible] 1/13/03

TABLE OF CONTENTS

EXECUTIVE SUMMARY

The risk impact of a one-time extension of the Clinton Power Station (CPS) integrated leak rate test (ILRT) interval from once in 10 years to once in 15 years is evaluated. The results demonstrate that a change in the ILRT test interval from 10 years to 15 years represents a "very small" impact on risk, as defined by Reg. Guide 1.174.

The CPS ILRT risk assessment uses CPS specific information to calculate the existing risk profile and the changes to the risk profile for radionuclide releases. The ex-plant consequences are then calculated by adjusting the ex-plant consequences from a surrogate Mark III plant (as allowed by the NEI Interim Guidance). The evaluation utilizes NUREG/CR-4551 50 mile dose risk for a Mark III plant (Grand Gulf). The total dose risk is subdivided into accident progression bins (APBs) based on NUREG/CR-4551. The dose risk for each APB is adjusted to account for population differences, containment leakage rate, and power level for applicability to Clinton. The Clinton Level 2 release modes are sorted to match the APBs and determine the Clinton specific accident frequency for each APB.

The Clinton accident frequency and dose for each APB is then converted to an equivalent EPRI category for consideration of the effects of ILRT interval changes. Only three of the EPRI categories are affected by ILRT interval changes (1, 3a, and 3b). Table ES-1 summarizes the results.

A one-time Drywell Bypass Test (DWBT) interval extension is also requested to be consistent with the change in the ILRT interval extension from 10 years to 15 years. Therefore, the incremental assessment of the risk change is performed for the case in which both the ILRT and DWBT intervals are extended from 10 years to 15 years. This is reported in Appendix C of this report.

The results demonstrate a small impact on risk associated with the one time extension of the ILRT and DWBT interval to 15 years. In addition, the DWBT interval extension by itself represents a very small impact on risk.

The evaluation approach for the assessment of the risk is based on EPRI-TR-104285, NEI Interim Guidance (dated November 2001), and previous ILRT risk assessment submittals.

Three risk metrics are evaluated using the CPS Rev 3 internal events PRA model for each of the assessments of changing the test interval from the currently approved 10 years to 15 years:

Risk Metrics	ILRT Interval Extension Risk Increase	ILRT and DWBT Interval Extension Risk Increase
Change in Large Early Release Frequency (LERF)[(1)]	9.3E-8/yr	1.4E-7/yr
Change in conditional containment failure probability	0.3%	0.5%
Change in population dose rate (person-rem/yr)	0.03	0.04

[(1)] It is reemphasized that the radionuclide release (e.g., CsI release fraction) calculated for Class 3b is significantly below that which has been attributed to LERF releases. [C-25] Therefore, the NEI/EPRI characterization of Category 3b as a LERF contributor is considered extremely conservative for a Mark III.

The first risk measure change (change in LERF) is considered by Reg. Guide 1.174 as a "very small" impact on risk for the ILRT Interval Extension. The LERF change for the combined ILRT and DWBT interval is slightly over the boundary in Reg. Guide 1.174 between the regions very small risk (Region III) and small risk (Region II). The other two risk measure changes do not have criteria in Reg. Guide 1.174, but based on past ILRT interval extension requests these changes are also considered to represent "very small" impacts on risk.

logbook.xls.lnk

Table ES-1

QUANTITATIVE RESULTS AS A FUNCTION OF ILRT INTERVAL

EPRI Category	Category Description	Dose (Person-Rem Within 50 miles)(1)	Quantitative Results as a Function of ILRT Interval			
			Current (1-per-10 year ILRT)		Proposed (1-per-15 year ILRT)	
			Accident Frequency (per year)	Population Dose Rate (Person-Rem/Year Within 50 miles)	Accident Frequency (per year)	Population Dose Rate (Person-Rem/ Year Within 50 miles)
1	No Containment Failure(2)	2.4E+3	3.41E-6	8.18E-3	2.39E-6	5.74E-3
2	Containment Isolation System Failure	5.1E+5	1.13E-7	5.76E-2	1.13E-7	5.76E-2
3a	Small Pre-Existing Failures(2), (3)	2.4E+4	1.87E-6	4.49E-2	2.80E-6	6.72E-2
3b	Large Pre-Existing Failures(2), (3)	8.4E+4	1.87E-7	1.57E-2	2.80E-7	2.35E-2
4	Type B Failures (LLRT)	N/A	N/A	N/A	N/A	N/A
5	Type C Failures (LLRT)	N/A	N/A	N/A	N/A	N/A
6	Other Containment Isolation System Failure	N/A	N/A	N/A	N/A	N/A
7a	Containment Failure Due to Severe Accident (a)(4)	5.1E+5	2.63E-7	1.34E-1	2.63E-7	1.34E-1
7b	Containment Failure Due to Severe Accident (b)(4)	3.5E+5	4.70E-6	1.65	4.7E-6	1.65
7c	Containment Failure Due to Severe Accident (c)(4)	3.7E+5	1.71E-5	6.33	1.71E-5	6.33
7d	Containment Failure Due to Severe Accident (d)(4)	3.0E+5	9.20E-7	2.76E-1	9.2E-7	2.76E-1
8	Containment Bypass Accidents	5.1E+5	1.21E-7	6.17E-2	1.21E-7	6.17E-2
TOTALS:			2.87E-5	8.57	2.87E-5	8.60
Increase in Dose Rate						0.3%
Increase in LERF					9.30E-8	
Increase in CCFP (%)					0.3%	

Notes to Table ES-1:

(1) The population dose associated with the Technical Specification Leakage is based on scaling the population data, the power level, and allowable Technical Specification leakage compared to the NUREG/CR-4551 reference plant.

(2) Only EPRI categories 1, 3a, and 3b are affected by ILRT (Type A) interval changes.

(3) Dose estimates for #3a and #3b, per the NEI Interim Guidance, are calculated as 10xCategory 1 dose and 35xCategory 1 dose, respectively.

(4) EPRI Category #7, containment failure due to severe accident, was subdivided into four subgroups based on Clinton Level 2 release modes for dose allocation purposes. Note that this EPRI category is not affected by ILRT interval changes.

Section 1
INTRODUCTION

1.1 PURPOSE

The purpose of this analysis is to provide an assessment of the risk associated with implementing a one-time extension of the Clinton Power Station (CPS) containment Type A integrated leak rate test (ILRT) interval from ten years to fifteen years. The extension would allow for substantial cost savings as the ILRT could be deferred for additional scheduled refueling outages. The risk assessment follows the guidelines from NEI 94-01 [1], the methodology used in EPRI TR-104285 [2], the NEI Interim Guidance for Performing Risk Impact Assessments In Support of One-Time Extensions for Containment Integrated Leakage Rate Test Surveillance Intervals [3], NEI Additional Information for ILRT Extensions [21], and the NRC regulatory guidance on the use of Probabilistic Risk Assessment (PRA) findings and risk insights in support of a request for a change in a plant's licensing basis as outlined in Regulatory Guide 1.174 [4].

1.2 BACKGROUND

Revisions to 10CFR50, Appendix J (Option B) allow individual plants to extend the Integrated Leak Rate Test (ILRT) Type A surveillance testing requirements from three-in-ten years to at least once per ten years. The revised Type A frequency is based on an acceptable performance history defined as two consecutive periodic Type A tests at least 24 months apart in which the calculated performance leakage is less than normal containment leakage of 1.0 L_a (allowable leakage).

The basis for the current 10-year test interval is provided in Section 11.0 of NEI 94-01, Revision 0, and was established in 1995 during development of the performance-based Option B to Appendix J. Section 11.0 of NEI 94-01 states that NUREG-1493 [5], "Performance-Based Containment Leak Test Program," September 1995, provides the technical basis to support rulemaking to revise leakage rate testing requirements contained in Option B to Appendix J. The basis consisted of qualitative and quantitative assessments of the risk impact (in terms of increased public dose) associated with a range of extended leakage rate test intervals. To supplement the NRC's rulemaking basis, NEI undertook a similar study. The results of that study are documented in Electric Power Research Institute (EPRI) Research Project Report TR-104285.

The NRC report, Performance Based Leak Test Program, NUREG-1493 [5], analyzed the effects of containment leakage on the health and safety of the public and the benefits realized from the containment leak rate testing. In that analysis, it was determined for a comparable BWR plant that increasing the containment leak rate from the nominal 0.5 percent per day to 5 percent per day leads to a barely perceptible increase in total population exposure, and increasing the leak rate to 50 percent per day increases the total population exposure by less than 1 percent. Consequently, extending the ILRT interval should not lead to any substantial increase in risk. The current analysis is being performed to confirm these conclusions based on Clinton specific models and available data.

Earlier ILRT frequency extension submittals have used the EPRI TR-104285 methodology to perform the risk assessment. In November and December 2001, NEI issued enhanced guidance (hereafter referred to as the NEI Interim Guidance) that builds on the EPRI TR-104285 methodology and is intended to provide for more consistent submittals to the NRC. [3,21] The NEI Interim Guidance was developed for NEI by EPRI using personnel who also developed the EPRI TR-104285 methodology. This Clinton ILRT interval extension risk assessment employs the NEI Interim Guidance methodology.

It should be noted that, in addition to ILRT tests, containment leak-tight integrity is also verified through periodic in-service inspections conducted in accordance with the requirements of the American Society of Mechanical Engineers Boiler and Pressure Vessel Code (ASME Code), Section XI. More specifically, Subsection IWE provides the rules and requirements for in-service inspection of Class MC pressure-retaining components and their integral attachments, and of metallic shell and penetration liners of Class CC pressure-retaining components and their integral attachments in light-water cooled plants. Furthermore, NRC regulations 10 CFR 50.55a(b)(2)(ix)(E), require licensees to conduct visual inspections of the accessible areas of the interior of the containment 3 times every 10 years. These requirements will not be changed as a result of the extended ILRT interval. In addition, Appendix J, Type B local leak tests performed to verify the leak-tight integrity of containment penetration bellows, airlocks, seals, and gaskets are also not affected by the change to the Type A test frequency.

1.3 CRITERIA

Based on previously approved ILRT extension requests, Clinton uses the following risk metrics to characterize the change in risk associated with the one time ILRT extension:

- Change in Large Early Release Frequency (LERF)
- Change in conditional containment failure probability
- Change in population dose rate (person-rem/yr)

Consistent with the NEI Interim Guidance, the acceptance guidelines in Regulatory Guide 1.174 [4] are used to assess the acceptability of this one-time extension of the Type A test interval beyond that established during the Option B rulemaking of Appendix J.

RG 1.174 defines very small changes in the risk-acceptance guidelines as increases in core damage frequency (CDF) less than 10^{-6} per reactor year and increases in large early release frequency (LERF) less than 10^{-7} per reactor year. Since the Type A test does not impact CDF, the relevant criterion is the change in LERF. RG 1.174 also discusses defense-in-depth and encourages the use of risk analysis techniques to show that key principles, such as the defense-in-depth philosophy, are met. Therefore, the increase in the conditional containment failure probability, which helps to ensure that the defense-in-depth philosophy is maintained, will also be calculated.

In addition, based on the precedent of other ILRT extension requests [6, 20, 22], the total annual risk (person-rem/yr population dose rate) and the conditional containment failure probability are examined to demonstrate the relative change in risk. (No threshold has been established for these parameter changes.)

Section 2
METHODOLOGY

This section provides the following methodology related items:

- A brief summary of available resource documents to support the methodology
- The NEI Interim Guidance for the analysis approach to be used
- The assumptions used in the evaluation
- The inputs required
  - Generic ex-plant consequence
  - Plant specific inputs

The following subsections address these items.

2.1 General Resources Available

This section summarizes the general resources available as input. Various industry studies on containment leakage risk assessment are briefly summarized here:

1) NUREG/CR-3539 [10]
2) NUREG/CR-4220 [11]
3) NUREG-1273 [12]
4) NUREG/CR-4330 [13]
5) EPRI TR-105189 [8]
6) NUREG-1493 [5]
7) EPRI TR-104285 [2]
8) NEI Interim Guidance [3]
9) NUREG-1150 [14] and NUREG/CR-4551 [9]

The first study is applicable because it provides one basis for the threshold that could be used in the Level 2 PSA for the size of containment leakage that is considered significant and to be included in the model. The second study is applicable because it provides a basis of the probability for significant pre-existing containment leakage at the time of a core damage accident. The third study is applicable because it is a subsequent study to NUREG/CR-4220 that undertook a more extensive evaluation of the same database. The fourth study provides an assessment of the impact of different containment leakage rates on plant risk. The fifth study provides an assessment of the impact on shutdown risk from ILRT test interval extension. The sixth study is the NRC's cost-benefit analysis of various alternative approaches regarding extending the test intervals and increasing the allowable leakage rates for containment integrated and local leak rate tests. The seventh and eighth studies are EPRI studies of the impact of extending ILRT and LLRT test intervals on at-power public risk. The ninth study provides consequence evaluations that can be used as surrogate results when corrected for CPS specific characteristics.

<u>NUREG/CR-3539 [10]</u>

Oak Ridge National Laboratory (ORNL) documented a study of the impact of containment leak rates on public risk in NUREG/CR-3539. This study uses information from WASH-1400 [15] as the basis for its risk sensitivity calculations. ORNL concluded that the impact of leakage rates on LWR accident risks is relatively small.

<u>NUREG/CR-4220 [11]</u>

NUREG/CR-4220 is a study performed by Pacific Northwest Laboratories (PNL) for the NRC in 1985. The study reviewed over two thousand LERs, ILRT reports and other related records to calculate the unavailability of containment due to leakage. The study calculated unavailabilities for Technical Specification leakages and "large" leakages.

NUREG/CR-4220 assessed the "large" containment leak probability to be in the range of 1E-3 to 1E-2, with 5E-3 identified as the point estimate based on 4 events in 740 reactor years and conservatively assuming a one-year duration for each event.

NUREG-1273 [12]

A subsequent NRC study, NUREG-1273, performed a more extensive evaluation of the NUREG/CR-4220 database. This assessment noted that about one-third of the reported events were leakages that were immediately detected and corrected. In addition, this study noted that local leak rate tests can detect "essentially all potential degradations" of the containment isolation system.

NUREG/CR-4330 [13]

NUREG/CR-4330 is a study that examined the risk impacts associated with increasing the allowable containment leakage rates. The details of this report have no direct impact on the modeling approach of the ILRT test interval extension, as NUREG/CR-4330 focuses on leakage rate and the ILRT test interval extension study focuses on the frequency of testing intervals. However, the general conclusions of NUREG/CR-4330 are consistent with NUREG/CR-3539 and other similar containment leakage risk studies:

> "...the effect of containment leakage on overall accident risk is small since risk is dominated by accident sequences that result in failure or bypass of containment."

EPRI TR-105189 [8]

The EPRI study TR-105189 is useful to the ILRT test interval extension risk assessment because this EPRI study provides insight regarding the impact of containment testing on shutdown risk. This study performed a quantitative evaluation (using the EPRI ORAM software) for two reference plants (a BWR/4 and a PWR) of the impact of extending ILRT and LLRT test intervals on shutdown risk.

The result of the study concluded that a small but measurable safety benefit is realized from extending the test intervals. For the BWR, the benefit from reducing the ILRT frequency from 3 per 10 years to 1 per 10 years was calculated to be a reduction of approximately 1E-7/yr in the shutdown core damage frequency. This risk reduction is due to the following issues:

- Reduced opportunity for draindown events
- Reduced time spent in configurations with impaired mitigating systems

The study identified 7 shutdown incidents (out of 463 reviewed) that were caused by ILRT or LLRT activities. Two of the 7 incidents were RCS draindown events caused by ILRT/LLRT activities. The other 5 events involved loss of RHR and/or SDC due to ILRT/LLRT activities. This information was used in the EPRI study to estimate the safety benefit from reductions in testing frequencies. This represents a valuable insight into the improvement in the safety due to extending the ILRT test interval.

NUREG-1493 [5]

NUREG-1493 is the NRC's cost-benefit analysis for proposed alternatives to reduce containment leakage testing intervals and/or relax allowable leakage rates. The NRC conclusions are consistent with other similar containment leakage risk studies:

- Reduction in ILRT frequency from 3 per 10 years to 1 per 20 years results in an "imperceptible" increase in risk.
- Increasing containment leak rates several orders of magnitude over the design basis would minimally impact (0.2 – 1.0%) population risk.
- Given the insensitivity of risk to the containment leak rate and the small fraction of leak paths detected solely by Type A testing, increasing the interval between integrated leak rate tests is possible with minimal impact on public risk.

EPRI TR-104285 [2]

Extending the risk assessment impact beyond shutdown (the earlier EPRI TR-105189 study), the EPRI TR-104285 study is a quantitative evaluation of the impact of extending Integrated Leak Rate Test (ILRT) and (Local Leak Rate Test) LLRT test intervals on at-power public risk. This study combined IPE Level 2 models with NUREG-1150 Level 3 population dose models to perform the analysis. The study also used the approach of NUREG-1493 in calculating the increase in pre-existing leakage probability due to extending the ILRT and LLRT test intervals.

EPRI TR-104285 used a simplified Containment Event Tree to subdivide representative core damage sequences into eight (8) categories of containment response to a core damage accident:

1. Containment intact and isolated
2. Containment isolation failures due to support system or active failures
3. Type A (ILRT) related containment isolation failures
4. Type B (LLRT) related containment isolation failures
5. Type C (LLRT) related containment isolation failures
6. Other penetration related containment isolation failures
7. Containment failure due to core damage accident phenomena
8. Containment bypass

Consistent with the other containment leakage risk assessment studies, this study concluded:

> "*These study results show that the proposed CLRT* [containment leak rate tests] *frequency changes would have a minimal safety impact. The change in risk determined by the analyses is small in both absolute and relative terms. For example, for the PWR analyzed, the change is about 0.02 person-rem per year . . .*"

NEI Interim Guidance [3, 21]

NEI "Interim Guidance for Performing Risk Impact Assessments in Support of One-Time Extensions of Containment Integrated Leakage Rate Test Surveillance Intervals" [3] has been developed to provide utilities with revised guidance regarding licensing submittals.

A nine step process is defined which includes changes in the following areas of the previous EPRI guidance [2]:

- Impact of extending surveillance intervals on dose
- Method used to calculate the frequencies of leakages detectable only by ILRTs
- Provisions for using NUREG-1150 dose calculations to support the population dose determination.

This NEI Guidance is used in the Clinton ILRT analysis.

NUREG-1150 [14] and NUREG/CR 4551 [9]

NUREG-1150 and the technical basis, NUREG/CR-4551, provide an ex-plant consequence analysis for a spectrum of accidents including a severe accident with the containment remaining intact (i.e., Technical Specification leakage). The ex-plant consequences from NUREG-1150 used for the Clinton ILRT evaluation are taken from Grand Gulf (another Mark III plant).

2.2 NEI INTERIM GUIDANCE

The Clinton risk assessment analysis uses the approach outlined in the NEI Interim Guidance. [3,21] The nine steps of the methodology are:

1. Quantify the baseline (nominal three year ILRT interval) frequency per reactor year for the EPRI accident categories of interest. Note

that EPRI categories 4, 5, and 6 are not affected by changes in ILRT test frequency.

2. Determine the containment leakage rates for EPRI categories 1, 3a and 3b.

3. Develop the baseline population dose (person-rem) for the applicable EPRI categories.

4. Determine the population dose rate (person-rem/year) by multiplying the dose calculated in Step (3) by the associated frequency calculated in Step (1).

5. Determine the change in probability of leakage detectable only by ILRT, and associated frequency for the new surveillance intervals of interest. Note that with increases in the ILRT surveillance interval, the size of the postulated leak path and the associated leakage rate are assumed not to change, however the probability of leakage detectable only by ILRT does increase.

6. Determine the population dose rate for the new surveillance intervals of interest.

7. Evaluate the risk impact (in terms of population dose rate and percentile change in population dose rate) for the interval extension cases.

8. Evaluate the risk impact in terms of LERF.

9. Evaluate the change in conditional containment failure probability.

The first seven steps of the methodology calculate the change in dose. The change in dose is the principal basis upon which the Type A ILRT interval extension was previously granted and is a reasonable basis for evaluating additional extensions. The eighth step in the interim methodology calculates the change in LERF and compares it to the guidelines in Regulatory Guide 1.174. Because there is no change in CDF, the change in LERF forms the quantitative basis for a risk informed decision per current NRC practice, namely Regulatory Guide 1.174. The ninth and final step of the interim methodology calculates the change in containment failure probability. The NRC has previously accepted similar calculations (Ref. [7], referred to as conditional containment failure probability, CCFP) as the basis for showing that the proposed change is consistent with the defense in depth philosophy. As such this last step suffices as the remaining basis for a risk informed decision per Regulatory Guide 1.174.

2.3 GROUND RULES

The following ground rules are used in the analysis:

- The surveillance frequency for Type A testing in NEI 94-01 is at least once per ten years based on an acceptable performance history (i.e., two consecutive periodic Type A tests at least 24 months apart where the calculated performance leakage rate was less than 1.0 L_a) and consideration of the performance factors in NEI 94-01, Section 11.3. Based on the consecutive successful ILRTs performed in the early 1990's, the current ILRT interval for Clinton Power Station is once per ten years. [16]

- The Clinton Level 1 and Level 2 internal events PRA model provides representative results for the analysis.

- It is appropriate to use the Clinton internal events PRA model as a gauge to effectively describe the risk change attributable to the ILRT extension. It is reasonable to assume that the impact from the ILRT extension (with respect to percent increases in population dose rate) will not substantially differ if fire and seismic events were to be included in the calculations.

- An evaluation of the risk impact of the ILRT on shutdown risk is addressed using the generic results from EPRI TR-105189 [8] as augmented by NEI Interim Guidance. [3, 21]

- Radionuclide release categories are defined consistent with the EPRI TR-104285 methodology. [2]

- The ex-plant consequence in terms of population dose results for the containment failures modeled in the PSA can be characterized by information provided in NUREG/CR-4551 [9]. They are estimated by scaling the NUREG/CR-4551 population dose results by power level, population, and Tech Spec leak rate differences for Clinton compared to the NUREG/CR-4551 Mark III reference plant, Grand Gulf.

- Per the NEI Interim Guidance [3], the representative containment leakage for EPRI Category 1 sequences is 1 L_a (L_a is the Technical Specification maximum allowable containment leakage rate).

- Per the NEI Interim Guidance [3], the representative containment leakage for EPRI Category 3a sequences is 10 L_a.

- Per the NEI Interim Guidance [3], the representative containment leakage for EPRI Category 3b sequences is 35 L_a.

- EPRI Category 3b is conservatively categorized as LERF based on the previously approved methodology [3].

- The impact on population doses from Interfacing System LOCAs is not altered by the proposed ILRT extension, but is accounted for in the EPRI methodology as a separate entry for comparison purposes. Since the ISLOCA contribution to population dose is fixed, no changes on the conclusions regarding increases in population dose from this analysis will result from this assumption.

- The containment isolation valve test frequency is not altered. Therefore, the reduction in ILRT frequency does not impact the reliability of containment isolation valves to close in response to a containment isolation signal.

2.4 PLANT SPECIFIC INPUTS

The inputs to the risk assessment include the following:

- Past Clinton ILRT results to demonstrate the adequacy of the administrative and hardware issues.

- Ex-plant consequence evaluation from NUREG-1150 for a Mark III plant

- Clinton specific adjustments to ex-plant consequence evaluation from NUREG-1150 (NUREG/CR 4551 Vol. 6 for Grand Gulf)

- Clinton specific inputs (Level 1 & 2)

2.4.1 Ex-Plant Consequences

Consistent with the NEI Interim Guidance [3] and the supplemental information [21], ex-plant consequence evaluations from NUREG-1150 can be used in the ILRT evaluation to support the population dose estimate.

Figure 2-1 is a simplified flow chart that shows the process for determining the Clinton specific population dose (person-rem) for comparable radionuclide release categories starting with the NUREG-1150 Mark III (Grand Gulf) ex-plant consequence evaluation and correcting for key differences.

The surrogate plant consequence analysis for Grand Gulf is calculated for the 50-mile radial area surrounding Grand Gulf (A). The ex-plant calculation is delineated by total person-rem for each identified Accident Progression Bin (APB) from NUREG/CR-4551 (B). The Clinton Level 2 model end states are assigned to one of the NUREG/CR-4551 APBs (C, E).

In order to convert the Grand Gulf population dose estimates for use in the Clinton consequence evaluation, the adjustments to these ex-plant consequences that are judged important to account for include the following (D, F, G):

- Population differences
- Containment leakage rate
- Power level

Finally, the Clinton specific ex-plant consequences are calculated by APBs and subsequently converted to EPRI categories (H).

The parameters that were used in the Grand Gulf analysis from NUREG/CR-4551 and will be compared with Clinton are the following:

- Grand Gulf Population out to 50 miles = 3.6E+5 persons (See Appendix A for derivation)
- Grand Gulf Power level 3833 MWt
- Grand Gulf Containment leak rate = 0.5%/day[(1)]

(While meteorology could play a role in the early health effects calculations, the meteorology and site topography for Grand Gulf and Clinton are assumed to be sufficiently similar that these differences are assumed not to play a significant role in this evaluation of total population dose.)

2.4.2 Plant Specific Inputs

The Clinton specific information used to perform this ILRT interval extension risk assessment includes the following:

- Clinton Level 1 PSA
- Clinton Level 2 PSA
  - Population
  - Power Level
  - Containment Leak Rate

(1) While the Grand Gulf Technical Specification leakage is 0.35%/day, the analysis performed in NUREG/CR-4551 used a leakage of 0.5%/day.

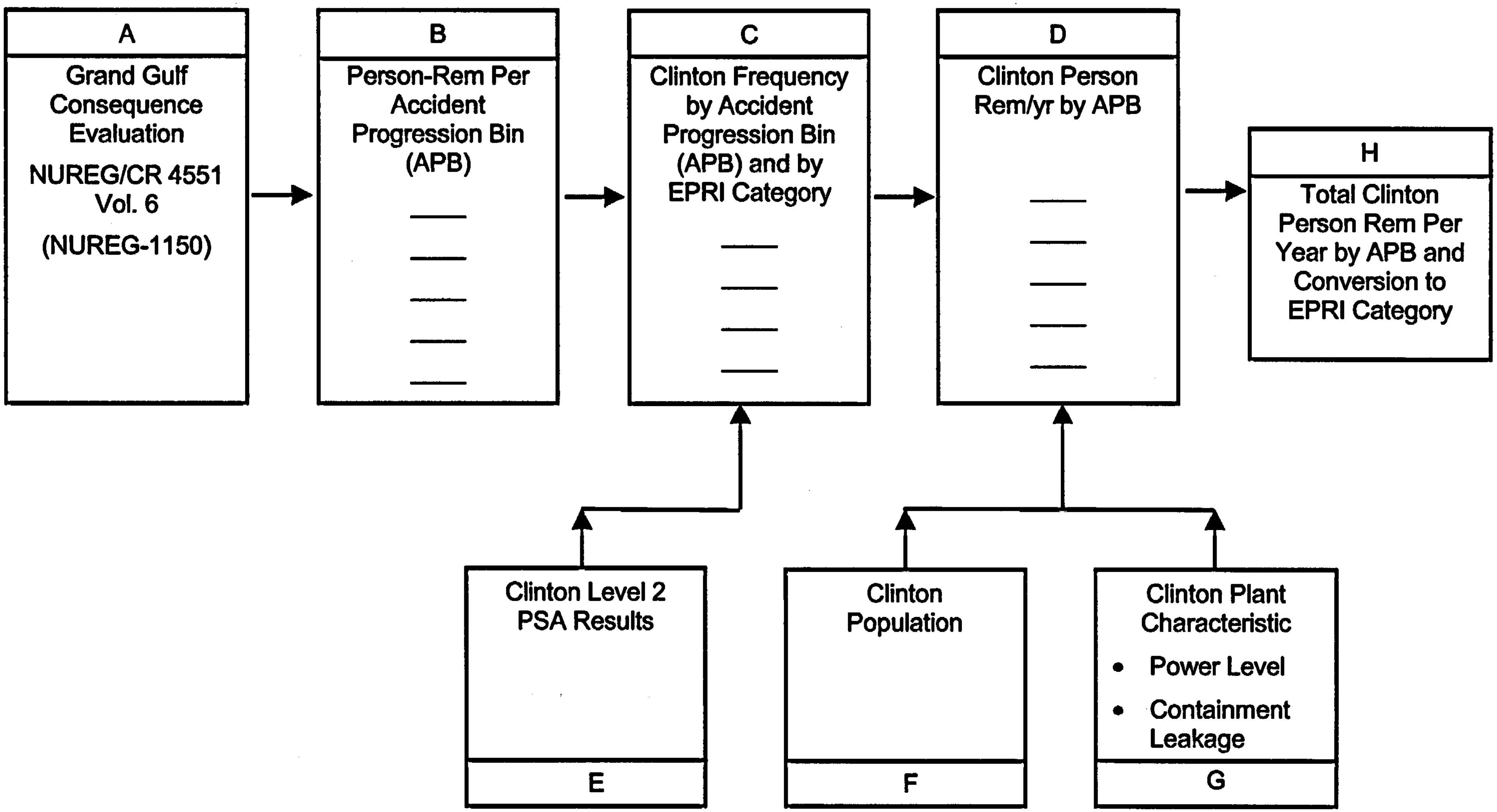

Figure 2-1 Process for Calculating Population Dose for Clinton Using the Surrogate Plant Results from NUREG-1150

2.4.2.1 Clinton Level 1 PSA

The Clinton Level 1 and 2 PSA (Rev. 3) used as input to this analysis is characteristic of the as-built, as-operated plant. The Rev. 3 PSA model has since been superseded by a Rev. 3a model that has a lower CDF and lower radionuclide release frequency. However, no detailed Level 2 is available for the Rev. 3a model. The use of the previous model, Rev. 3 model, with the higher CDF and release frequency will yield conservative estimates of the risk metrics investigated for the ILRT interval extension. Therefore, the Rev. 3 model is used in the analysis. The Rev. 3 and 3a models are developed in SETS. The total core damage frequency (CDF) as reported in the CPS Level 2 Results Report is 2.76E-5/yr. [18] Table 2-1a summarizes the Clinton Level 2 PSA results for containment failure. Table 2-1b summarizes the Clinton Level 1 PSA frequency results by core damage accident class.

2.4.2.2 Clinton Level 2 PRA

The Clinton Level 2 PRA is used to calculate the release frequencies for the accidents evaluated in this assessment. The Level 2 PRA is also developed in SETS. Table 2-2 summarizes the pertinent Clinton Level 2 PRA results in terms of release modes. [18]. The total release frequency is 2.25E-5/yr(1); with a total CDF of 2.76E-5/yr.

2.4.3. Adjustments to Ex-plant Consequence Calculations

This NUREG/CR-4551 ex-plant consequence analysis is calculated for the 50-mile radial area surrounding Grand Gulf, and is reported in total person-rem for discrete accident categories (termed Accident Progression Bins (APB) in NUREG/CR-4551). To use the NUREG/CR-4551 consequences in this ILRT risk assessment, the following steps should first be performed:

- Adjust the person-rem results to account for differences between the Grand Gulf analyses in NUREG/CR-4551 and the Clinton plant and its demographics:

(1) See Table 2.2. Note that the "No Release Mode" does not include the core damage sequences with a vented containment (i.e., CPS Release Mode A1).

  - Reactor Power Level
  - Technical Specification Allowed Containment Leakage Rate
  - Population

- Assign the adjusted NUREG/CR-4551 APB consequences to the EPRI categories used in this risk assessment

2.4.3.1 Surrounding Population

The 50-mile radius population used in the Grand Gulf NUREG/CR-4551 consequence calculations is 3.4E+5 persons (refer to Appendix A of this report).

For the Clinton population estimate, data is available for population by county from the US Census Bureau on the web site (http://quickfacts.census.gov/qfd/states/27000. html). This data is used to estimate the population within a 50-mile radius of the plant. If the entire county falls within the 50-mile radius based on a review of a map containing a mileage scale and county borders, then the entire population can be included in the population estimate. Otherwise, a fraction of the population is counted based on the percentage of the county within the 50-mile radius. The land area within the 50-mile radius is estimated based on visual inspection of the map and the population of that area is estimated assuming uniform distribution of the population within the county. The results of the population estimate for Clinton are presented in Table 2-3.

The year 2000 population within the 50-mile radius of Clinton is estimated in Appendix A of this report at 8.6E+5 persons.

The ratio of the population surrounding Clinton to that in the Grand Gulf analysis results in a factor increase of:

$$\frac{8.6E+5 \text{ persons}}{3.4E+5 \text{ persons}} \text{ or } 2.53$$

2.4.3.2 Reactor Power Level

The Grand Gulf reactor power level used in the NUREG/CR-4551 consequence calculations is 3833 MWt. By 2003, Clinton will have performed a power uprate of 20% over the originally licensed thermal power. The projected Clinton full power level is 3473 MWt.

The Clinton Power Level used in this ILRT evaluation is the extended power uprate power level of 3473 MWt. This represents a factor of .91 = (3473 MWt/3833 MWt) change in the population dose for each APB.

2.4.3.3 Technical Specification Containment Leakage

The Grand Gulf analysis in NUREG/CR-4551 deviates from the Grand Gulf plant in the following area:

- The Grand Gulf Tech Spec Leakage is 0.35% vol/day
- The NUREG/CR-4551 analysis (see Vol. 6, Rev. 1, Part 2, page B.2-9) used 0.5% vol/day

The Clinton Tech Spec leakage is 0.65% vol./ day. Because the leakage rates are in terms of the containment volume. These plant characteristics are also needed:

- Grand Gulf Containment Volume = 1.67E+6 ft^3
- Clinton Containment Volume = 1.8E+6 ft^3

Therefore, an adjustment in the ex-plant consequence calculation for the "INTACT" containment category is to be used between the analyzed surrogate "plant" (pseudo - Grand Gulf) and Clinton. The containment leakage rate used in the Grand Gulf NUREG/CR-4551 consequence calculations for core damage accidents with the containment intact is 0.5 vol. % of Grand Gulf over 24 hours. The Clinton maximum allowable containment leakage per Technical Specifications is 0.65 vol. % of Clinton per day (Clinton Technical Specifications).

For this comparison, the following factor can be developed to relate the leakage[1] impact between the two plants:

$$\frac{\text{Total Leakage Clinton}}{\text{Total Leakage Grand Gulf}} = \frac{0.65\ \text{Vol.}^{C}\ \%/\text{day}}{0.50\ \text{Vol.}^{GG}\ \%/\text{day}}$$

$$= \frac{0.65\%\ /\text{day}}{0.50\%\ /\text{day}} * \frac{1}{Z}$$

$$\frac{\text{Total Leakage for Clinton}}{\text{Total Leakage for Grand Gulf}} = 1.30 * 1.08 = 1.40$$

This represents a factor of 1.4 increase in the person rem consequence for the "intact" containment APB.

(1) Ratio of containment volumes is needed to relate the leakage rates:

$$\frac{\text{Vol}^{C}}{\text{Vol}^{GG}} = \frac{1}{Z}$$

Where

$\text{Vol}^{GG} = Z\ \text{Vol}^{C}$

$$Z = \frac{\text{Vol}^{GG}}{\text{Vol}^{C}} = \frac{1.67E{+}6ft^3}{1.8E{+}6ft^3} = .927,\ 1/Z = 1.08$$

Containment Vol of GG = $1.67E{+}6ft^3$

Containment Vol of Clinton = $1.8E{+}6ft^3$

2.4.4 Clinton ILRT Results

2.4.3.4 Summary

The factors that are calculated for use in adjusting the population dose (person-rem) of the surrogate plant (NUREG-1150 Grand Gulf) for the site and plant differences are as follows:

Consequence categories dependent on the "INTACT" Tech Spec Leakage

$F_{CAT\ 1,\ 3a,\ 3b} = F_{POWER} * F_{POPULATION} * F_{TS\ LEAK}$

$F_{CAT\ 1,\ 3a,\ 3b} = 0.91 * 2.53 * 1.4$

$F_{CAT\ 1,\ 3a,\ 3b} = 3.22$

Consequence categories not dependent on the Tech Spec Leakage:

$F_C = F_{POWER} * F_{POPULATION}$

$F_C = 0.91 * 2.53$

$F_C = 2.30$

Table 2-1a

SUMMARY OF CLINTON LEVEL 2 PSA RESULTS [18]

End State	Frequency (per year)	Percent
No Containment Failure (Release mode A0, No Release)	5.47E-6[(2)]	20
Containment Intact, Venting (Release mode A1)	7.81E-6[(2)]	27
Containment Failure (All other release modes)	1.47E-5	53
Total	2.76E-5[(1)]	100%

[(1)]Total CDF is based on CPS PSA Level 2 Results Report. [18] Sum of the release contributors is accurate to within a few percent of the quoted CDF. Differences are due to roundoff (See Note (2)).

[(2)]Sum of the A0 and A1 failure modes are taken from personal communication from A.J. Hable (CPS) to G.A. Teagarden (ERIN).

Table 2-1b

CORE DAMAGE FREQUENCY CONTRIBUTIONS BY ACCIDENT CLASS [19]

Contributing Accident Class		Core Damage Frequency[1]
<u>Transients</u>		
Class IA	Transients – Core Melt with Vessel at High Pressure	5.59E-6
Class IC	ATWS with Loss of Injection	1.55E-8
Class ID	Transients – Core Melt with Vessel at Low Pressure	9.97E-6
Class II	Core Melts After Containment Failure Because of Loss of DHR Capability	3.79E-6
<u>SBO</u>		
Class IB	Station Black Out	6.02E-6
<u>LOCAs</u>		
Class 3B	LOCA – Core Melt with Vessel at High Pressure	5.89E-10
Class 3C	Large, Medium, or Small LOCA – Core Melt with Vessel at Low Pressure	1.40E-7
Class V	Interfacing System LOCA	1.21E-7
<u>ATWS</u>		
Class IV	ATWS – Containment Fails Before Core Damage	9.90E-7
<u>Flooding</u>		
IFLD	Internal Flooding	7.44E-7

[1] All frequencies in events per year.

Table 2-2

SUMMARY OF CLINTON PSA LEVEL 2 RESULTS [18]

Release Mode	Description	Frequency (per year)(1)
A0	Containment Intact (no release)	5.47E-6(3)
A1	Containment intact, vented, no pool bypass	7.81E-6(3)
B1	Small containment failure, no pool bypass	1.13E-6
B2	Large containment failure, no pool bypass	8.16E-6
C1	Small containment failure before RPV failure, pool bypassed with containment spray	7.78E-8
C2	Large containment failure before RPV failure, pool bypassed with containment spray	1.80E-7
C6	Large containment failure before RPV failure, pool bypassed with injection to debris, Release prior to RPV breach is wetwell	3.51E-6
C8	Large containment failure before RPV failure, pool bypassed with injection to debris, Release location prior to RPV breach is drywell	9.41E-7
C10	Large containment failure before RPV failure, pool bypassed, no injection, wetwell release	2.62E-7
C12	Large containment failure before RPV failure, pool bypassed, no injection, DW release	1.82E-9
D5	Small containment failure after RPV failure, pool bypassed, no injection	9.91E-9
D6	Large containment failure after RPV failure, pool bypassed, no injection	9.07E-7
Total Release Frequency (A0 not included)		2.25E-5(4)
Total Frequency		2.76E-5(2)

(1) Based on Clinton PSA Rev. 3 model.

(2) Total CDF is based on CPS PSA Level 2 Results Report [18]

(3) Sum of the A0 and A1 failure modes are taken from personal communication from A.J. Hable (CPS) to G.A. Teagarden (ERIN).

(4) Release frequency is approximately 2.25E-5/yr as reported in the CPS Level 2 PSA Results Report. [18] Small differences in the summation are due to roundoff and varying truncation levels.

Table 2-3

POPULATION WITHIN 50 MILES OF CLINTON (200 US CENSUS)

(Source: http://quickfacts.census.gov/qfd/states/17000.html)

County	2000 Census Total Population of all Counties	Percent Area of County in 50 Mile Radius[1]	Population within 50 Mile Radius[2]
Champaign	179,669	80	156,000[3]
Christian	35,372	75	30,000[3]
Coles	53,196	10	1,200[3]
DeWitt	16,798	100	16,798
Douglas	19,922	50	9,961
Ford	14,241	40	5,696
Livingston	39,678	15	5,952
Logan	31,183	100	31,183
McLean	150,433	100	150,433
Macon	114,706	100	114,706
Mason	16,038	50	8,019
Menard	12,486	85	10,613
Moultrie	14,287	90	12,858
Piatt	16,365	100	16,365
Sangamon	188,951	60	160,000[3]
Shelby	22,893	40	9,157
Tazewell	128,485	95	95,000[3]
Woodford	35,469	65	23,055
TOTALS:	1,273,605		856,996

(1) Based on visual inspection of Illinois state maps.

(2) County Population multiplied by percentage within 50 mile zone, except when noted.

(3) Population density varied greatly in this region, an exception was made.

Section 3

ANALYSIS

This section provides a step-by-step summary of the NEI guidance as applied to the CPS ILRT interval extension risk assessment. Each subsection addresses a step or group of steps in the NEI guideline.

3.1 BASELINE ACCIDENT CATEGORY FREQUENCIES (STEP 1)

The first step of the NEI Interim Guidance is to quantify the baseline frequencies for each of the EPRI TR-104285 accident categories. This portion of the analysis is performed using the Clinton Level 1 and Level 2 PSA results. The results for each EPRI category are described below.

Tables 2-1a, 2-1b and 2-2 from the CPS Rev. 3 PSA are used for the inputs to the accident frequency assessment.

<u>Frequency of EPRI Category 1</u>

This group consists of all core damage accident sequences in which the containment is initially isolated and remains intact throughout the accident (i.e., containment leakage at or below maximum allowable Technical Specification leakage). The ILRT methodology artificially divides this category among the Tech Spec leakage case (Category 1) and two other categories that are used to simulate possible changes due to reduced ILRT frequencies (i.e., Categories 3a and 3b; see below for their definition). Per NEI Interim Guidance, the frequency per year for this category is calculated by subtracting the frequencies of EPRI Categories 3a and 3b (see below) from the sum of all severe accident sequence frequencies in which the containment is initially isolated and remains intact (i.e., accidents classified as "OK" in the Clinton Level 2 PSA).

As discussed previously in Section 2.4.2, the frequency of the Clinton Level 2 PSA "OK" or "No Release" accident bin is 5.47E-6/yr. As described below, the frequencies of EPRI Categories 3a and 3b are 5.6E-7/yr and 5.6E-8/yr, respectively. Therefore, the frequency of EPRI Category 1 is calculated as (5.47E-6/yr) – (5.6E-7/yr + 5.6E-8/yr) = 4.85E-6/yr.

Frequency of EPRI Category 2

This group consists of all core damage accident sequences in which the containment isolation system function fails during the accident progression (e.g., due to failures-to-close of large containment isolation valves initiated by support system failures, or random or common cause valve failures).

The frequency of this EPRI category is estimated by multiplying the conditional probability of containment isolation failure from the Clinton Level 2 PSA by the portion of the severe accident sequences (CDF) that would be challenged. The sequences that have containment isolation already failed are Class II, Class IV, and Class V. Therefore, the EPRI Category 2 CDF does not include CPS Level 1 Class II, Class IV, or Class V accident sequences. The following values are used for this calculation:

- Containment Isolation System failure probability = 4.99E-3
- Total CDF = 2.76E-5/yr [19]
- Class II sequences = 3.79E-6/yr [19]
- Class IV sequences = 9.9E-7/yr [19]
- Class V sequences = 1.21E-7/yr [19]

The frequency per year for this category is calculated as follows:

Frequency 2 = [containment isolation failure probability]
x [CDF – (CDF of Class II + CDF of Class IV + CDF of Class V)]

Frequency 2 = [4.99E-3] X [(2.76E-5/yr) – (3.79E-6/yr + 9.9E-7/yr +1.21E-7/yr)]

Frequency 2 = 1.13E-7/yr

Note that pre-existing isolation failures are included in Category 6.

The frequency of EPRI Category 2 is 1.13E-7/yr.

Frequency of EPRI Category 3a

This group consists of all core damage accident sequences in which the containment is failed due to a pre-existing "small" leak in the containment structure or liner that would be identifiable only from an ILRT (and thus affected by ILRT testing frequency). Consistent with NEI Interim Guidance [21], the frequency per year for this category is calculated as:

Frequency 3a = [3a conditional failure probability] x [CDF - (CDF with independent LERF + CDF that cannot cause LERF)]

The 3a conditional failure probability (2.7E-2) value is the conditional probability of having a pre-existing "small" containment leak that is detectable only by ILRTs. This value is derived in Reference [3] and is based on data collected by NEI from 91 plants. This value is also assumed reflective of ILRT testing frequencies of 3 tests in 10 years.

The pre-existing leakage probability is multiplied by the residual core damage frequency (CDF) determined as the total CDF minus the CDF for those individual sequences that either may already (independently) cause a LERF or could never cause a LERF. As discussed previously in Section 2.4.2, the Clinton total core damage frequency is 2.76E-5/yr. Of this total CDF, the following core damage accidents involve either LERF directly (containment bypass) or will never result in LERF:

- Long Term Station Blackout (SBO) scenarios = 3.0E-6/yr(1)
- Loss of Containment Heat Removal accidents (Clinton PRA Class II): 3.79E-6/yr [19]

(1) Table VI-1 [19]

- Containment Bypass accidents (Clinton PRA Class V): 1.21E-7/yr [19]

Therefore, the frequency of EPRI Category 3a is calculated as (2.70E-02) x [(2.76E-5/yr) – (3.0E-6/yr + 3.79E-6/yr + 1.21E-7/yr)] = 5.6E-7/yr.

Frequency of EPRI Category 3b

This group consists of all core damage accident sequences in which the containment is failed due to a pre-existing "large" leak in the containment structure or liner that would be identifiable only from an ILRT (and thus affected by ILRT testing frequency). Similar to Category 3a, the frequency per year for this category is calculated as:

Frequency 3b = [3b conditional failure probability] x [CDF - (CDF with independent LERF + CDF that cannot cause LERF)]

The 3b failure probability (2.7E-3) value is the conditional probability of having a pre-existing "large" containment leak that is detectable only by ILRTs. This value is derived in Reference [3] and is based on data collected by NEI from 91 plants. This value is also assumed reflective of ILRT testing frequencies of 3 tests in 10 years.

Therefore, similar to EPRI Category 3a, the frequency of Category 3b is calculated as (2.70E-03) x [(2.76E-5/yr) – (3.0E-6/yr + 3.79E-6/yr + 1.21E-7/yr)] = 5.6E-8/yr.

Frequency of EPRI Category 4

This group consists of all core damage accident sequences in which the containment isolation function is failed due to a pre-existing failure-to-seal of Type B component(s) that would not be identifiable by an ILRT. Per NEI Interim Guidance, because this category of failures is only detected by Type B tests and not by the Type A ILRT, this group is not evaluated further in this analysis.

Frequency of EPRI Category 5

This group consists of all core damage accident sequences in which the containment isolation function is failed due to a pre-existing failure-to-seal of Type C component(s) that would not be identifiable by an ILRT. Per NEI Interim Guidance, because this category of failures is only detected by Type C tests and not by the Type A ILRT, this group is not evaluated further in this analysis.

Frequency of EPRI Category 6

This group consists of all core damage accident sequences in which the containment isolation function is failed due to "other" pre-existing failure modes (e.g., pathways left open or valves that did not properly seal following test or maintenance activities) that would not be identifiable by containment leak rate tests. Per NEI Interim Guidance, because this category of failures is not impacted by leak rate tests, this group is not evaluated further in this analysis.

Frequency of EPRI Category 7

This group consists of all core damage accident progression bins in which containment failure is induced by severe accident phenomena (e.g., overpressure). Per NEI Interim Guidance, the frequency per year for this category is based on the plant Level 2 PSA results.

As the Clinton Level 2 PSA appropriately categorizes containment failure accident sequences into different release bins, EPRI Category 7 is sub-divided in this analysis to reflect the spectrum of the Clinton Level 2 PSA results. The subdivision represents the Release Modes (Table 2-2) determined in the CPS Level 2 [18] as they correlate with the consequence categories (APB) from the surrogate Mark III plant. Table 3-1 summarizes the severe accident release modes from the CPS PSA that contribute to Category 7 of the EPRI classification scheme. Other severe accidents such as intact containment leakage and containment bypass are accounted for in other EPRI categories.

The CPS Level 2 PRA release modes can be correlated or binned into similar groups that will then be characterized in terms of release magnitude and ex-plant consequence as categorized in NUREG/CR-4551 for the surrogate Mark III plant. This binning matches the similarity in release path and scenario definition between the CPS Level 2 PRA and NUREG/CR-4551. Tables 3-3 and 3-4 provide the release category definition from these two analyses, and Table 3-7 provides the correlation between the two.

The frequency of Category 7a is the total frequency of the Clinton Level 2 PRA release modes C9, C10, C11, C12, E1, and E2. Based on the Clinton Level 2 PRA results summarized earlier in Table 2-2, the frequency of Category 7a is 2.63E-7/yr.

The frequency of Category 7b is the total frequency of the Clinton Level 2 PRA release modes C1, C2, C6, and C8. Based on the Clinton Level 2 PRA results summarized earlier in Table 2-2, the frequency of Category 7b is 4.71E-6/yr.

The frequency of Category 7c is the total frequency of the Clinton Level 2 PRA release modes A1, B1 and B2. Based on the Clinton Level 2 PRA results summarized earlier in Table 2-2, the frequency of Category 7c is 1.71E-5/yr.

Based on the Clinton Level 2 results summarized earlier in Table 2-2, the frequency of Category 7d is the total frequency of the Clinton Level 2 PRA release modes D5 and D6, which is 9.2E-7/yr.

Frequency of EPRI Category 8

This group consists of all core damage accident progression bins in which the accident is initiated by a containment bypass scenario (i.e., Break Outside Containment LOCA or Interfacing Systems LOCA, ISLOCA). The frequency of Category 8 is the total frequency of the Clinton Level 1 PSA containment bypass scenarios (Class V). Based on the Clinton Level 1 PSA results summarized earlier in Table 2-1, the frequency of Category 8 is 1.21E-7/yr.

Summary of Frequencies of EPRI Categories

In summary, per the NEI Interim Guidance, the accident sequence frequencies that can lead to radionuclide releases to the public have been derived for accident categories defined in EPRI TR-104285. The accident sequence frequency results by EPRI category are summarized in Table 3-2.

Table 3-1

SUMMARY OF SEVERE ACCIDENT TYPES ALLOCATED TO CLASS 7 OF THE EPRI CLASSIFICATION SCHEME(1)

EPRI Severe Accident Type	APB	Definition	Frequency (per year)
7a	1	CD, vessel breach, Early CF, Early SP Bypass, CS Not Available Vessel breach occurs and both the containment and the drywell have failed either before or at the time of vessel breach. The containment sprays do not operate before or at the time of vessel breach.	2.6E-7
7b	2	CD, vessel breach, Early CF, Early SP Bypass, CS Available Vessel breach occurs and both the containment and the drywell fail either before or at the time of vessel breach. In this bin, however, the containment sprays operate before or at the time of vessel breach.	4.7E-6
7c	4	CD, vessel breach, Early CF, No SP Bypass Vessel breach occurs and the containment fails either before or at the time of vessel breach. The drywell does not fail and, therefore, all of the radionuclide releases pass through the suppression pool. Because the pool has not been bypassed, the availability of the sprays is not very important and, thus, the CS characteristic has been dropped.	1.7E-5
7d	5	CD, vessel breach, Late CF Vessel breach occurs, however, the containment does not fail until the late time period. If the containment did not fail early, it is unlikely that the drywell will fail early. Thus, the suppression pool bypass characteristic and the containment spray characteristic have been dropped.	9.2E-7
			2.30E-5

(1) Note the CPS PRA also calculates the following additional release conditions.

- Intact and Leakage State = 5.47E-6/yr
- Containment Bypass = 1.21E-7/yr

Other Release States are assessed in the CPS PSA to have a negligible contribution to release.

Table 3-2

SUMMARY OF CLINTON BASELINE RELEASE FREQUENCIES AS A FUNCTION OF EPRI CATEGORY

EPRI Category	Category Description	Frequency Estimation Methodology	Frequency (1/yr)
1	No Containment Failure: Accident sequences in which the containment remains intact and is initially isolated. Only affected by ILRT leak testing frequency due to the incorporation of categories 3a and 3b.	Per NEI Interim Guidance: *[Total Clinton "OK" release category frequency] – [Frequency EPRI Categories 3a and 3b]* *[5.47E-6/yr] – [5.6E-7/yr + 5.6E-8/yr] = 4.8E-6/yr*	4.8E-6
2	Containment Isolation System Failure: Accident sequences in which the containment isolation system function fails during the accident progression (e.g., due to failures-to-close of large containment isolation valves initiated by support system failures, or random or common cause failures). Not affected by ILRT leak testing frequency.	[Clinton containment isolation failure probability] X [(Total CDF) – (CDF of Class II + CDF of Class IV +CDF of Class V)] [4.99E-3] X [(2.76E-5/yr) – (3.79E-6/yr +9.9E-7/yr + 1.21E-7/yr)] = 1.13E-7/yr	1.1E-7
3a	Small Pre-Existing Failures: Accident sequences in which the containment is failed due to a pre-existing small leak in the containment structure or liner that would be identifiable only from an ILRT (and thus affected by ILRT testing frequency).	Per NEI Interim Guidance: *[Clinton CDF for accidents not involving containment failure/bypass] x [2.7E-2]* *[(2.76E-5/yr) – (3.0E-6/yr + 3.79E-6 /yr + 1.21E-7/yr) x [2.70E-02] = 5.60E-7/yr*	5.6E-7
3b	Large Pre-Existing Failures: Accident sequences in which the containment is failed due to a pre-existing large leak in the containment structure or liner that would be identifiable only from an ILRT (and thus affected by ILRT testing frequency).	Per NEI Interim Guidance: *[Clinton CDF for accidents not involving containment failure/bypass] x [2.7E-3]* *[(2.76E-5/yr) – (3.0E-6/yr + 3.79E-6/yr + 1.21E-7/yr) x [2.70E-03] = 5.60E-8/yr*	5.6E-8
4	Type B Failures: Accident sequences in which the containment is failed due to a pre-existing failure-to-seal of Type B components that would not be identifiable from a ILRT (and thus not affected by ILRT testing frequency).	Per NEI Interim Guidance: N/A (not affected by ILRT frequency)	N/A

Table 3-2

SUMMARY OF CLINTON BASELINE RELEASE FREQUENCIES AS A FUNCTION OF EPRI CATEGORY

EPRI Category	Category Description	Frequency Estimation Methodology	Frequency (1/yr)
5	Type C Failures: Accident sequences in which the containment is failed due to a pre-existing failure-to-seal of Type C components that would not be identifiable from a ILRT (and thus not affected by ILRT testing frequency).	Per NEI Interim Guidance: N/A (not affected by ILRT frequency)	N/A
6	Other Containment Isolation System Failure: Accident sequences in which the containment isolation system function fails due to "other" pre-existing failure modes not identifiable by leak rate tests (e.g., pathways left open or valves that did not properly seal following test or maintenance activities). Not affected by ILRT leak testing frequency.	Per NEI Interim Guidance: N/A (not affected by ILRT frequency)	N/A
7a	Containment Failure Due to Accident (a): CD, vessel breach, Early CF, Early SP Bypass, CS Not Available Vessel breach occurs and both the containment and the drywell have failed either before or at the time of vessel breach. The containment sprays do not operate before or at the time of vessel breach.	Total Clinton release mode frequency for: C9 ε C10 2.62E-7 C11 ε C12 1.82E-9 E1 ε E2 ε	2.6E-7
7b	Containment Failure Due to Accident (b): CD, vessel breach, Early CF, Early SP Bypass, CS Available Vessel breach occurs and both the containment and the drywell fail either before or at the time of vessel breach. In this bin, however, the containment sprays operate before or at the time of vessel breach.	*Total Clinton release mode frequency for:* C1 7.78E-8 C2 1.80E-7 C6 3.51E-6 C8 9.41E-7	4.7E-6

7c	Containment Failure Due to Accident (c): CD, vessel breach, Early CF, No SP Bypass Vessel breach occurs and the containment fails either before or at the time of vessel breach. The drywell does not fail and, therefore, all of the radionuclide releases pass through the suppression pool. Because the pool has not been bypassed, the availability of the sprays is not very important and, thus, the CS characteristic has been dropped.	*Total Clinton release mode frequency for:* B1 1.13E-6 B2 8.16E-6	1.7E-5
7d	Containment Failure Due to Accident (d): CD, vessel breach, Late CF Vessel breach occurs, however, the containment does not fail until the late time period. If the containment did not fail early, it is unlikely that the drywell will fail early. Thus, the suppression pool bypass characteristic and the containment spray characteristic have been dropped.	*Total Clinton release mode frequency for:* D5 9.91E-9 D6 9.07E-7	9.2E-7
8	Containment Bypass Accidents: Accident sequences in which the containment is bypassed. Such accidents are initiated by LOCAs outside containment (i.e., Break Outside Containment LOCA, or Interfacing Systems LOCA). Not affected by ILRT leak testing frequency.	*[Total Clinton Containment Bypass release frequency]*	1.2E-7
		TOTAL:	2.8E-5[(1)]

(1) Accurate to within a few percent of the total CDF (2.76E-5/yr). [18] Differences due to roundoff and EPRI calculational approach.

3.2 CONTAINMENT LEAKAGE RATES (STEP 2)

The second step of the NEI Interim Guidance is to define the containment leakage rates for EPRI Categories 3a and 3b. As discussed earlier, EPRI Categories 3a and 3b are accidents with pre-existing containment leakage pathways ("small" and "large", respectively) that would only be identifiable from an ILRT.

The NEI Interim Guidance recommends containment leakage rates of $10L_a$ and $35L_a$ for Categories 3a and 3b, respectively. The NEI Interim Guidance describes these two recommended containment leakage rates as "conservative". These values are consistent with previous ILRT frequency extension submittal applications. L_a is the plant Technical Specification maximum allowable containment leak rate; for Clinton L_a is 0.65% of containment air weight per day (per Clinton Technical Specifications).

The NEI recommended values of $10L_a$ and $35L_a$ are used as is in this analysis to characterize the containment leakage rates for Categories 3a and 3b.

By definition, the containment leakage rate for Category 1 (i.e., accidents with containment leakage at or below maximum allowable Technical Specification leakage) is $1.0L_a$.

3.3 BASELINE POPULATION DOSE RATE ESTIMATES (STEPS 3-4)

The third and fourth steps of the NEI Interim Guidance are to estimate the baseline population dose (person-rem) for each EPRI category and to calculate the dose rate (person-rem/year) by multiplying the category frequencies by the estimated dose.

3.3.1 Population Dose Estimates (Step 3)

The NEI Interim Guidance recommends two options for calculating population dose for the EPRI categories:

- Use of NUREG-1150 dose calculations
- Use of plant-specific dose calculations

The NUREG-1150 [14] dose calculations were used in the EPRI TR-104285 study, as discussed previously in Section 2.1. The use of generic dose information for NUREG-1150 is recommended by NEI to make the ILRT risk assessment methodology more readily usable for plants that do not have a Level 3 PRA. As Clinton does not have a Level 3 PRA or associated plant-specific dose calculations, this ILRT risk assessment employs NUREG-1150 dose results calculated using the MACCS2 (MELCOR Accident Consequence Code System) consequence code; specifically, the doses for the Grand Gulf NUREG-1150 study (as documented in supporting report NUREG/CR-4551) are used. The following discussion summarizes the population dose calculation and results.

Grand Gulf NUREG-1150 Study Population Dose

The population dose is calculated by using data provided in NUREG/CR-4551 for Grand Gulf and adjusting the results for Clinton. Each accident sequence was associated with an applicable collapsed Accident Progression Bin (APB) from NUREG/CR-4551. The collapsed APBs are characterized by four attributes related to the accident progression. In the summary binning scheme, there are essentially four characteristics: vessel breach, containment failure, suppression pool bypass, and containment spray operation. Each of these characteristics and their associated attributes are defined in Table 3-4.

Information from the Clinton PRA Containment Event Trees (CETs) was used to classify each of the Level 2 sequences using these attributes. The definitions of the 8 collapsed APBs are provided in NUREG/CR-4551, Vol. 6 and are reproduced in Table 3-4 for reference purposes. Table 3-5 summarizes the calculated population dose associated with each APB from NUREG/CR-4551, Vol. 6, for Grand Gulf including the fraction of the population dose within 50 miles contributed by each APB and the frequency of release at the surrogate plant, Grand Gulf.

Adjustment of NUREG-4551 Doses to Clinton

As discussed in Section 2.4.3, the Grand Gulf NUREG/CR-4551 ex-plant consequence results are used as input to determine the population dose estimates of this risk assessment. The NUREG/CR-4551 50-mile radius ex-plant consequence results are summarized in Table 3-5 as a function of accident progression bins.

The NUREG/CR-4551 consequences summarized in Table 3-5 should be adjusted for use in this analysis to account for differences in the following parameters between NUREG-1150 analysis and the Clinton plant to obtain realistic estimates for Clinton:

- Population
- Reactor Power Level
- Technical Specification Allowed Containment Leakage Rate

Population Adjustment

As discussed in Section 2.4.3, the 50-mile radius Grand Gulf population used in the NUREG/CR-4551 consequence calculations is estimated at 3.4E+5 persons, whereas the year 2000 population within the 50-mile radius of Clinton is estimated at 8.6E+5 persons. This difference in population results in the adjustment factor to be applied to the NUREG/CR-4551 APB doses of 2.53.

Reactor Power Level Adjustment

As discussed in Section 2.4.3, the reactor power level used in the NUREG/CR-4551 Grand Gulf consequence calculations is 3833 MWth, whereas the Clinton Extended Power Uprate full power level is 3473 MWth. This difference in reactor power level results in the following adjustment factor to be applied to the NUREG/CR-4551 APB doses: 0.91.

Containment Leakage Rate Adjustment

As discussed in Section 2.4.3, the containment leakage rate used in the NUREG/CR-4551 consequence calculations for core damage accidents with the containment intact is 0.5 Vol^{GG} % over 24 hours[1], whereas the Clinton maximum allowable containment leakage per Technical Specifications is 0.65 Vol^{C} % per day. While use of a leakage rate below the maximum allowable may be reasonable, this analysis assumes that containment leakage is at the maximum allowable Technical Specification value. Additionally, a correction is required to account for differences in containment volumes. The containment volume of Grand Gulf is 1.67E+6 ft^3 while that of Clinton is slightly larger, 1.8E+6 ft^3. These differences result in the following adjustment factor to be applied to the NUREG/CR-4551 APB doses: 1.4.

Grand Gulf NUREG/CR-4551 Adjusted Doses

Table 3-6 summarizes the Grand Gulf NUREG/CR-4551 doses after adjustment for changes in population, reactor power level, and containment leakage rate for application to Clinton.

The factors that are calculated for use in adjusting the population dose (person-rem) of the surrogate plant (NUREG-4551 Grand Gulf) for the site and plant differences are as follows:

Consequence categories dependent on the "INTACT" Technical Specification Leakage

[1] Note that while the Grand Gulf Tech Spec leakage rate is 0.35% /day, the NRC contractor used a higher containment leakage rate of 0.5% /day for its analysis.

$$F_{CAT\ 1,\ 3a,\ 3b} = F_{POWER} * F_{POPULATION} * F_{TS\ LEAK}$$
$$F_{CAT\ 1,\ 3a,\ 3b} = 0.91 * 2.53 * 1.4$$
$$F_{CAT\ 1,\ 3a,\ 3b} = 3.22$$

Consequence categories not dependent on the Tech Spec Leakage:

$$F_C = F_{POWER} * F_{POPULATION}$$
$$F_C = 0.91 * 2.53$$
$$F_C = 2.30$$

Population Dose By APB for Clinton

Table 3-6 provides the translation of the surrogate analysis (Grand Gulf from NUREG-4551) to the Clinton plant and site based on APBs. This translation uses the adjustments to power, population, and containment leak rate to the NUREG/CR-4551 population dose results at 50 miles to obtain the adjusted population dose at 50 miles for Clinton for each APB.

Application of Clinton PRA Model Results to NUREG/CR-4551 Level 3 Output

A major factor related to the use of NUREG/CR-4551 in this evaluation is that the results of the Clinton PRA Level 2 model are not defined in the same terms as reported in NUREG/CR-4551. In order to use the Level 3 model presented in NUREG/CR-4551, it is necessary to convert the Clinton PRA Level 2 model results into a format which allows for the scaling of the Level 3 results based on current Level 2. As mentioned above, the Level 3 results are modified to reflect the difference in the plants and the site demographics that exist between the two sites. This subsection provides a description of the process used to convert the Clinton PRA Level 2 model results into a form that can be used to generate Level 3 results using the NUREG/CR-4551 documentation.

The basic process that was pursued to obtain Level 3 results based on the Clinton PRA Level 2 model and NUREG/CR-4551 was to define a useful relationship between the Level 2 and Level 3 results. The Clinton PRA provides a grouping of containment failure modes resulting from severe accident challenges. The grouping is referred to as the release modes. Table 3-3 provides this breakdown for the Clinton PRA. In addition, Clinton release modes of the PRA Level 2 were reviewed and assigned into one of the collapsed Accident Progression Bins (APBs) from NUREG/CR-4551.

The Clinton Level 2 model contains a significantly larger amount of information about the accident sequences than what is used in the collapsed APBs for the surrogate plant in NUREG/CR-4551 and this assignment process required simplification of accident progression information and assumptions related to categorizations of certain items. Note that each Level 2 sequence is characterized by a combination of three Plant Damage State categories. The first characteristic is based on the RPV pressure; the second characteristic is based on the Containment Failure Mode; and the third characteristic is based on the Timing of Release. The Clinton Release Modes can be correlated with the Grand Gulf Accident Progression Bins (APB) which are described in Table 3-4 and have the ex-plant population dose calculated in Table 3-5.

Table 3-7 summarizes the correlation between the Clinton Level 2 Release Modes and the Source Term/Ex-plant Consequences from the surrogate Mark III Plant in NUREG/CR-4551. This table provides a summary of the CPS Level 2 PRA results (Rev 3). The frequencies in the last two columns are those from the CPS Level 2 results. These frequencies are given by CPS "release mode" and are summed as appropriate, consistent with the Accident Progression Bin (APB) definitions from the surrogate Mark III analysis performed to support NUREG 1150 and reported in NUREG/CR-4551.

Table 3-3

SUMMARY OF CLINTON PSA RELEASE MODES

Release Mode	Description	Frequency (per year)(1)
A1	Containment intact, vented, no pool bypass	7.81E-6
B1	Small containment failure(2), no pool bypass	1.13E-6
B2	Large containment failure, no pool bypass	8.16E-6
C1	Small containment failure before RPV failure, pool bypassed with containment spray	7.78E-8
C2	Large containment failure before RPV failure, pool bypassed with containment spray	1.80E-7
C6	Large containment failure before RPV failure, pool bypassed with injection to debris, Release location prior to RPV breach is wetwell	3.51E-6
C8	Large containment failure before RPV failure, pool bypassed with injection to debris, Release location prior to RPV breach is Drywell	9.41E-7
C10	Large containment failure before RPV failure, pool bypassed, no injection, wetwell release	2.62E-7
C12	Large containment failure before RPV failure, pool bypassed, no injection, DW release	1.82E-9
D5	Small containment failure after RPV failure, pool bypassed, no injection	9.91E-9
D6	Large containment failure after RPV failure, pool bypassed, no injection	9.07E-7
		2.25E-5(3)(4)

(1) Based on Clinton PSA Rev.3 model.

(2) Before and after RPV failure.

(3) Release mode frequency A0 for containment intact (no release) is not included in total.

(4) Release frequency is approximately 2.25E-5/yr as reported in the CPS Level 2 PSA Results Report. [18] Small differences in the summation are due to roundoff and varying truncation levels.

Table 3-4

COLLAPSED ACCIDENT PROGRESSION BIN (APB) DESCRIPTIONS [9]

Collapsed APB Number	Description
1	CD, vessel breach, Early CF, Early SP Bypass, CS Not Available Vessel breach occurs and both the containment and the drywell have failed either before or at the time of vessel breach. The containment sprays do not operate before or at the time of vessel breach.
2	CD, vessel breach, Early CF, Early SP Bypass, CS Available Vessel breach occurs and both the containment and the drywell fail either before or at the time of vessel breach. In this bin, however, the containment sprays operate before or at the time of vessel breach.
3	CD, vessel breach, Early CF, Late SP Bypass Vessel breach occurs and the containment fails either before or at the time of vessel breach. The drywell does not fail until the late time period and, thus, both the in-vessel releases and the releases associated with vessel breach are scrubbed by the suppression pool. Therefore, the availability of containment sprays during the time period that the suppression pool is not bypassed is not very important and, thus, the CS characteristic has been dropped.
4	CD, vessel breach, Early CF, No SP Bypass Vessel breach occurs and the containment fails either before or at the time of vessel breach. The drywell does not fail and, therefore, all of the radionuclide releases pass through the suppression pool. Because the pool has not been bypassed, the availability of the sprays is not very important and, thus, the CS characteristic has been dropped.
5	CD, vessel breach, Late CF Vessel breach occurs, however, the containment does not fail until the late time period. If the containment did not fail early, it is unlikely that the drywell will fail early. Thus, the suppression pool bypass characteristic and the containment spray characteristic have been dropped.
6	CD, vessel breach, Vent This summary bin represents the case in which vessel breach occurs and the containment was vented during any of the time periods in the accident.

Table 3-4

COLLAPSED ACCIDENT PROGRESSION BIN (APB) DESCRIPTIONS [9]

Collapsed APB Number	Description
7	CD, VB, No CF Vessel breach occurs but there is no containment failure and any releases associated with normal containment leakage are minor. Thus, the suppression pool bypass characteristic and the containment spray characteristic have been dropped. The risk associated with this bin will be negligible.
8	CD, No vessel breach Vessel breach is averted. Thus, there are no releases associated with vessel breach and there are no CCI releases. It must be remembered, however, that the containment can fail even if vessel breach is averted. Thus, the potential exists for some of the in-vessel releases to be released to the environment. It follows that there will be some risk associated with this bin.

Legend

CD = Core Damage

VB = Vessel Breach

CF = Containment Failure

WW = Wetwell

DW = Drywell

Table 3-5

GRAND GULF NUREG/CR-4551 50-MILE RADIUS POPULATION DOSE (1)

APB #	APB Definition	APB Frequency (per year) (2)	APB Fractional Contribution to 50-Mile Radius Total Dose Risk(3)	APB 50-Mile Radius Dose Risk (person-rem/year) (4)	APB 50-Mile Radius Dose (Person-rem) (5)
1	CD, vessel breach, Early CF, Early SP Bypass, CS Not Available Vessel breach occurs and both the containment and the drywell have failed either before or at the time of vessel breach. The containment sprays do not operate before or at the time of vessel breach.	6.46E-7	.268	0.139	2.2E+5
2	CD, vessel breach, Early CF, Early SP Bypass, CS Available Vessel breach occurs and both the containment and the drywell fail either before or at the time of vessel breach. In this bin, however, the containment sprays operate before or at the time of vessel breach.	2.00E-7	.056	0.029	1.5E+5
3	CD, vessel breach, Early CF, Late SP Bypass Vessel breach occurs and the containment fails either before or at the time of vessel breach. The drywell does not fail until the late time period and, thus, both the in-vessel releases and the releases associated with vessel breach are scrubbed by the suppression pool. Therefore, the availability of containment sprays during the time period that the suppression pool is not bypassed is not very important and, thus, the CS characteristic has been dropped.	2.86E-8	.011	5.7E-3	2.0E+5

Table 3-5

GRAND GULF NUREG/CR-4551 50-MILE RADIUS POPULATION DOSE (1)

APB #	APB Definition	APB Frequency (per year) (2)	APB Fractional Contribution to 50-Mile Radius Total Dose Risk(3)	APB 50-Mile Radius Dose Risk (person-rem/year) (4)	APB 50-Mile Radius Dose (Person-rem) (5)
4	CD, vessel breach, Early CF, No SP Bypass Vessel breach occurs and the containment fails either before or at the time of vessel breach. The drywell does not fail and, therefore, all of the radionuclide releases pass through the suppression pool. Because the pool has not been bypassed, the availability of the sprays is not very important and, thus, the CS characteristic has been dropped.	8.92E-7	.267	0.139	1.6E+5
5	CD, vessel breach, Late CF Vessel breach occurs, however, the containment does not fail until the late time period. If the containment did not fail early, it is unlikely that the drywell will fail early. Thus, the suppression pool bypass characteristic and the containment spray characteristic have been dropped.	1.16E-6	.281	0.146	1.3E+5
6	CD, vessel breach, Vent This summary bin represents the case in which vessel breach occurs and the containment was vented during any of the time periods in the accident.	1.55E-7	.039	0.0203	1.3E+5
7	CD, VB, No CF Vessel breach occurs but there is no containment failure and any releases associated with normal containment leakage are minor. Thus, the suppression pool bypass characteristic and the containment spray characteristic have been dropped. The risk associated with this bin will be negligible.	2.05E-7	3E-4	1.56E-4	7.6E+2

Table 3-5

GRAND GULF NUREG/CR-4551 50-MILE RADIUS POPULATION DOSE (1)

APB #	APB Definition	APB Frequency (per year) (2)	APB Fractional Contribution to 50-Mile Radius Total Dose Risk(3)	APB 50-Mile Radius Dose Risk (person-rem/year) (4)	APB 50-Mile Radius Dose (Person-rem) (5)
8	CD, No vessel breach Vessel breach is averted. Thus, there are no releases associated with vessel breach and there are no CCI releases. It must be remembered, however, that the containment can fail even if vessel breach is averted. Thus, the potential exists for some of the in-vessel releases to be released to the environment. It follows that there will be some risk associated with this bin.	7.36E-7	.077	0.040	5.4E+4
	Total:	4.09E-6	1.0	0.52	

(1) This table is presented in the form of a calculation because NUREG/CR-4551 does not document dose results as a function of accident progression bin (APB); as such, the dose results as a function of APB must be back calculated from documented APB frequencies and APB dose risk results in NUREG/CR-4551.

(2) The total (i.e., internal accident sequences) CDF of 4.09E-6/yr and the CDF subtotals by APB are taken from Figure 2.5-7 of NUREG/CR-4551 Vol. 6 Rev.1 Part I.

(3) The individual APB contributions to total (i.e., internal accident sequences) 50-mile radius dose rate are taken from Table 5.1-3 of NUREG/CR-4551 Vol. 6 Rev.1 Part I.

(4) The APB 50-mile dose risk is calculated by multiplying the individual APB dose risk contributions (column 4) by the total 50-mile radius dose risk of 0.52 person-rem/yr (taken from Table 5.1-1 of NUREG/CR-4551 Vol. 6 Rev.1 Part I).

(5) The individual APB doses are calculated by dividing the individual APB dose risk by the APB frequencies.

Table 3-6

CLINTON POPULATION DOSE BY APB:

ADJUSTED GRAND GULF NUREG/CR-4551[1]
50-MILE RADIUS POPULATION DOSES

APB #	Grand Gulf 50-Mile Radius Dose (Person-rem) [1]	Population Adjustment Factor	Reactor Power Adjustment Factor	Containment Leak Rate Adjustment Factor	Clinton Population Dose Adjusted 50-Mile Radius Dose (Person-rem)
1	2.2E+05	2.53	0.91	n/a	5.1E+05
2	1.5E+05	2.53	0.91	n/a	3.5E+05
3	2.0E+05	2.53	0.91	n/a	4.6E+05
4	1.6E+05	2.53	0.91	n/a	3.7E+05
5	1.3E+05	2.53	0.91	n/a	3.0E+05
6	1.3E+05	2.53	0.91	n/a	3.0E+05
7	7.6E+02	2.53	0.91	1.4	2.4E+03
8	5.4E+04	2.53	0.91	n/a	1.2E+05

[1]The NUREG/CR-4551 evaluation of Grand Gulf is used as input to the assessment of population dose for Clinton.

Table 3-7

CORRELATION OF THE CLINTON LEVEL 2 END STATES WITH THE MARK III SURROGATE APBs FROM NUREG/CR-4551[(1), (4)]

APB #	APB Definition	Clinton Release Modes	Clinton Frequencies (per year)[(5)]	Total Clinton Frequency (per year)
1	CD, vessel breach, Early CF, Early SP Bypass, CS Not Available Vessel breach occurs and both the containment and the drywell have failed either before or at the time of vessel breach. The containment sprays do not operate before or at the time of vessel breach.	C10 C12 C9, C11, E1, E2	2.62E-7 1.82E-9 ε	2.6E-7
2	CD, vessel breach, Early CF, Early SP Bypass, CS Available Vessel breach occurs and both the containment and the drywell fail either before or at the time of vessel breach. In this bin, however, the containment sprays operate before or at the time of vessel breach.	C1 C2 C6 C8	7.78E-8 1.80E-7 3.51E-6 9.41E-7	4.7E-6
3	CD, vessel breach, Early CF, Late SP Bypass Vessel breach occurs and the containment fails either before or at the time of vessel breach. The drywell does not fail until the late time period and, thus, both the in-vessel releases and the releases associated with vessel breach are scrubbed by the suppression pool. Therefore, the availability of containment sprays during the time period that the suppression pool is not bypassed is not very important and, thus, the CS characteristic has been dropped.		ε	ε
4	CD, vessel breach, Early CF, No SP Bypass Vessel breach occurs and the containment fails either before or at the time of vessel breach. The drywell does not fail and, therefore, all of the radionuclide releases pass through the suppression pool. Because the pool has not been bypassed, the availability of the sprays is not very important and, thus, the CS characteristic has been dropped.	B1 B2 A1[(2)]	1.13E-6 8.16E-6 7.81E-6	1.7E-5

Table 3-7

CORRELATION OF THE CLINTON LEVEL 2 END STATES WITH THE MARK III SURROGATE APBs FROM NUREG/CR-4551[(1), (4)]

APB #	APB Definition	Clinton Release Modes	Clinton Frequencies (per year)[(5)]	Total Clinton Frequency (per year)
5	CD, vessel breach, Late CF Vessel breach occurs, however, the containment does not fail until the late time period. If the containment did not fail early, it is unlikely that the drywell will fail early. Thus, the suppression pool bypass characteristic and the containment spray characteristic have been dropped.	D5 D6	9.91E-9 9.07E-7	9.2E-7
6	CD, vessel breach, Vent This summary bin represents the case in which vessel breach occurs and the containment was vented during any of the time periods in the accident.	(2)	ε	ε
7	CD, VB, No CF Vessel breach occurs but there is no containment failure and any releases associated with normal containment leakage are minor. Thus, the suppression pool bypass characteristic and the containment spray characteristic have been dropped. The risk associated with this bin will be negligible.	A0	5.47E-6	5.47E-6
8	CD, No vessel breach Vessel breach is averted. Thus, there are no releases associated with vessel breach and there are no CCI releases. It must be remembered, however, that the containment can fail even if vessel breach is averted. Thus, the potential exists for some of the in-vessel releases to be released to the environment. It follows that there will be some risk associated with this bin.	n/a	ε	ε
	Total:	--	--	2.8E-5[(3)]

Notes to Table 3-7

(1) Differs by less than 1% due to round off of individual contributors.

(2) Release mode A1 has been grouped in APB #4 with release modes B1 and B2 in lieu of APB #6 due to A1's magnitude and equivalent CPS source term category (ST1).]

(3) Within a few percent of total release frequency of 2.76E-5/yr. [18] Slight differences are due to the EPRI calculational approach and round off.

(4) This table provides a summary of the CPS Level 2 PRA results (Rev. 3). The frequencies in the last two columns are those from the CPS Level 2 results. These frequencies are given by CPS "release mode" and are associated with the Accident Progression Bin (APB) definitions from the surrogate Mark III analysis performed to support NUREG 1150 and reported in NUREG/CR-4551.

(5) Derived directly from the CPS Level 2 PSA Results Report [18].

Clinton Population Dose By EPRI Category

Table 3-6 provides the conversion of the surrogate plant 50-mile radius population dose to the CPS 50-mile radius population dose. This conversion is performed by accident progression bin (APB). This then needs to be converted to population dose for each EPRI category.

Using the preceding information, the population dose for the 50 mile radius surrounding Clinton is summarized in Table 3-8. (Use of dose results for the 50 mile radius around the plant as a figure of merit in the risk evaluation is consistent with NUREG-1150, past ILRT frequency extension submittals, and the NEI Interim Guidance.) The following discussion provides the basis for the assignment of population dose for each EPRI category.

The dose for the "no containment failure" EPRI category #1 is based on NUREG/CR-4551 APB #7 as the one closest to the definition of an intact containment.

The dose for EPRI Category 2 is based on NUREG/CR-4551 APB #1. This assignment is based on assuming that the containment isolation failure of EPRI Category 2 occurs coincident with bypass of the drywell due to failure of containment spray. APB #1 results in the highest dose of all the Grand Gulf "containment failure" APBs (which is indicative of a containment failure with suppression pool and drywell bypass).

No separate assignment of NUREG/CR-4551 APBs is made for EPRI Categories 3a and 3b. Instead, per the NEI Interim Guidance, the doses for EPRI Categories #3a and #3b are taken as factors of 10 and 35, respectively, times the population dose of EPRI Category 1.

As EPRI Categories 4, 5, and 6 are not affected by ILRT frequency and not analyzed as part of this risk assessment (per NEI Interim Guidance), no assignment of NUREG/CR-4551 APBs is made for these categories.

The dose for EPRI Category 7a is based on NUREG/CR-4551 APB #1.

The dose for EPRI Category 7b is based on NUREG/CR-4551 APB #2.

The dose for EPRI Category 7c is also based on NUREG/CR-4551 APB #4.

The dose for EPRI Category 7d is based on NUREG/CR-4551 APB #5.

The dose for the containment bypass category, EPRI Category 8, is based on NUREG/CR-4551 APB #1. APB #1 results in the highest dose of all the NUREG/CR-4551 "containment failure" APBs, indicative of containment bypass scenarios.

3.3.2 Baseline Population Dose Rate Estimates (Step 4)

The baseline dose rates per EPRI accident category are calculated by multiplying the population dose estimates from Table 3-8 by the frequencies summarized in Table 3-2 The resulting baseline population dose rates by EPRI category are summarized in Table 3-9. As the conditional containment pre-existing leakage probabilities for EPRI Categories 3a and 3b are reflective of a 3-per-10 year ILRT frequency (refer to Section 3.1), the baseline results shown in Table 3-9 are indicative of a 3-per-10 year ILRT surveillance frequency. The impact of Clinton's currently allowed 1-per-10-year ILRT surveillance frequency is discussed in Section 3.4.1.

3.4 IMPACT OF PROPOSED ILRT INTERVAL (STEPS 5-9)

Steps 5 through 9 of the NEI Interim Guidance assess the impact on plant risk due to the new ILRT surveillance interval in the following ways:

- Determine change in probability of detectable leakage (Step 5)
- Determine population dose rate for new ILRT interval (Step 6)
- Determine change in dose rate due to new ILRT interval (Step 7)
- Determine change in LERF risk measure due to new ILRT interval (Step 8)
- Determine change in CCFP due to new ILRT interval (Step 9)

Table 3-8

CLINTON POPULATION DOSE ESTIMATES AS A FUNCTION OF EPRI CATEGORY WITHIN 50-MILE RADIUS

EPRI Category	Category Description	Person-Rem Within 50 miles
1	No Containment Failure	2.4E+03
2	Containment Isolation System Failure	5.1E+05
3a	Small Pre-Existing Failures	2.4E+04
3b	Large Pre-Existing Failures	8.4E+04
4	Type B Failures (LLRT)	n/a
5	Type C Failures (LLRT)	n/a
6	Other Containment Isolation System Failure	n/a
7a	Containment Failure Due to Severe Accident (a)	5.1E+5
7b	Containment Failure Due to Severe Accident (b)	3.5E+5
7c	Containment Failure Due to Severe Accident (c)	3.7E+5
7d	Containment Failure Due to Severe Accident (d)	3.0E+5
8	Containment Bypass Accidents	5.1E+05

Table 3-9

CLINTON DOSE RATE ESTIMATES AS A FUNCTION OF EPRI CATEGORY FOR POPULATION WITHIN 50-MILES
(Base Line 3/10 year ILRT)

EPRI Category	Category Description	Person-Rem Within 50 miles[6]	Baseline Frequency (per year)[7]	Dose Rate (Person-Rem/yr)
1	No Containment Failure[1]	2.4E+3	4.85E-6	1.16E-2
2	Containment Isolation System Failure[2]	5.1E+5	1.13E-7	5.76E-2
3a	Small Pre-Existing Failures[3]	2.4E+4	5.6E-7	1.34E-2
3b	Large Pre-Existing Failures[3]	8.4E+4	5.6E-8	4.70E-3
4	Type B Failures (LLRT)	n/a	n/a	n/a
5	Type C Failures (LLRT)	n/a	n/a	n/a
6	Other Containment Isolation System Failure	n/a	n/a	n/a
7a	Containment Failure Due to Severe Accident (a)[4]	5.1E+5	2.63E-7	1.34E-1
7b	Containment Failure Due to Severe Accident (b)[4]	3.5E+5	4.7E-6	1.65
7c	Containment Failure Due to Severe Accident (c)[4]	3.7E+5	1.71E-5	6.33
7d	Containment Failure Due to Severe Accident (d)[4]	3.0E+5	9.2E-7	2.76E-1
8	Containment Bypass Accidents[5]	5.1E+5	1.21E-7	6.17E-2
	Total		2.86E-5[8]	8.53

Notes to Table 3-9

(1) The population dose associated with the Technical Specification Leakage is based on scaling the population data, the power level, and allowable Technical Specification leakage compared to the NUREG/CR-4551 reference plant. The release for this EPRI category is assigned from APB#7 from Table 3-4.

(2) EPRI Category #2 (Containment Isolation failures) may include drywell isolation failures. Therefore, the release associated with this category is assigned to be equivalent to the release associated with APB#1 from Table 3-4.

(3) Dose estimates for #3a and #3b, per the NEI Interim Guidance, are calculated as 10xCategory 1 dose and 35xCategory 1 dose, respectively.

(4) Dose estimate for 7a, 7b, 7c, and 7d are taken from APB # 1, 2, 4, and 5, respectively.

(5) EPRI Category #8 sequences involve containment bypass failures; as a result, the person-rem dose is not based on normal containment leakage. The releases for this category are assumed to result in a direct path to the environment, and as such, are assigned to be equivalent to the highest release category from NUREG/CR-4551. APB#1 from Table 3-4 is therefore used.

(6) Table 3-6.

(7) Table 3-1.

(8) Within a few percent of total CDF of 2.76E-5/yr [18]. Slight differences are due to the EPRI calculational approach and round off. The use of slightly higher frequencies in Table 3-9 is conservative for assessing the risk metric of dose rate.

3.4.1 Change in Probability of Detectable Leakage (Step 5)

Step 5 of the NEI Interim Guidance is the calculation of the change in probability of leakage detectable only by ILRT (and associated re-calculation of the frequencies of the impacted EPRI categories). Note that with increases in the ILRT surveillance interval, the size of the postulated leak path and the associated leakage rates are assumed not to change; however, the probability of pre-existing leakage detectable only by ILRT does increase.

Per the NEI Interim Guidance, the calculation of the change in the probability of a pre-existing ILRT-detectable containment leakage is based on the relationship that relaxation of the ILRT interval results in increasing the average time that a pre-existing leak would exist undetected. Using the standby failure rate statistical model, the average time that a pre-existing containment leak would exist undetected is one-half the surveillance interval. For example, if the ILRT frequency is 1-per-10 years, then the average time that a leak would be undetected is 60 months (surveillance interval of 120 months divided by 2). The impact on the leakage probability due to the ILRT interval extension is then calculated by applying a multiplier determined by the ratio of the average times of undetection for the two ILRT interval cases.

As discussed earlier in Section 3.1, the conditional probability of a pre-existing ILRT-detectable containment leakage is divided into two categories. The calculated pre-existing ILRT-detectable leakage probabilities are reflective of a 3-per-10 year ILRT frequency and are as follows:

- "Small" pre-existing leakage (EPRI Category 3a): 2.70E-2
- "Large" pre-existing leakage (EPRI Category 3b): 2.70E-3

Since the latter half of the 1990's, the Clinton plant has been operating under a 1-per-10 year ILRT testing frequency consistent with the performance-based Option B of 10

CFR Part 50, Appendix J. [16] The baseline[1] leakage probabilities first need to be adjusted to reflect the current 1-per-10 year Clinton ILRT testing frequency, as follows:

- "Small" : 2.70E-2 x [(120 months/2) / (36 months/2)] = 9.00E-2
- "Large" : 2.70E-3 x [(120 months/2) / (36 months/2)] = 9.00E-3

Note that a nominal 36 month interval (i.e., as opposed to 40 months, 120/3) is used in the above adjustment calculation to reflect the 3-per-10 year ILRT frequency. This is consistent with operational practicalities and the NEI Interim Guidance.

Similarly, the pre-existing ILRT-detectable leakage probabilities for the 1-per-15 year ILRT frequency currently being pursued by Clinton (and the subject of this risk assessment) are calculated as follows:

- "Small" : 9.00E-2 x [(180 months/2) / (120 months/2)] = 1.35E-1
- "Large" : 9.00E-3 x [(180 months/2) / (120 months/2)] = 1.35E-2

Given the above adjusted leakage probabilities, the impacted frequencies of the EPRI categories are summarized below (refer to Table 3-2 for details regarding frequency calculations for the individual EPRI categories):

EPRI Category	EPRI Category Frequency as a Function of ILRT Interval		
	Baseline (3-per-10 year ILRT)	Current (1-per-10 year ILRT)	Proposed (1-per-15 year ILRT)
1	4.85E-6	3.41E-6	2.39E-6
3a	5.60E-7	1.87E-6	2.80E-6
3b	5.60E-8	1.87E-7	2.80E-7

(1) The baseline case uses data characteristic of the 3/10 year ILRT frequency of testing.

Note that, per the definition of the EPRI categories, only the frequencies of Categories 1, 3a, and 3b are impacted by changes in ILRT testing frequencies.

3.4.2 Population Dose Rate for New ILRT Interval (Step 6)

The dose rates per EPRI accident category as a function of ILRT interval are summarized in Table 3-10.

3.4.3 Change in Population Dose Rate Due to New ILRT Interval (Step 7)

As can be seen from the dose rate results summarized in Table 3-10, the calculated total dose rate increases imperceptibly (0.32%) from the current Clinton 1-per-10 year ILRT interval amount of 8.57 person-rem/year to the proposed 1-per-15 year ILRT interval amount of 8.60 person-rem/year.

Per the NEI Interim Guidance, the change in percentage contribution to total dose rate attributable to EPRI Categories 3a and 3b is also investigated here. Using the results summarized in Table 3-10, for the current Clinton 1-per-10 year ILRT interval, the percentage contribution to total dose rate from Categories 3a and 3b is shown to be very minor:

$$[\,(4.49E\text{-}2 + 1.57E\text{-}2) / 8.57\,] \times 100 = 0.71\%$$

For the proposed 1-per-15 year ILRT interval, the percentage contribution to total dose rate from Categories 3a and 3b increases slightly but remains very minor:

$$[\,(6.72E\text{-}2 + 2.35E\text{-}2) / 8.60\,] \times 100 = 1.05\%$$

Table 3-10

BASELINE DOSE RATE ESTIMATES BY EPRI ACCIDENT CATEGORY FOR POPULATION WITHIN 50-MILE

EPRI Category	Category Description	Dose Rate as a Function of ILRT Interval (Person-Rem/Yr)		
		Baseline (3-per-10 year ILRT)	Current (1-per-10 year ILRT)	Proposed (1-per-15 year ILRT)
1	No Containment Failure	1.16E-2	8.18E-3	5.74E-3
2	Containment Isolation System Failure	5.76E-2	5.76E-2	5.76E-2
3a	Small Pre-Existing Failures	1.34E-2	4.49E-2	6.72E-2
3b	Large Pre-Existing Failures	4.70E-3	1.57E-2	2.35E-2
4	Type B Failures (LLRT)	N/A	N/A	N/A
5	Type C Failures (LLRT)	N/A	N/A	N/A
6	Other Containment Isolation System Failure	N/A	N/A	N/A
7a	Containment Failure Due to Severe Accident (a)	1.34E-1	1.34E-1	1.34E-1
7b	Containment Failure Due to Severe Accident (b)	1.65	1.65	1.65
7c	Containment Failure Due to Severe Accident (c)	6.33	6.33	6.33
7d	Containment Failure Due to Severe Accident (d)	2.76E-1	2.76E-1	2.76E-1
8	Containment Bypass Accidents	6.17E-2	6.17E-2	6.17E-2
	TOTAL:	8.53	8.57	8.60

3.4.4 Change in LERF Due to New ILRT Interval (Step 8)

The risk increase associated with extending the ILRT interval involves the potential that a core damage event that normally would not result in a radionuclide release from an intact containment could in fact result in a release due to the increase in probability of failure to detect a pre-existing leak. Per the NEI Interim Guidance, only Category 3b sequences have the potential to result in large releases if a pre-existing leak were present. As such, the change in LERF (Large Early Release Frequency) is determined by the change in the frequency of Category 3b.

Category 1 accidents are not considered as potential large release pathways because the containment remains intact. Therefore, the containment leak rate is expected to be small. Similarly, Category 3a is a "small" pre-existing leak. Other accident categories such as 2, 6, 7, and 8 could result in large releases but these are not affected by the change in ILRT interval. Late releases are excluded regardless of the size of the leak because late releases are, by definition, not LERF contributors.

The impact on the LERF risk measure due to the proposed ILRT interval extension is calculated as follows:

delta LERF = (Frequency of EPRI Category 3b for 1-per-15 year ILRT interval) – (Frequency of EPRI Category 3b for 1-per-10 year ILRT interval)

= 2.80E-7/yr – 1.87E-7/yr

= 9.30E-8/yr

This delta LERF of 9.30E-8/yr falls into Region III, Very Small Change in Risk, of the acceptance guidelines in NRC Regulatory Guide 1.174. Therefore, increasing the ILRT interval at Clinton from the currently allowed 1-per-10 years to 1-per-15 years represents a very small change in risk, and is an acceptable plant change from a risk perspective.[1]

3.4.5 Impact on Conditional Containment Failure Probability (Step 9)

Another parameter that the NRC Guidance in Reg. Guide 1.174 states can provide input into the decision-making process is the consideration of change in the conditional containment failure probability (CCFP). The change in CCFP is indicative of the effect of the ILRT on all radionuclide releases, not just LERF. The conditional containment failure probability (CCFP) can be calculated from the risk calculations performed in this analysis. In this assessment, based on the NEI Interim Guidance, CCFP is defined such that containment failure includes all radionuclide release end states other than the intact state (EPRI Category 1) and small failures (EPRI Category 3a). The conditional part of the definition is conditional given a severe accident (i.e., core damage).

Consequently, the change in CCFP can be calculated by the following equation:

$$CCFP_{\%} = [1 - (\text{Intact Containment Frequency} / \text{Total CDF})] \times 100\%, \text{ or}$$
$$= [1 - ((\#1 \text{ Frequency} + \#3a \text{ Frequency}) / \text{CDF})] \times 100\%$$

For the 10-year interval:

$$CCFP_{10} = [1 - ((3.41E\text{-}6 + 1.87E\text{-}6) / 2.76E\text{-}5)] \times 100\%$$
$$= 80.9\%$$

For a 15-year interval:

$$CCFP_{15} = [1 - ((2.39E\text{-}6 + 2.80E\text{-}6) / 2.76E\text{-}5)] \times 100\%$$

(1) Note that if this conservative estimate of delta LERF due to internal events increases, then the ILRT extension would fall into Region II of the Reg. Guide 1.174 acceptance guideline, i.e., the small risk increase.

$$= 81.2\%$$

Therefore, the change in the conditional containment failure probability is:

$$\Delta\,CCFP = CCFP_{15} - CCFP_{10} = 0.3\%$$

This change in CCFP of less than 1% is insignificant from a risk perspective.

Section 4
RESULTS SUMMARY

The application of the approach based on NEI Interim Guidance [3, 21], EPRI-TR-104285 [2] and previous risk assessment submittals on this subject [6, 20, 22] have led to the quantitative results summarized in this section. These results demonstrate a very small impact on risk associated with the one time extension of the ILRT test interval to 15 years.

The analysis performed examined Clinton specific accident sequences in which the containment remains intact or the containment is impaired. The accidents are analyzed and the results are displayed according to the eight (8) EPRI accident categories defined in Reference [2]:

1. Containment intact and isolated
2. Containment isolation failures due to support system or active failures
3. Type A (ILRT) related containment isolation failures
4. Type B (LLRT) related containment isolation failures
5. Type C (LLRT) related containment isolation failures
6. Other penetration related containment isolation failures
7. Containment failure due to core damage accident phenomena
8. Containment bypass

The quantitative results are summarized in Table 4-1. The key results to this risk assessment are those for the ten year interval (current Clinton condition) and the fifteen year interval (proposed change). The 3-per-10 year ILRT is a baseline starting point for this risk assessment given that the pre-existing containment leakage probabilities (estimated based on industry experience - - refer to Section 3.1) are reflective of the 3-per-10 year ILRT testing.

The following is a brief summary of some of the key aspects of the ILRT test interval extension risk analysis:

- Increasing the current 10 year ILRT interval to 15 years results in an insignificant increase in total population dose rate of 0.3 percentage points.

- The increase in the LERF risk measure is also insignificant, a 9.30E-8/yr increase. This LERF increase is categorized as a "very small" increase per NRC Reg. Guide 1.174.

- Likewise, the conditional containment failure probability ($CCFP_{\%}$) increases insignificantly by 0.3 percentage points.

Table 4-1

QUANTITATIVE RESULTS AS A FUNCTION OF ILRT INTERVAL

EPRI Category	Dose (Person-Rem Within 50 miles)	Quantitative Results as a Function of ILRT Interval					
		Baseline (3-per-10 year ILRT)		Current (1-per-10 year ILRT)		Proposed (1-per-15 year ILRT)	
		Accident Frequency (per year)	Population Dose Rate (Person-Rem/Year Within 50 miles)	Accident Frequency (per year)	Population Dose Rate (Person-Rem/Year Within 50 miles)	Accident Frequency (per year)	Population Dose Rate (Person-Rem/Year Within 50 miles)
1	2.4 E+3	4.85E-6	1.16E-2	3.41E-6	8.18E-3	2.39E-6	5.74E-3
2	5.1E+5	1.13E-7	5.76E-2	1.13E-7	5.76E-2	1.13E-7	5.76E-2
3a	2.4E+4	5.6E-7	1.34E-2	1.87E-6	4.49E-2	2.80E-6	6.72E-2
3b	8.4E+4	5.6E-8	4.70E-3	1.87E-7	1.57E-2	2.80E-7	2.35E-2
4	N/A	N/A	N/A	N/A	N/A	N/A	N/A
5	N/A	N/A	N/A	N/A	N/A	N/A	N/A
6	N/A	N/A	N/A	N/A	N/A	N/A	N/A
7a	5.1E+5	2.63E-7	1.34E-1	2.63E-7	1.34E-1	2.63E-7	1.34E-1
7b	3.5E+5	4.7E-6	1.65	4.7E-6	1.65	4.7E-6	1.65
7c	3.7E+5	1.71E-5	6.33	1.71E-5	6.33	1.71E-5	6.33
7d	3.0E+5	9.2E-7	2.76E-1	9.2E-7	2.76E-1	9.2E-7	2.76E-1
8	5.1E+5	1.21E-7	6.17E-2	1.21E-7	6.17E-2	1.21E-7	6.17E-2
TOTALS:		2.87E-5(4)	8.53	2.87E-5(4)	8.57	2.87E-5(4)	8.60
Increase in Dose Rate (1)					0.5%		0.3%
Increase in LERF (2)				1.31E-7		9.30E-8	
Increase in CCFP (%) (3)				0.5%		0.3%	

Notes to Table 4-1:

(1) The increase in dose rate (person-rem/year) is with respect to the results for the preceding ILRT interval, as presented in the table. For example, the increase in dose rate for the proposed 1-per-15 ILRT is calculated as: total dose rate for 1-per-15 year ILRT, minus total dose rate for 1-per-10 year ILRT. For each case, the dose rate increase is insignificant.

(2) The increase in Large Early Release Frequency (LERF) is with respect to the results for the preceding ILRT interval, as presented in the table. As discussed in Section 3.4.4 of the report, the change in LERF is determined by the change in the accident frequency of EPRI Category 3b. For example, the increase in LERF for the proposed 1-per-15 ILRT is calculated as: 3b frequency for 1-per-15 year ILRT, 2.80E-7/yr, minus 3b frequency for 1-per-10 year ILRT, 1.87E-7/yr, equals 9.30E-8/yr.

(3) The increase in the conditional containment failure probability (CCFP) is with respect to the results for the preceding ILRT interval, as presented in the table. As discussed in Section 3.4.5, the conditional containment failure probability (CCFP) is calculated as:

$$CCFP_{\%} = [1 - ((\text{Category \#1 Frequency} + \text{Category \#3a Frequency}) / \text{CDF})] \times 100\%$$

(4) Due to the NEI methodology and round off, the total frequency of all severe accidents is slightly higher than the CPS Rev 3 reported CDF (approximately 4%). This in turn leads to slightly higher population dose rate estimates for the Baseline, the current, and the proposed ILRT frequencies.

Section 5

CONCLUSIONS

A risk assessment of the impact of changing the Clinton Integrated Leak Rate Test (ILRT) interval from the currently approved 10 year interval to a one-time extension to 15 years has been performed. Sections 5.1 to 5.4 summarize the conclusions regarding this risk assessment. In addition, a risk assessment of extending the test interval for both the Integrated Leak Rate Test (ILRT) and the Drywell Bypass Test (DWBT) is also provided (see Appendix C). This latter assessment is summarized in Section 5.5.

5.1 QUANTITATIVE CONCLUSIONS

The conclusions from the risk assessment of the one time ILRT extension can be characterized by the risk metrics used in previously approved ILRT test interval extensions. These include:

- Change in LERF
- Change in conditional containment failure probability
- Change in population dose rate

5.1.1 LERF

Based on the results from Sections 3 and 4, the main conclusion regarding the impact on plant risk associated with extending the Type A ILRT test frequency from ten years to fifteen years is:

> Reg. Guide 1.174 [4] provides guidance for determining the risk impact of plant-specific changes to the licensing basis. Reg. Guide 1.174 defines very small changes in risk as resulting in increases of CDF below 10^{-6}/yr and increases in LERF below 10^{-7}/yr. Since the ILRT does not impact CDF, the relevant criterion is LERF. The increase in LERF resulting from a change in the Type A ILRT test interval from once-per-ten years to once-per-fifteen years (using the change in the EPRI Category 3b frequency per the NEI Interim Guidance) is 9.30E-8/yr. Guidance in Reg. Guide 1.174 defines very small changes in LERF as below 10^{-7}/yr. Therefore, increasing the Clinton ILRT interval from 10 to 15 years results in a very small change in risk, and is an acceptable plant change from a risk perspective.

5.1.2 CCFP

The change in conditional containment failure probability (CCFP) is also calculated as an additional risk measure to demonstrate the impact on defense-in-depth. The ΔCCFP is found to be very small (0.3% increase) and represents a negligible change in the Clinton defense-in-depth.

5.1.3 Population Dose Rate

The change in population dose rate is also reported consistent with previously approved ILRT interval extension requests. The change in population dose rate from the current 1/10 year ILRT frequency to 1/15 year frequency is an insignificant 0.3% increase (0.03 person rem/yr).

5.2 RISK TRADE-OFF

The performance of an ILRT introduces risk. An EPRI study of operating experience events associated with the performance of ILRTs has indicated that there are real risk impacts associated with the setup and performance of the ILRT during shutdown operation [8]. While these risks have not been quantified for Clinton, it is judged that there is a positive (yet unquantified) safety benefit associated with the avoidance of frequent ILRTs.

The safety benefits relate to the avoidance of plant conditions and alignments associated with the ILRT which place the plant in a less safe condition leading to events related to drain down or loss of shutdown cooling. Therefore, while the focus of this evaluation has been on the negative aspects, or increased risk, associated with the ILRT extension, there are, in fact, positive safety benefits associated with reducing the risk contribution from shutdown risk configurations.

5.3 EXTERNAL EVENTS IMPACT

External hazards were evaluated in the Clinton Individual Plant Examination of External Events (IPEEE) Submittal in response to the NRC IPEEE Program (Generic Letter 88-20 Supplement 4). The IPEEE Program was a one-time review of external hazard risk to identify potential plant vulnerabilities and to understand severe accident risks. Clinton does not currently maintain external event PSA models and associated documentation. Although the external event hazards in the Clinton IPEEE were evaluated to varying levels of conservatism, the results of the Clinton IPEEE are nonetheless used in this risk assessment to provide a conservative comparison of the impact of external hazards on the conclusions of this ILRT interval extension risk assessment.

The impact of external events on this ILRT risk assessment is summarized in this section (refer to Appendix B for further details).

The purpose is to assess whether there are any unique insights or important quantitative information associated with the explicit consideration of external events in the risk assessment results.

Given the characteristics of the proposed plant change (i.e., ILRT interval extension), specific quantitative information regarding the impact on external event hazard risk measures is not a significant decision making input. The proposed ILRT interval extension impacts plant risk in a very specific and limited way. The probability of a pre-existing containment leak being the initial containment failure mode given a core damage accident is potentially higher when the ILRT interval is extended. This impact is manifested in the plant risk profile in a similar manner for both internal events and external events.

The spectrum of external hazards has been evaluated in the Clinton IPEEE by screening methods with varying levels of conservatism. Therefore, it is not possible at this time to incorporate realistic quantitative risk assessments of all external event hazards into the ILRT extension assessment. As a result, external events have been evaluated as a sensitivity case to show that the conclusions of this analysis would not be altered if external events were explicitly considered.

The quantitative consideration of external hazards is discussed in more detail in Appendix B of this report. As can be seen from Appendix B, if the external hazard risk results of the Clinton IPEEE are included in this assessment (i.e., in addition to internal events), the change in LERF associated with the increase in ILRT interval from 10 years to 15 years will be 1.08E-7/yr. This delta LERF is just slightly above the Region III boundary for LERF and falls within NRC RG 1.174 Region II ("Small Changes" in risk). As noted above, this can be attributed to the conservative screening nature of the external event methods available for their quantitative assessment at CPS. Consistent with Reg. Guide 1.174, the total Clinton LERF from internal and external events was then also calculated at 8.39E-7/yr to demonstrate that LERF is acceptable. This is significantly less than the Reg. Guide 1.174 acceptance guideline of 1E-5/yr (refer to Appendix B).

Therefore, incorporating external event accident sequence results into this analysis does not change the conclusion of this risk assessment (i.e., increasing the Clinton ILRT interval from 10 to 15 years is an acceptable plant change from a risk perspective).

5.4 PREVIOUS ASSESSMENTS

The NRC in NUREG-1493 [5] has previously concluded that:

- Reducing the frequency of Type A tests (ILRTs) from 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 existing requirements.

- Given the insensitivity of risk to containment leakage rate 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 impact of relaxing the ILRT frequency beyond one in 20 years has not been evaluated. Beyond testing the performance of containment penetrations, ILRTs also test the integrity of the containment structure.

The findings for Clinton confirm the above general findings on a plant specific basis when considering (1) Clinton severe accident risk profile, (2) the Clinton containment failure modes, and (3) the local population surrounding the Clinton site.

5.5 RISK METRICS FOR EXTENDING THE TEST INTERVAL OF BOTH ILRT AND DWBT

This section summarizes the <u>combined</u> risk increase associated with the change in <u>both</u> the ILRT <u>and</u> DWBT intervals. This combination of changes results in a very slight increase in the risk measures relative to those calculated for the ILRT interval change by itself.

As in the case of the ILRT, the conclusions from the risk assessment of the one-time ILRT and DWBT interval extension can be characterized by the risk metrics used in previously approved ILRT interval extensions for other plants. These include:

- Change in LERF
- Change in conditional containment failure probability
- Change in population dose rate

Based on the results from Appendix C, the main conclusion regarding the impact on plant risk associated with extending the ILRT and DWBT interval from ten years to fifteen years is:

Reg. Guide 1.174 [4] provides guidance for determining the risk impact of plant-specific changes to the licensing basis. Reg. Guide 1.174 defines

> very small changes in risk as resulting in increases of CDF below 10^{-6}/yr and increases in LERF below 10^{-7}/yr. Since the ILRT and DWBT do not impact CDF, the relevant criterion is LERF. The increase in LERF resulting from a change in the ILRT and DWBT interval from ten years to fifteen years (using the change in the EPRI Category 3b frequency per the NEI Interim Guidance) is 1.4E-7/yr. Guidance in Reg. Guide 1.174 defines small changes in LERF as above 10^{-7}/yr and less than 10^{-6}/yr. Therefore, increasing the Clinton ILRT and DWBT interval from 10 to 15 years results in a small change in risk, and is an acceptable plant change from a risk perspective.

Per Reg. Guide 1.174, when the calculated increase in LERF due to the proposed plant change is in the range of 1E-7 to 1E-6 per reactor year (Region II, "small change" in risk), the risk assessment must also reasonably show that the total LERF is less than 1E-5.

Per the Clinton internal events PSA (Rev. 3) documentation, the Clinton LERF due to internal event accidents is 2.63E-7/yr. Therefore, the total LERF for Clinton of 2.63E-7/yr is significantly less than the Reg. Guide 1.174 acceptance guideline of 1E-5/yr.

It is emphasized that the radionuclide release (e.g., CsI release fraction) calculated for Class 3b using MAAP (see Appendix C) is significantly below that which has been attributed to LERF releases. [C-25] Therefore, the NEI/EPRI characterization of Category 3b as a LERF contributor is considered extremely conservative for a Mark III.

<u>CCFP</u>

The change in conditional containment failure probability (CCFP) is also calculated as an additional risk measure to demonstrate the impact on defense-in-depth. The ΔCCFP is found to be very small (0.5% increase) and represents a negligible change in the Clinton defense-in-depth.

Population Dose Rate

The change in population dose rate is also reported consistent with previously approved ILRT interval extension requests. The change in population dose rate from the current 1/10 year ILRT and DWBT frequency to 1/15 year frequency is an insignificant 0.48% increase (.04 person rem/yr).

Summary

The findings for Clinton confirm that the risk change associated with extending the ILRT and DWBT interval from 10 years to 15 years is small when considering (1) Clinton severe accident risk profile, (2) the Clinton containment failure modes, and (3) the local population surrounding the Clinton site.

Section 6

REFERENCES

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[4] U.S. Nuclear Regulatory Commission, *"An Approach for Using Probabilistic Risk Assessment in Risk-Informed Decisions on Plant-Specific Changes to the Licensing Basis"*, Regulatory Guide 1.174, July 1998.

[5] *Performance-Based Containment Leak-Test Program,* NUREG-1493, September 1995.

[6] Letter from R.J. Barrett (Entergy) to U.S. Nuclear Regulatory Commission, IPN-01-007, dated January 18, 2001.

[7] United States Nuclear Regulatory Commission, Indian Point Nuclear Generating Unit No. 3 - Issuance of Amendment Re: Frequency of Performance-Based Leakage Rate Testing (TAC No. MB0178), April 17, 2001.

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[17] ERIN Engineering and Research, Inc., Identification of Risk Implications Due to Extended Power Uprate at Clinton, Doc. #C14670108-4441, August 2001.

[18] "Clinton PRA Update Level 2 Results", P21-00 (06-27)-6, Y-109204, dated June 27, 2000.

[19] "Clinton PRA Update Level 1 Results", P21-00 (05-31)-6, Y-109190, May 31, 2000.

[20] Letter from D.E. Young (Florida Power) to U.S. Nuclear Regulatory Commission, 3F0401-11, dated April 25, 2001

[21] Letter from A. Petrangelo (NEI) to NEI Administrative Points of Cont*act, "One-Time Extension of Containment Integrated Leak Rate Test Interval – Additional Information"*, November 30, 2001.

[22] Letter from J.A. Hutton (Exelon, Grand Gulf) to U.S. Nuclear Regulatory Commission, Docket No. 50-278, License No. DPR-56, LAR 01-00430, dated May 30, 2001.

Appendix A

CLINTON POPULATION DATA

This appendix includes the population estimates for the following:

- Appendix A.1: 50-Mile Radius Population Data Used to Characterize Grand Gulf Population Dose Calculations in NUREG/CR-4551
- Appendix A.2: 50-Mile Radius Population from Clinton Power Station

A.1 POPULATION DATA USED TO CHARACTERIZE GRAND GULF POPULATION DOSE CALCULATIONS IN NUREG/CR-4551

Background

NEI Interim Guidance for the ILRT internal extension licensing request includes the option to use NRC Ex-Plant consequences from NUREG-1150 if a plant does not have a plant specific Level 3 PRA. This approach is used for the Clinton ILRT analysis.

Analysis

The Population Dose (Person Rem) calculation for the Mark III surrogate source terms is derived from the NRC's landmark study of reactor risks in NUREG-1150 for the Grand Gulf plant. In order to relate that 50 mile population dose calculation from Grand Gulf to Clinton, the population information for both sites is needed to properly scale the calculated dose from Grand Gulf to Clinton.

This section derives the population within 50 miles of Grand Gulf used to support the NUREG-1150 risk estimates.

The following table gives the population within certain distances of the plant as summarized from the MACCS demographic input based on 1980 Census Tapes (P. 4.3 of NUREG/CR 4551 Vol. 6.)

Distance From Plant (Km)	(miles)	Population
1.6	1.0	34
4.8	3.0	879
16.1	10.0	10,255
48.3	30.0	97,395
160.9	100.0	1,614,883
563.3	350.0	22,259,422
1609.3	1000.0	142,024,448

Two methods are used for the estimate of the population within 50 miles of Grand Gulf:

- Method 1: Assume Direct proportion of the population with area
- Method 2: Interpolate between estimates for 30 miles and 100 miles as a function of area.

Method 1: Assume Direct proportion of the population with area

A) Assume direct proportion with 30 mile data

$$\frac{\pi R_1^2}{97,395} = \frac{\pi R_2^2}{X}$$

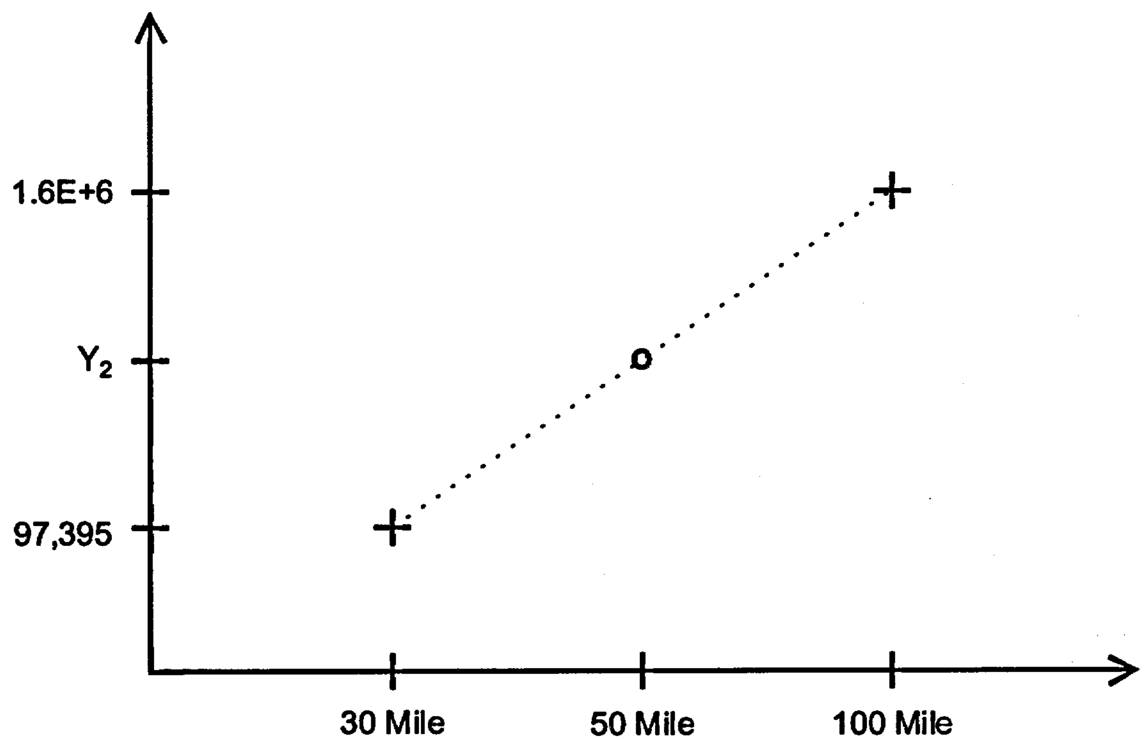
$X = 97,395 (R_2^2/R_1^2) = 2.7E+5$ persons

B) Assume direct proportion with 100 mile data

$Z = 1.61E+6 (50^2/100^2) = 4.02E+5$ persons

Avg = 3.4E+5 persons

<u>Method 2</u>: Interpolate Based on Area



2.83E+3 mi² 7.85E+3 mi² 3.14E+4 mi²

$Y = mx + b$

$M = \dfrac{Y_3 - Y_1}{X_3 - X_1}$

$Y_2 - Y_1 = m(X_2 - X_1)$

$Y_1 = 97{,}395$

$$Y_2 = 97{,}395 + \frac{1.61E{+}6 - 97{,}395}{3.14E{+}4 - 2.83E{+}3} * (7.85E{+}3 - 2.83E{+}3)$$

$Y_2 = 3.6E{+}5$ persons

The two methods yield estimates that are very close. The smaller estimate, 3.4E+5, is chosen since this will lead to a more conservative estimate of the risk at Clinton when the person-rem are scaled to the Clinton site.

A.2 YEAR 2000 50-MILE RADIUS POPULATION AROUND CLINTON

A calculation of the 2000 50-mile radius population around Clinton was performed in support of this risk assessment.

This calculation uses 2000 Census data, as reported by the US Census Bureau on the web site http://quickfacts.census.gov/qfd/states/17000.html, along with Illinois maps to perform the population estimation.

The Clinton plant is located 6 miles east of Clinton in De Witt County, Illinois. The location of the site and the 50-mile radius is illustrated in Figure A-1. If the entire county falls within the 50-mile radius, based on a review of a map containing a mileage scale and county borders, then the entire population was included in the population estimate. Otherwise, a fraction of the population was counted based on the percentage of the county within the 50-mile radius. The land area within the 50-mile radius was estimated based on visual inspection of the map and the population of that area was estimated assuming uniform distribution of the population within the county.

Five counties were completely inside the fifty-mile radius. For the other counties, their percentage included in the fifty-mile radius was estimated and then multiplied by their total population based on the 2000 Census data. Since the population densities within some counties varied greatly, exceptions were made for the following counties: Champaign, Christian, Coles, Sangamon, and Tazewell.

<u>Champaign County</u>: Champaign (65,000), and Urbana (35,000) both lie within the 50 mile zone. These two cities account for approximately 60% of the population of the county. The remaining population (70,000) is assumed to be uniformly distributed and can be multiplied by the percent of the county that is in the 50 mile zone (80%). As a result the total population that falls within the 50 mile zone is 156,000. This compares with the UFSAR estimate of approximately 137,661 in 1990.

Christian County: The town of Taylorville (12,500) lies within the 50 mile zone. This town accounts for approximately 33% of the population of the county. The remaining population (24,000) is assumed to be uniformly distributed and can be multiplied by the percent of the county that is in the 50 mile zone (75%). As a result the total population that falls within the 50 mile zone is approximately 30,000. This compares with the 17,243 reported in the UFSAR for 1990.

Coles County: The towns of Mattoon (18,000) and Charleston (21,000) lie outside of the 50 mile zone. The remaining population (12,000) is assumed to be uniformly distributed and can be multiplied by the percent of the county that is in the 50 mile zone (10%). As a result the total population that falls within the 50 mile zone is 1,200. This compares with the 513 reported in the UFSAR for 1990.

Sangamon County: The fifty-mile radius includes Springfield (117,000), which is approximately 60% of the population of the county. The remaining population (70,000) is assumed to be uniformly distributed and can be multiplied by the percent of the county that is in the 50 mile zone (60%). As a result the total population that falls within the 50 mile zone is approximately 160,000. This compares with the 118,675 reported in the UFSAR for 1990.

Tazewell County: The town of East Peoria (25,000) lies outside the 50 mile zone. The remaining population (100,000) is assumed to be uniformly distributed and can be multiplied by the percent of the county that is in the 50 mile zone (95%). As a result the total population that falls within the 50 mile zone is 95,000. This compares with the 90,535 reported in the UFSAR for 1990.

Figure A-1

ILLUSTRATION OF 50-MILE RADIUS AROUND CLINTON SITE

A list of the counties within the 50 mile radius of Clinton, along with their total population, the percent area the county lies within the 50 mile radius, and the population within the 50 mile zone is summarized in Table A-1. The total year 2000 population within a 50-mile radius of Clinton Nuclear Station is estimated at 856,996 persons.

The UFSAR includes estimates of population centers within the 50 mile radius from Clinton. Table 2.1-2 (Rev. 5) from the UFSAR is included for information. It includes population center estimates for both 1980 and 1990.

Table A-1

2000 CENSUS POPULATION FOR COUNTIES
WITHIN 50 MILE RADIUS OF CLINTON

(Source: http://quickfacts.census.gov/qfd/states/17000.html)

County	2000 Census Total Population of all Counties	Percent Area of County in 50 Mile Radius[1]	Population within 50 Mile Radius[2]
Champaign	179,669	80	156,000[3]
Christian	35,372	75	30,000[3]
Coles	53,196	10	1,200[3]
DeWitt	16,798	100	16,798
Douglas	19,922	50	9,961
Ford	14,241	40	5,696
Livingston	39,678	15	5,952
Logan	31,183	100	31,183
McLean	150,433	100	150,433
Macon	114,706	100	114,706
Mason	16,038	50	8,019
Menard	12,486	85	10,613
Moultrie	14,287	90	12,858
Piatt	16,365	100	16,365
Sangamon	188,951	60	160,000[3]
Shelby	22,893	40	9,157
Tazewell	128,485	95	95,000[3]
Woodford	35,469	65	23,055
TOTALS	1,273,605	—	856,996

(1) Based on visual inspection of Illinois state maps.

(2) County Population multiplied by percentage within 50 mile zone, except when noted.

(3) Population density varied greatly in this region, an exception was made.

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TABLE 2.1-2

CITIES, TOWNS AND VILLAGES WITHIN 50 MILES OF CLINTON POWER STATION

CITY OR TOWN	COUNTY	1980 POPULATION	1990 POPULATION	DISTANCE AND DIRECTION FROM THE SITE
DeWitt	DeWitt	232	122	2.5 miles ENE
Weldon	DeWitt	531	361	5.3 miles ESE
Clinton	DeWitt	8,014	7,437	6.3 miles W
Wapella	DeWitt	768	608	7.4 miles WNW
Deland	Piatt	509	458	10.1 miles ESE
Maroa	Macon	1,760	1,602	10.7 miles SW
Farmer City	DeWitt	2,252	2,114	11.2 miles ENE
Cisco	Piatt	333	282	11.7 miles SSE
Heyworth	McLean	1,598	1,627	12.3 miles NW
Argenta	Macon	994	940	12.4 miles S
LeRoy	McLean	2,870	2,777	13.1 miles NNE
Kenney	DeWitt	443	390	13.6 miles WSW
Oreana	Macon	999	847	15.6 miles S
Downs	McLean	561	620	15.7 miles N
Waynesville	DeWitt	569	440	15.7 miles WNW
Monticello	Piatt	4,753	4,549	16.6 miles SE
Mansfield	Piatt	921	929	17.0 miles E
Forsyth	Macon	1,072	1,275	17.0 miles SSW
Cerro Gordo	Piatt	1,553	1,436	19.6 miles SSE
Warrensburg	Macon	1,372	1,274	19.8 miles SW
McLean	McLean	836	797	19.9 miles WNW
Bellflower	McLean	421	405	19.9 miles NE
Ellsworth	McLean	244	224	20.2 miles NNE
Atlanta	Logan	1,807	1,616	21.3 miles WNW
Bement	Piatt	1,770	1,668	21.4 miles SE
Latham	Logan	564	482	21.5 miles SW
Arrowsmith	McLean	292	313	21.9 miles NNE
Mahomet	Champaign	1,986	3,103	22.1 miles E
Decatur	Macon	94,081	83,885	22.4 miles SSW
Bloomington	McLean	44,189	51,972	22.7 miles NNW
Saybrook	McLean	882	767	23.8 miles NE
Ivesdale	Champaign	339	339	24.5 miles SE
Normal	McLean	35,672	40,023	24.6 miles NNW
Foosland	Champaign	153	132	24.7 miles ENE
Mount Pulaski	Logan	1,783	1,610	25.3 miles WSW
Cooksville	McLean	259	211	26.3 miles NNE
Mount Zion	Macon	4,563	4,522	26.5 miles S
Fisher	Champaign	1,572	1,526	26.8 miles ENE
Stanford	McLean	720	620	26.8 miles NW
Armington	Tazewell	292	348	27.0 miles WNW
Lincoln	Logan	16,327	15,418	27.1 miles W
Niantic	Macon	761	647	27.2 miles SW
Towanda	McLean	630	856	27.2 miles N
Hammond	Piatt	556	527	28.1 miles SSE

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TABLE 2.1-2 (Cont'd)
CITIES, TOWNS AND VILLAGES WITHIN 50 MILES OF CLINTON POWER STATION

CITY OR TOWN	COUNTY	1980 POPULATION	1990 POPULATION	DISTANCE AND DIRECTION FROM THE SITE
Sadorus	Champaign	435	469	28.6 miles ESE
Colfax	McLean	920	854	29.4 miles NNE
Illiopolis	Sangamon	1,118	934	29.8 miles SW
Champaign	Champaign	58,133	63,502	29.9 miles E
Danvers	McLean	921	981	30.2 miles NW
Minier	Tazewell	1,261	1,155	30.3 miles NW
Savoy	Champaign	2,126	2,674	30.7 miles ESE
Dalton City	Mountrie	574	573	30.8 miles S
Hudson	McLean	929	1,006	30.9 miles NNW
Gibson City	Ford	3,498	3,396	31.0 miles NE
Anchor	McLean	192	178	31.2 miles NNE
Atwood	Douglas/Piatt	1,464	1,253	31.3 miles SE
Hartsburg	Logan	379	306	31.5 miles W
Tolono	Champaign	2,434	2,605	31.7 miles ESE
Broadwell	Logan	183	146	31.7 miles WSW
Carlock	McLean	410	418	32.0 miles NNW
Urbana	Champaign	35,978	36,344	32.2 miles E
Macon	Macon	1,300	1,282	32.2 miles SSW
Lovington	Moultrie	1,313	1,143	32.4 miles SSE
Lexington	McLean	1,806	1,809	32.4 miles N
Garrett	Douglas	205	169	32.7 miles SE
Pesotum	Champaign	651	558	33.5 miles ESE
Thomasboro	Champaign	1,242	1,250	33.5 miles E
Hopedale	Tazewell	913	805	34.2 miles WNW
Emden	Logan	527	459	34.3 miles WNW
Elkhart	Logan	493	475	34.5 miles WSW
Mount Auburn	Christian	598	544	34.8 miles SW
Blue Mound	Macon	1,338	1,161	35.0 miles SSW
Elliott	Ford	370	309	35.2 miles NE
Kappa	Woodford	170	134	35.8 miles NNW
Arthur	Douglas/Moultrie	2,122	2,112	35.9 miles SSE
Bethany	Moultrie	1,550	1,369	36.0 miles S
Congerville	Woodford	373	397	36.1 miles NNW
Buffalo	Sangamon	514	503	36.3 miles SW
Philo	Champaign	973	1,028	36.3 miles ESE
Mackinaw	Tazewell	1,354	1,331	36.5 miles NW
Rantoul	Champaign	20,161	17,212	36.6 miles ENE
Sibley	Ford	370	359	36.9 miles NE
Tuscola	Douglas	3,839	4,155	37.6 miles SE
Mechanicsburg	Sangamon	515	538	37.7 miles SW
Moweaqua	Shelby	1,922	1,785	38.0 miles SSW
New Holland	Logan	295	330	38.1 miles W
Dawson	Sangamon	532	536	38.2 miles WSW
Delavan	Tazewell	1,973	1,642	38.8 miles WNW
Goodfield	Woodford	500	454	38.8 miles NNW
Middletown	Logan	503	436	38.8 miles W
Williamsville	Sangamon	996	1,140	39.2 miles WSW

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TABLE 2.1-2 (Cont'd)
CITIES, TOWNS AND VILLAGES WITHIN 50 MILES OF CLINTON POWER STATION

CITY OR TOWN	COUNTY	1980 POPULATION	1990 POPULATION	DISTANCE AND DIRECTION FROM THE SITE
Gridley	McLean	1,246	1,304	39.2 miles N
Ludlow	Champaign	397	323	39.3 miles ENE
Chenoa	McLean	1,847	1,732	39.8 miles N
El Paso	Woodford	2,676	2,499	40.1 miles NNW
Villa Grove	Douglas	2,707	2,734	40.2 miles ESE
Sullivan	Moultrie	4,526	4,354	40.2 miles SSE
Stonington	Christian	1,184	1,006	40.3 miles SSW
Sidney	Champaign	886	1,027	40.3 miles ESE
Deer Creek	Tazewell	688	630	40.3 miles NW
San Jose	Logan/Mason	784	519	40.4 miles WNW
Melvin	Ford	519	466	40.6 miles NE
St. Joseph	Champaign	1,900	2,052	40.7 miles E
Spaulding	Sangamon	428	440	41.4 miles WSW
Tremont	Tazewell	2,096	2,088	41.4 miles NW
Riverton	Sangamon	2,783	2,638	42.0 miles WSW
Secor	Christian	488	389	42.1 miles NNW
Camargo	Douglas	428	372	42.3 miles SE
Arcola	Douglas	2,714	2,678	42.4 miles SE
Paxton	Ford	4,258	4,289	42.7 miles ENE
Fairbury	Livingston	3,544	3,643	42.7 miles NNE
Gifford	Champaign	848	845	42.7 miles ENE
Strawn	Livingston	143	132	42.7 miles NNE
Panola	Coles	31	43	43.0 miles NNW
Sherman	Sangamon	1,501	2,080	43.5 miles WSW
Eureka	Woodford	4,306	4,435	43.6 miles NNW
Morton	Tazewell	14,178	13,799	43.7 miles NW
Longview	Champaign	207	180	43.8 miles ESE
Mason City	Mason	2,719	2,323	44.0 miles W
Findlay	Shelby	868	787	44.1 miles S
Royal	Champaign	274	217	44.1 miles E
Allenville	Moultrie	204	166	44.2 miles SSE
Green Valley	Tazewell	768	745	44.3 miles WNW
Clear Lake	Sangamon	236	193	44.4 miles WSW
Edinburg	Christian	1,231	982	44.6 miles SW
Roberts	Ford	422	397	45.1 miles NE
Forrest	Livingston	1,246	1,124	45.2 miles NNE
Ogden	Champaign	818	671	45.2 miles E
Assumption	Christian	1,283	1,244	45.3 miles SSW
Loda	Iroquois	486	390	45.3 miles ENE
Cantrall	Sangamon	141	123	45.7 miles WSW
Homer	Champaign	1,279	1,264	44.8 miles ESE
Rochester	Sangamon	2,488	2,676	45.9 miles SW
Greenview	Menard	830	848	46.3 miles W
Broadlands	Champaign	346	340	46.4 miles ESE
Humboldt	Coles	499	470	46.4 miles SE
Grandview	Sangamon	1,794	1,647	46.4 miles WSW

TABLE 2.1-2 (Cont'd)

CITIES, TOWNS AND VILLAGES WITHIN 50 MILES OF CLINTON POWER STATION

CITY OR TOWN	COUNTY	1980 POPULATION	1990 POPULATION	DISTANCE AND DIRECTION FROM THE SITE
Roanoke	Woodford	2,001	1,910	46.7 miles NNW
Washington	Tazewell	10,364	10,099	46.8 miles NW
South Pekin	Tazewell	1,243	1,184	47.2 miles WNW
Athens	Menard	1,371	1,404	47.4 miles WSW
Taylorville	Christian	11,386	11,133	48.1 miles SSW
Flanagan	Livingston	978	987	48.3 miles N
Hindsboro	Douglas	407	346	48.6 miles SE
Springfield	Sangamon	99,637	105,227	48.6 miles WSW
Benson	Woodford	460	410	48.7 miles NNW
Chatsworth	Livingston	1,187	1,186	48.7 miles NE
Pekin	Tazewell	33,967	32,254	49.0 miles NW
Kincaid	Christian	1,591	1,353	49.1 miles SW
Allerton	Vermillion	303	274	49.1 miles ESE
Fithian	Vermillion	540	512	49.1 miles E
Bulpitt	Christian	301	206	49.3 miles SW
Pontiac	Livingston	11,227	11,428	49.5 miles NNE
East Peoria	Tazewell	22,385	21,378	49.5 miles NW
Jeiseyville	Christian	178	126	49.5 miles SW
Marquette Heights	Tazewell	3,386	3,077	49.8 miles NW
Owaneco	Christian	285	260	49.9 miles SSW

Source: 1980 - U.S. Department of Commerce, 1981.
1990 - Illinois Counties and Incorporated Municipalities.

Appendix B
EXTERNAL EVENT ASSESSMENT

B.1 INTRODUCTION

This appendix discusses the external events assessment in support of the Clinton ILRT interval extension risk assessment.

External hazards were evaluated in the Clinton Individual Plant Examination of External Events (IPEEE) Submittal in response to the NRC IPEEE Program (Generic Letter 88-20 Supplement 4). The IPEEE Program was a one-time review of external hazard risk to identify potential plant vulnerabilities and to understand severe accident risks. Clinton does not currently maintain external event PSA models and associated documentation. Although the external event hazards in the Clinton IPEEE were evaluated to varying levels of conservatism, the results of the Clinton IPEEE are nonetheless used in this risk assessment to provide a conservative comparison of the impact of external hazards on the conclusions of this ILRT interval extension risk assessment.

B.2 CLINTON IPEEE INTERNAL FIRES ANALYSIS

The Clinton plant risk due to internal fires was evaluated in 1995 as part of the CPS Individual Plant Examination of External Events (IPEEE) Submittal. The EPRI FIVE Methodology and Fire PRA Implementation Guide screening approaches and data were used to perform the CPS IPEEE fire PRA study. The CDF contribution due to internal fires was calculated at 3.26E-6/yr.

The IPEEE documentation for the fire induced core damage scenarios and the associated frequency results were reviewed in support of this assessment. Based on review of the critical fire areas, the approximate breakdown of the CPS fire risk profile is as follows:

• Fire-induced loss of inventory control scenarios	~70%
• Fire-induced loss of decay heat removal scenarios	~30%
• Fire-induced loss of reactivity control (ATWS) scenarios	ε [1]

This information is used in Section B.5 of this appendix to provide insight into the impact of external hazard risk on the conclusions of this ILRT risk assessment.

B.3 CLINTON IPEEE SEISMIC ANALYSIS

The Clinton seismic risk analysis was performed as part of the Individual Plant Examination of External Events (IPEEE). Clinton performed a seismic margins assessment (SMA) following the guidance of NUREG-1407 and EPRI NP-6041. The SMA is a deterministic evaluation process that does not calculate risk on a probabilistic basis. No core damage frequency sequences were quantified as part of the IPEEE seismic risk evaluation.

The conclusions of the Clinton IPEEE seismic risk analysis are as follows:

> *"No improvements to the plant were identified as a result of the Seismic Margins Assessment ... the plant was determined to be fully capable of attaining safe shutdown conditions after the Review Level Earthquake (RLE)."*

Although quantitative risk information is not directly available from the Clinton SMA IPEEE analysis, Reference [A-1] provides a simple method (called the Simplified Hybrid Method) for obtaining a seismic-induced CDF estimate based on results of an SMA analysis. Reference [A-1] has shown that only the plant HCLPF (High Confidence Low Probability of Failure) seismic capacity is needed in order to estimate the seismic CDF within a precision of approximately a factor of two. The approach is as follows:

---

[1] ε = negligible

Step 1: Determine the plant HCLPF seismic capacity C_{HCLPF} from the SMA analysis

Step 2: Estimate the 10% conditional probability of failure capacity $C_{10\%}$ from:

$$C_{10\%} = F_{\beta} C_{HCLPF}$$
$$F_{\beta} = e^{1.044\beta}$$

where 1.044 is the difference between the 10% NEP standard normal variable (-1.282) and the 1% NEP standardized normal variable (-2.326).

Experience gained from high quality seismic PRA studies indicates that the plant damage state fragility determined by rigorous convolution will tend to have β_c values in the range of 0.30 to 0.35 (the plant damage state β_c value is equal to or less than the β_c values for the fragilities of the individual components that dominate the seismic risk). As such, the Simplified Hybrid method recommends:

$$C_{10\%} = 1.4 C_{HCLPF}$$

Step 3: Determine hazard exceedance frequency $H_{10\%}$ that corresponds to $C_{10\%}$ from hazard curve.

Step 4: Determine seismic risk PF from:

$$PF = 0.5\ H_{10\%}$$

Using the Simplified Hybrid Method, an approximation of the Clinton seismic-induced CDF is performed here.

Step 1: If the SMA analysis screens out every component on the Seismic Safe Shutdown Paths at the Review Level Earthquake (RLE), the plant HCLPF is equal to the RLE. Such is the case with the Clinton IPEEE SMA analysis. As the Clinton RLE is 0.30g PGA (Peak Ground Acceleration), the Clinton plant HCLPF is 0.30g PGA.

Step 2: Using the relationship recommended above, the plant 10% capacity point (C10%) is estimated as 1.4 x 0.3g PGA = 0.42g PGA.

Step 3: The seismic hazard curve for the Clinton site, based upon EPRI NP-6395-D, is summarized in tabular form in Table A-1. As can be seen from Table A-1, the seismic hazard frequency associated with the 10% capacity point (0.42g PGA) is approximately 5E-6/yr.

Step 4: Using the relationship recommended above, the seismic-induced CDF is approximated as 0.50 x 5E-6/yr = 2.5E-6/yr.

The Simplified Hybrid Method only provides an overall seismic-induced CDF estimate and does not provide information as to the breakdown of seismic accident sequence types. A more rigorous analysis (e.g., a seismic PRA, or the Rigorous Hybrid Method referred to in Reference [A-1]) is required for such information. Such an analysis was not performed as part of this ILRT risk assessment. However, a Rigorous Hybrid Method calculation was recently completed for another Exelon BWR plant (Limerick). [A-2] The results of that study (Case #2 of Reference [A-2]) are used here to provide a reasonable approximation of the breakdown of seismic accident sequence types, they are as follows:

- Seismic-induced loss of decay heat removal scenarios ~35%
- Wide-spread failure of seismic safe shutdown SSCs ~20%
- Seismic-induced ATWS scenarios ~15%
- Other seismic-induced accidents (e.g., SBO, loss of coolant makeup, etc.) ~30%

This information is used in Section B.5 of this appendix to provide quantitative insights into the impact of external hazard risk on the conclusions of this ILRT risk assessment.

Table B-1

CLINTON SITE SEISMIC HAZARD CURVE
- EPRI NP-6395-D(1)

Peak Ground Acceleration		EPRI Exceedance Frequency (1/yr, mean)
cm/s^2	g	
8	0.01	2.7E-2
80	0.08	6.9E-4
160	0.16	1.8E-4
305	0.31	3.1E-5
420	0.43	4.8E-6
570	0.58	8.0E-7
800	0.82	2.0E-7

(1)From Table 3-19 and Figure 3-55 of EPRI NP-6395-D, Appendix E.

B.4 OTHER EXTERNAL HAZARDS

In addition to internal fires and seismic events, the CPS IPEEE Submittal analyzed a variety of other external hazards:

- High Winds/Tornadoes
- External Flooding
- Transportation and Nearby Facility Accidents
- Other External Hazards

The CPS IPEEE analysis of high winds, tornadoes, external floods, transportation accidents, nearby facility accidents, and other external hazards was accomplished by reviewing the plant environs against regulatory requirements regarding these hazards. Based upon this review, it was concluded that CPS meets the applicable Standard Review Plan requirements and therefore has an acceptably low risk with respect to these hazards. As such, these hazards were determined in the Clinton IPEEE to be negligible contributors to overall plant risk.

Accordingly, these other external event hazards are not included explicitly in this appendix and are reasonably assumed not to impact the results or conclusions of the ILRT interval extension risk assessment.

B.5 IMPACT OF EXTERNAL HAZARD RISK ON ILRT RISK ASSESSMENT

The NEI Interim Guidance calculation of delta LERF performed in Section 3 of this report is re-performed here including, in addition to internal event information, the Clinton IPEEE external event risk information discussed in the previous sections.

Per the NEI Interim Guidance, the impact on the LERF risk measure due to the proposed ILRT interval extension is calculated as follows:

delta LERF = (Frequency of EPRI Category 3b for 1-per-15 year ILRT interval) – (Frequency of EPRI Category 3b for 1-per-10 year ILRT interval)

As discussed in Section 3.1, the frequency per year for EPRI Category 3b is calculated as:

Frequency 3b = [3b conditional failure probability] x [CDF - (CDF with independent LERF + CDF that cannot cause LERF)]

Based on the previous discussion in Sections B.2 through B.4, the Clinton external event initiated CDF is approximately 3.26E-6/yr (internal fires) + 2.50E-6/yr (seismic) = 5.76E-6/yr. In addition, the following external event accident scenarios are excluded from the 3b frequency calculation because they cannot result in a LERF release or independently result in LERF:

- Fire-induced loss of decay heat removal scenarios (9.80E-7/yr)
  0.30 x 3.26E-6/yr = 9.80E-7/yr
- Seismic-induced loss of decay heat removal scenarios (8.75E-7/yr)
  0.35 x 2.50E-6/yr = 8.75E-7/yr
- Wide-spread failure of seismic safe shutdown SSCs (5.00E-7/yr)
  0.20 x 2.50E-6/yr = 5.00E-7/yr

Therefore, the baseline frequency of category 3b due to external events is calculated as (2.70E-03) x [(5.76E-6/yr) – (9.80E-7/yr + 8.75E-7/yr + 5.00E-7/yr)] = 9.19E-9/yr.

Using the relationship described in Section 3.4.1 for the impact on 3b frequency due to increases in the ILRT surveillance interval, the EPRI Category 3b frequency for the 1-per-10 year and 1-per-15 year ILRT intervals are calculated as 3.06E-8/yr and 4.60E-8/yr, respectively. Therefore, the change in the LERF risk measure due to extending the ILRT from 1-per-10 years to 1-per-15 years, including both internal and external hazard risk, is estimated as:

	3b Frequency (1-per-10 year ILRT)	3b Frequency (1-per-15 year ILRT)	LERF Increase
External Events Contribution	3.06E-8/yr	4.60E-8/yr	1.54E-8/yr
Internal Events Contribution	1.87E-7/yr	2.80E-7/yr	9.30E-8/yr
Combined (Internal + External)	2.18E-7/yr	3.26E-7/yr	1.08E-7/yr

Comparison to RG 1.174 Acceptance Guidelines

NRC Regulatory Guide 1.174, "An Approach for Using PRA in Risk-Informed Decisions on Plant-Specific Changes to the Licensing Basis", provides NRC recommendations for using risk information in support of applications requesting changes to the license basis of the plant. As discussed in Section 2 of this report, the risk acceptance criteria of RG 1.174 is used here to assess the ILRT interval extension.

The 1.08E-7/yr increase in LERF from extending the Clinton ILRT frequency from 1-per-10 years to 1-per-15 years falls into Region II ("Small Change" in risk) of the RG 1.174 acceptance guidelines. Per RG 1.174, when the calculated increase in LERF due to the proposed plant change is in the range of 1E-7 to 1E-6 per reactor year, the risk assessment must also reasonably show that the total LERF is less than 1E-5/yr.

Per the Clinton internal events PSA (Rev. 3) documentation, the Clinton LERF due to internal event accidents is 2.63E-7/yr. Explicit information on LERF due to external events is not available from the Clinton IPEEE. However, assuming a conservative relationship that approximately 10% of CDF represents LERF (note that the Clinton internal events LERF vs. CDF relationship is approximately 1%), the Clinton LERF due to external events can be approximated by 0.10 x 5.76E-6/yr = 5.76E-7/yr. Therefore, the total LERF for Clinton is estimated at 2.63E-7/yr + 5.76E-7/yr = 8.39E-7/yr, which is significantly less than the RG 1.174 acceptance guideline of 1E-5/yr.

REFERENCES

[B-1] Kennedy, R.P., "Overview of Methods for Seismic PRA and Margin Analysis Including Recent Innovations", Proceedings of the OECD-NEA Workshop on Seismic Risk, Tokyo, Japan, August 1999. Available from OECD Nuclear Energy Agency, La Seine St.-Germain, 12 Boulevard des Iles, F-92130 Issy-les-Moulineaus, France

[B-2] ERIN Engineering and Structural Mechanics Consulting, Summary of Limerick Generating Station Seismic Margins Insights Evaluation, ERIN Report No. C0467010033-4801, June 2002.

[B-3] Electric Power Research Institute, Probabilistic Seismic Hazard Evaluations at Nuclear Plant Sites in the Central and Eastern United States: Resolution of the Charleston Earthquake Issue, NP-6395-D, April 1989.

Appendix C

RISK ASSESSMENT OF EXTENDING CLINTON ILRT AND DWBT INTERVAL

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Appendix C

RISK ASSESSMENT OF EXTENDING CLINTON ILRT AND DWBT INTERVAL

C.1 PURPOSE

The purpose of this analysis is to provide an assessment of the combined risk increase associated with implementing a one-time extension of the Clinton Power Station (CPS) Drywell Bypass Test (DWBT) and Integrated Leak Rate Test (ILRT) interval from ten years to fifteen years[(1)]. The extension would allow for substantial cost savings as the DWBT could be deferred for additional scheduled refueling outages to keep it on the same schedule as the Type A Containment Integrated Leak Rate Test (ILRT). The risk assessment follows the guidelines from NEI 94-01 [C-1], the methodology used in EPRI TR-104285 [C-2], the NEI Interim Guidance for Performing Risk Impact Assessments In Support of One-Time Extensions for Containment Integrated Leakage Rate Test Surveillance Intervals [C-3], and the NRC regulatory guidance on the use of Probabilistic Risk Assessment (PRA) findings and risk insights in support of a request for a change in a plant's licensing basis as outlined in Regulatory Guide 1.174 [C-4].

C.1.1 Background

Revisions to 10CFR50, Appendix J (Option B) allow individual plants to extend the Integrated Leak Rate Test (ILRT) Type A surveillance testing requirements from three-in-ten years to at least once per ten years. The revised Type A frequency is based on an acceptable performance history defined as two consecutive periodic Type A tests at least 24 months apart in which the calculated performance leakage is less than normal containment leakage of 1.0 L_a (allowable leakage).

[(1)] Note that the extension of the ILRT interval alone is evaluated in the main report.

In addition to the ILRT interval extensions from 3 to 10 years, the NRC has also approved the same time interval extensions for the BWR Mark III plants for the Drywell Bypass Test (DWBT).

The DWBT is to verify that pre-existing drywell bypass leakage does not exceed the minimum requirements. The DWBT thus affects the likelihood of a suppression pool bypass in the Level 1 and 2 PSA analysis.

The industry and the NRC have agreed upon an approach to the risk assessment for a similar exemption request, i.e., the one-time ILRT interval extension from 10 years to 15 years. This methodology is applied to the ILRT in the main report. However, no such precedent has been established for the DWBT interval extension.

The CPS ILRT interval extension risk assessment workscope is discussed in the main report. It employs the NEI Guidance methodology for ILRT interval extension. [C-1] The NEI ILRT methodology has been accepted by the NRC through their exemption approval process on plant specific cases. The same approach is used in this appendix to address the risk changes associated with the DWBT interval extension.

C.1.2 Criteria

Based on previously approved ILRT extension requests, Clinton uses the following risk metrics to characterize the change in risk associated with the one time ILRT extension:

- Change in Large Early Release Frequency (LERF)
- Change in conditional containment failure probability
- Change in population dose rate (person-rem/yr)

Consistent with the NEI Interim Guidance on ILRT extensions, the acceptance guidelines in Regulatory Guide 1.174 [C-4] are used to assess the acceptability of this one-time extension of the DWBT interval beyond that established by exemption for the Mark III plants during the Option B rulemaking of Appendix J. The CPS DWBT interval extension from 3 to 10 years was requested by CPS [C-23] and approved by NRC [C-24].

Based on the precedent of other ILRT extension requests [C-6, C-20, C-22], the total annual risk (person-rem/yr population dose rate) and the conditional containment failure probability are examined to demonstrate the relative change in risk. (No threshold has been established for these parameter changes.) The LERF criteria used here are those from Reg. Guide 1.174 and are the same as discussed in the main report.

C. 2 METHODOLOGY

Section 2 of the main report provides available references related to the ILRT risk assessment methodology. This section provides the following methodology related items for the DWBT:

- The basic approach for applying the ILRT extension methodology to the DWBT interval extension
- The steps to be used in the analysis (analogous to those in the NEI guidance for the ILRT interval extension requests)
- The assumptions used in the evaluation
- The inputs required
    - Generic ex-plant consequence
    - Plant specific inputs

The following subsections address these items.

C.2.1 Basic Approach

The DWBT extension methodology makes maximum use of the EPRI ILRT extension methodology as described in the main report. The same failure frequencies, release categories, consequence calculations, and acceptance criteria are used. The impact of drywell leakage is to allow drywell atmosphere, including fission products, to be passed at some rate directly to the containment, without benefit of quenching and fission product retention in the suppression pool. The key augmentation needed to the ILRT method is to determine the impact of this suppression pool bypass on the containment and on the radionuclide releases.

It is assumed in this augmented methodology that the special leakage categories established by EPRI for use in ILRT risk assessments can also be applied to the drywell for the DWBT risk assessment. The Mark III containment has a different arrangement from either PWR containments or BWR Mark I/II containments. The difference is that the drywell which includes the RPV is completely enclosed by the outer containment. As such, the drywell leakage (the subject of this analysis) does not leak directly to the environment but is further mitigated by the outer containment leakage barrier. Because of this "dual" containment, there are several possible leakage path combinations that must be considered. The drywell can be intact (base leakage assumed), it can have a small pre-existing failure (10 times base leakage), or it can have a large pre-existing failure (35 times base leakage). The probability of each of these drywell failure categories is taken from the industry experience evaluation of containment failures and is the same as that established in the main report of the corresponding category for containment. As further discussed below, this leads to nine combinations of drywell and containment leakage sizes. Each combination will have an impact on radionuclide releases that corresponds approximately to one of the original containment failure categories.

This assignment of each of these combinations to an original containment failure category depends on two things. The first is whether the drywell leakage results in significantly higher pressures in containment than would occur with no drywell leakage. The second is the increased concentration of fission products in containment due to the drywell leakage. To determine these compound effects, a Clinton specific MAAP 4.0 model is used. By examining CsI concentrations in containment and the amount of CsI released, one can assign each drywell-failure-containment-failure combination to one of the original containment failure bins. Once this is done, the calculation process is identical to that used for the ILRT interval extension.

The following are believed to be the most important aspects associated with the DWBT:

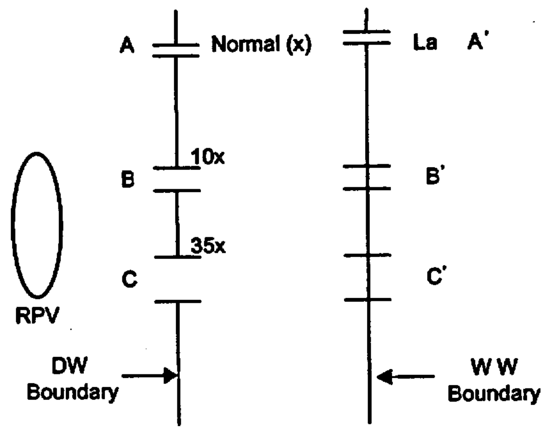
- The potential leak paths from the drywell into the wetwell airspace
- The Clinton historical evidence of low drywell leakage and the Clinton methods to identify adverse trends
- The ability to withstand bypass leakage under postulated accident conditions (this includes passive capability and active systems)

C.2.2. Process Steps/Approach

It is noted that there is no approved methodology developed solely for the risk assessment of the DWBT interval extension. The process steps for the DWBT interval extension risk evaluation are patterned after the EPRI/NEI methodologies used for the ILRT interval extension requests and used here for Clinton in the main report. The process steps are the following:

1. Determine whether there are increases in the potential for core damage arising from the change in DWBT interval. This relates principally to the potential for increased containment rupture frequency that could then fail RPV injection. The increase in containment rupture frequency may occur due to suppression pool bypass. Deterministic calculations using MAAP form the basis for characterizing the potential for such induced severe accidents. Adjust the core damage frequency, if necessary.

2. Determine the containment leakage rates for EPRI categories 1, 3a and 3b. The Mark III containment has a different arrangement from either PWR containments or BWR Mark I/II containments. The difference is that the drywell which includes the RPV is completely enclosed by the outer containment. As such, the drywell leakage (the subject of this analysis) does not leak directly to the environment but is further mitigated by the outer containment leakage path. Because of this "dual" containment, there are several possible leakage path combinations that must be considered. These combinations are identified below. Note that Categories 3a and 3b will be made up of multiple combinations of DW and WW leakage pathways:



where x = Assumed Baseline Leakage Rate

The failure combinations are provided below for the assumed baseline condition of ILRT and DWBT every 3.3 years.

Failures DW	Failures WW	Probability	Consequence Characterization
A	A'	--	Already included in the ILRT Assessment (Class I)
A	B'	--	Already included in the ILRT Assessment (Class 3a)
A	C'	--	Already included in the ILRT Assessment (Class 3b)
B	A'	2.7E-2	Leak Failure Mode with small DW Bypass
B	B'	$(2.7E-2)^2$	Small pre-existing failure mode with small DW Bypass
B	C'	2.7E-2 * 2.7E-3	Large pre-existing failure mode with small DW Bypass
C	A'	2.7E-3	Leakage Failure Mode with large DW Bypass
C	B'	2.7E-3 * 2.7E-2	Small pre-existing failure mode with large DW Bypass
C	C'	$(2.7E-3)^2$	Large pre-existing failure mode with large DW Bypass

Each of these failure combinations can be correlated to the impacts on the containment failure categories 1, 3a, and 3b, by using MAAP calculations.

3. Quantify the baseline (nominal three year ILRT <u>and</u> DWBT interval) frequency for the EPRI accident classes of interest. Use the conditional probability of failure of the drywell as determined from operating experience data per the EPRI methodology.[1]

[1] The one-time interval extension is requested to be consistent with the change in the ILRT interval extension from 10 years to 15 years. Therefore, in the incremental assessment of the risk change due to the DWBT interval extension, the comparisons are made for both test intervals to be extended together.

4. Develop the baseline population dose (person-rem) for the applicable EPRI categories from Grand Gulf (NUREG/CR-4551) translation to Clinton.

5. Determine the population dose rate (person-rem/year) by multiplying the dose calculated in Step (4) by the associated frequency calculated in Step (3).

6. Determine the change in probability of leakage detectable only by DWBT, and associated frequency for the new surveillance intervals of interest. Note that the DWBT interval extension will be made simultaneously with the ILRT.

7. Determine the population dose rate for the new surveillance intervals of interest.

8. Evaluate the risk impact (in terms of population dose rate and percentile change in population dose rate) for the interval extension cases.

9. Evaluate the risk impact in terms of LERF.

10. Evaluate the change in conditional containment failure probability.

C.2.3 Ground Rules

The following ground rules are used in the analysis:

- The Clinton Level 1 and Level 2 internal events PRA model provides representative results for the analysis.
- It is appropriate to use the Clinton internal events PRA model as a gauge to effectively describe the risk change attributable to the ILRT extension. It is reasonable to assume that the impact from the ILRT extension (with respect to percent increases in population dose rate) will not substantially differ if fire and seismic events were to be included in the calculations.
- An evaluation of the risk impact of the DWBT on shutdown risk is addressed using the generic results from EPRI TR-105189 [C-8] as augmented by NEI Interim Guidance. [C-3]
- Radionuclide release categories are defined consistent with the EPRI TR-104285 methodology. [C-2]

The integral change in risk measures when both the ILRT <u>and</u> DWBT intervals are extended from the currently approved ten (10) years to the one time interval extension of fifteen (15) years provides this risk perspective.

- The ex-plant consequence in terms of population dose results for the containment failures modeled in the PSA can be characterized by information provided in NUREG/CR-4551 [C-9]. They are estimated by scaling the NUREG/CR-4551 population dose results by power level, population, and leak rate differences for Clinton compared to the NUREG/CR-4551 Mark III reference plant, Grand Gulf.
- Per the NEI Interim Guidance [C-3], the representative drywell leakage for EPRI Category 1 sequences is 1 DWL_C (DWL_C is the conservative characterization of observed DW leakage rate).
- Per the NEI Interim Guidance [C-3], the representative drywell leakage for EPRI Category 3a sequences is 10 DWL_C.
- Per the NEI Interim Guidance [C-3], the representative drywell leakage for EPRI Category 3b sequences is 35 DWL_C.
- EPRI Category 3b is conservatively categorized as LERF based on the previously approved methodology [C-3].
- The impact on population doses from Interfacing System LOCAs is not altered by the proposed ILRT extension, but is accounted for in the EPRI methodology as a separate entry for comparison purposes. Since the ISLOCA contribution to population dose is fixed, no changes on the conclusions regarding increases in population dose from this analysis will result from this assumption.
- The containment isolation valve test frequency is not altered. Therefore, the reduction in DWBT frequency does not impact the reliability of containment isolation valves to close in response to a containment isolation signal.

C.2.4 PLANT SPECIFIC DESIGN, TESTING, AND INPUTS AFFECTING THE DWBT

C.2.4.1 Design Overview

As described in Updated Safety Analysis Report (USAR) Section 6.2.1, the Mark III containment design at CPS incorporates the drywell/pressure-suppression feature of previous BWR containment designs (Mark I and II) into a dry-containment type structure. The Mark III containment has three main features: (1) a drywell surrounding the reactor pressure vessel and a large part of the reactor coolant pressure boundary, (2) a suppression pool that serves as a heat sink during normal operational transients and accident conditions, and (3) a primary containment structure to prevent the uncontrolled release of radioactivity to the environment.

The drywell is a Class 1 seismic structure and features reinforced concrete walls and floor in a vertical right cylinder geometry. The ceiling is also reinforced concrete with a removable steel dome known as the drywell head. The floor is common with the primary containment basemat. The drywell encloses the reactor pressure vessel (RPV), the reactor coolant recirculation loops, and branch connections of the reactor coolant system (RCS). The function of the drywell is to maintain a pressure boundary that forces steam from a loss of coolant accident (LOCA) through the 102 horizontal vents in the drywell wall into the suppression pool. The steam is condensed in the suppression pool, and the air forced from the drywell is released into the primary containment (i.e., wetwell). The pressure-suppression capability of the suppression pool assures that the peak LOCA temperature and pressure in the primary containment are kept below the design limits of 185 °F and 15 psig, respectively. The drywell also shields accessible areas of the primary containment from radiation originating in the reactor core and RCS.

Penetrations through the drywell enable the passage of piping, ventilation, and electrical cables. Electrical penetrations feature a sealing medium which surrounds the cables that pass through the penetration. Ventilation and piping penetrations feature manual, automatic, or check valves for isolation. Valves which prevent leakage from the drywell into the primary containment are considered drywell isolation valves since leakage through these valves contributes to the maximum allowable drywell leakage. Valves which prevent leakage from the drywell and primary containment to the secondary containment or environment are considered primary containment isolation valves. Leakage through these valves is determined in accordance with 10CFR50, Appendix J and contributes to the maximum allowable primary containment leakage rate. Leakage through primary containment isolation valves is not considered drywell leakage in the design basis analyses.

The drywell equipment hatch and drywell personnel air lock also penetrate the drywell boundary. The drywell equipment hatch is designed to be removed during plant outages and utilizes two compression seals to maintain leaktightness. The drywell air lock is designed to provide personnel access (ingress and egress) to the drywell for maintenance, while its safety function is to maintain drywell integrity. The drywell air lock features two doors. Each air lock door closes positively against the air lock structure by means of a latching mechanism. The drywell air lock door latching mechanisms are interlocked to each other to ensure that at least one door is maintained in the latched closed position, ensuring that the drywell air lock does not provide a gross leakage path and compromise drywell integrity. Each of the two drywell air lock doors utilizes two compression seals to minimize leakage. The drywell is not accessed during full power operation.

The structural integrity of the primary containment is largely dependent on the drywell's ability to perform its safety function. Steam from a LOCA that bypasses the suppression pool would compress the air in the wetwell (i.e., the space between the exterior wall of the drywell and the interior wall of primary containment) and could result in excessive primary containment pressures. As described in USAR Section 6.2.1.1.5, the effect of steam bypass of the suppression pool on primary containment integrity has been evaluated. The allowable drywell leakage was evaluated for a spectrum of reactor system rupture sizes (areas), with and without containment spray and heat sinks. The limiting case was determined to be a small reactor system break which would not result in reactor depressurization. Assuming the containment spray system and other heat sinks are available, the maximum allowable leakage path area ($A/\sqrt{k}$) was calculated to be 1.18 ft^2. [Drywell bypass leakage area is expressed in terms of the parameter $A/\sqrt{k}$, where A is the flow area of leakage (ft^2)and k is the geometric and friction loss coefficient.] An $A/\sqrt{k}$ of 1.18 ft^2 is equivalent to a bypass leakage rate of 136,400 scfm at a drywell design pressure of 30 psid and 43,120 scfm at 3 psid. For large break LOCA events, larger bypass leakage areas are allowable since the break would rapidly depressurize the reactor and terminate the blowdown. The maximum allowable leakage path area ($A/\sqrt{k}$)

for the large break LOCA case was calculated to be 10.15 ft^2, a factor of eight larger than for the small break LOCA case. CPS TS 3.6.5.1, "Drywell," requires the drywell bypass leakage rate to be maintained less than that corresponding to the design limit area. The value for the design limit has been reduced from a previously identified value of 1.18 ft^2 to a more restrictive value of 1.0 $ft.^2$[(1)] For this DWBT interval risk assessment, the value of 1.18 $ft.^2$ is conservatively used where the design limit area is cited.

Surveillance activities presently in effect to ensure the drywell safety function include drywell penetration configuration surveillances (i.e., valve line-ups); drywell structural integrity inspections; DWBTs; multiple drywell air lock tests, including overall air lock leakage rate testing, air lock door seal leak rate testing, and air lock door interlock mechanism functional verification; monitoring drywell temperature; monitoring drywell differential pressure relative to the primary containment; and monitoring suppression pool temperature and level.

C.2.4.2 Inputs

The inputs to the risk assessment include the following:

- Past Clinton DWBT results to demonstrate the adequacy of the administrative and hardware issues.
- Ex-plant consequence evaluation from NUREG-1150 for a Mark III plant
- Clinton specific adjustments to ex-plant consequence evaluation from NUREG-1150 (NUREG/CR 4551 Vol. 6 for Grand Gulf)
- Clinton specific inputs (Level 1 & 2)

[(1)] Letter from J.P. McElwain (CNO - Illinois Power co.) to US NRC, Document 50-461, U-603212 (8E.100a) dated June 24, 1999 [C-26]

C.2.4.2.1 DWBT Results

Clinton Power Station (CPS) Technical Specification (TS) Surveillance Requirement (SR) 3.6.5.1.1 currently requires the drywell bypass leakage test (DWBT) to be performed at least once per 10 years.

The results of DWBTs conducted since initial plant startup, including four periodic tests, have revealed an $A/\sqrt{k}$ that is two orders of magnitude less than the Technical Specification limit. Table C.2-1 provides these DWBT results. In addition, the on-line monitoring capability has continued to verify this very low bypass leakage. In other words, based on the testing performed over the last ten years, the performance of the drywell structure at CPS has been excellent. Based on this demonstrated performance, Exelon believes that a one-time reduction in the testing requirements for the drywell is warranted.

C. 2.4.2.2 <u>On-Line Qualitative Monitoring Capability</u>

Due to the demonstrated leaktight performance of the drywell, CPS is able to monitor the integrity of the drywell during normal plant operation. This is possible due to the normal operation of pneumatic controls and operators in the drywell that pressurize the drywell, plus the existence of small instrument air system leaks. These effects create a differential pressure between the drywell and primary containment that is monitored, and periodic operation action is required to vent the drywell.

For example, in 1994, the drywell was being pressurized at a rate of approximately 0.04 psi/hr. The drywell was being vented approximately once per day when pressure approached the upper TS limit of 1.0 psid. Based on application of the ideal gas law and known data, such as the drywell pressurization rate and the drywell leakage measured during the fourth refueling outage (RF-4), the total amount of instrument air in-leakage was calculated to be between 21.5 and 22.5 scfm. The rate of drywell pressurization remained essentially constant since drywell closeout from RF-4. Pressurization rates following RF-5 have also remained consistent with those observed following RF-4.

This steady drywell pressurization rate allows qualitative monitoring of the drywell leakage rate. An increase in this rate would be indication of an increase in the instrument air system leakage into the drywell since it is improbable that the drywell would become more leaktight. Conversely, a decrease in this rate would be evidence of a larger drywell leakage area. The maximum drywell leakage rate that would still maintain a differential pressure between the drywell and wetwell must be less than the instrument air in-leakage rate (which after RF-4 was 23 scfm). The $A/\sqrt{k}$ for a 23 scfm leak at 0.2 psid is 0.0025 ft^2 or 0.2 % of the allowable leakage area. Because of this large margin to the allowable drywell leakage rate, it has been concluded that as long as the drywell continues to pressurize, regardless of the rate, drywell integrity is always assured. This ability to qualitatively assess the integrity of the drywell during normal plant operation provides further support to extending the DWBT interval.

In order to provide added assurance that the drywell has not seriously degraded between the performance of DWBTs, a qualitative assessment of the drywell leak tightness is performed at least once per operating cycle. The first assessment was performed prior to Operating Cycle 7. By checking for gross leakage, this assessment will provide an indication of the ability of the drywell to perform its design function. As a check for gross leakage, the assessment may not identify drywell leakage that is masked by plant conditions, or identify leakage through systems that are not communicating with the drywell atmosphere at the time of the assessment. For example, minor increases in drywell bypass leakage could be masked by a small leak in the instrument air system inside the drywell. The assessment is not detailed enough to account for such minor changes. However, as demonstrated above, as long as the drywell continues to pressurize, regardless of the rate, drywell integrity is always assured.

Table C.2-1

PREVIOUS RESULTS OF CPS DRYWELL BYPASS LEAKAGE RATE TEST

Test Date	Leak Rate (at 3.0 psig)	Ratio to Design Limit	Calculated $A/\sqrt{k}$
01/86	273.0 scfm[1]	0.63%	0.0075 ft^2
11/86	20.8 scfm	0.05%	0.0006 ft^2
04/89 (RF-1)	18.8 scfm	0.04%	0.0005 ft^2
03/91 (RF-2)	21.9 scfm	0.05%	0.0006 ft^2
05/92 (RF-3)	18.0 scfm	0.04%	0.0005 ft^2
11/93 (RF-4)	30.2 scfm	0.07%	0.0008 ft^2

[1]The leakage rate from the initial test was primarily attributed to a defective electrical penetration seal that was later repaired. Subsequent tests have found the drywell leakage to consistently be between 18 and 30 scfm.

C. 2.4.2.3 Ex-Plant Consequences

The same process used for the ILRT interval extension in the main report is used here for the DWBT interval extension evaluation. This process makes use of surrogate Mark III consequences adjusted for Clinton specific power level, population, and leakage rates.

C.2.4.2.4 Clinton Scaling Factors

The Clinton specific information used to perform this DWBT interval extension risk assessment is the same as that for the ILRT interval extension risk assessment. The factors that are calculated for use in adjusting the population dose (person-rem) of the surrogate plant (NUREG-1150 Grand Gulf) for the site and plant differences are the same as in the main report and are as follows:

Consequence categories dependent on the "intact" Tech Spec Leakage

$$F_{CAT\ 1,\ 3a,\ 3b} = F_{POWER} * F_{POPULATION} * F_{TS\ LEAK}$$
$$F_{CAT\ 1,\ 3a,\ 3b} = 0.91 * 2.53 * 1.4$$
$$F_{CAT\ 1,\ 3a,\ 3b} = 3.22$$

Consequence categories not dependent on the Tech Spec Leakage:

$$F_C = F_{POWER} * F_{POPULATION}$$
$$F_C = 0.91 * 2.53$$
$$F_C = 2.30$$

Note that the same adjustment factor for the ex-plant consequence calculation for the "intact" containment category as used in the ILRT evaluation is to be used between the analyzed surrogate "plant" (Grand Gulf) and Clinton for the DWBT risk assessment. These factors represent the increase in the person rem consequence for the containment release bins for Clinton compared with the surrogate plant (Grand Gulf).

C.3 ANALYSIS

This section provides a step-by-step summary of the NEI guidance as applied to the CPS DWBT interval extension risk assessment. Each subsection addresses a step or group of steps in the NEI guideline for ILRT risk assessment, which is used here as a guide to the risk assessment for the comparable DWBT interval extension.

C.3.1 POTENTIAL FOR INCREASE IN CDF (STEP 1)

One of the areas investigated as part of the risk assessment of the DWBT interval extension is the potential that the core damage frequency could increase due to the increased bypass area. Probabilistic risk assessments of BWRs have postulated that a catastrophic failure of containment due to overpressure could result in failures of the RPV injection capability to maintain adequate core cooling. This may result from an adverse environment outside containment, disruption of injection pipe penetrations, injection valve misalignments, or the loss of effective control. Therefore, an assessment of the possibility that Clinton overpressure containment failures may increase in frequency due to the extension of the DWBT interval is performed by examining those sequences with the highest potential to cause such containment pressure increases. The USAR was reviewed to identify that the limiting condition was a 2" primary system LOCA in the drywell. Using this information and the identified allowable leak areas, several confirmatory MAAP cases were performed to demonstrate the containment challenges for varying bypass flow areas.

Table C.3-1 summarizes some of the key results of the deterministic calculations. These calculations provide insights with regard to the following:

- The peak containment pressure for different LOCAs and as a function of DW bypass leakage
- The radionuclide release as characterized by Noble Gas and CsI release fraction if a severe accident with core damage is assumed.

Attachment C-1 provides the deterministic plots for key parameters for these cases.

Table C.3-1 summarizes the results of these calculations. The results in the Figures C.3-1 through C.3-11 and Table C.3-1 indicate the following:

- The containment pressurization due to a LOCA is insensitive to relatively large variations in the DW Bypass area and does not exceed 20 psia except for the "worst case" postulated condition of a 2" LOCA and maximum Technical Specification Bypass. [1]
- The pressure suppression capability of the containment is robust.
- The large volume in the outer containment minimizes the effects of changes in the drywell bypass flow area.
- Any effects of the containment pressurization due to drywell bypass leakage can be effectively terminated by:

  a) RPV depressurization which is directed by the EOPs on exceeding the pressure suppression pressure

  <u>or</u>

  b) Containment sprays which are directed by the EOPs upon exceeding relatively low containment pressures

  Both of these operating crew actions can be completed over many hours and therefore their success probability is very high.
- Subsequent peaks of 30-40 psia in the containment pressure are due to hydrogen combustion events.

The conclusion from this investigation is that containment failure induced by containment pressurization aggravated by the drywell bypass leakage change is highly unlikely. The relatively small changes postulated due to the DWBT interval extension make no appreciable change in the containment pressurization compared to its ultimate capability.

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[1] Clinton Pressure Capability is as follows:

	Design	Ultimate
Containment	15 psig	94 psig
Drywell	30 psig	(Not Limiting)

The containment overpressure challenges due to the loss of containment heat removal capability are already accounted for in the Clinton PSA. As such, the perturbation on these sequences caused by slight changes in the drywell bypass area are considered negligible contributors to CDF.

There is no change in CDF due to the small increases in drywell bypass leakage associated with the DWBT interval extension.

The remaining steps in the risk assessment process examine the radionuclide release effects associated with combinations of pre-existing drywell and containment leakage. These effects are characterized by changes in LERF, population dose rate, and conditional containment failure probability.

Note: as previously cited a value of 1.18 $ft.^2$ is conservatively used to represent this bypass area. [C-26]

Table C.3-1

SUMMARY OF CLINTON THERMAL HYDRAULIC CALCULATIONS FOR VARYING LOCAs AND DRYWELL BYPASS LEAKAGE

Figure No.	LOCA Size (dia)	DW Bypass Leakage Area (ft^2)[1]	Peak Containment Pressure Before RPV Breach	Margin to Containment Failure at LOCA	Cont. Leakage	Fraction of Noble Gas Released	Fraction of CsI Released
C.3-1	2 in.	7.56E-3	< 20 psia	89 psig	ε	NA	64 psig
C.3-2	18 in. (DBA)	7.56E-3	< 20 psia	89 psig	ε	NA	57 psig
C.3-3	2 in.	0.265	< 20 psia	89 psig	ε	NA	64 psig
C.3-4	18 in. (DBA)	0.265	< 20 psia	89 psig	ε	NA	57 psig
C.3-5	18 in. (DBA)	1.18	< 20 psia	89 psig	ε	NA	57 psig
C.3-6	DBA	7.56E-3	< 20 psia	89 psig	1 La	6.5E-3	1.7E-5
C.3-7	DBA	7.56E-3	< 20 psia	89 psig	35 La	0.2	2.3E-4
C.3-8	DBA	0.256	< 20 psia	89 psig	1 La	6.7E-3	3.9E-5
C.3-9	DBA	0.256	< 20 psia	89 psig	35 La	0.21	1.3E-4
C.3-10	DBA	1.18	< 20 psia	89 psig	1 La	6.9E-4	6.1E-5
C.3-11	2"	1.18	26 psia	83 psig	—	—	—

[1] 7.56E-3ft^2 = 300 SCFM @ 3 psid

0.265 ft^2 = 10,500 SCFM @ 3 psid

1.18 ft^2 = 43,120 SCFM @ 3 psid

Figure C.3-1 Containment Pressure for a 2 in. LOCA with 7.56E-3 ft^2 DW Bypass Leakage Area

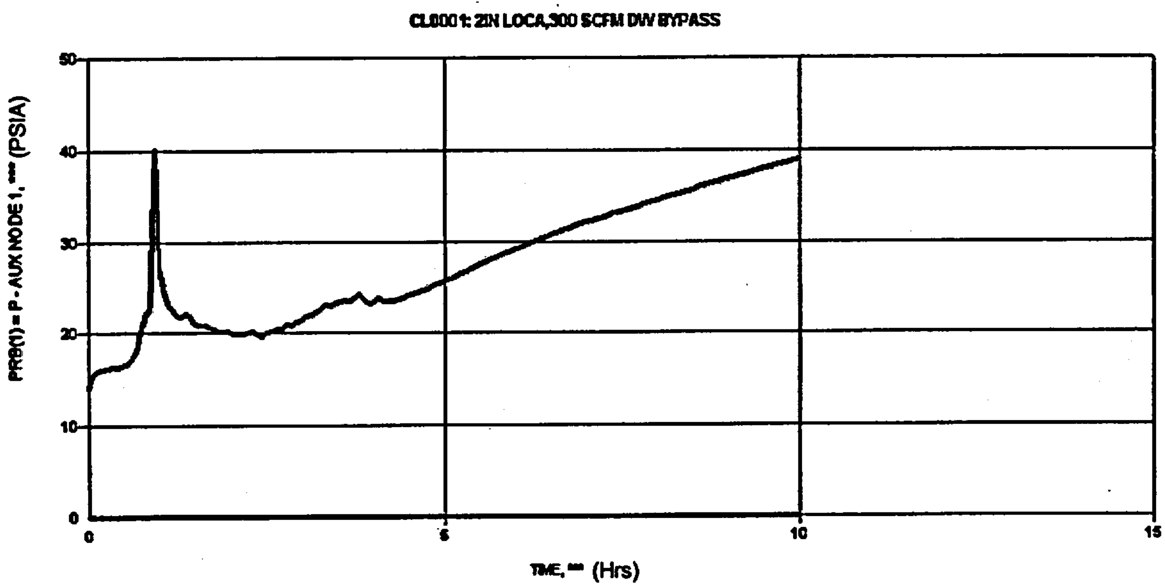


Figure C.3-2 Containment Pressure for DBA LOCA with 7.56E-3 ft^2 DW Bypass Leakage Area

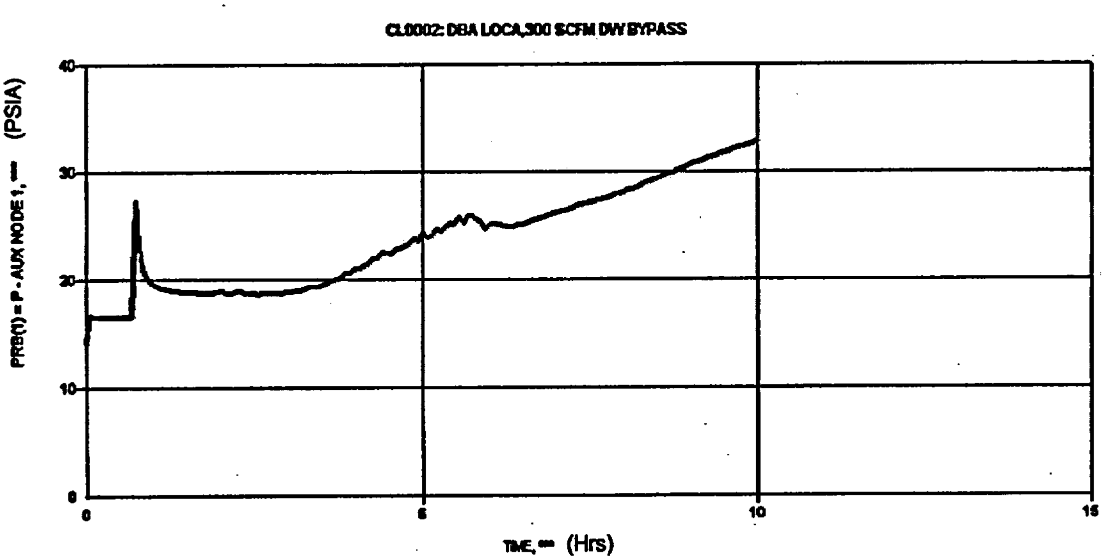


Figure C.3-3 Containment Pressure for a 2" LOCA with 0.265 ft^2 DW Bypass Leakage

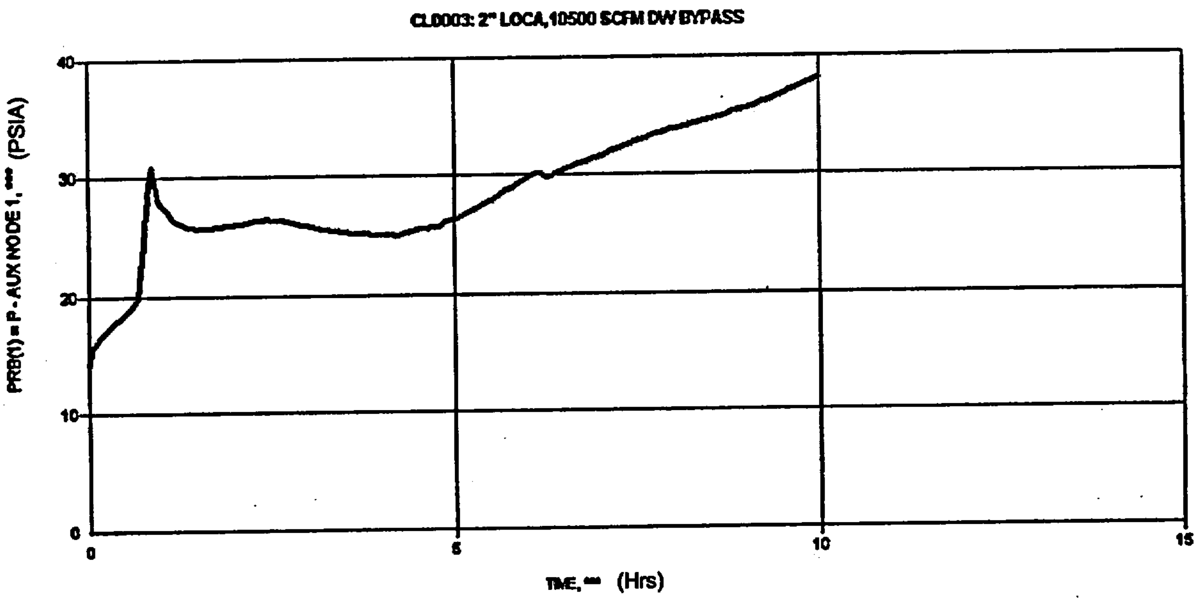


Figure C.3-4 Containment Pressure for a DBA LOCA with a 0.265 ft^2 DW Bypass Leakage

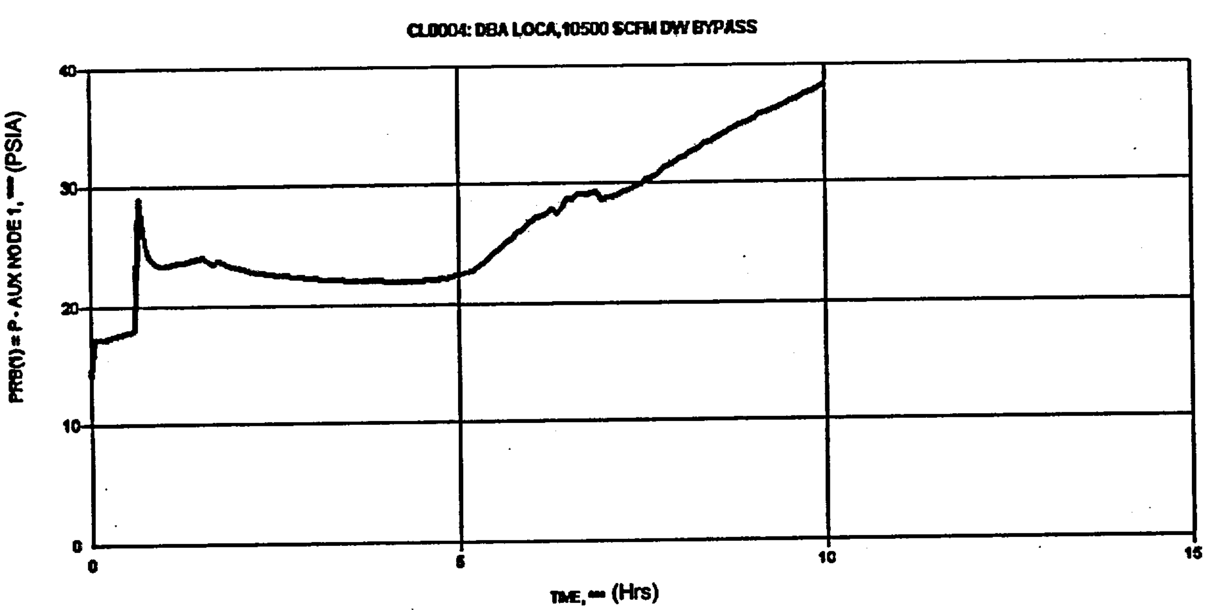


Figure C.3-5 Containment Pressure for a DBA LOCA with a 1.18 ft^2 DW Bypass Leakage

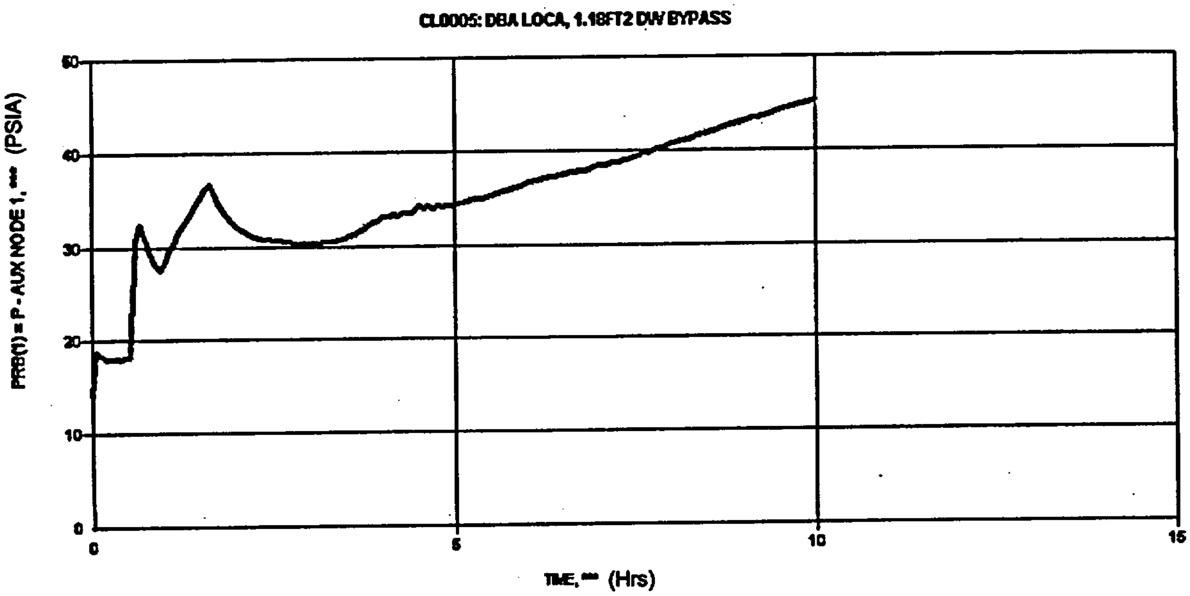


Figure C.3-6 Containment Pressure for a DBA LOCA with a 7.56E-3 ft^2 DW Bypass Leakage (Cont. Leakage of 1 La)

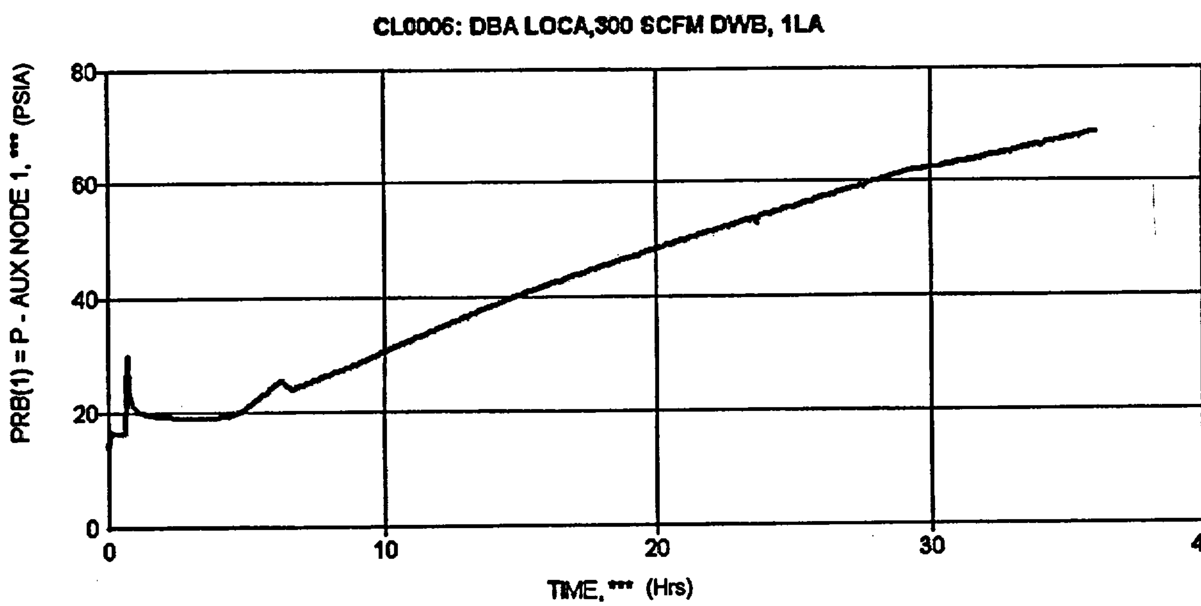


Figure C.3-7 Containment Pressure for a DBA LOCA with a 7.56E-3 ft² DW Bypass Leakage (Cont. Leakage of 35 La)

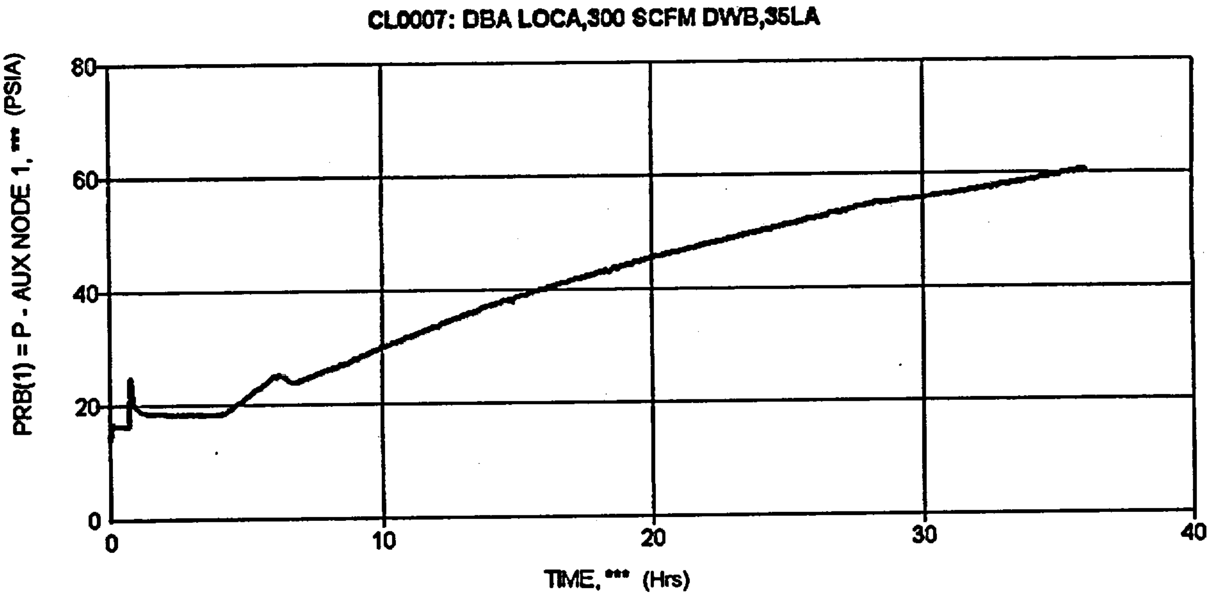


Figure C.3-8 Containment Pressure for a DBA LOCA with a 0.256 ft² DW Bypass Leakage (Cont. Leakage of 1 La)

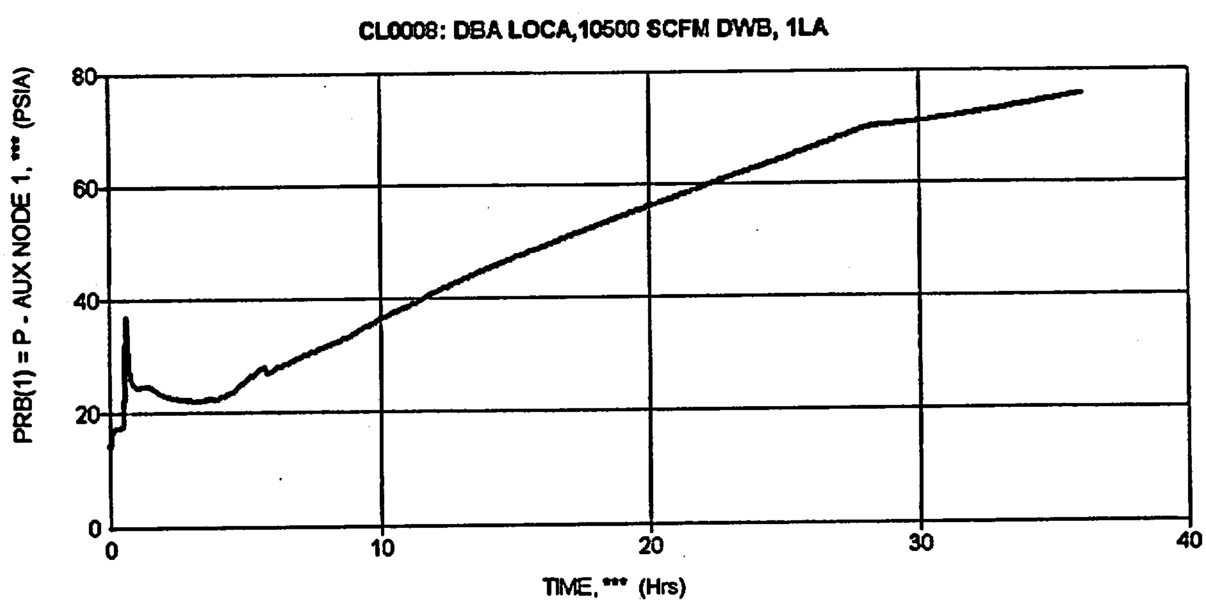


Figure C.3-9 Containment Pressure for a DBA LOCA with a 0.256 ft^2 DW Bypass Leakage (Cont. Leakage of 35 La)

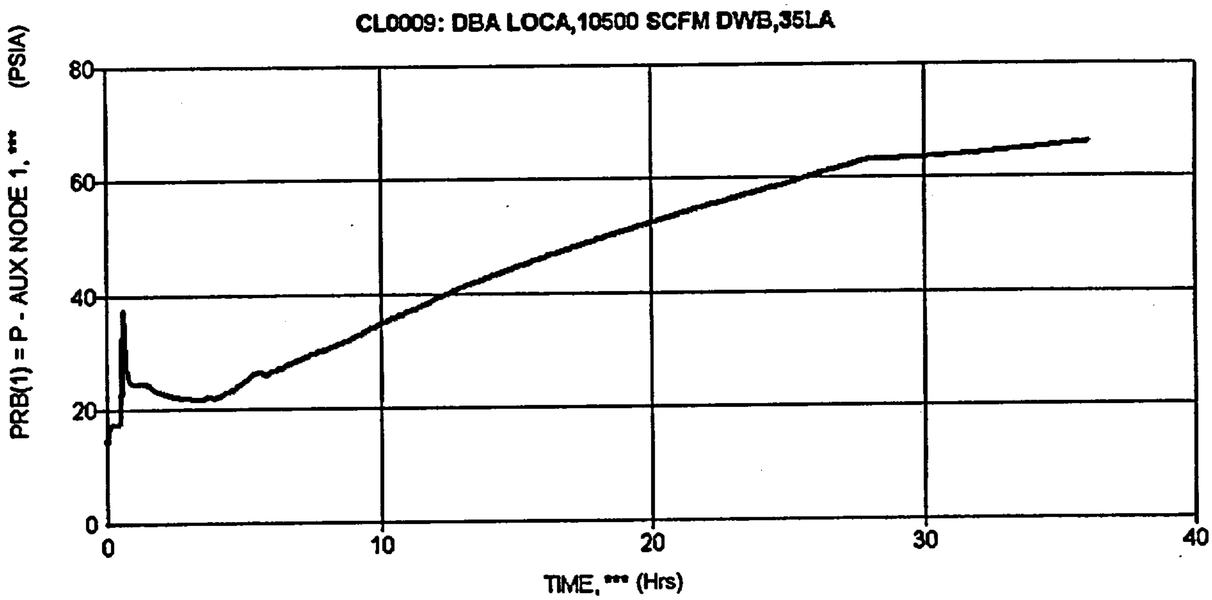


Figure C.3-10 Containment Pressure for a DBA LOCA with a 1.18 ft^2 DW Bypass Leakage (Cont. Leakage of 1 La)

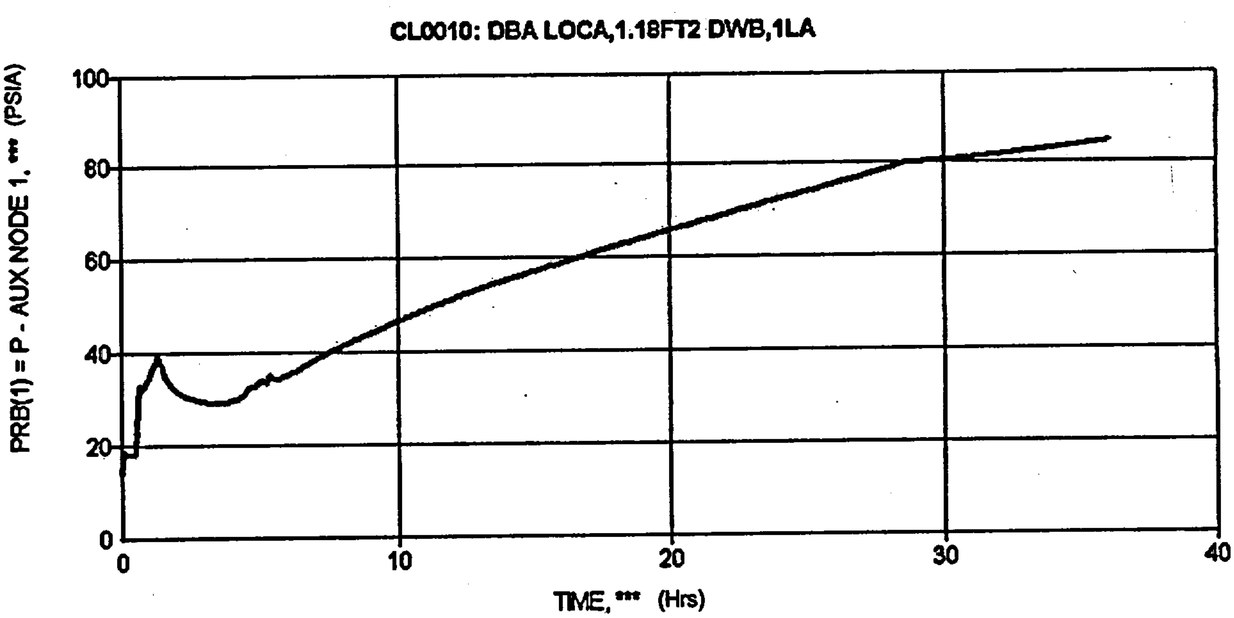
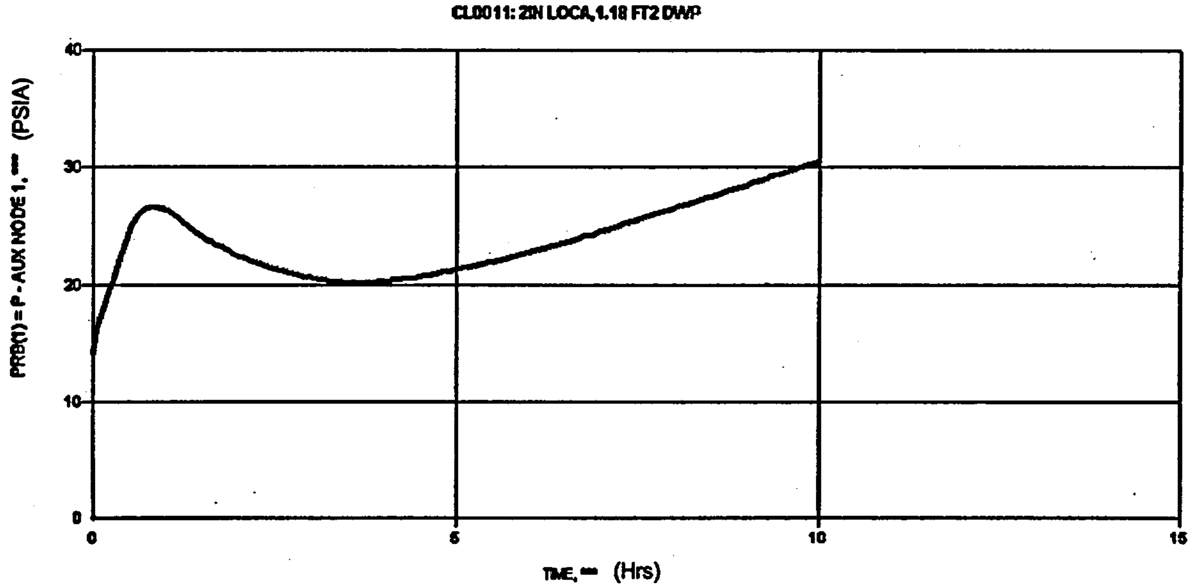


Figure C.3-11 Containment Pressure for a 2 inch LOCA with a 1.18 ft^2 DW Bypass Leakage



C.3.2 DRYWELL LEAKAGE RATES (STEP 2)

The second step of the NEI Interim Guidance is to define the drywell leakage rates for EPRI Categories 3a and 3b. As discussed earlier, EPRI Categories 3a and 3b are accidents with pre-existing drywell leakage pathways ("small" and "large", respectively) that would only be identifiable from a DWBT.

The NEI Interim Guidance recommends containment leakage rates of $10L_a$ and $35L_a$ for Categories 3a and 3b, respectively. The NEI Interim Guidance describes these two recommended containment leakage rates as "conservative". These values are consistent with previous ILRT frequency extension submittal applications. L_a is the plant Technical Specification maximum allowable containment leak rate; for Clinton L_a is 0.65% of containment air weight per day (per Clinton Technical Specifications).

The calculations performed to assess the risk changes use reasonable yet conservative estimates of the drywell and containment leakage. The NEI recommended values of $10L_a$ and $35L_a$ are used "as is" in this analysis to characterize the containment leakage rates for Categories 3a and 3b. For the containment, this translates into multiples of the allowable Technical Specification leakage (La). However for the drywell, such an assumption would appear unreasonable based upon the historical evidence for the drywell bypass leakage. The "as found" DW Bypass leakage historical tests are shown in the previous subsection in Table C.2-1. This information clearly identifies that the historical leakage rate is quite low, i.e., in the range of 20-30 scfm which is a factor of > 1000 below the design limit.

Therefore, the analysis is performed using the leakage characteristic of the "as found" state of the drywell. This recognizes both the historical results of the DWBT and the fact that Clinton continuously monitors the DW leakage. CPS is committed to trending this monitored information and noting any adverse trends (which there have been none). Based on these results and the continuous on-line monitoring, it is considered appropriate to use the conservatively high leakage rate of 300 scfm (DWL_c)[1] as the baseline leakage characteristic of a 3/10 year DWBT frequency. This is conservative, but is not as large as the Technical Specification allowable. The rationale for using a conservative but more realistic value than the Technical Specification leakage for the drywell is that the last five DWBTs show that the drywell leakage is below 31 scfm which is more than two orders of magnitude below the Technical Specification limit. The conservative analysis characterization of the DWBT using 300 scfm bounds even the initial drywell leakage which had defective electrical penetrations. These defective electrical penetrations were subsequently repaired.

The Technical Specification allowable leakage for the drywell is not used because of the on-line monitoring that is established by the past DWBT. Use of the Technical Specification limit would mischaracterize the Clinton drywell integrity and would make the decision <u>not</u> risk-informed. Therefore, the DW leakage is characterized in the analysis to be 1 times, 10 times, or 35 times a conservative characterization of the drywell leakage, which is referred to in this analysis as DWL_C.

This leads to the specification of the drywell leakage rates consistent with the EPRI ILRT methodology:

Minimal leakage case	300 SCFM @ 3 psid (DWL_C)
10 DWL_C case	3000 SCFM @ 3 psid
35 DWL_C case	10,500 SCFM @ 3 psid

These represent very conservative characterizations of the "as found" drywell bypass leakage and are interpreted in terms of an equivalent leakage area in Table C.3-1 and in the deterministic calculations performed to support these analyses.

(1) A realistic estimate would be closer to 30 scfm. This conservatism affects the population dose estimates.

By definition, the containment leakage rate for Category 1 (i.e., accidents with containment leakage at or below maximum allowable Technical Specification leakage) is $1.0L_a$ (or 1.0 DWL_C for the drywell).

The data available from NUREG/CR 4551 (Grand Gulf) support the ILRT assessment. However, the examination of the potentially small differences associated with the DWBT changes are not within the NUREG/CR 4551 level of discrimination. To allow an estimation of the potential impact, the effects of DW bypass are scaled using MAAP computer code calculations.

Deterministic calculations summarized in Table C.3-1 are then used to assist in the identification of those combinations of drywell and wetwell leakage that are to be assigned to the EPRI categories, i.e., Category 1, 3a, or 3b. Table C.3-2 provides this link between the EPRI categories and the combinations of DW and WW leakage. Table C.3-2 summarizes the population dose as a function of the possible combinations of DW and WW leakage.

Table C.3-2 makes use of the deterministic calculations from the surrogate plant discussed in the main report to characterize the population dose for leakage combination Cases AA', AB', and AC'. The remaining leakage combination cases represent various combinations of DW and WW leakage that can lead to variable population doses. These cases are benchmarked using MAAP and then the population dose is interpolated using the CsI release calculated by MAAP for the appropriate cases. These estimates are based on LOCA accidents with no containment sprays. These CsI radionuclide release assumptions create conservative estimates of the fission product releases that bypass the drywell. The population dose is then inferred for each of these cases as discussed in the table notes based on the fractional CsI released.

Once this is completed, each case is assigned to an EPRI category. Those that meet or exceed the definition of 3b, consistent with the ILRT method, are cited as a LERF contributor.

The threshold developed for LERF is generally characterized by greater than 10% CsI release. [C-25] As can be seen from the Clinton specific MAAP calculations, the CsI release fraction for Category 3b is very low, much lower than the threshold used for LERF. However, consistent with the EPRI ILRT methodology, Category 3b is still used as a surrogate measure of the LERF changes.

This appears to be severely conservative when characterizing the change in LERF associated with the ILRT and DWBT interval extensions.

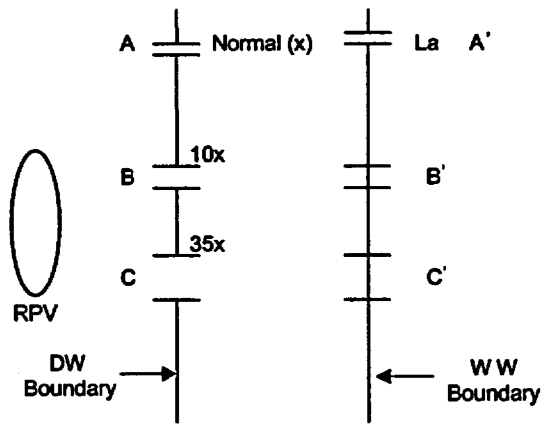
Conditional Probabilities of Category 3a and 3b

The characterization of EPRI Categories 3a and 3b are dictated in the EPRI and NEI guidance for the ILRT risk evaluation for PWRs and BWR Mark I and II containments. However, for Mark III containments, the interpretation of the Categories 3a and 3b are not straightforward because of the significantly different Mark III containment and drywell relationship. Therefore, the deterministic calculations presented in Table C.3-1 are used to provide a logical method of categorizing the combinations of leakage paths into Category 3a or 3b.

As noted previously in Section C.2:

> The Mark III containment has a different arrangement from either PWR containments or BWR Mark I/II containments. The difference is that the drywell which includes the RPV is completely enclosed by the outer containment. As such, the drywell leakage (the subject of this analysis) does not leak directly to the environment but is further mitigated by the outer containment leakage path. Because of this "dual" containment, there are several possible leakage path combinations that must be considered. These combinations are identified below. Note that Categories 3a and 3b

will be made up of multiple combinations of DW and WW leakage pathways:



where x = Assumed Baseline Leakage Rate

The failure combinations are provided in the following table for the baseline case of DWBT and ILRT every 3.3 years:

Failures DW	Failures WW	Probability	Consequence Characterization
A	A'	--	Already included in the ILRT Assessment (Class I)
A	B'	--	Already included in the ILRT Assessment (Class 3a)
A	C'	--	Already included in the ILRT Assessment (Class 3b)
B	A'	2.7E-2	Leak Failure Mode with small DW Bypass
B	B'	$(2.7E\text{-}2)^2$	Small pre-existing failure mode with small DW Bypass
B	C'	2.7E-2 * 2.7E-3	Large pre-existing failure mode with small DW Bypass
C	A'	2.7E-3	Leakage Failure Mode with large DW Bypass
C	B'	2.7E-3 * 2.7E-2	Small pre-existing failure mode with large DW Bypass
C	C'	$(2.7E\text{-}3)^2$	Large pre-existing failure mode with large DW Bypass

Table C.3-2

SUMMARY OF POPULATION DOSE AS A FUNCTION OF THE COMBINATIONS OF DW AND WW LEAKAGE

Case	DW Bypass Leakage[10]	WW Leakage[11]	Person Rem[8]	Factional CsI Release	Equivalent EPRI Category	LERF Characterization
AA'[1]	1 DWL_C	1 La	2.4E+3[1]	1.7E-5[2]	1	Non-LERF
AB'[1]	1 DWL_C	10 La	2.4E+4[1]	NA	3a	Non-LERF
AC'[1]	1 DWL_C	35 La	8.4E+4[1]	2.3E-4[2]	3b[9]	LERF
BA'	10 DWL_C	1 La	3.9E+3[4]	2.1E-5[3]	1	Non-LERF
BB'	10 DWL_C	10 La	3.9E+4[5]	NA	3a	Non-LERF
BC'	10 DWL_C	35 La	1.4E+5[6]	NA	3b[9]	LERF
CA'	35 DWL_C	1 La	1.0E+4[4]	3.9E-5[2]	3a	Non-LERF
CB'	35 DWL_C	10 La	1.0E+5[7]	NA	3b[9]	LERF
CC'	35 DWL_C	35 La	4.9E+5[4]	1.35E-3[2]	3b[9]	LERF

NA = Not available from the MAAP cases performed to model the Leakage cases.

Notes to Table C.3-2

(1) These are the cases developed in the ILRT submittal (see the Main Report) assuming a 1/15 year ILRT interval. The Person Rem are assigned using the EPRI methodology.

The EPRI accident classes are assigned according to the methodology developed by EPRI as are the LERF characterization.

(2) Based on deterministic thermal hydraulic calculations on a Clinton-specific basis.

(3) CsI release based on an assumed linear relationship of the CsI release to the DW leakage rate.

(4) Derived using the calculated CsI release fraction and the calculated relationship between Clinton CsI fractional release and the person-rem dose. (See Attachment C-2.)

From this relationship, the person rem associated with different leakage cases is estimated as follows:

CsI Fraction Release (x)	*Person Rem* (y)
3.9E-5	1.0E+4
1.3E-3	4.9E+5
2.1E-5	3.9E+3

(5) Calculated according to EPRI method (10 times Case D which is 1 La WW leakage).

(6) Calculated according to EPRI method (35 times Case D which is 1 La WW leakage).

(7) Calculated according to EPRI method (10 times Case G which is 1 La WW leakage).

(8) Person rem is adjusted for CPS power, leakage rate, and population.

(9) It is important to note that the CsI release for Category 3b is substantially below that typically assigned to LERF. However, to be consistent with the NEI guidelines, the Category 3b is treated here as a LERF contributor. This appears to be severely conservative when characterizing the change in LERF associated with the ILRT and DWBT interval extensions.

(10) DW leakage is based on multiples of a conservative estimate of the DW leakage from the "as found" DWBT results and subsequent trending information (DWL_c).

(11) WW leakage is based on multiples of the Tech Spec Leakage (La). This is consistent with the NEI methodology and operating experience.

C.3.3 BASELINE FREQUENCY ESTIMATES (STEP 3)

This section summarizes the derivation of the baseline frequencies (ILRT and DWBT intervals of 3.3 years). In addition, the accident frequency summary tables for the ILRT and DWBT intervals of 10 years and 15 years are presented here.

The cases developed to examine the change in risk metrics due to changes in the ILRT and DWBT interval include the following:

Case No.	Cases With the Same ILRT and DWBT Interval	*Tables of Inputs and Results*		
		Conditional Probability of Categories 1, 3a, and 3b	Frequencies of EPRI Categories	Dose Rate
1	3/10 years (Baseline)	Table C.3-3a	Table C.3-3b	Table C.3-3c
2	1/10 years	Table C.3-4a	Table C.3-4b	Table C.3-4c
3	1/15 years	Table C.3-5a	Table C.3-5b	Table C.3-5c

Note that these are cases that account for the integral effect of changing both the ILRT and DWBT frequencies simultaneously.

As noted in the above table, there are a series of three tables associated with each of the cases. These series of tables provide the following:

- Table -a: Gives the conditional probabilities of the various leakage pathways depending upon the test interval assumed (i.e., the case)
- Table -b: Gives the calculated frequencies for each EPRI category for the case
- Table -c: Provides the population dose rate calculation in person rem/year for the case. (Note that the population dose derivation is included in Section C.3.3 and Table C.3-6.)

Case 1 is the Baseline evaluation required for Step 2. This baseline evaluation follows the EPRI ILRT methodology while also accommodating the unique Mark III arrangement with the drywell enclosed by the containment.

Cases 2 and 3, while provided here, are discussed in detail in Steps 5, 6, and 7 (Section 3.4) where the changes in leakage probability and its effect on dose rate are calculated when the ILRT and DWBT frequency is reduced from the baseline (3/10 years) to 1/10 years (Case 2) and 1/15 years (Case 3).

Table C.3-3a

SUMMARY OF THE CONDITIONAL PROBABILITY OF OCCURRENCE
FOR THE VARIOUS POSTULATED LEAKAGE CASES

(Case No. 1: ILRT and DWBT Frequencies at 3/10 Years)

1	*2*	*3*	*4*	*5*	*6*	*7*
			PROBABILITY OF CASE			
Leakage Combinations	DW Bypass Leakage[(1)]	WW Leakage[(2)]	DW	WW	Combined	EPRI Class
AA'	1 DWL_c	1 La	1.0	1.0	1.0	1
AB'	1 DWL_c	10 La	1.0	2.7E-2	2.7E-2	3a
AC'	1 DWL_c	35 La	1.0	2.7E-3	2.7E-3	3b
BA'	10 DWL_c	1 La	2.7E-2	~ 1.0	2.7E-2	1
BB'	10 DWL_c	10 La	2.7E-2	2.7E-2	7.3E-4	3a
BC'	10 DWL_c	35 La	2.7E-2	2.7E-3	7.3E-5	3b
CA'	35 DWL_c	1 La	2.7E-3	~ 1.0	2.7E-3	3a
CB'	35 DWL_c	10 La	2.7E-3	2.7E-2	7.3E-5	3b
CC'	35 DWL_c	35 La	2.7E-3	2.7E-3	7.3E-6	3b

(1) DW leakage is based on multiples of a conservative estimate of the DW leakage from the "as found" DWBT results and subsequent trending information (DWLc).

(2) WW leakage is based on multiples of the Tech Spec Leakage (La). This is consistent with the NEI methodology and operating experience.

Table C.3-3b

SUMMARY OF CLINTON REVISED BASELINE RELEASE FREQUENCIES AS A FUNCTION OF EPRI CATEGORY

(Case No. 1 ILRT and DWBT Frequency = 3/10 yrs)

EPRI Category	Category Description	Frequency Estimation Methodology	Frequency (1/yr)
1	No Containment Failure: Accident sequences in which the containment remains intact and is initially isolated. Only affected by ILRT leak testing frequency due to the incorporation of categories 3a and 3b.	Per NEI Interim Guidance: [Total Clinton "OK" release category frequency] – [Frequency EPRI Categories 3a and 3b] [5.47E-6/yr] – [6.29E-7/yr + 5.90E-8/yr] = 4.78E-6/yr	4.78E-6
2	Containment Isolation System Failure: Accident sequences in which the containment isolation system function fails during the accident progression (e.g., due to failures-to-close of large containment isolation valves initiated by support system failures, or random or common cause failures). Not affected by ILRT leak testing frequency.	[Clinton containment isolation failure probability] X [(Total CDF) – (CDF of Class II + CDF of Class IV +CDF of Class V)] [4.99E-3] X [(2.76E-5/yr) – (3.79E-6/yr +9.9E-7/yr + 1.21E-7/yr)] = 1.13E-7/yr	1.1E-7
3a	Small Pre-Existing Failures: Accident sequences in which the containment is failed due to a pre-existing small leak in the containment structure or liner that would be identifiable only from an ILRT (and thus affected by ILRT testing frequency).	Per NEI Interim Guidance: DW leakage = 1DWL$_c$[(4)] [Clinton CDF for accidents not involving containment failure/bypass] x [2.7E-2][(3)] [(2.76E-5/yr) – (3.0E-6/yr + 3.79E-6 /yr + 1.21E-7/yr) x [2.7E-2][(3)] = 5.59E-3/yr Other Contributors[(5)] In addition, the following incremental effect associated with the combination of DW and WW leakage is included: [(2.76E-5/yr) – (3.0E-6/yr + 3.79E-6 /yr + 1.21E-7/yr) * (7.3E-4 + 2.7E-3)[(2)] = 2.07E-5/yr * 3.4E-3 = 7.03E-08/yr	6.29E-7

Table C.3-3b

SUMMARY OF CLINTON REVISED BASELINE RELEASE FREQUENCIES AS A FUNCTION OF EPRI CATEGORY

(Case No. 1 ILRT and DWBT Frequency = 3/10 yrs)

EPRI Category	Category Description	Frequency Estimation Methodology	Frequency (1/yr)
3b	Large Pre-Existing Failures: Accident sequences in which the containment is failed due to a pre-existing large leak in the containment structure or liner that would be identifiable only from an ILRT (and thus affected by ILRT testing frequency).	Per NEI Interim Guidance: (DW Leakage = 1 DWLc)[4] [Clinton CDF for accidents not involving containment failure/bypass] x [2.7E-3][3] [(2.76E-5/yr) – (3.0E-6/yr + 3.79E-6/yr + 1.21E-7/yr) x [2.7E-3][3] = 5.59E-8/yr Other Contributors[5] In addition, the frequency incremental effect associated with the combination of DW and WW leakage is included: [(2.76E-5/yr) – (3.0E-6/yr + 3.79E-6/yr + 1.21E-7/yr) * (7.3E-6 + 1.46E-4)[2] = 2.07E-5/yr * 1.53E-4 = 3.17E-9/yr	5.90E-8
4	Type B Failures: Accident sequences in which the containment is failed due to a pre-existing failure-to-seal of Type B components that would not be identifiable from a ILRT (and thus not affected by ILRT testing frequency).	Per NEI Interim Guidance: N/A (not affected by ILRT frequency)	N/A
5	Type C Failures: Accident sequences in which the containment is failed due to a pre-existing failure-to-seal of Type C components that would not be identifiable from a ILRT (and thus not affected by ILRT testing frequency).	Per NEI Interim Guidance: N/A (not affected by ILRT frequency)	N/A
6	Other Containment Isolation System Failure: Accident sequences in which the containment isolation system function fails due to "other" pre-existing failure modes not identifiable by leak rate tests (e.g., pathways left open or valves that did not properly seal following test or maintenance activities). Not affected by ILRT leak testing frequency.	Per NEI Interim Guidance: N/A (not affected by ILRT frequency)	N/A

Table C.3-3b

SUMMARY OF CLINTON REVISED BASELINE RELEASE FREQUENCIES AS A FUNCTION OF EPRI CATEGORY

(Case No. 1 ILRT and DWBT Frequency = 3/10 yrs)

EPRI Category	Category Description	Frequency Estimation Methodology	Frequency (1/yr)
7a	Containment Failure Due to Accident (a): CD, vessel breach, Early CF, Early SP Bypass, CS Not Available Vessel breach occurs and both the containment and the drywell have failed either before or at the time of vessel breach. The containment sprays do not operate before or at the time of vessel breach.	Total Clinton release mode frequency for: C9 ε C10 2.62E-7 C11 ε C12 1.82E-9 E1 ε E2 ε	2.6E-7
7b	Containment Failure Due to Accident (b): CD, vessel breach, Early CF, Early SP Bypass, CS Available Vessel breach occurs and both the containment and the drywell fail either before or at the time of vessel breach. In this bin, however, the containment sprays operate before or at the time of vessel breach.	Total Clinton release mode frequency for: C1 7.78E-8 C2 1.80E-7 C6 3.51E-6 C8 9.41E-7	4.7E-6
7c	Containment Failure Due to Accident (c): CD, vessel breach, Early CF, No SP Bypass Vessel breach occurs and the containment fails either before or at the time of vessel breach. The drywell does not fail and, therefore, all of the radionuclide releases pass through the suppression pool. Because the pool has not been bypassed, the availability of the sprays is not very important and, thus, the CS characteristic has been dropped.	Total Clinton release mode frequency for: B1 1.13E-6 B2 8.16E-6 A1 7.81E-6	1.7E-5

7d	Containment Failure Due to Accident (d): CD, vessel breach, Late CF Vessel breach occurs, however, the containment does not fail until the late time period. If the containment did not fail early, it is unlikely that the drywell will fail early. Thus, the suppression pool bypass characteristic and the containment spray characteristic have been dropped.	Total Clinton release mode frequency for: D5 9.91E-9 D6 9.07E-7	9.2E-7
8	Containment Bypass Accidents: Accident sequences in which the containment is bypassed. Such accidents are initiated by LOCAs outside containment (i.e., Break Outside Containment LOCA, or Interfacing Systems LOCA). Not affected by ILRT leak testing frequency.	[Total Clinton Containment Bypass release frequency]	1.2E-7
		TOTAL:	2.8E-5[1]

(1) Accurate to within a few percent of the total CDF (2.76E-5/yr). [18] Differences due to roundoff and EPRI calculational approach.

(2) The derivation of the conditional probability of a Category 3a and 3b release given an otherwise isolated accident sequence is derived in Table C.3-3a.

(3) As derived in Section 3.4.1, the following conditional probabilities have been derived as part of the NEI/EPRI ILRT methodology for containment:

Test Frequency	10 La	35 La
3/10 yr	2.7E-2	2.7E-3
1/10 yr	9.0E-2	9.0E-3
1/15 yr	0.135	0.0135

These same values are applied to both containment and DW leakage where DW leakage is characterized as multiplies of DWL_c.

(4) Contributor calculated in main report for ILRT interval extension.

(5) Other contributors associated with extension of both ILRT and DWBT intervals.

Table C.3-3c

CLINTON DOSE RATE ESTIMATES AS A FUNCTION OF EPRI CATEGORY FOR POPULATION WITHIN 50-MILES

(Case No 1: 3/10 year DWBT and ILRT)

EPRI Category	Category Description	Person-Rem Within 50 miles[6]	Category Frequency (per year)[7]	Dose Rate (Person-Rem/yr)
1	No Containment Failure[1]	2.4E+3	4.78E-6	1.15E-2
2	Containment Isolation System Failure[2]	5.1E+5	1.13E-7	5.76E-2
3a	Small Pre-Existing Failures[3]	2.4E+4	6.29E-7	1.51E-2
3b	Large Pre-Existing Failures[3]	8.4E+4[9]	5.90E-8[9]	5.02E-3[9]
4	Type B Failures (LLRT)	n/a	n/a	n/a
5	Type C Failures (LLRT)	n/a	n/a	n/a
6	Other Containment Isolation System Failure	n/a	n/a	n/a
7a	Containment Failure Due to Severe Accident (a)[4]	5.1E+5	2.63E-7	1.34E-1
7b	Containment Failure Due to Severe Accident (b)[4]	3.5E+5	4.7E-6	1.65
7c	Containment Failure Due to Severe Accident (c)[4]	3.7E+5	1.71E-5	6.33
7d	Containment Failure Due to Severe Accident (d)[4]	3.0E+5	9.2E-7	2.76E-1
8	Containment Bypass Accidents[5]	5.1E+5	1.21E-7	6.17E-2
	Total		2.86E-5[8]	8.54

Notes to Table C.3-3c

(1) The population dose associated with the Technical Specification Leakage is based on scaling the population data, the power level, and allowable Technical Specification leakage compared to the NUREG/CR-4551 reference plant. The release for this EPRI category is assigned from APB#7 from Table 3-4 of the main report.

(2) EPRI Category #2 (Containment Isolation failures) may include drywell isolation failures. Therefore, the release associated with this category is assigned to be equivalent to the release associated with APB#1 from Table 3-4 of the main report.

(3) Dose estimates for #3a and #3b, per the NEI Interim Guidance, are calculated as 10xCategory 1 dose and 35xCategory 1 dose, respectively.

(4) Dose estimate for 7a, 7b, 7c, and 7d are taken from APB # 1, 2, 4, and 5, respectively. (See main Report.)

(5) EPRI Category #8 sequences involve containment bypass failures; as a result, the person-rem dose is not based on normal containment leakage. The releases for this category are assumed to result in a direct path to the environment, and as such, are assigned to be equivalent to the highest release category from NUREG/CR-4551. APB#1 from Table 3-4 is therefore used of the main report.

(6) Table C.3-2.

(7) Table C.3-3b.

(8) Within a few percent of total CDF of 2.76E-5/yr [18]. Slight differences are due to the EPRI calculational approach and round off. The use of slightly higher frequencies in Table C.3-6 is conservative for assessing the risk metric of dose rate.

(9) Large pre-existing failure estimates of the population dose are based on using the extrapolated person rem for Case CC' (Table C.3-2) in the calculation of those contributors with 35DWLc and 35 La and adding these results to the other large pre-existing failures.

Table C.3-4a

SUMMARY OF THE CONDITIONAL PROBABILITY OF OCCURRENCE
FOR THE VARIOUS POSTULATED LEAKAGE CASES

(Case No. 2: ILRT and DWBT Frequencies at 1/10 Years)

1	*2*	*3*	*4*	*5*	*6*	*7*
			PROBABILITY OF CASE			
Leakage Combinations	DW Bypass Leakage[1]	WW Leakage[2]	DW	WW	Combined	EPRI Class
AA'	1 DWL_c	1 La	1.0	1.0	1.0	1
AB'	1 DWL_c	10 La	1.0	9.0E-2	9.0E-2	3a
AC'	1 DWL_c	35 La	1.0	9.0E-3	9.0E-3	3b
BA'	10 DWL_c	1 La	9.0E-2	~ 1.0	9.0E-2	1
BB'	10 DWL_c	10 La	9.0E-2	9.0E-2	8.1E-3	3a
BC'	10 DWL_c	35 La	9.0E-2	9.0E-3	8.1E-4	3b
CA'	35 DWL_c	1 La	9.0E-3	~ 1.0	9.0E-3	3a
CB'	35 DWL_c	10 La	9.0E-3	9.0E-2	8.1E-4	3b
CC'	35 DWL_c	35 La	9.0E-3	9.0E-3	8.1E-5	3b

(1) DW leakage is based on multiples of a conservative estimate of the DW leakage from the "as found" DWBT results and subsequent trending information (DWLc).

(2) WW leakage is based on multiples of the Tech Spec Leakage (La). This is consistent with the NEI methodology and operating experience.

Table C.3-4b

SUMMARY OF CLINTON RELEASE FREQUENCIES AS A FUNCTION OF EPRI CATEGORY

(Case No. 2: ILRT and DWBT Frequency = 1/10 yrs)

EPRI Category	Category Description	Frequency Estimation Methodology	Frequency (1/yr)
1	No Containment Failure: Accident sequences in which the containment remains intact and is initially isolated. Only affected by ILRT leak testing frequency due to the incorporation of categories 3a and 3b.	Per NEI Interim Guidance: [Total Clinton "OK" release category frequency] – [Frequency EPRI Categories 3a and 3b] [5.47E-6/yr] – [2.21E-6/yr + 2.21E-7/yr] = 3.04E-6/yr	3.04E-6
2	Containment Isolation System Failure: Accident sequences in which the containment isolation system function fails during the accident progression (e.g., due to failures-to-close of large containment isolation valves initiated by support system failures, or random or common cause failures). Not affected by ILRT leak testing frequency.	[Clinton containment isolation failure probability] X [(Total CDF) – (CDF of Class II + CDF of Class IV +CDF of Class V)] [4.99E-3] X [(2.76E-5/yr) – (3.79E-6/yr +9.9E-7/yr + 1.21E-7/yr)] = 1.13E-7/yr	1.1E-7
3a	Small Pre-Existing Failures: Accident sequences in which the containment is failed due to a pre-existing small leak in the containment structure or liner that would be identifiable only from an ILRT (and thus affected by ILRT testing frequency).	Per NEI Interim Guidance: (DW Leakage = 1 DWL_c)(4) [Clinton CDF for accidents not involving containment failure/bypass] x [9.0E-2](3) [(2.76E-5/yr) – (3.0E-6/yr + 3.79E-6 /yr + 1.21E-7/yr)] x [9.0E-2](3) = 1.86E-6/yr Other Contributors(5) In addition, the following incremental effect associated with the combination of DW and WW leakage is included: [(2.76E-5/yr) – (3.0E-6/yr + 3.79E-6 /yr + 1.21E-7/yr)] * (8.1E-3 + 9.0E-3)(2) = 2.07E-5/yr * 1.71E-2 = 3.5E-7/yr	2.21E-6

Table C.3-4b

SUMMARY OF CLINTON RELEASE FREQUENCIES AS A FUNCTION OF EPRI CATEGORY

(Case No. 2: ILRT and DWBT Frequency = 1/10 yrs)

EPRI Category	Category Description	Frequency Estimation Methodology	Frequency (1/yr)
3b	Large Pre-Existing Failures: Accident sequences in which the containment is failed due to a pre-existing large leak in the containment structure or liner that would be identifiable only from an ILRT (and thus affected by ILRT testing frequency).	Per NEI Interim Guidance: (DW Leakage = 1 DWL_o)[4] [Clinton CDF for accidents not involving containment failure/bypass] x [9.0E-3][3] [(2.76E-5/yr) – (3.0E-6/yr + 3.79E-6/yr + 1.21E-7/yr)] x [9.0E-3][3] = 1.86E-7/yr Other Contributors[5] In addition, the frequency incremental effect associated with the combination of DW and WW leakage is included: [(2.76E-5/yr) – (3.0E-6/yr + 3.79E-6/yr + 1.21E-7/yr)] * (1.62E-3 + 8.1E-5)[2] = 2.07E-5/yr * 1.70E-3 = 3.52E-8/yr	2.21E-7
4	Type B Failures: Accident sequences in which the containment is failed due to a pre-existing failure-to-seal of Type B components that would not be identifiable from a ILRT (and thus not affected by ILRT testing frequency).	Per NEI Interim Guidance: N/A (not affected by ILRT frequency)	N/A
5	Type C Failures: Accident sequences in which the containment is failed due to a pre-existing failure-to-seal of Type C components that would not be identifiable from a ILRT (and thus not affected by ILRT testing frequency).	Per NEI Interim Guidance: N/A (not affected by ILRT frequency)	N/A
6	Other Containment Isolation System Failure: Accident sequences in which the containment isolation system function fails due to "other" pre-existing failure modes not identifiable by leak rate tests (e.g., pathways left open or valves that did not properly seal following test or maintenance activities). Not affected by ILRT leak testing frequency.	Per NEI Interim Guidance: N/A (not affected by ILRT frequency)	N/A

Table C.3-4b

SUMMARY OF CLINTON RELEASE FREQUENCIES AS A FUNCTION OF EPRI CATEGORY

(Case No. 2: ILRT and DWBT Frequency = 1/10 yrs)

EPRI Category	Category Description	Frequency Estimation Methodology	Frequency (1/yr)
7a	<u>Containment Failure Due to Accident (a):</u> CD, vessel breach, Early CF, Early SP Bypass, CS Not Available Vessel breach occurs and both the containment and the drywell have failed either before or at the time of vessel breach. The containment sprays do not operate before or at the time of vessel breach.	Total Clinton release mode frequency for: C9 ε C10 2.62E-7 C11 ε C12 1.82E-9 E1 ε E2 ε	2.6E-7
7b	<u>Containment Failure Due to Accident (b):</u> CD, vessel breach, Early CF, Early SP Bypass, CS Available Vessel breach occurs and both the containment and the drywell fail either before or at the time of vessel breach. In this bin, however, the containment sprays operate before or at the time of vessel breach.	Total Clinton release mode frequency for: C1 7.78E-8 C2 1.80E-7 C6 3.51E-6 C8 9.41E-7	4.7E-6
7c	<u>Containment Failure Due to Accident (c):</u> CD, vessel breach, Early CF, No SP Bypass Vessel breach occurs and the containment fails either before or at the time of vessel breach. The drywell does not fail and, therefore, all of the radionuclide releases pass through the suppression pool. Because the pool has not been bypassed, the availability of the sprays is not very important and, thus, the CS characteristic has been dropped.	Total Clinton release mode frequency for: B1 1.13E-6 B2 8.16E-6 A1 7.81E-6	1.7E-5

7d	Containment Failure Due to Accident (d): CD, vessel breach, Late CF Vessel breach occurs, however, the containment does not fail until the late time period. If the containment did not fail early, it is unlikely that the drywell will fail early. Thus, the suppression pool bypass characteristic and the containment spray characteristic have been dropped.	Total Clinton release mode frequency for: D5 9.91E-9 D6 9.07E-7	9.2E-7
8	Containment Bypass Accidents: Accident sequences in which the containment is bypassed. Such accidents are initiated by LOCAs outside containment (i.e., Break Outside Containment LOCA, or Interfacing Systems LOCA). Not affected by ILRT leak testing frequency.	[Total Clinton Containment Bypass release frequency]	1.2E-7
		TOTAL:	2.8E-5(1)

(1) Accurate to within a few percent of the total CDF (2.76E-5/yr). [18] Differences due to roundoff and EPRI calculational approach.

(2) The derivation of the conditional probability of a Category 3a and 3b release given an otherwise isolated accident sequence is derived in Table C.3-4a.

(3) As derived in Section 3.4.1, the following conditional probabilities have been derived as part of the NEI/EPRI ILRT methodology for containment:

Test Frequency	10 La	35 La
3/10 yr	2.7E-2	2.7E-3
1/10 yr	9.0E-2	9.0E-3
1/15 yr	0.135	0.0135

These same values are applied to both containment and DW leakage where DW leakage is characterized as multiples of DWL_c.

(4) Contributor calculated in main report for ILRT interval extension.

(5) Other contributors associated with extension of both ILRT and DWBT intervals.

Table C.3-4c

CLINTON DOSE RATE ESTIMATES AS A FUNCTION OF EPRI CATEGORY FOR POPULATION WITHIN 50-MILES

(Case No 2: ILRT and DWBT With Frequency of 1/10 year)

EPRI Category	Category Description	Person-Rem Within 50 miles[6]	Category Frequency (per year)[7]	Dose Rate (Person-Rem/yr)
1	No Containment Failure[1]	2.4E+3	3.04E-6	7.30E-3
2	Containment Isolation System Failure[2]	5.1E+5	1.13E-7	5.76E-2
3a	Small Pre-Existing Failures[3]	2.4E+4	2.21E-6	5.30E-2
3b	Large Pre-Existing Failures[3]	8.4E+4[9]	2.21E-7[91)]	1.93E-2[9]
4	Type B Failures (LLRT)	n/a	n/a	n/a
5	Type C Failures (LLRT)	n/a	n/a	n/a
6	Other Containment Isolation System Failure	n/a	n/a	n/a
7a	Containment Failure Due to Severe Accident (a)[4]	5.1E+5	2.63E-7	1.34E-1
7b	Containment Failure Due to Severe Accident (b)[4]	3.5E+5	4.7E-6	1.65
7c	Containment Failure Due to Severe Accident (c)[4]	3.7E+5	1.71E-5	6.33
7d	Containment Failure Due to Severe Accident (d)[4]	3.0E+5	9.2E-7	2.76E-1
8	Containment Bypass Accidents[5]	5.1E+5	1.21E-7	6.17E-2
	Total		2.86E-5[8]	8.59

Notes to Table C.3-4c

(1) The population dose associated with the Technical Specification Leakage is based on scaling the population data, the power level, and allowable Technical Specification leakage compared to the NUREG/CR-4551 reference plant. The release for this EPRI category is assigned from APB#7 from Table 3-4 of the main report.

(2) EPRI Category #2 (Containment Isolation failures) may include drywell isolation failures. Therefore, the release associated with this category is assigned to be equivalent to the release associated with APB#1 from Table 3-4 of the main report.

(3) Dose estimates for #3a and #3b, per the NEI Interim Guidance, are calculated as 10xCategory 1 dose and 35xCategory 1 dose, respectively.

(4) Dose estimate for 7a, 7b, 7c, and 7d are taken from APB # 1, 2, 4, and 5, respectively. (See main Report.)

(5) EPRI Category #8 sequences involve containment bypass failures; as a result, the person-rem dose is not based on normal containment leakage. The releases for this category are assumed to result in a direct path to the environment, and as such, are assigned to be equivalent to the highest release category from NUREG/CR-4551. APB#1 from Table 3-4 is therefore used of the main report.

(6) Table C.3-2.

(7) Table C.3-4b.

(8) Within a few percent of total CDF of 2.76E-5/yr [18]. Slight differences are due to the EPRI calculational approach and round off. The use of slightly higher frequencies in Table C.3-6 is conservative for assessing the risk metric of dose rate.

(9) Large pre-existing failure estimates of the population dose are based on using the extrapolated person rem for Case CC' (Table C.3-2) in the calculation of those contributors with 35DWLc and 35 La and adding these results to the other large pre-existing failures.

Table C.3-5a

SUMMARY OF THE CONDITIONAL PROBABILITY OF OCCURRENCE
FOR THE VARIOUS POSTULATED LEAKAGE CASES

(Case No. 3: ILRT and DWBT Frequencies of 1/15 yr)

1	2	3	4	5	6	7
			PROBABILITY OF CASE			
Leakage Combinations	DW Bypass Leakage	WW Leakage	DW[(2)]	WW[(1)]	Combined	EPRI Class
A	1 DWL_C	1 La	1.0	1.0	1.0	1
B	1 DWL_C	10 La	1.0	.135	.135	3a
C	1 DWL_C	35 La	1.0	.0135	.0135	3b
D	10 DWL_C	1 La	0.135	~ 1.0	0.135	1
E	10 DWL_C	10 La	0.135	0.135	1.8E-2	3a
F	10 DWL_C	35 La	0.135	0.0135	1.8E-3	3b
G	35 DWL_C	1 La	0.0135	~ 1.0	0.0135	3a
H	35 DWL_C	10 La	0.0135	0.135	1.8E-3	3b
I	35 DWL_C	35 La	0.0135	0.0135	1.8E-4	3b

[(1)] The WW leakage cases for the DWBT evaluation use the assessed probabilities characteristic of the 1 per 15 year ILRT frequency.

10 La = 0.135
35 La = 0.0135

[(2)] The DW leakage estimates are using the 1/15 year DWBT frequency.

Table C.3-5b

SUMMARY OF CLINTON RELEASE FREQUENCIES AS A FUNCTION OF EPRI CATEGORY

(Case No 3: ILRT and DWBT Frequencies of 1/15 yrs)

EPRI Category	Category Description	Frequency Estimation Methodology	Frequency (1/yr)
1	<u>No Containment Failure</u>: Accident sequences in which the containment remains intact and is initially isolated. Only affected by ILRT leak testing frequency due to the incorporation of categories 3a and 3b.	Per NEI Interim Guidance: [Total Clinton "OK" release category frequency] – [Frequency EPRI Categories 3a and 3b] [5.47E-6/yr] – [3.45E-6/yr + 3.59E-7/yr] = 1.66E-6/yr	1.66E-6
2	<u>Containment Isolation System Failure</u>: Accident sequences in which the containment isolation system function fails during the accident progression (e.g., due to failures-to-close of large containment isolation valves initiated by support system failures, or random or common cause failures). Not affected by ILRT leak testing frequency.	[Clinton containment isolation failure probability] X [(Total CDF) – (CDF of Class II + CDF of Class IV +CDF of Class V)] [4.99E-3] X [(2.76E-5/yr) – (3.79E-6/yr +9.9E-7/yr + 1.21E-7/yr)] = 1.13E-7/yr	1.1E-7
3a	<u>Small Pre-Existing Failures</u>: Accident sequences in which the containment is failed due to a pre-existing small leak in the containment structure or liner that would be identifiable only from an ILRT (and thus affected by ILRT testing frequency).	Per NEI Interim Guidance: [(2.76E-5/yr) – (3.0E-6/yr + 3.79E-6 /yr + 1.21E-7/yr)] x [0.135]$^{(3)}$ = 2.8E-6/yr In addition, the following incremental effect associated with the combination of DW and WW leakage is included: [(2.76E-5/yr) – (3.0E-6/yr + 3.79E-6 /yr + 1.21E-7/yr)] * (1.8E-2+ 1.35E-2)$^{(2)}$ = 2.07E-5/yr * 3.15E-2 = 6.52E-7/yr	3.45E-6

Table C.3-5b

SUMMARY OF CLINTON RELEASE FREQUENCIES AS A FUNCTION OF EPRI CATEGORY

(Case No 3: ILRT and DWBT Frequencies of 1/15 yrs)

EPRI Category	Category Description	Frequency Estimation Methodology	Frequency (1/yr)
3b	Large Pre-Existing Failures: Accident sequences in which the containment is failed due to a pre-existing large leak in the containment structure or liner that would be identifiable only from an ILRT (and thus affected by ILRT testing frequency).	Per NEI Interim Guidance: [(2.76E-5/yr) – (3.0E-6/yr + 3.79E-6/yr + 1.21E-7/yr)] x [0.0135][(3)] = 2.8E-7/yr In addition, the frequency incremental effect associated with the combination of DW and WW leakage is included: [(2.76E-5/yr) – (3.0E-6/yr + 3.79E-6/yr + 1.21E-7/yr)] * (3.6E-3 + 1.8E-4) = 2.07E-5/yr * 3.78E-3 = 7.82E-8/yr	3.59E-7
4	Type B Failures: Accident sequences in which the containment is failed due to a pre-existing failure-to-seal of Type B components that would not be identifiable from a ILRT (and thus not affected by ILRT testing frequency).	Per NEI Interim Guidance: N/A (not affected by ILRT frequency)	N/A
5	Type C Failures: Accident sequences in which the containment is failed due to a pre-existing failure-to-seal of Type C components that would not be identifiable from a ILRT (and thus not affected by ILRT testing frequency).	Per NEI Interim Guidance: N/A (not affected by ILRT frequency)	N/A
6	Other Containment Isolation System Failure: Accident sequences in which the containment isolation system function fails due to "other" pre-existing failure modes not identifiable by leak rate tests (e.g., pathways left open or valves that did not properly seal following test or maintenance activities). Not affected by ILRT leak testing frequency.	Per NEI Interim Guidance: N/A (not affected by ILRT frequency)	N/A

Table C.3-5b

SUMMARY OF CLINTON RELEASE FREQUENCIES AS A FUNCTION OF EPRI CATEGORY

(Case No 3: ILRT and DWBT Frequencies of 1/15 yrs)

EPRI Category	Category Description	Frequency Estimation Methodology	Frequency (1/yr)
7a	<u>Containment Failure Due to Accident (a):</u> CD, vessel breach, Early CF, Early SP Bypass, CS Not Available Vessel breach occurs and both the containment and the drywell have failed either before or at the time of vessel breach. The containment sprays do not operate before or at the time of vessel breach.	Total Clinton release mode frequency for: C9 ε C10 2.62E-7 C11 ε C12 1.82E-9 E1 ε E2 ε	2.6E-7
7b	<u>Containment Failure Due to Accident (b):</u> CD, vessel breach, Early CF, Early SP Bypass, CS Available Vessel breach occurs and both the containment and the drywell fail either before or at the time of vessel breach. In this bin, however, the containment sprays operate before or at the time of vessel breach.	Total Clinton release mode frequency for: C1 7.78E-8 C2 1.80E-7 C6 3.51E-6 C8 9.41E-7	4.7E-6
7c	<u>Containment Failure Due to Accident (c):</u> CD, vessel breach, Early CF, No SP Bypass Vessel breach occurs and the containment fails either before or at the time of vessel breach. The drywell does not fail and, therefore, all of the radionuclide releases pass through the suppression pool. Because the pool has not been bypassed, the availability of the sprays is not very important and, thus, the CS characteristic has been dropped.	Total Clinton release mode frequency for: B1 1.13E-6 B2 8.16E-6 A1 7.81E-6	1.7E-5
7d	<u>Containment Failure Due to Accident (d):</u> CD, vessel breach, Late CF Vessel breach occurs, however, the containment does not fail until the late time period. If the containment did not fail early, it is unlikely that the drywell will fail early. Thus, the suppression pool bypass characteristic and the containment spray characteristic have been dropped.	Total Clinton release mode frequency for: D5 9.91E-9 D6 9.07E-7	9.2E-7

Table C.3-5b

SUMMARY OF CLINTON RELEASE FREQUENCIES AS A FUNCTION OF EPRI CATEGORY

(Case No 3: ILRT and DWBT Frequencies of 1/15 yrs)

EPRI Category	Category Description	Frequency Estimation Methodology	Frequency (1/yr)
8	<u>Containment Bypass Accidents</u>: Accident sequences in which the containment is bypassed. Such accidents are initiated by LOCAs outside containment (i.e., Break Outside Containment LOCA, or Interfacing Systems LOCA). Not affected by ILRT leak testing frequency.	[Total Clinton Containment Bypass release frequency]	1.2E-7
		TOTAL:	2.8E-5(1)

(1) Accurate to within a few percent of the total CDF (2.76E-5/yr). [18] Differences due to roundoff and EPRI calculational approach.

(2) Table C.3-2b provides the "combined" probability of the DW <u>and</u> WW leakage that needs to be accounted for in Category 3a and 3b.

(3) Reflective of a 1/15 year frequency for the ILRT.

Table C.3-5c

CLINTON DOSE RATE ESTIMATES AS A FUNCTION OF EPRI CATEGORY FOR POPULATION WITHIN 50-MILES

(Case No 3: DWBT and ILRT Frequency of 1/15 yrs))

EPRI Category	Category Description	Person-Rem Within 50 miles[6]	Category Frequency (per year)[7]	Dose Rate (Person-Rem/yr)
1	No Containment Failure[1]	2.4E+3	1.66E-6	3.98E-3
2	Containment Isolation System Failure[2]	5.1E+5	1.13E-7	5.76E-2
3a	Small Pre-Existing Failures[3]	2.4E+4	3.45E-6	8.28E-2
3b	Large Pre-Existing Failures[3]	8.4E+4[9]	3.59E-7[9]	3.15E-2[9]
4	Type B Failures (LLRT)	n/a	n/a	n/a
5	Type C Failures (LLRT)	n/a	n/a	n/a
6	Other Containment Isolation System Failure	n/a	n/a	n/a
7a	Containment Failure Due to Severe Accident (a)[4]	5.1E+5	2.63E-7	1.34E-1
7b	Containment Failure Due to Severe Accident (b)[4]	3.5E+5	4.7E-6	1.65
7c	Containment Failure Due to Severe Accident (c)[4]	3.7E+5	1.71E-5	6.33
7d	Containment Failure Due to Severe Accident (d)[4]	3.0E+5	9.2E-7	2.76E-1
8	Containment Bypass Accidents[5]	5.1E+5	1.21E-7	6.17E-2
	Total		2.86E-5[8]	8.63

Notes to Table C.3-5c

(1) The population dose associated with the Technical Specification Leakage is based on scaling the population data, the power level, and allowable Technical Specification leakage compared to the NUREG/CR-4551 reference plant. The release for this EPRI category is assigned from APB#7 from Table 3-4 of the main report.

(2) EPRI Category #2 (Containment Isolation failures) may include drywell isolation failures. Therefore, the release associated with this category is assigned to be equivalent to the release associated with APB#1 from Table 3-4 of the main report.

(3) Dose estimates for #3a and #3b, per the NEI Interim Guidance, are calculated as 10xCategory 1 dose and 35xCategory 1 dose, respectively.

(4) Dose estimate for 7a, 7b, 7c, and 7d are taken from APB # 1, 2, 4, and 5, respectively. (See main Report.)

(5) EPRI Category #8 sequences involve containment bypass failures; as a result, the person-rem dose is not based on normal containment leakage. The releases for this category are assumed to result in a direct path to the environment, and as such, are assigned to be equivalent to the highest release category from NUREG/CR-4551. APB#1 from Table 3-4 is therefore used of the main report.

(6) Table C.3-2.

(7) Table C.3-5b.

(8) Within a few percent of total CDF of 2.76E-5/yr [18]. Slight differences are due to the EPRI calculational approach and round off. The use of slightly higher frequencies in Table C.3-6 is conservative for assessing the risk metric of dose rate.

(9) Large pre-existing failure estimates of the population dose are based on using the extrapolated person rem for Case CC' (Table C.3-2) in the calculation of those contributors with 35DWLc and 35 La and adding these results to the other large pre-existing failures.

C.3.4 BASELINE POPULATION DOSE ESTIMATES (STEP 4)

The fourth step of the process (see Section C.2) is to estimate the baseline population dose (person-rem) for each EPRI category.

The NEI Interim Guidance recommends two options for calculating population dose for the EPRI categories:

- Use of NUREG-1150 dose calculations
- Use of plant-specific dose calculations

The NUREG-1150 [14] dose calculations were used in the EPRI TR-104285 study, as discussed previously in Section 2.1 of the main report. The use of generic dose information for NUREG-1150 is recommended by NEI to make the ILRT (and DWBT) risk assessment methodology more readily usable for plants that do not have a Level 3 PRA. As Clinton does not have a Level 3 PRA or associated plant-specific dose calculations, this ILRT risk assessment employs NUREG-1150 dose results calculated using the MACCS2 (MELCOR Accident Consequence Code System) consequence code; specifically, the doses for the Grand Gulf NUREG-1150 study (as documented in supporting report NUREG/CR-4551) are used. The following discussion summarizes the population dose calculation and results.

<u>Clinton Population Dose By EPRI Category</u>

As discussed in Section C.2, MAAP has been used to allow the consistent characterization of combinations of drywell and containment leakage cases for assignment to Categories 1, 3a, and 3b. Using this information and that developed from NUREG-1150 and Clinton and used in the ILRT assessment, the population dose for the 50 mile radius surrounding Clinton is summarized in Table C.3-6 by EPRI category. (Use of dose results for the 50 mile radius around the plant as a figure of merit in the risk evaluation is consistent with NUREG-1150, past ILRT frequency extension submittals, and the NEI Interim Guidance.)

As EPRI Categories 4, 5, and 6 are not affected by DWBT frequency, they are not analyzed as part of this risk assessment (per NEI Interim Guidance).

C.3.5 BASELINE POPULATION DOSE RATE ESTIMATES (STEP 5)

The baseline dose rates per EPRI accident category are calculated by multiplying the population dose estimates from Table C.3-6 by the frequencies summarized in Table C.3-3b. The resulting baseline population dose rates by EPRI category are summarized in Table C.3-3c.

Table C.3-6

CLINTON POPULATION DOSE ESTIMATES AS A FUNCTION OF EPRI CATEGORY WITHIN 50-MILE RADIUS

EPRI Category	Category Description	Person-Rem Within 50 miles
1	No Containment Failure	2.4E+03
2	Containment Isolation System Failure	5.1E+05
3a	Small Pre-Existing Failures	2.4E+04
3b	Large Pre-Existing Failures	8.4E+04
4	Type B Failures (LLRT)	n/a
5	Type C Failures (LLRT)	n/a
6	Other Containment Isolation System Failure	n/a
7a	Containment Failure Due to Severe Accident (a)	5.1E+5
7b	Containment Failure Due to Severe Accident (b)	3.5E+5
7c	Containment Failure Due to Severe Accident (c)	3.7E+5
7d	Containment Failure Due to Severe Accident (d)	3.0E+5
8	Containment Bypass Accidents	5.1E+05

C.3.6 IMPACT OF PROPOSED DWBT INTERVAL (STEPS 6-9)

Steps 6 through 9 of the NEI Interim Guidance assess the impact on plant risk due to the new and proposed ILRT and DWBT surveillance intervals (used here for the DWBT interval assessment) in the following ways:

- Determine change in probability of detectable leakage (Step 6)
- Determine population dose rate for new ILRT and DWBT intervals (Step 7)
- Determine change in dose rate due to new ILRT and DWBT intervals (Step 8)
- Determine change in LERF risk measure due to new ILRT and DWBT intervals (Step 9)
- Determine change in CCFP due to new ILRT and DWBT intervals (Step 10)

C.3.6.1 Change in Probability of Detectable Leakage (Step 6)

Step 6 of the NEI Interim Guidance for ILRT risk assessment, is the calculation of the change in probability of leakage detectable by DWBT (and associated re-calculation of the frequencies of the impacted EPRI categories). Note that with increases in the DWBT surveillance interval, the size of the postulated leak path and the associated leakage rates are assumed not to change; however, the probability of pre-existing leakage detectable by DWBT does increase.

Per the NEI Interim Guidance for the ILRT risk assessment, the calculation of the change in the probability of a pre-existing DWBT-detectable leakage is based on the relationship that relaxation of the DWBT interval results in increasing the average time that a pre-existing leak would exist undetected. Using the standby failure rate statistical model, the average time that a pre-existing drywell leak would exist undetected is one-half the surveillance interval. For example, if the DWBT frequency is 1-per-10 years, then the average time that a leak would be undetected is 60 months (surveillance interval of 120 months divided by 2). The impact on the leakage probability due to the DWBT interval extension is then calculated by applying a multiplier determined by the ratio of the average times of non-detection for the two DWBT interval cases.

As discussed earlier in Section C.3.1, the conditional probability of a pre-existing DWBT-detectable containment leakage is divided into two categories. The calculated pre-existing DWBT-detectable leakage probabilities are reflective of a 3-per-10 year DWBT frequency and are as follows:

- "Small" pre-existing leakage (EPRI Category 3a): 2.70E-2
- "Large" pre-existing leakage (EPRI Category 3b): 2.70E-3

Since the latter half of the 1990's, the Clinton plant has been operating under a 1-per-10 year ILRT and DWBT testing frequency consistent with the performance-based Option B of 10 CFR Part 50, Appendix J. [16] The baseline(1) leakage probabilities first need to be adjusted to reflect the currently allowed 1-per-10 year Clinton DWBT testing frequency, as follows:

- "Small" : 2.70E-2 x [(120 months/2) / (36 months/2)] = 9.00E-2
- "Large" : 2.70E-3 x [(120 months/2) / (36 months/2)] = 9.00E-3

Note that a nominal 36 month interval (i.e., as opposed to 40 months, 120/3) is used in the above adjustment calculation to reflect the 3-per-10 year DWBT frequency. This is consistent with operational practicalities and the NEI Interim Guidance.

Similarly, the pre-existing DWBT-detectable leakage probabilities for the 1-per-15 year DWBT frequency currently being pursued by Clinton (and the subject of this risk assessment) are calculated as follows:

- "Small" : 9.00E-2 x [(180 months/2) / (120 months/2)] = 1.35E-1
- "Large" : 9.00E-3 x [(180 months/2) / (120 months/2)] = 1.35E-2

(1) The baseline case uses data characteristic of the 3/10 year ILRT frequency of testing.

Given the above adjusted leakage probabilities, the impacted frequencies of the EPRI categories are summarized below in Table C.3-7.

Table C.3-7
SUMMARY OF EPRI CATEGORY FREQUENCIES
AS A FUNCTION OF ILRT AND DWBT INTERVAL

EPRI Category	EPRI Category Frequency as a Function of ILRT and DWBT Interval		
	Case No 1 Revised Baseline[1] (3-per-10 year)	Current (1-per-10 year)	Proposed (1-per-15 year)
1	4.78E-6	3.04E-6	1.66E-6
3a	6.29E-7	2.21E-6	3.45E-6
3b	5.90E-8	2.21E-7	3.59E-7

[1]The revised baseline case uses data characteristic of the 3/10 year ILRT and DWBT frequency of testing for characterizing the containment leakage for EPRI Categories 1, 3a, and 3b.

Note that, per the definition of the EPRI categories, only the frequencies of Categories 1, 3a, and 3b are impacted by changes in ILRT and DWBT testing frequencies.

C. 3.6.2 Population Dose Rate for New DWBT Interval (Step 7)

The dose rates per EPRI accident category as a function of ILRT and DWBT interval are summarized in Table C.3-8. Table C.3-8 is merely a compilation of the information developed in Tables C.3-3c, -4c, and -5c.

C.3.6.3 Change in Population Dose Rate Due to New DWBT Interval (Step 8)

As can be seen from the dose rate results summarized in Table C.3-8, the calculated total dose rate increases imperceptibly (0.47%) from the current Clinton 1-per-10 year DWBT interval amount of 8.59 person-rem/year to the proposed 1-per-15 year DWBT interval amount of 8.626 person-rem/year.

Table C.3-8

DOSE RATE ESTIMATES BY EPRI ACCIDENT CATEGORY FOR POPULATION WITHIN 50-MILES

EPRI Category	Category Description	Dose Rate as a Function of ILRT and DWBT (Person-Rem/Yr)		
		Revised Baseline (3-per-10 year) Case No. 1	Current (1-per-10 year) Case No. 2	Proposed (1-per-15 year) Case No. 3
1	No Containment Failure	1.15E-2	7.30E-3	3.98E-3
2	Containment Isolation System Failure	5.76E-2	5.76E-2	5.76E-2
3a	Small Pre-Existing Failures	1.51E-2	5.30E-2	8.28E-2
3b	Large Pre-Existing Failures	5.02E-3	1.93E-2	3.15E-2
4	Type B Failures (LLRT)	N/A	N/A	N/A
5	Type C Failures (LLRT)	N/A	N/A	N/A
6	Other Containment Isolation System Failure	N/A	N/A	N/A
7a	Containment Failure Due to Severe Accident (a)	1.34E-1	1.34E-1	1.34E-1
7b	Containment Failure Due to Severe Accident (b)	1.65	1.65	1.65
7c	Containment Failure Due to Severe Accident (c)	6.33	6.33	6.33
7d	Containment Failure Due to Severe Accident (d)	2.76E-1	2.76E-1	2.76E-1
8	Containment Bypass Accidents	6.17E-2	6.17E-2	6.17E-2
	TOTAL:	8.54	8.59	8.63

Per the NEI Interim Guidance, the change in percentage contribution to total dose rate attributable to EPRI Categories 3a and 3b is also investigated here. Using the results summarized in Table C.3-8, for the current Clinton 1-per-10 year DWBT interval, the percentage contribution to total dose rate from Categories 3a and 3b is shown to be very minor:

$$[(5.30E\text{-}2 + 1.93E\text{-}2) / 8.59] \times 100 = 0.842\%$$

For the proposed 1-per-15 year DWBT interval, the percentage contribution to total dose rate from Categories 3a and 3b increases slightly but remains very minor:

$$[(8.28E\text{-}2 + 3.15E\text{-}2) / 8.63] \times 100 = 1.32\%$$

C.3.6.4 Change in LERF Due to Proposed ILRT and DWBT Interval (Step 9)

The risk increase associated with extending the ILRT and DWBT interval involves the potential that a core damage event that normally would not result in a radionuclide release from an intact containment could in fact result in a release due to the increase in probability of failure to detect a pre-existing leak. Per the NEI Interim Guidance, Category 3b sequences have been designated to have the potential to result in large releases if a pre-existing leak were present. Using the NEI/EPRI methodology, the change in LERF (Large Early Release Frequency) is determined by the change in the frequency of Category 3b.

Category 1 accidents are not considered as potential large release pathways because the containment remains intact. Therefore, the containment leak rate is expected to be small. Similarly, Category 3a is a "small" pre-existing leak. Other accident categories such as 2, 6, 7, and 8 could result in large releases but these are not affected by the change in ILRT or DWBT interval. Late releases are excluded regardless of the size of the leak because late releases are, by definition, not LERF contributors.

C.3.6.4.1. LERF Change Due to ILRT and DWBT Interval Change

The impact on the LERF risk measure due to the proposed ILRT and DWBT interval extension is calculated as follows:

delta LERF = Frequency of EPRI Category 3b for 1-per-15 year ILRT and DWBT interval) – (Frequency of EPRI Category 3b for 1-per-10 year ILRT and DWBT interval)

= 3.59E-7/yr – 2.21E-7/yr

= 1.4E-7/yr

The change in LERF due to the combined change in test intervals of ILRT and DWBT is 1.4E-7/yr which is right on the border line between "very small" (Region III) and "small" (Region II) risk change using the Reg. Guide 1.174 acceptance criteria.

At this point, it is reemphasized that the radionuclide release calculated for Class 3b is significantly below that which has been attributed to LERF releases. [C-25] Therefore, the NEI/EPRI characterization of Category 3b as a LERF contributor is considered extremely conservative.

C.3.6.5 <u>Impact on Conditional Containment Failure Probability (Step 10)</u>

Another parameter that the NRC Guidance in Reg. Guide 1.174 states can provide input into the decision-making process is the consideration of change in the conditional containment failure probability (CCFP). The change in CCFP is indicative of the effect of the ILRT and DWBT on all radionuclide releases, not just LERF. The conditional containment failure probability (CCFP) can be calculated from the risk calculations performed in this analysis. In this assessment, based on the NEI Interim Guidance, CCFP is defined such that containment failure includes all radionuclide release end states other than the intact state (EPRI Category 1) and small failures (EPRI Category 3a). The conditional part of the definition is conditional given a severe accident (i.e., core damage).

Consequently, the change in CCFP can be calculated by the following equation:

$$CCFP_{\%} = [1 - (\text{Intact Containment Frequency} / \text{Total CDF})] \times 100\%, \text{ or}$$
$$= [1 - ((\#1 \text{ Frequency} + \#3a \text{ Frequency}) / \text{CDF})] \times 100\%$$

For the DWBT and ILRT 10-year interval:

$$CCFP_{10}^{(3)} = [1 - ((3.04E\text{-}6 + 2.21E\text{-}6) / 2.76E\text{-}5)] \times 100\%$$
$$= 81.0\%$$

For a DWBT and ILRT 15-year interval:

$$CCFP_{15}^{(1)} = [1 - ((1.66E\text{-}6 + 3.45E\text{-}6) / 2.76E\text{-}5)] \times 100\%$$
$$= 81.5\%$$

The change in the conditional containment failure probability when both ILRT and DWBT intervals are extended from 10 to 15 years is:

$$\Delta CCFP = CCFP_{15}^{(1)} - CCFP_{10}^{(3)} = 0.5\%$$

This change in CCFP of less than 1% is insignificant from a risk perspective.

C.4 RESULTS SUMMARY

A one-time DWBT interval extension is also requested to be consistent with the change in the ILRT interval extension from 10 years to 15 years. Therefore, the incremental assessment of the risk change is performed for the case in which both the ILRT and DWBT intervals are extended from 10 years to 15 years.

The application of the approach based on NEI Interim Guidance [C-3, C-21], EPRI-TR-104285 [C-2] and previous risk assessment submittals on this subject [C-6, C-20, C-22] have led to the quantitative results summarized in this section. The results demonstrate a small impact on risk associated with the one time extension of the ILRT and DWBT interval to 15 years. In addition, the DWBT interval extension by itself represents a very small impact on risk.

The analysis performed examined Clinton specific accident sequences in which the containment remains intact or the containment is impaired. The accidents are analyzed and the results are displayed according to the eight (8) EPRI accident categories defined in Reference [2]:

1. Containment intact and isolated
2. Containment isolation failures due to support system or active failures
3. Type A (Combined ILRT and DWBT) related containment isolation failures
4. Type B (LLRT) related containment isolation failures
5. Type C (LLRT) related containment isolation failures
6. Other penetration related containment isolation failures
7. Containment failure due to core damage accident phenomena
8. Containment bypass

The quantitative results are summarized in Table C.4-1. The key results to this risk assessment are those for the ten year interval (current Clinton condition) and the fifteen year interval (proposed change). The 3-per-10 year DWBT is a baseline starting point for this risk assessment given that the pre-existing containment leakage probabilities (estimated based on industry experience - - refer to Section C.3.1) are reflective of the 3-per-10 year DWBT testing.

The following is a brief summary of some of the key aspects of the DWBT interval extension risk analysis:

- Increasing the current 10 year ILRT and DWBT interval to 15 years results in an insignificant increase in total population dose rate of 0.48 percentage points.
- The increase in the LERF risk measure is small, a 1.4E-7/yr increase. This LERF increase is categorized as right on the border between Region III and Region II per NRC Reg. Guide 1.174.
- Likewise, the conditional containment failure probability ($CCFP_{\%}$) increases insignificantly by 0.5 percentage points.

Table C.4-1

QUANTITATIVE RESULTS AS A FUNCTION OF ILRT AND DWBT INTERVAL

EPRI Category	Dose (Person-Rem Within 50 miles)	Quantitative Results as a Function of ILRT and DWBT Interval					
		Baseline (3-per-10 year)		Current (1-per-10 year)		Proposed (1-per-15 year)	
		Accident Frequency (per year)	Population Dose Rate (Person-Rem/Year Within 50 miles)	Accident Frequency (per year)	Population Dose Rate (Person-Rem/Year Within 50 miles)	Accident Frequency (per year)	Population Dose Rate (Person-Rem/Year Within 50 miles)
1	2.4 E+3	4.78E-6	1.15E-2	3.04E-6	7.3E-3	1.66E-6	3.98E-3
2	5.1E+5	1.13E-7	5.76E-2	1.13E-7	5.76E-2	1.13E-7	5.76E-2
3a	2.4E+4	6.29E-7	1.51E-2	2.21E-6	5.30E-2	3.45E-6	8.28E-2
3b	8.4E+4	3.90E-8	5.02E-3	2.21E-7	1.93E-2	3.59E-7	3.15E-2
4	N/A	N/A	N/A	N/A	N/A	N/A	N/A
5	N/A	N/A	N/A	N/A	N/A	N/A	N/A
6	N/A	N/A	N/A	N/A	N/A	N/A	N/A
7a	5.1E+5	2.63E-7	1.34E-1	2.63E-7	1.34E-1	2.63E-7	1.34E-1
7b	3.5E+5	4.7E-6	1.65	4.7E-6	1.65	4.7E-6	1.65
7c	3.7E+5	1.71E-5	6.33	1.71E-5	6.33	1.71E-5	6.33
7d	3.0E+5	9.2E-7	2.76E-1	9.2E-7	2.76E-1	9.2E-7	2.76E-1
8	5.1E+5	1.21E-7	6.17E-2	1.21E-7	6.17E-2	1.21E-7	6.17E-2
TOTALS:		2.87E-5[4]	8.54	2.87E-5[4]	8.59	2.87E-5[4]	8.63
Increase in Dose Rate [1]					0.58%		0.48%
Increase in LERF [2]				1.62E-7		1.4E-7	
Increase in CCFP (%) [3]				< 1%		0.5%	

Notes to Table C.4-1:

(1) The increase in dose rate (person-rem/year) is with respect to the results for the preceding ILRT and DWBT interval, as presented in the table. For example, the increase in dose rate for the proposed 1-per-15 ILRT and DWBT is calculated as: total dose rate for 1-per-15 year ILRT and DWBT, minus total dose rate for 1-per-10 year ILRT and DWBT. For each case, the dose rate increase is insignificant.

(2) The increase in Large Early Release Frequency (LERF) is with respect to the results for the preceding ILRT and DWBT interval, as presented in the table. As discussed in Section C.3.4.4 of the report, the change in LERF is determined by the change in the accident frequency of EPRI Category 3b. For example, the increase in LERF for the proposed 1-per-15 ILRT and DWBT is calculated as: 3b frequency for 1-per-15 year ILRT and DWBT.

(3) The increase in the conditional containment failure probability (CCFP) is with respect to the results for the preceding ILRT and DWBT interval, as presented in the table. As discussed in Section C.3.4.5, the conditional containment failure probability (CCFP) is calculated as:

$$CCFP_{\%} = [1 - ((\text{Category \#1 Frequency} + \text{Category \#3a Frequency}) / \text{CDF})] \times 100\%$$

(4) Due to the NEI Methodology and round off, the total frequency of all severe accidents is slightly higher than the CPS Rev 3 reported CDF (approximately 4%). This in turn leads to slightly higher population dose rate estimates for the Baseline, the current, and the proposed ILRT frequencies.

C.5 CONCLUSIONS

This appendix summarizes the combined risk increase associated with the change in both the ILRT and DWBT intervals. This combination of changes results in a very slight increase in the risk measures relative to those calculated in the main report for the ILRT interval change by itself.

C.5.1 QUANTITATIVE CONCLUSIONS

The conclusions from the risk assessment of the one-time ILRT and DWBT interval extension can be characterized by the risk metrics used in previously approved ILRT interval extensions for other plants. These include:

- Change in LERF
- Change in conditional containment failure probability
- Change in population dose rate

C.5.1.1 LERF

Based on the results from Sections 3 and 4, the main conclusion regarding the impact on plant risk associated with extending the ILRT and DWBT interval from ten years to fifteen years is:

> Reg. Guide 1.174 [4] provides guidance for determining the risk impact of plant-specific changes to the licensing basis. Reg. Guide 1.174 defines very small changes in risk as resulting in increases of CDF below 10^{-6}/yr and increases in LERF below 10^{-7}/yr. Since the ILRT and DWBT do not impact CDF, the relevant criterion is LERF. The increase in LERF resulting from a change in the ILRT and DWBT interval from ten years to fifteen years (using the change in the EPRI Category 3b frequency per the NEI Interim Guidance) is 1.4E-7/yr. Guidance in Reg. Guide 1.174 defines small changes in LERF as above 10^{-7}/yr and less than 10^{-6}/yr. Therefore, increasing the Clinton ILRT and DWBT interval from 10 to 15 years results in a small change in risk, and is an acceptable plant change from a risk perspective.

Per Reg. Guide 1.174, when the calculated increase in LERF due to the proposed plant change is in the range of 1E-7 to 1E-6 per reactor year (Region II, "small change" in risk), the risk assessment must also reasonably show that the total LERF is less than 1E-5.

Per the Clinton internal events PSA (Rev. 3) documentation, the Clinton LERF due to internal event accidents is 2.63E-7/yr. Therefore, the total LERF for Clinton of 2.63E-7/yr is significantly less than the Reg. Guide 1.174 acceptance guideline of 1E-5/yr.

It is emphasized that the radionuclide release (e.g., CsI release fraction) calculated for Class 3b is significantly below that which has been attributed to LERF releases. [C-25] Therefore, the NEI/EPRI characterization of Category 3b as a LERF contributor is considered extremely conservative for a Mark III.

C.5.1.2 CCFP

The change in conditional containment failure probability (CCFP) is also calculated as an additional risk measure to demonstrate the impact on defense-in-depth. The ΔCCFP is found to be very small (0.5% increase) and represents a negligible change in the Clinton defense-in-depth.

C.5.1.3 Population Dose Rate

The change in population dose rate is also reported consistent with previously approved ILRT interval extension requests. The change in population dose rate from the current 1/10 year ILRT and DWBT frequency to 1/15 year frequency is an insignificant 0.48% increase.

C.5.2 DWBT INTERVAL EFFECT

Appendix C.5.1 summarizes the combined risk effect of changing both the ILRT and DWBT interval. This appendix isolates the risk impacts associated with just the DWBT interval change.

By using the ILRT interval assessment in the main report, the incremental risk increase due to the DWBT interval extension can be isolated for each of the risk metrics. These are as follows:

	RISK METRICS DUE TO EXTENDING 10 YEAR INTERVAL TO A 15 YEAR INTERVAL		
	ILRT Only[1]	ILRT and DWBT[2]	Risk Metric Increase Due to DWBT Interval Extension[3]
Change in Population Dose Rate (Person Rem/yr)	0.03	0.04	0.01
LERF (per RX Yr)	9.3E-8	1.4E-7	4.7E-8
CCDP	0.3%	0.5%	0.2%

[1] Calculated in main Report.
[2] Calculated in Appendix C.4.
[3] Calculated by subtracting ILRT only column from the ILRT and DWBT column.

This shows that the DWBT interval extension effects on the risk metrics are smaller than the ILRT interval extension effects and that their combination with the ILRT is also a small risk effect.

C.5.3 EXTERNAL EVENTS IMPACT

External hazards were evaluated in the Clinton Individual Plant Examination of External Events (IPEEE) Submittal in response to the NRC IPEEE Program (Generic Letter 88-20 Supplement 4). The IPEEE Program was a one-time review of external hazard risk to identify potential plant vulnerabilities and to understand severe accident risks. Clinton does not currently maintain external event PSA models and associated documentation. Although the external event hazards in the Clinton IPEEE were evaluated to varying levels of conservatism, the results of the Clinton IPEEE are nonetheless used in this risk assessment to provide a conservative comparison of the impact of external hazards on the conclusions of this ILRT and DWBT interval extension risk assessment.

Given the characteristics of the proposed plant change (i.e., combined ILRT and DWBT interval extension), specific quantitative information regarding the impact on external event hazard risk measures is not judged to be a significant decision making input. The proposed ILRT and DWBT interval extension impacts plant risk in a very specific and limited way. The probability of a pre-existing suppression pool bypass given a core damage accident is potentially higher when the DWBT interval is extended. This impact is manifested in the plant risk profile in a similar manner for both internal events and external events.

The spectrum of external hazards has been evaluated in the Clinton IPEEE by screening methods with varying levels of conservatism. Therefore, it is not possible at this time to incorporate realistic quantitative risk assessments of all external event hazards into the ILRT and DWBT extension assessment.

While the external events impact on the ILRT and DWBT interval extension is not explicitly quantified, Appendix B has provided a sensitivity case for the ILRT interval extension to show that the LERF change is within the Reg. Guide 1.174 Region II for "small changes in risk." This analysis is considered to approximately characterize the ILRT and DWBT interval extension also.

C.5.4 SUMMARY

The findings for Clinton confirm that the risk change associated with extending the ILRT and DWBT interval from 10 years to 15 years is small when considering (1) Clinton severe accident risk profile, (2) the Clinton containment failure modes, and (3) the local population surrounding the Clinton site.

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