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NL-04-0563

U. S. Nuclear Regulatory Commission
ATTN: Document Control Desk
Washington, D. C. 20555-0001

Edwin I. Hatch Nuclear Plant Unit 2
Technical Specification Revision Request
Integrated Leakage Rate Testing Interval Extension

Ladies and Gentlemen:

In accordance with the provisions of 10 CFR 50.90, Southern Nuclear Operating Company (SNC) is proposing a change to the Edwin I. Hatch Nuclear Plant (HNP) Unit 2 Technical Specifications (TS). This proposed change will revise TS section 5.5.12, ("Primary Containment Leakage Rate Testing Program") to reflect a one-time deferral of the Type A Containment Integrated Leak Rate Test (ILRT). The ten (10) year interval between integrated leakage rate tests is to be extended to fifteen (15) years from the previous integrated leakage rate test, which was completed on November 2, 1995. This proposed change is based on and has been evaluated using the "risk informed" guidance in Regulatory Guide 1.174, "An Approach for Using Probabilistic Risk Assessment in Risk-Informed Decisions on Plant-Specific Changes to the Licensing Basis."

The "Risk Assessment for Edwin I. Hatch Nuclear Power Station Regarding ILRT (Type A) Extension Request" is provided as an attachment to this letter. This risk assessment is based on the Hatch Unit 1 Level 1 and Level 2 internal events PRA model and is judged to provide representative results for Hatch Unit 2.

Enclosure 1 provides a description of the proposed change and an explanation of the basis for the change. Also contained in Enclosure 1 is a list of typical NRC questions and the SNC response to those questions. Enclosure 2 details the basis for SNC's determination that the proposed change does not involve a significant hazards consideration. Enclosure 3 provides the revised Technical Specification page and the corresponding marked-up page.

Southern Nuclear Operating Company requests the proposed amendment be approved by January 3, 2005 to support the planning activities for the Unit 2 outage scheduled in February 2005.

AD17

A similar request was approved for Vogtle Units 1 and 2 in a letter dated January 12, 2004, Clinton Power Station Unit 1 in a letter dated January 8, 2004, Lasalle Units 1 and 2 in a letter dated November 19, 2003, and Hatch Unit 1 in a letter dated February 20, 2002.

Mr. H. L. Sumner, Jr. states he is a Vice President of Southern Nuclear Operating Company, is authorized to execute this oath on behalf of Southern Nuclear Operating Company and to the best of his knowledge and belief, the facts set forth in this letter are true.

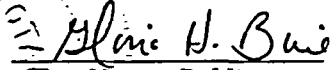
This letter contains no NRC commitments. If you have any questions, please advise.

Respectfully submitted,

SOUTHERN NUCLEAR OPERATING COMPANY


H. L. Sumner, Jr.

Sworn to and subscribed before me this 26 day of April, 2004.


Notary Public

My commission expires: 6-7-05

HLS/whc/daj

Enclosures: 1 – Basis for Change Request
2 – 10 CFR 50.92 Evaluation
3 – Marked up and Clean Typed TS Pages

Attachment: Risk Assessment for Edwin I. Hatch Nuclear Power Station Regarding
ILRT (Type A) Extension Request

cc: Southern Nuclear Operating Company
Mr. J. B. Beasley, Jr., Executive Vice President
Mr. G. R. Frederick, General Manager – Plant Hatch
RType: CHA02.004

U. S. Nuclear Regulatory Commission
Mr. L. A. Reyes, Regional Administrator
Mr. C. Gratton, NRR Project Manager – Hatch
Mr. D. S. Simpkins, Senior Resident Inspector – Hatch

State of Georgia
Mr. L. C. Barrett, Commissioner – Department of Natural Resources

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Enclosure 1

Basis for Change Request

and

NRC Questions with SNC Response

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Enclosure 1

Basis for Change Request

Proposed Change

Southern Nuclear Operating Company (SNC) is proposing a change to the Hatch Nuclear Plant (HNP) Unit 2 Technical Specifications (TS). This proposed change will revise TS section 5.5.12, "Primary Containment Leakage Rate Testing Program," to reflect a one-time deferral of the Type A Containment Integrated Leak Rate Test (ILRT). The ten (10) year interval between integrated leakage rate tests is to be extended to fifteen (15) years from the previous integrated leakage rate test, which was completed on November 2, 1995.

The proposed change involves a one-time exception to the ten (10) year frequency of the performance-based leakage rate testing program for Type A tests as required by Nuclear Energy Institute (NEI) 94-01, Revision 0, "Industry Guidelines for Implementing Performance-Based Option of 10 CFR 50, Appendix J." The current ten (10) year ILRT for HNP Unit 2 is due on November 2, 2005, which would require the test to be performed during Refueling Outage 2RF18. The proposed exception would allow the next ILRT for HNP Unit 2 to be performed within fifteen (15) years from the last ILRT as opposed to the current ten (10) year frequency.

The proposed change would revise Section 5.5.12, "Primary Containment Leakage Rate Testing Program" of the Hatch Unit 2 Technical Specifications to add the following statement:

... , as modified by the following exception to NEI 94-01, Rev. 0, "Industry Guideline for Implementing Performance-Based Option of 10 CFR 50, Appendix J":

Section 9.2.3: The first Type A test after the November 2, 1995 Type A test shall be performed no later than November, 2010.

This one-time exception will result in the following:

- For Unit 2, the Type A Containment ILRT will be performed during Refueling Outage 2RF20, currently scheduled for Spring 2009.
- A substantial cost savings will be realized, and unnecessary personnel radiation exposure will be avoided by deferring the Type A test for an additional five (5) years. Cost savings have been estimated at approximately \$1.95 million, which includes labor, equipment, and critical path outage time needed to perform the test. Personnel radiation exposure reduction is estimated at 750 mrem.

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Basis for Proposed Change

a. 10 CFR 50, Appendix J, Option B

The testing requirements of 10 CFR 50, Appendix J, provide assurance that leakage from the containment, including systems and components that penetrate the containment, does not exceed the allowable leakage values specified in Technical Specifications. The limitation on containment leakage provides assurance that the containment will perform its design function following plant design basis accidents.

10 CFR 50, Appendix J was revised, effective October 26, 1995, to allow licensees to perform containment leakage testing in accordance with the requirements of Option A, "Prescriptive Requirements," or Option B, "Performance-Based Requirements." Amendment No. 141 of the Hatch Unit 2 TS was issued March 6, 1996, to reflect the adoption of the requirements of 10 CFR Part 50, Appendix J, Option B. These amendments revised Technical Specifications to require Type A, B, and C testing in accordance with Regulatory Guide (RG) 1.163, "Performance-Based Containment Leak-Test Program." RG 1.163 specifies a method acceptable to the NRC for complying with 10 CFR 50, Appendix J, Option B by approving the use of NEI 94-01 and ANSI/ANS 56.8-1994, subject to several regulatory positions in the guide.

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

Adoption of the Option B performance-based containment leakage rate testing program did not alter the basic method by which Appendix J leakage rate testing is performed; however, it did alter the frequency at which Type A, B, and C containment leakage tests must be performed. Under the performance-based option of 10 CFR 50, Appendix J, test frequency is based upon an evaluation that reviews "as found" leakage and maintenance history to determine the frequency for leakage testing which provides assurance that leakage limits will be maintained.

The allowed frequency for Type A testing, as documented in NEI 94-01, is based, in part, upon a generic evaluation documented in NUREG-1493. The evaluation documented in NUREG-1493 included a study of the dependence of reactor accident risks on containment leak-tightness for five reactor/containment types including a GE designed boiling water reactor in Mark I containment. (HNP Unit 2 is a Mark I containment). NUREG-1493 made the following observations with regard to decreasing the test frequency.

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Basis for Proposed Change (continued)

- Reducing the Type A Integrated Leak Rate Test (ILRT) testing frequency to one per twenty (20) years was found to lead to imperceptible increase in risk. The estimated increase in risk is small because ILRTs 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 been only marginally above the existing requirements. Given the insensitivity of risk to containment leakage rate, and the small fraction of leakage detected solely by Type A testing, increasing the interval between ILRT testing has minimal impact on public risk.
- While Type B and C tests identify the vast majority (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.

NEI 94-01 requires that Type A testing be performed at least once per ten (10) years based upon an acceptable performance history. Acceptable performance history is defined as two consecutive periodic Type A tests at least 24 months apart where the calculated performance leakage rate was less than $1.0 L_a$. Based upon the acceptable ILRTs for Unit 2 (November 7, 1992 and November 2, 1995), the current test interval for HNP Unit 2 is once every ten (10) years, with the next test due to be performed by November 2, 2005.

b. HNP Integrated Leak Rate Test History

Type A testing is performed to verify the integrity of the containment structure in its Loss of Coolant Accident (LOCA) configuration. Industry test experience has demonstrated that Type B and C testing detect a large percentage of containment leakage and that the percentage of containment leakage that is detected only by integrated containment leakage testing is very small.

HNP Unit 2 has undergone five (5) operational Type A tests. The results of these tests demonstrate that the HNP Unit 2 containment structure remains an essentially leak-tight barrier and represents minimal risk to increased leakage. These plant-specific results support the conclusions of NUREG-1493. As specified in Hatch Technical Specifications Section 5.5.12, the maximum allowable containment leakage rate L_a , at P_a , is 1.2% of primary containment air weight per day. The HNP Unit 2 ILRT results are provided below.

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Refueling Outage (95% Upper Confidence Limit % Cnmt Air
Mass/Day)(Total Time Analysis)

May 2, 1982	.07890
November 20, 1986	0.5870
November 13, 1989	0.8000
November 7, 1992	0.8839
November 2, 1995	0.3175

c. Plant Operational Performance

HNP Unit 2 is a GE designed boiling water reactor in a Mark I containment. During power operation the primary containment atmosphere is inerted with nitrogen to ensure that no external sources of oxygen are introduced into containment. The containment inerting system is used during the initial purging of the primary containment prior to power operation and provides a supply of makeup nitrogen to maintain primary containment oxygen concentration within Technical Specification limits. As a result, the primary containment is maintained at a slightly positive pressure during power operation. Primary containment pressure is continuously recorded and verified by TS surveillance on a frequency of every 12 hours from the Main Control Room. Although this feature, that is inherent to the HNP BWR containment design, does not challenge the structural and leak tight integrity of the containment system at post-accident pressure, the fact that the containment is continuously pressurized by the containment inerting system, and is periodically monitored, provides assurance that gross containment leakage that may develop during power operation will be detected.

d. Containment Inspections

Containment leak tight integrity is also verified through periodic inservice inspections conducted in accordance with the requirements of the 1992 Edition through the 1992 Addenda of American Society of Mechanical Engineers Boiler and Pressure Vessel Code (ASME Code) Section XI. More specifically, subsection IWE provides the rules and requirements for inservice inspection of Class MC pressure retaining components and their integral attachments. Furthermore, NRC regulations, 10 CFR 50.55a(b)(2)(ix)(E), require licensees to conduct visual inspections of the accessible areas on the interior of the containment three 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 not affected by the change to the Type A test frequency. Likewise the Appendix J, Type C local leak tests, which are performed to verify the leak tight integrity of containment isolation valves, are not affected by the change to the Type A test frequency.

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e. Typical Questions

The NRC has sent Requests for Additional Information (RAI) to several licensees concerning their request for a technical specification revision allowing a one-time ILRT interval extension. These RAIs contain typical questions. Listed below are the typical questions with the SNC responses:

1. **NRC Question**

Since there is no description (or summarization) regarding the containment ISI program being implemented at HNP, please provide a description of the ISI methods that provide assurance that in the absence of an ILRT for 15 years, the containment structural and leak tight integrity will be maintained.

SNC Response

As described in Section d above, containment leak tight integrity is also verified through periodic inservice inspections conducted in accordance with the requirements of the 1992 Edition through the 1992 Addenda of ASME Code Section XI.

The ASME Code Section XI IWE containment inspections provide a high degree of assurance that any degradation of the containment structure is identified and corrected before a containment leakage path is introduced.

2. **NRC Question**

IWE-1240 requires licensees to identify the containment surface areas requiring augmented examinations. Please provide the locations of the containment liner surfaces that have been identified as requiring augmented examination and a summary of the findings of the examinations performed.

SNC Response

There are no areas of the Hatch Unit 2 containment liners that require augmented examinations per IWE-1240. General Visual examination of the entire containment structure has not identified any areas that are subject to the augmented examination requirements of IWE.

3. **NRC Question**

For the examination of seals and gaskets, and examination and testing of bolted connections associated with the primary containment pressure boundary (Examination Categories E-D and E-G), relief from the requirements of the Code had been requested. As an alternative, it was proposed to examine them during the leak rate testing of the primary containment. However, Option B of Appendix J for Type B and Type C testing (as per Nuclear Energy Institute 94-01 and

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Regulatory Guide 1.163) and the ILRT extension requested in this amendment for Type A testing provide flexibility in the scheduling of these inspections. Please provide your schedule for examination and testing of seals, gaskets, and bolts that provide assurance regarding the integrity of the containment pressure boundary.

SNC Response

Relief Request RR-MC-1 (Seals (including O-rings) and gaskets of Class MC pressure retaining components, Examination Category E-D, Item Numbers E5.10 and E5.20) was approved February 11, 2000. Leak-tightness of the seals and gaskets will be confirmed in accordance with 10 CFR 50 Appendix J. If a seal or gasket is replaced, it will be visually inspected by maintenance personnel before reassembly or closure. Also, an as-left Appendix J leakage test will be performed after installation to ensure leak tightness.

Relief Request RR-MC-6 (Class MC pressure retaining bolting requiring visual examination (VT-1) per Category E-G, Item E8.10, in accordance with Subarticle IWE-3515.1) was approved February 11, 2000. Bolting material will be examined in accordance with the inservice standards of the 1992 Edition, with 1992 Addenda of ASME Section XI, Subarticle IWB-3517.1 Standards for Examination Category B-G-1, Pressure Retaining Bolting Greater than 2 in. in Diameter, and Examination Category B-G-2, Pressure Retaining Bolting 2 in. and Less in Diameter.

Relief Request RR-MC-8 (Class MC pressure retaining bolting for bolted connections that have not been disassembled and reassembled during the inspection interval) was approved October 4, 2000. ASME Code Case N-604 will be used for alternate examination of pressure retaining bolting in lieu of torque or tension testing.

The one-time extension requested by SNC applies only to the 10 CFR 50, Appendix J, Type A integrated leak rate test that is currently on a 10-year interval pursuant to Appendix J, Option B, Performance Based Requirements. Appendix J, Type B and Type C tests are performed at the intervals required by Appendix J, Option B and will be tested at least once in the 10-year interval. This frequency of testing of seals, gaskets, and containment pressure retaining bolting provides reasonable assurance that the integrity of the containment pressure boundary is maintained during the period of the extension.

4. NRC Question

The stainless steel bellows have been found to be susceptible to trans-granular stress corrosion cracking and the leakage through them is not readily detectable by Type B testing (see Information Notice 92-20). If applicable, please provide information regarding inspection and testing of the bellows, and how such behavior has been factored into the risk assessment.

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SNC Response

NRC Information Notice 92-20, Inadequate Local Leak Rate Testing, discussed the inadequate local leak rate testing of two-ply stainless steel bellows. Stainless steel expansion bellows are typically covered by a guard plate which encloses the bellows and is welded to the penetration assembly. The guard plate must be removed in order to perform any meaningful examinations of the circumferential and longitudinal welds in the bellows assembly. Removing the guard plate poses the risk of damaging the bellows assembly which is not warranted just to perform examinations. Experience indicates that conventional examination techniques are not adequate to identify defects in the bellows and presently, Appendix J testing is the only practical test method currently being performed. We are presently monitoring on-going industry activities concerning this potential problem area and intend to remain proactive as developments unfold.

All bellows are Type B tested in accordance with 10 CFR 50 Appendix J, Option B. All bellows are also tested during the Appendix J, Type A test.

The risk submittal refers to the Type B category of test results which show excessive leakage but are detected by the test as Class 4. The Type A test extension does not affect the frequency of the Type B tests and as a result these are not considered in this analysis. Class 3 leakage, however, as referenced in the risk submittal, takes into account the probabilistic occurrence of small and large drywell liner or bellows leaks which may exist at the time of postulated core damage, possibly not detected by Type B testing. The leakage rates range from two times Tech Spec leakage (2La) to 35La for small leaks and greater than 35La for large leaks. In order to perform the calculations a value of 10La was used for small leakage (Class 3A in the risk submittal) and 35La for larger leakage referenced as Class 3B. This information is used with the core damage frequency to obtain accident frequency of occurrence.

5. NRC Question

Inspections of some reinforced concrete and steel containment structures have found degradation on the uninspectable (embedded) side of the drywell steel shell and steel liner of the primary containment. These degradations cannot be found by visual (i.e., VT-1 or VT-3) examinations unless they are through the thickness of the shell or liner, or 100% of the uninspectable surfaces are periodically examined by ultrasonic testing. Please provide information (additional analyses) addressing how potential leakage under high pressure during core damage accidents is factored into the risk assessment related to the extension of the ILRT.

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SNC Response

The attached "Risk Assessment for Edwin I. Hatch Nuclear Power Station Regarding ILRT (Type A) Extension Request" provides a sensitivity evaluation considering potential corrosion impacts within the framework of the ILRT interval extension risk assessment. The analysis confirms that the ILRT interval extension has a minimal impact on plant risk. Additionally, a series of parametric sensitivity studies regarding the potential age-related corrosion effects on the steel liner also indicate that even with very conservative assumptions, the conclusions from the original analysis would not change. That is, the ILRT interval extension is judged to have a minimal impact on plant risk and is therefore acceptable.

The attached analysis also clarifies the delta LERF for the original License Bases "three tests in 10 years" and the proposed "one test in 15 years." The analysis also provides a discussion on the effects ILRT interval extension would have on the total LERF (internal and external events) for Hatch. The conclusion shows that the total LERF for Hatch is well below the RG 1.174 acceptance criteria for total LERF of 1.0E-05.

Additionally, the drywell containment at HNP has a 2" air gap between the steel shell and concrete shield wall. The design includes drain lines (4), at the basemat elevation, which route any leakage into the air gap away from the drywell shell. SNC performed examination of the drain lines using a video probe to confirm that the drains were open and functional (unlike the drain lines at another plant which resulted in water accumulation). SNC performs visual examinations of the drains lines each outage when the refueling cavity is flooded to look for evidence of moisture or leakage. These examinations are performed to ensure there is no leakage from the refueling cavity bellows that would support corrosion.

6. NRC Question

Southern Nuclear Operating Company states that containment leak tight integrity is verified through periodic inservice inspections (ISI) conducted in accordance with the requirements of the 1992 Edition through the 1992 Addenda of the ASME Code Section XI. Provide a detailed summary of ISI and related containment testing activities including inspection/testing dates, findings, corrective actions, and maintenance/repair as well as containment modifications that may or may not be a result of the required ISIs.

SNC Response

In compliance with the rulemaking actions which revised 10CFR50.55a to invoke the requirements of the ASME Boiler and Pressure Vessel Code, Section XI, Subsections IWE and IWL (61FR41303; August 8, 1996 and 64FR51370; September 22, 1999), SNC has performed examinations of the Unit 2 containment in accordance with the 1992 Edition through 1992 Addenda of the ASME Boiler and Pressure Vessel Code as well as other supplemental exams.

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Containment Inspection History Summary Table

Outage / Date	Significant Inspection Activities
1994 - 1997	Implemented the BWROG Model Containment Inspection Program. The Program included visual examinations of the accessible Drywell interior surfaces, Torus interior and exterior surfaces, and Vent Header piping. Torus submerged surfaces were inspected by divers, grids were established for monitoring, and spot repairs of submerged surface coatings were performed. Drain lines from the drywell air gap region and sand pocket were examined to confirm functionality. Ultrasonic thickness exams of select locations of the Drywell shell were performed. No significant pressure boundary degradation was identified by any of these exams.
2R14 Fall 1998	E-A: VT-3 of Torus and Vent Lines Submerged Surfaces. Minor coating degradation and shell pitting observed. No repairs required. Supplemental: UT Thickness exams of select locations of the Torus shell. No significant pressure boundary degradation identified.
2R16 Fall 2001	E-A: General Visual of accessible Drywell, Torus, and Vent Header accessible interior and exterior surfaces. Minor coating degradation and shell pitting observed. Spot repairs of coatings were performed. E-D: VT-3 of Concrete Floor to Interior Drywell Shell Mastic Seal. Partially replaced seal. Supplemental: Torus submerged surface desludge and coatings inspection. Minor coating degradation and shell pitting observed. No repairs required. Supplemental: UT Thickness exams of select locations of the Torus shell. No significant pressure boundary degradation identified. E-G: VT-1 of bolting for any pressure boundary penetration that was disassembled during the outage.
2R17 Spring 2003	E-D: VT-3 of Concrete Floor to Interior Drywell Shell Mastic Seal. Fully replaced seal. E-G: VT-1 of bolting for any pressure boundary penetration that was disassembled during the outage. Supplemental: Torus submerged surface desludge and coatings inspection. Minor coating degradation and shell pitting observed. No repairs required. Supplemental: UT Thickness exams of select locations of the Drywell shell. No pressure boundary degradation identified.

All HNP-2 IWE examinations that were required by the rulemaking to 10CFR50.55a to be completed by September 9, 2001, were completed during the September 2001 through October 2001 timeframe.

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Containment Modifications

No modifications or repairs of the HNP-2 containment have been conducted as a result of ISI. Other modifications or repairs which were implemented after the last Type A Test (November 2, 1995) include:

- Torus penetration stiffening as part of the ECCS Suction Strainer upgrade (Fall 1998).
- Drywell interior attachments as part of SRV Transfer Monorail installation (Spring 2003).
- Drywell interior attachments as part of MSIV Maintenance Platform installation (Fall 2001).

These modifications only involved attachments to the pressure boundary and all modifications were performed in accordance with the ASME Section XI Repair and Replacement Program. The Repair and Replacement Program addresses the IWE examination requirements.

7. NRC Question

Provide a schedule of future ISI activities including, if any, planned major repairs and modifications during the ILRT extension period from 10 to 15 years.

SNC Response

Below is the tentative schedule for future Containment ISI activities through the proposed ILRT extension period which concludes Spring 2010. This schedule is believed to be true and accurate at the time of submittal; however, the examination activities for a given refueling outage are subject to change due to rulemaking, licensing actions, and licensee outage scheduling or ALARA considerations.

Examination in accordance with the 1992 Edition with 1992 Addenda ASME Boiler and Pressure Vessel Code, Section XI, Subsection IWE, IWE-2500-1, Categories E-B, Pressure Retaining Welds and E-F, Pressure Retaining Dissimilar Metal Welds was made optional per 10 CFR 50.55a(b)(2)(ix)(C). SNC has chosen not to perform such examinations at HNP. Additionally, as of the date of this submittal, no areas of HNP-2 have been identified as requiring examination in accordance with Category E-C, Containment Surfaces Requiring Augmented Examination. If any areas are identified at a later date, examinations will be conducted in accordance with the applicable rulemaking.

No major containment repairs or modifications are anticipated during the extended ILRT interval.

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Containment Inspection Summary Table

Outage / Date	Planned Containment ISI Activities
2R18 Spring 2005	E-A: Visual Examination E-D: None (100% inspection during 2R15 (S2000) and 2R16 (F2001)) E-G: VT-1 of any bolted connection that is disassembled + VT-1 of any bolted connections not disassembled previously (complete 10-year interval requirements). E-P: In accordance with 10 CFR 50 Appendix J, Option B.
December 31, 2005	End of 3 rd Inspection Period for 3 rd ISI Program Interval. HNP intends to update (via licensing submittal) IWE Interval coincident with ISI Interval in lieu of 9/9/2008 (7-years from end of 1 st IWE period as allowed by 10 CFR 50).
January 1, 2006	Beginning of new ISI and IWE Interval.
2R19 Spring 2007	E-A: None E-D: VT-3 of moisture barrier sufficient to complete examination of at least 16% of accessible moisture barriers per Table IWE-2412-1 Inspection Program B (100% is typically examined due to small size of moisture barrier). E-G: VT-1 of any bolted connection that is disassembled. E-P: In accordance with 10 CFR 50 Appendix J, Option B.
December 31, 2008	End of 1 st Period of Inspection Interval
2R20 Spring 2009	E-A: Visual Examination as required by updated IWE Program. E-D: None. E-G: VT-1 of any bolted connection that is disassembled. E-P: In accordance with 10 CFR 50 Appendix J, Option B.

8. NRC Question

Describe briefly the containment liners areas that can be inspected visually from both sides, inside only, or outside only, and also the areas that are uninspectable from both sides such as imbedded liner or basemat liner. In addition, provide their corresponding percentage of total containment liner area.

SNC Response

The containment for Hatch Unit 2 is a General Electric Mark I containment design. Provided below is a response for the various containment structures.

Drywell Shell

The exterior surface of the drywell shell, with the exception of the drywell head, is not accessible for visual examination and is exempt from examination per IWE-1220(b) and IWE-1232(a). The exterior surface is inaccessible due to the concrete shield wall and the 2" air gap.

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The drywell shell below the concrete basemat is inaccessible for examination from both sides. However, this area is exempt from examination per IWE-1220(b) and IWE-1232(a).

The interior surfaces of the drywell shell, above the basemat elevation are generally accessible for 100% visual examination. Therefore, slightly more than 50% of the total drywell shell is accessible for visual examination from one side with the drywell head being accessible from both sides.

Suppression Pool Exterior Surfaces

The outside surfaces of the suppression pool are generally 100% accessible for visual examination.

Suppression Pool Interior Surfaces

Virtually 100% of the interior non-submerged suppression pool surfaces (vapor space) are accessible for visual examination.

Suppression Pool Interior Submerged Surfaces

The submerged surfaces of the suppression pool are only accessible for visual examination using underwater divers or by draining the pool.

To satisfy the IWE requirements, visual examination by divers with VT-3 certifications is performed once each interval. 100% of the submerged area is examined.

Vent System

Virtually 100% of the vent system surfaces are accessible for visual examination.

f. Risk Assessment

Attached is a detailed performance based, risk informed assessment, "Risk Assessment for Edwin I. Hatch Nuclear Power Station Regarding ILRT (Type A) Extension Request," to support this request.

g. Similar Requests

(As noted in cover letter)

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h. Conclusion

Based on the attached risk assessment results, the containment leak rate test results, and containment inspection results, the requested change is concluded to be acceptable.

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Enclosure 2

10 CFR 50.92 Evaluation

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Enclosure 2

10 CFR 50.92 Evaluation

In 10 CFR 50.92(c), the NRC provides the following standards to be used in determining the existence of a significant hazards consideration:

...a proposed amendment to an operating license for a facility licensed under §50.21(b) or §50.22 or for a testing facility involves no significant hazards consideration, if operation of the facility in accordance with the proposed amendment would not: (1) Involve a significant increase in the probability or consequences of an accident previously evaluated; or (2) Create the possibility of a new or different kind of accident from any accident previously evaluated; or (3) Involve a significant reduction in a margin of safety.

Southern Nuclear Operating Company has reviewed the proposed license amendment request and determined its adoption does not involve a significant hazards consideration based on the following discussion.

Basis for no significant hazards consideration determination

1. *The proposed Technical Specification change does not involve a significant increase in the probability or consequences of an accident previously evaluated.*

The proposed revision to Technical Specification 5.5.12 ("Primary Containment Leakage Rate Testing Program") involves a one-time extension to the current interval for Type A containment testing. The current test interval of ten (10) years would be extended on a one-time basis to no longer than fifteen (15) years from the last Type A test. The proposed Technical Specification change does not involve a physical change to the plant or a change in the manner which the plant is operated or controlled. The reactor containment is designed to provide an essentially leak tight barrier against the uncontrolled release of radioactivity to the environment for postulated accidents. As such the reactor containment itself and the testing requirements invoked to periodically demonstrate the integrity of the reactor containment exist to ensure the plant's ability to mitigate the consequences of an accident, and do not involve the prevention or identification of any precursors of an accident. Therefore, the proposed Technical Specification change does not involve a significant increase in the probability of an accident previously evaluated.

The proposed change involves only the extension of the interval between Type A containment leakage tests. Type B and C containment leakage tests will continue to be performed at the frequency currently required by plant Technical Specifications. Industry experience has shown, as documented in NUREG-1493, that Type B and C containment leakage tests have identified a very large percentage of containment leakage paths and that the percentage of containment leakage paths that are detected only by Type A testing is very small. HNP Unit 2 ILRT test history supports this

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Integrated Leakage Rate Testing Interval Extension

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conclusion. NUREG-1493 concluded, in part, that reducing the frequency of Type A containment leak tests to once per twenty (20) years leads to an imperceptible increase in risk. The integrity of the reactor containment is subject to two types of failure mechanisms which can be categorized as (1) activity based and (2) time based. Activity based failure mechanisms are defined as degradation due to system and/or component modifications or maintenance. Local leak rate test requirements and administrative controls such as design change control and procedural requirements for system restoration ensure that containment integrity is not degraded by plant modifications or maintenance activities. The design and construction requirements of the reactor containment itself combined with the containment inspections performed in accordance with ASME Section XI, the Maintenance Rule and the containment coatings program serve to provide a high degree of assurance that the containment will not degrade in a manner that is detectable only by Type A testing. Therefore, the proposed Technical Specification change does not involve a significant increase in the consequences of an accident previously evaluated.

2. *The proposed TS change does not create the possibility of a new or different kind of accident from any accident previously evaluated.*

The proposed revision to the Technical Specifications involves a one-time extension to the current interval for Type A containment testing. The reactor containment and the testing requirements invoked to periodically demonstrate the integrity of the reactor containment exist to ensure the plant's ability to mitigate the consequences of an accident and do not involve the prevention or identification of any precursors of an accident. The proposed Technical Specification change does not involve a physical change to the plant or the manner in which the plant is operated or controlled. Therefore, the proposed Technical Specification change does not create the possibility of a new or different kind of accident from any accident previously evaluated.

3. *The proposed TS change does not involve a significant reduction in a margin of safety.*

The proposed revision to Technical Specifications involves a one-time extension to the current interval for Type A containment testing. The proposed Technical Specification change does not involve a physical change to the plant or a change in the manner in which the plant is operated or controlled. The specific requirements and conditions of the Primary Containment Leakage Rate Testing Program, as defined in Technical Specifications, exist to ensure that the degree of reactor containment structural integrity and leak-tightness that is considered in the plant safety analysis is maintained. The overall containment leakage rate limit specified by Technical Specifications is maintained. The proposed change involves only the extension of the interval between Type A containment leakage tests. Type B and C containment leakage tests will continue to be performed at the frequency currently required by plant Technical Specifications.

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HNP Unit 2 and industry experience strongly supports the conclusion that Type B and C testing detects a large percentage of containment leakage paths and that the percentage of containment leakage paths that are detected only by Type A testing is small. The containment inspections performed in accordance with ASME Section XI, the Maintenance Rule and the Coatings Program serve to provide a high degree of assurance that the containment will not degrade in a manner that is detectable only by Type A testing. Additionally, the on-line containment monitoring capability that is inherent to inerted BWR containments allows for the detection of gross containment leakage that may develop during power operation. The combination of these factors ensures that the margin of safety that is inherent in plant safety analysis is maintained. Therefore, the proposed Technical Specification change does not involve a significant reduction in a margin of safety.

ENVIRONMENTAL IMPACT

The proposed Technical Specification changes were reviewed against the criteria of 10 CFR 51.22 for environmental considerations. The proposed changes do not involve a significant hazards consideration, a significant increase in the amounts of effluents that may be released offsite, or a significant increase in individual or cumulative occupational radiation exposures. Based on the foregoing, Southern Nuclear Operating Company concludes the proposed Technical Specifications meet the criteria given in 10CFR51.22(c)(9) for a categorical exclusion from the requirements for an Environmental Impact Statement.

CONCLUSION

SNC has concluded that the proposed change to the Plant Hatch Unit 2 TS does not involve a Significant Hazards Consideration.

**Hatch Nuclear Plant Unit 2
Technical Specification Revision Request
Integrated Leakage Rate Testing Interval Extension**

Enclosure 3

Marked up and Clean Typed TS Pages

5.5 Programs and Manuals (continued)

5.5.12 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 exception to NEI 94-01, Rev. 0, "Industry Guideline for Implementing Performance-Based Option of 10 CFR 50, Appendix J":

Section 9.2.3: The first Type A test after the November 2, 1995 Type A test shall be performed no later than November, 2010.

The peak calculated primary containment internal pressure for the design basis loss of coolant accident, P_a , is 46.9 psig.

The maximum allowable primary containment leakage rate, L_a , at P_a is 1.2% of primary containment air weight per day.

Leakage rate acceptance criteria are:

- a. Primary containment overall leakage rate acceptance criterion is $\leq 1.0 L_a$. During the first unit startup following testing in accordance with this program, the leakage rate acceptance criteria are $\leq 0.60 L_a$ for the combined 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 $\leq 0.05 L_a$ when tested at $\leq P_a$,
 - 2) For each door, leakage rate is $\leq 0.01 L_a$ when the gap between the door seals is pressurized to ≥ 10 psig for at least 15 minutes.

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

(continued)

5.5 Programs and Manuals (continued)

5.5.12 Primary Containment Leakage Rate Testing Program

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

(continued)

Attachment 1

**Risk Assessment for Edwin I. Hatch
Nuclear Power Station Regarding ILRT
(Type A) Extension Request**

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ATTACHMENTS

- A. CONTAINMENT ISOLATION FAULT TREE
- B. CUTSETS FOR THE CONTAINMENT ISOLATION FAULT TREE

Section 1

PURPOSE OF ANALYSIS

1.0 PURPOSE

The purpose of this analysis is to provide a risk assessment of extending the currently allowed containment Type A integrated leak rate test (ILRT) from ten years to fifteen years for a one time extension for Hatch Unit 1 and Unit 2. The extension would allow for substantial cost savings as the ILRT could be deferred for additional scheduled refueling outages for each of the Hatch units. The risk assessment follows the guidelines from NEI 94-01 [1], the methodology used in EPRI TR-104285 [2], and the NRC regulatory guidance on the use of Probabilistic Risk Assessment (PRA) findings and risk insights in support of a request to change a plant's licensing basis as outlined in Regulatory Guide 1.174 [3].

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 was less than normal containment leakage of 1.0La. Both Hatch units meet these requirements.

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, "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 [4], which analyzed the effects of containment leakage on the health and safety of the public and the benefits realized from the containment leak rate testing determined 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. In addition, 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 Hatch specific models and available data.

EPRI TR-104285 (Risk Impact Assessment of Revised Containment Leak Rate Testing Intervals) is a follow-on report to NUREG-1493 that provides a methodology for use in preparing PRA analysis to support a submittal. This methodology is followed to determine the appropriate risk information for use in evaluating the impact of the proposed ILRT changes.

It should be noted that containment leak-tight integrity is also verified through periodic inservice 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 inservice 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 and C local leak tests performed to verify the leak-tight integrity of containment penetration valves, bellows, airlocks, seals, and gaskets are also not affected by the change to the Type A test frequency.

1.2 CRITERIA

The acceptance guidelines in RG 1.174 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 help ensure and 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, the total annual risk (person rem/yr population dose) is examined to demonstrate the relative change in this parameter. (No criteria has been established for this parameter change.)

Section 2

METHODOLOGY

A simplified bounding analysis approach consistent with the EPRI approach is used for evaluating the change in risk associated with increasing the test interval to fifteen years. The approach is consistent with that presented in EPRI TR-104285 [2] and NUREG-1493 [4]. The analysis uses the current Hatch Probabilistic Risk Assessment (PRA) model that includes the results from the Hatch Level 2 analysis of core damage scenarios and subsequent containment response resulting in various fission product release categories (including no release).

The four general steps of this risk assessment are as follows:

- 1) Quantify the baseline risk and sensitivity cases in terms of frequency events (per reactor year) for each of the eight containment release scenario types identified in the EPRI report.
- 2) Develop plant-specific person-rem (population dose) per reactor year for each of the eight containment release scenario types from plant specific consequence analyses (i.e., previously performed SAMA calculations using MACCS2).
- 3) Evaluate the risk impact (i.e., the change in containment release scenario type frequency and population dose) of extending the ILRT interval to fifteen years.
- 4) Determine the change in risk in terms of Large Early Release Frequency (LERF) in accordance with Regulatory Guide 1.174 [3] and compare with the acceptance guidelines of RG 1.174.

This approach is based on the information and approaches contained in the previously mentioned studies and further is consistent with the following:

- Consistent with the other industry risk assessments of extending the ILRT test interval, the Hatch assessment uses population dose as one of the risk measures. The other risk measures used in the Hatch assessment are Large Early Release Frequency (LERF) and Conditional Containment Failure Probability (CCFP) to demonstrate that the acceptance guidelines from RG 1.174 are met.
- Consistent with EPRI TR-104285 and NUREG-1493, the Hatch assessment uses information from NUREG-1273 [6] regarding the low percentage of containment leakage events that would only be detected by an ILRT as input to calculate the increase in the pre-existing containment leakage probability due to the testing interval extension.
- Consistent with the approach used in the Indian Point 3 risk-informed submittal for a one-time extension of the Type A test interval, the Hatch evaluation uses similar ground rules and methods to calculate changes in risk metrics. [14] The NRC approval was granted on April 17, 2001 (TAC No. MB0178). [22]

Section 3

GROUND RULES

The following ground rules are used in the analysis:

- The Hatch Level 1 and Level 2 internal events PRA model for Unit 1 provides representative results for the analysis. (A Unit 2 PRA model is available and the CDF and LERF are essentially the same, but it is judged that it will not provide any unique or additional insights compared to the results from the Unit 1 model.)
- It is appropriate to use the Hatch 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) will not substantially differ if fire and seismic events were to be included in the calculations.
- An evaluation of the risk trade-off impact of performing the ILRT during shutdown is addressed using the generic results from EPRI TR 105189. [10]
- Dose results for the containment failures modeled in the PRA can be characterized by the Hatch population dose results from MACCS2 calculations such as performed for SAMA.
- The lowest consequence calculations (i.e., intact containment and small leakages) are not available on a plant specific basis for Hatch; they are based on scaling the NUREG 1150 results for such cases relative to population and differences in Technical Specification Leakage.
- Accident classes describing radionuclide release end states are defined consistent with EPRI methodology [2] and are summarized in Section 4.2.
- The maximum containment leakage for Class 1 sequences is 1 L_a . Class 3 accounts for increased leakage due to Type A inspection failures.

- The maximum containment leakage for Class 3a sequences is $10 L_a$ based on the previously approved methodology [14, 22].
- The maximum containment leakage for Class 3b sequences is $35 L_a$ based on the previously approved methodology [14, 22]
- Class 3b is conservatively categorized as LERF based on the previously approved methodology [14, 22]
- 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 from this analysis will result from this assumption.
- The reduction in ILRT frequency does not impact the reliability of containment isolation valves to close in response to a containment isolation signal. Containment isolation valves that fail to close during an accident and in response to a containment isolation signal are calculated on a Hatch specific basis and made part of the overall population dose and LERF calculations.

Section 4

INPUTS

This section summarizes the general resources available as input (Section 4.1) and the plant specific resources required (Section 4.2).

4.1 General Resources Available

Various industry studies on containment leakage risk assessment are briefly summarized here:

- 1) NUREG/CR-3539 [7]
- 2) NUREG/CR-4220 [8]
- 3) NUREG-1273 [6]
- 4) NUREG/CR-4330 [9]
- 5) EPRI TR-105189 [10]
- 6) NUREG-1493 [4]
- 7) EPRI TR-104285 [2]

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 which 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 last study is an EPRI study of the impact of extending ILRT and LLRT test intervals on at-power public risk.

NUREG/CR-3539 [7]

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

NUREG/CR-4220 [8]

NUREG/CR-4220 is a study performed by Pacific Northwest Laboratories 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. It is the latter category that is applicable to containment isolation modeling that is the focus of this risk assessment.

NUREG/CR-4220 assessed the "large" containment leak probability to be in the range of $1\text{E-}3$ to $1\text{E-}2$, with $5\text{E-}3$ identified as the point estimate based on 4 events in 740 reactor years and conservatively assuming a one-year duration for each event. It should be noted that all of the 4 identified large leakage events were PWR events, and the assumption of a one-year duration is not applicable to an inerted containment such as Hatch. The NUREG identifies inerted BWRs as having significantly improved potential for leakage detection because of the requirement to remain inerted during power operation. This calculation presented in NUREG/CR-4220 is called an "upper bound" estimate for BWRs (presumably meaning "inerted" BWR containment designs).

NUREG-1273 [6]

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 [9]

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 [10]

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 extending the ILRT frequency from 3 per 10 years to 1 per 10 years was calculated to be a reduction of approximately $1\text{E-}7/\text{yr}$ 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, and the other 5 were events involving 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 [4]

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.

NUREG-1493 used information from NUREG-1273 regarding the low percentage of containment leakage events that would only be detected by an ILRT in the calculation of

the increase in the pre-existing containment leakage probability due to the testing interval extension. NUREG-1493 makes the following assumptions in this probability calculation:

- The average time that a pre-existing leakage may go undetected increases with the length of the testing interval (and is $\frac{1}{2}$ the length of the test interval)
- Only 3% of all pre-existing leaks can be detected only by an ILRT (i.e., and not by LLRTs)

This same approach that was used in a previously approved ILRT test interval extension submittal [14, 22] is also proposed here for the Hatch ILRT test interval extension risk assessment.

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 ILRT and 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 frequencies into eight (8) classes of containment response to a core damage accident:

1. Containment intact and isolated
2. Containment isolation failures dependent upon the core damage accident
3. Type A (ILRT) related containment isolation failures
4. Type B (LLRT) related containment isolation failures
5. Type C (LLRT) 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..."

NUREG-1150 [23] and NUREG/CR 4551 [5]

NUREG-1150 and the technical basis, NUREG/CR 4551[5], provide an ex-plant consequence analysis for a spectrum of accidents including a severe accident with the containment remaining intact (i.e., Tech Spec leakage). This ex-plant consequence calculation is calculated for the 50-mile radial area surrounding Peach Bottom and represents a very small contributor to the overall risk spectrum. Because it is a small contributor, this ex-plant calculation, total person-rem, is considered adequate to represent Hatch if the Tech Spec leakage and the population are scaled to represent Hatch. (The meteorology is assumed not to play a significant role in this evaluation.)

4.2 Plant Specific Inputs

The information used to perform the Hatch ILRT Extension Risk Assessment includes the following:

- Level 1 Model
- Level 2 Model
- Release Category definitions used in Level 2 or LERF

- Population Dose calculations by release category (e.g., MACCS2 code calculation results)
- ILRT results to demonstrate adequacy of the administrative and hardware issues.⁽¹⁾

Level 1 Model

The Level 1 Model that is used for Hatch Unit 1 is characteristic of the as-built plant. The Level 1 model is developed in CAFTA. Table 4.2-1 summarizes some of the quantitative results of the Hatch PRA model of record.

The Level 1 model was quantified with the total Core Damage Frequency (CDF) = $1.24\text{E}-5/\text{yr}$ at a truncation of $1\text{E}-11/\text{yr}$.

Level 2 Model

The Level 2 Model that is used for Hatch Unit 1 was developed to calculate the LERF contribution. The Level 2 model was quantified using the CAFTA model. The total Large Early Release Frequency (LERF) was found to be $2.19\text{E}-6/\text{yr}$ at a truncation of $1\text{E}-11/\text{yr}$. Table 4.2-1 summarizes some of the pertinent Hatch results.

The contributors to the LERF calculation were found as follows:

- Containment Bypass (LER_CB) = $1.65\text{E}-7$
- Containment Overpressure (LER_OPD) = $6.56\text{E}-7$
- Containment Overtemperature (LER_OT) = $1.37\text{E}-6$
- Containment Intact with DW Vent Open (LER_VD) = $8.1\text{E}-10$

⁽¹⁾ The two most recent Type A tests at Hatch 1 and Hatch 2 have been successful, so the current Type A test interval requirement is 10 years.

Therefore, several additional calculations were performed to allow the representation of elements of the risk profile that are not explicitly quantified as part of the Level 2 model. These include:

- Containment isolation failures
- Non-LERF contributors

Table 4.2-1

SUMMARY OF PRA MODEL RESULTS

Truncation (/yr)	Level 1 Results		Level 2 Results	
	CDF	# Cut Sets	LERF	# Cut Sets
1.00E-08	6.96E-06	166	8.59E-07	24
1.00E-09	9.85E-06	1234	1.53E-06	260
1.00E-10	1.15E-05	7172	1.94E-06	1787
1.00E-11	1.24E-05	37197	2.19E-06	10336
<i>Level 2 Subgate Results (@1E-11/yr truncation)</i>				
LERF Subgate	CET Sequence	LERF	# Cut Sets	LERF %
Gate LER_CB	5	1.65E-07	22	7.5
Gate LER_OPD	4, 11	6.56E-07	5711	30.0
Gate LER_VD	15	8.10E-10	16	ε
Gate LER_OT	2	1.37E-06	4587	62.5
Total LERF		2.19E-06	10336	100
<i>Late Containment Failure Results (@1E-11/yr truncation)</i>				
Level 2 Subgate	CET Sequence	LATE	# Cut Sets	LATE %
Gate LAT_OT ⁽¹⁾	9	6.12E-08	407	57
Gate LAT_OPD ⁽¹⁾	12	4.62E-08	142	43
Total Late		1.07E-07	549	100

(1) Level 2 subgates for late containment failure logic based on existing LERF fault tree logic.

Containment isolation failure is not included in the Hatch PRA Level 2 risk calculation because it is judged sufficiently small in probability to be deleted. However, as part of the ILRT evaluation, the detailed containment isolation fault tree has been quantified and used in conjunction with the CDF to calculate the containment isolation failure frequency under severe accident conditions for use in the EPRI ILRT categorization scheme for dose calculation purposes. Therefore, this risk contribution is added to the baseline risk profile. This quantification is summarized in Section 5.

Similarly, non-LERF contributors were also added to the containment evaluation by quantifying the appropriate non-LERF branches of the Hatch Containment Event Tree.

Population Dose Calculations

The population dose is calculated from MACCS2 calculations performed for the Hatch SAMA evaluation which is representative of power uprated operation for Hatch. Table 4.2-2 summarizes the calculated population dose/year when the frequencies of accident sequences contributing to each category were multiplied by the applicable MACCS2 calculated person-rem.

Table 4.2-3 provides the derivations of the annual population dose (person-rem/year) citing both the accident sequence frequencies used in the SAMA evaluation and the total population dose (person-rem) calculated by MACCS2. It is noted that the Hatch PRA model has been updated since the SAMA analysis and the accident sequence frequencies and the associated annual population dose has decreased from that used in the SAMA evaluation.

The population dose (person-rem) for each of the severe accident types modeled in the PRA from Table 4.2-3 provides the input to the calculation of the risk spectrum for the

various ILRT configurations calculated in Section 5 of this analysis. However, there is not a plant specific calculation of the person-rem dose associated with Technical Specification allowed leakage under a core damage accident. (This is typically much smaller than the person-rem dose associated with severe accidents involving containment failure states.) In order to approximate the intact containment dose (in person-rem), the NUREG/CR-4551 calculation for the Peach Bottom site using Accident Progression Bin 8 (Core is damaged, Vessel is breached, but no containment failure has occurred — Technical Specification leakage of 0.5%/day is assumed) is used. The resulting dose is 8,300 person rem for the Peach Bottom site which includes a population of 5,060,000 in the calculation. [15]

This can be used as an approximation of the dose for Hatch if it is corrected for the population surrounding Hatch and the difference in Technical Specification leak rate. The population within 50 miles used for Hatch is that projected for 2030 of 499,000. [20] This will be conservative for the period before 2020 which is the time applicable to the ILRT one time extension.

This leads to a dose for severe accidents with the containment intact of:

$$8,300 \text{ person-rem} \quad * \quad \frac{499,000}{5,060,000} \quad = \quad 818 \text{ person-rem}$$

However, a second correction factor is also required to the NUREG/CR-4551 calculation to account for differences in the Technical Specification leakage value.

The Technical Specification containment allowable leak rate for Hatch is 1.2% of Primary Containment air weight per day (L_a^H) versus the 0.5% of Primary Containment air weight per day (L_a^{PB}) for the NUREG-1150 plant, Peach Bottom. Therefore, the population dose due to allowable Technical Specification leakage in person-rem calculated for Peach Bottom given a severe accident that is scaled by population for the Hatch analysis must

also be multiplied by a factor of 2.4 ($= L_a^H / L_a^{PB}$) to account for the differences in Technical Specification leakage rates.

The Hatch "intact containment" leakage dose is then:

$$818 \text{ person-rem} * 2.4 = 1963 \text{ person-rem}$$

As can be seen by comparison with accidents that involve containment breach or bypass, the leakage dose is extremely small and would be expected to have little influence on the baseline risk or the change in risk.

Table 4.2-2

MACCS2 POPULATION DOSE CALCULATIONS FOR
SPECIFIC ACCIDENT SEQUENCES [21]

Release Mode	Sequence	Population Dose (Person-rem/yr)	Contribution (%)
Containment bypass	5 (Loss-of-coolant accident (LOCA) Outside Containment)	0.189	5.44
Early containment failure	2 (SBO), 4 (Loss of containment heat removal (CHR)/Drywell Failure), 11 (Anticipated transient without scram (ATWS) Drywell Failure)	3.18	91.21
Late containment failure	12 (High pressure transient w/loss of CHR), 14 (SBO w/containment isolation failure)	0.113	3.32
Intact containment (venting)	15 (High pressure transient w/Venting)	1.05E-03	0.03
TOTAL		3.48	100

Risk Impact Assessment of Extending the Containment Type A Test Interval

Table 4.2-3 SUMMARY OF SAMA/MACCS2 CALCULATIONS [20]				
Level 2 End State	SEQ #	Frequency (per yr) ⁽¹⁰⁾	Total Dose Person-Rem	Annual Risk (Person-Rem/Yr) [20, 21]
Containment Bypass	5	1.66E-7 ⁽⁶⁾	1.15E+6 ⁽²⁾	0.19
Early Cont. Failure	2	1.79E-6 ⁽⁶⁾	1.06E+6 ⁽³⁾	1.90
	4	7.43E-7 ⁽⁶⁾	1.02E+6 ⁽⁴⁾	0.76
	11	7.43E-7 ⁽⁶⁾	7.02E+5 ⁽⁵⁾	0.52
				3.18
Late Cont. Failure	12	2.0E-7 ⁽¹⁾	5.7E+5	0.112 ⁽⁸⁾
	14	3.1E-9 ⁽¹⁾		0.0008
Intact Cont. (DW Vent)	15	9.24E-10 ⁽⁶⁾	1.13E+6 ⁽⁹⁾	0.001
No Containment Failure				e ⁽⁷⁾
			TOTAL	3.48

⁽¹⁾ RAI response to Q#4 [20]

⁽²⁾ RAI response to Q#14; Sequence #5 [20] clarification provided to NRC by SNC [21]

⁽³⁾ RAI response to Q#14; Sequence #2 [20]

⁽⁴⁾ RAI response to Q#14; Sequence #4 [20]

⁽⁵⁾ RAI response to Q#14; Sequence #11 [20]

⁽⁶⁾ RAI response to Q#1.b-1 [20]

⁽⁷⁾ e - negligible; not calculated

⁽⁸⁾ RAI clarification provided by SNC to Question #5 [21]

⁽⁹⁾ RA response to Q#14; Sequence 15 [20]

⁽¹⁰⁾ It is noted that the Hatch PRA model has been updated since the SAMA analysis and the accident sequence frequencies and the associated annual population dose has decreased from that used in the SAMA evaluation.

Release Category Definitions

Table 4.2-4 defines the accident classes used in the ILRT extension evaluation consistent with the EPRI methodology [2].

Table 4.2-4
EPRI CONTAINMENT FAILURE CLASSIFICATIONS

Class	
1	Containment remains intact including accident sequences that do not lead to containment failure in the long term. The release of fission products (and attendant consequences) is determined by the maximum allowable leakage rate values L_m under Appendix J for that plant
2	Containment isolation failures include those accidents in which there is a failure to isolate the containment.
3	Independent (or random) isolation failures include those accidents in which the pre-existing isolation failure to seal (i.e., provide a leak-tight containment) is not dependent on the sequence in progress.
4	Independent (or random) isolation failures include those accidents in which the pre-existing isolation failure to seal is not dependent on the sequence in progress. This class is similar to Class 3 isolation failures, but is applicable to sequences involving Type B tests and their potential failures. These are the Type B-tested components that have isolated but exhibit excessive leakage.
5	Independent (or random) isolation failures include those accidents in which the pre-existing isolation failure to seal is not dependent on the sequence in progress. This class is similar to Class 4 isolation failures, but is applicable to sequences involving Type C tests and their potential failures.
6	Containment isolation failures include those leak paths covered in the plant test and maintenance requirements or verified per in service inspection and testing (ISI/IST) program.
7	Accidents involving containment failure induced by severe accident phenomena. Changes in Appendix J testing requirements do not impact these accidents.
8	Accidents in which the containment is bypassed (either as an initial condition or induced by phenomena) are included in Class 8. Changes in Appendix J testing requirements do not impact these accidents.

4.3 CONDITIONAL PROBABILITY OF ILRT FAILURE (SMALL AND LARGE)

The ILRT can detect a number of failures such as liner breach, failure of certain bellows arrangements, and failure of some sealing surfaces. The proposed ILRT test interval extension may influence the conditional probability associated with the ILRT failure. To ensure that this effect is properly accounted for, the Class 3 Accident Class is divided into two sub-classes, Class 3a and Class 3b, representing small and large leakage failures, respectively.

To calculate the probability that a liner leak will be large (Event CLASS-3B), use was made of the data presented in NUREG-1493 [4]. The data found in NUREG-1493 states that 144 ILRTs were conducted. The largest reported leak rate from those 144 tests was 21 times the allowable leakage rate (L_a). Because $21L_a$ does not constitute a large release, no releases have occurred based on the 144 ILRTs reported in NUREG-1493 [4].

To estimate the failure probability given that no failures have occurred, a conservative estimate is obtained from the 95th percentile of the χ^2 distribution. In statistical theory, the χ^2 distribution can be used for statistical testing, goodness-of-fit tests, and evaluating s-confidence [13]. The χ^2 distribution is really a family of distributions, which range in shape from that of the exponential to that of the normal distribution. Each distribution is identified by the degrees of freedom, ν . For time-truncated tests (versus failure-truncated tests), an estimate of the probability of a large leak using the χ^2 distribution can be calculated as $\chi^2_{95}(\nu = 2n+2)/2N$, where n represents the number of large leaks and N represents the number of ILRTs performed to date. With no large leaks ($n=0$) in 144 events ($N = 144$) and $\chi^2_{95}(2) = 5.99$, the 95th percentile estimate of the probability of a large leak is calculated as $5.99/(2*144) = 0.021$.

To calculate the probability that a liner leak will be small (Event CLASS-3A), use was made of the data presented in NUREG-1493 [4]. The data found in NUREG-1493

states that 144 ILRTs were conducted. The data reported that 23 of 144 tests had allowable leak rates in excess of $1.0L_a$. However, of these 23 "failures" only 4 were found by an ILRT; the others were found by Type B and C testing or errors in test alignments. Therefore, the number of failures considered for "small releases" are 4-of-144. Similar to the event CLASS-3B probability, the estimated failure probability for small release is found by using the χ^2 distribution. The χ^2 distribution is calculated by $n=4$ (number of small leaks) and $N=144$ (number of events) which yields a χ^2 (10)=18.3070. Therefore, the 95th percentile estimate of the probability of a small leak is calculated as $18.3070/(2*144) = 0.064$.

Using the methodology discussed above is conservative compared to the typical mean estimates used for PRA analysis. For example, the mean probability of a Class 3a failure would be the (number of failures) / (number of tests) or $4/144 = 0.03$ compared with 0.064 used here.

4.4 IMPACT OF EXTENSION ON LEAK DETECTION PROBABILITY

The NRC in NUREG-1493 [4] has determined from a review of operating experience data⁽¹⁾ that only 3% of the ILRT failures were found which local leakage-rate testing could not and did not detect. In NUREG-1493 [4], it is noted that based on a review of leakage-rate testing experience, a small percentage (3%) of leakages that exceed current requirements are detectable only by Type A testing (ILRT). Further, in NUREG-1493 it is noted that the leakage rates observed in these few Type A test failures were only marginally above currently prescribed limits and could be characterized by a leakage rate of about two times the allowable.

Also in NUREG-1493 [4], it was assumed that the characteristic magnitude of leakages detectable only by ILRTs would not change, but the probability of leakage would change

⁽¹⁾ Data collected at a time when the ILRT frequency was 3/10 years is represented in NUREG 1493 [4] and by EPRI [2] as every 3 years.

due to the longer intervals between tests. The change in probability was estimated by comparing the average time that a leak could exist without detection. For example, the average time that a leak could go undetected with a three-year test interval is 1.5 years (3 yrs/2), and the average time that a leak could exist without detection for a ten-year interval is 5 years (10 yrs/ 2). This change would lead to a non-detection probability that is a factor of 3.33 (5.0/1.5) higher for the probability of a leak that is detectable only by ILRT testing. However, since ILRTs have been demonstrated to improve the residual leak detection by only 3%⁽¹⁾, the interval change noted above would only lead to about a 10% (3.33 x 3%) non-detection probability of a leak. Correspondingly, an extension of the ILRT interval to fifteen years can be estimated to lead to about a 15% (7.5/1.5 x 3%) non-detection probability of a leak.⁽²⁾

Therefore, the failure rate of ILRTs for which the LLRTs do not provide adequate backup is .03/1.5 year average detection time. As the average detection time increases and using a constant failure rate model, the failure probability of ILRTs, P_i , can be estimated as follows:

for 3 Year Interval

$$P_i = \frac{1}{2} \lambda T = \frac{0.03}{1.5 \text{ yr}} * \frac{3 \text{ yrs}}{2} = 0.03$$

for 10 Year Interval

$$P_i = \frac{1}{2} \lambda T = \frac{0.03}{1.5 \text{ yr}} * \frac{10 \text{ yrs}}{2} = 0.10$$

for 15 Year Interval

$$P_i = \frac{1}{2} \lambda T = \frac{0.03}{1.5 \text{ yr}} * \frac{15 \text{ yrs}}{2} = 0.15$$

⁽¹⁾ Assumes that the Local Leak Rate Tests (LLRT) will continue to provide leak detection for the other 97% of leakages.

⁽²⁾ These are obviously approximations assumed by the NRC and EPRI because the current 3 ILRTs in 10 years would have a $T/2 = 1.67$ years instead of 1.5 years.

EPRI has previously interpreted this to mean that the failure to detect probabilities are as follows:

ILRT FAILURE TO DETECT PROBABILITY

ILRT Interval	EPRI Assessment [2]	IP3 [14]	Constant Failure Rate Model
3 yr	0.03	0.03	0.03
10 yr	0.13	0.13	0.10
15 yr	NA	0.18	0.15

In addition, IP3 [14] has used this same estimate of changes in detection probability in a submittal to extend the ILRT interval on a one-time basis. The IP3 request for a one-time ILRT extension was approved by the NRC on April 17, 2000 (TAC No. MB0178). [22]

The analysis included in this report follows the precedence set by the EPRI report and the IP3 analysis. The use of the constant failure rate model is conservatively represented by the assumed "failure to detect" probabilities used by EPRI and in the IP3 submittal.

Section 5

RESULTS

The application of the approach based on EPRI-TR-105189 [10] and previous risk assessment submittals on this subject [14] has established a clear process for the calculation and presentation of results.

The method chosen to display the results is according to the eight (8) accident classes consistent with these two reports. Table 5-1 lists these accident classes.

The analysis performed examined Hatch specific accident sequences in which the containment remains intact or the containment is impaired. Specifically, the break down of the severe accidents contributing to risk were considered in the following manner:

- Core damage sequences in which the containment remains intact initially and in the long term (EPRI TR-104285 Class 1 sequences).
- Core damage sequences in which containment integrity is impaired due to random isolation failures of plant components other than those associated with Type B or Type C test components. For example, liner breach or bellows leakage. (EPRI TR-104285 Class 3 sequences).
- Core damage sequences in which containment integrity is impaired due to containment isolation failures of pathways left "opened" following a plant post-maintenance test. (For example, a valve failing to close following a valve stroke test.) (EPRI TR-104285 Class 6 sequences).
- Accident sequences involving containment bypass (EPRI TR-104285 Class 8 sequences), large containment isolation failures ((EPRI TR-104285 Class 2 sequences), and small containment isolation "failure-to-seal" events (EPRI TR-104285 Class 4 and 5 sequences) are accounted for in this evaluation as part of the baseline risk profile. However, they are not affected by the ILRT frequency change.
- Class 4 and 5 sequences are impacted by changes in Type B and C test intervals; therefore, changes in the Type A test interval do not impact these sequences.

Table 5-1

ACCIDENT CLASSES

Accident Classes (Containment Release Type)	Description
1	No Containment Failure
2	Large Isolation Failures (Failure to Close)
3a	Small Isolation Failures (liner breach)
3b	Large Isolation Failures (liner breach)
4	Small Isolation Failures (Failure to seal—Type B)
5	Small Isolation Failures (Failure to seal—Type C)
6	Other Isolation Failures (e.g., dependent failures)
7	Failures Induced by Phenomena (Early and Late)
8	Bypass (Interfacing System LOCA)
CDF	All CET End states (including very low and no release)

The steps taken to perform this risk assessment evaluation are as follows:

- Step 1 - Quantify the base-line risk in terms of frequency per reactor year for each of the applicable eight accident classes presented in Table 5-1.
- Step 2 - Develop plant specific person-rem dose (population dose) per reactor year for each of the eight accident classes evaluated in EPRI TR-104285.
- Step 3 - Evaluate the risk impact of extending Type A test interval from 10 to 15 years.
- Step 4 - Determine the change in risk in terms of Large Early Release Frequency (LERF) in accordance with RG 1.174.

5.1 STEP 1 - QUANTIFY THE BASE-LINE RISK IN TERMS OF FREQUENCY PER REACTOR YEAR

The severe accident sequence frequencies that can result in offsite consequences are evaluated. The latest update of the Hatch Level 1 PRA model as documented by SNC is used in the ILRT evaluation.

This step involves the review of the Hatch containment event tree (CET) and Level 2 accident sequence frequency results. The CET characterizes the response of the containment to important severe accident sequences that can fail containment and release radionuclides to the environment. The CET used in this evaluation is based on important phenomena and systems-related events identified in NUREG-1335 [23] and on plant features that influence the phenomena.

The containment isolation model for Hatch examines the probability of containment isolation failure. Attachment A includes the Containment Isolation fault tree. The assessed probability of a large containment isolation failure is found to be $4.4\text{E-}4/\text{demand}$. See cutsets from Attachment B.

As previously described, the extension of the Type A test interval does not influence those accident progressions that involve large containment isolation failures, Type B or Type C testing, or containment failure induced by severe accident phenomena.

For the assessment of ILRT impacts on the risk profile, the potential for pre-existing leaks are included in the model. Specifically, a simplified model based on NUREG 1493 results is used to predict the likelihood of having a small/large breach in the containment liner that is undetected by the Type A ILRT test. These events are represented by the "Class 3" sequence depicted in EPRI TR-104285 [2]. The Class 3 leakage includes the probability of a liner breach or bellows failure (due to excessive leakage) at the time of core damage. Two failure modes were considered to ensure

proper representation of available data. These are Event Class-3A (small breach) and Event Class-3B (large breach).

After including the containment isolation fault tree model (Attachment A), Class 2, and including the respective "large" and "small" liner breach leak rate probabilities (Classes 3A and 3B), the eight severe accidents class frequencies were developed consistent with the definitions in Table 5-1 and described below.

Class 1 Sequences. This group consists of all core damage accident progression bins for which the containment remains intact (modeled as Technical Specification Leakage). The frequency per year for these sequences is $9.06\text{E-}6/\text{year}$ and is determined by subtracting all containment failure end states from the total CDF. After all accident class frequencies (Classes 2 through 8) were developed, frequencies for Classes 2 through 8 were summed (result = $3.3\text{E-}6/\text{yr}$). This was then subtracted from the total CDF ($1.24\text{E-}5/\text{yr}$) to obtain the Class 1 frequency of "No Containment Failure" of $9.0\text{E-}6/\text{yr}$. For this analysis, the associated maximum containment leakage for this group is 1L_s , consistent with an intact containment evaluation.

Class 2 Sequences. This group consists of all core damage accident progression bins for which a failure to isolate the containment occurs. These sequences are dominated by failure-to-close of large containment isolation valves (Appendices A and B). The frequency per year for these sequences is determined as follows:

$$\text{CLASS 2 FREQUENCY} = \text{PROB}_{\text{large CI}} * \text{CDF}$$

Where:

$$\begin{aligned} \text{PROB}_{\text{large CI}} &= \text{Random large containment isolation failure probability (e.g., large valves)} \\ &= 4.4\text{E-}4 \text{ (see Appendix B)} \\ \text{CDF} &= \text{Core damage frequency} = 1.24\text{E-}5/\text{year} \end{aligned}$$

$$\begin{aligned}\text{CLASS 2 FREQUENCY} &= 4.4\text{E-4} * 1.24\text{E-5/year} \\ \text{CLASS 2 FREQUENCY} &= 5.5\text{E-9/year}\end{aligned}$$

These failures are assumed to result in a LERF that is characterized as a containment bypass, i.e., the same as Class 8. This may be overly conservative.

Class 3 Sequences. This group consists of all core damage accident progression bins for which a pre-existing leakage in the containment structure (e.g., containment liner) exists. The containment leakage for these sequences can be either small ($2L_a$ to $35L_a$) or large ($>35L_a$).

The respective frequencies per year are determined as follows:

$$\begin{aligned}\text{PROB}_{\text{class_3a}} &= \text{probability of small pre-existing containment liner leakage} \\ &= 0.064 \quad \quad \quad [\text{see Section 4.3}]\end{aligned}$$

$$\begin{aligned}\text{PROB}_{\text{class_3b}} &= \text{probability of large pre-existing containment liner leakage} \\ &= 0.021 \quad \quad \quad [\text{see Section 4.3}]\end{aligned}$$

$$\text{CLASS_3A_FREQUENCY} = 0.064 * 1.24\text{E-5/year} = 7.9\text{E-7/year}$$

$$\text{CLASS_3B_FREQUENCY} = 0.021 * 1.24\text{E-5/year} = 2.6\text{E-7/year}$$

For this analysis, the associated containment leakage for Class 3A is $10L_a$ and for Class 3B is $35L_a$. These assignments are consistent with the Indian Point 3 ILRT submittal [14] which was approved by the NRC. [22]

Class 4 Sequences. This group consists of all core damage accident progression bins for which a containment isolation failure-to-seal of Type B test components occurs. Because these failures are detected by Type B tests which are unaffected by the Type A ILRT, this group is not evaluated any further in the analysis.

Class 5 Sequences. This group consists of all core damage accident progression bins for which a containment isolation failure-to-seal of Type C test components. Because the failures are detected by Type C tests which are unaffected by the Type A ILRT, this group is not evaluated any further in this analysis.

Class 6 Sequences. This group is similar to Class 2. These are sequences that involve core damage accident progression bins for which a failure-to-seal containment leakage due to failure to isolate the containment occurs. These sequences are dominated by misalignment of containment isolation valves following a test/maintenance evolution.

This group is similar to Class 2, and addresses additional failure modes of containment failure with low probability of occurrence due to the inerted Mark I containment requirements for leak tightness. The low failure probabilities are based on the need for multiple failures, the presence of automatic closure signals, and control room indication. Based on the fact that this failure class is not impacted by Type A testing, a screening value is considered appropriate for this low probability failure mode. This is consistent with the EPRI guidance. However, in order to maintain consistency with the previously approved methodology (i.e. $PROB_{class6} > 0$), a conservative screening value of $4E-4$ will be used to evaluate this class.

The frequency per year for these sequences is determined as follows:

$$CLASS_6_FREQUENCY = PROB_{largeT\&M} * CDF$$

Where:

$PROB_{largeT\&M}$ = random large containment isolation failure probability due to valve misalignment is estimated using NUREG/CR 1278

= 4E-4

$$CLASS_6_FREQUENCY = 4E-4 * 1.24E-5/year = 5.0E-9/year$$

For this analysis, the associated containment leakage for this group is represented by the direct release from containment, i.e., Class 8 consequences are assigned.

Class 7 Sequences. This group consists of all core damage accident progression bins in which containment failure is induced by severe accident phenomena (e.g., Mark I shell melt-through, overpressure). For this analysis, the associated radionuclide releases are based on MACCS2 calculations.

$$CLASS_7_FREQUENCY = LER_OPD + LER_OT + LATE + LER_VD$$

Where the latest model calculation results are summarized in Table 4.2-1 and yield the following:

$$LATE = \text{total late containment failure frequency} = 1.1E-7/year$$

LER_OT Early Containment Failure due to overtemperature of the Mark I drywell

= 1.37E-6/yr

LER_OPD Early Containment Failure due to overpressure of the Mark I drywell

= 6.56E-7/yr

LER_VD Early Containment Release due to Drywell Venting (containment otherwise intact)

$$= 8.1\text{E-}10/\text{yr}$$

Where:

$$\text{Total early containment failure frequency} = 2.0\text{E-}6^{(1)}$$

$$\text{CLASS_7_FREQUENCY} = 2.0\text{E-}6/\text{year} + 1.1\text{E-}7/\text{year}$$

$$\text{CLASS_7_FREQUENCY} = 2.11\text{E-}6$$

Class 8 Sequences. This group consists of all core damage accident progression bins in which containment bypass occurs. The containment bypass failure frequency (LER_CB) for this class is $1.65\text{E-}7/\text{year}$.

Summary of Accident Class Frequencies

In summary, the accident sequence frequencies that can lead to radionuclide release to the public have been derived consistent with the definition of Accident Classes defined in EPRI-TR-104285. Table 5-2 summarizes these accident frequencies by Accident Class.

⁽¹⁾ Note that the early containment failure frequency included here does not include the containment bypass contribution which is treated under Class 8.

Table 5-2
RADIONUCLIDE RELEASE FREQUENCIES AS A FUNCTION OF ACCIDENT CLASS

Accident Classes (Containment Release Type)	Description	Frequency (per Rx-yr)
1	No Containment Failure (Including Successful Venting)	9.06E-6
2	Large Isolation Failures (Failure to Close)	5.5E-9
3a	Small Isolation Failures (liner breach)	7.9E-7
3b	Large Isolation Failures (liner breach)	2.6E-7
4	Small Isolation Failures (Failure to seal—Type B)	NA
5	Small Isolation Failures (Failure to seal—Type C)	NA
6	Other Isolation Failures (e.g., dependent failures)	5.0E-9
7	Failures Induced by Phenomena (Early and Late)	2.11E-6
7a	Early 2.0E-6	
7b	Late 1.1E-7 ⁽¹⁾	
8	Bypass (Interfacing System LOCA)	1.65E-7
CDF	All CET End states (including very low and no release)	1.24E-5

⁽¹⁾ Late - Derived from the PRA model by manipulation of the LERF model (LATE = 1.1E-7/yr)

5.2 STEP 2 - DEVELOP PLANT-SPECIFIC PERSON-REM DOSE (POPULATION DOSE) PER REACTOR YEAR

Plant-specific release analysis was performed to evaluate the person-rem doses to the population, within a 50-mile radius from the plant. The releases are based on MACCS2 calculations for Hatch that were also used to support the Hatch Severe Accident Mitigation Alternative (SAMA) evaluation and submittal.

From the data section of this calculation, the person-rem (population dose) taken out to 50 miles is based on either: (1) Hatch specific MACCS2 calculations for severe accident end states for a failed containment; or, (2) the design-basis containment leak rate of 1.2%/day (or $1L_a$). This latter value is used to predict the person-rem dose for accident Classes 1 and 3 as follows:

Class 1	=	1963 person-rem (at $1.0L_a$)	=	1963 person-rem ⁽¹⁾
Class 2	=	$1.15E+6$ ⁽²⁾		
Class 3a	=	1963 person-rem x $10L_a$	=	19,630 person-rem ⁽³⁾
Class 3b	=	1963 person-rem x $35L_a$	=	68,705 person-rem ⁽³⁾
Class 4	=	Not analyzed		
Class 5	=	Not analyzed		
Class 6	=	$1.15E+6$ person-rem ⁽⁴⁾		
Class 7a	=	$1.06E+6$ person-rem ⁽⁵⁾		
Class 7b	=	$5.7E+5$ person-rem		
Class 8	=	$1.15E+6$ person-rem ⁽⁶⁾		

⁽¹⁾ The population dose associated with the Technical Specification Leakage is based on use of the ex-plant consequence calculation for the Mark I containment in NUREG-1150. The derivation is described in Section 4.2 for the Hatch using the NUREG-1150 information scaled by population and allowable Tech Spec Leakage.

⁽²⁾ Class 2 (Containment Isolation failures) may be drywell isolation failures. No specific MACCS2 calculation is available. Therefore, the containment bypass MACCS2 calculation is conservatively used to represent this accident class.

⁽³⁾ The population dose for Technical Specification Leakage is derived as discussed in Note (1) and the Class 3a and 3b releases are related to the Technical Specification Leakage rate as shown. This is consistent with the Indian Point 3 ILRT submittal. [14]

⁽⁴⁾ No available MACCS2 calculation is available for Isolation failure. Therefore, the containment bypass dose estimate is conservatively used to represent these failures.

- (5) For Class 7, the person-rem dose associated various contributors to the Class 7 varied from $7\text{E}+5$ to $1.06\text{E}+6$ person-rem. Either a weighted average or the maximum person-rem could be used. For this bounding assessment, the maximum person-rem dose of the contributing sequences is used.
- (6) Class 8 sequences involve containment bypass failures; as a result, the person-rem dose is not based on normal containment leakage. The releases for this class are expected to be released directly to the environment. Based on MACCS2 evaluations, the value used is $1.15\text{E}+6$ person-rem.

The population dose estimates derived for use in the risk evaluation are summarized in Table 5-3.

Table 5-3
HATCH POPULATION DOSE ESTIMATES FOR POPULATION WITHIN 50 MILES

Accident Classes (Containment Release Type)	Description	Person-Rem (50 miles)
1	No Containment Failure	1963
2	Large Isolation Failures (Failure to Close)	$1.15\text{E}+6^{(1)}$
3a	Small Isolation Failures (liner breach)	19,630
3b	Large Isolation Failures (liner breach)	68,705
4	Small Isolation Failures (Failure to seal—Type B)	NA
5	Small Isolation Failures (Failure to seal—Type C)	NA
6	Other Isolation Failures (e.g., dependent failures)	$1.15\text{E}+6^{(1)}$
7a	Failures Induced by Phenomena (Early)	$1.06\text{E}+6^{(1)}$
7b	Failures Induced by Phenomena (Late) ⁽²⁾	$5.7\text{E}+5^{(1)(2)}$
8	Bypass (Interfacing System LOCA)	$1.15\text{E}+6^{(1)}$

⁽¹⁾ The person-rem is calculated from MACCS2 calculations performed for the SAMA evaluation and the power uprate condition. The table from RAI#5 as clarified and shown in Table 4.2-3 is used as the basis.

⁽²⁾ Late Release Evaluation based on Table 4.2-3 person rem/yr estimate [21] and the accident sequence frequency of $2.0\text{E}-7/\text{yr}$ yields $5.7\text{E}+5$ person-rem.

The above results when combined with the results presented in Table 5-2 yield the Hatch baseline mean consequence measures for each accident class. These results are presented in Table 5-4.

Table 5-4

ANNUAL DOSE (PERSON-REM/YR)⁽¹⁾ AS A FUNCTION OF
ACCIDENT CLASS CHARACTERISTIC OF CONDITIONS
FOR ILRT REQUIRED 3/10 YEARS
(I.E., REPRESENTATIVE OF ILRT DATA)

Accident Classes (Containment Release Type)	Description	Frequency (per Rx-yr)	Person- Rem (50 miles)	Person- Rem/yr (50 miles)
1	No Containment Failure ⁽²⁾	9.06E-6	1963	1.78E-2
2	Large Isolation Failures (Failure to Close)	5.5E-9	1.15E+6	6.32E-3
3a	Small Isolation Failures (liner breach)	7.9E-7	19,630	1.55E-2
3b	Large Isolation Failures (liner breach)	2.6E-7	68,705	1.79E-2
4	Small Isolation Failures (Failure to seal—Type B)	NA	NA	NA
5	Small Isolation Failures (Failure to seal—Type C)	NA	NA	NA
6	Other Isolation Failures (e.g., dependent failures)	5.0E-9	1.15E+6	5.75E-3
7a	Failures Induced by Phenomena (Early)	2.0E-6	1.06E+6	2.12
7b	Failures Induced by Phenomena (Late)	1.1E-7	5.7E+5	6.27E-2
8	Bypass (Interfacing System LOCA)	1.65E-7	1.15E+6	1.90E-1
CDF	All CET End states (including very low and no release)	1.24E-5		2.436

⁽¹⁾ As noted earlier, the Hatch PRA has been updated since the SAMA evaluation and the Level 1 accident sequence frequencies are generally slightly lower. This results in reductions in the radionuclide release frequencies from the containment and the total calculated person rem/year when compared with the SAMA results discussed in Section 4 and shown in Table 4.2-3.

⁽²⁾ Characterized as 1L₁ release magnitude consistent with the derivation of the ILRT non-detection failure probability for ILRTs. Release Category 3a and 3b include failures of containment to meet the Technical Specification leak rate.

Because of the relatively small population, the total dose per year is relatively low compared with the other sites as shown below:

Plant	Annual Dose (Person-Rem/Yr)	Reference
Indian Point 3	14,515	14
Peach Bottom	6.2	15
Crystal River	1.4	16
Hatch	2.4	Table 5-4

Based on the risk values from Table 5-4, the percent risk contribution ($\%Risk_{BASE}$) for Class 3 (i.e., the Class affected by the ILRT interval change) is as follows:

$$\%Risk_{BASE} = [(CLASS3a_{BASE} + CLASS3b_{BASE}) / Total_{BASE}] \times 100$$

Where:

$CLASS3a_{BASE}$ = Class 3a person-rem/year = $1.55E-2$ person-rem/year [Table 5-4]

$CLASS3b_{BASE}$ = Class 3b person-rem/year = $1.79E-2$ person-rem/year [Table 5-4]

$TOTAL_{BASE}$ = Total person-rem/yr for baseline interval = 2.436 person-rem/yr [Table 5-4]

$$\%Risk_{BASE} = [(1.55E-2 + 1.79E-2) / 2.436] = (3.34E-2) / 2.436$$

$$\%Risk_{BASE} = 1.37\%$$

5.3 STEP 3 - EVALUATE RISK IMPACT OF EXTENDING TYPE A TEST INTERVAL FROM 10-TO-15 YEARS

According to NUREG-1493 [4], relaxing the Type A ILRT interval from 3-in-10 years to 1-in-10 years will increase the average time that a leak detectable only by an ILRT goes undetected from 1.5 years to 5 years. The average time for failure to detect is calculated using the approximation $\frac{1}{2} \lambda T$ where T is the Test Interval and λ , the leakage failure rate, is (3%)/1.5 year. If the test interval is extended to 1 in 15 years, the

average time that a leak detectable only by an ILRT test goes undetected increases to 7.5 years ($1/2 * 15$ years). Because ILRTs only detect about 3% of leaks (the rest are identified during LLRTs), the result for a 10-yr ILRT interval is a 10% undetectable rate in the overall probability of leakage $\frac{1}{2} * \frac{3\%}{1.5 \text{ yrs}} * 10 \text{ years}$.

This value is determined by multiplying 3% and the ratio of the average time for non-detection for the increased ILRT test interval to the baseline average time for non-detection. For a 15-yr-test interval, the result is a 15% overall probability of leakage (i.e., $\frac{1}{2} * \frac{3\%}{1.5 \text{ yrs}} * 15 \text{ years}$). Thus, increasing the ILRT test interval from 10 years to 15 years results in a 5% increase in the overall probability of leakage.

Risk Impact due to 10-year Test Interval

As previously stated, Type A tests impact only Class 3 sequences. For Class 3 sequences, the release magnitude is not impacted by the change in test interval, (a small or large breach remains the same, even though the probability of not detecting the breach increases). Thus, only the frequency of Class 3 sequences are impacted. Therefore, for Class 3 sequences, the risk contribution is determined by multiplying the Class 3 accident frequency by the increase in probability of leakage of 1.1 (7% which is approximated here as a factor of 1.1 consistent with the approach used by Indian Point 3 [14]). Specifically, there is a factor of 1.1 increase in Class 3a and 3b frequencies relative to the baseline associated with increasing the ILRT test interval from 3 yrs to 10 yrs. (See Section 4.4.) The results of this calculation are presented in Table 5-5. Based on the Table 5-5 values, the Type A 10-year test frequency percent risk contribution (%Risk₁₀) for Class 3 is as follows:

$$(\%Risk_{10}) = [(CLASS3a_{10} + CLASS3b_{10}) / Total_{10}] * 100$$

Where:

CLASS3a₁₀ = Class 3a person-rem/year = 1.71E-2 person-rem/year [Table 5-5]

CLASS3b₁₀ = Class 3b person-rem/year = 1.96E-2 person-rem/year [Table 5-5]

TOTAL₁₀ = Total person-rem/yr for 10-year interval = 2.439 person-rem/yr [Table 5-5]

$$\%Risk_{10} = [(1.71E-2 + 1.96E-2) / 2.439] \times 100 = (3.67E-2) / 2.439 \times 100$$

$$\%Risk_{10} = 1.5\%$$

Therefore, the Total Type A 10-year ILRT interval risk contribution of leakage, represented by Class 3 accident scenarios is 1.5%.

The percent risk increase ($\Delta\%Risk_{10}$) due to a ten-year ILRT over the baseline case is as follows:

$$\Delta\%Risk_{10} = [(Total_{10} - Total_{BASE}) / Total_{BASE}] \times 100.0$$

TOTAL_{BASE} = Total person-rem/yr for baseline interval = 2.436 person-rem/yr [Table 5-5]

TOTAL₁₀ = Total person-rem/yr for 10 yr ILRT interval = 2.439 person-rem/yr [Table 5-5]

$$\Delta\%Risk_{10} = [(2.439 - 2.436) / 2.436] \times 100.0$$

$$\Delta\%Risk_{10} = 0.12\%$$

Therefore, the increase in risk contribution because of the change to the already approved ten-year ILRT test frequency from three-in-ten-years to 1-in-ten-years is 0.12%.

Table 5-5
ANNUAL DOSE (PERSON-REM/YR) AS A FUNCTION OF
ACCIDENT CLASS CHARACTERISTIC OF CONDITIONS
FOR ILRT REQUIRED EVERY 10 YEARS ⁽²⁾

Accident Classes (Containment Release Type)	Description	Frequency (per Rx-yr)	Person- Rem (50 miles)	Person- Rem/yr (50 miles)
1	No Containment Failure ⁽¹⁾	8.97E-6	1963	1.76E-2
2	Large Isolation Failures (Failure to Close)	5.5E-9	1.15E+6	6.32E-3
3a	Small Isolation Failures (liner breach)	8.69E-7	19,630	1.71E-2
3b	Large Isolation Failures (liner breach)	2.86E-7	68,705	1.96E-2
4	Small Isolation Failures (Failure to seal—Type B)	NA	NA	NA
5	Small Isolation Failures (Failure to seal—Type C)	NA	NA	NA
6	Other Isolation Failures (e.g., dependent failures)	5.0E-9	1.15E+6	5.75E-3
7a	Failures Induced by Phenomena (Early)	2.0E-6	1.06E+6	2.12
7b	Failures Induced by Phenomena (Late)	1.1E-7	5.7E+5	6.27E-2
8	Bypass (Interfacing System LOCA)	1.65E-7	1.15E+6	1.90E-1
CDF	All CET End states (including very low and no release)	1.24E-5		2.439

⁽¹⁾ Characterized as 1L₀ release magnitude consistent with the derivation of the ILRT non-detection failure probability for ILRTs.

⁽²⁾ A 10% increase in Classes 3a and 3b frequencies are used consistent with the method developed by EPRI [2] and [14].

Risk Impact Due to 15-Year Test Interval

The risk contribution for a 15-year interval is calculated in a manner similar to the 10-year interval. The difference is in the increase in probability of leakage in Classes 3a and 3b. For this case, the value used in the analysis is 15 percent or 1.15 consistent with previously approved method [14,22]. Specifically, there is a factor of 1.15 increase in Class 3a and 3b frequencies relative to the baseline associated with increasing the ILRT test interval from 3 yrs to 15 yrs. (See Section 4.4.) The results for this calculation are presented in Table 5-6.

Based on the values from Table 5-6, the Type A 15-year test frequency percent risk contribution ($\%Risk_{15}$) for Class 3 is as follows:

$$(\%Risk_{15}) = [(CLASS3a_{15} + CLASS3b_{15}) / Total_{15}] \times 100$$

Where:

CLASS3a₁₅ = Class 3a person-rem/year = 1.78E-2 person-rem/year [Table 5-6]

CLASS3b₁₅ = Class 3b person-rem/year = 2.06E-2 person-rem/year [Table 5-6]

TOTAL₁₅ = Total person-rem/yr for 15-year interval = 2.4407 person-rem/yr [Table 5-6]

$$\%Risk_{15} = [(1.78E-2 + 2.06E-2) / 2.4407] \times 100 = (3.84E-2) / 2.4407 \times 100$$

$$\%Risk_{15} = 1.57\%$$

Therefore, the Total Type A 15-year ILRT interval risk contribution of leakage, represented by Class 3 accident scenarios is 1.57%.

The percent increase in risk (in terms of person-rem/yr) of these associated specific sequences when the ILRT test interval is increased from 10 years to 15 years is computed as follows:

Table 5-6

ANNUAL DOSE (PERSON-REM/YR) AS A FUNCTION OF
ACCIDENT CLASS CHARACTERISTIC OF CONDITIONS
FOR ILRT REQUIRED EVERY 15 YEARS⁽²⁾

Accident Classes (Containment Release Type)	Description	Frequency (per Rx-yr)	Person- Rem (50 miles)	Person- Rem/yr (50 miles)
1	No Containment Failure ⁽¹⁾	8.91E-6	1963	1.75E-2
2	Large Isolation Failures (Failure to Close)	5.5E-9	1.15E+6	6.32E-3
3a	Small Isolation Failures (liner breach)	9.09E-7	19,630	1.78E-2
3b	Large Isolation Failures (liner breach)	3.00E-7	68,705	2.06E-2
4	Small Isolation Failures (Failure to seal—Type B)	NA	NA	NA
5	Small Isolation Failures (Failure to seal—Type C)	NA	NA	NA
6	Other Isolation Failures (e.g., dependent failures)	5.0E-9	1.15E+6	5.75E-3
7a	Failures Induced by Phenomena (Early)	2.0E-6	1.06E+6	2.12
7b	Failures Induced by Phenomena (Late)	1.1E-7	5.7E+5	6.27E-2
8	Bypass (Interfacing System LOCA)	1.65E-7	1.15E+6	1.90E-1
CDF	All CET End states (including very low and no release)	1.24E-5		2.4407

⁽¹⁾ Characterized as 1L₁ release magnitude consistent with the derivation of the ILRT non-detection failure probability for ILRTs.

⁽²⁾ A 15% increase in Classes 3a and 3b frequencies are used consistent with the method developed by IP3 [14] based on EPRI evaluation [2]. This results in a 5% delta risk in Classes 3a and 3b when comparing the risk associated with the 10-year period for the ILRT to that of a 15-year ILRT period.

$$\%Risk_{10-15} = [(PER-REM_{15} - PER-REM_{10}) / PER-REM_{10}] \times 100$$

Where:

$$\begin{aligned} PER-REM_{10} &= \text{person-rem/year of ten years interval (for Classes 3a and 3b)} \\ &= 3.67E-2 \text{ person-rem/yr} \end{aligned}$$

$$\begin{aligned} PER-REM_{15} &= \text{person-rem/year of fifteen years interval (for Classes 3a and 3b)} \\ &= 3.84E-2 \text{ person-rem/yr} \end{aligned}$$

$$\%Risk_{10-15} = [(3.84E-2 - 3.67E-2) / 3.67E-2] \times 100$$

$$\%Risk_{10-15} = 4.6\%$$

Therefore, the change in Type A test frequency from once-per-ten-years to once-per-fifteen-years increases the risk of those associated specific accident sequences of Class 3 by 4.6%.

However, the more appropriate comparison is the change in the total integrated plant risk. The percent increase on the total integrated plant risk when the ILRT is extended from 10 years to 15 years is computed as follows:

$$\%TOTAL_{10-15} = [(TOTAL_{15} - TOTAL_{10}) / TOTAL_{10}] \times 100$$

Where:

$$TOTAL_{10} \quad \text{Total person-rem/year for 10-year interval} = 2.439 \text{ person-rem/year} \\ \text{[Table 5-5]}$$

$$TOTAL_{15} \quad \text{Total person-rem/year for 15-year interval} = 2.4407 \text{ person-rem/year} \\ \text{[Table 5-5]}$$

$$\%TOTAL_{10-15} = [(2.4407 - 2.439) / 2.439] \times 100$$

$$\%TOTAL_{10-15} = 0.07\%$$

Therefore, the risk impact on the total integrated plant risk for these accident sequences influenced by Type A testing is only 0.07%.

The percent risk increase (ΔRisk_{15}) due to a fifteen-year ILRT over the baseline case is as follows:

$$\Delta\text{Risk}_{15} = [(\text{Total}_{15} - \text{Total}_{\text{BASE}}) / \text{Total}_{\text{BASE}}] \times 100.0$$

Where:

$\text{TOTAL}_{\text{BASE}}$ = Total person-rem/year for baseline interval = 2.436 person-rem/year [Table 5-5]

TOTAL_{15} = Total person-rem/year for 15-year interval = 2.4407 person-rem/year [Table 5-5]

$$\% \Delta\text{Risk}_{\text{BASE-15}} = [(2.4407 - 2.436) / 2.436] \times 100$$

$$\% \Delta\text{Risk}_{\text{BASE-15}} = 0.19\%$$

Therefore, the total increase in risk contribution associated with relaxing the ILRT test frequency from three in ten years to once-per-fifteen years is 0.19%.

5.4 STEP 4 - DETERMINE THE CHANGE IN RISK IN TERMS OF LARGE EARLY RELEASE FREQUENCY (LERF)

The risk increase associated with extending the ILRT interval involves the potential that a core damage event that normally would result in only a small radioactive release from an intact containment could in fact result in a larger release due to the increase in probability of failure to detect a pre-existing leak. Class 3b radionuclide release person-rem is significantly less than a typical LERF contributor as seen by comparing the relative population dose for Class 3b/Class 7 ($6.87\text{E}+4$ person-rem/ $1.06\text{E}+6$ person-rem) or 6.5%. Nevertheless, Class 3b is treated in this analysis as a potential LERF contributor. Class 3a is even less than Class 3b and is treated here as not a "large" release. Therefore, for this evaluation, only Class 3b sequences have the potential to result in

large releases if a pre-existing leak were present. Class 1 sequences are not considered as potential large release pathways because the containment remains intact. Therefore, the containment leak rate is expected to be small. Other accident classes 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 a LERF. (See also the discussion in Section 5.5 regarding the conditional containment failure probability to assess the defense-in-depth.) Therefore, the frequency of Class 3B sequences is used as the LERF estimate. This frequency, based on a ten-year test interval, is $2.86\text{E-}7/\text{yr}$.

Reg. Guide 1.174 [17] 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 core damage frequency (CDF) below $10^{-8}/\text{yr}$ and increases in LERF below $10^{-7}/\text{yr}$. Because the ILRT does not impact CDF, the relevant metric is LERF. Calculating the increase in LERF requires determining the impact of the ILRT interval on the leakage probability.

As described in Step 3, extending the ILRT interval from once-per-10 years to once-per-15 years will increase the average time that a leak detectable only by an ILRT goes undetected from 60 to 90 months. ILRTs only detect about 3% of leaks (the rest are identified during LLRTs). Increasing the ILRT test interval from 10 to 15 years results in a 5% increase in the overall probability of leakage. Multiplying the 10-year interval LERF frequency ($2.86\text{E-}7/\text{yr}$) by the increase in overall probability of leakage (0.05) gives an increase in LERF of $1.43\text{E-}8/\text{yr}$. Guidance in Reg. Guide 1.174 defines very small changes in LERF as below $1\text{E-}7/\text{yr}$. Therefore, using this NRC guidance, increasing the ILRT interval from the current authorized 10 years to 15 years represents a very small change in risk.

It should be noted that if the risk increase is measured from the original 3-in-10 year interval, the increase in LERF is $2.86\text{E-}7/\text{yr}$ multiplied by the 12% incremental increase in overall probability for a fifteen-year test interval (i.e., 15% - 3%) is $3.4\text{E-}8/\text{yr}$, which is also well below the $1.0\text{E-}7/\text{yr}$ screening criterion in Reg. Guide 1.174 and represents a very small change in risk.

5.5 IMPACT ON THE CONDITIONAL CONTAINMENT FAILURE PROBABILITY (CCFP)

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. One of the difficult aspects of this calculation is providing a definition of the "failed containment." In this assessment, the CCFP is defined such that containment failure includes all radionuclide release end states other than the intact state. The conditional part of the definition is conditional given a severe accident (i.e., core damage).

Because the only classes that are increasing are Classes 3a and 3b, the change in CCFP can be calculated by the difference in these classes.

$$\begin{aligned}\Delta\text{CCFP} &= \text{CCFP}_{15} - \text{CCFP}_{10} = \frac{(\text{Class 3a} + \text{Class 3b})_{15} - (\text{Class 3a} + \text{Class 3b})_{10}}{\text{CDF}} \\ &= 0.435\%\end{aligned}$$

This change in CCFP of less than 1% is judged to be insignificant and reflects sufficient defense-in-depth.

5.6 RESULTS SUMMARY

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

1. The baseline risk contribution (person-rem) associated with containment leakage affected by the ILRT and represented by Class 3 accident scenarios is 1.37% of the total risk. The majority of the risk (98%) is associated with severe accident phenomena during core melt progression.
2. When the ILRT interval is 10 years, the risk contribution of leakage (person-rem) represented by Class 3 accident scenarios is increased to 1.5% of the total risk.
3. When the ILRT interval is 15 years, the risk contribution of leakage represented by Class 3 accident scenarios is increased to 1.57% of the total risk.
4. The person-rem/year increase in risk contribution based solely on the affected sequences (Class 3) from extending the ILRT test frequency from the current once-per-ten-year frequency to once-per-fifteen years is 4.6%.
5. The total integrated increase in risk contribution from reducing the ILRT test frequency from the current once-per-10-year frequency to once-per-15 years is 0.07%.
6. There is no change in the at-power CDF associated with the ILRT extension. Therefore, this is within the Reg. Guide 1.174 acceptance guidelines.
7. The risk increase in LERF from reducing the ILRT test frequency from the current once-per-10 years to once-per-15 years is $1.43\text{E-}8$. This is determined to be very small using the acceptance guidelines of Reg. Guide 1.174.
8. The risk increase in LERF from the original 3-in-10 years test frequency, to once-per-15 years is $3.14\text{E-}8/\text{yr}$. This is also found to be "very small" using the acceptance guidelines in Reg. Guide 1.174.
9. This change in CCFP of less than 1% is judged to be insignificant and reflects sufficient defense-in-depth.
10. Other salient results are summarized in Table 5-7.

Table 5-7

SUMMARY OF RISK IMPACT ON TYPE A ILRT TEST FREQUENCY

Class ⁽¹⁾	Risk Impact (Base) ⁽²⁾	Risk Impact (10-years) ⁽³⁾	Risk Impact (15-years) ⁽⁴⁾
3a and 3b	1.37% of integrated value 3.34E-2 person-rem/yr	1.50% of integrated value 3.67E-2 person-rem/yr	1.57% of integrated value 3.84E-2 person-rem/yr
Total Integrated Risk	2.436 person-rem/year	2.439 person-rem/year	2.4407 person-rem/year
Reference	Section 5.2	Section 5.3	Section 5.3

⁽¹⁾ Only accident sequences increased by a change in Type A test frequency are evaluated. These are sequences 3A and 3B.

⁽²⁾ Hatch IPE baseline values.

⁽³⁾ Type A ILRT test frequency of 1-in-10-years

⁽⁴⁾ Type A ILRT test frequency of 1-in-15-years

Section 6

CONCLUSIONS

This section provides the principal conclusions of the ILRT test interval extension risk assessments as reported for the following:

- Previous generic risk assessment by the NRC
- Plant Specific Hatch risk assessment for the at-power case
- General conclusions regarding the beneficial effects on shutdown risk

6.1 PREVIOUS ASSESSMENTS

The NRC in NUREG-1493 has previously concluded that:

- Reducing the frequency of Type A tests (ILRTs) from the current three per 10 years to one per 20 years was found to lead to an imperceptible increase in risk. The estimated increase in risk is very small because ILRTs identify only a few potential containment leakage paths that cannot be identified by Type B and C testing, and the leaks that have been found by Type A tests have been only marginally above 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 liner.

6.2 HATCH SPECIFIC RISK RESULTS

The findings for Hatch confirm the general findings of previous studies on a plant specific basis considering the severe accidents evaluated for Hatch, the Hatch containment failure modes, the Hatch Technical Specification allowed leakage, and the local population surrounding Hatch.

Based on the results from Section 5, the following conclusions regarding the assessment of the plant risk are associated with extending the Type A ILRT test from ten years to fifteen years:

- There is no change in the at-power CDF associated with the ILRT test interval extension. Therefore, this is within the Reg. Guide 1.174 acceptance guidelines.
- Reg. Guide 1.174 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 $10^{-6}/\text{yr}$ and increases in LERF below $10^{-7}/\text{yr}$. 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 frequency from once-per-ten years to once-per-fifteen years is $1.43\text{E-}8/\text{yr}$. Guidance in Reg. Guide 1.174 defines very small changes in LERF as below $1\text{E-}7/\text{yr}$. Therefore, increasing the ILRT interval from 10 to 15 years is considered to result in a very small change to the Hatch risk profile.
- The change in Type A test frequency from once-per-ten-years to once-per-fifteen-years increases the total integrated plant risk by only 0.07%. Therefore, the risk impact change when compared to other severe accident risks is negligible.
- This change in Conditional Containment Failure Probability (CCFP) of less than 1% is judged to be insignificant and reflects sufficient defense-in-depth.

6.3 RISK TRADE-OFF

The performance of an ILRT occurs during plant shutdown and introduces some small residual risk. An EPRI study of operating experience events associated with the performance of ILRTs has indicated that there are real shutdown risk impacts associated with the setup and performance of the ILRT during shutdown operation. [10] While these risks have not been quantified for Hatch, 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 test interval extension, there are, in fact, positive safety benefits that reduce the already small risk associated with the extension of the ILRT test interval.

Section 7

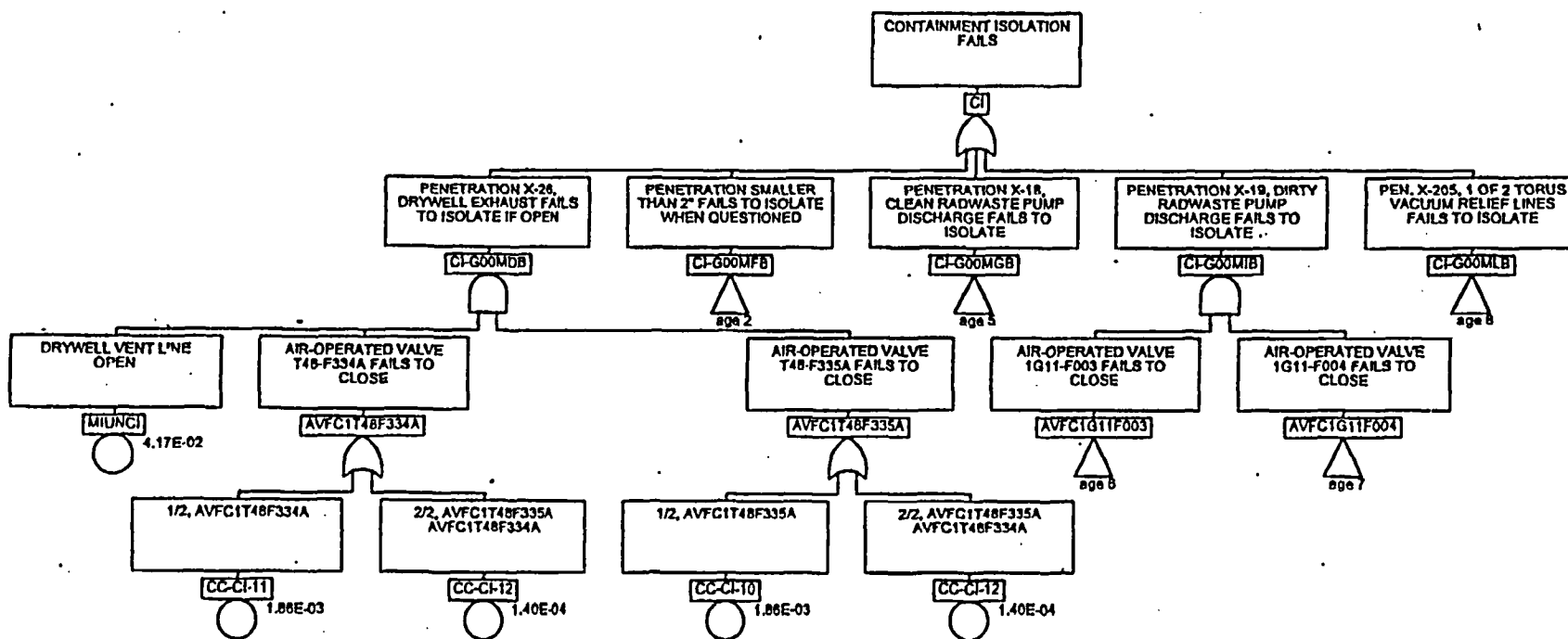
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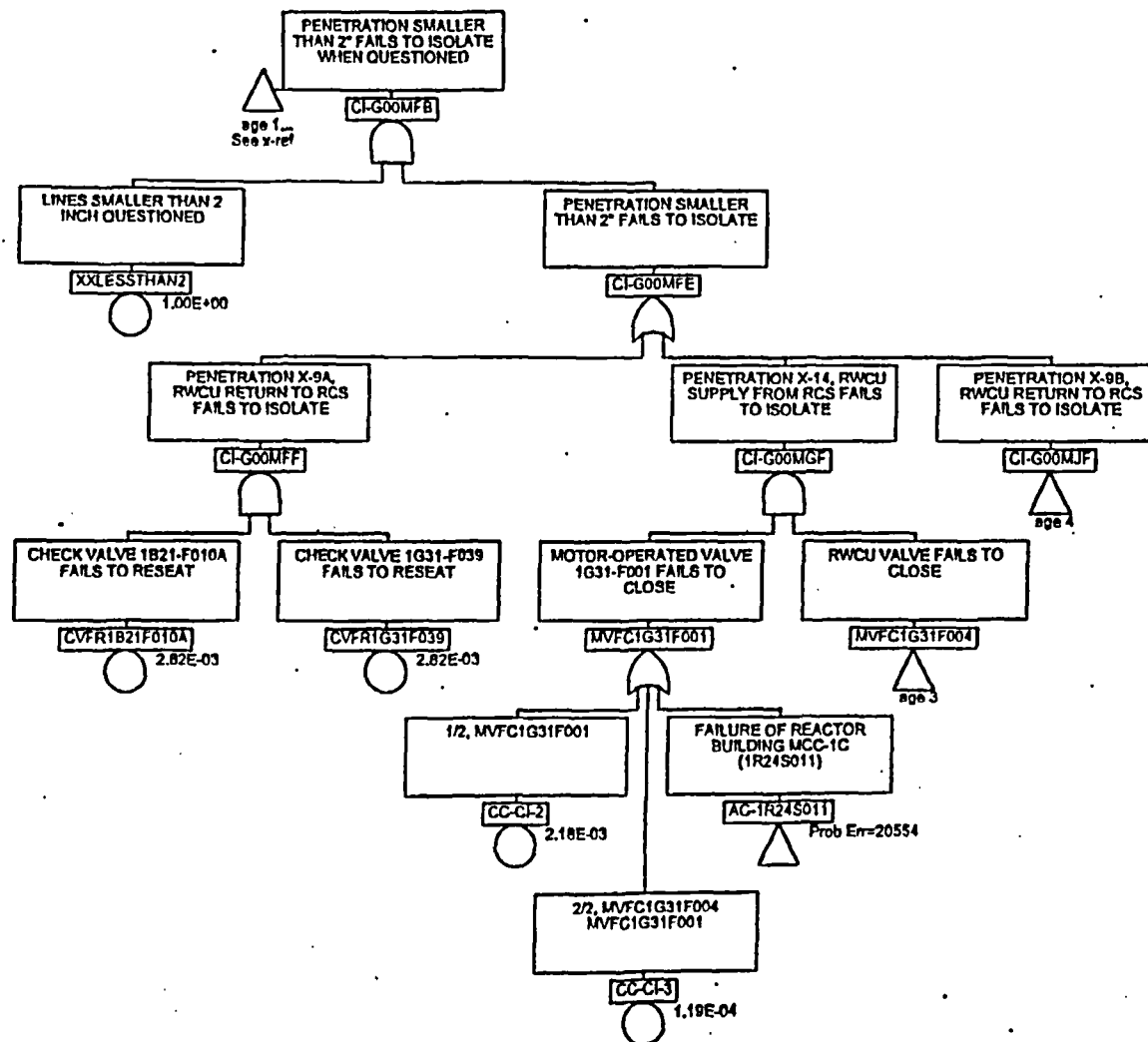
- [1] *Industry Guideline for Implementing Performance-Based Option of 10 CFR Part 50, Appendix J*, NEI 94-01, July 1995.
- [2] *Risk Impact Assessment of Revised Containment Leak Rate Testing Intervals*, EPRI, Palo Alto, CA EPRI TR-104285, August 1994.
- [3] *An Approach for Using Probabilistic Risk Assessment in Risk-Informed Decisions on Plant-Specific Changes to the Licensing Basis*, Regulatory Guide 1.174, July 1998.
- [4] *Performance-Based Containment Leak-Test Program*, NUREG-1493, September 1995.
- [5] *Evaluation of Severe Accident Risks: Peach Bottom, Unit 2*, Main Report NUREG/CR-4551, SAND86-1309, Volume 4, Revision 1, Part 1, December 1990.
- [6] *Technical Findings and Regulatory Analysis for Generic Safety Issue II.E.4.3 'Containment Integrity Check'*, NUREG-1273, April 1988.
- [7] *Impact of Containment Building Leakage on LWR Accident Risk*, Oak Ridge National Laboratory, NUREG/CR-3539, ORNL/TM-8964, April 1984.
- [8] *Reliability Analysis of Containment Isolation Systems*, Pacific Northwest Laboratory, NUREG/CR-4220, PNL-5432, June 1985.
- [9] *Review of Light Water Reactor Regulatory Requirements*, Pacific Northwest Laboratory, NUREG/CR-4330, PNL-5809, Vol. 2, June 1986.
- [10] *Shutdown Risk Impact Assessment for Extended Containment Leakage Testing Intervals Utilizing ORAM™*, EPRI, Palo Alto, CA TR-105189, Final Report, May 1995.
- [11] *Individual Plant Examination Peach Bottom Atomic Power Station Units 2 and 3*, Volumes 1 and 2, Philadelphia Electric Company, 1992.
- [12] *ALWR Severe Accident Dose Analysis*. DE-ACOG-87RL11313, March 1989.
- [13] Patrick D.T. O'Connor, *Practical Reliability Engineering*, John Wiley & Sons, 2nd Edition, 1985.

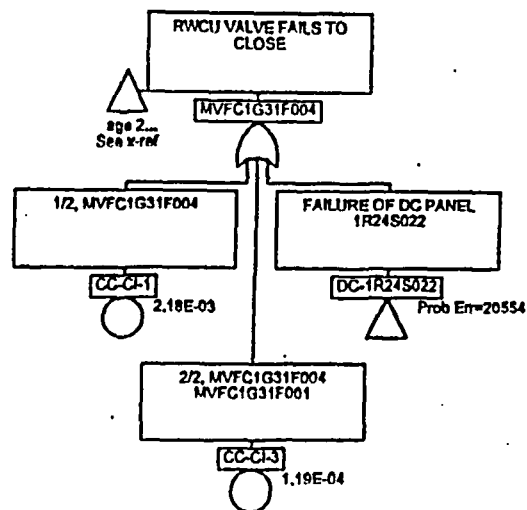
- [14] Letter from R.J. Barrett (Entergy) to U.S. Nuclear Regulatory Commission, IPN-01-007, dated January 18, 2001.
- [15] Letter from J.A. Hutton (Exelon, Peach Bottom) to U.S. Nuclear Regulatory Commission, Docket No. 50-278, License No. DPR-56, LAR 01-00430, dated May 30, 2001.
- [16] Letter from D.E. Young (Florida Power) to U.S. Nuclear Regulatory Commission, 3F0401-11, dated April 25, 2001.
- [17] Regulatory Guide 1.174, *An Approach for Using Probabilistic Risk Assessment in Risk-Informed Decisions on Plant-Specific Changes to the Licensing Basis*, July 1998.
- [18] Burns, T.J., *Impact of Containment Building Leakage on LWR Accident Risk*, Oak Ridge National Laboratory, NUREG/CR-3539, April 1984.
- [19] United States Nuclear Regulatory Commission, Reactor Safety Study, WASH-1400, October 1975.
- [20] Letter from SNC (H.L. Summer, Jr.) to USNRC dated July 26, 2000.
- [21] Letter from SNC (H.L. Summer, Jr.) to USNRC, HL-5982.
- [22] 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.
- [23] *Severe Accident Risks: An Assessment for Five U.S. Nuclear Power Plants*, NUREG -1150, December 1990.

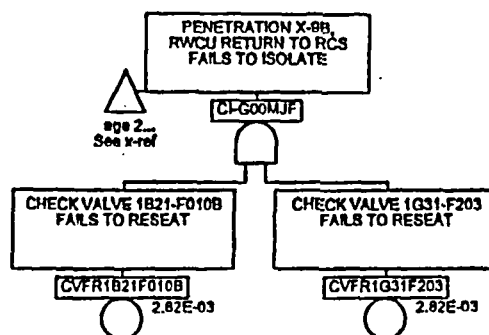
Attachment A

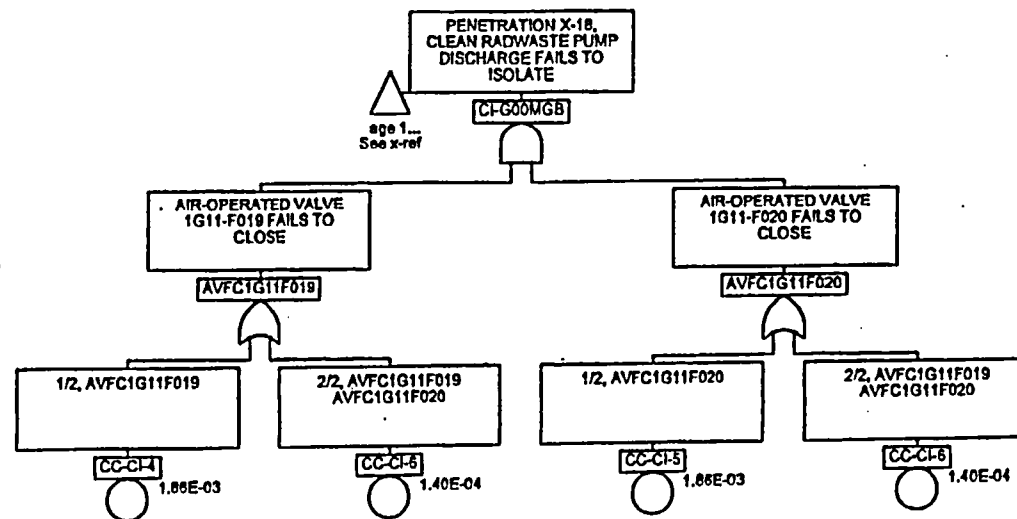
***CONTAINMENT ISOLATION FAULT
TREE***

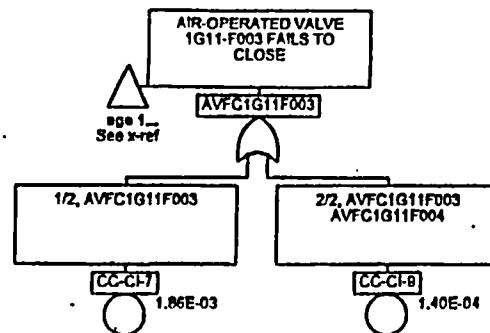


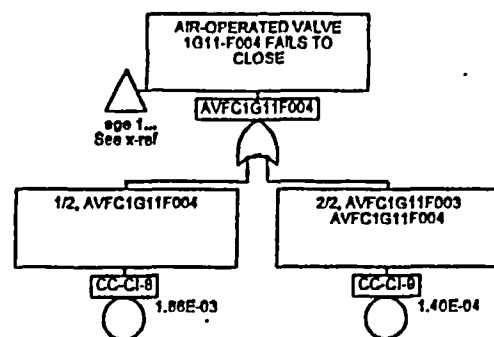


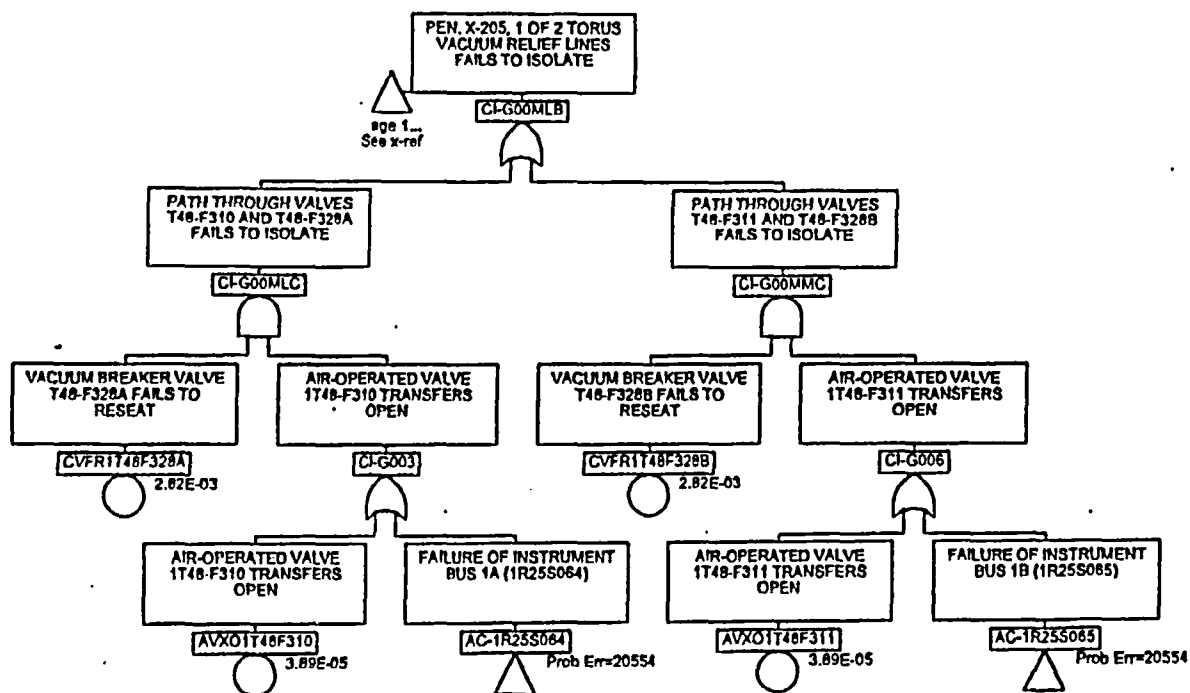












Name	Page	Zone	Name	Page	Zone	Name	Page	Zone
AC-1R24S011	2	4	CVFR1T48F328A	8	1			
AC-1R25S064	8	3	CVFR1T48F328B	8	3			
AC-1R25S065	8	5	DC-1R24S022	3	2			
AVFC1G11F003	1	5	MIUNCI	1	1			
AVFC1G11F003	6	2	MVFC1G31F001	2	3			
AVFC1G11F004	1	6	MVFC1G31F004	2	4			
AVFC1G11F004	7	2	MVFC1G31F004	3	2			
AVFC1G11F019	5	2	XXLESSTHAN2	2	1			
AVFC1G11F020	5	4						
AVFC1T48F334A	1	2						
AVFC1T48F335A	1	4						
AVXO1T48F310	8	2						
AVXO1T48F311	8	4						
CC-CI-1	3	1						
CC-CI-10	1	4						
CC-CI-11	1	2						
CC-CI-12	1	3						
CC-CI-12	1	5						
CC-CI-2	2	3						
CC-CI-3	2	3						
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CI	1	5						
CI-G003	8	2						
CI-G006	8	4						
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CI-G00MFB	1	4						
CI-G00MFB	2	2						
CI-G00MFE	2	3						
CI-G00MFF	2	2						
CI-G00MGB	1	5						
CI-G00MGB	5	3						
CI-G00MGF	2	4						
CI-G00MIB	1	6						
CI-G00MJF	2	5						
CI-G00MJF	4	2						
CI-G00MLB	1	7						
CI-G00MLB	8	3						
CI-G00MLC	8	2						
CI-G00MMC	8	4						
CVFR1B21F010A	2	1						
CVFR1B21F010B	4	1						
CVFR1G31F039	2	2						
CVFR1G31F203	4	2						
				C:\CAFTA-WHATCH\CI.CAF		11/11/97		Page 9

Attachment B

CUTSETS FOR THE CONTAINMENT ISOLATION FAULT TREE

Cutsets with Descriptions Report

CI = 4.40E-04

#	Inputs	Description	Rate	Exposure	Event Prob	Probability
1	CC-CI-6	2/2, AVFC1G11F019 AVFC1G11F020		1.40E-04	1.40E-04	1.40E-04
2	CC-CI-9	2/2, AVFC1G11F003 AVFC1G11F004		1.40E-04	1.40E-04	1.40E-04
3	CC-RWISO-3	2/2, MVFC1G31F001 MVFC1G31F004		1.19E-04	1.19E-04	1.19E-04
	XXLESSTHAN2	LINES SMALLER THAN 2 INCH QUESTIONED		1.00E+00	1.00E+00	
4	CVFR1B21F010A	CHECK VALVE 1B21-F010A FAILS TO RESEAT	2.82E-03	1.00E+00	2.82E-03	7.95E-06
	CVFR1G31F039	CHECK VALVE 1G31-F039 FAILS TO RESEAT	2.82E-03	1.00E+00	2.82E-03	
	XXLESSTHAN2	LINES SMALLER THAN 2 INCH QUESTIONED		1.00E+00	1.00E+00	
5	CVFR1B21F010B	CHECK VALVE 1B21-F010B FAILS TO RESEAT	2.82E-03	1.00E+00	2.82E-03	7.95E-06
	CVFR1G31F203	CHECK VALVE 1G31-F203 FAILS TO RESEAT	2.82E-03	1.00E+00	2.82E-03	
	XXLESSTHAN2	LINES SMALLER THAN 2 INCH QUESTIONED		1.00E+00	1.00E+00	
6	%FL-BUSC	FLAG FOR LOSS OF 600-V BUS C INITIATING EVENT		1.00E+00	1.00E+00	6.26E-06
	BSSH1R23S003_I	600-V BUS C FAILS	3.76E-07	1.00E+00	1.00E+00	
	CC-RWISO-2	1/2, MVFC1G31F004		8.76E+03	3.29E-03	
	HATCHAVAIL	HATCH AVAILABILITY		2.18E-03	2.18E-03	
	XXLESSTHAN2	LINES SMALLER THAN 2 INCH QUESTIONED		8.72E-01	8.72E-01	
7	CC-CI-12	2/2, AVFC1T48F335A AVFC1T48F334A		1.40E-04	1.40E-04	5.82E-06
	MIUNCI	DRYWELL VENT LINE OPEN		4.17E-02	4.17E-02	
8	CC-RWISO-1	1/2, MVFC1G31F001		2.18E-03	2.18E-03	4.76E-06
	CC-RWISO-2	1/2, MVFC1G31F004		2.18E-03	2.18E-03	
	XXLESSTHAN2	LINES SMALLER THAN 2 INCH QUESTIONED		1.00E+00	1.00E+00	
9	CC-CI-4	1/2, AVFC1G11F019		1.86E-03	1.86E-03	3.44E-06
	CC-CI-5	1/2, AVFC1G11F020		1.86E-03	1.86E-03	
10	CC-CI-7	1/2, AVFC1G11F003		1.86E-03	1.86E-03	3.44E-06
	CC-CI-8	1/2, AVFC1G11F004		1.86E-03	1.86E-03	
11	%FL-BUSC	FLAG FOR LOSS OF 600-V BUS C INITIATING EVENT		1.00E+00	1.00E+00	3.16E-07
	BSSH1R23S003_I	600-V BUS C FAILS	3.76E-07	1.00E+00	1.00E+00	
	HATCHAVAIL	HATCH AVAILABILITY		8.72E-01	8.72E-01	
	MNUNRWISO_OUT	RWCU OUTBOARD MOV INOP DUE TO MAINTENANCE		1.10E-04	1.10E-04	
	XXLESSTHAN2	LINES SMALLER THAN 2 INCH QUESTIONED		1.00E+00	1.00E+00	
12	CC-RWISO-1	1/2, MVFC1G31F001		2.18E-03	2.18E-03	2.40E-07
	MNUNRWISO_OUT	RWCU OUTBOARD MOV INOP DUE TO MAINTENANCE		1.10E-04	1.10E-04	
	XXLESSTHAN2	LINES SMALLER THAN 2 INCH QUESTIONED		1.00E+00	1.00E+00	
13	CC-RWISO-2	1/2, MVFC1G31F004		2.18E-03	2.18E-03	2.40E-07
	MNUNRWISO_IN	RWCU INBOARD MOV INOP DUE TO MAINTENANCE		1.10E-04	1.10E-04	
	XXLESSTHAN2	LINES SMALLER THAN 2 INCH QUESTIONED		1.00E+00	1.00E+00	
14	%FL-BUSC	FLAG FOR LOSS OF 600-V BUS C INITIATING EVENT		1.00E+00	1.00E+00	1.62E-07
	BSSH1R23S003_I	600-V BUS C FAILS	3.76E-07	1.00E+00	1.00E+00	
	CVFR1T48F328A	VACUUM BREAKER VALVE T48-F328A FAILS TO RESEAT	2.82E-03	1.00E+00	2.82E-03	
	HATCHAVAIL	HATCH AVAILABILITY		8.72E-01	8.72E-01	
	OPHES064/S065	OPERATOR ACTION TO MANUALLY TRANSFER INSTRUMENT BUS POWER SUPPLIES		2.00E-02	2.00E-02	

#	Inputs	Description	Rate	Exposure	Event Prob	Probability
15	CC-CI-10	1/2, AVFC1T48F335A		1.86E-03	1.86E-03	1.44E-07
	CC-CI-11	1/2, AVFC1T48F334A		1.86E-03	1.86E-03	
	MIUNCI	DRYWELL VENT LINE OPEN		4.17E-02	4.17E-02	
16	AVXO1T48F310	AIR-OPERATED VALVE 1T48-F310 TRANSFERS OPEN	1.62E-06	2.40E+01	3.89E-05	1.10E-07
	CVFR1T48F328A	VACUUM BREAKER VALVE T48-F328A FAILS TO RESEAT	2.82E-03	1.00E+00	2.82E-03	
17	AVXO1T48F311	AIR-OPERATED VALVE 1T48-F311 TRANSFERS OPEN	1.62E-06	2.40E+01	3.89E-05	1.10E-07
	CVFR1T48F328B	VACUUM BREAKER VALVE T48-F328B FAILS TO RESEAT	2.82E-03	1.00E+00	2.82E-03	
18	%FL-BUSC	FLAG FOR LOSS OF 600-V BUS C INITIATING EVENT		1.00E+00	1.00E+00	5.12E-08
	CC-RWISO-2	1/2, MVFC1G31F004		2.18E-03	2.18E-03	
	HATCHAVAIL	HATCH AVAILABILITY		8.72E-01	8.72E-01	
	OPHEEPA	OPERATOR FAILS TO ALIGN 600-V BUS TO BACKUP 4160-V BUS		5.91E-03	5.91E-03	
	XROR1R23S003___I	STATION SERVICE TRANSFORMER C FAILS TO OPERATE	5.20E-07	8.76E+03	4.56E-03	
	XXLESSTHAN2	LINES SMALLER THAN 2 INCH QUESTIONED		1.00E+00	1.00E+00	
19	%FL-LOBUSE	FLAG FOR LOSS OF BUS E OR SUPPLY HARDWARE INITIATING EVENT		1.00E+00	1.00E+00	3.70E-08
	BSSH1R22S005___I	4KV BUS E FAILS TO OPERATE	3.76E-07	8.76E+03	3.29E-03	
	CC-RWISO-2	1/2, MVFC1G31F004		2.18E-03	2.18E-03	
	HATCHAVAIL	HATCH AVAILABILITY		8.72E-01	8.72E-01	
	OPHEEPA	OPERATOR FAILS TO ALIGN 600-V BUS TO BACKUP 4160-V BUS		5.91E-03	5.91E-03	
	XXLESSTHAN2	LINES SMALLER THAN 2 INCH QUESTIONED		1.00E+00	1.00E+00	
20	%FL-BUSD	FLAG FOR INITIATING EVENT CAUSED BY LOSS OF 600V BUS D		1.00E+00	1.00E+00	3.24E-08
	BSSH1R23S004___I	600-V BUS D FAILS DURING OPERATION	3.76E-07	8.76E+03	3.29E-03	
	CVFR1T48F328B	VACUUM BREAKER VALVE T48-F328B FAILS TO RESEAT	2.82E-03	1.00E+00	2.82E-03	
	HATCHAVAIL	HATCH AVAILABILITY		8.72E-01	8.72E-01	
	OPHES064/S065	OPERATOR ACTION TO MANUALLY TRANSFER INSTRUMENT BUS POWER SUPPLIES		2.00E-02	2.00E-02	
	XXBD TRANSIENT	LOSS OF BUS D CAUSES INITIATING EVENT (TRIP)		2.00E-01	2.00E-01	
21	%FL-BUSC	FLAG FOR LOSS OF 600-V BUS C INITIATING EVENT		1.00E+00	1.00E+00	2.59E-08
	BSSH1R22S017	DC SWITCHGEAR S017 FAILS DURING OPERATION	3.76E-07	2.40E+01	9.02E-06	
	BSSH1R23S003___I	600-V BUS C FAILS	3.76E-07	8.76E+03	3.29E-03	
	HATCHAVAIL	HATCH AVAILABILITY		8.72E-01	8.72E-01	
	XXLESSTHAN2	LINES SMALLER THAN 2 INCH QUESTIONED		1.00E+00	1.00E+00	
22	%FL-BUSC	FLAG FOR LOSS OF 600-V BUS C INITIATING EVENT		1.00E+00	1.00E+00	2.28E-08
	BSSH1R23S003___I	600-V BUS C FAILS	3.76E-07	8.76E+03	3.29E-03	
	HATCHAVAIL	HATCH AVAILABILITY		8.72E-01	8.72E-01	
	MCOR1R24S022	DC MCC S022 FAILS DURING OPERATION	3.31E-07	2.40E+01	7.94E-06	
	XXLESSTHAN2	LINES SMALLER THAN 2 INCH QUESTIONED		1.00E+00	1.00E+00	
23	BSSH1R22S017	DC SWITCHGEAR S017 FAILS DURING OPERATION	3.76E-07	2.40E+01	9.02E-06	1.97E-08
	CC-RWISO-1	1/2, MVFC1G31F001		2.18E-03	2.18E-03	
	XXLESSTHAN2	LINES SMALLER THAN 2 INCH QUESTIONED		1.00E+00	1.00E+00	
24	BSSH1R23S003	600-V BUS C FAILS	3.76E-07	2.40E+01	9.02E-06	1.97E-08
	CC-RWISO-2	1/2, MVFC1G31F004		2.18E-03	2.18E-03	
	XXLESSTHAN2	LINES SMALLER THAN 2 INCH QUESTIONED		1.00E+00	1.00E+00	
25	%FL-BUSC	FLAG FOR LOSS OF 600-V BUS C INITIATING EVENT		1.00E+00	1.00E+00	1.85E-08
	BSSH1R23S003___I	600-V BUS C FAILS	3.76E-07	8.76E+03	3.29E-03	
	C2XO1R22S017_4B	CIRCUIT BREAKER (LOW VOLTAGE) TRANSFERS OPEN	2.68E-07	2.40E+01	6.43E-06	
	HATCHAVAIL	HATCH AVAILABILITY		8.72E-01	8.72E-01	
	XXLESSTHAN2	LINES SMALLER THAN 2 INCH QUESTIONED		1.00E+00	1.00E+00	
26	CC-RWISO-1	1/2, MVFC1G31F001	3.31E-07	2.18E-03	2.18E-03	1.73E-08
	MCOR1R24S022	DC MCC S022 FAILS DURING OPERATION		2.40E+01	7.94E-06	
	XXLESSTHAN2	LINES SMALLER THAN 2 INCH QUESTIONED		1.00E+00	1.00E+00	

#	Inputs	Description	Rate	Exposure	Event Prob	Probability
27	CC-RWISO-2 MCOR1R24S011 XXLESSTHAN2	1/2, MVFC1G31F004 RX BLDG 600-V MCC 1C FAILS LINES SMALLER THAN 2 INCH QUESTIONED	3.31E-07	2.18E-03 2.40E+01 1.00E+00	2.18E-03 7.94E-06 1.00E+00	1.73E-08
28	%FL-BUSC CBX01R22S005_10I CC-RWISO-2 HATCHAVAIL	FLAG FOR LOSS OF 600-V BUS C INITIATING EVENT 4160-V SUPPLY BRKR TO XFMR C XFERS OPEN 1/2, MVFC1G31F004 HATCH AVAILABILITY	1.74E-07	1.00E+00 8.76E+03 2.18E-03 8.72E-01	1.00E+00 1.52E-03 2.18E-03 8.72E-01	1.71E-08
29	OPHEEPA XXLESSTHAN2 %FL-BUSC CBX01R23S003_2MI CC-RWISO-2 HATCHAVAIL	OPERATOR FAILS TO ALIGN 600-V BUS TO BACKUP 4160-V BUS LINES SMALLER THAN 2 INCH QUESTIONED FLAG FOR LOSS OF 600-V BUS C INITIATING EVENT 600-V LOAD BRKR FROM XFMR C TRANSFERS OPEN 1/2, MVFC1G31F004 HATCH AVAILABILITY	1.74E-07	5.91E-03 1.00E+00 1.00E+00 8.76E+03 2.18E-03 8.72E-01	5.91E-03 1.00E+00 1.00E+00 1.52E-03 2.18E-03 8.72E-01	1.71E-08
30	OPHEEPA XXLESSTHAN2 C2X01R22S017_4B CC-RWISO-1 XXLESSTHAN2	OPERATOR FAILS TO ALIGN 600-V BUS TO BACKUP 4160-V BUS LINES SMALLER THAN 2 INCH QUESTIONED CIRCUIT BREAKER (LOW VOLTAGE) TRANSFERS OPEN 1/2, MVFC1G31F001 LINES SMALLER THAN 2 INCH QUESTIONED	2.68E-07	5.91E-03 1.00E+00 2.40E+01 2.18E-03 1.00E+00	5.91E-03 1.00E+00 6.43E-06 2.18E-03 1.00E+00	1.40E-08
31	%LOSP CC-RWISO-2 DUR3 MNUNPS_TRNA UOL3 XXLESSTHAN2	LOSP INITIATING EVENT 1/2, MVFC1G31F004 OFFSITE POWER RESTORED AFTER 30 MINUTES, WITHIN 2.5 HOURS MAINT ON PSW PUMP C001A LOCA SIGNAL ON OPPOSITE UNIT, LOSP FOR 3 HOURS LINES SMALLER THAN 2 INCH QUESTIONED		1.89E-02 2.18E-03 4.90E-01 1.57E-02 3.33E-02 1.00E+00	1.89E-02 2.18E-03 4.90E-01 1.57E-02 3.33E-02 1.00E+00	1.06E-08
32	%LOSP CC-DGS-2 CC-RWISO-2 DUR24 UOL24 XXLESSTHAN2	LOSP INITIATING EVENT 1/3, DGLR1R43S001A 1/2, MVFC1G31F004 LOSP EXCEEDS 2.5 HOURS (24 HOURS ASSUMED) LOCA SIGNAL ON OPPOSITE UNIT, LOSP FOR 24 HOURS LINES SMALLER THAN 2 INCH QUESTIONED		1.89E-02 3.18E-02 2.18E-03 2.10E-01 3.78E-02 1.00E+00	1.89E-02 3.18E-02 2.18E-03 2.10E-01 3.78E-02 1.00E+00	1.04E-08
33	CBX01R23S003_7M CC-RWISO-2 XXLESSTHAN2	SUPPLY BREAKER TO RX BLDG 600-V MCC 1C TRANSFERS OPEN 1/2, MVFC1G31F004 LINES SMALLER THAN 2 INCH QUESTIONED	1.74E-07	2.40E+01 2.18E-03 1.00E+00	4.18E-06 2.18E-03 1.00E+00	9.11E-09
34	%LOSP CC-DGS-2 CC-DGS-3 CC-RWISO-2 DUR24 XXLESSTHAN2	LOSP INITIATING EVENT 1/3, DGLR1R43S001A 1/3, DGLR1R43S001B 1/2, MVFC1G31F004 LOSP EXCEEDS 2.5 HOURS (24 HOURS ASSUMED) LINES SMALLER THAN 2 INCH QUESTIONED		1.89E-02 3.18E-02 3.18E-02 2.18E-03 2.10E-01 1.00E+00	1.89E-02 3.18E-02 3.18E-02 2.18E-03 2.10E-01 1.00E+00	8.76E-09
35	%LOSP CC-DGS-22 CC-RWISO-2 DUR3 UOL3 XXLESSTHAN2	LOSP INITIATING EVENT 1/3, DGSS1R43S001A 1/2, MVFC1G31F004 OFFSITE POWER RESTORED AFTER 30 MINUTES, WITHIN 2.5 HOURS LOCA SIGNAL ON OPPOSITE UNIT, LOSP FOR 3 HOURS LINES SMALLER THAN 2 INCH QUESTIONED		1.89E-02 1.27E-02 2.18E-03 4.90E-01 3.33E-02 1.00E+00	1.89E-02 1.27E-02 2.18E-03 4.90E-01 3.33E-02 1.00E+00	8.52E-09

#	Inputs	Description	Rate	Exposure	Event Prob	Probability
36	%FL-BUSC	FLAG FOR LOSS OF 600-V BUS C INITIATING EVENT		1.00E+00	1.00E+00	8.33E-09
	CBFC1R23S003_9M	600-V ALT SUPPLY BRKR FROM XFMR CD FAILS TO CLOSE	9.62E-04	1.00E+00	9.62E-04	
	CC-RWISO-2	1/2, MVFC1G31F004		2.18E-03	2.18E-03	
	HATCHAVAIL	HATCH AVAILABILITY		8.72E-01	8.72E-01	
	XROR1R23S003_I	STATION SERVICE TRANSFORMER C FAILS TO OPERATE	5.20E-07	8.76E+03	4.56E-03	
	XXLESSTHAN2	LINES SMALLER THAN 2 INCH QUESTIONED		1.00E+00	1.00E+00	
37	%FL-BUSC	FLAG FOR LOSS OF 600-V BUS C INITIATING EVENT		1.00E+00	1.00E+00	7.79E-09
	BSSH1R23S003_I	600-V BUS C FAILS	3.76E-07	8.76E+03	3.29E-03	
	CBFC1R25S064_39	CROSS TIE CIRCUIT BREAKER FAILS TO CLOSE	9.62E-04	1.00E+00	9.62E-04	
	CVFR1T48F328A	VACUUM BREAKER VALVE T48-F328A FAILS TO RESEAT	2.82E-03	1.00E+00	2.82E-03	
	HATCHAVAIL	HATCH AVAILABILITY		8.72E-01	8.72E-01	
38	%FL-BUSC	FLAG FOR LOSS OF 600-V BUS C INITIATING EVENT		1.00E+00	1.00E+00	7.79E-09
	BSSH1R23S003_I	600-V BUS C FAILS	3.76E-07	8.76E+03	3.29E-03	
	CBFC1R25S064_40	CROSS TIE CIRCUIT BREAKER FAIL TO CLOSE	9.62E-04	1.00E+00	9.62E-04	
	CVFR1T48F328A	VACUUM BREAKER VALVE T48-F328A FAILS TO RESEAT	2.82E-03	1.00E+00	2.82E-03	
	HATCHAVAIL	HATCH AVAILABILITY		8.72E-01	8.72E-01	
39	%FL-BUSC	FLAG FOR LOSS OF 600-V BUS C INITIATING EVENT		1.00E+00	1.00E+00	7.79E-09
	BSSH1R23S003_I	600-V BUS C FAILS	3.76E-07	8.76E+03	3.29E-03	
	CBFC1R25S065_39	CROSS TIE CIRCUIT BREAKER FAILS TO CLOSE	9.62E-04	1.00E+00	9.62E-04	
	CVFR1T48F328A	VACUUM BREAKER VALVE T48-F328A FAILS TO RESEAT	2.82E-03	1.00E+00	2.82E-03	
	HATCHAVAIL	HATCH AVAILABILITY		8.72E-01	8.72E-01	
40	%FL-BUSC	FLAG FOR LOSS OF 600-V BUS C INITIATING EVENT		1.00E+00	1.00E+00	7.79E-09
	BSSH1R23S003_I	600-V BUS C FAILS	3.76E-07	8.76E+03	3.29E-03	
	CBFC1R25S065_40	CROSS TIE CIRCUIT BREAKER FAILS TO CLOSE	9.62E-04	1.00E+00	9.62E-04	
	CVFR1T48F328A	VACUUM BREAKER VALVE T48-F328A FAILS TO RESEAT	2.82E-03	1.00E+00	2.82E-03	
	HATCHAVAIL	HATCH AVAILABILITY		8.72E-01	8.72E-01	
41	%FL-LOBUSE	FLAG FOR LOSS OF BUS E OR SUPPLY HARDWARE INITIATING EVENT		1.00E+00	1.00E+00	6.03E-09
	BSSH1R22S005_I	4KV BUS E FAILS TO OPERATE	3.76E-07	8.76E+03	3.29E-03	
	CBFC1R23S003_9M	600-V ALT SUPPLY BRKR FROM XFMR CD FAILS TO CLOSE	9.62E-04	1.00E+00	9.62E-04	
	CC-RWISO-2	1/2, MVFC1G31F004		2.18E-03	2.18E-03	
	HATCHAVAIL	HATCH AVAILABILITY		8.72E-01	8.72E-01	
	XXLESSTHAN2	LINES SMALLER THAN 2 INCH QUESTIONED		1.00E+00	1.00E+00	
42	%LOSP	LOSP INITIATING EVENT		1.89E-02	1.89E-02	5.14E-09
	CC-RWISO-2	1/2, MVFC1G31F004		2.18E-03	2.18E-03	
	DUR24	LOSP EXCEEDS 2.5 HOURS (24 HOURS ASSUMED)		2.10E-01	2.10E-01	
	MNUNPS_TRNA	MAINT ON PSW PUMP C001A		1.57E-02	1.57E-02	
	UOL24	LOCA SIGNAL ON OPPOSITE UNIT, LOSP FOR 24 HOURS		3.78E-02	3.78E-02	
	XXLESSTHAN2	LINES SMALLER THAN 2 INCH QUESTIONED		1.00E+00	1.00E+00	
43	%LOSP	LOSP INITIATING EVENT		1.89E-02	1.89E-02	4.60E-09
	CC-DGS-15	1/3, DG1R1R43S001A		6.84E-03	6.84E-03	
	CC-RWISO-2	1/2, MVFC1G31F004		2.18E-03	2.18E-03	
	DUR3	OFFSITE POWER RESTORED AFTER 30 MINUTES, WITHIN 2.5 HOURS		4.90E-01	4.90E-01	
	UOL3	LOCA SIGNAL ON OPPOSITE UNIT, LOSP FOR 3 HOURS		3.33E-02	3.33E-02	
	XXLESSTHAN2	LINES SMALLER THAN 2 INCH QUESTIONED		1.00E+00	1.00E+00	

#	Inputs	Description	Rate	Exposure	Event Prob	Probability
44	%LOSP	LOSP INITIATING EVENT		1.89E-02	1.89E-02	4.46E-09
	CC-DGS-2	1/3, DGLR1R43S001A		3.18E-02	3.18E-02	
	CC-RWISO-2	1/2, MVFC1G31F004		2.18E-03	2.18E-03	
	DUR24	LOSP EXCEEDS 2.5 HOURS (24 HOURS ASSUMED)		2.10E-01	2.10E-01	
	OPHEEPB	OPERATOR FAILS TO ALIGN 600-V BUS TO BACKUP 4160-V BUS		1.62E-02	1.62E-02	
	XXLESSTHAN2	LINES SMALLER THAN 2 INCH QUESTIONED		1.00E+00	1.00E+00	
45	%LOSP	LOSP INITIATING EVENT		1.89E-02	1.89E-02	4.15E-09
	CC-DGS-22	1/3, DGSS1R43S001A		1.27E-02	1.27E-02	
	CC-RWISO-2	1/2, MVFC1G31F004		2.18E-03	2.18E-03	
	DUR24	LOSP EXCEEDS 2.5 HOURS (24 HOURS ASSUMED)		2.10E-01	2.10E-01	
	UOL24	LOCA SIGNAL ON OPPOSITE UNIT, LOSP FOR 24 HOURS		3.78E-02	3.78E-02	
	XXLESSTHAN2	LINES SMALLER THAN 2 INCH QUESTIONED		1.00E+00	1.00E+00	
46	%LOSP	LOSP INITIATING EVENT		1.89E-02	1.89E-02	4.14E-09
	CC-DGS-22	1/3, DGSS1R43S001A		1.27E-02	1.27E-02	
	CC-RWISO-2	1/2, MVFC1G31F004		2.18E-03	2.18E-03	
	DUR3	OFFSITE POWER RESTORED AFTER 30 MINUTES, WITHIN 2.5 HOURS		4.90E-01	4.90E-01	
	OPHEEPB	OPERATOR FAILS TO ALIGN 600-V BUS TO BACKUP 4160-V BUS		1.62E-02	1.62E-02	
	XXLESSTHAN2	LINES SMALLER THAN 2 INCH QUESTIONED		1.00E+00	1.00E+00	
47	%LOSP	LOSP INITIATING EVENT		1.89E-02	1.89E-02	3.70E-09
	CC-RWISO-2	1/2, MVFC1G31F004		2.18E-03	2.18E-03	
	DUR3	OFFSITE POWER RESTORED AFTER 30 MINUTES, WITHIN 2.5 HOURS		4.90E-01	4.90E-01	
	MNUN1R43S001A	DGA MAINTENANCE		5.51E-03	5.51E-03	
	UOL3	LOCA SIGNAL ON OPPOSITE UNIT, LOSP FOR 3 HOURS		3.33E-02	3.33E-02	
	XXLESSTHAN2	LINES SMALLER THAN 2 INCH QUESTIONED		1.00E+00	1.00E+00	
48	%LOSP	LOSP INITIATING EVENT		1.89E-02	1.89E-02	3.69E-09
	CC-RWISO-2	1/2, MVFC1G31F004		2.18E-03	2.18E-03	
	CC-SW-1	1/4, PMOS1P41C001A		5.49E-03	5.49E-03	
	DUR3	OFFSITE POWER RESTORED AFTER 30 MINUTES, WITHIN 2.5 HOURS		4.90E-01	4.90E-01	
	UOL3	LOCA SIGNAL ON OPPOSITE UNIT, LOSP FOR 3 HOURS		3.33E-02	3.33E-02	
	XXLESSTHAN2	LINES SMALLER THAN 2 INCH QUESTIONED		1.00E+00	1.00E+00	
49	%FL-BUSC	FLAG FOR LOSS OF 600-V BUS C INITIATING EVENT		1.00E+00	1.00E+00	3.64E-09
	CBFO1R23S003_2M	600-V LOAD BRKR FROM XFMR C FAILS TO OPEN	4.20E-04	1.00E+00	4.20E-04	
	CC-RWISO-2	1/2, MVFC1G31F004		2.18E-03	2.18E-03	
	HATCHAVAIL	HATCH AVAILABILITY		8.72E-01	8.72E-01	
	XROR1R23S003_I	STATION SERVICE TRANSFORMER C FAILS TO OPERATE	5.20E-07	8.76E+03	4.56E-03	
	XXLESSTHAN2	LINES SMALLER THAN 2 INCH QUESTIONED		1.00E+00	1.00E+00	
50	%LOSP	LOSP INITIATING EVENT		1.89E-02	1.89E-02	3.49E-09
	CC-DGS-2	1/3, DGLR1R43S001A		3.18E-02	3.18E-02	
	CC-DGS-23	1/3, DGSS1R43S001B		1.27E-02	1.27E-02	
	CC-RWISO-2	1/2, MVFC1G31F004		2.18E-03	2.18E-03	
	DUR24	LOSP EXCEEDS 2.5 HOURS (24 HOURS ASSUMED)		2.10E-01	2.10E-01	
	XXLESSTHAN2	LINES SMALLER THAN 2 INCH QUESTIONED		1.00E+00	1.00E+00	

Report Summary:

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Importance Measure Report

CI = 4.40E-04

Event Name	Probability	Fus Ves	BlmBm	Red W	Ach W	Description
XXLESSTHAN2	1.00E+00	3.35E-01	1.47E-04	1.504	1.00	LINES SMALLER THAN 2 INCH QUESTIONED
CC-CI-6	1.40E-04	3.17E-01	1.00E+00	1.465	2.27E+03	2/2, AVFC1G11F019 AVFC1G11F020
CC-CI-9	1.40E-04	3.17E-01	1.00E+00	1.465	2.27E+03	2/2, AVFC1G11F003 AVFC1G11F004
CC-RWISO-3	1.19E-04	2.71E-01	1.00E+00	1.372	2.27E+03	2/2, MVFC1G31F001 MVFC1G31F004
CC-RWISO-2	2.18E-03	2.63E-02	5.30E-03	1.027	13.02	1/2, MVFC1G31F004
CVFR1B21F010A	2.82E-03	1.81E-02	2.82E-03	1.018	7.39	CHECK VALVE 1B21-F010A FAILS TO RESEAT
CVFR1B21F010B	2.82E-03	1.81E-02	2.82E-03	1.018	7.39	CHECK VALVE 1B21-F010B FAILS TO RESEAT
CVFR1G31F039	2.82E-03	1.81E-02	2.82E-03	1.018	7.39	CHECK VALVE 1G31-F039 FAILS TO RESEAT
CVFR1G31F203	2.82E-03	1.81E-02	2.82E-03	1.018	7.39	CHECK VALVE 1G31-F203 FAILS TO RESEAT
HATCHAVAIL	8.72E-01	1.60E-02	8.07E-06	1.016	1.00	HATCH AVAILABILITY
%FL-BUSC	1.00E+00	1.58E-02	6.95E-06	1.016	1.00	FLAG FOR LOSS OF 600-V BUS C INITIATING EVENT
BSSH1R23S003_I	3.29E-03	1.56E-02	2.08E-03	1.016	5.70	600-V BUS C FAILS
MIUNCI	4.17E-02	1.35E-02	1.43E-04	1.014	1.31	DRYWELL VENT LINE OPEN
CC-CI-12	1.40E-04	1.32E-02	4.17E-02	1.013	95.67	2/2, AVFC1T48F335A AVFC1T48F334A
CC-RWISO-1	2.18E-03	1.15E-02	2.31E-03	1.012	6.25	1/2, MVFC1G31F001
CC-CI-4	1.86E-03	7.83E-03	1.86E-03	1.008	5.21	1/2, AVFC1G11F019
CC-CI-5	1.86E-03	7.83E-03	1.86E-03	1.008	5.21	1/2, AVFC1G11F020
CC-CI-7	1.86E-03	7.83E-03	1.86E-03	1.008	5.21	1/2, AVFC1G11F003
CC-CI-8	1.86E-03	7.83E-03	1.86E-03	1.008	5.21	1/2, AVFC1G11F004
MNUNRWISO_OUT	1.10E-04	1.27E-03	5.09E-03	1.001	12.56	RWCU OUTBOARD MOV INOP DUE TO MAINTENANCE
CVFR1T48F328A	2.82E-03	7.07E-04	1.10E-04	1.001	1.25	VACUUM BREAKER VALVE T48-F328A FAILS TO RESEAT
MNUNRWISO_IN	1.10E-04	5.45E-04	2.18E-03	1.001	5.96	RWCU INBOARD MOV INOP DUE TO MAINTENANCE
OPHES064/S065	2.00E-02	4.54E-04	9.99E-06	1.000	1.02	OPERATOR ACTION TO MANUALLY TRANSFER INSTRUMENT BUS POWER
CVFR1T48F328B	2.82E-03	3.48E-04	5.44E-05	1.000	1.12	VACUUM BREAKER VALVE T48-F328B FAILS TO RESEAT
CC-CI-10	1.86E-03	3.26E-04	7.73E-05	1.000	1.18	1/2, AVFC1T48F335A
CC-CI-11	1.86E-03	3.26E-04	7.73E-05	1.000	1.18	1/2, AVFC1T48F334A
OPHEEPA	5.91E-03	2.92E-04	2.17E-05	1.000	1.05	OPERATOR FAILS TO ALIGN 600-V BUS TO BACKUP 4160-V BUS
%LOSP	1.89E-02	2.70E-04	6.29E-06	1.000	1.01	LOSP INITIATING EVENT
AVXO1T48F310	3.89E-05	2.49E-04	2.82E-03	1.000	7.41	AIR-OPERATED VALVE 1T48-F310 TRANSFERS OPEN
AVXO1T48F311	3.89E-05	2.49E-04	2.82E-03	1.000	7.41	AIR-OPERATED VALVE 1T48-F311 TRANSFERS OPEN
XROR1R23S003_I	4.56E-03	1.52E-04	1.47E-05	1.000	1.03	STATION SERVICE TRANSFORMER C FAILS TO OPERATE
DUR24	2.10E-01	1.39E-04	2.91E-07	1.000	1.00	LOSP EXCEEDS 2.5 HOURS (24 HOURS ASSUMED)
DUR3	4.90E-01	1.32E-04	1.18E-07	1.000	1.00	OFFSITE POWER RESTORED AFTER 30 MINUTES, WITHIN 2.5 HOURS
%FL-LOBUSE	1.00E+00	1.08E-04	4.76E-08	1.000	1.00	FLAG FOR LOSS OF BUS E OR SUPPLY HARDWARE INITIATING EVENT
BSSH1R22S005_I	3.29E-03	1.08E-04	1.44E-05	1.000	1.03	4KV BUS E FAILS TO OPERATE
BSSH1R22S017	9.02E-06	1.04E-04	5.04E-03	1.000	12.47	DC SWITCHGEAR S017 FAILS DURING OPERATION
MCOR1R24S022	7.94E-06	9.12E-05	5.04E-03	1.000	12.47	DC MCC S022 FAILS DURING OPERATION
%FL-BUSD	1.00E+00	8.78E-05	3.86E-08	1.000	1.00	FLAG FOR INITIATING EVENT CAUSED BY LOSS OF 600V BUS D
BSSH1R23S004_I	3.29E-03	8.78E-05	1.17E-05	1.000	1.03	600-V BUS D FAILS DURING OPERATION
XXBD_TRANSIENT	2.00E-01	8.78E-05	1.93E-07	1.000	1.00	LOSS OF BUS D CAUSES INITIATING EVNET (TRIP)
UOL3	3.33E-02	7.53E-05	9.94E-07	1.000	1.00	LOCA SIGNAL ON OPPOSITE UNIT, LOSP FOR 3 HOURS

Event Name	Probability	Fus Ves	BlmBm	Red W	Ach W	Description
CC-DGS-2	3.18E-02	7.41E-05	1.02E-06	1.000	1.00	1/3, DGLR1R43S001A
C2X01R22S017_4B	6.43E-06	7.38E-05	5.04E-03	1.000	12.47	CIRCUIT BREAKER (LOW VOLTAGE) TRANSFERS OPEN
CC-DGS-22	1.27E-02	7.22E-05	2.51E-06	1.000	1.01	1/3, DGSS1R43S001A
UOL24	3.78E-02	5.80E-05	6.75E-07	1.000	1.00	LOCA SIGNAL ON OPPOSITE UNIT, LOSP FOR 24 HOURS
CBX01R22S005_10I	1.52E-03	4.81E-05	1.39E-05	1.000	1.03	4160-V SUPPLY BRKR TO XFMR C XFERS OPEN
CBX01R23S003_2MI	1.52E-03	4.81E-05	1.39E-05	1.000	1.03	600-V LOAD BRKR FROM XFMR C TRANSFERS OPEN
CBFC1R23S003_9M	9.62E-04	4.53E-05	2.07E-05	1.000	1.05	600-V ALT SUPPLY BRKR FROM XFMR CD FAILS TO CLOSE
BSSH1R23S003	9.02E-06	4.47E-05	2.18E-03	1.000	5.96	600-V BUS C FAILS
MCOR1R24S011	7.94E-06	3.94E-05	2.18E-03	1.000	5.96	RX BLDG 600-V MCC 1C FAILS
MNUNPS TRNA	1.57E-02	3.57E-05	9.99E-07	1.000	1.00	MAINT ON PSW PUMP C001A
CC-DGS-3	3.18E-02	3.56E-05	4.92E-07	1.000	1.00	1/3, DGLR1R43S001B
OPHEEPB	1.62E-02	3.27E-05	8.90E-07	1.000	1.00	OPERATOR FAILS TO ALIGN 600-V BUS TO BACKUP 4160-V BUS
CC-DGS-15	6.84E-03	2.89E-05	1.86E-06	1.000	1.00	1/3, DG1R1R43S001A
CC-DGS-23	1.27E-02	2.56E-05	8.90E-07	1.000	1.00	1/3, DGSS1R43S001B
MNUN1R43S001A	5.51E-03	2.33E-05	1.86E-06	1.000	1.00	DGA MAINTENANCE
CBFC1R25S064_39	9.62E-04	2.12E-05	9.72E-06	1.000	1.02	CROSS TIE CIRCUIT BREAKER FAILS TO CLOSE
CBFC1R25S064_40	9.62E-04	2.12E-05	9.72E-06	1.000	1.02	CROSS TIE CIRCUIT BREAKER FAIL TO CLOSE
CBFC1R25S065_39	9.62E-04	2.12E-05	9.72E-06	1.000	1.02	CROSS TIE CIRCUIT BREAKER FAILS TO CLOSE
CBFC1R25S065_40	9.62E-04	2.12E-05	9.72E-06	1.000	1.02	CROSS TIE CIRCUIT BREAKER FAILS TO CLOSE
CBX01R23S003_7M	4.18E-06	2.07E-05	2.18E-03	1.000	5.96	SUPPLY BREAKER TO RX BLDG 600-V MCC 1C TRANSFERS OPEN
CBFO1R23S003_2M	4.20E-04	1.98E-05	2.07E-05	1.000	1.05	600-V LOAD BRKR FROM XFMR C FAILS TO OPEN
CC-SW-1	5.49E-03	1.25E-05	9.99E-07	1.000	1.00	1/4, PMOS1P41C001A
FAILRATERATIO	1.00E-01	1.02E-05	4.48E-08	1.000	1.00	ASSUMED RATIO OF PANEL TO MCC FAILURE RATES. (RISKMAN MOD
MNUN1R43S001B	7.21E-03	8.70E-06	5.31E-07	1.000	1.00	DGB MAINTENANCE
CC-DGS-16	6.84E-03	8.25E-06	5.31E-07	1.000	1.00	1/3, DG1R1R43S001B
CBFO1R25S036_25	4.20E-04	7.73E-06	8.10E-06	1.000	1.02	FEEDER BREAKER FAILS TO OPEN
MIUNDGS_DGB	5.84E-03	7.05E-06	5.31E-07	1.000	1.00	DIESEL B ALIGNED TO UNIT 2 AND UNIT 2 ALSO IN LOSP
MCOR1R25S064	7.94E-06	5.09E-06	2.82E-04	1.000	1.64	R25S064 FAILS DURING OPERATION
MCOR1R25S065	7.94E-06	5.09E-06	2.82E-04	1.000	1.64	R25S065 FAILS DURING OPERATION
CC-DGS-9	3.03E-03	4.63E-06	6.72E-07	1.000	1.00	1/3, DGS1R43S001A
CC-DGS-6	1.92E-04	3.78E-06	8.65E-06	1.000	1.02	2/3, DGLR1R43S001A DGLR1R43S001B
CC-DGS-7	1.89E-04	3.72E-06	8.65E-06	1.000	1.02	3/3, DGLR1R43S001C DGLR1R43S001A DGLR1R43S001B
%FL-LOBUSG	1.00E+00	3.68E-06	1.62E-09	1.000	1.00	FLAG FOR LOSS OF BUS G INITIATING EVENT
BSSH1R22S007_I	3.29E-03	3.68E-06	4.92E-07	1.000	1.00	4KV BUS G FAILS DURING OPERATION
OPHEEPANOLINK	5.00E-02	3.68E-06	3.24E-08	1.000	1.00	OPERATOR FAILS TO ALIGN 600-V BUS TO BACKUP 4160-V BUS
XXBG_TRANSIENT	2.00E-01	3.68E-06	8.10E-09	1.000	1.00	LOSS OF BUS G CAUSES AN INITIATING EVENT (TRIP)
CC-DGS-39	6.65E-05	3.05E-06	2.02E-05	1.000	1.05	2/3, CBFC1R22S005_5 CBFC1R22S006_6
CC-DGS-42	6.60E-05	3.03E-06	2.02E-05	1.000	1.05	3/3, CBFC1R22S005_5 CBFC1R22S006_6 CBFC1R22S007_6
CC-DGS-28	6.40E-05	2.94E-06	2.02E-05	1.000	1.05	3/3, DGSS1R43S001A DGSS1R43S001B DGSS1R43S001C
FUS01R25S064	2.21E-05	2.83E-06	5.64E-05	1.000	1.13	SUPPLY FUSE PREMATURELY OPENS
FUS01R25S065	2.21E-05	2.83E-06	5.64E-05	1.000	1.13	SUPPLY FUSE PREMATURELY OPENS
CC-DGS-25	5.87E-05	2.70E-06	2.02E-05	1.000	1.05	2/3, DGSS1R43S001A DGSS1R43S001B

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