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Docket Nos. 50-321

HL-6118

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

Edwin I. Hatch Nuclear Plant Unit 1
Request to Revise Technical Specifications:
Deferral of Type A Containment Integrated Leak Rate Test (ILRT)

Ladies and Gentlemen:

In accordance with the provisions of 10 CFR 50.90, as required by 10 CFR 50.59(c)(1), Southern Nuclear Operating Company (SNC) is proposing a change to the Edwin I. Hatch Nuclear Plant (HNP) Unit 1 Technical Specifications, Appendix A to Operating License DPR-57. This proposed change will revise Technical Specifications (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) to no later than April 2008. 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." Attachment 1 provides the "Risk Assessment for Edwin I. Hatch Nuclear Power Station Regarding ILRT (Type A) Extension Request."

Enclosure 1 provides a description of the proposed change and an explanation of the basis for the change. Enclosure 2 details the bases for SNC's determination that the proposed change does not involve a significant hazards consideration. Enclosure 3 provides page change instructions for incorporating the proposed change. Following Enclosure 3 are the revised Technical Specifications page and the corresponding marked-up page.

Southern Nuclear Operating Company requests the proposed amendment to be issued by January 2002, with the amendment to be effective prior to the HNP Unit 1 outage currently scheduled to begin in March 2002.

In accordance with the requirements of 10 CFR 50.91, a copy of this letter and all applicable enclosures will be sent to the designated State official of the Environmental Protection Division of the Georgia Department of Natural Resources.

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Mr. H. L. Sumner, Jr. states he is Vice President of Southern Nuclear Operating Company and 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.

Respectfully submitted,

H. L. Sumner, Jr.

Sworn to and subscribed before me this 31 *day of* August *2001.*

Notary Public

Commission Expiration Date: MY COMMISSION EXPIRES JAN. 12, 2005

IFL/eb

Enclosures:
1. Basis for Change Request
2. 10 CFR 50.92 Evaluation
3. Page Change Instructions

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

cc: Southern Nuclear Operating Company
Mr. P. H. Wells, Nuclear Plant General Manager
SNC Document Management (R-Type A02.001)

U.S. Nuclear Regulatory Commission, Washington, D.C.
Mr. L. N. Olshan, Project Manager - Hatch

U.S. Nuclear Regulatory Commission, Region II
Mr. L. A. Reyes, Regional Administrator
Mr. J. T. Munday, Senior Resident Inspector - Hatch

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

Enclosure 1

Edwin I. Hatch Nuclear Plant Unit 1
Request to Revise Technical Specifications:
Deferral of Type A Containment Integrated Leak Rate Test (ILRT)

Basis for Change Request

Proposed Change

SNC requests that the Technical Specifications (TS) contained in Appendix A to the HNP Unit 1 Operating License DPR-57 be amended to revise Technical Specifications Section 5.5.12 to reflect a one-time deferral of the Type A Containment Integrated Leak Rate Test (ILRT) to no later than April 2008.

The proposed change involves a one-time exception to the ten (10) year frequency of the performanced-based leakage rate testing program for Type A tests as required by Nuclear Energy Institute (NEI) 94-01, Revision 0, "Industry Guideline for Implementing Performanced-Based Option of 10 CFR 50, Appendix J." The current ten (10) year ILRT for HNP Unit 1 is due in April 2003, which would require it to be performed during Refueling Outage (RFO) 1R20 in April 2002. The proposed exception would allow the next ILRT for HNP Unit 1 to be performed within fifteen (15) years (April 2008) 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 HNP Unit 1 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":
>
> a. Section 9.2.3: The first Type A test after the April 1993 Type A test shall be performed no later than April 2008.

This one-time exception will result in the following:

- Perform a Type A Containment ILRT during RFO 1R23, currently scheduled for March 2008.

- 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 for this outage 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 450 mrem.

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 primary 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 primary 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 200 was issued for HNP Unit 1 (dated March 6, 1996) to permit implementation of 10 CFR 50, Appendix J, Option B. Amendment 200 revised Technical Specification Section 5.5 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 Nuclear Regulatory Commission (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 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 a Mark I containment. (HNP Unit 1 is a Mark I containment.) NUREG-1493 made the following observations with regard to decreasing the test frequency.

- 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 $1.0L_a$. Based upon the acceptable November 1988 and April 1993 ILRTs, the current test interval for HNP Unit 1 is once every ten (10) years, with the next test due to be performed by April 2003.

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 leakages and that the percentage of containment leakages that are detected only by integrated containment leakage testing is very small.

HNP, Unit 1 has undergone 6 operational Type A tests in addition to the pre-operational Type A test. The results of these tests demonstrate that the HNP Unit 1 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 HNP Unit1 Technical Specifications Section 5.5.12, the maximum allowable primary containment leakage rate, L_a, at P_a , is 1.2% of primary containment air weight per day. The HNP Unit 1 ILRT results are provided below.

Unit 1 completed April 28, 1993
Total Time Analysis (95% upper confidence limit) = 0.3608 (weight % per day)
Mass Point Analysis (95% upper confidence limit) = 0.3488 (weight % per day)

Unit 1 completed November 30, 1988
Total Time Analysis (95% upper confidence limit) = 0.57 (weight % per day)
Mass Point Analysis (95% upper confidence limit) = 0.4968 (weight % per day)

Unit 1 completed April 19, 1986
Total Time Analysis (95% upper confidence limit) = 0.611 (weight % per day)
Mass Point Analysis (95% upper confidence limit) = 0.428 (weight % per day)

Unit 1 completed February 2, 1982
Total Time Analysis (95% upper confidence limit) = 0.555 (weight % per day)
Mass Point Analysis (95% upper confidence limit) = 0.442 (weight % per day)

Unit 1 completed June 30, 1978
Total Time Analysis (95% upper confidence limit) = 0.558 (weight % per day)
Mass Point Analysis (95% upper confidence limit) = 0.456 (weight % per day)

Unit 1 completed May 26, 1974
Total Time Analysis (95% upper confidence limit) = 0.37 (weight % per day)
Mass Point Analysis: Not Calculated

c. Plant Operational Performance

HNP Unit 1 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

Effective September, 1996, the NRC endorsed Subsections IWE and IWL of ASME Section XI, 1992 Edition including 1992 Addenda. These subsections contain inservice inspection and repair and replacement rules for metal containment vessels (Class MC) and concrete containment vessels (Class CC), respectively. The reactor containments at Hatch are free-standing steel containments, to which only the requirements of Subsection IWE apply. The Hatch, Unit 1 IWE inspection program was established in 1998. The initial 40-month inspection period specified by ASME XI, Subsection IWE, as modified by 10CFR50.55(a)(2)(x), is required to be implemented by September 9, 2001 and runs essentially concurrent with the second ISI 40-month inspection period. The IWE program will be updated early, at the completion of the third ISI 40-month inspection period, to run concurrent with the fourth ISI inspection interval.

Hatch was given an exemption to perform the first general visual examination of the Unit 1 primary containment in spring of 2002 (Refueling Outage 1R20), to meet the intent of the expedited implementation requirement of 10CFR50.55a(b). VT-3 visual examination was performed on the submerged surfaces in the Spring of 1999 (Refueling Outage 1R18). Reinspection of the nonsubmerged primary containment, per IWE, is scheduled in order to meet the 10 CFR 50 requirement of IWE containment examination, each inspection period (i.e., three times per inspection interval).

Prior to the inception of the containment inservice inspection program, visual inspection of the accessible areas of the primary containment was performed in accordance with the requirements of 10 CFR 50.65, "Requirements for Monitoring the Effectiveness of Maintenance at Nuclear Power Plants" and 10 CFR 50, Appendix J prior to each Type A leakage test.

Inspections of the submerged interior surfaces of the suppression pool have been performed and documented at Hatch Unit 1 since June of 1994 through 1997 at which time the IWE requirements were considered applicable to Unit 1.

Visual examination of the accessible and immersed surfaces of the containment is also performed periodically to assess the condition of containment coatings in accordance with the requirements of 10 CFR 50.65 and licensing commitments for Generic Letter 98-04 ("Potential for Degradation of the Emergency Core Cooling System and the Containment Spray System After a Loss-of-Coolant Accident Because of Construction and Protective Coating Deficiencies and Foreign Material in Containment"). These periodic inspections serve to identify coating distress that may be indicative of degradation of containment structural integrity. Inspections performed to date have resulted in some localized repairs, but the amount of shell degradation has been minimal.

The ASME Section XI IWE/IWL 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.

IWE-1240 Examinations

Surfaces that should be evaluated as requiring augmented examinations are defined by IWE-1240. The basic premise of IWE-1240 is: (a) containment surfaces that are subject to accelerated corrosion with no or minimal corrosion allowance or areas where the absence or repeated loss of protective coatings has resulted in substantial corrosion and pitting, and (b) containment surfaces subjected to excessive wear from abrasion or erosion that causes a loss of protective coatings, deformation, or material loss. The areas described in IWE-1240 were considered for their applicability at Plant Hatch and a discussion of each area is provided below.

- Interior Submerged Surfaces of Suppression Pool (Torus)

  The torus design accounted for maintenance of a specific water level during all modes of plant operation and post accident. The interior and exterior surfaces were initially provided with protective coatings which have been inspected and patch coated as needed since commercial operation of the plant. Plant Hatch implements periodic VT-3 visual examination, utilizing underwater divers, of the submerged surfaces to determine any areas of coating or shell degradation. A recoating process has been implemented for any areas that indicated coating degradation. Pitting depth measurements were taken in conjunction with these examinations to determine torus shell corrosion rates. Test specimens have also been placed inside the torus, below the water level, to provide additional information relative to coating degradation and potential shell corrosion rates.

Evaluation of the examination results to date, does not indicate that the submerged areas of the torus have experienced any significant degradation that presently warrants classification as IWE Category E-C. The interior submerged surfaces of the torus have been included in the examination plan as IWE Category E-A subject to VT-3 visual examination. The results of future VT-3 examinations, performed by underwater divers, will be evaluated to determine if these areas should be re-categorized.

- Interior Torus Surfaces Exposed to Periodic Wetting and Drying

  The containment spray mode of RHR system operation is only used infrequently to control suppression pool pressure. This has resulted in discoloration of the protective coating on the areas adjacent to the spray nozzles. However, periodic examination of the interior torus surfaces (each refueling outage) has not indicated any significant degradation of the protective coating or the shell surface. Some minor areas of the coating have been cleaned and recoated, but no significant shell degradation has been identified. These surfaces are visually examined in accordance with Category E-A, and the performance of augmented examinations per Category E-C is not presently warranted for these surface areas.

- Bottom Interior of Torus Adjacent to SRV Discharge Lines

  The SRV discharge lines at Plant Hatch were modified in the early 1980s incorporating a "T-quencher" design which evenly distributes the discharged steam and prevents steam-jets that could damage the protective coating or the shell surface. Periodic VT-3 visual examination by underwater divers has not indicated any significant coating degradation which would indicate potential shell degradation. These surfaces are included within the scope of the Category E-A examinations and are periodically inspected by underwater divers. Therefore, the performance of augmented examinations per Category E-C is not presently warranted for these surface areas.

- Torus Seismic Restraints (Earthquake Ties - 4)

  The torus is provided with 4 seismic restraints to account for the possibility of any seismic loads that could be experienced. These restraints are located at the 87 foot floor elevation in the torus room and are accessible during the general and VT-3 examinations (Category E-A) of the containment surfaces. The torus room is not a harsh environment and Plant Hatch has not been subjected to any seismic events that would affect the torus or the restraints. These seismic restraints are included within the scope of the Category E-A examinations and performance of augmented examinations per Category E-C is not presently warranted for these restraint areas.

- Exterior Drywell Shell Below The 114 Foot Floor Elevation

  The exterior of the drywell shell at and below the 114 foot elevation was considered as possibly subject to accelerated degradation due to problems reported at Oyster Creek. This area at Oyster Creek was found to be severely corroded due to exposure to water and corrosive chemicals that had accumulated in the air gap region because of a leak in

the refueling bellows, chemicals in the gap forming material (that was left in place), and drain lines that were not functional.

The refueling bellows at Plant Hatch is of a different design than that at Oyster Creek and virtually all of the gap forming material was removed during construction. The sand cushion drain lines have been modified at Plant Hatch (discharge elbows removed) to facilitate visual examination. The air gap drain lines and the sand cushion drain lines have been examined using a video probe to assure that they are functional and that any water that might leak into the drywell air gap region would be discharged from the area. These visual examinations did not indicate the present or past existence of moisture in these areas. The discharge of each drain is also examined for evidence of moisture during each refueling outage, while the reactor cavity is flooded, to assure that no water is present in this area. Therefore, the performance of augmented examinations per Category E-C is not presently warranted for these surface areas.

- Drywell Equipment Hatches and Personnel Air Lock

  The equipment hatches and personnel air lock are used as entry/exit openings during refueling outages for equipment and personnel. The air lock is provided with floor grating which prevents abrasion of the lower portion of the shell. Wooden platforms are constructed in the equipment hatches to prevent abrasion of the shell. Therefore, there is no reason to expect accelerated degradation in these areas and the visual examination requirements of Category E-A, general and VT-3, are adequate to monitor the structural integrity.

  Based on a review of IWE-1240 relative to the containment design at Plant Hatch and the results of previous examinations related to the integrity of the containment, there are no areas that should be designated for augmented examination per Category E-C at the present time. The results of future containment examinations, related to IWE, Appendix J, and the Maintenance Rule, will be evaluated to determine if any areas are experiencing degradation that would result in the need to implement augmented examinations.

Examinations of Seals and Bolts

Maintenance personnel, trained in the installation of seals and gaskets and the proper assembly of these closure devices, examine the seals and/ or gaskets as well as the mating surfaces during the assembly process. Appendix J leak rate testing after re-assembly then provides a positive confirmation of leak tight integrity.

- Electrical Penetrations And Containment Penetrations Whose Design Incorporates Resilient Seals, Gaskets, Or Sealant Compounds

  For those penetrations that are disassembled or opened, an Appendix J test is required upon final assembly prior to start-up. Additionally, if a seal (including O-rings or gaskets) is reinstalled of replaced, it will be visually inspected by maintenance personnel before re-assembly or closure. These tests and inspections will assure the

leak tightness of primary containment and provide an acceptable level of quality and safety.

- Airlocks And Containment Equipment Hatches

  The personnel airlocks are opened as needed during maintenance outages and refueling outages. Prior to final closure, the accessible portions of gaskets and the door sealing faces are inspected for damage that could affect the leak tightness of the seal. If gasket reinstallation is performed or replacement is necessary, the existing or new gasket will be visually inspected by maintenance personnel before re- assembly or closure. Door seals will be tested, as required by Technical Specifications, in accordance with Appendix J within seven days of opening and once every 30 days during periods of frequent opening.

  The containment equipment hatch is normally removed during refueling outages. If gasket replacement is necessary, the new gasket will be visually inspected by maintenance personnel before re-assembly or closure. Prior to establishing containment integrity following the refueling outage, the containment equipment hatch is leak rate tested in accordance with Appendix J.

- Bolt Examination

  Testing of bolting associated with the primary containment pressure boundary is accomplished by VT-1 examination in accordance with the requirements of Section IWE, when disassembled, or examined in place if not disassembled during the interval. Bolting that has not been disassembled and reassembled during the inspection will also be VT-1 examined in place at the end of the interval and also VT-1 examined in the event that the bolting is disassembled.

Stainless Steel Bellows Examination

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.

Features That Mitigate Containment Degradation

A review was performed in response to GL 87-05, as applicable to Plant Hatch. The results of the review indicated that Plant Hatch was not subject to the same conditions which caused the drywell shell degradation at other plants having a Mark I containment.

- Construction drawings indicated that the gap forming material was removed at Plant Hatch except for narrow rings at the elevation of each concrete pour.

- The refueling bellows does not contain any mechanical joints subject to degradation and subsequent leakage.

- The air gap drain lines were inspected utilizing a video probe and were all found to be functional. Video inspection of the air gap drains did not reveal any evidence of moisture or collection of water.

- The sand cushion at Hatch was constructed with a metal seal plate which would have directed any water into the air gap drain lines and prevented collection in the sand cushion.

In 1996, sample ultrasonic thickness measurements were added to the inspection program for the drywell shell. These initial measurements did not indicate any degradation of the drywell. These measurements are presently scheduled to be performed on a supplemental program basis once each inspection interval.

e. 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.

f. Conclusion

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

Enclosure 2

Edwin I. Hatch Nuclear Plant Unit 1
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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 1 ILRT test history supports this 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.

HNP Unit 1 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 1 TS does not involve a Significant Hazards Consideration.

Enclosure 3

Edwin I. Hatch Nuclear Plant Unit 1
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Page Change Instructions

Unit 1

Page	Replace
5.0-16a	5.0-16a

5.5 Programs and Manuals

5.5.11 Technical Specifications (TS) Bases Control Program (continued)

d. Proposed changes that meet the criteria of b. above shall be reviewed and approved by the NRC prior to implementation. Changes to the Bases implemented without prior NRC approval shall be provided to the NRC on a frequency consistent with 10 CFR 50.71(e).

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 April 1993 Type A test shall be performed no later than April 2008.

The peak calculated primary containment internal pressure for the design basis loss of coolant accident, P_a, is 50.5 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 $\geq P_a$,

(continued)

5.5 Programs and Manuals

5.5.12 Primary Containment Leakage Rate Testing Program (continued)

2) For each door, leakage rate is $\leq$ 0.01 L_a when the gap between the door seals is pressurized to $\geq$ 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.

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

(continued)

5.5 Programs and Manuals

5.5.11 Technical Specifications (TS) Bases Control Program (continued)

d. Proposed changes that meet the criteria of b. above shall be reviewed and approved by the NRC prior to implementation. Changes to the Bases implemented without prior NRC approval shall be provided to the NRC on a frequency consistent with 10 CFR 50.71(e).

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.

Insert A

The peak calculated primary containment internal pressure for the design basis loss of coolant accident, P_a, is 50.5 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 $\geq P_a$,

   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)

Insert A

, 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 April 1993 Type A test shall be performed no later than April 2008.

Attachment 1

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

TABLE OF CONTENTS

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 1E-3 to 1E-2, with 5E-3 identified as the point estimate based on 4 events in 740 reactor years and conservatively assuming a one-year duration for each event. 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 1E-7/yr in the shutdown core damage frequency. This risk reduction is due to the following issues:

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

The study identified 7 shutdown incidents (out of 463 reviewed) that were caused by ILRT or LLRT activities. Two of the 7 incidents were RCS draindown events caused by ILRT/LLRT activities, 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 ½ 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) = 1.24E-5/yr at a truncation of 1E-11/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 2.19E-6/yr at a truncation of 1E-11/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) = 1.65E-7
- Containment Overpressure (LER_OPD) = 6.56E-7
- Containment Overtemperature (LER_OT) = 1.37E-6
- Containment Intact with DW Vent Open (LER_VD) = 8.1E-10

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(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
Level 2 Subgate Results (@1E-11/yr truncation)				
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
Late Containment Failure Results (@1E-11/yr truncation)				
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 person-rem * 2.4 = 1963 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**

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				ε[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) ε - 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_a, 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 χ^2_{95} (ν = 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 χ^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_f, can be estimated as follows:

for <u>3 Year Interval</u>

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

for <u>10 Year Interval</u>

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

for <u>15 Year Interval</u>

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

(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 4.4E-4/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 9.06E-6/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 = 3.3E-6/yr). This was then subtracted from the total CDF (1.24E-5/yr) to obtain the Class 1 frequency of "No Containment Failure" of 9.0E-6/yr. For this analysis, the associated maximum containment leakage for this group is $1L_a$, 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:

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

CLASS 2 FREQUENCY = 4.4E-4 * 1.24E-5/year
CLASS 2 FREQUENCY = 5.5E-9/year

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:

$PROB_{class\_3a}$ = probability of small pre-existing containment liner leakage
= 0.064 [see Section 4.3]

$PROB_{class\_3b}$ = probability of large pre-existing containment liner leakage
= 0.021 [see Section 4.3]

CLASS_3A_FREQUENCY = 0.064 * 1.24E-5/year = 7.9E-7/year
CLASS_3B_FREQUENCY = 0.021 * 1.24E-5/year = 2.6E-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-though, 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 = 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)

= 8.1E-10/yr

Where:

Total early containment failure frequency = 2.0E-6[1]

CLASS_7_FREQUENCY = 2.0E-6/year + 1.1E-7/year

CLASS_7_FREQUENCY = 2.11E-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 1.65E-7/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 7E+5 to 1.06E+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 1.15E+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)	1.15E+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)	1.15E+6(1)
7a	Failures Induced by Phenomena (Early)	1.06E+6(1)
7b	Failures Induced by Phenomena (Late)(2)	5.7E+5(1) (2)
8	Bypass (Interfacing System LOCA)	1.15E+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 2.0E-7/yr yields 5.7E+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 $1L_a$ 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]

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

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

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 ½ λT 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 $\frac{1}{2} * \frac{3\%}{1.5 \text{ yrs}} * 10$ 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., $\frac{1}{2} * \frac{3\%}{1.5 \text{ yrs}} * 15$ 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 ($\%Risk_{10}$) for Class 3 is as follows:

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

Where:

$CLASS3a_{10}$ = Class 3a person-rem/year = 1.71E-2 person-rem/year [Table 5-5]
$CLASS3b_{10}$ = Class 3b person-rem/year = 1.96E-2 person-rem/year [Table 5-5]

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

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

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

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]

$TOTAL_{10}$ = 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 $1L_a$ 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:

$CLASS3a_{15}$ = Class 3a person-rem/year = 1.78E-2 person-rem/year [Table 5-6]

$CLASS3b_{15}$ = Class 3b person-rem/year = 2.06E-2 person-rem/year [Table 5-6]

$TOTAL_{15}$ = Total person-rem/yr for 15-year interval = 2.4407 person-rem/yr [Table 5-6]

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

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

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 $1L_a$ 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.

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

Where:

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

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

$$\%Risk_{10\text{-}15} = [(3.84E\text{-}2 - 3.67E\text{-}2) / 3.67E\text{-}2)] \times 100$$
$$\%Risk_{10\text{-}15} = 4.6\%$$

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 <u>total integrated plant risk</u> when the ILRT is extended from 10 years to 15 years is computed as follows:

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

Where:

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

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

$$\%TOTAL_{10\text{-}15} = [(2.4407 - 2.439) / 2.439] \times 100$$
$$\%TOTAL_{10\text{-}15} = 0.07\%$$

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 ($\Delta Risk_{15}$) due to a fifteen-year ILRT over the baseline case is as follows:

$$\Delta Risk_{15} = [(Total_{15} - Total_{BASE} \ / \ Total_{BASE}] \times 100.0$$

Where:

$TOTAL_{BASE}$ = 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]

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

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

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 (6.87E+4 person-rem/1.06E+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 2.86E-7/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^{-6}/yr and increases in LERF below 10^{-7}/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 (2.86E-7/yr) by the increase in overall probability of leakage (0.05) gives an increase in LERF of 1.43E-8/yr. Guidance in Reg. Guide 1.174 defines very small changes in LERF as below 1E-7/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 2.86E-7/yr multiplied by the 12% incremental increase in overall probability for a fifteen-year test interval (i.e., 15% - 3%) is 3.4E-8/yr, which is also well below the 1.0E-7/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.

$$\Delta CCFP = CCFP_{15} - CCFP_{10} = \frac{(\text{Class 3a + Class 3b})_{15} - (\text{Class 3a + Class 3b})_{10}}{CDF}$$

$$= 0.435\%$$

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 1.43E-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 3.14E-8/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}/yr and increases in LERF below 10^{-7}/yr. Since the ILRT does not impact CDF, the relevant criterion is LERF. The increase in LERF resulting from a change in the Type A ILRT test frequency from once-per-ten years to once-per-fifteen years is 1.43E-8/yr. Guidance in Reg. Guide 1.174 defines very small changes in LERF as below 1E-7/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

REFERENCES

[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.

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[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.

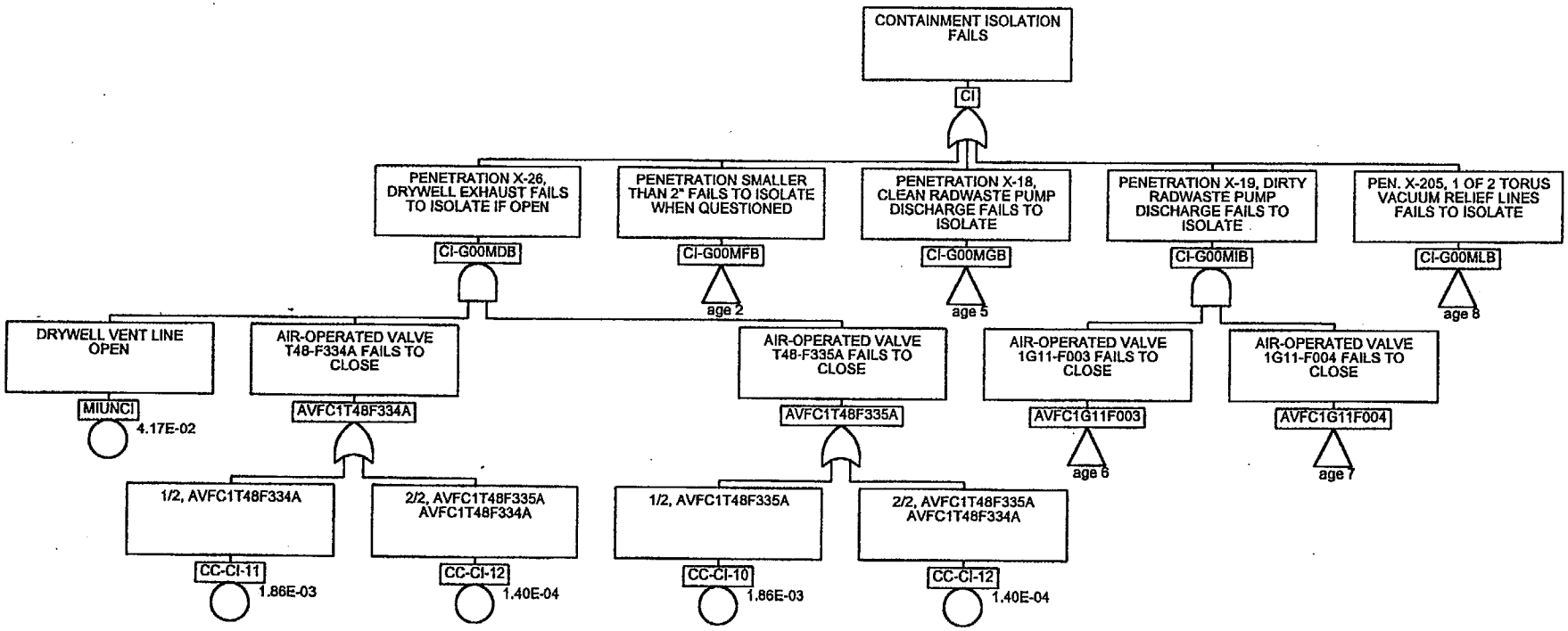
[21] Letter from SNC (H.L. Summer, Jr.) to USNRC, HL-5982.

[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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CI
PENETRATION X-26, DRYWELL EXHAUST FAILS TO ISOLATE IF OPEN
CI-G00MDB
PENETRATION SMALLER THAN 2" FAILS TO ISOLATE WHEN QUESTIONED
CI-G00MFB
age 2
PENETRATION X-18, CLEAN RADWASTE PUMP DISCHARGE FAILS TO ISOLATE
CI-G00MGB
age 5
PENETRATION X-19, DIRTY RADWASTE PUMP DISCHARGE FAILS TO ISOLATE
CI-G00MIB
PEN. X-205, 1 OF 2 TORUS VACUUM RELIEF LINES FAILS TO ISOLATE
CI-G00MLB
age 8
DRYWELL VENT LINE OPEN
MIUNCI
4.17E-02
AIR-OPERATED VALVE T48-F334A FAILS TO CLOSE
AVFC1T48F334A
1/2, AVFC1T48F334A
CC-CI-11
1.86E-03
2/2, AVFC1T48F335A AVFC1T48F334A
CC-CI-12
1.40E-04
AIR-OPERATED VALVE T48-F335A FAILS TO CLOSE
AVFC1T48F335A
1/2, AVFC1T48F335A
CC-CI-10
1.86E-03
2/2, AVFC1T48F335A AVFC1T48F334A
CC-CI-12
1.40E-04
AIR-OPERATED VALVE 1G11-F003 FAILS TO CLOSE
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age 6
AIR-OPERATED VALVE 1G11-F004 FAILS TO CLOSE
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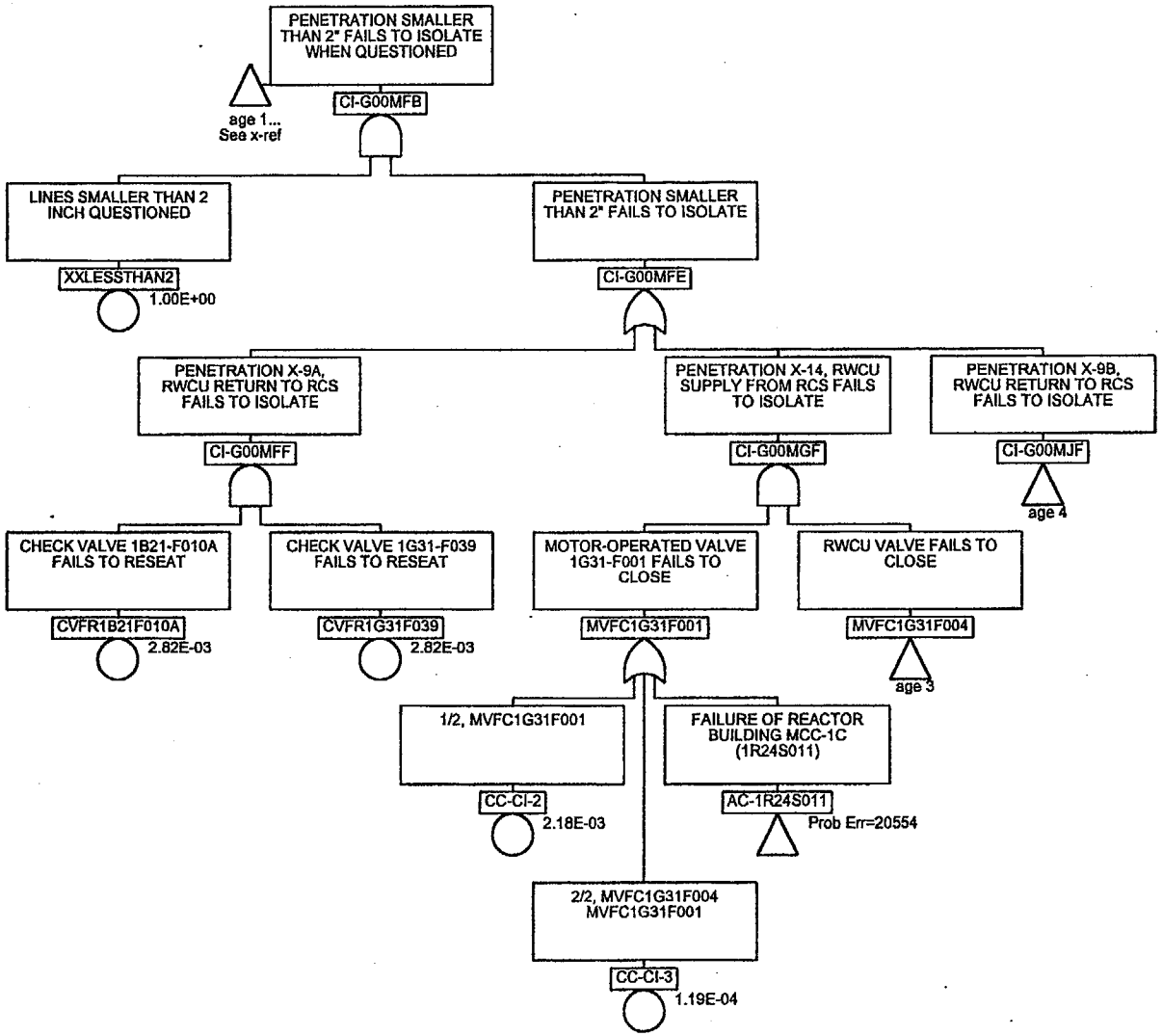
age 7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CI-G00MFB
age 1... See x-ref
LINES SMALLER THAN 2 INCH QUESTIONED
XXLESSTHAN2
1.00E+00
PENETRATION SMALLER THAN 2" FAILS TO ISOLATE
CI-G00MFE
PENETRATION X-9A, RWCU RETURN TO RCS FAILS TO ISOLATE
CI-G00MFF
PENETRATION X-14, RWCU SUPPLY FROM RCS FAILS TO ISOLATE
CI-G00MGF
PENETRATION X-9B, RWCU RETURN TO RCS FAILS TO ISOLATE
CI-G00MJF
age 4
CHECK VALVE 1B21-F010A FAILS TO RESEAT
CVFR1B21F010A
2.82E-03
CHECK VALVE 1G31-F039 FAILS TO RESEAT
CVFR1G31F039
2.82E-03
MOTOR-OPERATED VALVE 1G31-F001 FAILS TO CLOSE
MVFC1G31F001
RWCU VALVE FAILS TO CLOSE
MVFC1G31F004
age 3
1/2, MVFC1G31F001
CC-CI-2
2.18E-03
FAILURE OF REACTOR BUILDING MCC-1C (1R24S011)
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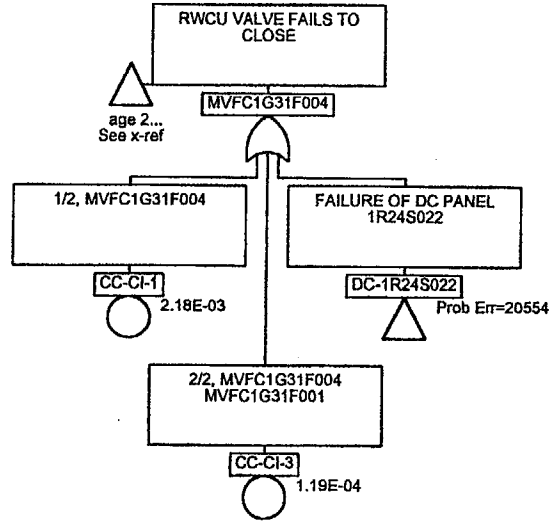
CC-CI-3
1.19E-04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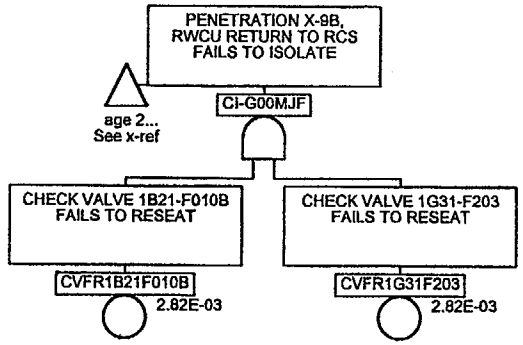
CC-CI-3
1.19E-04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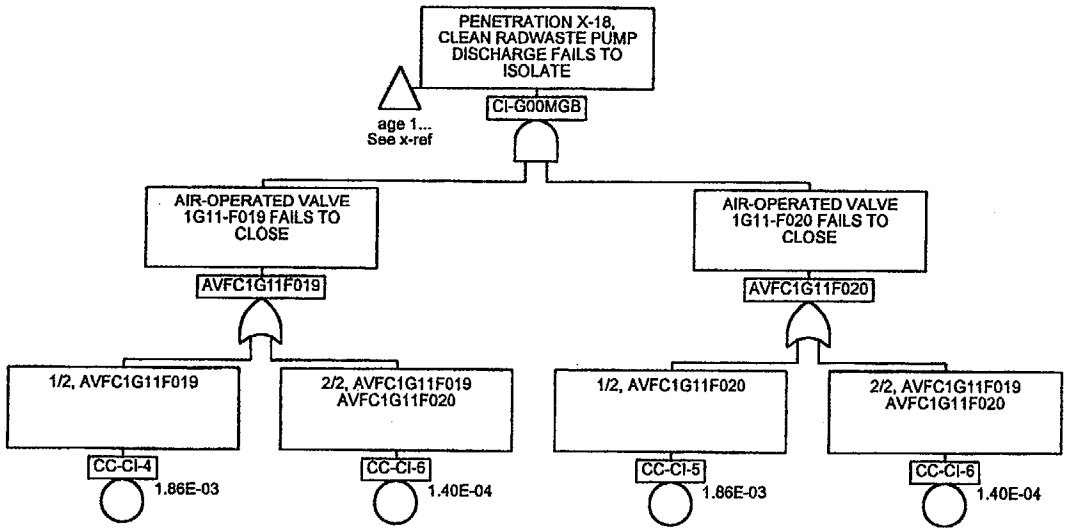
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CI-G00MGB
age 1...
See x-ref
AIR-OPERATED VALVE 1G11-F019 FAILS TO CLOSE
AVFC1G11F019
AIR-OPERATED VALVE 1G11-F020 FAILS TO CLOSE
AVFC1G11F020
1/2, AVFC1G11F019
CC-CI-4
1.86E-03
2/2, AVFC1G11F019 AVFC1G11F020
CC-CI-6
1.40E-04
1/2, AVFC1G11F020
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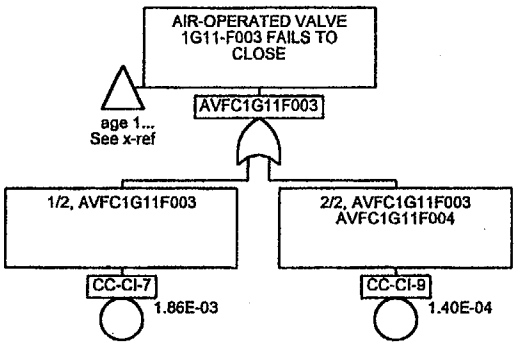
CC-CI-6
1.40E-04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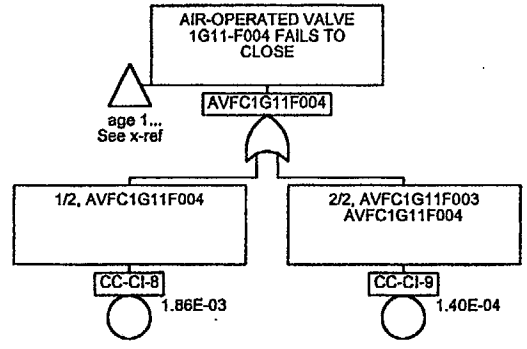
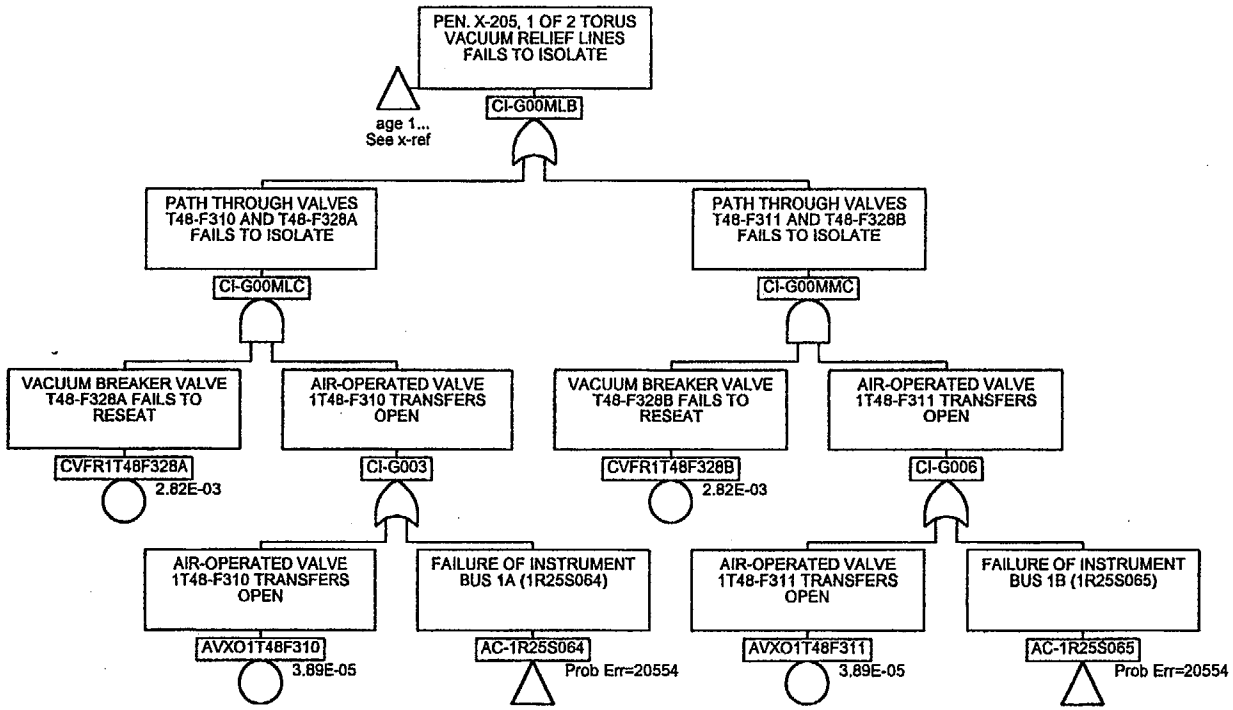
AVFC1G11F003
age 1...
See x-ref
1/2, AVFC1G11F003
2/2, AVFC1G11F003 AVFC1G11F004
CC-CI-7
1.86E-03
CC-CI-9
1.40E-04


AIR-OPERATED VALVE
1G11-F004 FAILS TO
CLOSE
AVFC1G11F004
age 1...
See x-ref
1/2, AVFC1G11F004
2/2, AVFC1G11F003
AVFC1G11F004
CC-CI-8
1.86E-03
CC-CI-9
1.40E-04


PEN. X-205, 1 OF 2 TORUS VACUUM RELIEF LINES FAILS TO ISOLATE
CI-G00MLB
age 1...
See x-ref
PATH THROUGH VALVES T48-F310 AND T48-F328A FAILS TO ISOLATE
CI-G00MLC
PATH THROUGH VALVES T48-F311 AND T48-F328B FAILS TO ISOLATE
CI-G00MMC
VACUUM BREAKER VALVE T48-F328A FAILS TO RESEAT
CVFR1T48F328A
2.82E-03
AIR-OPERATED VALVE 1T48-F310 TRANSFERS OPEN
CI-G003
VACUUM BREAKER VALVE T48-F328B FAILS TO RESEAT
CVFR1T48F328B
2.82E-03
AIR-OPERATED VALVE 1T48-F311 TRANSFERS OPEN
CI-G006
AIR-OPERATED VALVE 1T48-F310 TRANSFERS OPEN
AVXO1T48F310
3.89E-05
FAILURE OF INSTRUMENT BUS 1A (1R25S064)
AC-1R25S064
Prob Err=20554
AIR-OPERATED VALVE 1T48-F311 TRANSFERS OPEN
AVXO1T48F311
3.89E-05
FAILURE OF INSTRUMENT BUS 1B (1R25S065)
AC-1R25S065
Prob Err=20554

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						
CC-CI-3	3	2						
CC-CI-4	5	1						
CC-CI-5	5	3						
CC-CI-6	5	2						
CC-CI-6	5	4						
CC-CI-7	6	1						
CC-CI-8	7	1						
CC-CI-9	6	2						
CC-CI-9	7	2						
CI	1	5						
CI-G003	8	2						
CI-G006	8	4						
CI-G00MDB	1	3						
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						

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	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	
	XXLESSTHAN2	LINES SMALLER THAN 2 INCH QUESTIONED		1.00E+00	1.00E+00	
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	8.76E+03	3.29E-03	
	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	8.76E+03	3.29E-03	
	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 EVNET (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		2.18E-03	2.18E-03	1.73E-08
	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	

#	Inputs	Description	Rate	Exposure	Event Prob	Probability
27	CC-RWISO-2	1/2, MVFC1G31F004		2.18E-03	2.18E-03	1.73E-08
	MCOR1R24S011	RX BLDG 600-V MCC 1C FAILS	3.31E-07	2.40E+01	7.94E-06	
	XXLESSTHAN2	LINES SMALLER THAN 2 INCH QUESTIONED		1.00E+00	1.00E+00	
28	%FL-BUSC	FLAG FOR LOSS OF 600-V BUS C INITIATING EVENT		1.00E+00	1.00E+00	1.71E-08
	CBXO1R22S005_10I	4160-V SUPPLY BRKR TO XFMR C XFERS OPEN	1.74E-07	8.76E+03	1.52E-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	
29	%FL-BUSC	FLAG FOR LOSS OF 600-V BUS C INITIATING EVENT		1.00E+00	1.00E+00	1.71E-08
	CBXO1R23S003_2MI	600-V LOAD BRKR FROM XFMR C TRANSFERS OPEN	1.74E-07	8.76E+03	1.52E-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	
30	C2XO1R22S017_4B	CIRCUIT BREAKER (LOW VOLTAGE) TRANSFERS OPEN	2.68E-07	2.40E+01	6.43E-06	1.40E-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	
31	%LOSP	LOSP INITIATING EVENT		1.89E-02	1.89E-02	1.06E-08
	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	
	MNUNPS_TRNA	MAINT ON PSW PUMP C001A		1.57E-02	1.57E-02	
	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	
32	%LOSP	LOSP INITIATING EVENT		1.89E-02	1.89E-02	1.04E-08
	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	
	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	
33	CBXO1R23S003_7M	SUPPLY BREAKER TO RX BLDG 600-V MCC 1C TRANSFERS OPEN	1.74E-07	2.40E+01	4.18E-06	9.11E-09
	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	
34	%LOSP	LOSP INITIATING EVENT		1.89E-02	1.89E-02	8.76E-09
	CC-DGS-2	1/3, DGLR1R43S001A		3.18E-02	3.18E-02	
	CC-DGS-3	1/3, DGLR1R43S001B		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	
	XXLESSTHAN2	LINES SMALLER THAN 2 INCH QUESTIONED		1.00E+00	1.00E+00	
35	%LOSP	LOSP INITIATING EVENT		1.89E-02	1.89E-02	8.52E-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	
	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
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	BirnBm	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 EVEN
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	BirnBm	Red W	Ach W	Description
CC-DGS-2	3.18E-02	7.41E-05	1.02E-06	1.000	1.00	1/3, DGLR1R43S001A
C2XO1R22S017_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
CBXO1R22S005_10I	1.52E-03	4.81E-05	1.39E-05	1.000	1.03	4160-V SUPPLY BRKR TO XFMR C XFERS OPEN
CBXO1R23S003_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
CBXO1R23S003_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, DGSR1R43S001A
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
FUSO1R25S064	2.21E-05	2.83E-06	5.64E-05	1.000	1.13	SUPPLY FUSE PREMATURELY OPENS
FUSO1R25S065	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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