NMC
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DAEC Plant Support Center
Operated by Nuclear Management Company, LLC

March 29, 2002
NG-02-0232

Office of Nuclear Reactor Regulation
U.S. Nuclear Regulatory Commission
Attn: Document Control Desk
Mail Station 0-P1-17
Washington, DC 20555-0001

Subject: Duane Arnold Energy Center
Docket No: 50-331
Op. License No: DPR-49
Technical Specification Change Request (TSCR-055): Deferral of Type A Containment Integrated Leak Rate Test (ILRT)
File: A-117

In accordance with the Code of Federal Regulations, Title 10, Sections 50.59 and 50.90, Nuclear Management Company, LLC (NMC) hereby requests revision to the Technical Specifications for the Duane Arnold Energy Center (DAEC). This proposed change will revise Technical Specifications 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 September 2008.

This proposed change is similar to an amendment approved for Indian Point 3 to allow a one-time change in the Type A test interval from 10 years to a test interval of 15 years (NRC letter to Entergy Nuclear Operations, Inc. for Indian Point Nuclear Generating Unit No.3, TAC No. MB0178, dated April 17, 2001.) Similar amendments have also been approved for several other facilities, including Brunswick, Unit 1 and Hatch, Unit 1 (NRC letter to Carolina Power and Light Company for Brunswick Steam Electric Plant, Unit 1, TAC No. MB3470, dated March 6, 2002; and NRC letter to Southern Nuclear Operating Company, Inc. for Edwin I. Hatch Nuclear Plant, Unit 1, TAC No. MB2842, dated February 20, 2002).

Attachment 1 details the bases for NMC's determination that the proposed change does not involve a significant hazards consideration. Attachment 2 provides the marked-up page. Attachments 3 and 4 provide the safety assessment and Environmental Consideration, respectively. Attachment 5 provides a risk assessment for the DAEC regarding ILRT (Type A) extension. This risk assessment was performed consistent with the assessment performed for Indian Point 3's ILRT deferral submittal. In addition, a sensitivity calculation was performed using methodology developed by NEI for the performance of risk assessments in support of one-time extensions for ILRT surveillance intervals.

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This application has been reviewed by the DAEC Operations Committee and the Offsite Review Committee. A copy of this submittal, along with the 10 CFR 50.92 evaluation of "No Significant Hazards Consideration," is being forwarded to our appointed state official pursuant to 10 CFR Section 50.91.

NMC requests approval of the proposed amendment prior to December 31, 2002, in order to facilitate planning and scheduling for the next DAEC refueling outage, currently scheduled to begin in March 2003.

This letter is true and accurate to the best of my knowledge and belief.

NUCLEAR MANAGEMENT COMPANY, LLC

By Gary Van Middlesworth
Gary Van Middlesworth
DAEC Site Vice-President

State of Iowa
(County) of Linn

Signed and sworn to before me on this 29th day of March, 2002,

by Gary Van Middlesworth.

Nancy S. Franck
Notary Public in and for the State of Iowa

NANCY S. FRANCK
Commission Number 011041
MY COMMISSION EXPIRES
9-28-04

Commission Expires

Attachments: 1) Evaluation of Change Pursuant to 10 CFR Section 50.92
2) Proposed Change TSCR-055 to the Duane Arnold Energy Center Technical Specifications
3) Safety Assessment
4) Environmental Consideration
5) Risk Assessment

cc: C. Rushworth
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EVALUATION OF CHANGE PURSUANT TO 10 CFR SECTION 50.92

Background:

The Duane Arnold Energy Center (DAEC) Technical Specifications (TS) currently specify that Containment Integrated Leakage Rate Tests (ILRTs) be performed in accordance with the guidelines contained in Regulatory Guide 1.163, "Performance-Based Containment Leak-Test Program," dated September 1995. This results in a TS requirement to perform an ILRT prior to September 2003.

Nuclear Management Company, LLC, Docket No. 50-331
Duane Arnold Energy Center, Linn County, Iowa
Date of Amendment Request: March 29, 2002

Description of Amendment Request:

The DAEC TS would be revised to allow a one-time extension of the ILRT frequency from 10 years to 15 years.

Basis for proposed No Significant Hazards Consideration:

The Commission has provided standards (10 CFR Section 50.92(c)) for determining whether a significant hazards consideration exists. A proposed amendment to an operating license for a 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; (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.

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

1. The proposed amendment will not involve a significant increase in the probability or consequences of an 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 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 in 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.

Therefore, the proposed Technical Specification change does not involve a significant increase in the probability of an accident previously evaluated.

The consequences of the evaluated accidents are the amount of radioactivity that is released to secondary containment and subsequently to the public. 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. The DAEC ILRT test history supports this conclusion. NUREG-1493, Performance-Based Containment Leak-Test Program, 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 containment inspections performed in accordance with ASME Section XI, the Maintenance Rule and the DAEC's response to NRC Generic Letter 98-04 ("Potential for Degradation of the Emergency Core Cooling System (ECCS) and Containment Spray System (CSS) after a Loss-of-Coolant Accident (LOCA) because of Construction and Protective Coating Deficiencies and Foreign Material in Containment") 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, thus maintaining containment leakage low. 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.

Therefore, the proposed Technical Specification change does not involve a significant increase in the consequences of an accident previously evaluated.

2. The proposed amendment will 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. Primary containment is designed to contain energy and fission products during and after an event. 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. Revision to the Type A test interval does not change the events that could lead to containment failure. There are no physical changes being made to the plant and there are no changes to the operation of the plant that could introduce a new failure mode creating an accident.

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 amendment will not involve a significant reduction in a margin of safety.

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 proposed change involves only the extension of the interval between Type A containment leakage tests. The current interval of 10 years, based on past performance, would be extended on a one-time basis to 15 years from the last Type A test. Type B and C containment leakage tests will continue to be performed at the frequency currently required by plant Technical Specifications.

The NUREG-1493 generic study of the effects of extending containment leakage testing found that a 20-year interval in Type A leakage testing resulted in an imperceptible increase in risk to the public. NUREG-1493 found that, generically, the design containment leakage rate contributes about 0.1% to the individual risk and that increasing the Type A test interval would have minimal affect on this risk since about 95% of the potential leakage paths are detected by Type B and Type C testing. The DAEC 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 DAEC's response to NRC Generic Letter 98-04 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.

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. Additionally, the on-line containment monitoring capability that is inherent to inerted BWR containments allows for the detection of gross containment leakage should it develop during power operation.

Therefore, the proposed Technical Specification change will not involve a significant reduction in a margin of safety.

Based upon the above, NMC has determined that the proposed amendment will not involve a significant hazards consideration.

Local Public Document Room Location: Cedar Rapids Public Library, 500 First Street SE, Cedar Rapids, Iowa 52401

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PROPOSED CHANGE TSCR-055 TO THE DUANE ARNOLD ENERGY CENTER TECHNICAL SPECIFICATIONS

The holders of license DPR-49 for the Duane Arnold Energy Center propose to amend the Technical Specifications by deleting the referenced page and replacing it with the enclosed new page. Following this page is the marked-up page for this change.

SUMMARY OF CHANGES:

Page	Description of Changes
5.0-17	Revise TS 5.5.12, Primary Containment Leakage Rate Testing Program, to allow the first Type A test after the September 1993 Type A test to be performed no later than September 2008.

5.5 Programs and Manuals

5.5.11 Safety Function Determination Program (SFDP) (continued)

2. Provisions for ensuring the plant is maintained in a safe condition if a loss of function condition exists;

3. Provisions to ensure that an inoperable supported system's Completion Time is not inappropriately extended as a result of multiple support system inoperabilities; and

4. Other appropriate limitations and remedial or compensatory actions.

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

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

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

3. A required system redundant to support system(s) for the supported systems (1) and (2) above is also inoperable.

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

5.5.12 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

(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":

The first Type A test after the September 1993 Type A test shall be performed no later than September 2008.

SAFETY ASSESSMENT

Background:

By letter dated March 29, 2002, Nuclear Management Company, LLC (NMC) submitted a request for revision of the Technical Specifications (TS) for the Duane Arnold Energy Center (DAEC). The proposed amendment revises TS 5.5.12 to reflect a one-time deferral of the Type A Containment Integrated Leak Rate Test (ILRT) to no later than September 2008.

The proposed change involves a one-time exception to the ten (10) year frequency of the performance-based leakage rate testing program for Type A tests as required by Nuclear Energy Institute (NEI) 94-01, Revision 0, "Industry Guideline for Implementing Performance-Based Option of 10 CFR 50, Appendix J." The current ten (10) year ILRT for the DAEC is due in September 2003, which would require it to be performed during Refueling Outage (RFO) 18, presently scheduled to begin in March 2003. The proposed exception would allow the next ILRT for the DAEC to be performed within fifteen (15) years (September 2008) from the last ILRT as opposed to the current ten (10) year frequency.

ILRTs have been required of operating nuclear power plants to ensure the public health and safety in the case of an accident that releases radioactivity to the primary containment. Conservative design and construction combined with stringent configuration control have led to very few primary containment concerns identified during Type A testing. The NRC has extended the allowable ILRT test period from three times in ten years to once in ten years using a performance based evaluation of past successful tests. NUREG-1493 that supported the change also states that test periods of up to twenty years do not significantly increase public risk.

The justification for this request is based on past successful Type A, B, and C tests, and American Society of Mechanical Engineers (ASME) Section XI inspections at the DAEC. Further justification is based on research documented in NUREG-1493 that very few potential containment leakage paths fail to be identified by Type B and C tests. 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. In fact, only 5 out of 180 ILRTs had failures that could not be detected by Type B and C tests. NUREG-1493 documents no failures of the containment liner.

This request is similar to an amendment approved for Indian Point 3 (NRC letter to Entergy Nuclear Operations, Inc. for Indian Point Nuclear Generating Unit No.3, TAC No. MB0178, dated April 17, 2001.) Similar amendments have also been approved for Brunswick, Unit 1 and Hatch, Unit 1 (NRC letter to Carolina Power and Light Company for Brunswick Steam Electric Plant, Unit 1, TAC No. MB3470, dated March 6, 2002; and NRC letter to Southern Nuclear Operating Company, Inc. for Edwin I. Hatch Nuclear Plant, Unit 1, TAC No. MB2842, dated February 20, 2002).

Without approval of this proposed TS change, an ILRT would be required to be performed at the DAEC during the next refueling outage (RFO 18), currently scheduled to begin in March of 2003. With approval of this request, NMC will reschedule the ILRT for a later RFO. A substantial cost savings will be realized during this outage, 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 RFO 18 at approximately $850,000, which includes labor, equipment and critical path outage time needed to perform the test. The requested approval date of December 31, 2002 allows NMC to finalize the DAEC outage schedule and vendor support in advance of the Spring 2003 RFO start date.

Basis for Change:

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 219 was issued for the DAEC (dated October 4, 1996) to permit implementation of 10 CFR 50, Appendix J, Option B. Amendment 219 revised Technical Specifications to establish a Primary Containment Leakage Rate Testing Program in accordance with Regulatory Guide (RG) 1.163, "Performance-Based Containment Leak-Test Program." RG 1.163 specifies a method acceptable to the NRC for complying with 10 CFR 50, Appendix J, Option B by approving the use of NEI 94-01 and ANSI/ANS 56.8-1994, subject to several regulatory positions in the guide. Exceptions to the requirements of RG 1.163 are permitted by 10 CFR 50, Appendix J, Option B, as discussed in Section V.B, "Implementation." Therefore, this application does not require an exemption from 10 CFR 50, Appendix J, Option B.

Adoption of the Option B performance-based containment leakage rate testing program did not alter the basic method by which Appendix J leakage rate testing is performed; however, it did alter the frequency at which Type A, B, and C containment leakage tests must be performed. Under the performance-based option of 10 CFR 50, Appendix J, test frequency is based upon an evaluation that reviews "as-found" leakage 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. (The DAEC 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 1.0 L_a. Based upon the DAEC's ILRT history (outlined below), the current test interval for the DAEC is once every ten (10) years, with the next test due to be performed by September 2003.

<u>DAEC Integrated Leak Rate Test History</u>

The DAEC has performed one preoperational Type A test and 7 operational Type A tests. The results of these tests demonstrate that the DAEC containment structure remains an essentially leaktight barrier and represents minimal risk to increased leakage. These plant specific results support the conclusions of NUREG-1493. As specified in the DAEC Technical Specifications Section 5.5.12, the as-left criterion is 0.75 of the maximum allowable primary containment leak rate (L_a). The as-found acceptance criterion is 1.0 L_a. L_a is 2.0% by weight of the containment air per 24 hours at P_a. The as-left criterion in weight % per day is 1.5% per day and the as-found acceptance criterion is 2.0% per day. The DAEC ILRT results are provided below.

DAEC Type A Test Results

Test Date	As-Left Test Result (Total Time Upper Confidence Limit)	As-Found Test Result (Total Time Upper Confidence Limit)	As-Left Type B & C Min. Path Leakage and water level correction	Type A Leakage not attributed to Type B & C Penetrations
Dec. 29, 1973	0.129 %/day	Pre		
April 15, 1978	0.380 %/day		Not calculated	Not calculated
April 18, 1983	0.626 %/day	1.701 %/day	0.082 %/day	0.544 %/day
July 5, 1985	0.478 %/day	Not quantified **	0.061 %/day	0.417 %/day
June 1, 1987	0.503 %/day	Not quantified **	0.070 %/day	0.433 %/day
Dec. 15, 1988	0.229 %/day	1.353 %/day	0.033 %/day	0.196 %/day
August, 29,1990	1.146 %/day	1.633 %/day	0.082 %/day	1.064 %/day
Sept. 20, 1993	0.254 %/day	0.511 %/day	0.100 %/day	0.154 %/day

** - Penetration X-9B (feedwater) as-found leakage was not quantified.

DAEC Containment Design

The DAEC is a GE designed boiling water reactor in a Mark I containment. The drywell is a steel pressure vessel (0.75 to 3.0 inches thick, with a spherical lower portion and cylinder upper portion. It is enclosed in reinforced concrete, 4 to 7 feet thick, for shielding, and to provide additional resistance to deformation and buckling over areas where the concrete backs up the steel shell. Above the foundation transition zone, and below the flange, the drywell is separated from the reinforced concrete by a gap of approximately 2 inches to allow for thermal expansion. Shielding over the top of the drywell is provided by removable, segmented, reinforced concrete shield plugs.

The drywell vessel is provided with a removable head to facilitate refueling, one combination double door personnel access lock/equipment lock, one equipment hatch, one personnel access hatch, and one control rod drive removal hatch. The head and hatches are all bolted in place and have double seals and test tap for leak tests.

Special bellows seals are provided between the reactor vessel, the drywell vessel, and the reactor well to form a watertight seal and enable flooding of the upper portion of the drywell during refueling operations. To protect the outer circumference of the bellows a backing plate is provided which has a test tap for leakage monitoring. During normal operation, six watertight hinged covers are opened and removable air supply and return ducts permit circulation of ventilation air in the region above the reactor well seal bulkhead plate.

The pressure suppression chamber is a steel pressure vessel in the shape of a torus located below and encircling the drywell. The suppression chamber will transmit seismic loading to the reinforced concrete foundation slab of the Reactor Building. Space is provided outside the chamber for inspection. Access to the chamber is provided by two 4-foot diameter manhole entrances with double gasketed (leak testable) bolted covers connected to the chamber by 4 foot diameter steel pipe inserts.

Eight 4 foot 9 inch diameter vent pipes connect the drywell and the pressure suppression chamber. Jet deflectors are provided in the drywell at the entrance of each vent pipe to prevent possible damage to the vent pipes from jet forces or projectiles which might accompany a pipe break in the drywell. The vent pipes are provided with two-ply expansion bellows to accommodate differential motion between the drywell and suppression chamber. These bellows have test connections which allow for leak testing and for determining that the passages between the two-ply bellows are not obstructed.

The drywell vents are connected to a 3 foot 6 inch diameter vent header in the form of a torus which is contained within the air space of the suppression chamber. Projecting downward from the header are 48 downcomer pipes, 24 inches in diameter and terminating a minimum of 3 feet below the water surface of the pool and approximately 7 feet above the bottom of the Torus.

Plant Operational Performance

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 Atmosphere Control 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 recorded and periodically monitored in the Main Control Room. Although this feature, inherent to the DAEC 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, and is periodically monitored, provides assurance that if gross containment leakage developed during power operation, it would be detected.

Containment Inspections

The leak rate testing requirements (ILRT and LLRTs) of Option B of Appendix J, and the containment inservice inspection requirements mandated by 10 CFR 50.55a complement each other in ensuring the leaktightness and structural integrity of the containment. Therefore, additional information related to containment inspection is provided below, along with information, which addresses issues raised by the Staff during their review of similar ILRT extension requests.

IWE Program

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 DAEC containment is a free-standing steel containment, to which only the requirements of Subsection IWE apply.

The DAEC IWE Program meets the requirements of the 1992 Edition with the 1992 Addenda of ASME Section XI. The First Ten-Year Containment Inspection Interval started September 9, 1996 with the first period examinations completed by September 9, 2001 (as required by 10 CFR 50.55a(g)(6)(ii)(B)(1)). The three inspection periods during the containment inspection interval are as follows:

First Period:	September 9, 1996 - September 8, 2001
Second Period:	September 9, 2001 - December 8, 2004
Third Period:	December 9, 2004 - April 7, 2007

The ASME IWE inspections include the interior liner and the exterior concrete surfaces. In general, the areas and items subject to inspection include the accessible class MC pressure retaining containment surface areas, including structural attachments and penetrations, seals, gaskets, moisture barriers, pressure retaining bolting, and Class MC supports. Exceptions taken to the ASME Section XI requirements have been documented and approved by the NRC as requests for relief. Inaccessible areas are evaluated for degradation when conditions in accessible areas indicate the presence of or result in degradation not meeting the established acceptance standards.

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

IWE-1240 Examinations

Subsubarticle IWE- 1240 of Subsection IWE of Section XI of the ASME Boiler and Pressure Vessel Code requires the identification of surface areas requiring augmented examinations. IWE-1241 provides the selection criteria for those areas requiring augmented examinations. Surface areas likely to experience accelerated degradation and aging require augmented examinations. Such areas include:

- interior and exterior containment surface areas 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. Typical locations of such areas are those exposed to standing water, repeated wetting and drying, persistent leakage, and those with geometries that permit water accumulation, condensation, and microbiological attack.

Such areas may include penetration sleeves, surfaces wetted during refueling, concrete to steel shell or liner interfaces, embedment zones, leak chase channels, drain areas, or sump liners.

- interior and exterior containment surface areas that are subject to excessive wear from abrasion or erosion that causes a loss of protective coatings, deformation, or material loss. Typical locations of such areas are those subject to substantial traffic, sliding pads or supports, pins or clevises, shear lugs, seismic restraints, surfaces exposed to water jets from testing operations or safety relief valve discharge, and areas that experience wear from frequent vibrations.

IWE-2500-1 requires a VT-1 on 100% of Surface Areas Identified by IWE-1242.

Torus Augmented Examination

The DAEC Torus was initially coated in 1973 with an inorganic zinc protective coating CarboZinc 11 with a four foot wide band epoxy coating Phenoline 368 at the waterline. NMC has a very proactive inspection program for the torus interior. The torus has been inspected and coating repairs performed in 1977, 1980, 1981, and 1983. In 1985, all internal surfaces of the torus shell and external surfaces of the vent system were recoated with CarboZinc 11. Additional inspections and repairs to the coating were performed in 1987, 1988, 1990, 1992, and 1993. During the 1993 inspection, a quantitative inspection was performed on a one square foot evaluation area representative of the worst-case corrosion observed during the qualitative inspections. A grid was established and the coordinates of each pit in the evaluation area were recorded so that the rate of corrosion (pit depth) could be monitored and trended in the future. Repairs to the coating are made as necessary.

NMC will continue to perform a general visual examination on 100% of the torus exterior and interior surface (above the water line) each period. In addition a VT-1 visual examination on 100% of the torus interior surfaces (below the water line) will be performed twice per interval. Since both sides are accessible for visual examination, no volumetric examination is required.

This examination provides an acceptable way to monitor the coating on the interior surface of the torus. Areas which are detected to have a "corrosion cell" (small area of uncoated metal, typically 1/4" to 1/2") will be repaired. Performing a general visual of 100% of the exterior and interior surfaces (above the water line), VT-1 of 100% of the interior surfaces (below the water line) twice per interval provides an assurance that the structural integrity of the torus is acceptable. Repairs to the coating are performed when necessary.

During RFO 16, a visual examination (VT-3) was conducted on the exterior surface of Torus Bay 15. This examination revealed a degraded condition in the coating (1"x2" area) which required engineering evaluation. The evaluation accepted the degraded condition, however the cause of the degraded condition was determined to be leakage of water and grease from MO-2001, which is located above. A repair was performed on the coating. This 1"x2" area is currently included in the Augmented Category E-C, Item E4.11.

Torus Repair (RFO 17)

During RFO 17, inspections of welds performed in accordance with the DAEC IWE Program, identified a portion of approximately 26 1/4" incomplete weld located between the torus shell and a ring girder. The weld was repaired with underwater welding (NG-01-0589 dated May 1, 2001, NDE-R042 Request for Authorization to Use Code Case N-516-1).

IWE Program Relief Requests

By letter dated October 19, 1999, the NRC approved several relief requests regarding the DAEC's IWE Program, including MC-R002, MC-R003 and MC-R007.

Relief Request MC-R002 provides relief from the requirements of IWE-2500, Table IWE-2500-1, Category E-D, Item Numbers E5.10 and E5.20. The Code requires seals and gaskets to be visually examined once each interval. As an alternative, the relief allows the leak-tightness of seals and gaskets to be tested in accordance with 10 CFR 50, Appendix J. As discussed in MC-R002, for those penetrations that are routinely disassembled, the gaskets are considered safety-related, and inspected during receiving inspection looking for cuts or tears and maintenance personnel perform a final examination prior to installation. In addition, a Type B test is required upon final assembly and prior to start-up. The Type B test will assure the leak tight integrity of primary containment.

Relief Request MC-R003 provides relief from the requirements of IWE-2500, Table IWE-2500-1, Category E-G, Item Number E8.20 for all Class MC pressure retaining bolts. The Code requires a bolt torque or tension test for bolted connections that have not been disassembled and reassembled during the inspection interval. As an alternative, the relief allows the use of the 10 CFR 50, Appendix J, Type B test. As discussed in MC-R003, each penetration receives a 10CFR 50 Appendix J Type B test in accordance with the testing frequencies specified in Appendix J. The following examinations and tests required by Subsection IWE also ensure the structural integrity and the leak-tightness of Class MC pressure retaining bolting.

(1) Exposed surfaces of bolted connections shall be visually examined in accordance with requirements of Table IWE-2500-1, Examination Category E-G, Pressure Retaining Bolting, Item No. E8.10, and
(2) Bolted connections shall meet the pressure test requirements of Table IWE-2500-1, Examination Category E-P, All Pressure Retaining Components, Item E9.40, and
(3) A General Visual Examination of the entire containment once each inspection period shall be conducted in accordance with 10CFR50.55a(b)(2)(x)(E).

The Type B test frequencies discussed in MC-R002 and MC-R003 are in accordance with the DAEC Performance Based Containment Testing Program Manual. Initially, Type B testing for components other than airlocks is performed each refueling cycle, not to exceed 30 months, until acceptable performance is demonstrated. Following the completion of two consecutive periodic

tests with results within allowable limits, the testing interval may be extended to 120 months. If a test result is greater than the allowable limit, then Type B testing must be performed each refueling cycle until acceptable performance is demonstrated.

Relief Request MC-R007 provides relief from the requirements of Table IWE-2500-1, Examination of Category E-A, Items E1.12 and E1.20. Instead of performing visual examinations of the accessible surface areas of the containment and vent system at the end of the interval, the alternative allows the examinations to be performed in accordance with Code Case N-601. Code Case N-601 "Extent and Frequency of VT-3 Visual Examination for Inservice Inspection of Metal Containments, Section XI, Division 1" provides an alternative to perform the visual examinations at any time during the interval. As discussed in the Staff's SE, performing visual examinations on the accessible surfaces of the containment structure and vent system during the course of the inspection interval, as recommended in Code Case N-601, is more technically sound than performing all the visual examinations at the end of the interval. In doing this, the integrity of the containment and vent system can be better monitored between the 10 CFR 50, Appendix J testing, and the visual examinations required by Table IWE-2500-1.

By letter dated December 13, 2000, the NRC approved Relief Request MC-R008 regarding the limited examination of Drywell Stabilizer X-58A. During RFO 16, it was discovered that well water piping prohibited the removal of the bolting associated with the drywell stabilizer. Without removal of the bolting, the integral attachment and the associated reinforcing structure could not be examined. 10 CFR 50.55a(b)(2)(x)(A) of the 1992 Code of Federal Regulations (section 50.55a(b)(2)(ix)(A) in the 2000 edition of the Federal Code of Regulations) states that for Class MC applications, the licensee shall evaluate the acceptability of inaccessible areas when conditions exist in accessible areas that could indicate the presence of or result in degradation to such inaccessible areas. Relief Request MC-R008 provides an alternative of performing general visual examination of the accessible surfaces once per examination interval. As required by the approved Relief Request MC-R003, NMC will examine the bolting associated with the Drywell Stabilizer X-58A once per examination interval.

10 CFR 50.65 and GL 98-04

Prior to the inception of the containment in service 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.

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 the DAEC response to 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. Structural walkdown inspections of the interior and exterior of the primary containment conducted for the DAEC Maintenance Rule in 1996 and 2001 found no deficiencies related to containment integrity.

GL 87-05

A review was performed in response to GL 87-05, "Request for Additional Information - Assessment of Licensee Measures to Mitigate and/or Identify Potential Degradation of Mark I Drywells," dated March 12, 1987. The results of the review indicated that the DAEC was not subject to the same conditions that caused drywell shell degradation at another plant having a Mark I containment.

As discussed in the DAEC response to Generic Letter 87-05, the original air gap filler material was Polyurethane foam, which was removed during original plant construction. There are Ethafoam rings embedded into the concrete on the outside of the drywell air gap. The design of the sand cushion area at the DAEC uses an 18 gage galvanized sheet metal plate sealed to the drywell shell to cover the sand pocket. Any leakage of water into the air gap between the drywell and the surrounding concrete shield wall above the sheet metal plate is directed to the Torus Room basement via four 4-inch drain lines. If water does penetrate the sheet metal plate or seal and enters the sand pocket, four additional 2-inch sand-filled drain lines will drain the sand pocket to the Torus Room basement. In response to the GL, the four 4-inch drain lines were verified to be unplugged and the four 2-inch sand filled drain lines were visually inspected at the mesh screen caps for the presence of any water. No evidence of leakage was found.

The design of the drywell to reactor building refueling bellows prevents the leakage of water into the drywell air gap. Four 2-inch bellows area drain lines are seal welded to a carbon steel plate below the refueling bellows. Any leakage past the bellows area will be directed through 8-inch drain lines, which run concentric with the previously mentioned 2-inch lines for a large portion of their runs. A 2-inch lip between the air gap and the 8" drain lines prevents bellows leakage from entering the air gap. The 8-inch lines also serve to drain any other leakage past couplings in the refueling bellows drain lines. This leakage is subsequently directed past flow switches which will alarm upon excessive leakage.

Risk Assessment

A detailed performance based, risk-informed assessment "Risk Assessment for Duane Arnold Energy Center Regarding ILRT (Type A) Extension Request" was performed to support this request. This risk assessment was performed consistent with the assessment performed for Indian Point 3's ILRT deferral submittal.

The following conclusions are drawn from this assessment.

- Regulatory Guide 1.174 provides guidance for determining the risk impact of plant-specific

changes to the licensing basis. Regulatory Guide 1.174 defines very small changes in risk as resulting in increases of core damage frequency (CDF) below 10^{-6}/year and increases in Large Early Release Frequency (LERF) below 10^{-7}/year. Since the ILRT does not impact CDF, the relevant criterion is LERF. The increase in LERF resulting from a change in the Type A ILRT test interval from once-per-10 years to once-per-15 years is 1.20E-8. Guidance in Regulatory Guide 1.174 defines very small changes in LERF as below 10^{-7}/year. Therefore, increasing the ILRT interval from 10 to 15 years is considered to result in a very small change to the DAEC risk profile.

- The change in Type A test frequency from once-per-10 years to once-per-15 years increases the total integrated plant risk for those accident sequences influenced by Type A testing by only 0.08%. Therefore, the risk impact when compared to other severe accident risks is negligible.

A sensitivity calculation was also performed using methodology developed by NEI for the performance of risk assessments in support of one-time extensions of ILRT surveillance intervals. If the approach from the NEI methodology is used instead of the Indian Point 3 methodology, a slightly different measured potential impact on LERF, population dose and conditional containment failure probability (CCFP) from the proposed ILRT extension is calculated compared to the original analyses, but it does not change the conclusions.

Conclusion

The Containment Inspection Program at the DAEC was developed in accordance with the requirements of Subsection IWE of ASME Section XI, 1992 Edition (with the 1992 Addenda), including the NRC-approved requests for relief from certain code requirements. The combination of examinations under the DAEC Containment Inspection Program and visual examination of the accessible and immersed surfaces of the containment in accordance with the requirements of 10 CFR 50.65 and the DAEC's response to Generic Letter 98-04 will provide assurance that the leaktight integrity and the containment structural integrity will be maintained during the extended ILRT period.

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.

ENVIRONMENTAL CONSIDERATION

10 CFR Section 51.22(c)(9) identifies certain licensing and regulatory actions which are eligible for categorical exclusion from the requirement to perform an environmental assessment. A proposed amendment to an operating license for a facility requires no environmental assessment if operation of the facility in accordance with the proposed amendment would not: (1) involve a significant hazards consideration; (2) result in a significant change in the types or significant increase in the amounts of any effluents that may be released offsite; and (3) result in a significant increase in individual or cumulative occupational radiation exposure. Nuclear Management Company (NMC) has reviewed this request and determined that the proposed amendment meets the eligibility criteria for categorical exclusion set forth in 10 CFR Section 51.22(c)(9). Pursuant to 10 CFR Section 51.22(b), no environmental impact statement or environmental assessment needs to be prepared in connection with the issuance of the amendment. The basis for this determination follows:

Basis

The change meets the eligibility criteria for categorical exclusion set forth in 10 CFR Section 51.22(c)(9) for the following reasons:

1. As demonstrated in Attachment 1 to this letter, the proposed amendment does not involve a significant hazards consideration.

2. There is no significant change in the types or significant increase in the amounts of any effluents that may be released offsite. The operation of the plant is not being changed by this extension.

3. There is no significant increase in individual or cumulative occupational radiation exposure. The activities of plant personnel are not being changed by this extension.

Attachment 5 to
NG-02-0232

RISK ASSESSMENT FOR

DUANE ARNOLD ENERGY CENTER

REGARDING

ILRT (TYPE A) EXTENSION REQUEST

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December 2001

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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) to fifteen years. The extension would allow for substantial cost savings as the ILRT could be deferred for additional scheduled refueling outages for the Duane Arnold Energy Center (DAEC). 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 for 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.

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.

Previously, the NRC published a 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. In that analysis, it was 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, and increasing the leak rate to 50 percent per day increases the total population exposure by less than 1 percent. Consequently, extending the ILRT interval should not lead to any substantial increase in risk. The current analysis is being performed to confirm these conclusions based on Duane Arnold 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 PSA 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 local leak tests performed to verify the leak-tight integrity of containment penetration bellows, airlocks, seals, and gaskets are also not affected by the change to the Type A test frequency.

1.2 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 their parameters. (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 Duane Arnold PSA model that includes the results from the Duane Arnold 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.

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 containment leak risk assessments, the Duane Arnold assessment uses population dose as one of the risk measures. The other risk measures used in the Duane Arnold 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 Duane Arnold assessment uses information from NUREG-1273 [5] regarding the low percentage of containment leakage events that would only be detected by an ILRT 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 Duane Arnold evaluation uses similar ground rules and methods to calculate changes in risk metrics [6]. The NRC approval was granted on April 17, 2001 (TAC No. MB0178) [7].

Section 3
GROUND RULES

The following ground rules are used in the analysis:

- The Duane Arnold Level 1 and Level 2 internal events PSA model provides representative results for the analysis.
- It is appropriate to use the Duane Arnold internal events PSA 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 impact of the ILRT on shutdown risk is addressed using the generic results from EPRI TR 105189 [8].
- Dose results for the containment failures modeled in the PSA can be characterized by information provided in NUREG/CR-4551 [9]. They are estimated by scaling the NUREG/CR-4551 results by population differences for Duane Arnold compared to the NUREG/CR-4551 reference plant.
- The lowest consequence calculations (i.e., intact containment and small leakages) are also based on scaling the NUREG/CR-4551 results for such cases using population differences, and also based on differences in the allowable 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 [6, 7].
- The maximum containment leakage for Class 3b sequences is 35 L_a. based on the previously approved methodology [6, 7]

- Class 3b is conservatively categorized as LERF based on the previously approved methodology [6, 7]

- 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 included in the Duane Arnold Level 2 model, and made part of the LERF calculation.

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

The first study is applicable because it provides one basis for the threshold that could be used in the Level 2 PSA for the size of containment leakage that is considered significant and to be included in the model. The second study is applicable because it provides a basis of the probability for significant pre-existing containment leakage at the time of a core damage accident. The third study is applicable because it is a subsequent study to NUREG/CR-4220 that undertook a more extensive evaluation of the same database. The fourth study provides an assessment of the impact of different containment leakage rates on plant risk. The fifth study provides an assessment of the impact on shutdown risk from ILRT test interval extension. The sixth study is the NRC's cost-benefit analysis of various alternative approaches regarding extending the test intervals and increasing the allowable leakage rates for containment integrated and

local leak rate tests. The seventh study is an EPRI study of the impact of extending ILRT and LLRT test intervals on at-power public risk. Finally, the last studies provide an ex-plant consequence analysis for a 50-mile radius surrounding a plant that is used as the bases for the consequence analysis of the ILRT interval extension for DAEC.

NUREG/CR-3539 [10]

Oak Ridge National Laboratory (ORNL) documented a study of the impact of containment leak rates on public risk in NUREG/CR-3539. This study uses information from WASH-1400 [19] 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 [11]

NUREG/CR-4220 is a study performed by Pacific Northwest Laboratories (PNL) for the NRC in 1985. The study reviewed over two thousand LERs, ILRT reports and other related records to calculate the unavailability of containment due to leakage. The study calculated unavailabilities for Technical Specification leakages and "large" leakages. 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 Duane Arnold. NUREG/CR-4220 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 [5]

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 [12]

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

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

EPRI TR-105189 [8]

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

The result of the study concluded that a small but measurable safety benefit is realized from extending the test intervals. For the BWR, the benefit from 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 <u>improvement</u> in the safety due to extending the ILRT test interval.

<u>NUREG-1493 [4]</u>

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 <u>only</u> by an ILRT (i.e., and not by LLRTs)

This same approach is proposed in the Duane Arnold ILRT test interval extension risk assessment.

<u>EPRI TR-104285 [2]</u>

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
6. Other penetration related containment isolation failures

7. Containment failure due to core damage accident phenomena
8. Containment bypass

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

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

<u>NUREG-1150 [13] and NUREG/CR 4551 [9]</u>

NUREG-1150 and the technical basis, NUREG/CR-4551 [9], provide an ex-plant consequence analysis for a spectrum of accidents including a severe accident with the containment remaining intact (i.e., Technical Specification leakage). This ex-plant consequence analysis is calculated for the 50-mile radial area surrounding Peach Bottom. The ex-plant calculation can be delineated to total person-rem for each identified Accident Progression Bin (APB) from NUREG/CR-4551. With the Duane Arnold Level 2 model end-states assigned to one of the NUREG/CR-4551 APBs, it is considered adequate to represent Duane Arnold if the Technical Specification leakage and the population are scaled to represent Duane Arnold. (The meteorology and site differences other than population are assumed not to play a significant role in this evaluation.)

4.2 PLANT SPECIFIC INPUTS

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

- Level 1 Model
- Level 2 Model
- Release Category definitions used in the Level 2 Model
- Population Dose calculations by release category

- ILRT results to demonstrate adequacy of the administrative and hardware issues.[(1)]

Level 1 Model

The Level 1 PSA model that is used for Duane Arnold is characteristic of the as-built plant. The current Level 1 model is developed in CAFTA, and was quantified with the total Core Damage Frequency (CDF) = 1.18E-5/yr at a truncation of 3E-11/yr. [14]

Level 2 Model

The Level 2 Model that is used for Duane Arnold was developed to calculate the LERF contribution as well as the other release categories evaluated in the model. The Level 2 model was quantified using the CAFTA model. The total Large Early Release Frequency (LERF) which corresponds to the H/E release category in Table 4.2-1 was found to be 1.14E-6/yr at a truncation of 1E-12/yr[(2)]. Table 4.2-1 summarizes the pertinent Duane Arnold results in terms of release category. The total release frequency is 9.18E-6/yr. With a total CDF of 1.18E-5/yr, this corresponds to an "OK" release limited to normal leakage of 2.62E-6/yr [14].

(1) The two most recent Type A tests at Duane Arnold have been successful, so the current Type A test interval requirement is 10 years. In fact, the last 3 ILRT results at Duane Arnold have been successful [18].

(2) A truncation level of 1E-12 is generally used for the DAEC Level 2 quantification although a few sequences were quantified at slightly higher values due to code processing limitations.

Table 4.2-1
Duane Arnold Level 2 PSA Model Release Categories and Frequencies

Category	Frequency/yr
H/E – High Early (LERF)	1.14E-06
M/E – Medium Early	3.81E-06
L/E – Low Early	4.95E-07
LL/E – Low Low Early	1.12E-07
H/I – High Intermediate	3.74E-07
M/I – Medium Intermediate	1.92E-06
L/I – Low Intermediate	4.07E-07
LL/I – Low Low Intermediate	2.00E-08
H/L – High Late	1.99E-07
M/L – Medium Late	7.50E-08
L/L – Low Late	5.36E-07
LL/L – Low Low Late	9.48E-08
Total Release Frequency	**9.18E-06**
Core Damage Frequency	**1.18E-05**

Population Dose Calculations

The population dose is calculated by using data provided in NUREG/CR-4551 and adjusting the results for Duane Arnold. Each accident sequence was associated with an applicable collapsed Accident Progression Bin (APB) from NUREG/CR-4551. The collapsed APBs are characterized by 5 attributes related to the accident progression. Unique combinations of the 5 attributes result in a set of 10 bins that are relevant to the analysis. Information from the Duane Arnold PSA Containment Event Trees (CETs) was used to classify each of the Level 2 sequences using these attributes. The definitions of the 10 collapsed APBs are provided in NUREG/CR-4551 and are reproduced in Table 4.2-2 for references purposes. Table 4.2-3 summarizes the calculated population dose associated with each APB from NUREG/CR-4551.

Table 4.2-2
Collapsed Accident Progression Bin (APB) Descriptions [9]

Collapsed APB Number	Description
1	CD, VB, Early CF, WW Failure, RPV Pressure > 200 psi at VB Core damage occurs followed by vessel breach. The containment fails early in the wetwell (i.e., either before core damage, during core damage, or at vessel breach) and the RPV pressure is greater than 200 psi at the time of vessel breach (this means Direct Containment Heating (DCH) is possible).
2	CD, VB, Early CF, WW Failure, RPV Pressure < 200 psi at VB Core Damage occurs followed by vessel breach. The containment fails early in the wetwell (i.e., either before core damage, during core damage, or at vessel breach) and the RPV pressure is less than 200 psi at the time of vessel breach (this means DCH is not possible).
3	CD, VB, Early CF, DW Failure, RPV Pressure > 200 psi at VB Core damage occurs followed by vessel breach. The containment fails early in the drywell (i.e., either before core damage, during core damage, or at vessel breach) and the RPV pressure is greater than 200 psi at the time of vessel breach (this means DCH is possible).
4	CD, VB, Early CF, DW Failure, RPV Pressure < 200 psi at VB Core Damage occurs followed by vessel breach. The containment fails early in the drywell (i.e., either before core damage, during core damage, or at vessel breach) and the RPV pressure is less than 200 psi at the time of vessel breach (this means DCH is not possible).
5	CD, VB, Late CF, WW Failure, N/A Core Damage occurs followed by vessel breach. The containment fails late in the wetwell (i.e., after vessel breach during Molten Core-Concrete Interaction (MCCI)) and the RPV pressure is not important since, even if DCH occurred, it did not fail containment at the time it occurred.
6	CD, VB, Late CF, DW Failure, N/A Core Damage occurs followed by vessel breach. The containment fails late in the drywell (i.e., after vessel breach during MCCI) and the RPV pressure is not important since, even if DCH occurred, it did not fail containment at the time it occurred.
7	CD, VB, No CF, Vent, N/A Core Damage occurs followed by vessel breach. The containment never structurally fails, but is vented sometime during the accident progression. RPV pressure is not important (characteristic 5 is N/A) since, even if it occurred, DCH does not significantly affect the source term as the containment does not fail and the vent limits its effect.

Table 4.2-2
Collapsed Accident Progression Bin (APB) Descriptions [9]

Collapsed APB Number	Description
8	CD, VB, No CF, N/A, N/A Core damage occurs followed by vessel breach. The containment never fails structurally (characteristic 4 is N/A) and is not vented. RPV pressure is not important (characteristic 5 is N/A) since, even if it occurred, DCH did not fail containment. Some nominal leakage from the containment exists and is accounted for in the analysis so that while the risk will be small it is not completely negligible.
9	CD, No VB, N/A, N/A, N/A Core damage occurs but is arrested in time to prevent vessel breach. There are no releases associated with vessel breach or MCCI. It must be remembered, however, that the containment can fail due to overpressure or venting even if vessel breach is averted. Thus, the potential exists for some of the in-vessel releases to be released to the environment.
10	No CD, N/A, N/A, N/A, N/A Core damage did not occur. No in-vessel or ex-vessel release occurs. The containment may fail on overpressure or be vented. The RPV may be at high or low pressure depending on the progression characteristics. The risk associated with this bin is negligible.

Table 4.2-3
Calculation of PBAPS Population Dose Risk at 50 Miles

Collapsed Bin #	Fractional APB Contributions to Risk (MFCR) (1)	NUREG/CR-4551 Population Dose Risk at 50 miles (From a total of 7.9 person-rem/yr, mean) (2)	NUREG/CR-4551 Collapsed Bin Frequencies (per year) (3)	NUREG/CR-4551 Population Dose at 50 miles (Person-rem) (4)
1	0.021	0.1659	9.55E-08	1.74E+06
2	0.0066	0.05214	4.77E-08	1.09E+06
3	0.556	4.3924	1.48E-06	2.97E+06
4	0.226	1.7854	7.94E-07	2.25E+06
5	0.0022	0.01738	1.30E-08	1.34E+06
6	0.059	0.4661	2.04E-07	2.28E+06
7	0.118	0.9322	4.77E-07	1.95E+06
8	0.0005	0.00395	7.99E-07	4.94E+03
9	0.01	0.079	3.86E-07	2.05E+05
10	0	0	4.34E-08	0
Totals	**1.0**	**7.9**	**4.34E-6**	

(1) Mean Fractional Contribution to Risk from Table 5.2-3 of NUREG/CR-4551

(2) The total population dose risk at 50 miles from internal events in person-rem is provided in Table 5.1-1 of NUREG/CR-4551. The contribution for a given APB is the product of the total PDR50 and the fractional APB contribution.

(3) NUREG/CR-4551 provides the conditional probabilities of the collapsed APBs in Figure 2.5-6. These conditional probabilities are multiplied by the total internal CDF to calculate the collapsed APB frequency.

(4) Obtained from dividing the population dose risk shown in the third column of this table by the collapsed bin frequency shown in the fourth column of this table.

Population Estimate Methodology

The person-rem results in Table 4.2-3 can be used as an approximation of the dose for Duane Arnold if it is corrected for the population surrounding Duane Arnold and the difference in Technical Specifications leak rate. For the updated population estimate, data is available for population by county from the US Census Bureau on the Iowa State University web site (http://www.soc.iastate.edu/census/counties.html). This data is used to estimate the population within a 50-mile radius of the plant. If the entire county falls within the 50-mile radius based on a review of an atlas containing a mileage scale and

county borders, then the entire population can be included in the population estimate. Otherwise, a fraction of the population is counted based on the percentage of the county within the 50-mile radius. The land area within the 50-mile radius is estimated based on visual inspection of the map and the population of that area is estimated assuming uniform distribution of the population within the county. The results of this updated population estimate are presented in Table 4.2-4.

Table 4.2-4
Population Within 50 Miles of DAEC (2000 US Census)

COUNTY NAME	County Population		Population Within 50 Miles of DAEC
	Total	Percent Within 50 Miles of DAEC	
Benton	25308	100%	25308
Black Hawk	128012	100%	128012
Bremer	23325	25%	5831
Buchanan	21093	100%	21093
Cedar	18187	90%	16368
Clayton	18678	20%	3736
Clinton	50149	10%	5015
Delaware	18404	100%	18404
Dubuque	89143	40%	35657
Fayette	22008	30%	6602
Grundy	12369	25%	3092
Iowa	15671	100%	15671
Jackson	20296	15%	3044
Johnson	111006	100%	111006
Jones	20221	100%	20221
Keokuk	11400	25%	2850
Linn	191701	100%	191701
Muscatine	41722	25%	10431
Poweshiek	18815	60%	11289
Tama	18103	100%	18103
Washington	20670	40%	8268
		Total =	**661703**

The population data shown above in Table 4.2-4 is compared to the population data that is provided in NUREG/CR-4551 in order to get a "Population Dose Factor" that can be applied to the APBs to get dose estimates for Duane Arnold.

Total DAEC Population = 6.62E+05

PBAPS Population from NUREG/CR-4551 = 3.02E+06 [15]

Population Dose Factor = 6.62E+05 / 3.02E+06 = 0.21

This population dose factor then can be applied to the APBs from NUREG/CR-4551. Additionally, a second correction factor is also required to be applied to the NUREG/CR-4551 calculation to account for differences in the Technical Specification leakage value for Accident Progression Bin 8. The Technical Specification containment available leak rate for Duane Arnold is 2.0% (L_a^{DA}) versus the 0.5% (L_a^{PB}) for the NUREG-1150 plant, Peach Bottom. Therefore, the leakage (L_a^{PB}) person-rem calculated for Peach Bottom that is scaled by population for the Duane Arnold analysis must be multiplied by a factor of 4.0 (= L_a^{DA} / L_a^{PB}) to account for the differences in Technical Specification leakage rates.

Table 4.2-5 shows the results of applying the population dose factor and the allowable leakage factor to the NUREG/CR-4551 population dose results at 50 miles to obtain the adjusted population dose at 50 miles for Duane Arnold.

Table 4.2-5
Calculation of Duane Arnold Population Dose Risk at 50 Miles

Collapsed Bin #	NUREG/CR-4551 Population Dose at 50 miles (Person-rem)	Bin Mulitplier used to obtain DAEC Population Dose	DAEC Adjusted Population Dose at 50 miles (Person-rem)
1	1.74E+06	0.21	3.65E+05
2	1.09E+06	0.21	2.29E+05
3	2.97E+06	0.21	6.24E+05
4	2.25E+06	0.21	4.73E+05
5	1.34E+06	0.21	2.81E+05
6	2.28E+06	0.21	4.79E+05
7	1.95E+06	0.21	4.10E+05
8	4.94E+03	4 x 0.21	4.15E+03
9	2.05E+05	0.21	4.31E+04
10	0	0	0

Application of Duane Arnold PSA Model Results to NUREG/CR-4551 Level 3 Output

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

The basic process that was pursued to obtain Level 3 results based on the Duane Arnold PSA Level 2 model and NUREG/CR-4551 was to define a useful relationship between the Level 2 and Level 3 results. Consequently, each sequence of the Duane

Arnold PSA Level 2 model was reviewed and assigned into one of the collapsed Accident Progression Bins (APBs) from NUREG/CR-4551. The Level 2 model contains a significantly larger amount of information about the accident sequences than what is used in the collapsed APBs in NUREG/CR-4551 and this assignment process required simplification of accident progression information and assumptions related to categorizations of certain items. The assumptions used for these assignments are shown in Table 4.2-6.

Table 4.2-6
Duane Arnold Level 2 Model Nodal Assumptions for Application to the NUREG/CR-4551 Accident Progression Bins

DAEC PSA Containment Event Tree Node	Assumption
IS – Containment Isolation	If the containment is not isolated, it is assumed that it will be open for the equivalent of an un-scrubbed release as soon as the vessel is breached. No depressurization is asked prior to this node; it is assumed that RPV pressure is >= 200 psi for these sequences. This is APB #3.
OP – Operator depressurizes the RPV	It is assumed that success on this branch results in RPV pressure below 200 psi that is then used to distinguish between APB #1 versus APB #2, or APB #3 versus APB #4.
RX – Core Melt Arrested in Vessel	A success on this branch signifies that there is no vessel breach. The sequences following this path are grouped in APB #9. However, this assignment is overridden if the containment still fails due to subsequent CZ or HR-CV-MU failures. In these cases, CZ failures are assigned to APB #3 or APB #4 depending on the status of OP, and APB #5 or APB #6 is assigned for HR-CV-MU failures depending on the status of the SP node.
CX, CZ, DI, NC, SI – Containment Intact Nodes	Failure of containment is assumed to result in an un-scrubbed release. The timing is assumed to be "early" for all but loss of containment heat removal (Level 2 Accident Class 2) events and is grouped in APB #3 or APB #4 depending on RPV pressure. For the Level 2 Accident Class 2 events, the timing is assumed to be "late" and is grouped in APB #5 or #6 depending on whether the suppression pool is not bypassed in the SP node.
FD – Flooding Completed	If containment flooding is initiated and successfully completed without other containment failures, this is assigned to APB #7 based on the interpretation that the successful completion of flooding requires RPV venting. RPV venting is assumed to result in a release characteristic similar to the venting scenarios from APB #7.

DAEC PSA Containment Event Tree Node	Assumption
CV, GV – Containment Venting Nodes	Success of these nodes is used to indicate assignment to APB #7 for venting as long as the suppression pool is not bypassed in the SP node, and other containment failure nodes are not failed. This assignment applies to sequences with RX failures.
SP – Suppression Pool Not Bypassed	The suppression pool bypass node is considered in the DAEC CETs to determine whether the vent volume passes through the suppression pool or not. This node is used to distinguish between a WW or DW failure as described in the other node assumption descriptions above.
RB – Release Mitigated in Reactor Building	The RB node, release mitigated in reactor building, is not credited as a scrubbing mechanism. The only scrubbing accounted for in the collapsed bins is distinguished by indicating a WW release (with the success of the SP node) and the amount of scrubbing that the reactor building is capable of providing is not considered to be the equivalent a WW scrub. This may be slightly conservative because the DAEC Level 2 PSA provides some credit for the RB node as a scrubbing mechanism.

Release Category Definitions

Table 4.2-7 defines the accident classes used in the ILRT extension evaluation consistent with the EPRI methodology [2].

Table 4.2-7
EPRI CONTAINMENT FAILURE CLASSIFICATIONS

Class	Description
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 (as reported in the IPEs) 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.

These containment failure classifications are used in this analysis to determine the risk impact of extending the Containment Type A test interval as described in Section 5 of this report.

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 effort is properly accounted for, the Class 3 Accident Class as defined in Table 4.2-7 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 [16]. The χ^2 distribution is actually a family of distributions, which range in shape from that of the exponential distribution 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.

It should be noted that 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 due to the longer intervals between tests. The change in probability was estimated by

(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.

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)]

In addition, Indian Point 3 (IP3) [6] has used these same estimates 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) [7].

The analysis included in this report follows the precedence set by the EPRI report and the IP3 analysis. The use of the failure rate model is represented by the assumed "failure to detect" probabilities used by EPRI and in the IP3 submittal. That is, the extension of the ILRT interval from 3-in-10 to 1-in-10 years leads to a 10% increase in the probability of an undetected leak, and the extension from 3-in-10 to 1-in-15 years leads to a 15% increase in the probability of an undetected leak.

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(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.

Section 5

RESULTS

The application of the approach based on EPRI-TR-104285 [2] and previous risk assessment submittals on this subject [6] have led to the following 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 Duane Arnold 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 bypassed (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 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 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 Duane Arnold Level 2 PSA model is used in the ILRT evaluation [14].

This step involves the review of the Duane Arnold containment event trees (CETs) and Level 2 accident sequence frequency results. The CETs characterize the response of the containment to important severe accident sequences. As described in Section 4.2, the Duane Arnold CETs were examined and each endstate was applied to one of the Accident Progression Bins as defined in NUREG/CR-4551. This application forms the basis for estimating the population dose for Duane Arnold.

As previously described, the extension of the Type A 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. As a result, the CET containment isolation model was reviewed for applicable isolation failures and their impact on the overall plant risk. The containment isolation model for Duane Arnold 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 5.083E-3/demand. See cutsets from Attachment B.

For the assessment of ILRT impacts on the risk profile, the potential for pre-existing leaks are included in the model. (These events are represented by the "Class 3" sequence depicted in EPRI TR-104285 [2]). The question on containment integrity was modified to include the probability of a liner breach or bellows failure (due to excessive leakage) at the time of core damage. Two additional failure modes were considered in addition to large containment failure modes. 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, the eight severe accidents class frequencies were developed consistent with the definitions in Table 5-1 as 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 1.61E-6/year and is determined by subtracting all containment failure end states from the total CDF. 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 (>2-inch diameter) containment isolation valves (Attachments A and B). The frequency per year for these sequences is 4.33E-8/yr and is determined from the sum of all Level 2 end states involving containment isolation failure from the base model results.

Note that the frequency per year for the EPRI Class 2 sequences is slightly less than that which would be obtained by multiplying the independent containment failure probability by the total core damage frequency.

(5.083E-3 x 1.18E-5/yr = 6.00E-8/yr)

The difference is due to the fact that some of the Level 1 core damage sequences assume that containment failure occurs prior to vessel failure (e.g. loss of containment heat removal sequences) such that the failed containment makes the questioning of isolation failure unnecessary. This difference is judged to be appropriate, and in any event, will not impact the results or conclusions from this analysis. These failures are

assumed to result in a LERF that is characterized as a containment bypass, i.e., the same as Class 8.

<u>Class 3 Sequences</u>. 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 * 1.18E-5/year = 7.55E-7/year
CLASS_3B_FREQUENCY = 0.021 * 1.18E-5/year = 2.48E-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 calculations [6].

<u>Class 4 Sequences</u>. This group consists of all core damage accident progression bins for which a containment isolation failure-to-seal of Type B test components occurs. Becaαse 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.

<u>Class 5 Sequences</u>. 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.

<u>Class 6 Sequences</u>. 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-1273 [5]

= 4E-4

$$CLASS\_6\_FREQUENCY = 4E\text{-}4 * 1.18E\text{-}5/year = 4.72E\text{-}9/year$$

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 induced by severe accident phenomena occurs (e.g., Mark I shell melt-though, overpressure). For this analysis, the associated radionuclide releases are based on the application of the Level 2 endstates to the Accident Progression Bins from NUREG/CR-4551 as described in Section 4.2. The Class 7 Sequences are divided into 8 categories which can be mapped directly to Bins 1-7, and 9 from NUREG/CR-4551. The failure frequency and population dose for each specific APB is shown below in Table 5-2. The total release frequency and total dose are then used to determine a weighted average person-rem for use as the representative EPRI Class 7 dose in the subsequent analysis. Note that the total frequency and dose associated from this EPRI class does not change as part of the ILRT extension request.

Table 5-2
ACCIDENT CLASS 7 FAILURE FREQUENCIES AND POPULATION DOSES (DAEC BASE CASE LEVEL 2 MODEL)

Accident Class (APB umber)	Release Frequency/yr	Population Dose (50 miles) Person-Rem [1]	Population Dose Risk (50 Miles) (Person-Rem/yr) [2]
7a (APB #1)	1.76E-09	3.65E+05	6.43E-04
7b (APB #2)	1.10E-07	2.29E+05	2.52E-02
7c (APB #3)	5.23E-07	6.24E+05	3.26E-01
7d (APB #4)	5.30E-06	4.73E+05	2.50E+00
7e (APB #5)	2.02E-07	2.81E+05	5.69E-02
7f (APB #6)	2.97E-07	4.79E+05	1.42E-01
7g (APB #7)	1.15E-06	4.10E+05	4.73E-01
7h (APB #9)	1.46E-06	4.31E+04	6.28E-02
Class 7 Total	**9.05E-06**	**3.97E+05** [3]	**3.59E+00**

(1) Population dose values obtained from Table 4.2-5 based on the Accident Progression Bin.
(2) Obtained by multiplying the Release Frequency value from the second column of this table by the Population dose value from the third column of this table.
(3) The weighted average population dose for Class 7 is obtained by dividing the total population dose risk by the total release frequency.

Class 8 Sequences. This group consists of all core damage accident progression bins in which containment bypass occurs. The containment bypass failure frequency from the base model Level 2 results is 8.97E-8/yr

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 5-3 summarizes these accident frequencies by Accident Class.

Table 5-3
RADIONUCLIDE RELEASE FREQUENCIES AS A FUNCTION OF ACCIDENT CLASS (DAEC BASE CASE)

Accident Classes (Containment Release Type)	Description	Frequency (per Rx-yr)
1	No Containment Failure (Including Successful Venting)	1.61E-06
2	Large Isolation Failures (Failure to Close)	4.33E-08
3a	Small Isolation Failures (liner breach)	7.55E-07
3b	Large Isolation Failures (liner breach)	2.48E-07
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)	4.72E-09
7	Failures Induced by Phenomena (Early and Late)	9.05E-06
8	Bypass (Interfacing System LOCA)	8.97E-08
CDF	**All CET End states (including very low and no release)**	**1.18E-05**

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

Plant-specific release analyses were performed to estimate the person-rem doses to the population within a 50-mile radius from the plant. The releases are based on information provided by NUREG/CR-4551 with adjustments made for the site demographic differences and allowable leakage compared to the reference plant as described in Section 4.2, and summarized in Table 4.2-5. The results of applying these releases to the EPRI containment failure classification are shown below.

Class 1	=	4150 person-rem (at $1.0L_a$)	=	4150 person-rem (1)
Class 2	=	6.24E+5 person-rem (2)		
Class 3a	=	4150 person-rem x $10L_a$	=	4.15E+4 person-rem (3)
Class 3b	=	4150 person-rem x $35L_a$	=	1.45E+5 person-rem (3)
Class 4	=	Not analyzed		
Class 5	=	Not analyzed		
Class 6	=	6.24E+5 person-rem		
Class 7	=	3.97E+5 person-rem (4)		
Class 8	=	6.24E+5 person-rem (5)		

(1) The population dose associated with the Technical Specification Leakage is based on scaling both the population data and allowable Technical Specification leakage compared to the NUREG/CR-4551 reference plant. The derivation is described in Section 4.2 for Duane Arnold.

(2) Class 2 (Containment Isolation failures) may be drywell isolation failures.

(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 [6].

(4) This is the weighted average person-rem for Class 7 as derived in Table 5-2.

(5) 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.

The population dose estimates derived for use in the risk evaluation per the EPRI methodology [2] containment failure classification are summarized in Table 5-4.

Table 5-4
DUANE ARNOLD POPULATION DOSE ESTIMATES FOR POPULATION WITHIN 50 MILES

Accident Classes (Containment Release Type)	Description	Person-Rem (50 miles)
1	No Containment Failure	4.15E+03
2	Large Isolation Failures (Failure to Close)	6.24E+05
3a	Small Isolation Failures (liner breach)	4.15E+04
3b	Large Isolation Failures (liner breach)	1.45E+05
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)	6.24E+05
7	Failures Induced by Phenomena (Early and Late)	3.97E+05
8	Bypass (Interfacing System LOCA)	6.24E+05

The above results when combined with the results presented in Table 5-3 yield the Duane Arnold baseline mean consequence measures for each accident class. These results are presented in Table 5-5 below:

Table 5-5
ANNUAL DOSE (PERSON-REM/YR) 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 [(1)]	1.61E-06	4.15E+03	6.68E-03
2	Large Isolation Failures (Failure to Close)	4.33E-08	6.24E+05	2.70E-02
3a	Small Isolation Failures (liner breach)	7.55E-07	4.15E+04	3.13E-02
3b	Large Isolation Failures (liner breach)	2.48E-07	1.45E+05	3.60E-02
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)	4.72E-09	6.24E+05	2.94E-03
7	Failures Induced by Phenomena (Early and Late)	9.05E-06	3.97E+05	3.59E+00
8	Bypass (Interfacing System LOCA)	8.97E-08	6.24E+05	5.59E-02
CDF	**All CET End states (including very low and no release)**	**1.18E-05**		**3.752**

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(1) 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 is relatively low compared with the other sites as shown below:

Plant	Annual Dose (Person-Rem/Yr)	Reference
Indian Point 3	14,515	[6]
Peach Bottom	6.2	[15]
Crystal River	1.4	[17]
Duane Arnold	3.75	[Table 5-5]

Based on the risk values from Table 5-5, the percent risk contribution ($\%Risk_{BASE}$) for Class 3 is as follows:

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

Where:

$CLASS3a_{BASE}$ = Class 3a person-rem/year = 3.13E-2 person-rem/year [Table 5-5]

$CLASS3b_{BASE}$ = Class 3b person-rem/year = 3.60E-2 person-rem/year [Table 5-5]

$TOTAL_{BASE}$ = Total person-rem/yr for baseline interval = 3.752 person-rem/yr [Table 5-5]

$$\%Risk_{BASE} = [(3.13E\text{-}2 + 3.60E\text{-}2) / 3.752] \times 100 = (6.73E\text{-}2) / 3.752$$

$$\%Risk_{BASE} = 1.79\%$$

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 * 12). 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 [i.e., (½) * (3% / 1.5 years) * 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., (½) * (3% / 1.5 years) * 15 years]. Thus increasing the ILRT test interval from 10 years to 15 years results in an additional 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 by a factor of 1.1 (which is consistent with the approach used by Indian Point 3 [6]). The results of this calculation are presented in Table 5-6.

Based on the Table 5-6 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 = 3.45E-2 person-rem/year [Table 5-6]

$CLASS3b_{10}$ = Class 3b person-rem/year = 3.96E-2 person-rem/year [Table 5-6]

Table 5-6
ANNUAL DOSE (PERSON-REM/YR) AS A FUNCTION OF ACCIDENT CLASS CHARACTERISTIC OF CONDITIONS FOR ILRT REQUIRED EVERY 10 YEARS

Accident Classes (Containment Release Type)	Description	Frequency (per Rx-yr)	Person-Rem (50 miles)	Person-Rem/yr (50 miles)
1	No Containment Failure(1)	1.51E-06	4.15E+03	6.26E-03
2	Large Isolation Failures (Failure to Close)	4.33E-08	6.24E+05	2.70E-02
3a	Small Isolation Failures (liner breach) (2)	8.31E-07	4.15E+04	3.45E-02
3b	Large Isolation Failures (liner breach) (2)	2.73E-07	1.45E+05	3.96E-02
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)	4.72E-09	6.24E+05	2.94E-03
7	Failures Induced by Phenomena (Early and Late)	9.05E-06	3.97E+05	3.59E+00
8	Bypass (Interfacing System LOCA)	8.97E-08	6.24E+05	5.59E-02
CDF	**All CET End states (including very low and no release)**	**1.18E-05**		**3.758**

(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 [6].

$TOTAL_{10}$ = Total person-rem/yr for 10-year interval = 3.758 person-rem/yr [Table 5-6]

$\%Risk_{10}$ = [(3.45E-2 + 3.96E-2) / 3.758] x 100 = (7.41E-2) / 3.758 x 100

$\%Risk_{10}$ = 1.97%

Therefore, the Total Type A 10-year ILRT interval risk contribution of leakage, represented by Class 3a and Class 3b accident scenarios is 1.97%.

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

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

$TOTAL_{BASE}$ = Total person-rem/yr for baseline interval = 3.752 person-rem/yr [Table 5-5]

$TOTAL_{10}$ = Total person-rem/yr for 10 yr ILRT interval = 3.758 person-rem/yr [Table 5-6]

$\Delta\%Risk_{10}$ = [(3.758 - 3.752) / 3.752] x 100.0

$\Delta\%Risk_{10}$ = 0.16%

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 0.16%.

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 [6,7]. The results for this calculation are presented in Table 5-7.

Table 5-7
ANNUAL DOSE (PERSON-REM/YR) AS A FUNCTION OF ACCIDENT CLASS CHARACTERISTIC OF CONDITIONS FOR ILRT REQUIRED EVERY 15 YEARS

Accident Classes (Containment Release Type)	Description	Frequency (per Rx-yr)	Person-Rem (50 miles)	Person-Rem/yr (50 miles)
1	No Containment Failure[1]	1.46E-06	4.15E+03	6.06E-03
2	Large Isolation Failures (Failure to Close)	4.33E-08	6.24E+05	2.70E-02
3a	Small Isolation Failures (liner breach)[2]	8.68E-07	4.15E+04	3.60E-02
3b	Large Isolation Failures (liner breach)[2]	2.85E-07	1.45E+05	4.14E-02
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)	4.72E-09	6.24E+05	2.94E-03
7	Failures Induced by Phenomena (Early and Late)	9.05E-06	3.97E+05	3.59E+00
8	Bypass (Interfacing System LOCA)	8.97E-08	6.24E+05	5.59E-02
CDF	**All CET End states (including very low and no release)**	**1.18E-05**		**3.761**

(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 [6] 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.

Based on the values from Table 5-7, 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 = 3.60E-2 person-rem/year [Table 5-7]

$CLASS3b_{15}$ = Class 3b person-rem/year = 4.14E-2 person-rem/year [Table 5-7]

$TOTAL_{15}$ = Total person-rem/yr for 15-year interval = 3.761 person-rem/yr [Table 5-7]

$$\%Risk_{15} = [(3.60E\text{-}2 + 4.14E\text{-}2) / 3.761] \times 100 = (7.74E\text{-}2) / 3.761 \times 100$$

$$\%Risk_{15} = 2.06\%$$

Therefore, the Total Type A 15-year ILRT interval risk contribution of leakage, represented by Class 3a and Class 3b accident scenarios is 2.06%.

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:

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

Where:

$PER\text{-}REM_{10}$ = person-rem/year for ten year interval (for Classes 3a and 3b)
= 7.41E-2 person-rem/yr

$PER\text{-}REM_{15}$ = person-rem/year for fifteen year interval (for Classes 3a and 3b)
= 7.74E-2 person-rem/yr

$$\%Risk_{10\text{-}15} = [(7.74E\text{-}2 - 7.41E\text{-}2) / 7.41E\text{-}2)] \times 100$$

$\%Risk_{10\text{-}15} = 4.45\%$

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 4.45%.

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 = 3.758 person-rem/year [Table 5-6]

$TOTAL_{15}$ Total person-rem/year for 15-year interval = 3.761 person-rem/year [Table 5-7]

$$\%TOTAL_{10\text{-}15} = [(3.761 - 3.758) / 3.758] \times 100$$
$$\%TOTAL_{10\text{-}15} = 0.08\%$$

Therefore, the risk impact on the total integrated plant risk for these accident sequences influenced by Type A testing is only 0.08%.

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 = 3.752 person-rem/year [Table 5-5]

$TOTAL_{15}$ = Total person-rem/year for 15-year interval = 3.761 person-rem/year [Table 5-7]

$\%\Delta Risk_{BASE\text{-}15}$ = [(3.761 - 3.752) / 3.752] x 100

$\%\Delta Risk_{BASE\text{-}15}$ = 0.24%

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 0.24%.

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 2 (1.45E+5 person-rem / 6.24E+5 person-rem) or 23%. 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.

Reg. Guide 1.174 [3] 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.

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 three-year test interval, is 2.48E-7/yr [Table 5-5]; based on a ten-year test interval, it is 2.73E-7 [Table 5-6]; and, based on a fifteen-year test interval, it is 2.85E-7 [Table 5-7]. Thus, increasing the ILRT test interval from 10 to 15 years results in an additional 1.20E-8/yr increase in the overall probability of LERF due to Class 3b sequences. 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 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 3.70E-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 class that is increasing are Classes 3a and 3b, the change in CCFP can be calculated by the difference in these classes.

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

$$= 0.42\%$$

This change in CCFP of less than 1% is judged to be insignificant.

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) of leakage, represented by Class 3 accident scenarios is 1.79% where the majority of the risk 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 1.97%.

3. When the ILRT interval is 15 years, the risk contribution of leakage represented by Class 3 accident scenarios is 2.06%.

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 interval to once-per-fifteen years is 4.45%.

5. The total integrated increase in risk contribution from extending the ILRT test frequency from the current one-per-10-year interval to once-per-15 years is 0.08%.

6. The risk increase in LERF from extending the ILRT test frequency from the current once-per-10 year interval to once-per-15 years is 1.20E-8. This is determined to be very small using the acceptance guidelines of Reg. Guide 1.174.

7. The risk increase in LERF from the original 3-in-10 year test frequency, to once-per-15 years is 3.70E-8/yr. This is also found to be "very small" using the acceptance guidelines in Reg. Guide 1.174.

8. The change in the conditional containment failure frequency from the current once-per-10 year interval to once-per-15 years is 0.42%. Though no official acceptance criteria exists for this risk metric, it is also judged to be very small.

9. Other salient results are summarized in Table 5-8.

Table 5-8
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	1.79% of integrated value 6.73E-2 person-rem/yr	1.97% of integrated value 7.41E-2 person-rem/yr	2.06% of integrated value 7.74E-2 person-rem/yr
Total Integrated Risk	3.752 person-rem/year	3.758 person-rem/year	3.761 person-rem/year

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

(2) Duane Arnold IPE baseline values

(3) Type A ILRT test interval of 1-in10 years.

(4) Type A ILRT test interval of 1-in15 years.

Section 6

CONCLUSIONS

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 frequency from ten-years to fifteen years:

- Reg. Guide 1.174 [3] provides guidance for determining the risk impact of plant-specific changes to the licensing basis. Reg. Guide 1.174 defines very small changes in risk as resulting in increases of CDF below 10^{-6}/yr and increases in LERF below 10^{-7}/yr. Since the ILRT does not impact CDF, the relevant criterion is LERF. The increase in LERF resulting from a change in the Type A ILRT test interval from a once-per-ten years to a once-per-fifteen years is 1.20E-8. Guidance in Reg. Guide 1.174 defines very small changes in LERF as below 10^{-7}/yr. Therefore, increasing the ILRT interval from 10 to 15 years is considered to result in a very small change to the Duane Arnold 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 for those accident sequences influenced by Type A testing by only 0.08%. Therefore, the risk impact when compared to other severe accident risks is negligible.

<u>Risk Trade-Off</u>

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

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

Previous Assessments

The NRC in NUREG-1493 [4] 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 failure.

The findings for Duane Arnold confirm the above general findings on a plant specific basis when considering (1) the severe accidents evaluated for Duane Arnold, (2) the Duane Arnold containment failure modes, and (3) the local population surrounding Duane Arnold.

Section 7

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

[8] *Shutdown Risk Impact Assessment for Extended Containment Leakage Testing Intervals Utilizing ORAM™,* EPRI, Palo Alto, CA TR-105189, Final Report, May 1995.

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[12] *Review of Light Water Reactor Regulatory Requirements*, Pacific Northwest Laboratory, NUREG/CR-4330, PNL-5809, Vol. 2, June 1986.

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[16] Patrick D.T. O'Connor, *Practical Reliability Engineering,* John Wiley & Sons, 2nd Edition, 1985.

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

[18] Letter from L. Christiansen (DAEC) to D. Vanover (ERIN), *Information for ILRT Extension Request,* NG-01-1145, File A-351, September 25, 2001.

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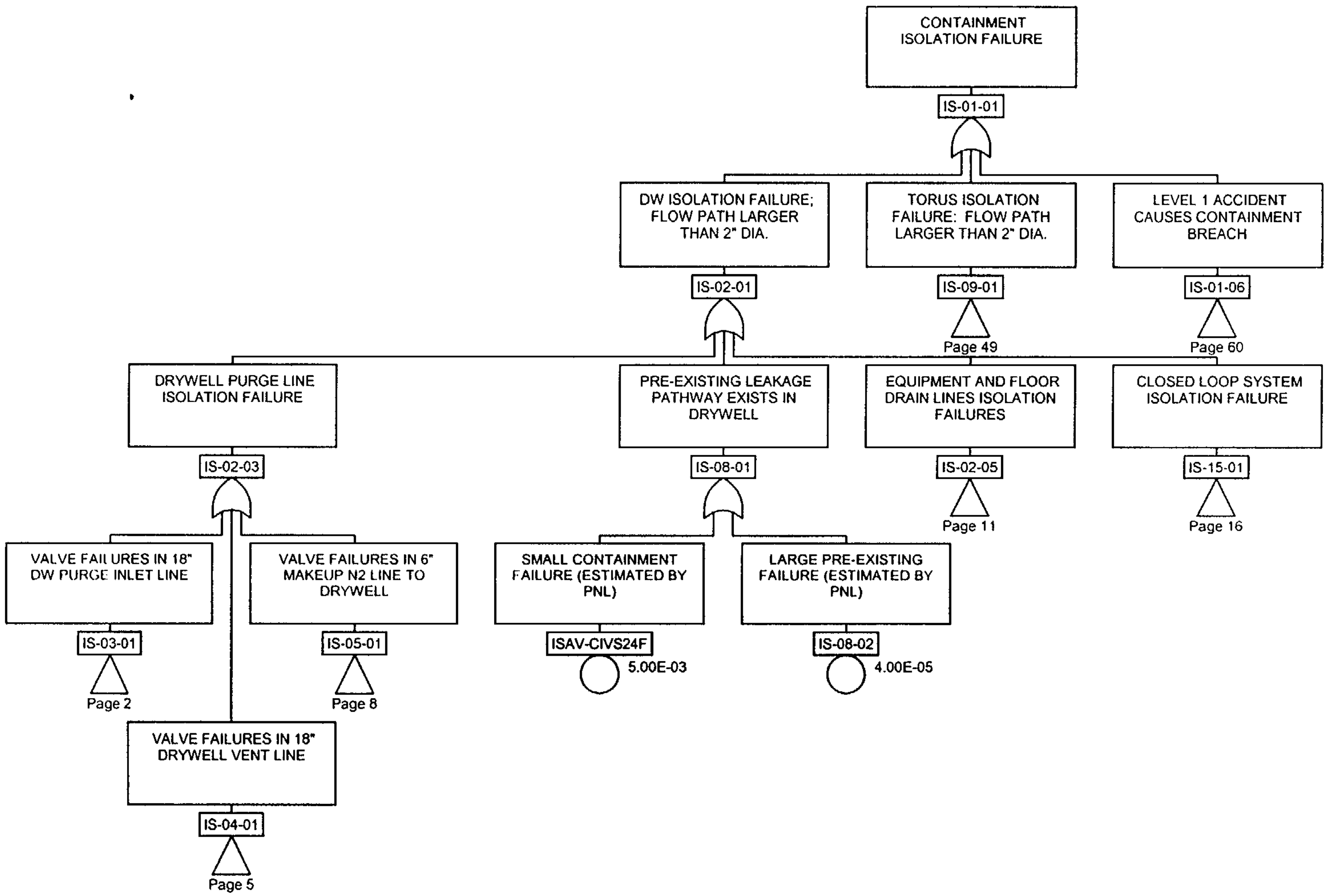
Attachment A

CONTAINMENT ISOLATION FAULT TREE



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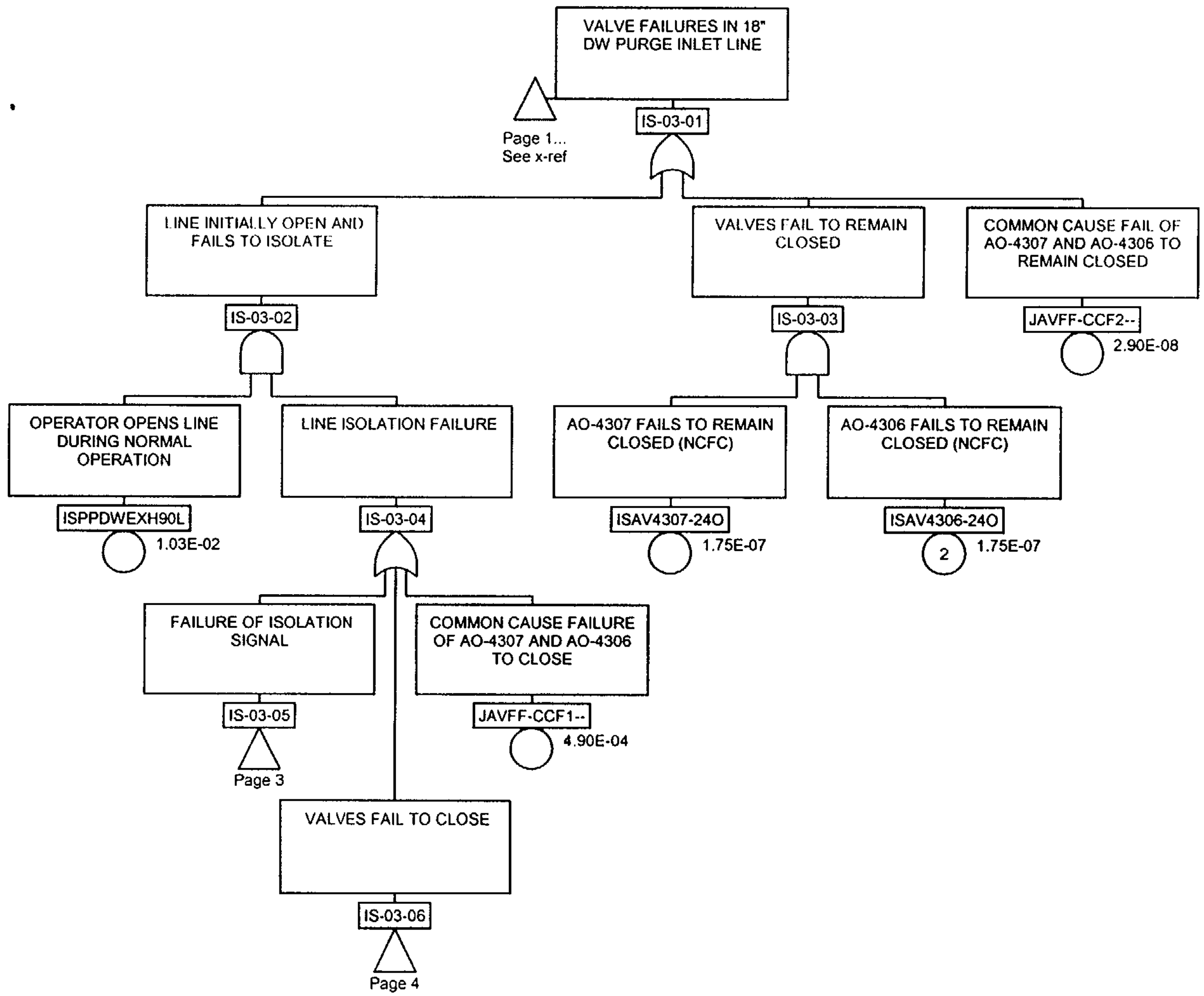

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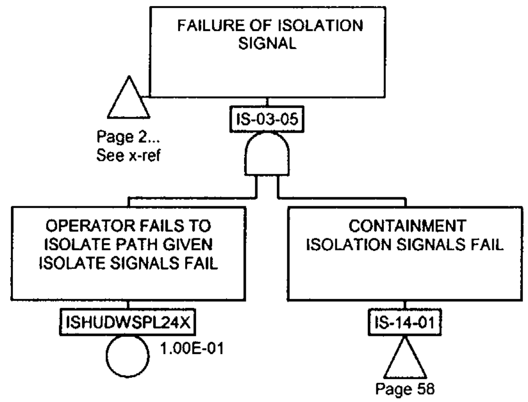

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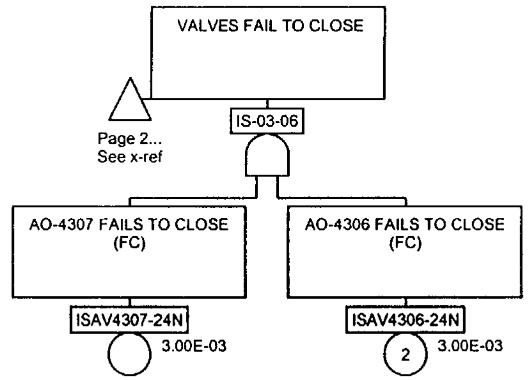

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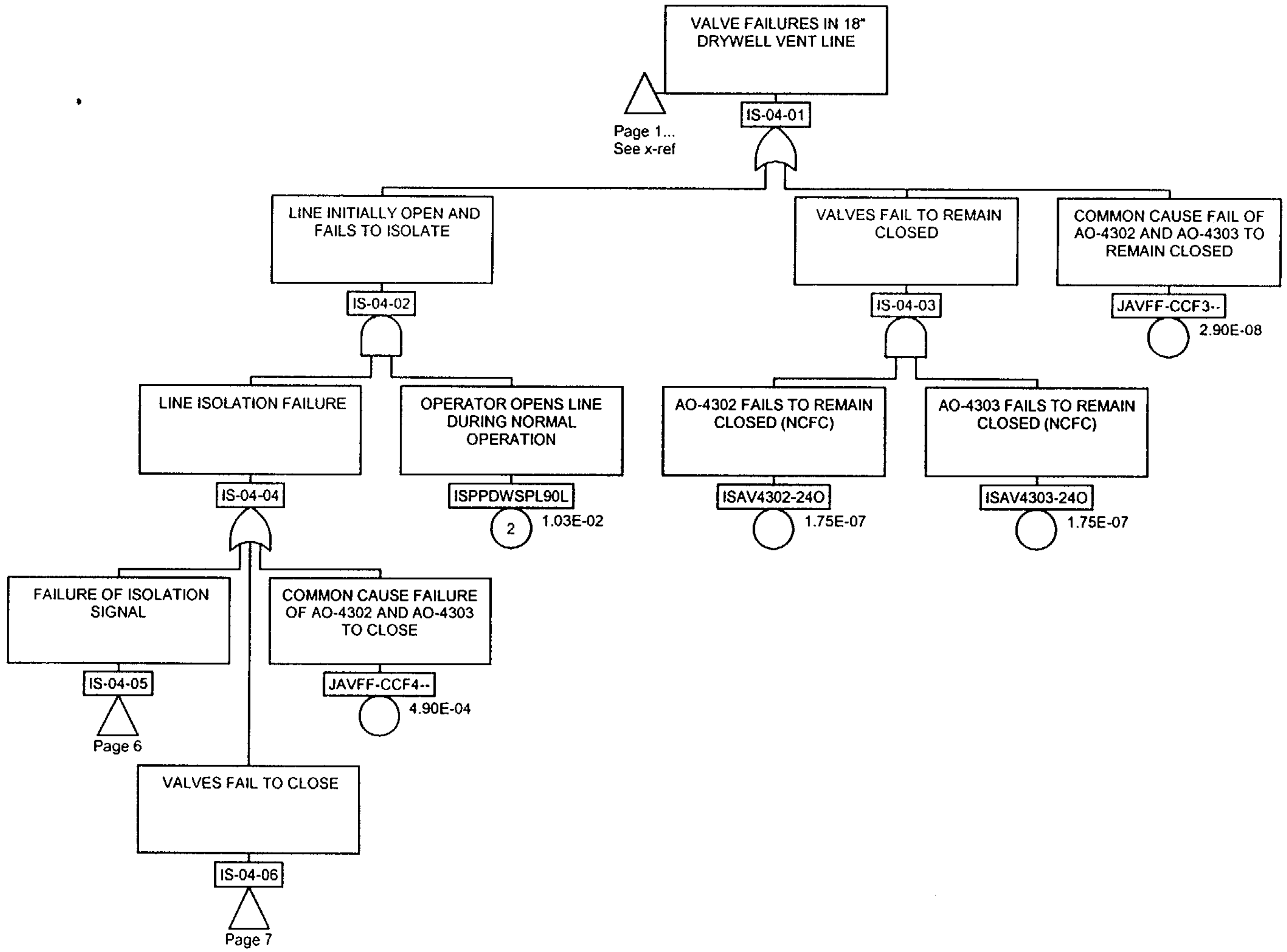

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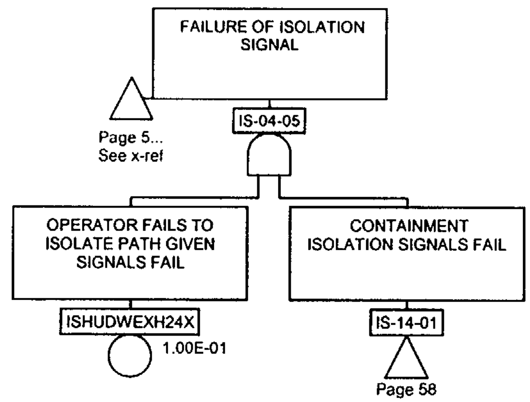

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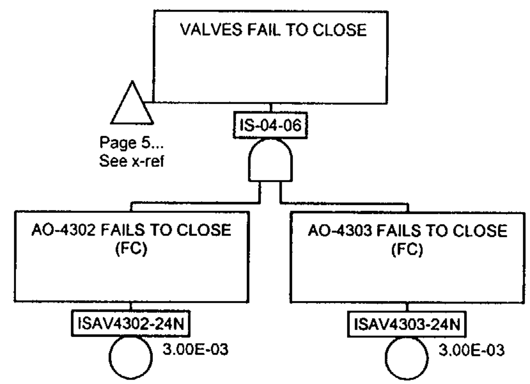

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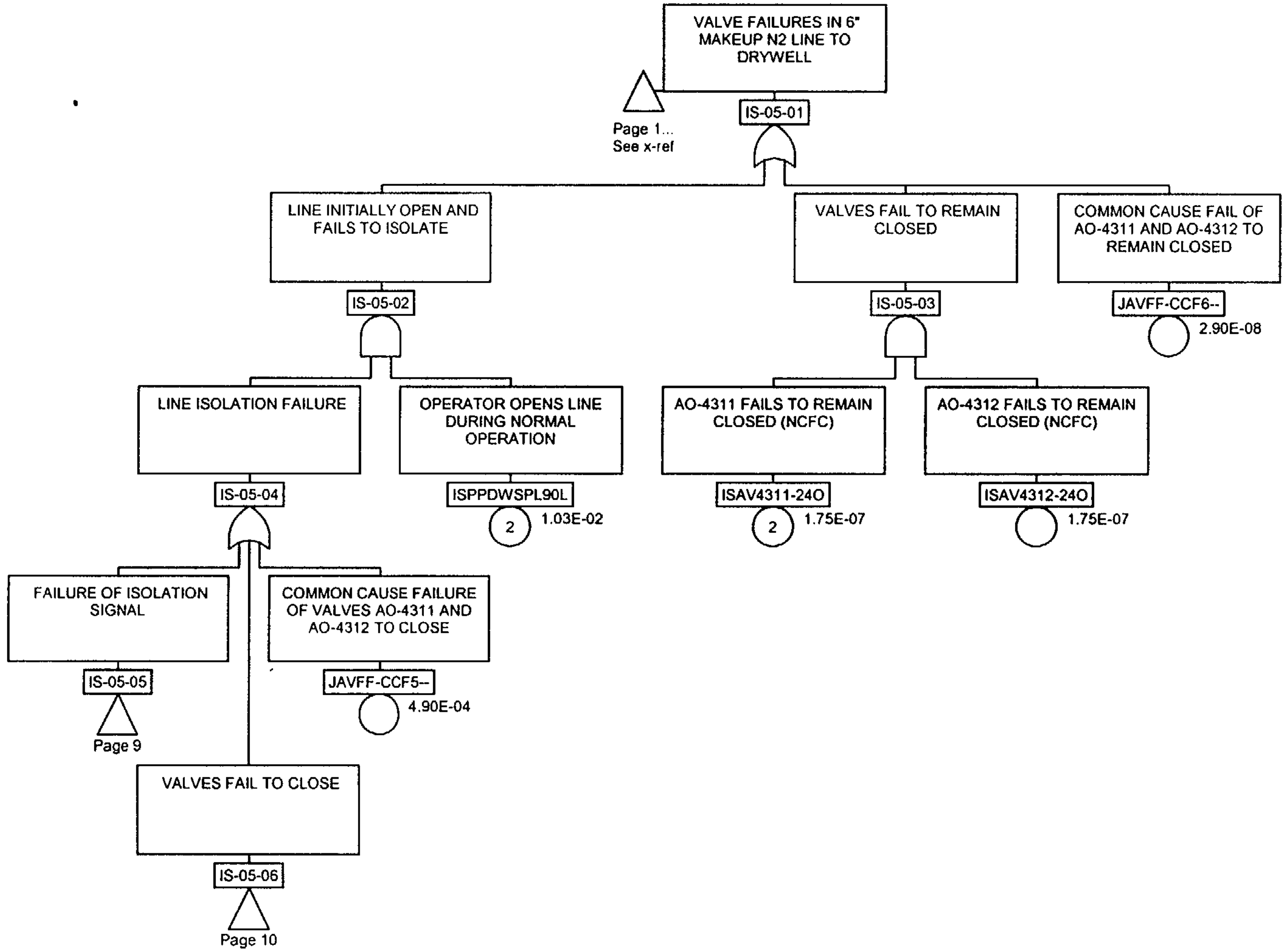

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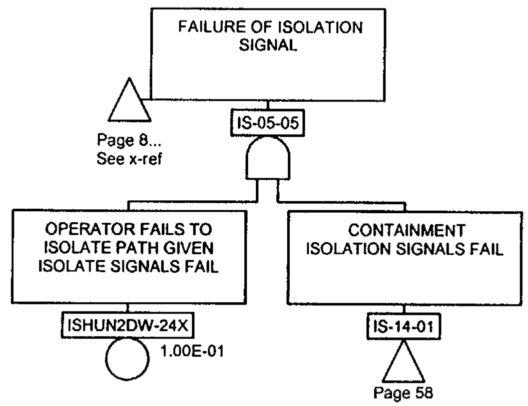

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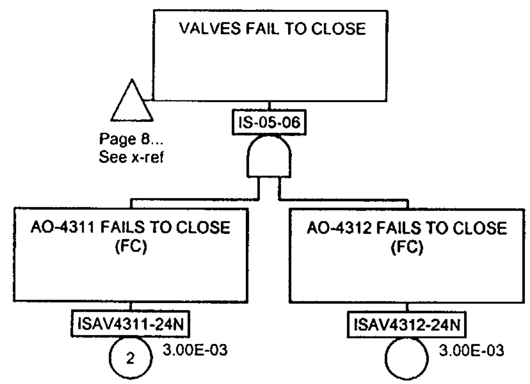

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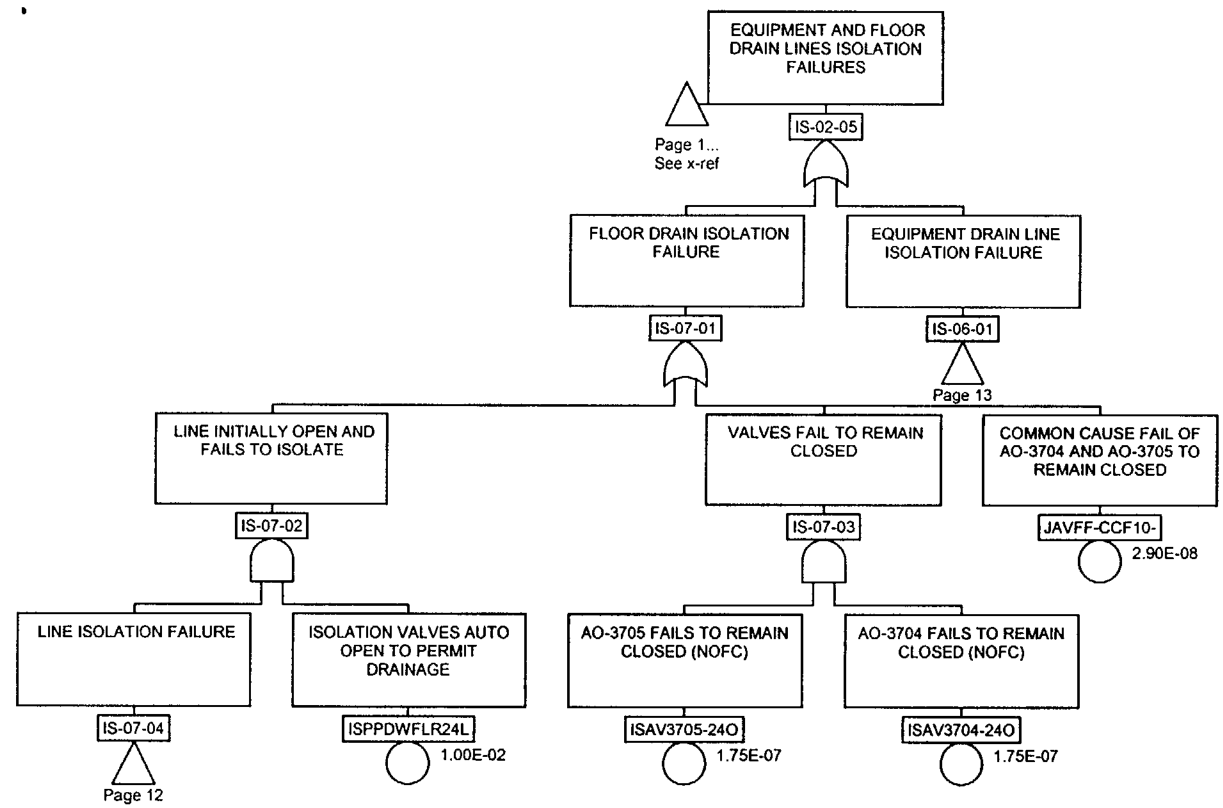

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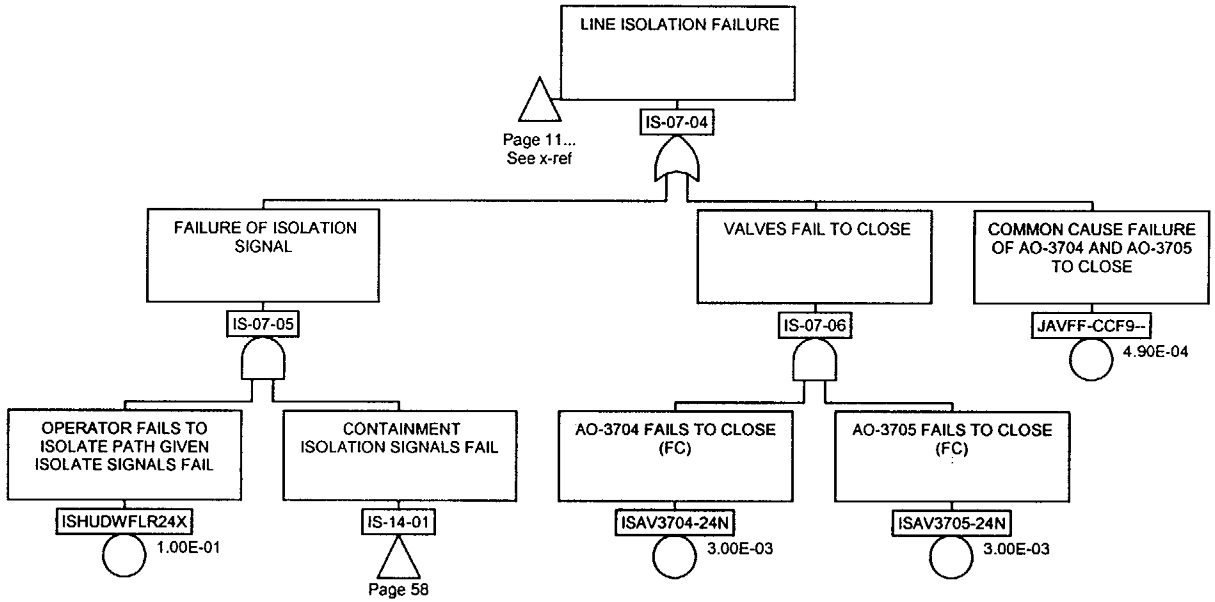

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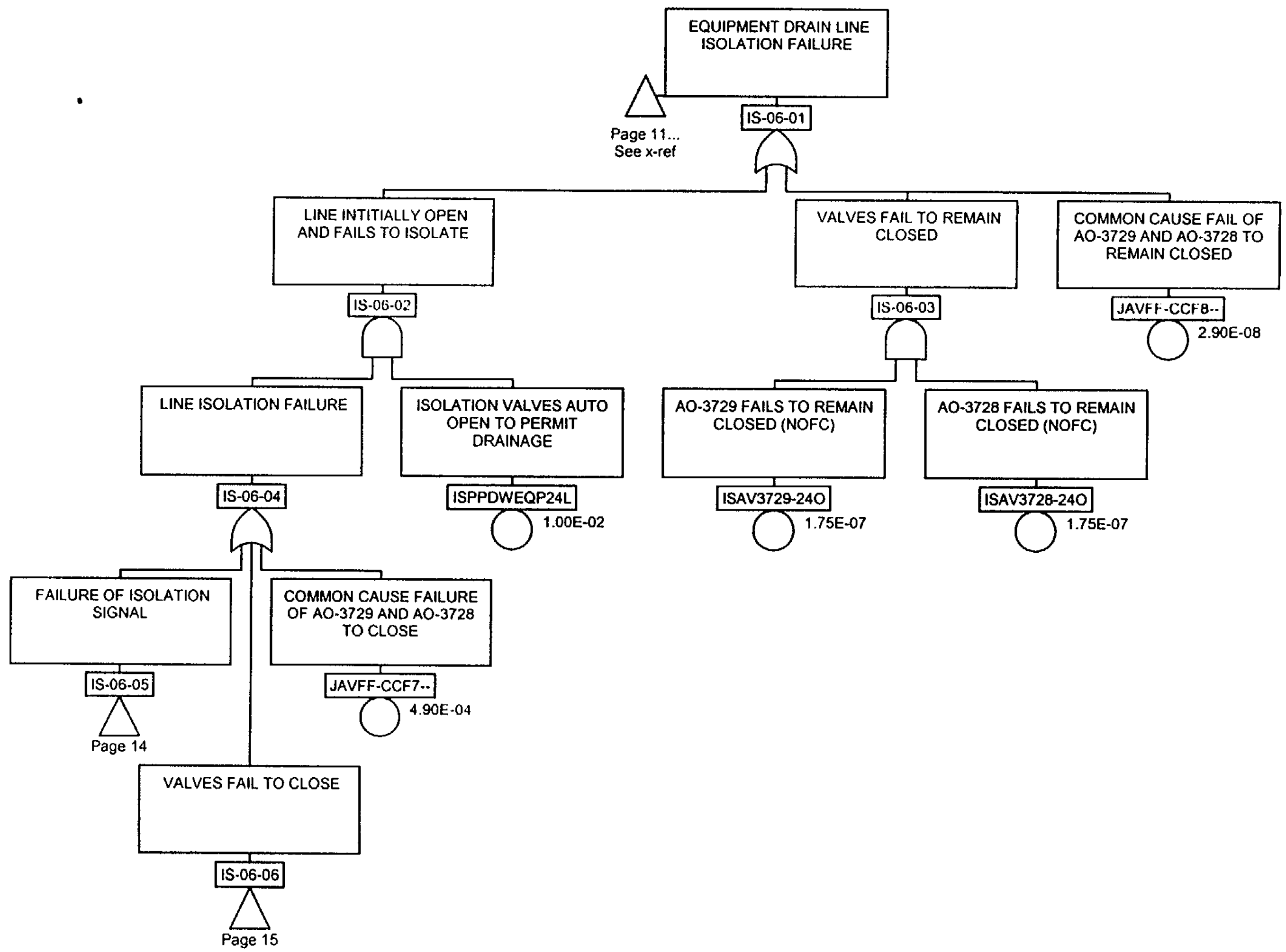

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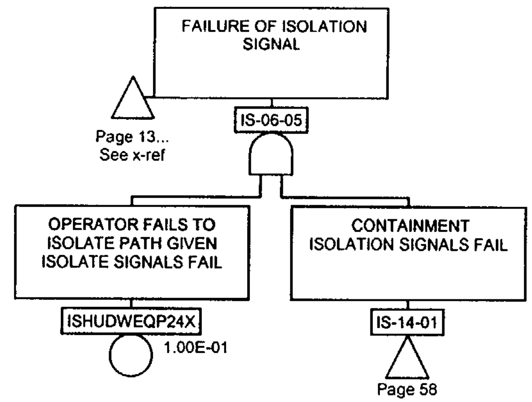

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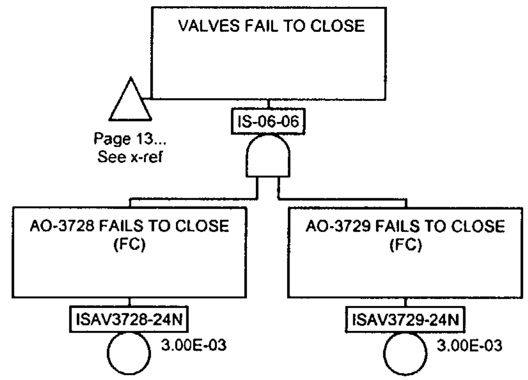

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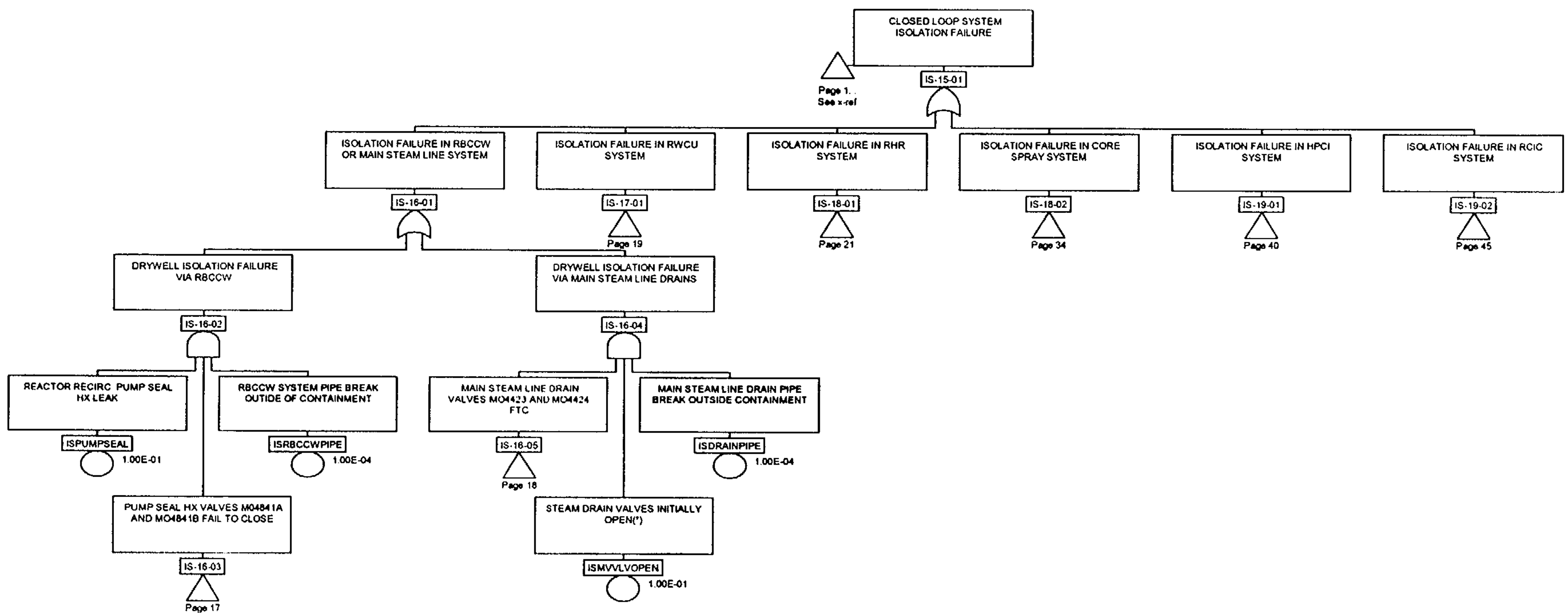

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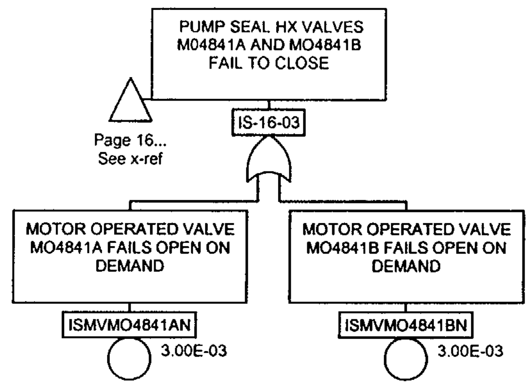

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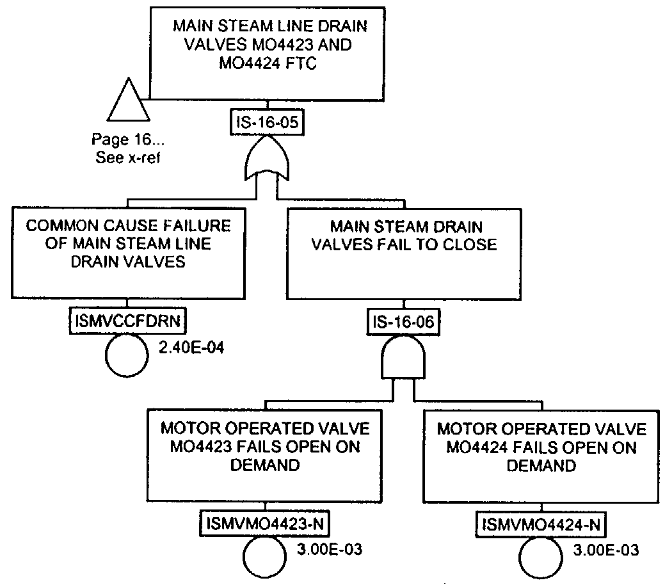

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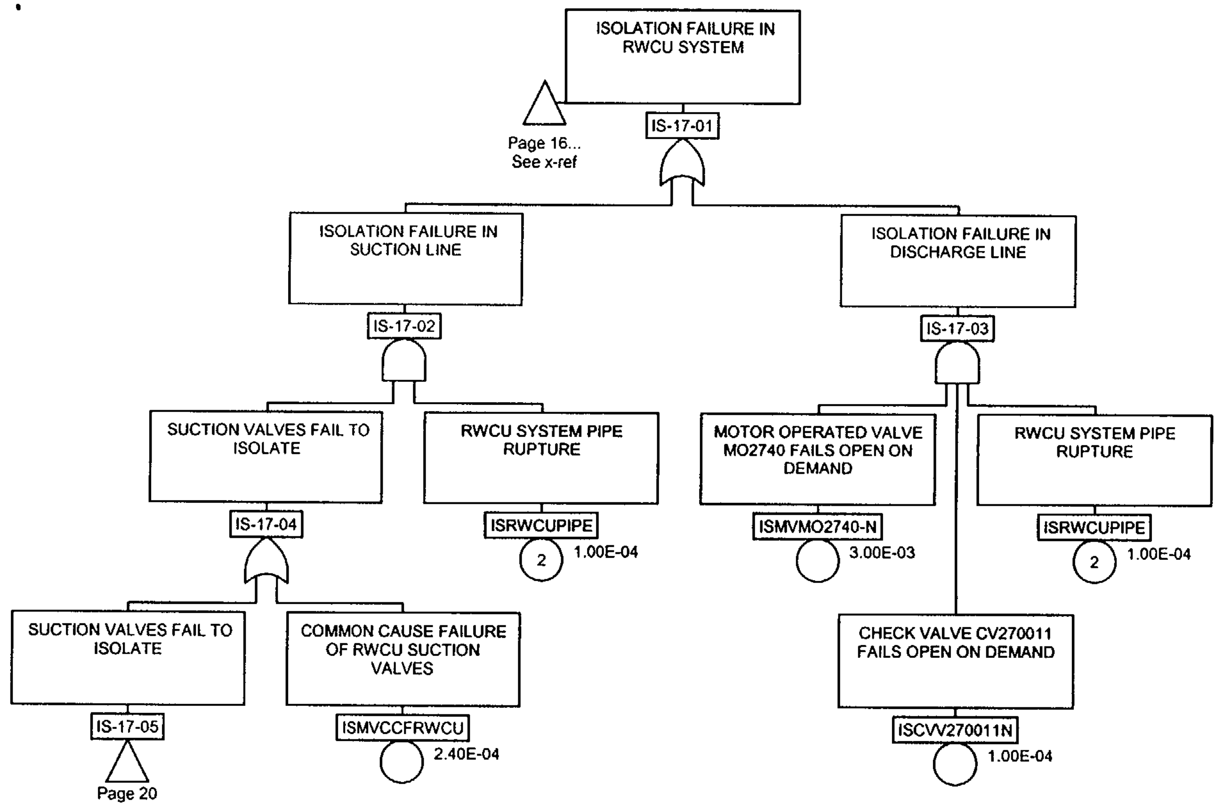

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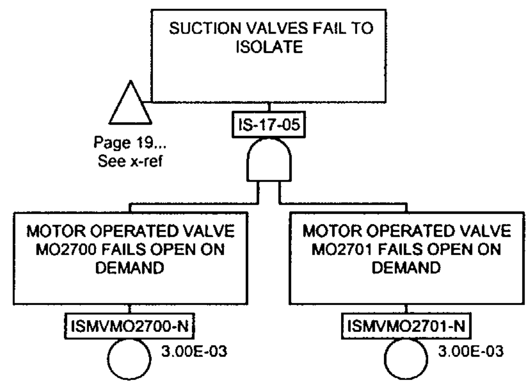

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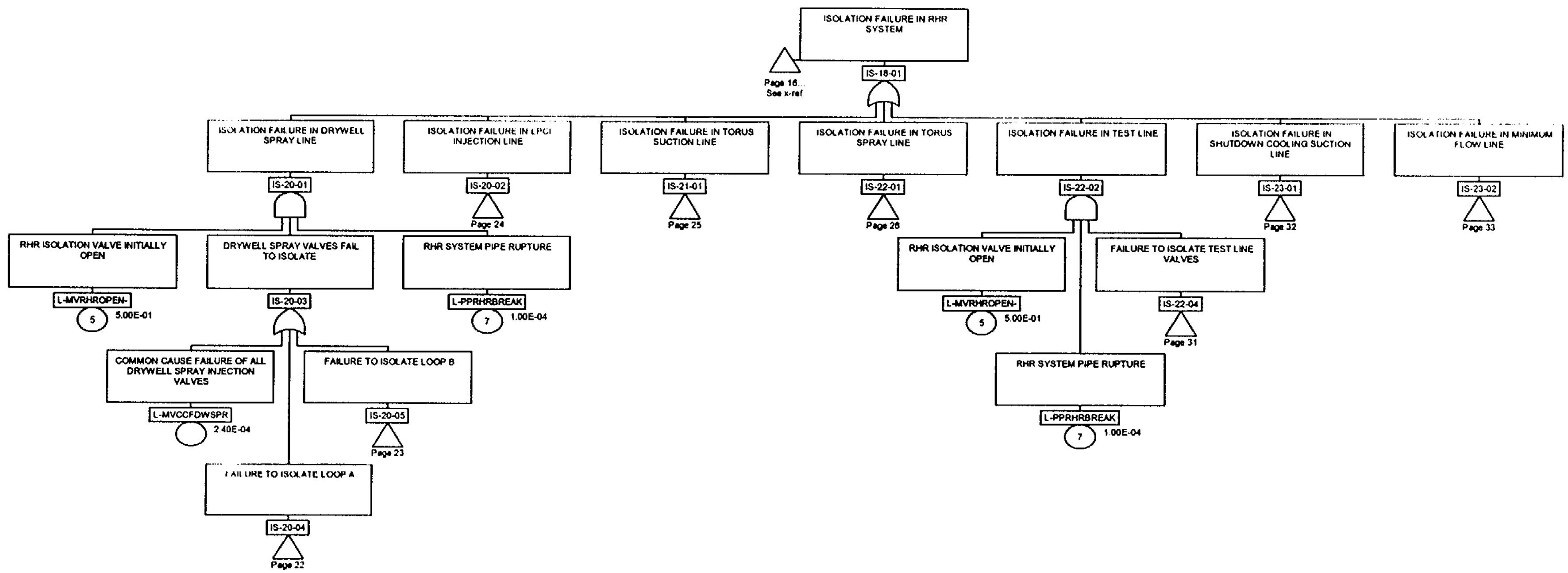

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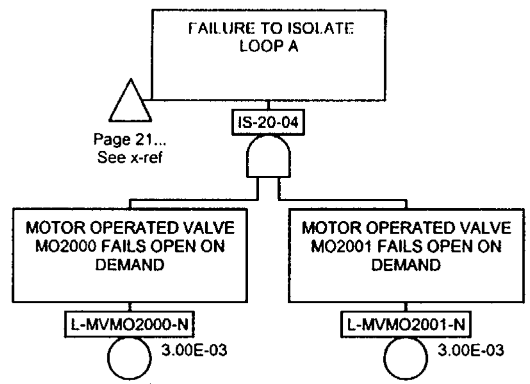

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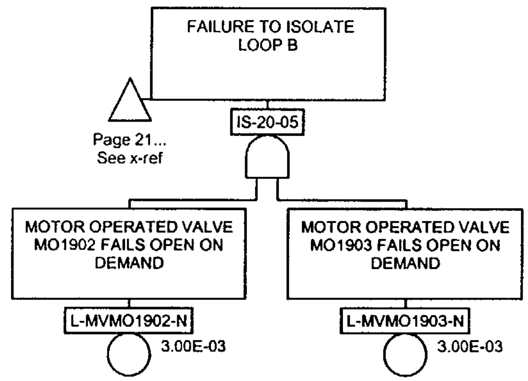

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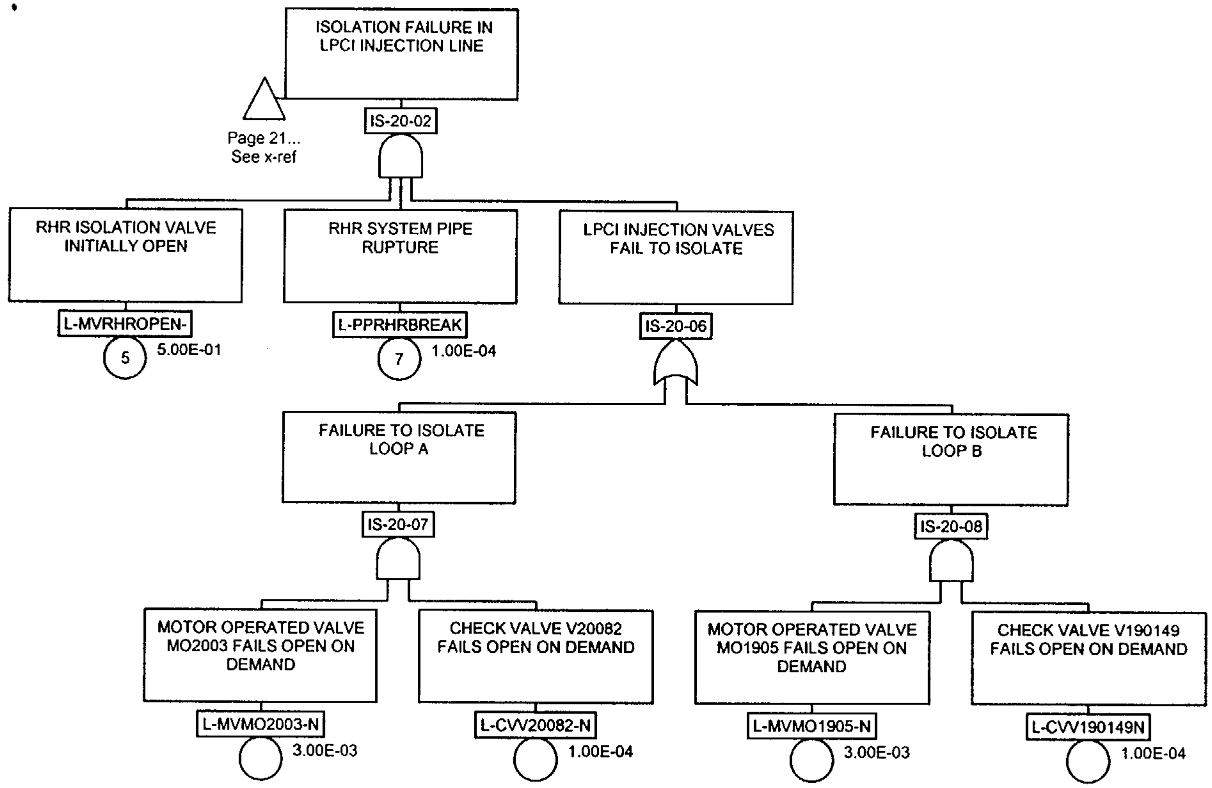

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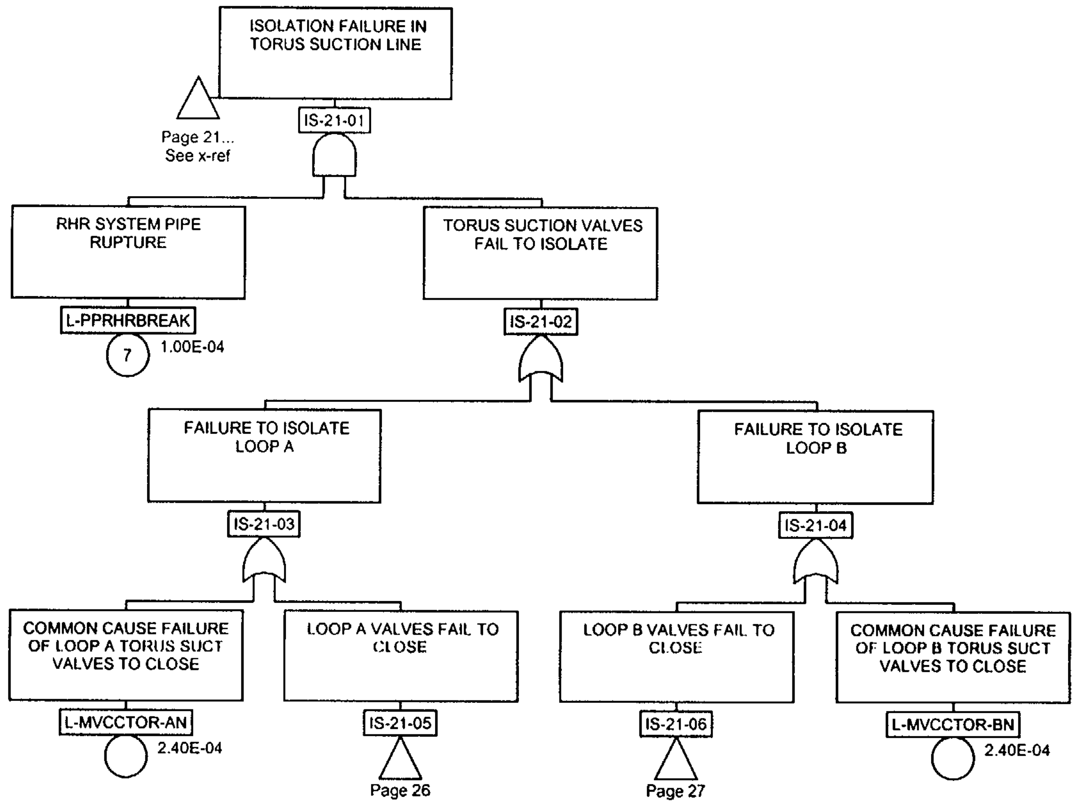

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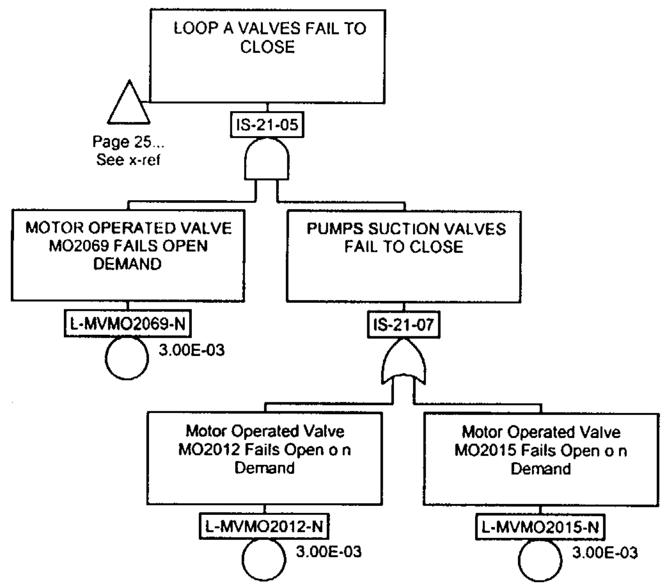

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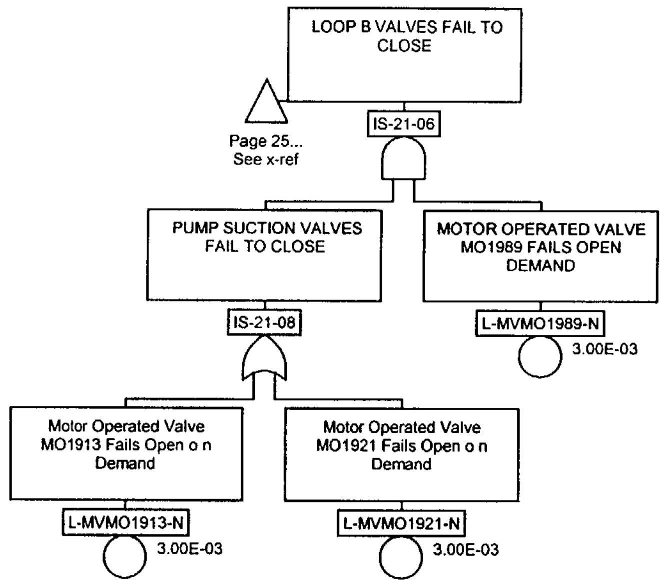

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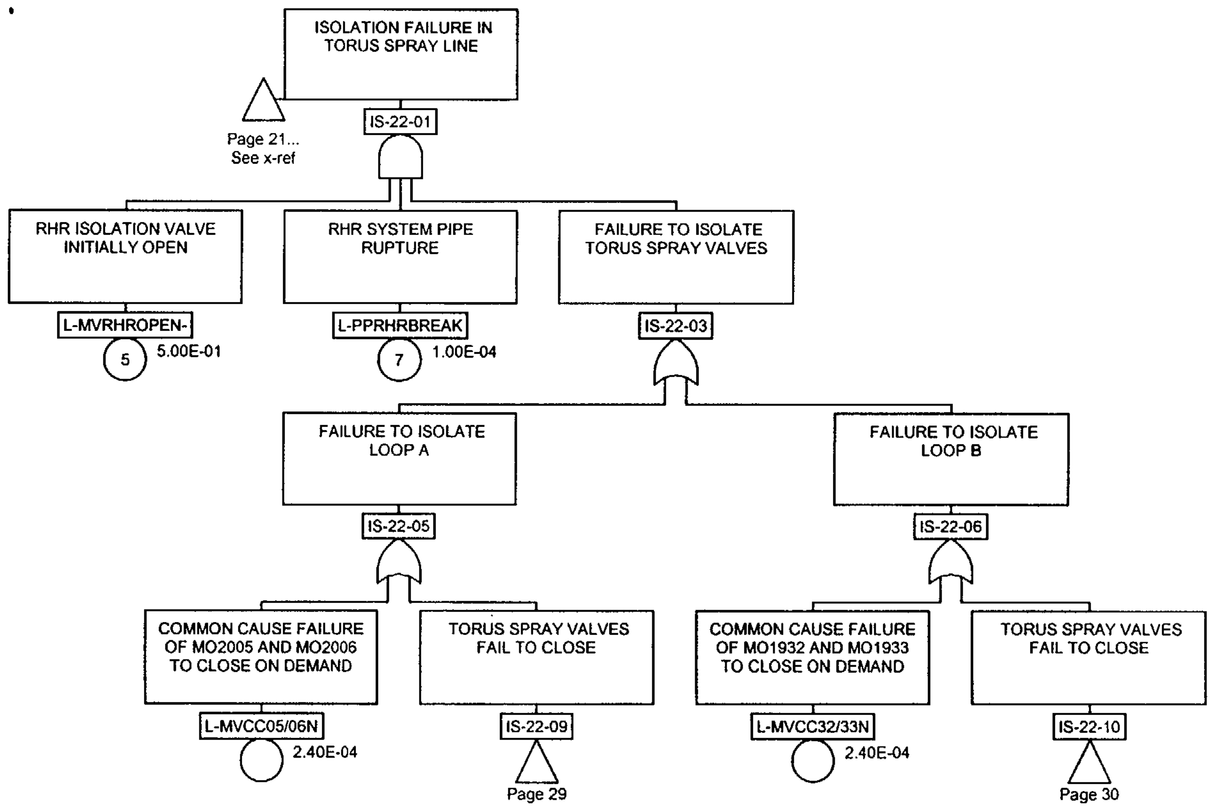

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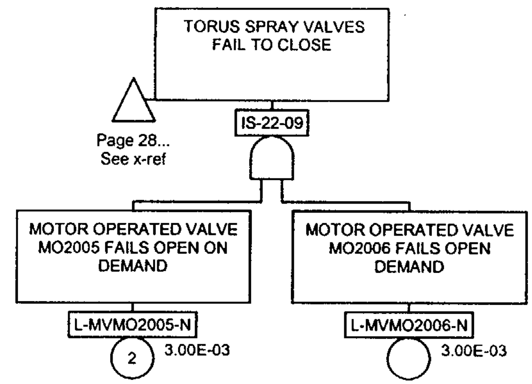

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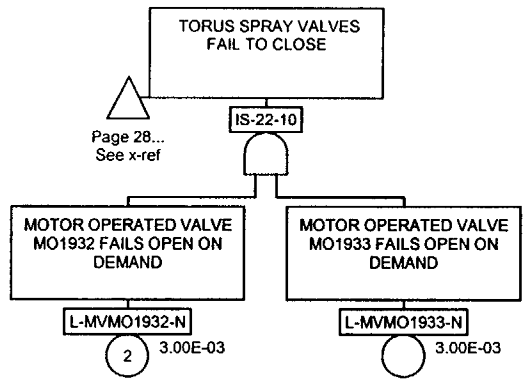

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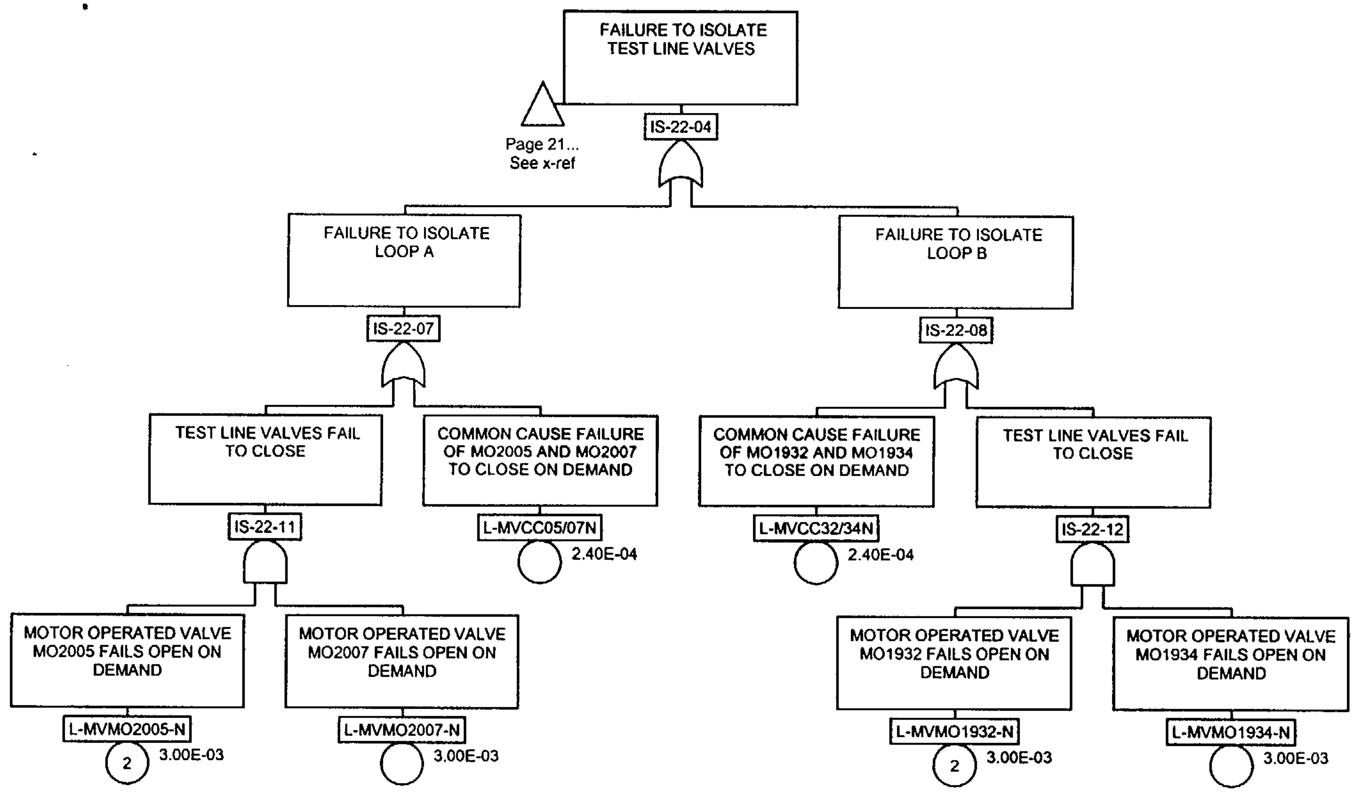

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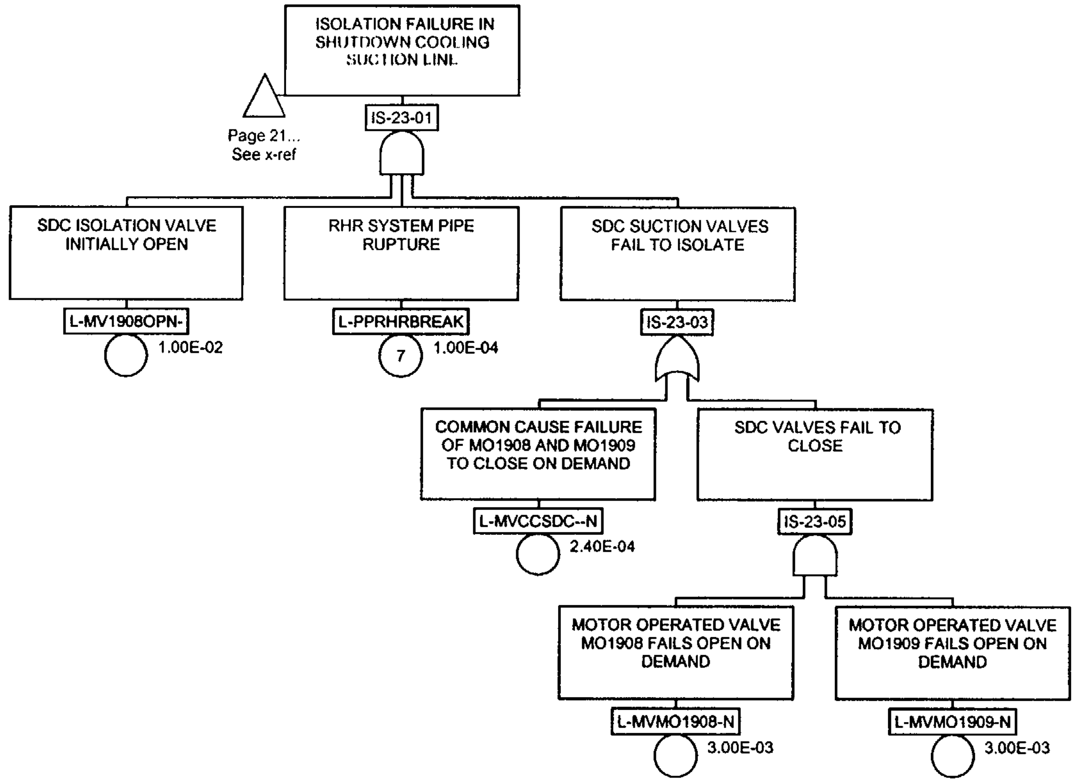

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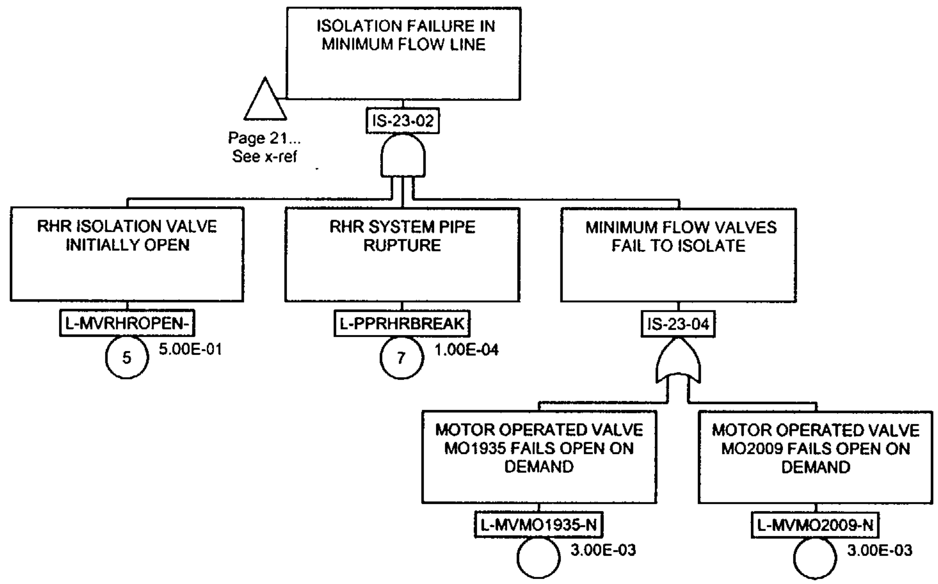

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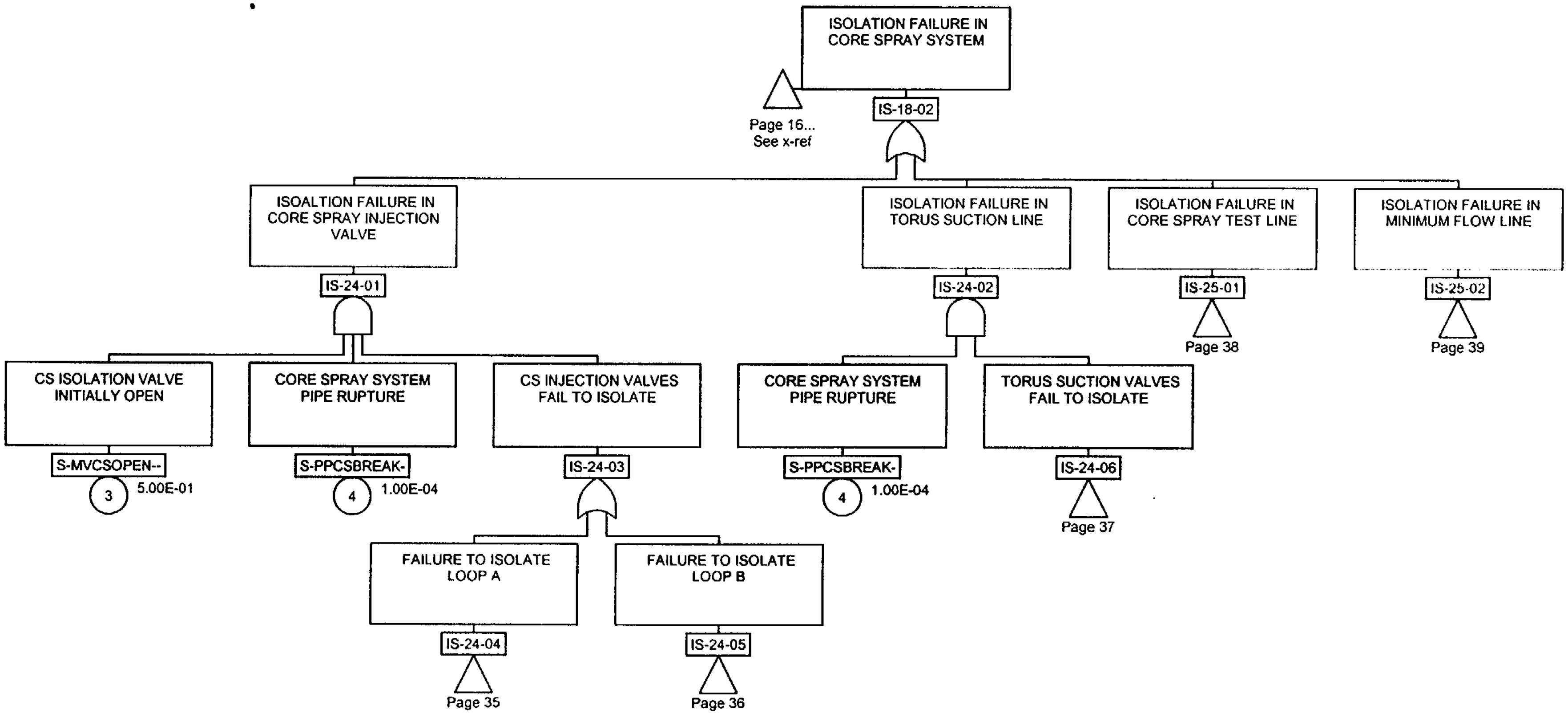

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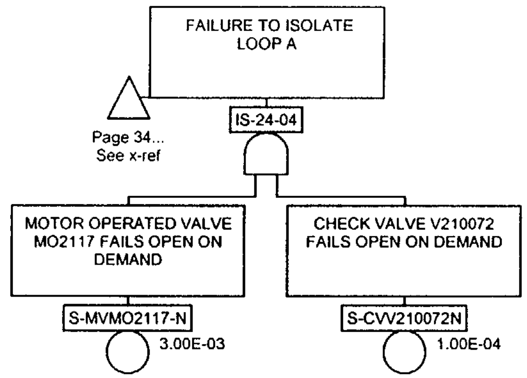

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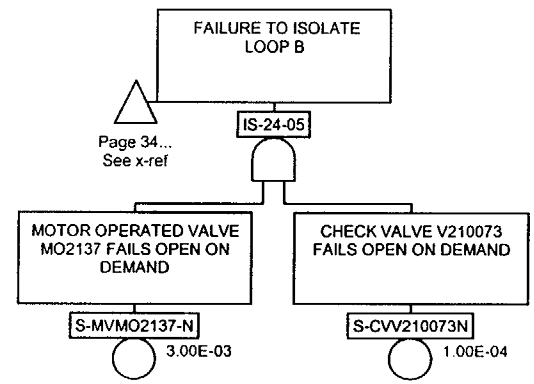

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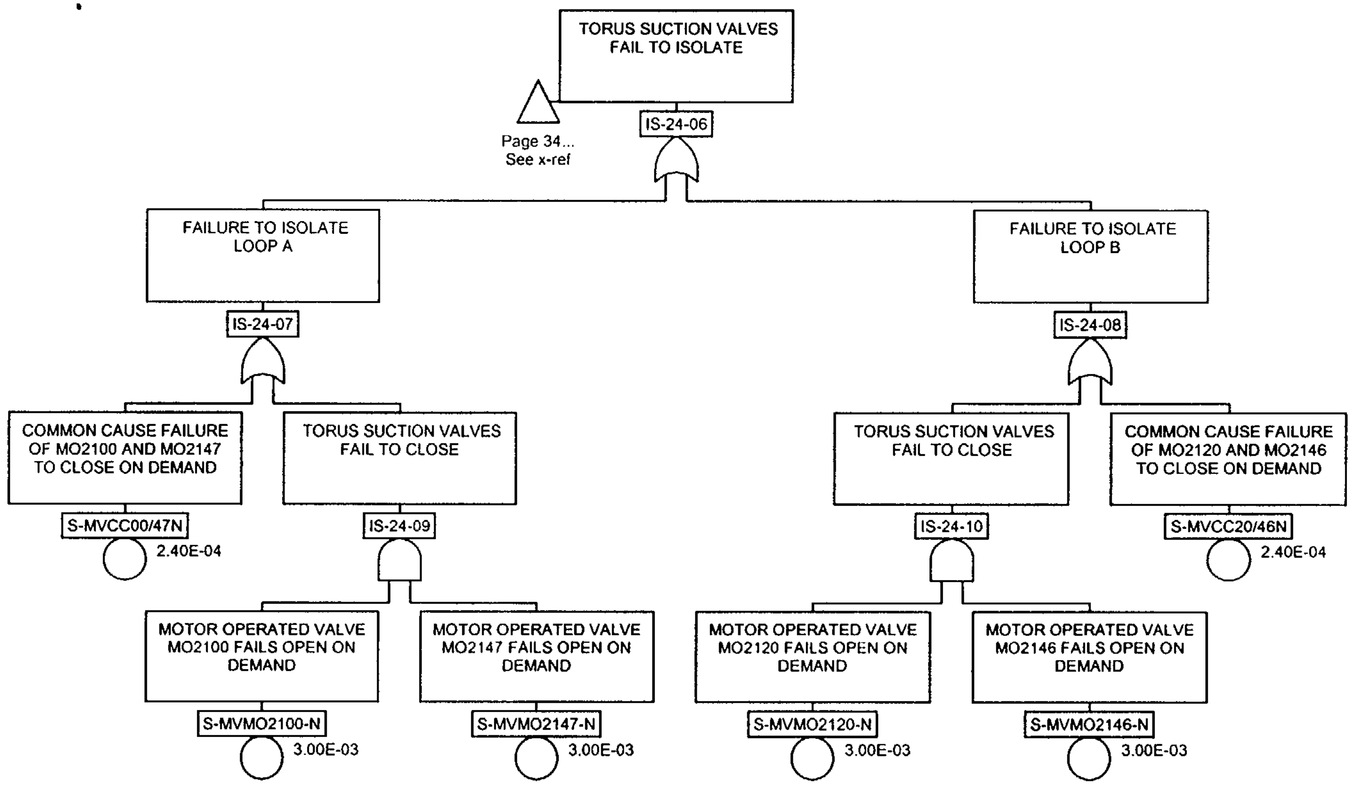

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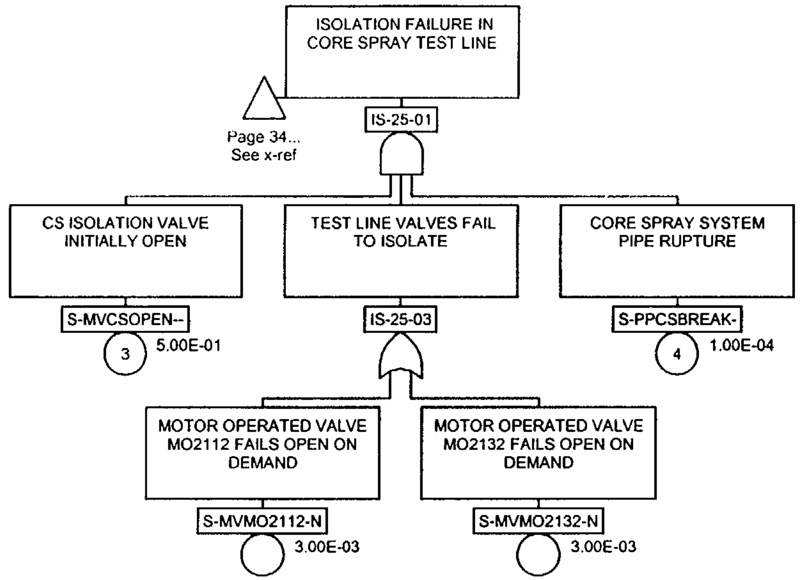

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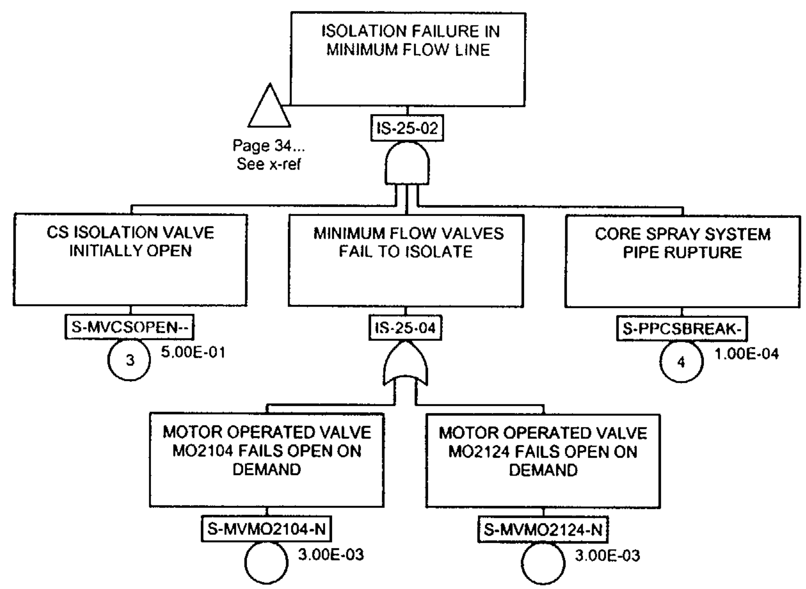

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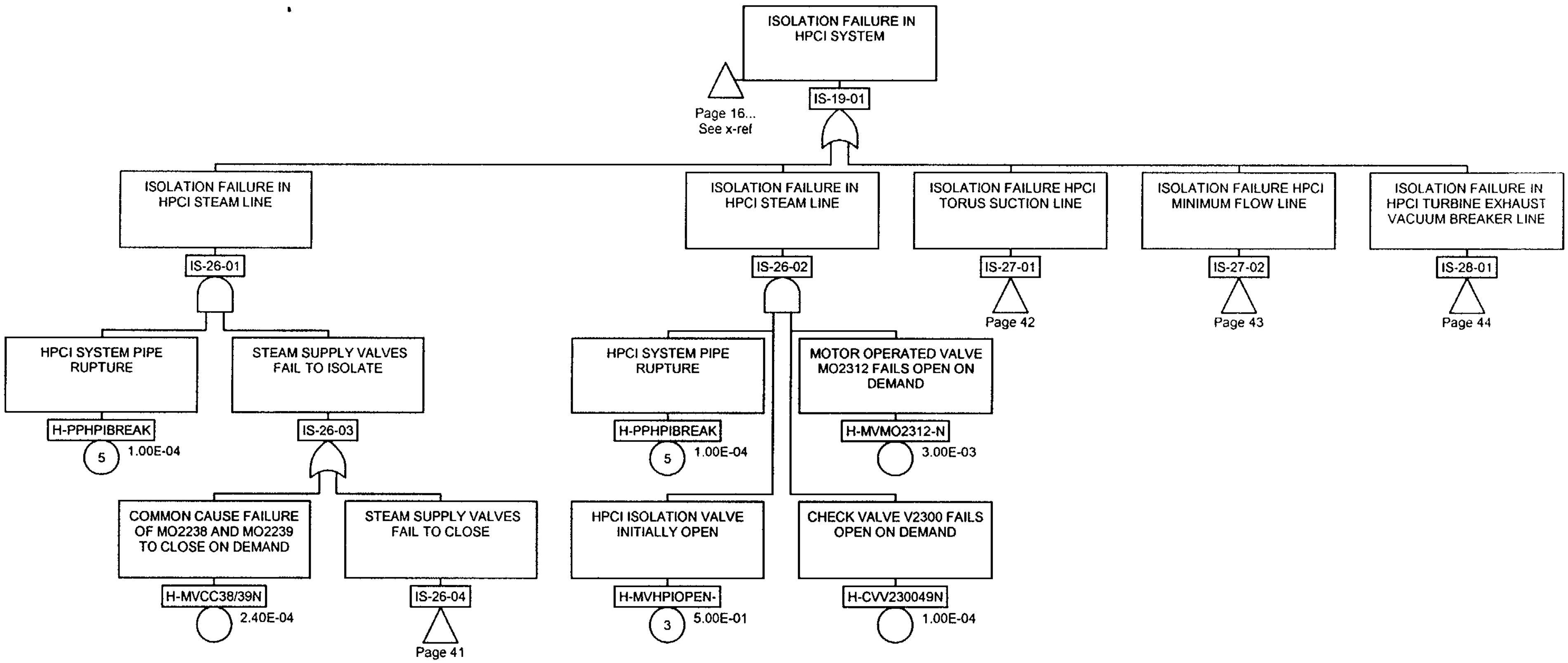

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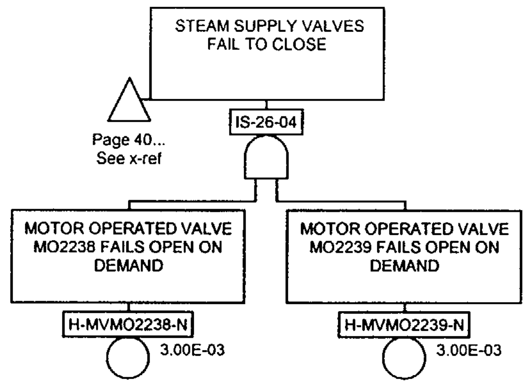

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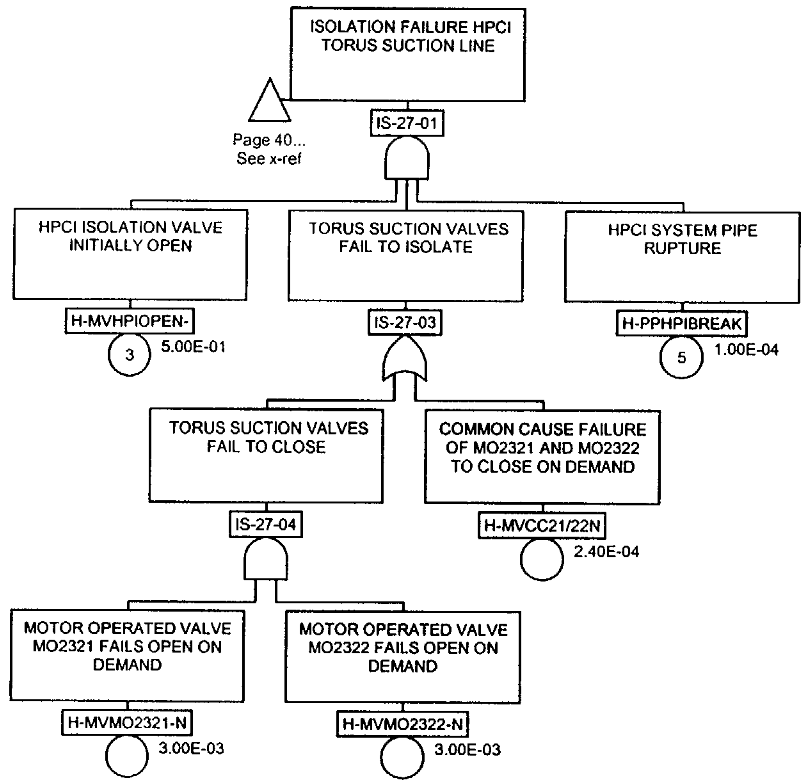

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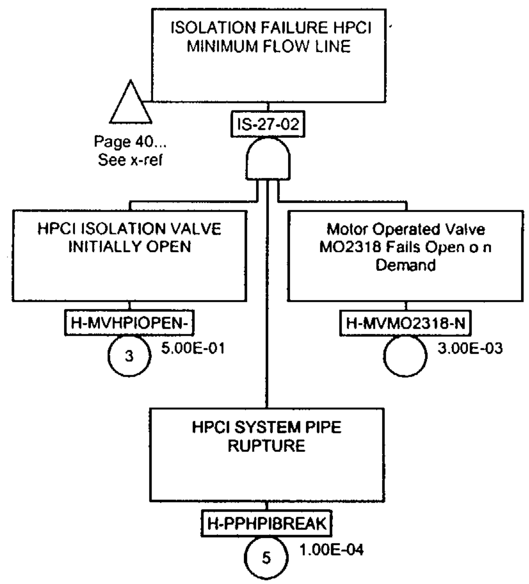

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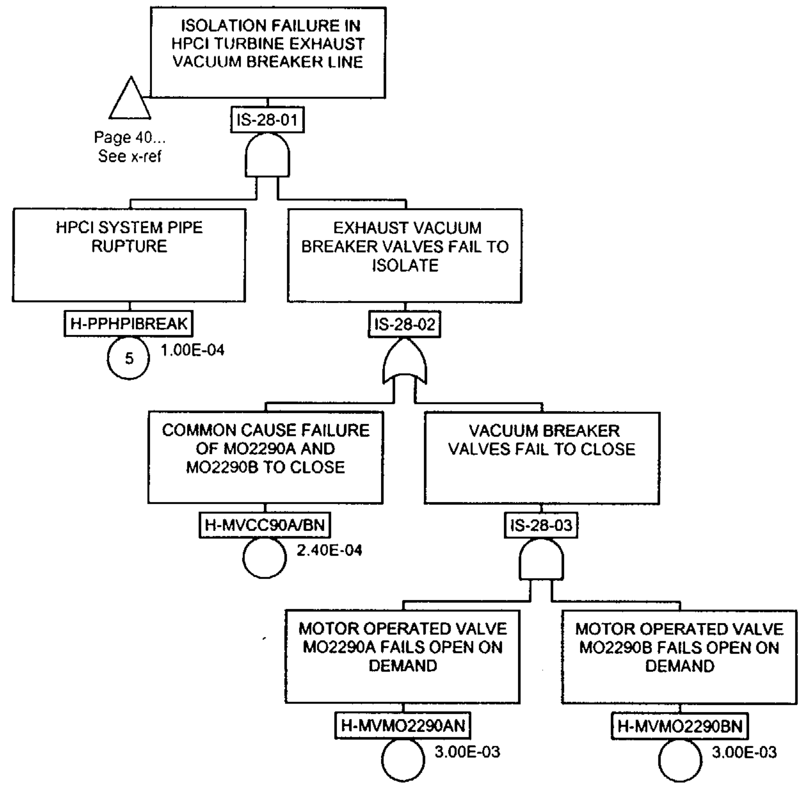

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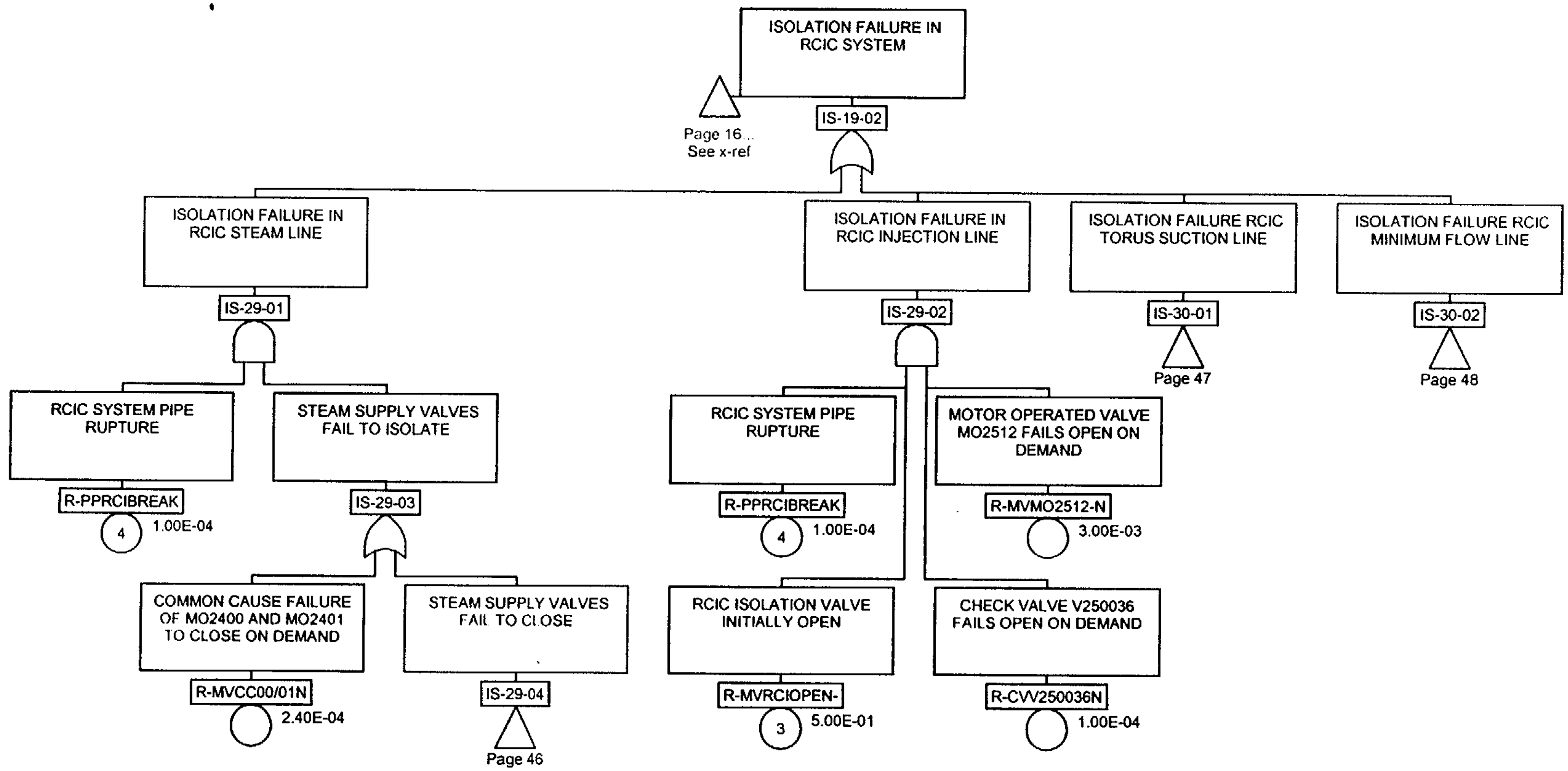

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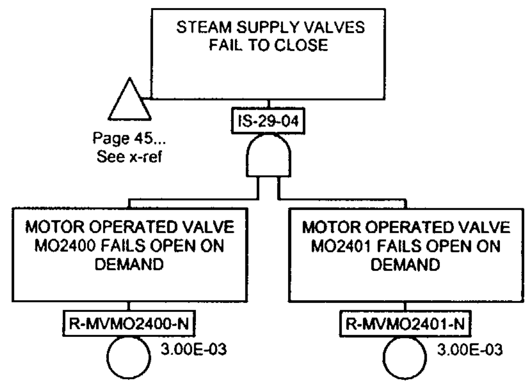

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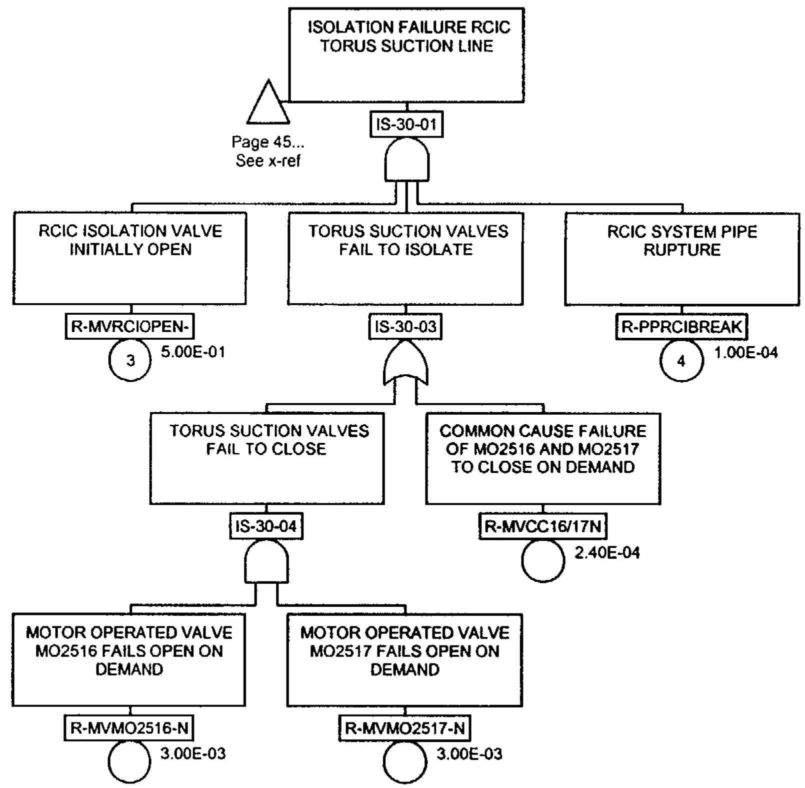

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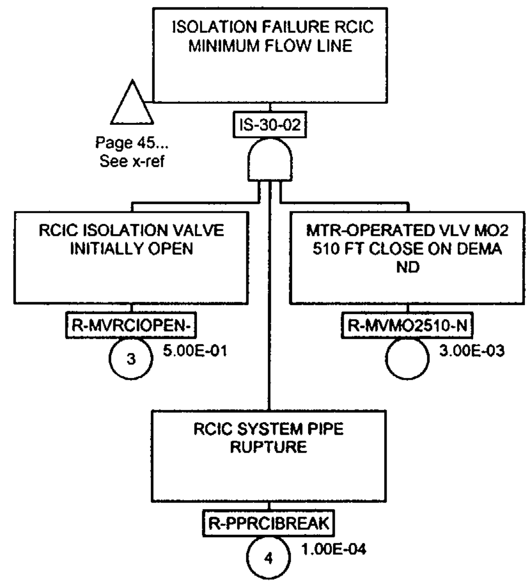

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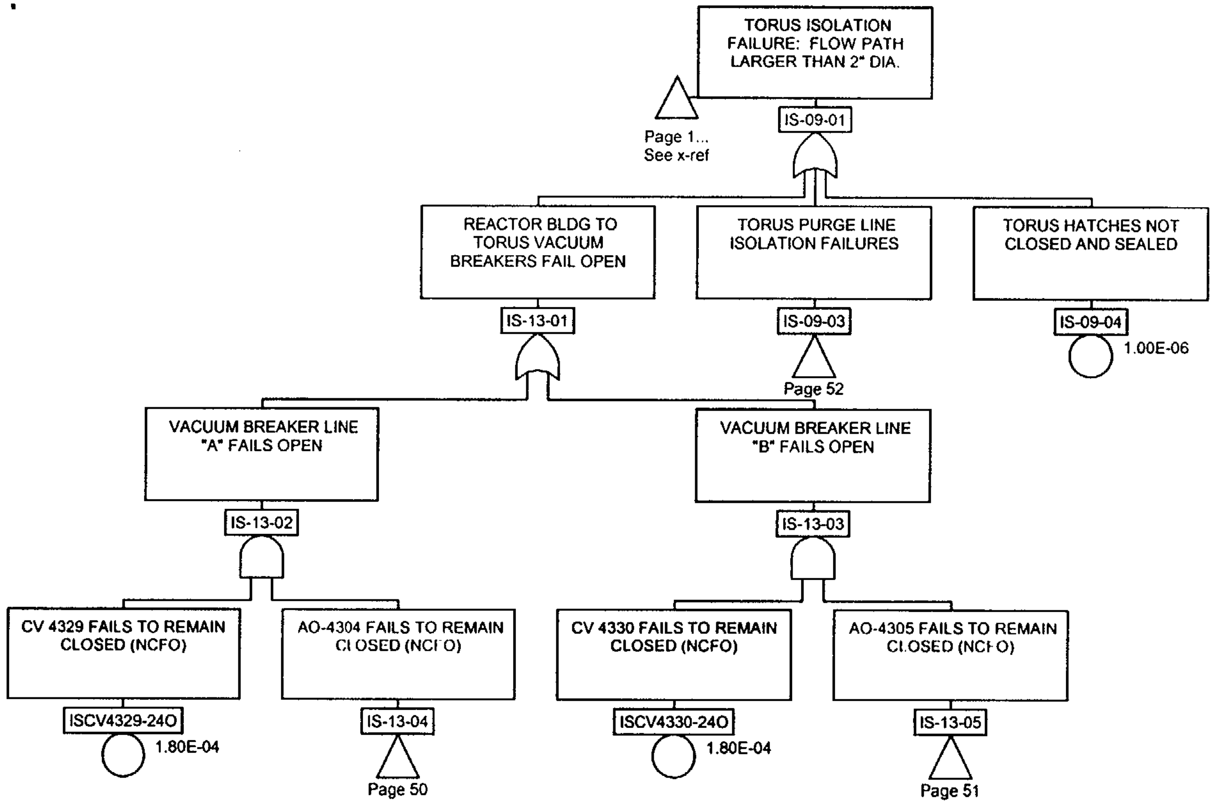

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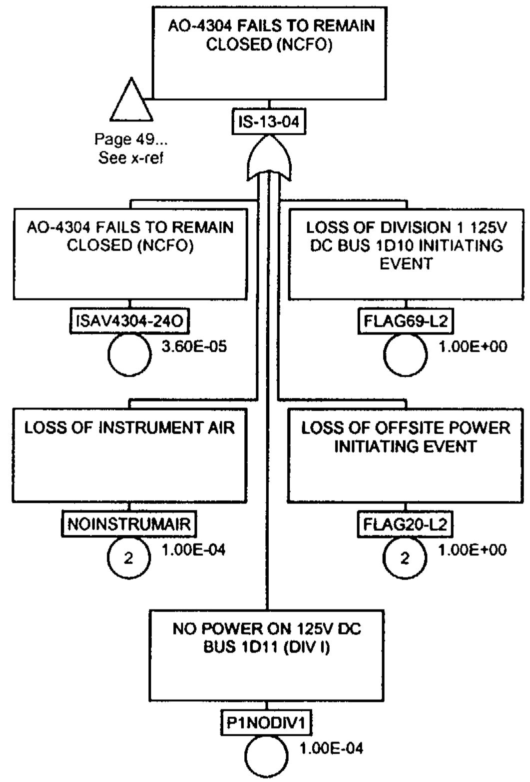

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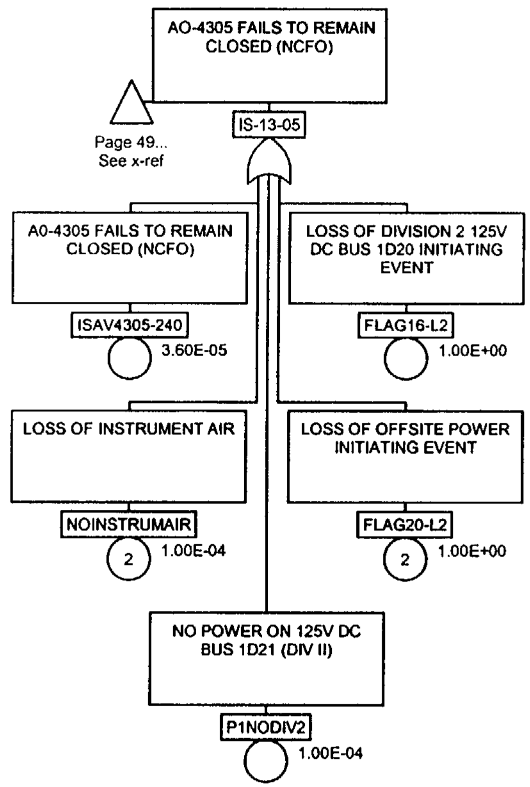

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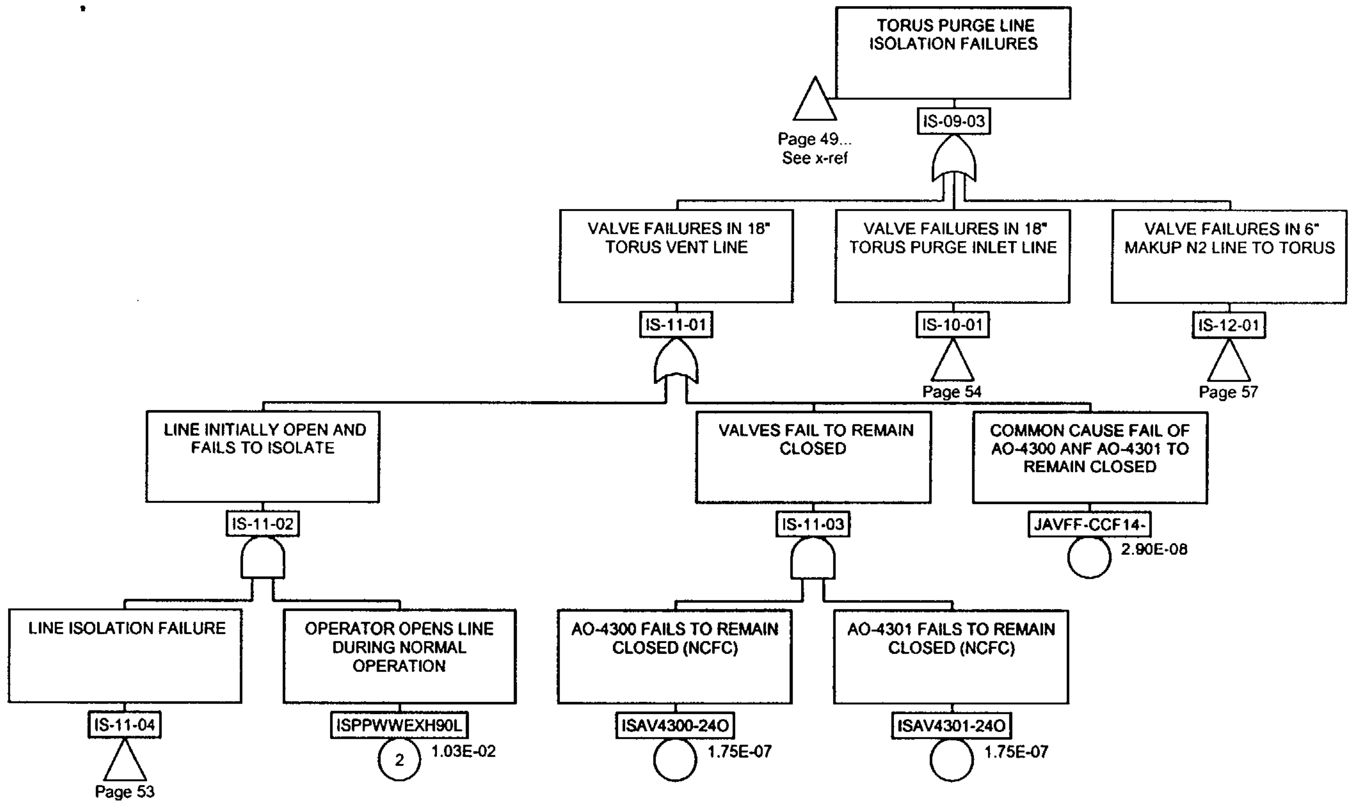

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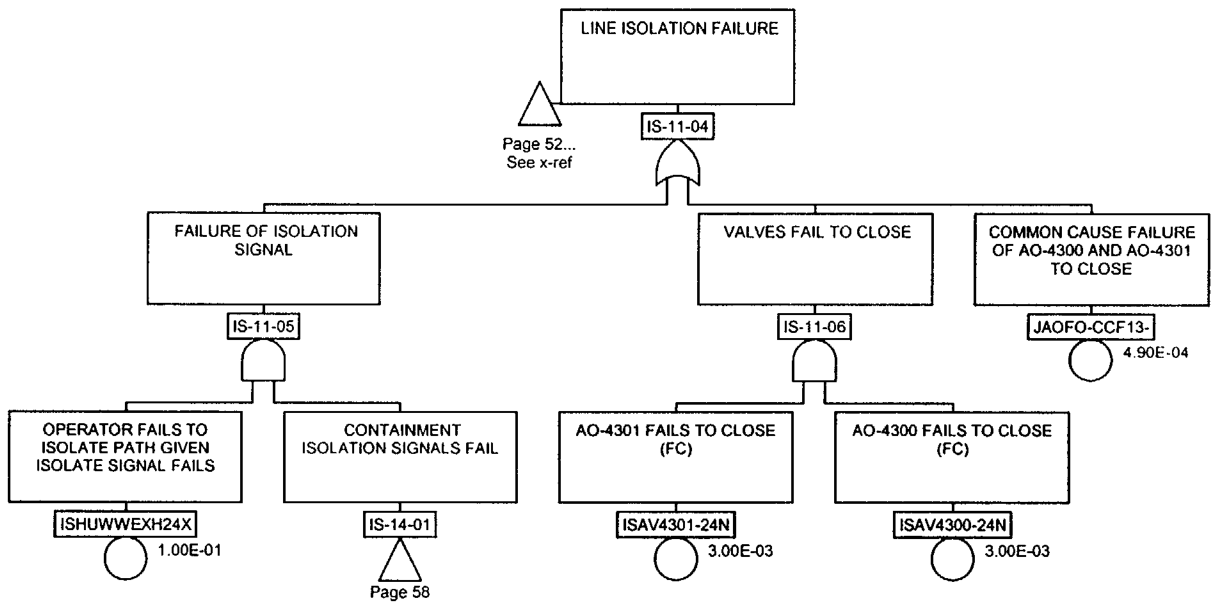

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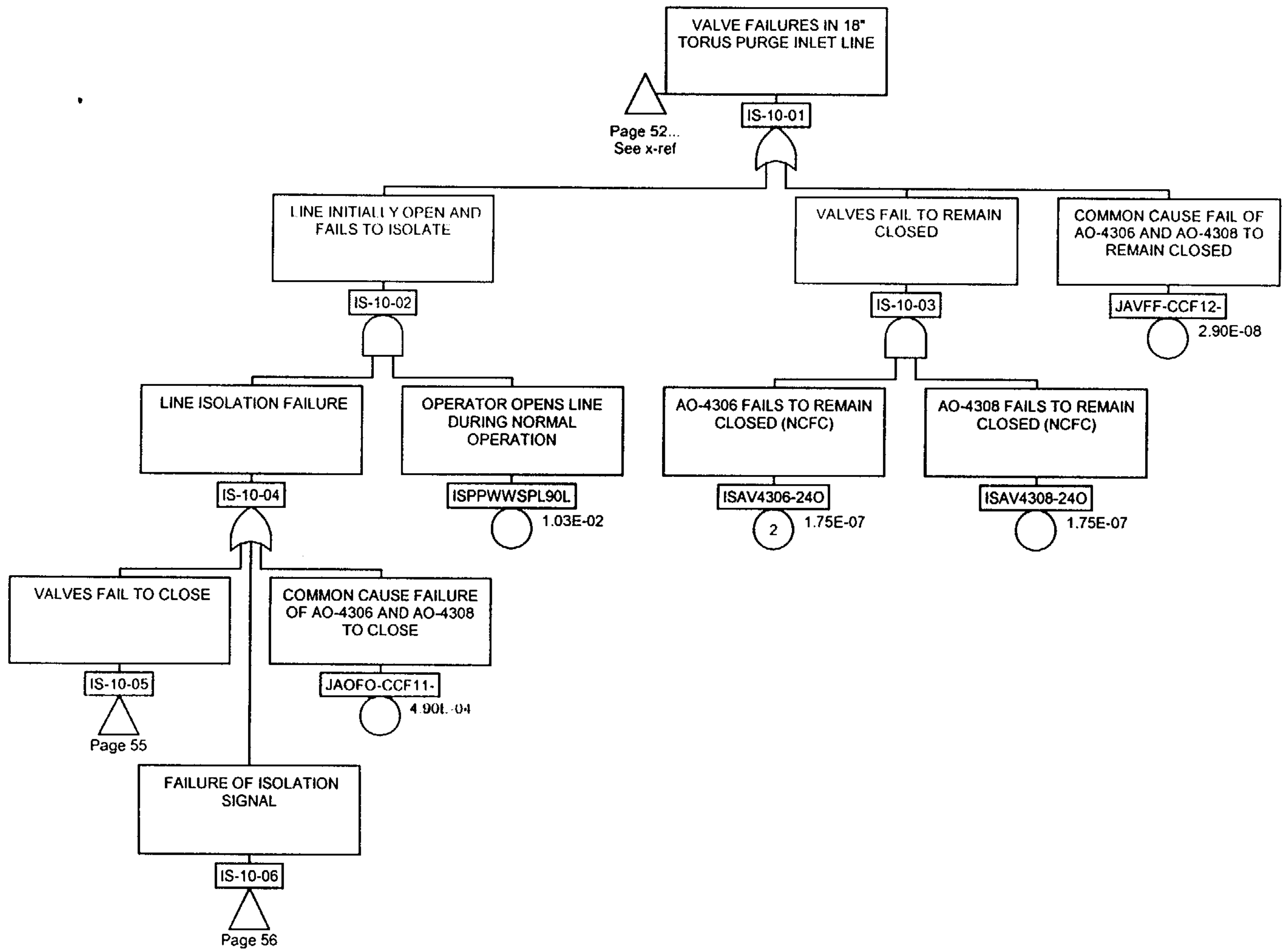

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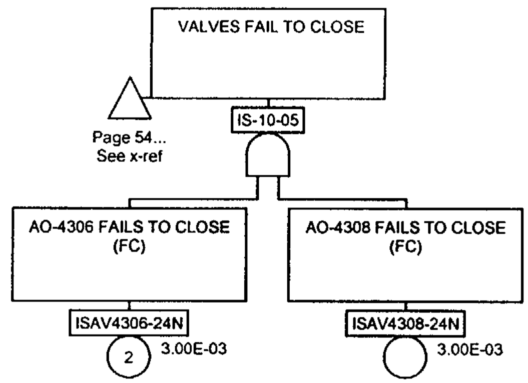

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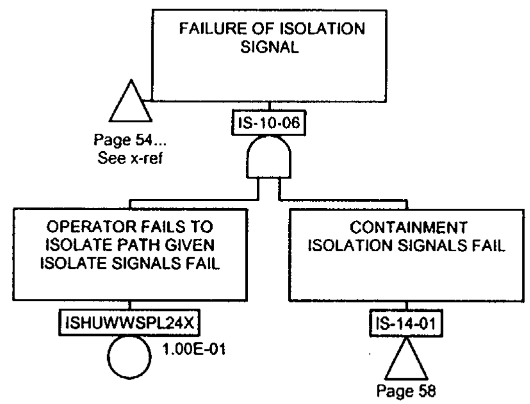

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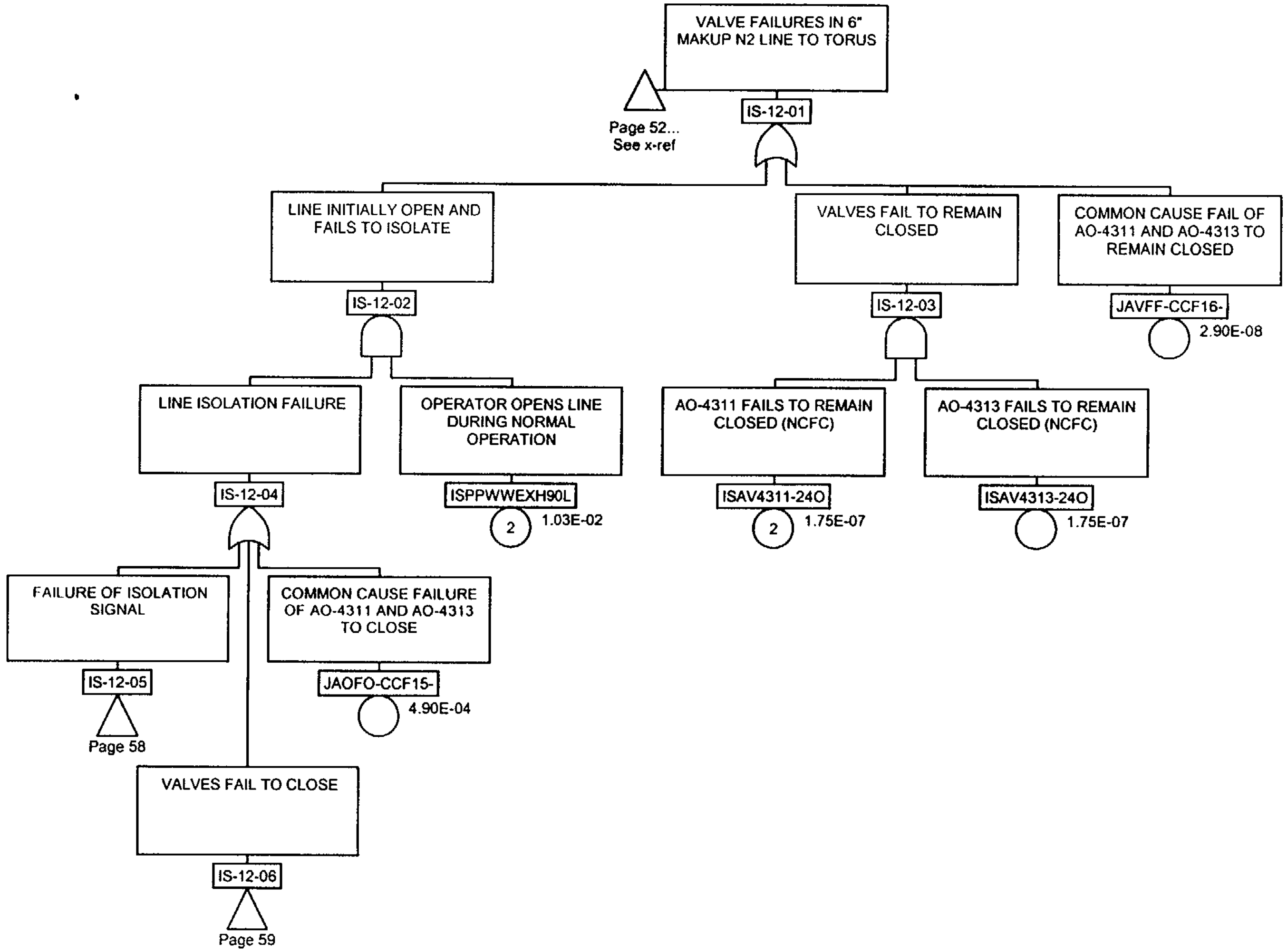

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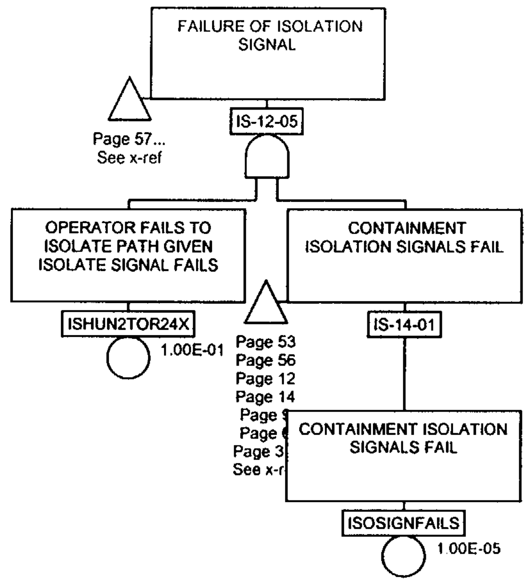

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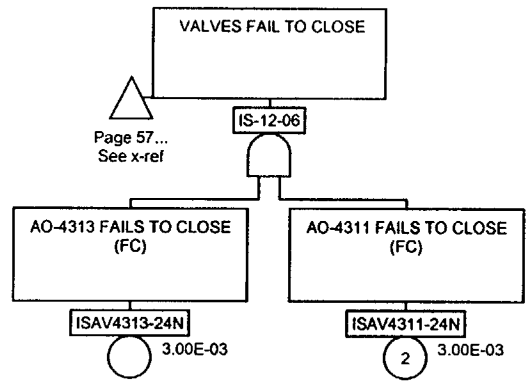

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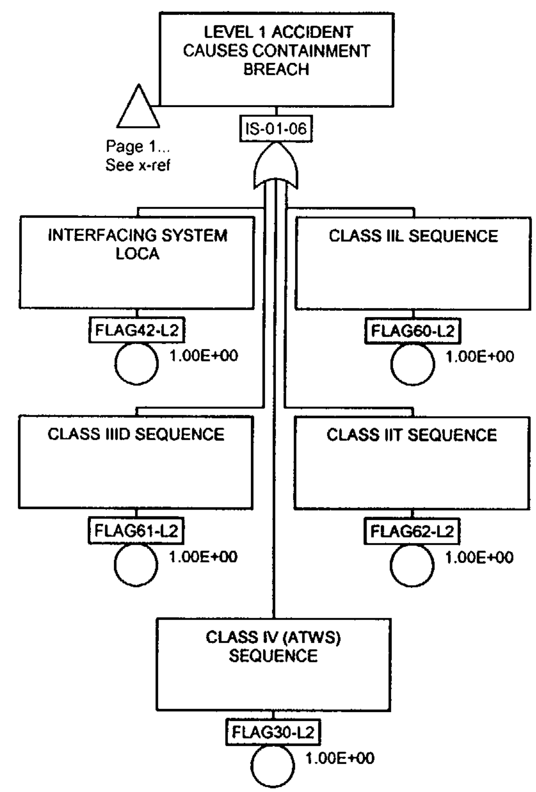

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Name	Page	Zone	Name	Page	Zone	Name	Page	Zone	Name	Page	Zone
FLAG16-L2	51	2	IS-03-02	2	2	IS-07-04	12	3	IS-13-04	49	2
FLAG20-L2	50	2	IS-03-03	2	4	IS-07-05	12	2	IS-13-04	50	2
FLAG20-L2	51	2	IS-03-04	2	2	IS-07-06	12	4	IS-13-05	49	4
FLAG30-L2	60	2	IS-03-05	2	2	IS-08-01	1	4	IS-13-05	51	2
FLAG42-L2	60	1	IS-03-05	3	2	IS-08-02	1	4	IS-14-01	3	2
FLAG60-L2	60	2	IS-03-06	2	2	IS-09-01	1	5	IS-14-01	6	2
FLAG61-L2	60	1	IS-03-06	4	2	IS-09-01	49	4	IS-14-01	9	2
FLAG62-L2	60	2	IS-04-01	1	2	IS-09-03	49	4	IS-14-01	12	2
FLAG69-L2	50	2	IS-04-01	5	4	IS-09-03	52	4	IS-14-01	14	2
H-CVV230049N	40	5	IS-04-02	5	2	IS-09-04	49	5	IS-14-01	53	2
H-MVCC21/22N	42	3	IS-04-03	5	4	IS-10-01	52	4	IS-14-01	56	2
H-MVCC38/39N	40	2	IS-04-04	5	2	IS-10-01	54	4	IS-14-01	58	2
H-MVCC90A/BN	44	2	IS-04-05	5	1	IS-10-02	54	2	IS-15-01	1	6
H-MVHPIOPEN-	40	4	IS-04-05	6	2	IS-10-03	54	4	IS-15-01	16	5
H-MVHPIOPEN-	42	1	IS-04-06	5	2	IS-10-04	54	2	IS-16-01	16	3
H-MVHPIOPEN-	43	1	IS-04-06	7	2	IS-10-05	54	1	IS-16-02	16	2
H-MVMO2238-N	41	1	IS-05-01	1	2	IS-10-05	55	2	IS-16-03	16	2
H-MVMO2239-N	41	2	IS-05-01	8	4	IS-10-06	54	2	IS-16-03	17	2
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H-MVMO2290BN	44	3	IS-05-03	8	4	IS-11-01	52	3	IS-16-05	16	3
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H-MVMO2318-N	43	2	IS-05-05	8	1	IS-11-03	52	4	IS-16-06	18	2
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IS-02-05	11	4	IS-07-02	11	2	IS-13-01	49	3	IS-19-02	16	8
IS-03-01	1	1	IS-07-03	11	4	IS-13-02	49	2	IS-19-02	45	4
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Name	Page	Zone	Name	Page	Zone	Name	Page	Zone	Name	Page	Zone
IS-20-02	21	3	IS-23-01	32	2	IS-28-02	44	2	ISAV4308-24N	55	2
IS-20-02	24	2	IS-23-02	21	8	IS-28-03	44	3	ISAV4308-24O	54	5
IS-20-03	21	2	IS-23-02	33	2	IS-29-01	45	2	ISAV4311-24N	10	1
IS-20-04	21	2	IS-23-03	32	3	IS-29-02	45	4	ISAV4311-24N	59	2
IS-20-04	22	2	IS-23-04	33	3	IS-29-03	45	2	ISAV4311-24O	8	4
IS-20-05	21	3	IS-23-05	32	4	IS-29-04	45	3	ISAV4311-24O	57	4
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IS-20-07	24	2	IS-24-03	34	3	IS-30-01	47	2	ISAV4313-24N	59	1
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IS-21-02	25	3	IS-24-05	36	2	IS-30-04	47	2	ISCVV270011N	19	4
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IS-21-04	25	4	IS-24-06	37	3	ISAV3704-24N	12	3	ISHUDWEQP24X	14	1
IS-21-05	25	2	IS-24-07	37	2	ISAV3704-24O	11	4	ISHUDWEXH24X	6	1
IS-21-05	26	2	IS-24-08	37	5	ISAV3705-24N	12	4	ISHUDWFLR24X	12	1
IS-21-06	25	3	IS-24-09	37	2	ISAV3705-24O	11	3	ISHUDWSPL24X	3	1
IS-21-06	27	2	IS-24-10	37	4	ISAV3728-24N	15	1	ISHUN2DW-24X	9	1
IS-21-07	26	2	IS-25-01	34	6	ISAV3728-24O	13	5	ISHUN2TOR24X	58	1
IS-21-08	27	2	IS-25-01	38	2	ISAV3729-24N	15	2	ISHUWWEXH24X	53	1
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IS-22-03	28	3	IS-25-04	39	2	ISAV4301-24N	53	3	ISMVMO2700-N	20	1
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IS-22-06	28	4	IS-26-04	40	3	ISAV4303-24N	7	2	ISMVMO4424-N	18	3
IS-22-07	31	2	IS-26-04	41	2	ISAV4303-24O	5	5	ISMVMO4841AN	17	1
IS-22-08	31	4	IS-27-01	40	5	ISAV4304-24O	50	1	ISMVMO4841BN	17	2
IS-22-09	28	3	IS-27-01	42	2	ISAV4305-240	51	1	ISMVVLVOPEN	16	4
IS-22-09	29	2	IS-27-02	40	6	ISAV4306-24N	4	2	ISOSIGNFAILS	58	2
IS-22-10	28	5	IS-27-02	43	2	ISAV4306-24N	55	1	ISPPDWEQP24L	13	3
IS-22-10	30	2	IS-27-03	42	2	ISAV4306-24O	2	4	ISPPDWEXH90L	2	1
IS-22-11	31	2	IS-27-04	42	2	ISAV4306-24O	54	4	ISPPDWFLR24L	11	2
IS-22-12	31	5	IS-28-01	40	7	ISAV4307-24N	4	1	ISPPDWSPL90L	5	3
IS-23-01	21	7	IS-28-01	44	2	ISAV4307-24O	2	3	ISPPDWSPL90L	8	3

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Name	Page	Zone	Name	Page	Zone	Name	Page	Zone	Name	Page	Zone
ISPPWWEXH90L	52	2	L-MVMO1909-N	32	4	R-MVCC16/17N	47	3			
ISPPWWEXH90L	57	3	L-MVMO1913-N	27	1	R-MVMO2400-N	46	1			
ISPPWWSPL90L	54	3	L-MVMO1921-N	27	2	R-MVMO2401-N	46	2			
ISPUMPSEAL	16	1	L-MVMO1932-N	30	1	R-MVMO2510-N	48	2			
ISRBCCWPIPE	16	2	L-MVMO1932-N	31	4	R-MVMO2512-N	45	5			
ISRWCUPIPE	19	3	L-MVMO1933-N	30	2	R-MVMO2516-N	47	1			
ISRWCUPIPE	19	5	L-MVMO1934-N	31	5	R-MVMO2517-N	47	2			
JAOFO-CCF11-	54	2	L-MVMO1935-N	33	3	R-MVRCIOPEN-	45	4			
JAOFO-CCF13-	53	5	L-MVMO1989-N	27	3	R-MVRCIOPEN-	47	1			
JAOFO-CCF15-	57	2	L-MVMO2000-N	22	1	R-MVRCIOPEN-	48	1			
JAVFF-CCF1--	2	3	L-MVMO2001-N	22	2	R-PPRCIBREAK	45	1			
JAVFF-CCF10-	11	5	L-MVMO2003-N	24	2	R-PPRCIBREAK	45	4			
JAVFF-CCF12-	54	5	L-MVMO2005-N	29	1	R-PPRCIBREAK	47	3			
JAVFF-CCF14-	52	5	L-MVMO2005-N	31	1	R-PPRCIBREAK	48	2			
JAVFF-CCF16-	57	5	L-MVMO2006-N	29	2	S-CVV210072N	35	2			
JAVFF-CCF2--	2	5	L-MVMO2007-N	31	2	S-CVV210073N	36	2			
JAVFF-CCF3--	5	5	L-MVMO2009-N	33	4	S-MVCC00/47N	37	1			
JAVFF-CCF4--	5	2	L-MVMO2012-N	26	2	S-MVCC20/46N	37	5			
JAVFF-CCF5--	8	2	L-MVMO2015-N	26	3	S-MVCSOPEN--	34	1			
JAVFF-CCF6--	8	5	L-MVMO2069-N	26	1	S-MVCSOPEN--	38	1			
JAVFF-CCF7--	13	2	L-MVRHROPEN-	21	1	S-MVCSOPEN--	39	1			
JAVFF-CCF8--	13	5	L-MVRHROPEN-	21	6	S-MVMO2100-N	37	2			
JAVFF-CCF9--	12	5	L-MVRHROPEN-	24	1	S-MVMO2104-N	39	2			
L-CVV190149N	24	5	L-MVRHROPEN-	28	1	S-MVMO2112-N	38	2			
L-CVV20082-N	24	3	L-MVRHROPEN-	33	1	S-MVMO2117-N	35	1			
L-MV1908OPN-	32	1	L-PPRHRBREAK	21	3	S-MVMO2120-N	37	4			
L-MVCC05/06N	28	2	L-PPRHRBREAK	21	6	S-MVMO2124-N	39	3			
L-MVCC05/07N	31	3	L-PPRHRBREAK	24	2	S-MVMO2132-N	38	3			
L-MVCC32/33N	28	4	L-PPRHRBREAK	25	1	S-MVMO2137-N	36	1			
L-MVCC32/34N	31	4	L-PPRHRBREAK	28	2	S-MVMO2146-N	37	5			
L-MVCCFDWSPR	21	2	L-PPRHRBREAK	32	2	S-MVMO2147-N	37	3			
L-MVCCSDC--N	32	3	L-PPRHRBREAK	33	2	S-PPCSBREAK-	34	2			
L-MVCCTOR-AN	25	1	NOINSTRUMAIR	50	1	S-PPCSBREAK-	34	4			
L-MVCCTOR-BN	25	4	NOINSTRUMAIR	51	1	S-PPCSBREAK-	38	3			
L-MVMO1902-N	23	1	P1NODIV1	50	2	S-PPCSBREAK-	39	3			
L-MVMO1903-N	23	2	P1NODIV2	51	2						
L-MVMO1905-N	24	4	R-CVV250036N	45	5						
L-MVMO1908-N	32	3	R-MVCC00/01N	45	2						

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Attachment B

CUTSETS FOR THE CONTAINMENT ISOLATION FAULT TREE

Cutset Report

IS-01-01 = 5.08E-03 (Probability)

Probability	%	Class	Inputs
5.00E-03	98.4%		ISAV-CIVS24F
4.00E-05	99.1%		IS-08-02
5.05E-06	99.2%		ISPPDWEXH9JAVFF-CCF1--
5.05E-06	99.3%		ISPPDWSPL9UAVFF-CCF4--
5.05E-06	99.4%		ISPPDWSPL9UAVFF-CCF5--
5.05E-06	99.5%		ISPPWWEXHSJAOFO-CCF13-
5.05E-06	99.6%		ISPPWWEXHSJAOFO-CCF15-
5.05E-06	99.7%		ISPPWWSPL9JAOFO-CCF11-
4.90E-06	99.8%		ISPPDWEQP2JAVFF-CCF7--
4.90E-06	99.9%		ISPPDWFLR2JAVFF-CCF9--
1.00E-06	99.9%		IS-09-04
1.50E-07	100.0%		H-MVHPIOPEIH-MVMO2318-H-PPHPIBREAK
1.50E-07	100.0%		L-MVMO1935-L-MVRHROPEL-PPRHRBREAK
1.50E-07	100.0%		L-MVMO2009-L-MVRHROPEL-PPRHRBREAK
1.50E-07	100.0%		R-MVMO2510-R-MVRCIOPEIR-PPRCIBREAK
1.50E-07	100.0%		S-MVCSOPENS-MVMO2104-S-PPCSBREAK-
1.50E-07	100.0%		S-MVCSOPENS-MVMO2112-S-PPCSBREAK-
1.50E-07	100.0%		S-MVCSOPENS-MVMO2124-S-PPCSBREAK-
1.50E-07	100.0%		S-MVCSOPENS-MVMO2132-S-PPCSBREAK-
9.27E-08	100.0%		ISAV4300-24NISAV4301-24NISPPWWEXH90L
9.27E-08	100.0%		ISAV4302-24NISAV4303-24NISPPDWSPL90L
9.27E-08	100.0%		ISAV4306-24NISAV4307-24NISPPDWEXH90L
9.27E-08	100.0%		ISAV4306-24NISAV4308-24NISPPWWSPL90L
9.27E-08	100.0%		ISAV4311-24NISAV4312-24NISPPDWSPL90L
9.27E-08	100.0%		ISAV4311-24NISAV4313-24NISPPWWEXH90L
9.00E-08	100.0%		ISAV3704-24NISAV3705-24NISPPDWFLR24L
9.00E-08	100.0%		ISAV3728-24NISAV3729-24NISPPDWEQP24L
3.00E-08	100.0%		ISMVMO4841AISPUMPSEAL ISRBCCWPIPE
3.00E-08	100.0%		ISMVMO4841EISPUMPSEAL ISRBCCWPIPE
2.90E-08	100.0%		JAVFF-CCF10-
2.90E-08	100.0%		JAVFF-CCF12-
2.90E-08	100.0%		JAVFF-CCF14-
2.90E-08	100.0%		JAVFF-CCF16-
2.90E-08	100.0%		JAVFF-CCF2--
2.90E-08	100.0%		JAVFF-CCF3--
2.90E-08	100.0%		JAVFF-CCF6--
2.90E-08	100.0%		JAVFF-CCF8--

Probability	%	Class	Inputs
2.40E-08	100.0%		H-MVCC38/39IH-PPHPIBREAK
2.40E-08	100.0%		H-MVCC90A/BH-PPHPIBREAK
2.40E-08	100.0%		ISMVCCFRW(ISRWCUPIPE
2.40E-08	100.0%		L-MVCCTOR-/L-PPRHRBREAK
2.40E-08	100.0%		L-MVCCTOR-EL-PPRHRBREAK
2.40E-08	100.0%		R-MVCC00/01IR-PPRCIBREAK
2.40E-08	100.0%		S-MVCC00/47IS-PPCSBREAK-
2.40E-08	100.0%		S-MVCC20/46IS-PPCSBREAK-
1.80E-08	100.0%		ISCV4329-24CNOINSTRUMAIR
1.80E-08	100.0%		ISCV4329-24CP1NODIV1
1.80E-08	100.0%		ISCV4330-24CNOINSTRUMAIR
1.80E-08	100.0%		ISCV4330-24CP1NODIV2
1.20E-08	100.0%		H-MVCC21/22IH-MVHPIOPEIH-PPHPIBREAK
1.20E-08	100.0%		L-MVCC05/06IL-MVRHROPEL-PPRHRBREAK
1.20E-08	100.0%		L-MVCC05/07IL-MVRHROPEL-PPRHRBREAK
1.20E-08	100.0%		L-MVCC32/33IL-MVRHROPEL-PPRHRBREAK
1.20E-08	100.0%		L-MVCC32/34IL-MVRHROPEL-PPRHRBREAK
1.20E-08	100.0%		L-MVCCFDWSL-MVRHROPEL-PPRHRBREAK
1.20E-08	100.0%		R-MVCC16/17IR-MVRCIOPEIR-PPRCIBREAK
1.03E-08	100.0%		ISHUDWEXH2ISOSIGNFAILSISPPDWSPL90L
1.03E-08	100.0%		ISHUDWSPL2ISOSIGNFAILSISPPDWEXH90L
1.03E-08	100.0%		ISHUN2DW-24ISOSIGNFAILSISPPDWSPL90L
1.03E-08	100.0%		ISHUN2TOR24ISOSIGNFAILSISPPWWEXH90L
1.03E-08	100.0%		ISHUWWEXH2ISOSIGNFAILSISPPWWEXH90L
1.03E-08	100.0%		ISHUWWSPL2ISOSIGNFAILSISPPWWSPL90L
1.00E-08	100.0%		ISHUDWEQP2ISOSIGNFAILSISPPDWEQP24L
1.00E-08	100.0%		ISHUDWFLR2ISOSIGNFAILSISPPDWFLR24L
6.48E-09	100.0%		ISAV4304-24CISCV4329-24O
6.48E-09	100.0%		ISAV4305-240ISCV4330-24O
2.40E-09	100.0%		ISDRAINPIPE ISMVCCFDRNISMVVLVOPEN
9.00E-10	100.0%		H-MVMO2238-H-MVMO2239-H-PPHPIBREAK
9.00E-10	100.0%		H-MVMO2290/H-MVMO2290IH-PPHPIBREAK
9.00E-10	100.0%		ISMVMO2700-ISMVMO2701-ISRWCUPIPE
9.00E-10	100.0%		L-MVMO1913-L-MVMO1989-L-PPRHRBREAK
9.00E-10	100.0%		L-MVMO1921-L-MVMO1989-L-PPRHRBREAK
9.00E-10	100.0%		L-MVMO2012-L-MVMO2069-L-PPRHRBREAK
9.00E-10	100.0%		L-MVMO2015-L-MVMO2069-L-PPRHRBREAK
9.00E-10	100.0%		R-MVMO2400-R-MVMO2401-R-PPRCIBREAK
9.00E-10	100.0%		S-MVMO2100-S-MVMO2147-S-PPCSBREAK-
9.00E-10	100.0%		S-MVMO2120-S-MVMO2146-S-PPCSBREAK-
4.50E-10	100.0%		H-MVHPIOPEIH-MVMO2321-H-MVMO2322-H-PPHPIBREAK
4.50E-10	100.0%		L-MVMO1902-L-MVMO1903-L-MVRHROPEL-PPRHRBREAK
4.50E-10	100.0%		L-MVMO1932-L-MVMO1933-L-MVRHROPEL-PPRHRBREAK
4.50E-10	100.0%		L-MVMO1932-L-MVMO1934-L-MVRHROPEL-PPRHRBREAK
4.50E-10	100.0%		L-MVMO2000-L-MVMO2001-L-MVRHROPEL-PPRHRBREAK

Probability	%	Class	Inputs
4.50E-10	100.0%		L-MVMO2005-L-MVMO2006-L-MVRHROPEL-PPRHRBREAK
4.50E-10	100.0%		L-MVMO2005-L-MVMO2007-L-MVRHROPEL-PPRHRBREAK
4.50E-10	100.0%		R-MVMO2516-R-MVMO2517-R-MVRCIOPEIR-PPRCIBREAK
2.40E-10	100.0%		L-MV1908OPNL-MVCCSDC--L-PPRHRBREAK
9.00E-11	100.0%		ISDRAINPIPE ISMVMO4423-ISMVMO4424-ISMVVLVOPEN
3.00E-11	100.0%		ISCVV270011IISMVMO2740-ISRWCUPIPE
1.50E-11	100.0%		H-CVV230049IH-MVHPIOPEIH-MVMO2312-H-PPHPIBREAK
1.50E-11	100.0%		L-CVV190149IL-MVMO1905-L-MVRHROPEL-PPRHRBREAK
1.50E-11	100.0%		L-CVV20082-NL-MVMO2003-L-MVRHROPEL-PPRHRBREAK
1.50E-11	100.0%		R-CVV250036IR-MVMO2512-R-MVRCIOPEIR-PPRCIBREAK
1.50E-11	100.0%		S-CVV210072IS-MVCSOPENS-MVMO2117-S-PPCSBREAK-
1.50E-11	100.0%		S-CVV210073IS-MVCSOPENS-MVMO2137-S-PPCSBREAK-
9.00E-12	100.0%		L-MV1908OPNL-MVMO1908-L-MVMO1909-L-PPRHRBREAK
3.06E-14	100.0%		ISAV3704-24CISAV3705-24O
3.06E-14	100.0%		ISAV3728-24CISAV3729-24O
3.06E-14	100.0%		ISAV4300-24CISAV4301-24O
3.06E-14	100.0%		ISAV4302-24CISAV4303-24O
3.06E-14	100.0%		ISAV4306-24CISAV4307-24O
3.06E-14	100.0%		ISAV4306-24CISAV4308-24O
3.06E-14	100.0%		ISAV4311-24CISAV4312-24O
3.06E-14	100.0%		ISAV4311-24CISAV4313-24O
0.00E+00	100.0%		FLAG16-L2 ISCV4330-24O
0.00E+00	100.0%		FLAG20-L2 ISCV4329-24O
0.00E+00	100.0%		FLAG20-L2 ISCV4330-24O
0.00E+00	100.0%		FLAG30-L2
0.00E+00	100.0%		FLAG42-L2
0.00E+00	100.0%		FLAG60-L2
0.00E+00	100.0%		FLAG61-L2
0.00E+00	100.0%		FLAG62-L2
0.00E+00	100.0%		FLAG69-L2 ISCV4329-24O

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DUANE ARNOLD ENERGY CENTER

SENSITIVITY CALCULATION FOR THE ILRT EXTENSION RISK ASSESSMENT

OVERVIEW

An analysis was recently completed for the Duane Arnold Energy Center (DAEC) to provide a risk impact assessment of obtaining a one-time extension of the Integrated Leakage Rate Test (ILRT) interval from its current one-in-ten year requirement to a one-in-fifteen year requirement [1]. The analysis followed the methodology that was utilized by Entergy in their successful ILRT extension request for their Indian Point 3 (IP3) plant [2,3].

Subsequently, a report has been issued from NEI providing interim guidance for performing risk assessments in support of one-time extensions for containment Integrated Leakage Rate Test (ILRT) surveillance intervals [4]. This guidance was developed following a review of various utility submittals (including the IP3 submittal) to provide guidance for other utilities wishing to obtain an extension to their required test interval.

Both the IP3 methodology and the NEI guidance document present the risk assessment results in terms of the release scenario types that had previously been developed by EPRI [5]. The table below summarizes the treatment of each of the EPRI Release Scenario Types performed in the DAEC ILRT extension risk assessment [1] using the previously approved IP3 methodology [2], and provides a comparison with the approach from the NEI Interim Guidance document [4].

Table 1
Treatment of EPRI Release Types in the ILRT Extension Assessments

Release Type (1)	Description	IP3 Methodology [2]	NEI Methodology [4]
1	No Containment Failure	Frequency is reduced as Type 3 releases increase. Assumed to be characterized by the Tech Spec allowable leakage, $1L_a$.	Frequency is reduced as Type 3 releases increase. Assumed to be characterized by the Tech Spec allowable leakage, $1L_a$.
2	Large Isolation Failures (Failure to Close)	Not affected by ILRT leak testing frequency.	Not affected by ILRT leak testing frequencies.
3	Pre-existing leaks from containment structure or liner	Categorized as Release Types 3a ("Small", $\sim 10L_a$, non-LERF) and 3b ("Large", $\sim 35L_a$, assumed to be LERF). Small and Large Failure probabilities developed from 95th percentile of the χ^2 distribution of data from NUREG-1493 [6]. (0.064 and 0.021, respectively)	Categorized as Release Types 3a ("Small", $\sim 10L_a$, non-LERF) and 3b ("Large", $\sim 35L_a$, assumed to be LERF). Small and Large Failure probabilities developed from non-informative prior distribution of data from NUREG-1493 supplemented with 1 failure in 38 additional data points. (0.027 and 0.0027, respectively)

Table 1
Treatment of EPRI Release Types in the ILRT Extension Assessments

Release Type [1]	Description	IP3 Methodology [2]	NEI Methodology [4]
4	Type B tested components fail to seal	Not affected by ILRT leak testing frequency.	Not affected by ILRT leak testing frequency.
5	Type C tested components fail to seal	Not affected by ILRT leak testing frequency.	Not affected by ILRT leak testing frequency.
6	Other Isolation Failures	Not affected by ILRT leak testing frequency.	Not affected by ILRT leak testing frequency.
7	Failures Induced by Phenomena (Ealry and Late)	Not affected by ILRT leak testing frequency.	Not affected by ILRT leak testing frequency.
8	Bypass	Characterized by bypass scenarios – not impacted by ILRT extension	Characterized by bypass scenarios – not impacted by ILRT extension

[1] i.e., the EPRI TR-104285 Containment Response Class

As can be seen in Table 1, the major difference in the development of the consequence measures involves the assumptions used in the determination of the failure probability of the EPRI Class 3a and 3b scenarios. The IP3 methodology utilized baseline values of 0.064 and 0.021, respectively whereas the NEI guidance recommends baseline values of 0.027 and 0.0027, respectively, for the baseline EPRI Class 3a and 3b failure probabilities.

The second major difference in the methodologies then arises from the assumptions used to estimate the increases in the failure probabilities of the EPRI Class 3a and 3b scenarios resulting from the ILRT extension. Without reproducing all of the details here, the IP3 methodology estimates a 10% increase in the failure probabilities from extending the ILRT interval from 3-in-10 to 1-in-10 years, and a 15% increase in the failure probabilities from extending the ILRT interval from 3-in-10 to 1-in-15 years. On the other hand, the NEI guidance has interpreted the interval extensions to result in a factor of 3.33 increase in the failure probabilities as the interval goes from 3-in-10 to 1-in-10 years, and to a factor of 5.0 increase as the interval goes from 3-in-10 to 1-in-15 years.

Each of the methodologies can then be used to perform the risk assessment of extending the ILRT interval. These results are summarized below.

RESULTS

Table 2 provides the baseline consequence measures using the DAEC data in the manner utilized in the IP3 submittal, and also shows the revised results using the DAEC data with the NEI methodology. As expected, the baseline consequence measures are lower using the NEI method as compared to the IP3 method since the baseline EPRI Class 3a and 3b probabilities are smaller using the NEI methodology.

Table 2
Base Case Mean Frequencies and Consequence Measures

Release Type	Description	DAEC [1] Person-Rem (50 miles)	IP3 Methodology [2]		NEI Methodology [4]	
			Frequency (per Rx-yr)	Person-Rem/yr (50 miles)	Frequency (per Rx-yr)	Person-Rem/yr (50 miles)
1	No Containment Failure	4.15E+03	*1.61E-06*	*6.68E-03*	*2.26E-6*	*9.39E-03*
2	Large Isolation Failures (Fail to Close)	6.24E+05	4.33E-08	2.70E-02	4.33E-08	2.70E-02
3a	Small Isolation Failures (liner breach)	4.15E+04	*7.55E-07*	*3.13E-02*	*3.19E-07*	*1.32E-2*
3b	Large Isolation Failures (liner breach)	1.45E+05	*2.48E-07*	*3.60E-02*	*3.19E-08*	*4.63E-03*
4	Small Isolation Failures (to seal - Type B)	NA	NA	NA	NA	NA
5	Small Isolation Failures (to seal - Type C)	NA	NA	NA	NA	NA
6	Other Isolation Failures	6.24E+05	4.72E-09	2.94E-03	4.72E-9	2.94E-03
7	Failures Induced by Phenomena	4.05E+05	9.05E-06	3.59E+00	9.05E-6	3.59E+00
8	Bypass (ISLOCA)	6.24E+05	8.97E-08	5.59E-02	8.97E-8	5.59E-02
CDF	**All CET End states**		**1.18E-05**	***3.752***	**1.18E-05**	***3.705***

Table 3 provides the revised calculated consequence measures for extending the interval from 3-in-10 years to 1-in-10 years using the two methodologies. As noted previously, the IP3 methodology assumes that a 10% increase in the 3a and 3b frequencies occurs, whereas the NEI methodology assumes that a factor of 3.33 increase would occur.

Table 3
Mean Frequencies and Consequence Measures
(ILRT Interval Set to Once Every Ten Years)

Release Type	Description	DAEC [1] Person-Rem (50 miles)	IP3 Methodology [2]		NEI Methodology [4]	
			Frequency (per Rx-yr)	Person-Rem/yr (50 miles)	Frequency (per Rx-yr)	Person-Rem/yr (50 miles)
1	No Containment Failure	4.15E+03	*1.51E-06*	*6.26E-03*	*1.45E-06*	*6.01E-03*
2	Large Isolation Failures (Fail to Close)	6.24E+05	4.33E-08	2.70E-02	4.33E-08	2.70E-02
3a	Small Isolation Failures (liner breach)	4.15E+04	*8.31E-07*	*3.45E-02*	*1.06E-06*	*4.41E-2*
3b	Large Isolation Failures (liner breach)	1.45E+05	*2.73E-07*	*3.96E-02*	*1.06E-07*	*1.54E-02*
4	Small Isolation Failures (to seal - Type B)	NA	NA	NA	NA	NA
5	Small Isolation Failures (to seal - Type C)	NA	NA	NA	NA	NA
6	Other Isolation Failures	6.24E+05	4.72E-09	2.94E-03	4.72E-9	2.94E-03
7	Failures Induced by Phenomena	4.05E+05	9.05E-06	3.59E+00	9.05E-6	3.59E+00
8	Bypass (ISLOCA)	6.24E+05	8.97E-08	5.59E-02	8.97E-8	5.59E-02
CDF	**All CET End states**		**1.18E-05**	***3.758***	**1.18E-05**	***3.743***

Table 4 provides the revised calculated consequence measures for extending the interval from 3-in-10 years to 1-in-15 years using the two methodologies. As noted previously, the IP3 methodology assumes that a 15% increase in the 3a and 3b frequencies occurs, whereas the NEI methodology assumes that a factor of 5.0 increase would occur.

Table 4
Mean Frequencies and Consequence Measures
(ILRT Interval Set to Once Every Fifteen Years)

Release Type	Description	DAEC [1] Person-Rem (50 miles)	IP3 Methodology [2] Frequency (per Rx-yr)	IP3 Methodology [2] Person-Rem/yr (50 miles)	NEI Methodology [4] Frequency (per Rx-yr)	NEI Methodology [4] Person-Rem/yr (50 miles)
1	No Containment Failure	4.15E+03	*1.46E-06*	*6.06E-03*	*8.60E-07*	*3.57E-03*
2	Large Isolation Failures (Fail to Close)	6.24E+05	4.33E-08	2.70E-02	4.33E-08	2.70E-02
3a	Small Isolation Failures (liner breach)	4.15E+04	*8.68E-07*	*3.60E-02*	*1.59E-06*	*6.61E-2*
3b	Large Isolation Failures (liner breach)	1.45E+05	*2.85E-07*	*4.14E-02*	*1.59E-07*	*2.31E-02*
4	Small Isolation Failures (to seal - Type B)	NA	NA	NA	NA	NA
5	Small Isolation Failures (to seal - Type C)	NA	NA	NA	NA	NA
6	Other Isolation Failures	6.24E+05	4.72E-09	2.94E-03	4.72E-9	2.94E-03
7	Failures Induced by Phenomena	4.05E+05	9.05E-06	3.59E+00	9.05E-6	3.59E+00
8	Bypass (ISLOCA)	6.24E+05	8.97E-08	5.59E-02	8.97E-8	5.59E-02
CDF	**All CET End states**		**1.18E-05**	***3.761***	**1.18E-05**	***3.770***

SUMMARY AND CONCLUSIONS

This analysis provides a comparison of the results obtained for the risk impact assessment of extending the ILRT interval based on using the IP3 Methodology and the recently released NEI interim guidance methodology. The key figures of merit for this analysis are reported in Table 5

Table 5
SUMMARY OF DAEC RISK IMPACT ON TYPE A ILRT TEST FREQUENCY USING THE IP3 METHODOLOGY AND THE NEI METHODOLOGY

Figure Of Merit	Risk Impact (Base)		Risk Impact (10-years)		Risk Impact (15-years)	
	IP3	NEI	IP3	NEI	IP3	NEI
Release Type 3a and 3b Frequency	1.00E-6	3.51E-7	1.10E-6	1.17E-6	1.15E-6	1.75E-6
Change in Type 3a and 3b Frequency (from 3-in-10 years)	-	-	1.00E-7	8.19E-7	1.50E-7	1.40E-6
Change in Type 3a and 3b Frequency (from 1-in-10 years)	-	-	-	-	5.00E-8	5.80E-7
Release Type 3b Frequency	2.48E-7	3.19E-8	2.73E-7	1.06E-7	2.85E-7	1.59E-7
Change in Release Type 3b (LERF, from 3-in-10 years)	-	-	2.50E-8	7.41E-8	3.70E-8	1.27E-7
Change in Release Type 3b (LERF, from 1-in-10 years)	-	-	-	-	1.20E-8	5.30E-8
Total Integrated Risk (Person-rem/yr)	3.752	3.705	3.758	3.743	3.761	3.770
Change in Integrated Risk (Person-rem/yr, from 3-in-10 years)	-	-	0.006	0.038	0.009	0.065
Change in Integrated Risk (Person-rem/yr, from 1-in-10 years)	-	-	-	-	0.003	0.027
Change in CCFP (1) (from 3-in-10 years)	-	-	0.8%	6.9%	1.3%	11.9%
Change in CCFP (1) (from 1-in-10 years)	-	-	-	-	0.4%	4.9%

(1) CCFP = Conditional Containment Failure Probability. Conservatively characterized from the change in the Type 3a and 3b frequencies divided by the core damage frequency of 1.18E-5.

If the approach from the NEI methodology for the failure probabilities is used instead of the IP3 methodology values used by DAEC, a slightly different measured potential impact on LERF, population dose, and CCFP from the proposed ILRT extension is calculated compared to the original analysis, but it does not change the conclusions. The results from the original submittal and from the additional sensitivity case explored here lead to the conclusion that the ILRT extension is of low risk significance based on the LERF criteria in Regulatory Guide 1.174, and based on relatively small changes to the other figures of merit.

REFERENCES

[1] *Risk Assessment For Duane Arnold Energy Center Regarding ILRT (Type A) Extension Request,* ERIN P0124010003-1875, December 2001.

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[5] *Risk Impact Assessment of Revised Containment Leak Rate Testing Intervals,* EPRI, Palo Alto, CA EPRI TR-104285, August 1994.

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