Exelon Nuclear

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10 CFR 50.90

RS-02-164

October 24, 2002

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

LaSalle County Station, Units 1 and 2
Facility Operating License Nos. NPF-11 and NPF-18
NRC Docket Nos. 50-373 and 50-374

Subject: Request for Amendment to Technical Specification 5.5.13, "Primary Containment Leakage Rate Testing Program"

In accordance with 10 CFR 50.90, Exelon Generation Company, LLC, (EGC) hereby requests the following amendment to Appendix A, Technical Specifications (TS), of Facility Operating License Nos. NPF-11 and NPF-18. Specifically, the proposed changes will revise TS 5.5.13, "Primary Containment Leakage Rate Testing Program" to reflect a one-time deferral of the primary containment Type A test to no later than June 13, 2009 for Unit 1 and no later than December 7, 2008 for Unit 2.

TS Section 5.5.13 establishes the leakage rate testing of the primary containments as required by 10 CFR 50.54(o) and 10 CFR 50, Appendix J, Option B, "Performance Based Requirements," as modified by approved exemptions. Additionally, the testing conforms with the guidelines contained in Regulatory Guide 1.163, "Performance-Based Containment Leak-Testing Program," dated September 1995.

EGC is requesting this one-time amendment in anticipation of a rule change to 10 CFR 50 extending the Type A testing frequency to at least 15 years. Approval of the proposed changes will allow sufficient time for this rule change to be processed and incorporated into LaSalle County Station TS.

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

Attachment 1 is the notarized affidavit.
Attachment 2 provides our evaluation supporting the proposed changes.
Attachment 3 contains the copy of the marked up TS page.
Attachment 4 provides the retyped TS page.
Attachment 5 provides the risk assessment supporting the proposed changes.

A017

The proposed TS changes have been reviewed by the LaSalle County Station Plant Operations Review Committee (PORC) and approved by the Nuclear Safety Review Board (NSRB) in accordance with the Quality Assurance Program.

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

We request approval of the proposed changes by October 1, 2003 with an implementation period of 60 days.

Should you have any questions concerning this submittal, please contact Mr. T. W. Simpkin at (630) 657-2821.

Sincerely,

T.W. Simpkin

for Keith R. Jury
Director-Licensing
Mid-West Regional Operating Group

Attachments:

Attachment 1. Affidavit
Attachment 2. Evaluation of Proposed Changes
Attachment 3. Markup of Proposed Technical Specification Page Changes
Attachment 4. Retyped Page for Technical Specification Changes
Attachment 5. Risk Assessment for LaSalle to Support ILRT (Type A) Interval Extension Request

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

ATTACHMENT 1
Affidavit

STATE OF ILLINOIS	)	
COUNTY OF DUPAGE	)	
IN THE MATTER OF:	)	
EXELON GENERATION COMPANY, LLC	)	**Docket Numbers**
LASALLE COUNTY STATION - UNIT 1 and UNIT 2	)	**50-373 and 50-374**

SUBJECT: **Request for Amendment to Technical Specification 5.5.13, "Primary Containment Leakage Rate Testing Program"**

AFFIDAVIT

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

T. W. Simpkin
T. W. Simpkin
Manager-Licensing
Mid-West Regional Operating Group

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

for the State above named, this 24th day of

October, 2002

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

Timothy A. Byam
Notary Public

1.0 INTRODUCTION

2.0 DESCRIPTION OF PROPOSED AMENDMENT

3.0 BACKGROUND

4.0 REGULATORY REQUIREMENTS & GUIDANCE

5.0 TECHNICAL ANALYSIS

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7.0 NO SIGNIFICANT HAZARDS CONSIDERATION (NSHC)

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1.0 INTRODUCTION

In accordance with 10 CFR 50.90, Exelon Generation Company, LLC, (EGC) hereby requests the following amendment to Appendix A, Technical Specifications (TS), of Facility Operating License Nos. NPF-11 and NPF-18. Specifically, the proposed changes will revise TS 5.5.13, "Primary Containment Leakage Rate Testing Program" to reflect a one-time deferral of the primary containment Type A test to no later than June 13, 2009 for Unit 1 and no later than December 7, 2008 for Unit 2.

EGC is requesting this one-time amendment in anticipation of a rule change to 10 CFR 50 extending the Type A testing frequency to at least 15 years. Approval of the proposed changes will allow sufficient time for this rule change to be processed and incorporated into LaSalle County Station TS.

2.0 DESCRIPTION OF PROPOSED AMENDMENT

The proposed changes add two new exceptions to TS 5.5.13 that modify the schedule for the next Type A test for LaSalle County Station, Units 1 and 2, to a 15-year interval The proposed wording associated with the exceptions to be added to TS 5.5.13 are identified below in bold type.

5.5.13 Primary Containment Leakage Rate Testing Program

a. This program shall establish 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-Testing Program," dated September 1995 **as modified by the following exceptions:**

1. **NEI 94-01 – 1995, Section 9.2.3: The first Unit 1 Type A test performed after June 14, 1994 Type A test shall be performed no later than June 13, 2009.**

2. **NEI 94-01 – 1995, Section 9.2.3: The first Unit 2 Type A test performed after December 8, 1993 Type A test shall be performed no later than December 7, 2008.**

3.0 BACKGROUND

LaSalle County Station, Units 1 and 2, are General Electric BWR/5 plants with Mark II primary containments. The Mark II primary containment consists of two compartments, the drywell and the suppression chamber. The drywell has the shape of a truncated cone, and is located above the cylindrically shaped suppression chamber. The drywell floor separates the drywell and the suppression chamber. The primary containment is penetrated by access, piping and electrical penetrations.

The integrity of the primary containment penetrations and isolation valves is verified through Type B and Type C local leak rate tests (LLRTs) and the overall leak tight integrity of the primary containment is verified by a Type A integrated leak rate test (ILRT) as required by 10 CFR 50, Appendix J, "Primary Reactor Containment Leakage Testing for Water-Cooled Power Reactors." These tests are performed to verify the essentially leak tight characteristics of the primary containment at the design basis accident pressure. The last Type A test for LaSalle County Station Unit 1 was June 14, 1994 and Unit 2 was December 8, 1993. The proposed changes will require the next Type A test for Unit 1 to be performed by June 13, 2009 and for Unit 2 to be performed by December 7, 2008.

Option B, "Performance Based Requirements," of Appendix J to 10 CFR 50 requires that a Type A test be conducted at a periodic interval based on historical performance of the overall primary containment system. LaSalle County Station TS 5.5.13 requires that a program be established to comply with the primary containment leakage rate testing requirements of 10 CFR 50.54(o) and 10 CFR 50, Appendix J, Option B, as modified by exemptions. Additionally, this program is in accordance with the guidelines contained in Regulatory Guide (RG) 1.163, "Performance-Based Containment Leak-Testing Program," dated September 1995. RG 1.163 endorses, with certain exceptions, Nuclear Energy Institute (NEI) 94-01, Revision 0., "Industry Guideline for Implementing Performance-Based Option of 10 CFR 50, Appendix J," dated July 26, 1995.

NEI 94-01 specifies for Type A tests, an initial test interval of 48 months and allows an extension of the interval to 10 years based on two consecutive successful tests. LaSalle County Station, Units 1 and 2 are currently on 10-year intervals.

The proposed changes add two exceptions to TS 5.5.13 to allow a one-time deferral from the guidelines contained in RG 1.163 and NEI 94-01 regarding the Type A test interval. The proposed changes will extend the next Type A test for Units 1 and 2 to a 15-year interval.

4.0 REGULATORY REQUIREMENTS & GUIDANCE

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

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

5.0 TECHNICAL ANALYSIS

5.1 Primary Containment Pressure Suppression Testing

The function of the primary containment is to isolate and contain fission products released from the Primary Coolant System (PCS) following a design basis Loss of Coolant Accident (LOCA) and to confine the postulated release of radioactive material to within limits. The primary containment incorporates a drywell section and a suppression chamber section. The drywell is located over the suppression chamber and is separated by the drywell floor. The suppression chamber contains a pool of water. The drywell floor is penetrated by downcomers, penetrations, and safety/relief valve (SRV) discharge lines. The downcomers originate in the drywell air space and terminate below the water level of the suppression chamber pool of water. The SRV discharge lines originate at the SRVs located on the steam lines within the drywell and terminate below the water level of the suppression chamber pool of water. The floor penetrations have blind flanges installed during plant operation.

The Suppression Chamber-Drywell Vacuum Breakers are vacuum relief valves that are located outside the primary containment in special piping and form an extension of the primary containment boundary. The vacuum breakers connect the drywell airspace and suppression chamber airspace to prevent exceeding the drywell floor negative differential design pressure and backflooding of the suppression pool water into the drywell.

During a LOCA, the downcomers direct steam from the drywell airspace to below the water level of the suppression chamber pool of water to condense the steam and thus, limit the containment pressure response. Steam that enters the suppression chamber air space directly from the drywell airspace will bypass the condensing capabilities of the suppression chamber pool of water, thereby causing a higher containment pressure response. The Drywell-to-Suppression Chamber bypass leakage test verifies that the total bypass leakage between the drywell airspace and suppression chamber airspace is consistent with analysis assumptions.

In an amendment dated November 7, 2001, the NRC approved TS revisions to the scheduling of the drywell to suppression chamber bypass test and the suppression chamber to drywell vacuum breaker leakage testing. The amendments require the drywell to suppression chamber bypass test to be conducted on a 10-year frequency and the drywell to suppression chamber to drywell vacuum breaker leakage tests to be conducted on a 24-month frequency. The drywell to suppression chamber bypass tests for Unit 1 was recently successfully conducted on November 11, 1999 and Unit 2 on November 10, 2000. The next drywell to suppression chamber bypass test for Unit 1 is required by November 2009 and Unit 2 by November 2010.

The proposed changes do not modify either of these test frequencies as the next required testing of the drywell to suppression chamber bypass test is consistent with the proposed changes and the suppression chamber to drywell vacuum breaker test is conducted independently of the Type A primary containment test. Additionally, the proposed changes do not modify the acceptance criteria of either of these tests.

Therefore, the proposed changes do not modify the current test frequencies or test acceptance criteria of the primary containment pressure suppression components and systems.

5.2 10CFR 50, Appendix J, Option B

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

10 CFR 50, Appendix J was revised, effective October 26, 1995, to allow licensees to choose primary containment leakage testing under Option A "Prescriptive Requirements" or Option B. Amendments Nos. 110 and 95 for Units 1 and 2, respectively, were issued to permit implementation of 10 CFR 50, Appendix J, Option B. TS 5.5.13 currently requires the establishment of a Primary Containment Leakage Testing Program in accordance with 10 CFR 50.54(o) and 10 CFR 50, Appendix J, Option B, as modified by approved exemptions. This program implements the guidelines contained in RG 1.163 which specifies a method acceptable to the NRC for complying with Option B by approving the use of NEI 94-01, subject to several regulatory positions stated in the RG.

10 CFR 50, Appendix J, Section V. B specifies that the regulatory guide or other implementing documents used to develop a performance-based leakage testing program must be included, by general reference, in the plant's TS. Additionally, deviations from guidelines endorsed in a regulatory guide are to be submitted as a revision to the plant's TS. Therefore, this application does not require an exemption from 10CFR 50, Appendix J, Option B.

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

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

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

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

The required surveillance frequency for Type A testing in NEI 94-01 is at least once per ten years based on an acceptable performance history (i.e., two consecutive periodic Type A tests at least 24 months apart or refueling cycles where the calculated performance leakage rate was less than 1.0 L_a) and consideration of the performance factors in NEI 94-01, Section 11.3. The proposed changes are requesting a one-time amendment in anticipation of a rule change to 10 CFR 50 extending the Type A test frequency to at least 15 years.

5.3 Integrated Leak Rate History

Type A testing is performed to verify the integrity of the containment structure in its Loss of Coolant Accident (LOCA) configuration. Industry test experience has demonstrated that Type B & C testing detect a large percentage of containment leakages and that the percentage of containment leakages detected only by integrated containment leakage testing is very small. Results of previous ILRT's, presented below, demonstrate both containment structures remain essentially leak tight barriers and represents minimal risk to increased leakage. These plant specific results support the conclusions of NUREG-1493.

10 CFR 50 Appendix J, Option B Test Information

Unit One

Test Date	Total Leakage (Note 1)	Acceptance Limit (Note 1)
06/14/94	0.2355%	0.634%
01/14/93	0.4243%	0.634%
12/23/89	0.3200%	0.634%
06/04/86	0.3107%	0.634%
05/14/82	0.3933%	0.634%

Unit Two

Test Date	Total Leakage (Note 1)	Acceptance Limit (Note 1)
12/08/93	0.3794%	0.634%
03/28/92	0.3760%	0.634%
06/03/90	0.5042%	0.634%
06/01/87	0.5395%	0.634%
06/24/83	0.2309%	0.634%

Note 1: Leakage rates are expressed in units of containment air weight percent per day at test pressure equal to the calculated peak containment internal pressure related to the DBA 39.6 psig (Pa). Calculated results are expressed at a 95% confidence level plus leakage attributed to non-vented penetrations. The maximum allowable primary containment leakage rate allowed by Option B during containment leak rate testing is 0.634% containment air weight percent per day ($1.0L_a$).

5.4 Type B and C Testing

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

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

5.5 Containment Inspections

a) Appendix J Visual Inspections

The Appendix J Program performs visual inspections of accessible interior and exterior surfaces of the containment system for structural problems which may affect either the containment structural leakage integrity or performance of the Type A Test. These examinations are conducted prior to initiating a Type A test, and during two refueling outages before the next Type A test based on a ten-year frequency.

The inspection requirements and ten-year frequency will not be changed as a result of the proposed changes.

b) Containment Inservice Inspection Program

A comprehensive primary containment inspection is performed to the requirements of American Society of Mechanical Engineers (ASME) Section XI, "Inservice Inspection," Subsections IWE, "Requirements for Class MC and Metallic Liners of Class CC Components of Light-Water Cooled Power Plants," and Subsection IWL, "Requirements of Class CC Concrete Components of Light-Water Cooled Power Plants." The Containment Inservice Inspection Program (CISI) was established in 1996 and the initial inspections were completed for both units by September 2001. The containment components subject to inspection are associated with the leak tight barrier including integral attachments and structural integrity. The program also inspects the Class MC pressure retaining components, including metallic shell and penetration liners of Class CC pressure retaining components and their integral attachments. The current inspection plan was developed in accordance with the requirements of the 1992 Edition with the 1992 Addenda of the ASME Boiler and Pressure Vessel Code, Section XI, Division 1, Subsections IWE and IWL, as modified by NRC final rulemaking to 10 CFR 50.55a published in the Federal Register on August 8, 1996. Future CISI inspections will be performed to the 1998 Edition of the ASME Code Section XI, Subsections IWE and IWL as modified by approved NRC relief requests.

The initial inspections of the Unit 1 and 2 Metal / Concrete Containment have been completed. Various indications were observed, documented, evaluated and determined to be acceptable. The inspections identified that no areas of the containment liner surfaces require augmented examination and no loss of structural integrity of the primary containments were observed.

There will be no change to the schedule for these inspections as a result of the proposed changes.

c) Coatings Inspections

The containment coatings inspection program was developed in accordance with the requirements of the 1998 ASME Code Edition of Subsection IWE and IWL as supplemented by specific details contained in the CISI. The inspection results for Unit 1 and Unit 2, performed in November 1999 and November 2000 respectively, found the containment coatings to be in good condition with no observed extensive coating indications. The inspections did identify some minor physical damage on various containment liners and other surfaces. The damage was characterized as small chips in the topcoat causing exposure of the primer coating. In areas where this type of indication was observed, the primer is intact, with no rusting of the substrate.

The inspection requirements of the containment coatings program will not be changed as a result of the proposed changes.

d) Maintenance Rule Inspections

Maintenance Rule Baseline Inspections required by 10CFR 50.65, Requirements for monitoring the effectiveness of maintenance at nuclear power plants," were completed in March of 1999. The inspections included the Reactor Buildings and Containment Structures. It was concluded that these structures are being adequately maintained and capable of performing their intended functions. This program ensures that containment structures are evaluated and maintained in conditions to perform their intended functions.

There will be no changes to the Maintenance Rule Program as a result of the proposed changes.

5.6 Information Notice 92-20

NRC Information Notice 92-20,"Inadequate Local Leak Rate Testing," discussed the inadequate local leak rate testing of two-ply stainless steel bellows. LaSalle County Station does not have any bellows that act as a part of the containment.

5.7 Risk Information

The risk analysis performed to support this submittal is contained in Attachment 5. The risk analysis used the LaSalle County Station PRA, Revision 2001a.

The risk analysis determined that the proposed changes result in:

- an insignificant increase in total population dose rate,
- a "very small" increase in the Large Early Release Frequency (LERF) risk measure based on criteria from NRC Regulatory Guide 1.174, "An Approach for Using Probabilistic Risk Assessment in Risk-Informed Decisions on Plant-Specific Changes to the Current Licensing Basis," and
- an insignificant increase in the conditional containment failure probability (CCFP).

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

6.0 REGULATORY ANALYSIS

10 CFR 50.36, "Technical specifications," provides the regulatory requirements for the content required in a licensee's TS. 10 CFR 36(c)(5), "Administrative controls," requires provisions relating to organization and management, procedures, recordkeeping, review and audit, and reporting necessary to assure operation of the facility in a safe manner will be included in a licensee's TS.

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

The proposed changes will revise TS Section 5.5.13 to reflect a one-time deferral from the program requirements for the Type A test for LaSalle County Station Units 1 and 2. The one-time deferral deviates from the guidelines contained in RG 1.163 and NEI 94-01. Thus, the proposed changes are consistent with the requirements of 10 CFR 36(c)(5) and 10 CFR 50, Appendix J, Section V. B.

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

7.0 NO SIGNIFICANT HAZARDS CONSIDERATION

EGC has evaluated the proposed changes to the TS for LaSalle County Station, Unit 1 and Unit 2, and has determined that the proposed changes do not involve a significant hazards consideration and is providing the following information to support a finding of no significant hazards consideration.

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

Response: No

The proposed changes will revise LaSalle County Station, Units 1 and 2, Technical Specification (TS) 5.5.13, "Primary Containment Leakage Rate Testing Program" to reflect a one-time deferral of the primary containment Type A test to no later than June 13, 2009 for Unit 1 and no later than December 7, 2008 for Unit 2. The current Type A test interval of ten years, based on past performance, would be extended on a one-time basis to 15 years from the last Type A test.

The function of the primary containment is to isolate and contain fission products released from the reactor Primary Coolant System (PCS) following a design basis Loss of Coolant Accident (LOCA) and to confine the postulated release of radioactive material to within limits. The test interval associated Type A testing is not a precursor of any accident previously evaluated. Type A testing does provide assurance that the LaSalle County Station primary containments will not exceed allowable leakage rate values specified in the Technical Specifications and will continue to perform their design function following an accident. The risk assessment of the proposed changes has concluded that there is an insignificant increase in total population dose rate and an insignificant increase in the conditional containment failure probability.

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

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

Response: No

The proposed changes for a one-time extension of the Type A tests for LaSalle County Station, Units 1 and 2 will not affect the control parameters governing unit operation or the response of plant equipment to transient and accident conditions. The proposed changes do not introduce any new equipment, modes of system operation or failure mechanisms.

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

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

Response: No

LaSalle County Station, Units 1 and 2, are General Electric BWR/5 plants with Mark II primary containments. The Mark II primary containment consists of two compartments, the drywell and the suppression chamber. The drywell has the shape of a truncated cone, and is located above the cylindrically shaped suppression chamber. The drywell floor separates the drywell and the suppression chamber. The primary containment is penetrated by access, piping and electrical penetrations.

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

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

Based upon the above, EGC concludes that the proposed amendment presents no significant hazards consideration under the standards set forth in 10 CFR50.92(c), and, accordingly, a finding of "no significant hazards consideration" is justified.

8.0 ENVIRONMENTAL CONSIDERATION

A review has determined that the proposed amendment would change a requirement with respect to installation or use of a facility component located within the restricted area, as defined in 10 CFR 20, or would change an inspection or surveillance requirement. However, the proposed amendment does not involve (i) a significant hazards consideration, (ii) a significant change in the types or significant increase in the amounts of any effluent that may be released offsite, or (iii) a significant increase in individual or cumulative occupational radiation exposure. Accordingly, the proposed amendment meets the eligibility criterion for categorical exclusion set forth in 10 CFR 51.22(c)(9). Therefore, pursuant to 10 CFR 51.22(b), no environmental impact statement or environmental assessment need be prepared in connection with the proposed amendment.

9.0 PRECEDENT

The proposed amendment incorporates into the LaSalle County Station changes that are similar to changes approved by the NRC for Susquehanna Steam Electric Station on March 8, 2002 and Seabrook Station on April 11, 2002.

ATTACHMENT 3

MARKUP OF PROPOSED TECHNICAL SPECIFICATION PAGE CHANGES

Revised TS Pages

5.5 Programs and Manuals

5.5.12 Safety Function Determination Program (SFDP) (continued)

b. A loss of safety function exists when, assuming no concurrent single failure, and assuming no concurrent loss of offsite power or loss of onsite diesel generator(s), 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 described in b.1 and b.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. When a loss of safety function is caused by the inoperability of a single Technical Specification support system, the appropriate Conditions and Required Actions to enter are those of the support system.

5.5.13 Primary Containment Leakage Rate Testing Program

a. This program shall establish 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-Testing Program," dated September 1995. INSERT

b. The peak calculated primary containment internal pressure for the design basis loss of coolant accident, P_a, is 39.9 psig.

(continued)

INSERT

as modified by the following exceptions:

1. **NEI 94-01 – 1995, Section 9.2.3: The first Unit 1 Type A test performed after June 14, 1994 Type A test shall be performed no later than June 13, 2009.**

2. **NEI 94-01 – 1995, Section 9.2.3: The first Unit 2 Type A test performed after December 8, 1993 Type A test shall be performed no later than December 7, 2008.**

ATTACHMENT 4

RETYPED PAGE
FOR
TECHNICAL SPECIFICATION CHANGES

<u>Retyped TS Pages</u>

5.5 Programs and Manuals

5.5.12 Safety Function Determination Program (SFDP) (continued)

b. A loss of safety function exists when, assuming no concurrent single failure, and assuming no concurrent loss of offsite power or loss of onsite diesel generator(s), 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 described in b.1 and b.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. When a loss of safety function is caused by the inoperability of a single Technical Specification support system, the appropriate Conditions and Required Actions to enter are those of the support system.

5.5.13 Primary Containment Leakage Rate Testing Program

This program shall establish 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-Testing Program," dated September 1995 as modified by the following exceptions:

1. NEI 94-01 - 1995, Section 9.2.3: The first Unit 1 Type A test performed after June 14, 1994 Type A test shall be performed no later than June 13, 2009.

(continued)

5.5 Programs and Manuals

5.5.13 Primary Containment Leakage Rate Testing Program (continued)

2. NEI 94-01 - 1995, Section 9.2.3: The first Unit 2 Type A test performed after December 8, 1993 Type A test shall be performed no later than December 7, 2008.

b. The peak calculated primary containment internal pressure for the design basis loss of coolant accident, P_a, is 39.9 psig.

c. The maximum allowable primary containment leakage rate, L_a, at P_a, is 0.635% of primary containment air weight per day.

d. Leakage rate acceptance criteria are:

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

2. Air lock testing acceptance criteria are:

a) Overall air lock leakage rate is $\leq 0.05\ L_a$ when tested at $\geq P_a$.

b) For each door, the seal leakage rate is ≤ 5 scf per hour when the gap between the door seals is pressurized to ≥ 10 psig.

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

ATTACHMENT 5

Risk Assessment for LaSalle to Support ILRT (Type A)

Interval Extension Request

ERIN Engineering And Research, Inc

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

ERIN Report No: C4670213-4900

Principal Contributors

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Prepared for

Exelon
LaSalle Nuclear Power Station

July 2002

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

ERIN Report No. C4670213-4900

Prepared by: [illegible] Date: July 8, 2002

Reviewed by: [illegible] Date: 8 July 2002

Approved by: E T Burns Date: 8 July 2002

Accepted by: [illegible] Date: 24 October 2002

Revisions:

Rev.	Description	Preparer/Date	Reviewer/Date	Approver/Date
1	CLARIFICATION OF THE USE OF TABLE A-2	E T Burns 10-23-02	[illegible] 10-23-02	Lawrence Lee 29 October 2002

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Section 1
INTRODUCTION

1.1 PURPOSE

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

1.2 BACKGROUND

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

The basis for the current 10-year test interval is provided in Section 11.0 of NEI 94-01, Revision 0, and was established in 1995 during development of the performance-based Option B to Appendix J. Section 11.0 of NEI 94-01 states that NUREG-1493 [5], "Performance-Based Containment Leak-Test Program," September 1995, provides the technical basis to support rulemaking to revise leakage rate testing requirements contained in Option B to Appendix J. The basis consisted of qualitative and quantitative assessments of the risk impact (in terms of increased public dose) associated with a range of extended leakage rate test intervals. To supplement the NRC's rulemaking basis, NEI undertook a similar study. The results of that study are documented in Electric Power Research Institute (EPRI) Research Project Report TR-104285. [2].

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

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

It should be noted that, in addition to ILRT tests, containment leak-tight integrity is also verified through periodic in-service inspections conducted in accordance with the requirements of the American Society of Mechanical Engineers Boiler and Pressure

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

1.3 CRITERIA

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

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

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

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

In addition, based on the precedent of other ILRT extension requests [6,18,20], the total annual risk (person-rem/yr population dose) is examined to demonstrate the relative change in risk. (No threshold has been established for this parameter change.)

Section 2
METHODOLOGY

This section provides the following methodology related items:

- Brief summary of available resource documents to support the methodology
- NEI Interim Guidance for the analysis approach to be used
- General assumptions used in the evaluation
- Plant-specific inputs

2.1 General Resources Available

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

1) NUREG/CR-3539 [10]
2) NUREG/CR-4220 [11]
3) NUREG-1273 [12]
4) NUREG/CR-4330 [13]
5) EPRI TR-105189 [8]
6) NUREG-1493 [5]
7) EPRI TR-104285 [2]
8) NEI Interim Guidance [3,21]

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.

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

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

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

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

NUREG/CR-4220 assessed the "large" containment leak probability to be in the range of 1E-3 to 1E-2, with 5E-3 identified as the point estimate based on 4 events in 740 reactor years and conservatively assuming a one-year duration for each event. It should be noted that all 4 of the identified large leakage events were PWR events, and the assumption of a one-year duration is not applicable to an inerted containment such as LaSalle. 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 [12]

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

NUREG/CR-4330 [13]

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

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

EPRI TR-105189 [8]

The EPRI study TR-105189 is useful to the ILRT test interval extension risk assessment because this EPRI study provides insight regarding the impact of containment testing on shutdown risk. This study performed a quantitative evaluation (using the EPRI

ORAM software) for two reference plants (a BWR-4 and a PWR) of the impact of extending ILRT and LLRT test intervals on shutdown risk.

The result of the study concluded that a small but measurable safety benefit is realized from extending the test intervals. For the BWR, the benefit from 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. The other 5 events involved loss of RHR and/or SDC due to ILRT/LLRT activities. This information was used in the EPRI study to estimate the safety benefit from reductions in testing frequencies. This represents a valuable insight into the <u>improvement</u> in the safety due to extending the ILRT test interval.

<u>NUREG-1493 [5]</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.

- Given the insensitivity of risk to the containment leak rate and the small fraction of leak paths detected solely by Type A testing, increasing the interval between integrated leak rate tests is possible with minimal impact on public risk.

EPRI TR-104285 [2]

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

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

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

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

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

<u>NEI Interim Guidance [3,21]</u>

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

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

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

This NEI Guidance is used in the LaSalle ILRT risk assessment.

2.2 NEI INTERIM GUIDANCE

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

1. Quantify the baseline (nominal three year ILRT interval) frequency per reactor year for the EPRI accident categories of interest. Note that EPRI categories 4, 5, and 6 are not affected by changes in ILRT test frequency.

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

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

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

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

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

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

8. Evaluate the risk impact in terms of LERF.

9. Evaluate the change in conditional containment failure probability.

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

2.3 ASSUMPTIONS

The following ground rules are used in the analysis:

- LaSalle Unit 2 is used explicitly in this risk assessment. Due to the similarity of Units 1 and 2, the results of this Unit 2 ILRT risk assessment apply to Unit 1, as well.
- Ex-plant consequence performed for LaSalle by Sandia National Laboratories provide representative offsite dose estimates when updated to account for changes (e.g., increase in population) since the 1992 study.
- The use of year 2000 population data is adequate for this analysis. Scaling the year 2000 population data to May 2002 (the date of this report) would not significantly impact the quantitative results, nor would it change the conclusions.
- An evaluation of the risk impact of the ILRT on shutdown risk is addressed using the generic results from EPRI TR-105189 [8].
- Radionuclide release categories are defined consistent with the EPRI TR-104285 methodology. [2]
- Per the NEI Interim Guidance, the representative containment leakage for EPRI Category 1 sequences is 1 L_a (L_a is the Technical Specification maximum allowable containment leakage rate). [3]
- Per the NEI Interim Guidance, the representative containment leakage for EPRI Category 3a sequences is 10 L_a. [3]
- Per the NEI Interim Guidance, the representative containment leakage for EPRI Category 3b sequences is 35 L_a. [3]
- EPRI Category 3b is conservatively categorized as LERF based on the previously approved methodology [3].

- The reduction in ILRT frequency does not impact the reliability of containment isolation valves to close in response to a containment isolation signal.

2.4 PLANT-SPECIFIC INPUTS

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

- LaSalle Unit 2 Level 1 PSA
- LaSalle Unit 2 Level 2 PSA
- Ex-plant consequence
- Past LaSalle ILRT results to demonstrate adequacy of the administrative and hardware issues.

2.4.1 LaSalle Unit 2 Level 1 PSA

The LaSalle Unit 2 Level 1 PSA (Rev. 2001A) used as input to this analysis is characteristic of the as-built, as-operated plant. The current Level 1 PSA model is developed in CAFTA. The total core damage frequency (CDF) for Unit 2 is 5.66E-6/yr. Table 2-1 summarizes the LaSalle Unit 2 Level 1 PSA frequency results by core damage functional accident class.

The Revision 2001A LaSalle Level 1 PSA models internal transients, LOCAs, internal flooding scenarios, and seismic-induced accident sequences[1].

(1) Other external events (e.g., external floods, tornadoes, etc.) are not included in the Revision 2001A PSA.

Table 2-1

SUMMARY OF LASALLE UNIT 2 LEVEL 1 PSA RESULTS

Core Damage Accident Class	Description	CDF(1)(2)(3) (/yr)	% of CDF
IA	Loss of Makeup at High RPV Pressure	3.68E-08	0.65
IBE	Early Station Blackout	4.35E-07	7.69
IBL	Late Station Blackout	9.87E-07	17.43
IC	Loss of Makeup accidents involving mitigated ATWS scenarios	6.47E-09(5)	0.11
ID	Loss of Makeup at Low RPV Pressure (transient Initiators)	1.87E-06	33.05
IE	Loss of Makeup due to DC power failures	6.50E-08	1.15
II(6)	Loss of Decay Heat Removal	1.84E-06	32.56
IIIB	SLOCA or MLOCA accidents in which RPV pressure is high at the time of core damage	4.39E-09	0.08
IIIC	Loss of Makeup at Low RPV Pressure (large LOCA Initiators)	9.09E-08	1.61
IIID	Large LOCA accidents with failure of the vapor suppression function	6.96E-08	1.23
IV(4)	ATWS	1.81E-07(5)	3.20
V	Containment Bypass	7.12E-08	1.26
Total		5.66E-06	100%

NOTES:

(1) LaSalle Unit 2 total CDF based on quantification of the "single-top" Rev. 2001A model at a truncation limit of 1E-11/year.

(2) As the "single-top" model does not provide accident class subtotals, the accident class subtotals were determined by running the "accident sequence" Rev. 2001A model, and then applying the resulting accident class percentage contributions of total CDF to the 5.66E-6/yr single-top CDF.

(3) The LaSalle Revision 2001A Level 1 PSA models internal transients, LOCAs, internal flooding scenarios, and seismic-induced accident sequences.

(4) Class IV results include contributions from Class IVL.

(5) A "labeling" error in the Rev. 2001A sequence model was identified and corrected for this risk application. The Rev. 2001A PRAQuant file contains minor inconsistencies in the accident class labeling of a dozen ATWS core damage sequences. A URE was created to track this error for correction in the next LaSalle PSA update.

(6) Includes all Class II subcategories.

2.4.2 LaSalle Unit 2 Level 2 PSA

The LaSalle Level 2 PSA models internal transients, LOCAs, internal flooding scenarios, and seismic-induced accident sequences[1].

The LaSalle Level 2 PSA is a LERF-model that calculates the Large Early Release Frequency risk measure; it does not provide specific frequency information for other release categories. As this ILRT risk assessment requires evaluation of the full range of release magnitudes and timings, the LaSalle LERF model has been extended in support of this analysis so that other release categories may be calculated (refer to Appendix B).

Table 2-2 summarizes the pertinent LaSalle Unit 2 Level 2 PSA results in terms of release category. The total Large Early Release Frequency (LERF), which corresponds to the LaSalle "H/E" release category, is calculated to be 2.70E-7/yr. The total release frequency is 4.72E-6/yr. The total frequency of accidents in which the containment remains intact (i.e., containment leakage within Technical Specifications – "OK" endstate) is 9.43E-7/yr. Refer to Appendix B for further details.

2.4.3 LaSalle Ex-Plant Consequences

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

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

The NUREG-1150 [9, 14] dose calculations were used in the EPRI TR-104285 study, as discussed previously in Section 2.1. The use of generic dose information for NUREG-1150 was recommended by NEI to make the ILRT risk assessment methodology more readily usable for plants that do not have a Level 3 PSA.

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(1) Other external events (e.g., external floods, tornadoes, etc.,) are not currently included in the PSA.

Table 2-2

SUMMARY OF LASALLE UNIT 2 LEVEL 2 PSA RESULTS

Level 1 CDF		LaSalle Level 2 PSA Release Bin Frequencies (1)(2)													
Class	CDF	Intact (OK)	LL/E	LL/I	LL/L	L/E	L/I	L/L	M/E	M/I	M/L	H/E	H/I	H/L	Total
IA	3.68E-08	3.34E-08	0.00E+00	0.00E+00	5.73E-10	0.00E+00	0.00E+00	0.00E+00	0.00E+00	2.23E-09	1.08E-10	4.56E-10	0.00E+00	0.00E+00	3.37E-09
IBE	4.35E-07	2.85E-07	0.00E+00	0.00E+00	1.47E-10	0.00E+00	0.00E+00	0.00E+00	0.00E+00	1.43E-07	1.39E-10	6.19E-09	0.00E+00	0.00E+00	1.50E-07
IBL	9.87E-07	5.49E-07	0.00E+00	0.00E+00	2.52E-10	0.00E+00	0.00E+00	0.00E+00	0.00E+00	4.24E-07	2.97E-10	0.00E+00	1.47E-08	0.00E+00	4.39E-07
IC	6.47E-09	6.26E-09	0.00E+00	0.00E+00	1.18E-10	0.00E+00	0.00E+00	0.00E+00	0.00E+00	2.01E-12	1.02E-11	7.44E-11	0.00E+00	0.00E+00	2.04E-10
ID	1.87E-06	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	1.84E-06	0.00E+00	2.81E-08	0.00E+00	0.00E+00	1.87E-06
IE	6.50E-08	3.70E-08	0.00E+00	0.00E+00	5.02E-10	0.00E+00	0.00E+00	0.00E+00	0.00E+00	2.62E-08	2.53E-10	9.16E-10	0.00E+00	0.00E+00	2.79E-08
II (3)	1.84E-06	0.00E+00	0.00E+00	1.35E-07	0.00E+00	0.00E+00	1.21E-06	0.00E+00	0.00E+00	4.44E-08	0.00E+00	0.00E+00	4.50E-07	0.00E+00	1.84E-06
IIIB	4.39E-09	4.33E-09	0.00E+00	0.00E+00	3.92E-12	0.00E+00	0.00E+00	0.00E+00	0.00E+00	6.05E-13	4.19E-13	5.04E-11	0.00E+00	0.00E+00	5.54E-11
IIIC	9.09E-08	2.77E-08	0.00E+00	0.00E+00	1.30E-08	0.00E+00	0.00E+00	0.00E+00	0.00E+00	3.41E-08	1.47E-08	1.30E-09	0.00E+00	0.00E+00	6.31E-08
IIID	6.96E-08	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	6.96E-08	0.00E+00	0.00E+00	6.96E-08
IV (4)	1.81E-07	0.00E+00	1.23E-08	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	7.63E-08	0.00E+00	0.00E+00	9.24E-08	0.00E+00	0.00E+00	1.81E-07
V	7.12E-08	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	7.12E-08	0.00E+00	0.00E+00	7.12E-08
Total:	5.66E-06	9.43E-07	1.23E-08	1.35E-07	1.46E-08	0.00E+00	1.21E-06	0.00E+00	7.63E-08	2.52E-06	1.55E-08	2.70E-07	4.64E-07	0.00E+00	4.72E-06
% of Total CDF:		16.7	0.2	2.4	0.3	0.0	21.5	0.0	1.3	44.5	0.3	4.8	8.2	0.0	100.0
% of Total Release:		n/a	0.3	2.9	0.3	0.0	25.7	0.0	1.6	53.3	0.3	5.7	9.8	0.0	100.0

NOTES TO TABLE 2-2:

(1) Release bin nomenclature is [Release Magnitude]/[Timing of Release], where:

LL:	Low-Low	E:	Early
L:	Low	I:	Intermediate
M:	Moderate	L:	Late
H:	High		

(2) The LaSalle Revision 2001A Level 2 PSA models internal transients, LOCAs, internal flooding scenarios, and seismic-induced accident sequences.

(3) Includes all Class II subcategories.

(4) Includes contributions from Class IVL.

Although LaSalle does not maintain a Level 3 PSA, a plant-specific Level 3 PSA was performed for the LaSalle plant by Sandia National Laboratories in the 1990 time frame. This study is documented in NUREG/CR-5305. [19]

This NUREG/CR-5305 ex-plant consequence analysis is calculated for the 50-mile radial area surrounding LaSalle, and is reported in total person-rem for discrete accident categories (termed Accident Progression Bins (APB) in NUREG/CR-5305). To use the NUREG/CR-5305 consequences in this ILRT risk assessment, the following steps should first be performed:

- Adjust the person-rem results to account for changes in:
  - Population
  - Reactor Power Level
  - Technical Specification Allowed Containment Leakage Rate
- Assign the adjusted NUREG/CR-5305 APB consequences to the EPRI categories used in this risk assessment

<u>LaSalle Surrounding Population</u>

The 50-mile radius population used in the 1992 NUREG/CR-5305 consequence calculations is 1,131,512 persons (refer to Appendix A of this report). The year 2000 population within the 50-mile radius of LaSalle is estimated in Appendix A of this report at 1,553,566 persons.

<u>LaSalle Reactor Power Level</u>

The LaSalle reactor power level used in the 1992 NUREG/CR-5305 consequence calculations is 3293 MWth (p. S-3 of Reference [19]). LaSalle recently performed a power uprate of 5% over the originally licensed thermal power; the current LaSalle full power level is 3489 MWth. [17]

LaSalle Technical Specification Containment Leakage

The containment leakage rate used in the 1992 NUREG/CR-5305 consequence calculations for core damage accidents with the containment intact is 0.5% over 24 hours (Tables 4.4-35 and 4.4-42 of Reference [19]). The LaSalle maximum allowable containment leakage per Technical Specifications is 0.635% per day (p. B 3/4 6-1 of LaSalle Technical Specifications).

2.4.4 LaSalle ILRT Results

The surveillance frequency for Type A testing in NEI 94-01 is at least once per ten years based on an acceptable performance history (i.e., two consecutive periodic Type A tests at least 24 months apart where the calculated performance leakage rate was less than1.0 L_a) and consideration of the performance factors in NEI 94-01, Section 11.3. Based on the consecutive successful ILRTs performed in the early 1990's, the current ILRT interval for both LaSalle Units 1 and 2 is once per ten years. [16]

Section 3
ANALYSIS

3.1 BASELINE ACCIDENT CATEGORY FREQUENCIES (STEP 1)

The first step of the NEI Interim Guidance (refer to Section 2.2 for outline of all steps) is to quantify the baseline frequencies for each of the EPRI TR-104285 accident categories. This portion of the analysis is performed using the LaSalle Level 1 and Level 2 PSA results. The results for each EPRI category are described below.

Frequency of EPRI Category 1

This group consists of all core damage accident sequences in which the containment is initially isolated and remains intact throughout the accident (i.e., containment leakage at or below maximum allowable Technical Specification leakage). Per NEI Interim Guidance, the frequency per year for this category is calculated by subtracting the frequencies of EPRI Categories 3a and 3b (see below) from the sum of all severe accident sequence frequencies in which the containment is initially isolated and remains intact (i.e., accidents classified as "OK" in the LaSalle Level 2 PSA).

As discussed previously in Section 2.4.2, the frequency of the LaSalle Level 2 PSA "OK" accident bin is 9.43E-7/yr. As described below, the frequencies of EPRI Categories 3a and 3b are 7.45E-8/yr and 7.45E-9/yr, respectively. Therefore, the frequency of EPRI Category 1 is calculated as (9.43E-7/yr) – (7.45E-8/yr + 7.45E-9/yr) = 8.61E-7/yr.

Frequency of EPRI Category 2

This group consists of all core damage accident sequences in which the containment isolation system function fails due to failures-to-close of large containment isolation valves (either due to support system failures; or random or common cause valve failures).

The frequency of this EPRI category is estimated as follows:

- Results (i.e., cutsets) of containment isolation failure fault tree (IS) are used as input
- All basic events, except those related to support system failure or random or common cause valve failures-to-close, are set to 0.00.
- Fraction of IS probability due to support system failure or random or common cause valve failures-to-close is then calculated. This value is then multiplied by the sum of the accident frequencies of the Level 2 containment isolation failure sequences (i.e., IA15, IBE15, IBL15, IC15, ID15, IE15, IIIA14, IIIB14, and IIIC14).

This process resulted in a fraction of 0.156 of the containment isolation failure probability due to support system failure or random or common cause valve failures-to-close. The sum of the LaSalle Level 2 containment isolation failure sequences is 2.07E-8/yr. Therefore, the frequency of EPRI Category 2 is 0.156 x 2.07E-8/yr = 3.22E-9/yr.

Note that all of the Level 2 containment isolation failure sequences outlined above except IBL15 are H/E sequences. Sequence IBL15 (representing 9.10E-10/yr of the EPRI Category #2 total frequency) is classified in the LaSalle Level 2 as a H/I release. This information is used in the calculation of the frequencies of EPRI Categories 7c and 7d.

Frequency of EPRI Category 3a

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

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

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

The pre-existing leakage probability is multiplied by the residual core damage frequency (CDF) determined as the total CDF minus the CDF for those individual sequences that either may already (independently) cause a LERF or could never cause a LERF. As discussed previously in Section 2.4.1, the LaSalle total core damage frequency is 5.66E-6/yr. Of this total CDF, the following core damage accidents involve either LERF directly (containment bypass) or will never result in LERF:

- Long-term Station Blackout (SBO) scenarios (LaSalle PSA Class IBL): 9.87E-7/yr
- Loss of Containment Heat Removal accidents (LaSalle PSA Class II): 1.84E-6/yr[(1)]
- Containment Bypass accidents (LaSalle PSA Class V): 7.12E-8/yr

(1) The current LaSalle Level 2 PSA models Class II accidents as proceeding to early releases, on the assumption that a General Emergency would not be declared for such accidents until very late in the accident sequence. Based on a re-evaluation by Exelon, this assumption has been proven to be conservative. [24] As such, the LERF model modifications performed in support of this risk application

Therefore, the frequency of EPRI Category 3a is calculated as (2.70E-02) x [(5.66E-6/yr) – (9.87E-7/yr + 1.84E-6/yr + 7.12E-8/yr)] = 7.45E-8/yr.

<u>Frequency of EPRI Category 3b</u>

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

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

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

Therefore, similar to EPRI Category 3a, the frequency of Category 3b is calculated as (2.70E-03) x [(5.66E-6/yr) – (9.87E-7/yr + 1.84E-6/yr + 7.12E-8/yr)] = 7.45E-9/yr.

<u>Frequency of EPRI Category 4</u>

This group consists of all core damage accident sequences in which the containment isolation function is failed due to a pre-existing failure-to-seal of Type B component(s) that would <u>not</u> be identifiable by an ILRT. Per NEI Interim Guidance, because this category of

(refer to Appendix B) include reclassifying releases for Class II accidents to the Intermediate time frame. Therefore, Class II accidents can not result in LERF releases.

failures is only detected by Type B tests and not by the Type A ILRT, this group is not evaluated further in this analysis.

Frequency of EPRI Category 5

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

Frequency of EPRI Category 6

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

Frequency of EPRI Category 7

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

As the LaSalle Level 2 PSA enhanced for this analysis (refer to Appendix B) appropriately categorizes containment failure accident sequences into different release bins, EPRI

Category 7 is sub-divided in this analysis to reflect the spectrum of the LaSalle Level 2 PSA results. The following sub-categories are defined here:

- Category 7a: severe accident induced containment failure resulting in Low magnitude releases in Intermediate time frame (LaSalle "L/I" release bin).
- Category 7b: severe accident induced containment failure resulting in Moderate magnitude releases in Intermediate time frame (LaSalle "M/I" release bin).
- Category 7c: severe accident induced containment failure resulting in High magnitude releases in Intermediate time frame (LaSalle "H/I" release bin).
- Category 7d: severe accident induced containment failure resulting in High magnitude releases in Early time frame (LaSalle "H/E" release bin).
- Category 7e: all other severe accident induced containment failure scenarios not represented by categories 7a-7d.

The frequency of Category 7a is the total frequency of the LaSalle Level 2 PSA "L/I" release bin. Based on the LaSalle Level 2 PSA results summarized earlier in Table 2-2, the frequency of Category 7a is 1.21E-6/yr.

The frequency of Category 7b is the total frequency of the LaSalle Level 2 PSA "M/I" release bin. Based on the LaSalle Level 2 PSA results summarized earlier in Table 2-2, the frequency of Category 7b is 2.52E-6/yr.

The frequency of Category 7c is the total frequency of the LaSalle Level 2 PSA "H/I" release bin minus the portion of the EPRI Category 2 frequency resulting in H/I releases. Based on the LaSalle Level 2 PSA results summarized earlier in Table 2-2 and the information presented earlier for the frequency of EPRI Category 2, the frequency of Category 7c is calculated as 4.64E-7/yr - 9.10E-10/yr = 4.63E-7/yr.

The frequency of Category 7d is determined by subtracting from the total frequency of the LaSalle Level 2 PSA "H/E" release bin the frequency of EPRI Category 8 and the portion of the EPRI Category 2 frequency resulting in H/E releases. Based on the LaSalle Level 2 results summarized earlier in Table 2-2, the frequency of the LaSalle Level 2 PSA "H/E" release bin is 2.70E-7/yr. As described previously, the frequency of EPRI Category 2 resulting in H/E releases is 2.31E-9/yr. As described below, the frequency of EPRI Category 8 is 7.12E-8/yr. Therefore, the frequency of Category 7d is calculated as (2.70E-7/yr) – (7.12E-8/yr + 2.31E-9/yr) = 1.96E-7/yr.

The frequency of Category 7e, 2.54E-7/yr, is determined by summing the frequencies of the remaining LaSalle Level 2 PSA release bins:

- LL/I: 1.35E-7
- LL/L: 1.46E-8
- M/E: 7.63E-8
- M/L: 1.55E-8
- LL/E: 1.23E-8
- L/E: 0.00
- L/L: 0.00
- H/L: 0.00

The release characteristics of Category 7e is conservatively modeled by the Moderate/Early (M/E) LaSalle release bin.

<u>Frequency of EPRI Category 8</u>

This group consists of all core damage accident progression bins in which the accident is initiated by a containment bypass scenario (i.e., Break Outside Containment LOCA or Interfacing Systems LOCA, ISLOCA). The frequency of Category 8 is the total frequency of the LaSalle Level 1 PSA containment bypass scenarios (Class V). Based on the

LaSalle Level 1 PSA results summarized earlier in Table 2-1, the frequency of Category 8 is 7.12E-8/yr.

Summary of Frequencies of EPRI Categories

In summary, per the NEI Interim Guidance, the accident sequence frequencies that can lead to radionuclide releases to the public have been derived for accident categories defined in EPRI TR-104285. The results are summarized in Table 3-1.

Table 3-1

BASELINE RELEASE FREQUENCY AS A FUNCTION OF EPRI CATEGORY

EPRI Category	Category Description	Frequency Estimation Methodology	Frequency (1/yr)
1	<u>No Containment Failure</u>: Accident sequences in which the containment remains intact and is initially isolated.	Per NEI Interim Guidance: *[Total LaSalle "OK" release category frequency]– [Frequency EPRI Categories 3a and 3b]* *[9.43E-7/yr]– [7.45E-8/yr+ 7.45E-9yr] = 8.61E-7/yr*	8.61E-07
2	<u>Containment Isolation System Failure</u>: Accident sequences in which the containment isolation system function fails due to failures-to-close of large containment isolation valves (either due to support system failures, or random or common cause failures). Not affected by ILRT leak testing frequency.	Cutsets of all LaSalle containment isolation fault tree used as input. All failure modes, except those related to support system failures or random and common cause valve failures-to-close, set to 0.00. Resulting fraction of IS failure probability due to support system or random or common cause FTC failures (0.156) multiplied by frequency sum of LaSalle CET containment isolation failure sequences (IA15, IBE15, IBL15, IC15, ID15, IE15, IIIA14, IIIB14, and IIIC14).	3.22E-09
3a	<u>Small Pre-Existing Failures</u>: Accident sequences in which the containment is failed due to a pre-existing small leak in the containment structure or liner that would be identifiable only from an ILRT (and thus affected by ILRT testing frequency).	Per NEI Interim Guidance: *[LaSalle CDF for accidents not involving containment failure/bypass]x [2.7E-2]* *[(5.66E-6/yr)– (9.87E-7/yr+ 1.84E-6/yr+ 7.12E-8/yr)]x [2.70E-02] = 7.45E-8/yr*	7.45E-08

Table 3-1

BASELINE RELEASE FREQUENCY AS A FUNCTION OF EPRI CATEGORY

EPRI Category	Category Description	Frequency Estimation Methodology	Frequency (1/yr)
3b	Large Pre-Existing Failures: Accident sequences in which the containment is failed due to a pre-existing large leak in the containment structure or liner that would be identifiable only from an ILRT (and thus affected by ILRT testing frequency).	Per NEI Interim Guidance: *[LaSalle CDF for accidents not involving containment failure/bypass] x [2.7E-3]* *[(5.66E-6/yr) – (9.87E-7/yr + 1.84E-6/yr + 7.12E-8/yr)] x [2.70E-03] = 7.45E-9/yr*	7.45E-09
4	Type B Failures: Accident sequences in which the containment is failed due to a pre-existing failure-to-seal of Type B components that would not be identifiable from an ILRT (and thus not affected by ILRT testing frequency).	Per NEI Interim Guidance: N/A (not affected by ILRT frequency)	n/a
5	Type C Failures: Accident sequences in which the containment is failed due to a pre-existing failure-to-seal of Type C components that would not be identifiable from an ILRT (and thus not affected by ILRT testing frequency).	Per NEI Interim Guidance: N/A (not affected by ILRT frequency)	n/a
6	Other Containment Isolation System Failure: Accident sequences in which the containment isolation system function fails due to "other" pre-existing failure modes not identifiable by leak rate tests (e.g., pathways left open or valves that did not properly seal following test or maintenance activities).	Per NEI Interim Guidance: N/A (not affected by ILRT frequency)	n/a
7a	Containment Failure Due to Accident (a): EPRI Category 7 applies to accident sequences in which the containment is failed due to the severe accident progression. Category 7a is defined in this analysis to apply to LaSalle PSA accidents that result in L/I releases. Not affected by ILRT leak testing frequency.	*[Total LaSalle "L/I" release category frequency]*	1.21E-06

Table 3-1

BASELINE RELEASE FREQUENCY AS A FUNCTION OF EPRI CATEGORY

EPRI Category	Category Description	Frequency Estimation Methodology	Frequency (1/yr)
7b	Containment Failure Due to Accident (b): EPRI Category 7 applies to accident sequences in which the containment is failed due to the severe accident progression. Category 7b is defined in this analysis to apply to LaSalle PSA accidents that result in M/I releases. Not affected by ILRT leak testing frequency.	*[Total LaSalle "M/I" release category frequency]*	2.52E-06
7c	Containment Failure Due to Accident (c): EPRI Category 7 applies to accident sequences in which the containment is failed due to the severe accident progression. Category 7c is defined in this analysis to apply to LaSalle PSA accidents that result in H/I releases. Not affected by ILRT leak testing frequency.	*[Total LaSalle "H/I" release category frequency] – [Portion of EPRI Categories #2 frequency resulting in H/I]* *[4.64E-7/yr] – [9.10E-10/yr] = 4.63E-7/yr*	4.63E-07
7d	Containment Failure Due to Accident (d): EPRI Category 7 applies to accident sequences in which the containment is failed due to the severe accident progression. Category 7d is defined in this analysis to apply to LaSalle PSA accidents that result in H/E releases (excluding contributions from EPRI Categories 2 and 8). Not affected by ILRT leak testing frequency.	*[Total LaSalle "H/E" release category frequency] – [(Frequency EPRI Category #8) + (Portion of EPRI Category #2 frequency resulting in H/E)]* *[2.70E-7/yr] – [7.12E-8/yr + 2.31E-9/yr] = 1.96E-7/yr*	1.96E-07
7e	Containment Failure Due to Accident (e): EPRI Category 7 applies to accident sequences in which the containment is failed due to the severe accident progression. Category 7e is defined in this analysis to apply to LaSalle PSA accidents that result in all other remaining release categories (consequences modeled in this assessment by M/E releases). Not affected by ILRT leak testing frequency.	Calculated as the sum of all other remaining LaSalle release categories: • LL/I: 1.35E-7 • LL/L: 1.46E-8 • M/E: 7.63E-8 • M/L: 1.55E-8 • LL/E: 1.23E-8 • L/E: 0.00 • L/L: 0.00 • H/L: 0.00	2.54E-07

Table 3-1

BASELINE RELEASE FREQUENCY AS A FUNCTION OF EPRI CATEGORY

EPRI Category	Category Description	Frequency Estimation Methodology	Frequency (1/yr)
8	Containment Bypass Accidents: Accident sequences in which the containment is bypassed. Such accidents are initiated by LOCAs outside containment (i.e., Break Outside Containment LOCA, or Interfacing Systems LOCA). Not affected by ILRT leak testing frequency.	*[Total LaSalle Containment Bypass (Accident Class V) release frequency]*	7.12E-08
		TOTAL:	5.66E-06

3.2 CONTAINMENT LEAKAGE RATES (STEP 2)

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

The NEI Interim Guidance recommends containment leakage rates of 10La and 35La for Categories 3a and 3b, respectively. These values are consistent with previous ILRT frequency extension submittal applications. La is the plant Technical Specification maximum allowable containment leak rate; for LaSalle La is 0.635% of containment air weight per day (per LaSalle Technical Specifications p. B 3/4 6-1).

The NEI Interim Guidance describes these two recommended containment leakage rates as "conservative". The NEI recommended values of 10La and 35La are used as is in this analysis to characterize the containment leakage rates for Categories 3a and 3b.

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

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

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

3.3.1 Population Dose Estimates (Step 3)

As discussed in Section 2.4.3, the LaSalle NUREG/CR-5305 ex-plant consequence results are used as input to determine the population dose estimates of this risk assessment. The NUREG/CR-5305 50-mile radius ex-plant consequence results are summarized in Table 3-2 as a function of accident progression bins (APBs).

The NUREG/CR-5305 consequences summarized in Table 3-2 must be adjusted for use in this analysis to account for changes in the following parameters:

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

Population Adjustment

As discussed in Section 2.4.3, the 50-mile radius population used in the 1992 NUREG/CR-5305 consequence calculations is 1,131,512 persons, whereas the year 2000 population within the 50-mile radius of LaSalle is estimated at 1,553,566 persons. This increase in population results in the following adjustment factor to be applied to the NUREG/CR-5305 APB doses: 1,553,566/1,131,512 = 1.37.

Reactor Power Level Adjustment

As discussed in Section 2.4.3, the reactor power level used in the NUREG/CR-5305 consequence calculations is 3293 MWth, whereas the current LaSalle full power level is 3489 MWth. This increase in reactor power level results in the following adjustment factor to be applied to the NUREG/CR-5305 APB doses: 3489/3293 = 1.06.

Table 3-2

LASALLE NUREG/CR-5305 50-MILE RADIUS POPULATION DOSE (1)

APB #	APB Definition	APB Frequency (per year) (2)	APB Contribution to 50-Mile Radius Total Dose Rate(3) (Fraction of Total)	APB 50-Mile Radius Dose Rate (person-rem/year) (4)	APB 50-Mile Radius Dose (Person-rem) (5)
1	**VB, Early CF, RPV at Low Pressure:** Vessel breach occurs, the containment fails either before or at the time of vessel breach, and the reactor pressure vessel is at low pressure at the time of vessel breach.	1.53E-05	0.18	12.012	7.85E+05
2	**VB, Early CF, RPV at High Pressure:** Vessel breach occurs, the containment fails either before or at the time of vessel breach, and the reactor pressure vessel is at high pressure at the time of vessel breach.	1.94E-05	0.25	16.5	8.51E+05
3	**VB, Late CF:** Vessel breach occurs and the containment fails late in the accident (i.e., hours after vessel breach).	9.46E-06	0.10	6.864	7.26E+05
4	**VB, Early or Late Venting:** Vessel breach occurs and the containment is either vented before vessel breach or late in the accident.	3.84E-05	0.43	28.314	7.37E+05
5	**VB, No CF:** Vessel breach occurs; however, the containment neither fails nor is vented during the accident.	5.82E-06	0.001	0.066	1.13E+04
6	**No VB, CF:** The core damage process is arrested (i.e., no vessel breach); however, the containment still fails during the accident due to the generation of steam and non-condensibles during the accident.	0.00E+00	0.00	0	n/a (6)

Table 3-2

LASALLE NUREG/CR-5305 50-MILE RADIUS POPULATION DOSE [1]

APB #	APB Definition	APB Frequency (per year) [2]	APB Contribution to 50-Mile Radius Total Dose Rate[3] (Fraction of Total)	APB 50-Mile Radius Dose Rate (person-rem/year) [4]	APB 50-Mile Radius Dose (Person-rem) [5]
7	**No VB, Venting:** The core damage process is arrested before vessel failure. However, the containment is vented either before the onset of core damage or during the core damage process.	9.05E-06	0.03	1.914	2.11E+05
8	**No VB, No CF, No Venting:** The core damage process is arrested and the containment remains intact.	6.76E-06	0.001	0.066	9.76E+03
	Total:	1.04E-04	1.00	66	—

(1) This table is presented in the form of a calculation because NUREG/CR-5305 does not document dose results as a function of accident progression bin (APB); as such, the dose results as a function of APB must be back calculated from documented APB frequencies and APB dose rate results.

(2) The total (i.e., internal plus external accident sequences) CDF of 1.04E-4/yr and the CDF subtotals by APB are taken from Figure 3.5-8 of NUREG/CR-5305.

(3) The individual APB contributions to total (i.e., internal plus external accident sequences) 50-mile radius dose rate are taken from Table 6.3-2 of NUREG/CR-5305.

(4) The individual APB 50-mile dose rates are calculated by multiplying the individual APB dose rate contributions by the total 50-mile radius dose rate of 66 person-rem/yr (taken from Table 6.2-1 of NUREG/CR-5305).

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

(6) As the frequency of APB#6 was calculated as negligible (i.e., no frequency results survived the quantification truncation limit) in NUREG/CR-5305, no dose result can be estimated for APB#6.

Containment Leakage Rate Adjustment

As discussed in Section 2.4.3, the containment leakage rate used in the 1992 NUREG/CR-5305 consequence calculations for core damage accidents with the containment intact is 0.5% over 24 hours, whereas the LaSalle maximum allowable containment leakage per Technical Specifications is 0.635% per day. While use of a leakage rate below the maximum allowable may be reasonable, this analysis assumes that containment leakage is at the maximum allowable Technical Specification value. As such, this difference in allowable containment leakage rate results in the following adjustment factor to be applied to the NUREG/CR-5305 APB doses: 0.635/0.5 = 1.27. The adjustment factor applies only to the "no containment failure" cases (i.e., APBs #5 and #8).

NUREG/CR-5305 Adjusted Doses

Table 3-3 summarizes the NUREG/CR-5305 doses after adjustment for changes in population, reactor power level, and containment leakage rate.

LaSalle Population Dose By EPRI Category

The NUREG/CR-5305 dose results summarized in Table 3-3 are then assigned to the EPRI accident categories based on similarity of accident characteristics. The LaSalle 50-mile population dose by EPRI accident category are summarized in Table 3-4.

The dose for the "no containment failure" category (EPRI Category 1) is based on NUREG/CR-5305 APB #5. Two "no containment failure" APBs, one with RPV breach (APB #5) and one without RPV breach (APB #8), are analyzed in NUREG/CR-5305. The APB with the highest calculated 50-mile radius dose (i.e., the case with RPV breach, APB #5) is assigned to EPRI Category 1.

Table 3-3

ADJUSTED NUREG/CR-5305 50-MILE RADIUS POPULATION DOSES

APB #	50-Mile Radius Dose (Person-rem) [5]	Population Adjustment Factor	Reactor Power Adjustment Factor	Containment Leak Rate Adjustment Factor	Adjusted 50-Mile Radius Dose (Person-rem)
1	7.85E+05	1.37	1.06	n/a	1.14E+06
2	8.51E+05	1.37	1.06	n/a	1.24E+06
3	7.26E+05	1.37	1.06	n/a	1.05E+06
4	7.37E+05	1.37	1.06	n/a	1.07E+06
5	1.13E+04	1.37	1.06	1.27	2.09E+04
6	n/a	n/a	n/a	n/a	n/a
7	2.11E+05	1.37	1.06	n/a	3.07E+05
8	9.76E+03	1.37	1.06	1.27	1.80E+04

Table 3-4

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

EPRI Category	Category Description	Person-Rem Within 50 miles
1	No Containment Failure	2.09E+04
2	Containment Isolation System Failure	1.24E+06
3a	Small Pre-Existing Failures	2.09E+05
3b	Large Pre-Existing Failures	7.32E+05
4	Type B Failures	n/a
5	Type C Failures	n/a
6	Other Containment Isolation System Failure	n/a
7a	Containment Failure Due to Severe Accident (a)	1.07E+06
7b	Containment Failure Due to Severe Accident (b)	1.05E+06
7c	Containment Failure Due to Severe Accident (c)	1.05E+06
7d	Containment Failure Due to Severe Accident (d)	1.24E+06
7e	Containment Failure Due to Severe Accident (e)	1.14E+06
8	Containment Bypass Accidents	1.24E+06

The dose for EPRI Category 2 is based on NUREG/CR-5305 APB #2. This assignment is based on assuming that the containment isolation failure of EPRI Category 2 occurs in the drywell. While APB #2 does not specify containment failure location, it results in the highest dose of all the NUREG/CR-5305 "containment failure" APBs (which is indicative of a drywell containment failure).

No assignment of NUREG/CR-5305 APBs is made for EPRI Categories 3a and 3b. Per the NEI Interim Guidance, the doses for EPRI Categories #3a and #3b are taken as a factor of 10 and 35, respectively, times the dose of EPRI Category 1.

As EPRI Categories 4, 5, and 6 are not affected by ILRT frequency and not analyzed as part of this risk assessment (per NEI Interim Guidance), no assignment of NUREG/CR-5305 APBs is made for these categories.

The dose for EPRI Category 7a is based on NUREG/CR-5305 APB #4. The majority of EPRI Category 7a is due to long-term loss of decay heat removal accidents in which core damage, vessel breach, and containment failure in the wetwell airspace occur many hours after accident initiation.

The dose for EPRI Category 7b is based on NUREG/CR-5305 APB #3. The majority of EPRI Category 7b is due to loss of coolant make-up accidents in which core damage and vessel breach occur at low vessel pressure early in the accident, and containment failure in the drywell occurs many hours later.

The dose for EPRI Category 7c is also based on NUREG/CR-5305 APB #3. The majority of EPRI Category 7c is due to long-term loss of decay heat removal accidents in which core damage, vessel breach, and containment failure in the drywell occur many hours after accident initiation.

The dose for EPRI Category 7d is based on NUREG/CR-5305 APB #2. The LaSalle accident scenarios comprising EPRI Category 7d result in H/E release (the most severe release category). Accordingly, the most severe NUREG/CR-5305 dose case (i.e., APB #2) is used to characterize this category.

The dose for EPRI Category 7e is based on NUREG/CR-5305 APB #1. The majority of EPRI Category 7e is due to unmitigated ATWS accidents in which containment failure in the wetwell airspace, and subsequent core damage and vessel breach occur early in the accident scenario.

The dose for the containment bypass category, EPRI Category 8, is based on NUREG/CR-5305 APB #2. APB #2 results in the highest dose of all the NUREG/CR-5305 "containment failure" APBs, indicative (i.e., in a relative comparison to other accidents) of containment bypass scenarios.

3.3.2 Baseline Population Dose Rate Estimates (Step 4)

The baseline dose rates per EPRI accident category are calculated by multiplying the dose estimates summarized in Table 3-4 by the frequencies summarized in Table 3-1. The resulting baseline population dose rates by EPRI category are summarized in Table 3-5. As the conditional containment pre-existing leakage probabilities for EPRI Categories 3a and 3b are reflective of a 3-per-10 year ILRT frequency (refer to Section 3.1), the baseline results shown in Table 3-5 are indicative of a 3-per-10 year ILRT surveillance frequency.

Table 3-5

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

EPRI Category	Category Description	Person-Rem Within 50 miles	Accident Frequency (Per Year)	Population Dose Rate (Person-Rem/Year Within 50 miles)
1	No Containment Failure	2.09E+04	8.61E-07	1.80E-02
2	Containment Isolation System Failure	1.24E+06	3.22E-09	3.98E-03
3a	Small Pre-Existing Failures	2.09E+05	7.45E-08	1.56E-02
3b	Large Pre-Existing Failures	7.32E+05	7.45E-09	5.45E-03
4	Type B Failures	n/a	n/a	n/a
5	Type C Failures	n/a	n/a	n/a
6	Other Containment Isolation System Failure	n/a	n/a	n/a
7a	Containment Failure Due to Severe Accident (a)	1.07E+06	1.21E-06	1.30E+00
7b	Containment Failure Due to Severe Accident (b)	1.05E+06	2.52E-06	2.65E+00
7c	Containment Failure Due to Severe Accident (c)	1.05E+06	4.63E-07	4.88E-01
7d	Containment Failure Due to Severe Accident (d)	1.24E+06	1.96E-07	2.43E-01
7e	Containment Failure Due to Severe Accident (e)	1.14E+06	2.54E-07	2.89E-01
8	Containment Bypass Accidents	1.24E+06	7.12E-08	8.79E-02
		TOTAL:	5.66E-06	5.1039

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

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

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

3.4.1 Change in Probability of Detectable Leakage (Step 5)

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

Per the NEI Interim Guidance, the calculation of the change in the probability of a pre-existing ILRT-detectable containment leakage is based on the relationship that relaxation of the ILRT interval results in increasing the average time that a pre-existing leak would exist undetected. Using the standby failure rate statistical model, the average time that a pre-existing containment leak would exist undetected is one-half the surveillance interval. For example, if the ILRT frequency is 1-per-10 years, then the average time that a leak would be undetected is 60 months (surveillance interval of 120 months divided by 2). The impact on the leakage probability due to the ILRT interval extension is then calculated by applying a multiplier determined by the ratio of the average times of undetection for the two ILRT interval cases. This is the approach used in this step to calculate the changes in the 3a and 3b category frequencies as a function of increased ILRT interval.

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

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

Since the latter half of the 1990's, the LaSalle plant has been operating under a 1-per-10 year ILRT testing frequency consistent with the performance-based Option B of 10 CFR Part 50, Appendix J. [16] As such, the NEI baseline 3-per-10 year based leakage probabilities first need to be adjusted to reflect the current 1-per-10 year LaSalle ILRT testing frequency. Using the standby failure rate model relationship discussed above, the 1-per-10 year pre-existing leakage probabilities are calculated as follows:

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

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

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

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

Given the above adjusted pre-existing containment leakage probabilities, the impacted frequencies of the EPRI categories are summarized below (refer to Table 3-1 for details regarding frequency calculations for the individual EPRI categories):

EPRI Category	EPRI Category Frequency as a Function of ILRT Interval		
	Baseline (3-per-10 year ILRT)	Current (1-per-10 year ILRT)	Proposed (1-per-15 year ILRT)
1	8.61E-07	6.69E-07	5.33E-07
3a	7.45E-08	2.48E-07	3.73E-07
3b	7.45E-09	2.48E-08	3.73E-08

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

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

Using the revised EPRI category frequencies due to ILRT interval extension (Step 5), the revised dose rates are then calculated (i.e., category frequency x category dose). The dose rates per EPRI accident category as a function of ILRT interval are summarized in Table 3-6.

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

As can be seen from the dose rate results summarized in Table 3-6, the calculated total dose rate changes imperceptibly from the current LaSalle 1-per-10 year ILRT interval to

Table 3-6

DOSE RATE ESTIMATES AS A FUNCTION OF ILRT INTERVAL FOR POPULATION WITHIN 50-MILE RADIUS

EPRI Category	Category Description	Dose Rate as a Function of ILRT Interval (Person-Rem/Yr)		
		Baseline (3-per-10 year ILRT)	Current (1-per-10 year ILRT)	Proposed (1-per-15 year ILRT)
1	No Containment Failure	1.80E-02	1.40E-02	1.11E-02
2	Containment Isolation System Failure	3.98E-03	3.98E-03	3.98E-03
3a	Small Pre-Existing Failures	1.56E-02	5.19E-02	7.79E-02
3b	Large Pre-Existing Failures	5.45E-03	1.82E-02	2.73E-02
4	Type B Failures	n/a	n/a	n/a
5	Type C Failures	n/a	n/a	n/a
6	Other Containment Isolation System Failure	n/a	n/a	n/a
7a	Containment Failure Due to Severe Accident (a)	1.30E+00	1.30E+00	1.30E+00
7b	Containment Failure Due to Severe Accident (b)	2.65E+00	2.65E+00	2.65E+00
7c	Containment Failure Due to Severe Accident (c)	4.88E-01	4.88E-01	4.88E-01
7d	Containment Failure Due to Severe Accident (d)	2.43E-01	2.43E-01	2.43E-01
7e	Containment Failure Due to Severe Accident (e)	2.89E-01	2.89E-01	2.89E-01
8	Containment Bypass Accidents	8.79E-02	8.79E-02	8.79E-02
	TOTAL:	5.1039	5.1490	5.1812

the proposed 1-per-15 year ILRT interval. The total dose increases from 5.1490 person-rem/year to 5.1812 person-rem/year (an increase of <1%).

Per the NEI Interim Guidance, the change in percentage contribution to total dose rate attributable to EPRI Categories 3a and 3b is also investigated here. Using the results summarized in Table 3-6, for the current LaSalle 1-per-10 year ILRT interval, the percentage contribution to total dose rate from Categories 3a and 3b is shown to be very minor:

$$[(5.19E\text{-}2 + 1.82E\text{-}2) / 5.1490] \times 100 = 1.4\%$$

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

$$[(7.79E\text{-}2 + 2.73E\text{-}2) / 5.1812] \times 100 = 2.0\%$$

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

The risk increase associated with extending the ILRT interval involves the potential that a core damage event that normally would 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. Per the NEI Interim Guidance, only Category 3b sequences have the potential to result in large releases if a pre-existing leak were present. As such, the change in LERF (Large Early Release Frequency) is determined by the change in the frequency of Category 3b.

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

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

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

= 3.73E-8/yr – 2.48E-8/yr

= 1.24E-8/yr[(1)]

This delta LERF of 1.24E-8/yr falls into Region III, Very Small Change in Risk, of the acceptance guidelines in NRC Regulatory Guide 1.174. Therefore, increasing the ILRT interval at LaSalle from the currently allowed 1-per-10 years to 1-per-15 years represents a very small change in risk, and is an acceptable plant change from a risk perspective.

3.4.5 Impact on Conditional Containment Failure Probability (Step 9)

Another parameter that the NRC Guidance in Reg. Guide 1.174 states can provide input into the decision-making process is the consideration of change in the conditional containment failure probability (CCFP). The change in CCFP is indicative of the effect of the ILRT on all radionuclide releases, not just LERF. The conditional containment failure probability (CCFP) can be calculated from the risk calculations performed in this analysis. In this assessment, based on the NEI Interim Guidance, CCFP is defined such that containment failure includes all radionuclide release end states other than the intact state

(1) The 1.24E-8/yr value, as are all calculated values in this analysis, is performed using a spreadsheet calculation of summed frequencies that contain additional significant figures beyond the 2 digits shown in the two numbers subtracted above.

(EPRI Category 1) and small failures (EPRI Category 3a). The conditional part of the definition is conditional given a severe accident (i.e., core damage).

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

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

For the 10-year interval:

$$\begin{aligned} CCFP_{10} &= [1 - ((6.69E\text{-}7 + 2.48E\text{-}7) / 5.66E\text{-}6)] \times 100\% \\ &= 83.8\% \end{aligned}$$

And for a 15-year interval:

$$\begin{aligned} CCFP_{15} &= [1 - ((5.33E\text{-}7 + 3.73E\text{-}7) / 5.66E\text{-}6)] \times 100\% \\ &= 84.0\% \end{aligned}$$

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

$$\Delta\ CCFP_{\%} = CCFP_{15} - CCFP_{10} = 0.2 \text{ percentage points}$$

This change in $CCFP_{\%}$ of less than 1 percentage point is insignificant from a risk perspective.

Section 4
RESULTS SUMMARY

The application of the approach based on NEI Interim Guidance [3, 21], EPRI-TR-104285 [2] and previous risk assessment submittals on this subject [6, 18, 20] have led to the quantitative results summarized in this section. These results demonstrate a very small impact on risk associated with the one time extension of the ILRT test interval to 15 years.

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

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

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

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

- Increasing the current 10 year ILRT interval to 15 years results in an insignificant increase in total population dose rate from 5.1490 person-rem/year to 5.1812 person-rem/year.
- The increase in the LERF risk measure is also insignificant, a 1.24E-8/yr increase. This LERF increase is categorized as a "very small" increase per NRC Reg. Guide 1.174.
- Likewise, the conditional containment failure probability ($CCFP_{\%}$) increases insignificantly by 0.2 percentage points.

Table 4-1

QUANTITATIVE RESULTS AS A FUNCTION OF ILRT INTERVAL

EPRI Category	Dose (Person-Rem Within 50 miles)	Quantitative Results as a Function of ILRT Interval					
		Baseline (3-per-10 year ILRT)		Current (1-per-10 year ILRT)		Proposed (1-per-15 year ILRT)	
		Accident Frequency (per year)	Population Dose Rate (Person-Rem/Year Within 50 miles)	Accident Frequency (per year)	Population Dose Rate (Person-Rem/Year Within 50 miles)	Accident Frequency (per year)	Population Dose Rate (Person-Rem/Year Within 50 miles)
1	2.09E+04	8.61E-07	1.80E-02	6.69E-07	1.40E-02	5.33E-07	1.11E-02
2	1.24E+06	3.22E-09	3.98E-03	3.22E-09	3.98E-03	3.22E-09	3.98E-03
3a	2.09E+05	7.45E-08	1.56E-02	2.48E-07	5.19E-02	3.73E-07	7.79E-02
3b	7.32E+05	7.45E-09	5.45E-03	2.48E-08	1.82E-02	3.73E-08	2.73E-02
4	n/a	n/a	n/a	n/a	n/a	n/a	n/a
5	n/a	n/a	n/a	n/a	n/a	n/a	n/a
6	n/a	n/a	n/a	n/a	n/a	n/a	n/a
7a	1.07E+06	1.21E-06	1.30E+00	1.21E-06	1.30E+00	1.21E-06	1.30E+00
7b	1.05E+06	2.52E-06	2.65E+00	2.52E-06	2.65E+00	2.52E-06	2.65E+00
7c	1.05E+06	4.63E-07	4.88E-01	4.63E-07	4.88E-01	4.63E-07	4.88E-01
7d	1.24E+06	1.97E-07	2.43E-01	1.97E-07	2.43E-01	1.97E-07	2.43E-01
7e	1.14E+06	2.54E-07	2.89E-01	2.54E-07	2.89E-01	2.54E-07	2.89E-01
8	1.24E+06	7.12E-08	8.79E-02	7.12E-08	8.79E-02	7.12E-08	8.79E-02
TOTALS:		5.66E-06	5.1039	5.66E-06	5.1490	5.66E-06	5.1812
Increase in Dose Rate (1)					5E-2		3E-2
Increase in LERF (2)				1.74E-8		1.24E-8	
Increase in $CCFP_{\%}$ (3)				0.3		0.2	

NOTES TO TABLE 4-1:

(1) The increase in dose rate (person-rem/year) is with respect to the results for the preceding ILRT interval, as presented in the table. For example, the increase in dose rate for the proposed 1-per-15 ILRT is calculated as: total dose rate for 1-per-15 year ILRT, 5.1812, minus total dose rate for 1-per-10 year ILRT, 5.1490, equals 3E-2.

(2) The increase in Large Early Release Frequency (LERF) is with respect to the results for the preceding ILRT interval, as presented in the table. As discussed in Section 3.4.4 of the report, the change in LERF is determined by the change in the accident frequency of EPRI Category 3b. For example, the increase in LERF for the proposed 1-per-15 ILRT is calculated as: 3b frequency for 1-per-15 year ILRT, 3.73E-08/yr, minus 3b frequency for 1-per-10 year ILRT, 2.48E-08/yr, equals 1.24E-08/yr.

(3) The increase in the conditional containment failure probability ($CCFP_{\%}$) is with respect to the results for the proceeding ILRT internal as presented in the table. As discussed in Section 3.4.5, the $CCFP_{\%}$ is calculated as:

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

Section 5

CONCLUSIONS

5.1 QUANTITATIVE CONCLUSIONS

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

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

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

> Reg. Guide 1.174 [4] provides guidance for determining the risk impact of plant-specific changes to the licensing basis. Reg. Guide 1.174 defines very small changes in risk as resulting in increases of CDF below 10^{-6}/yr and increases in LERF below 10^{-7}/yr. Since the ILRT does not impact CDF, the relevant criterion is LERF. The increase in LERF resulting from a change in the Type A ILRT test interval from once-per-ten years to once-per-fifteen years (using the change in the EPRI Category 3b frequency per the NEI Interim Guidance) is 1.24E-8/yr. Guidance in Reg. Guide 1.174 defines very small changes in LERF as below 10^{-7}/yr. Therefore, increasing the LaSalle ILRT interval from 10 to 15 years results in a very small change in risk, and is an acceptable plant change from a risk perspective.

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

The change in population dose is also reported consistent with previously approved ILRT interval extension requests. The change in population dose from the current 1/10 year ILRT frequency to 1/15 year frequency is 0.6%.

5.2 RISK TRADE-OFF

The performance of an ILRT introduces risk. An EPRI study of operating experience events associated with the performance of ILRTs has indicated that there are real risk impacts associated with the setup and performance of the ILRT during shutdown operation [8]. While these risks have not been quantified for LaSalle, 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.

5.3 EXTERNAL EVENTS IMPACT

The impact of external events on this ILRT risk assessment is summarized in this section (refer to Appendix C for further detail). The following categories of external events are discussed:

- Seismic
- Internal Fires
- High winds/tornadoes
- External Floods
- Other

5.3.1 Overview of LaSalle External Events

Seismic Events

Seismic-induced accident sequences are included in the LaSalle Revision 2001A PSA; as such, they are included explicitly in the quantification of this ILRT risk assessment.

Internal Fires

LaSalle does not currently maintain PSA models for internal fires. The impact of internal fires on this ILRT risk assessment is based on review of the internal fires PSA work performed for LaSalle as part of the RMIEP study (NUREG/CR-4832). Refer to Appendix C.2 for a detailed discussion.

The LaSalle fire risk, as evaluated in the RMIEP study, is dominated by long term core damage accidents. The risk impact (LERF) of ILRT frequency changes is dominated by short term core damage accidents. As such, explicit inclusion of internal fire accident frequency information in this ILRT risk assessment would not significantly alter the LERF quantitative results nor would it change the conclusions of this assessment.

High Winds/Tornadoes

The LaSalle plant design with respect to high wind and tornado loadings meets all the applicable criteria of the NRC Standard Review Plan (SRP). Core damage accidents induced by high winds or tornadoes are not significant contributors to plant risk (approximately 1% of the Revision 2001A PSA CDF).

External Floods

The LaSalle plant design with respect to external flooding meets all the applicable criteria of the NRC Standard Review Plan (SRP). Core damage accidents induced by external flooding are negligible contributors to plant risk.

Other External Hazards

The LaSalle site characteristics and design meet all the applicable criteria of the NRC Standard Review Plan (SRP). Core damage accidents induced by transportation accidents, nearby facility accidents, turbine missiles, and other miscellaneous external hazards are not significant contributors to plant risk.

5.3.2 Qualitative Assessment of Impact on External Event Risk

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

Although it is not possible at this time to incorporate quantitative risk assessments of all[(1)] external event hazards into this assessment, it is judged that if all external hazards were modeled in detail and a quantitative evaluation were performed in support of this proposed plant change, the calculated risk increase for both internal and external hazards would remain "very small".

5.4 PREVIOUS ASSESSMENTS

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

- Reducing the frequency of Type A tests (ILRTs) from three per 10 years to one per 20 years was found to lead to an imperceptible increase in risk. The estimated increase in risk is very small because ILRTs identify only a few potential containment leakage paths that cannot be identified by Type B and C testing, and the leaks that have been found by Type A tests have been only marginally above existing requirements.
- Given the insensitivity of risk to containment leakage rate and the small fraction of leakage paths detected solely by Type A testing, increasing the interval between integrated leakage-rate tests is possible with minimal impact on public risk. The impact of relaxing the ILRT frequency beyond one in 20 years has not been evaluated. Beyond testing the performance of containment penetrations, ILRTs also test the integrity of the containment failure.

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

(1) As discussed earlier, seismic-induced accident sequences are included explicitly in the quantitative analyses of this risk assessment.

Section 6

REFERENCES

[1] Nuclear Energy Institute, Industry Guideline for Implementing Performance-Based Option of 10 CFR Part 50, Appendix J, NEI 94-01, July 1995.

[2] Electric Power Research Institute, Risk Impact Assessment of Revised Containment Leak Rate Testing Intervals, EPRI TR-104285, August 1994.

[3] Letter from A. Petrangelo (NEI) to NEI Administrative Points of Contact, *"Interim Guidance for Performing Risk Impact Assessments in Support of One-Time Extensions for Containment Integrated Leak Rate Test Surveillance Intervals"*, November 13, 2001.

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Appendix A
LASALLE POPULATION DATA

The 50-mile radius population dose (person-rem) estimates used in this ILRT risk assessment are based on the LaSalle-specific accident consequence calculations documented in the 1992 NUREG/CR-5305 study. In order to use these 1992 LaSalle consequence results, they must first be scaled upward to account for the growth in population around the LaSalle site in the past decade.

A.1 NUREG/CR-5305 POPULATION

While the 1992 LaSalle NUREG/CR-5305 study reports population dose rate results for the 50-mile radius around the LaSalle site, the NUREG/CR-5305 documentation does not report the population total of the 50-mile radius used in the analysis. The purpose of this appendix is to estimate the 50-mile radius population total that was used in the NUREG/CR-5305 study, so that it may be used in this ILRT risk assessment for scaling and estimating population dose rates.

Table A-1 summarizes the population data around the LaSalle site as reported in the NUREG/CR-5305 study. As can be seen from Table A-1, this population data is for various radial distances around the plant, and does not include explicit information for the 50-mile radius.

Three methods are used here to estimate the 50-mile radius population used in the NUREG/CR-5305 study:

Method 1: Using the NUREG/CR-5305 reported population data points, assume direct proportion of population with area

Table A-1

LASALLE POPULATION DATA REPORTED IN NUREG/CR-5305 [19]

Radius From Site		
Miles	Kilometers	Population (persons) (1)
1	1.6	24
3	4.8	309
10	16.1	14, 730
30	48.3	217, 620
100	160.9	10, 372, 934
350	563.3	48, 584, 604
1000	1609.3	179, 831, 712

(1) The NUREG/CR-5305 population estimates are based on 1980 census information, updated to reflect the time period of the NUREG/CR-5305 study.

Method 2: Using the NUREG/CR-5305 reported population data points, interpolate between estimates for 30 miles and 100 miles as a function of area

Method 3: Using U.S. Census 2000 data and associated percentage changes in municipality populations compared to 1990 Census data, calculate the 1990 50-mile radius population

Method 1

This method assumes a constant population density, thus calculating the population of one area as a direct proportion of another. This population estimation method is performed for both the NUREG/CR-5305 30-mile radius data point and the 100-mile radius data point.

Using the population density indicated by the 30-mile radius data point produces the following 50-mile radius population estimate:

$$\frac{\pi R_{30}^2}{217{,}620} = \frac{\pi R_{50}^2}{Pop_{50}}$$

$$Pop_{50} = 217{,}620 \times (R_{50}^2/R_{30}^2) = 604{,}500 \text{ persons}$$

Using the population density indicated by the 100-mile radius data point produces the following 50-mile radius population estimate:

$$\frac{\pi R_{50}^2}{Pop_{50}} = \frac{\pi R_{100}^2}{10{,}372{,}934}$$

$$Pop_{50} = 10{,}372{,}934 \times (R_{50}^2/R_{100}^2) = 2{,}593{,}233 \text{ persons}$$

Using the 30-mile radius data point to calculate the 50-mile radius population produces a lower end value, as the population density closer to the site is comparatively low. Using the 100-mile radius data point produces a higher end value, as the population density for the 100-mile radius includes the highly populated Chicago area. The more correct value lies between these estimates.

Method 2

This population estimation method is an interpolation assuming a linearly increasing population with distance (refer to Figure A-1). Interpolating, using areas corresponding to the distances, results in the following 50-mile radius estimate;

$$\frac{(10{,}372{,}934 - 217{,}620)}{(3.14E{+}4 - 2.83E{+}3)} = \frac{(Pop_{50} - 217{,}620)}{(7.85E{+}3 - 2.83E{+}3)}$$

$$Pop_{50} = 2{,}001{,}998 \text{ persons}$$

Method 3

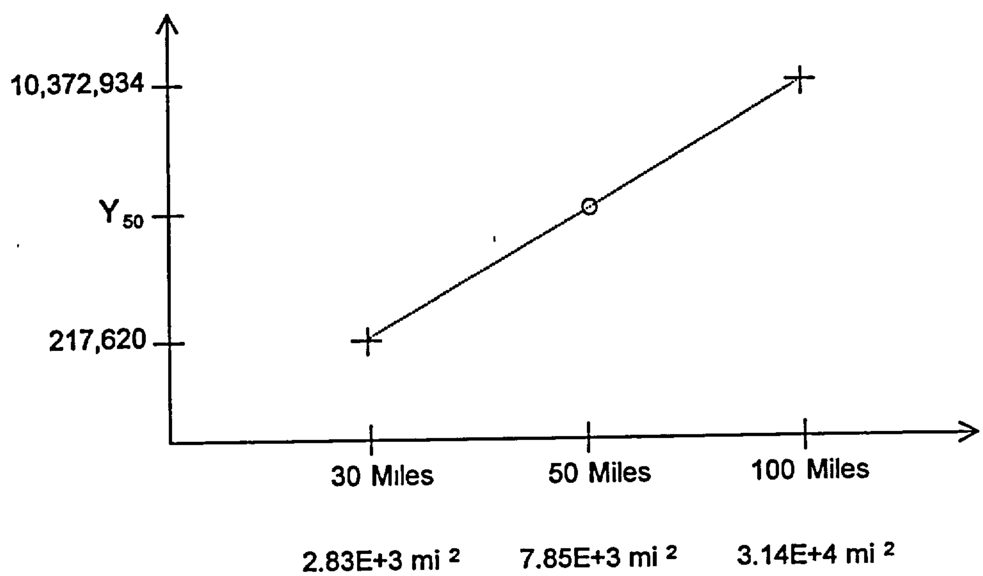
This population estimation method makes use of the 2000 U.S. Census information to back calculate the 50-mile radius population around the LaSalle site in the 1990 time frame. As discussed in the next section, the 2000 U.S. Census information has been analyzed in support of this study to estimate the 50-mile radius population for 2000. From that analysis the following information is available:

- 50-mile radius population around LaSalle for 2000
- Population change compared to 1990

As described in the following section, the 50-mile radius population around LaSalle for 2000 is estimated at 1,553,566 persons.

Figure A-1

LINEAR RELATIONSHIP USED IN
NUREG/CR-5305 POPULATION ESTIMATION METHOD #2



The 2000 U.S. Census data also provides population changes (compared to 1990 U.S. Census data) for discrete municipalities. Table A-2 provides a summary of discrete municipalities within the 50-mile radius of the LaSalle plant along with the population changes between 1990 and 2000. Table A-2 contains the majority of the city population within the 50 mile radius from LaSalle. The population of these discrete municipalities represents approximately 50-55% of the total population within the 50-mile radius of LaSalle. The total percentage change in population of the municipalities in Table A-2 is assumed here to apply uniformly across the entire 50-mile radius. The assumption is made that the growth rate of these municipalities can be taken to be the growth rate for the entire population within 50 miles of LaSalle.

As can be seen from Table A-2, the percentage population change from 1990 to 2000 for the municipalities within the 50-mile radius of LaSalle is +37.3%. Using the 2000 50-mile radius population calculated in the next section, the 1990 50-mile radius population around LaSalle is calculated as follows:

1,553,566 persons / 1.373 = 1,131,512 persons

<u>Summary of NUREG/CR-5305 50-mile Radius Population Estimation</u>

The 50-mile radius population used in the LaSalle NUREG/CR-5305 consequence calculations is required to determine the current consequence estimates to be used in this ILRT risk assessment. As the NUREG/CR-5305 study does not report the 50-mile radius population, three methods have been used here to estimate the population used in the NUREG/CR-5305 study.

The best estimate of the 1990 population within 50 miles can be obtained by using the approximate growth rate for the specific area around LaSalle as determined from Table A-2 which is based on the 1990 and 2000 census.

The best estimate of these three approaches for the 1990 population within 50 miles of LaSalle is judged to be 1,131,512 persons. The value of 1,131,512 persons is used in this risk assessment as the NUREG/CR-5305 50-mile radius population.

Table A-2

2000 CENSUS POPULATION COMPARED TO 1990 FOR MUNICIPALITIES WITHIN 50 MILE RADIUS OF THE LASALLE SITE[1]

(Source: US Census 2000 Redistricting Data Summary File, PL 94-171)

	2000 Census	1990 Census	1990-2000	1990-2000
Municipality	Total Population	Total Population	Population Change	% Change
Aurora city	142,990	99,581	43,409	43.6%
Naperville city	128,358	85,351	43,007	50.4%
Joliet city	106,221	76,836	29,385	38.2%
Bolingbrook village	56,321	40,843	15,478	37.9%
DeKalb city	39,018	34,925	4,093	11.7%
Woodridge village	30,934	26,256	4,678	17.8%
Kankakee city	27,491	27,575	(84)	-0.3%
Batavia city	23,866	17,076	6,790	39.8%
Lisle village	21,182	19,512	1,670	8.6%
Romeoville village	21,153	14,074	7,079	50.3%
Geneva city	19,515	12,617	6,898	54.7%
Ottawa city	18,307	17,451	856	4.9%
New Lenox village	17,771	9,627	8,144	84.6%
Bourbonnais village	15,256	13,934	1,322	9.5%
Lockport city	15,191	9,401	5,790	61.6%
Mokena village	14,583	6,128	8,455	138.0%
Streator city	14,190	14,121	69	0.5%
Crest Hill city	13,329	10,643	2,686	25.2%
Oswego village	13,326	3,876	9,450	243.8%
Lemont village	13,098	7,348	5,750	78.3%
Plainfield village	13,038	4,557	8,481	186.1%
Sycamore city	12,020	9,708	2,312	23.8%
Morris city	11,928	10,270	1,658	16.1%
Pontiac city	11,864	11,428	436	3.8%
North Aurora village	10,585	5,940	4,645	78.2%

[1] The municipalities used in this growth rate determination represent the majority of the city population within 50 miles of the LaSalle plant.

Table A-2

2000 CENSUS POPULATION COMPARED TO 1990 FOR MUNICIPALITIES WITHIN 50 MILE RADIUS OF THE LASALLE SITE[(1)]

(Source: US Census 2000 Redistricting Data Summary File, PL 94-171)

Municipality	2000 Census Total Population	1990 Census Total Population	1990-2000 Population Change	1990-2000 % Change
Frankfort village	10,391	7,180	3,211	44.7%
Marseilles city	4,655	4,811	(156)	-3.2%
Seneca village	2,053	1,878	175	9.3%
Grand Ridge village	546	560	(14)	-2.5%
Ransom village	409	438	(29)	-6.6%
Verona village	257	242	15	6.2%
Kinsman village	109	112	(3)	-2.7%
TOTALS:	829,955	604,299	225,656	37.3%

(1) The municipalities used in this growth rate determination represent the majority of the city population within 50 miles of the LaSalle plant.

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

A calculation of the 2000 50-mile radius population around LaSalle was performed in support of this risk assessment. The calculation is documented in Exelon RM Documentation No. 843. [22]

Calculation RM No. 843 used 2000 Census data, as reported by the US Census Bureau on the web site http://quickfacts.census.gov/qfd/states/17000.html, along with Illinois maps to perform the population estimation.

The LaSalle plant is located in the town of Marseilles in LaSalle County, Illinois. The location of the site and the 50-mile radius is illustrated in Figure A-2 (Figure A-2 is an illustration for discussion purposes – more detailed maps were used in Calculation RM No. 843 to apportion populations). If the entire county falls within the 50-mile radius, based on a review of a map containing a mileage scale and county borders, then the entire population was included in the population estimate. Otherwise, a fraction of the population was counted based on the percentage of the county within the 50-mile radius. The land area within the 50-mile radius was estimated based on visual inspection of the map and the population of that area was estimated assuming uniform distribution of the population within the county.

Five counties were completely inside the fifty-mile radius. For the other counties, their percentage included in the fifty-mile radius was estimated and then multiplied by their total population. Since the population densities within some counties varied greatly, exceptions were made for the following counties: McLean, Kankakee, DeKalb, Cook, Lee, and Will.

Figure A-3

ILLUSTRATION OF 50-MILE RADIUS AROUND LASALLE SITE

McLean County: The fifty-mile radius does not include the cities of Bloomington and Normal with populations of 64,808 and 45,586, respectively (www.suntimes.com/census/cities/). The population of those cities was subtracted from the total population of McLean County then multiplied by 40% for a more accurate count.

Lee County: The only area densely populated is the city of Dixon, which is not included in the fifty-mile radius. The population of Dixon (15,941) was subtracted from the total population of Lee County before multiplying that figure by 60%.

Kankakee County: The major cities of Kankakee, Bradley, and Bourbonnais (27,491, 12,784, and 15,256, respectively) were all included inside the fifty-mile radius in the county of Kankakee, so the total population was multiplied by a higher percentage, 80%.

Dekalb County: The large cities of DeKalb and Sycamore were both included inside the fifty-mile radius in DeKalb County. DeKalb's population not including those two cities was multiplied by 70% and then added to DeKalb and Sycamore's total population.

Cook County: The small portion of Cook County included inside the fifty-mile radius was comprised almost completely of the town, Romeoville. The population of Romeoville (21,153) was used for the Cook County population estimate.

Will County: All major cities were included within the 50 mile zone. The area within the zone was adjusted from 80% to 90% to account for the higher density within the zone.

Based on Exelon RM Documentation No. 843, the total year 2000 population within a 50-mile radius of LaSalle Nuclear Station is estimated at 1,553,566 persons.

Appendix B

LASALLE LERF CET EXTENSION

This appendix discusses modification of the LaSalle Revision 2001A Level 2 PSA LERF models for the purposes of this ILRT risk assessment to obtain additional release categories.

The LaSalle Level 2 PSA containment event tree structure and supporting documentation and analysis are based on the NRC specified requirements in RG 1.174 [B-14] to calculate a Large Early Release Frequency (LERF). The LaSalle Level 2 PSA provides the necessary information in risk-informed application submittals to the NRC as defined by RG 1.174. However, in seeking an exemption to the Integrated Leak Rate Test (ILRT) interval requirements, the NRC staff has requested additional information beyond the LERF estimate. This information includes the frequency of intact containment states along with radionuclide release effects for non-LERF end states. As this ILRT risk assessment requires evaluation of the full range of release magnitudes and timings, the LaSalle LERF model is extended here to address other release categories.

B.1 SUPPLEMENTARY CET NODES

Although the LaSalle Level 2 addresses specifically the LERF risk measure, the model structure and the Level 2 documentation also allows information to be developed regarding other (less severe) types of contributors to radionuclide release.

The approach used to extend the LaSalle LERF Containment Event Tree (CET) models adds additional CET nodes to ask and resolve questions related to other critical safety functions that address the less severe (non-LERF) accident sequences. These supplementary CET nodes are added to the non-LERF accident sequences.

B.1.1 Radionuclide Release Categories

The radionuclide release category definitions are developed in the LaSalle Level 2 PSA documentation. The source term assignments are made using LaSalle specific calculations and BWR Mark II radionuclide release calculations from other industry studies.

The LaSalle Level 2 PSA uses the release severity and timing classification scheme described in Table B-1. The LaSalle LERF model of record is structured to explicitly track and quantify accident sequences resulting in the H/E (High magnitude Early release, i.e., LERF) release category.

B.1.2 Supplementary CET Nodes

The non-LERF accident sequences can be allocated to radionuclide release categories other than LERF (and including intact containment) through the development of supplementary CET nodes. These supplementary CET nodes can be quantified approximately based on the Level 1 cutsets, the previous failures in the CET, and the additional system and phenomenological effects associated with the supplemental nodes.

Figure B-1 shows the supplementary CET nodes that are considered appropriate for the allocation of non-LERF sequences. This CET development is based on numerous previous BWR Mark I and II containment CETs [B-1, 2, 3, 4, 5, 6, 7, 8, 9]. Table B-2 summarizes the definitions of these supplemental nodes.

The supplemental CET structure shown in Figure B-2 is sufficient to establish and answer the critical questions needed to distinguish among non-LERF radionuclide release end states. The quantification of the supplemental nodes (refer to Section B.2) and the assignment of release categories varies with the core damage accident class and CET sequence.

Table B-1

RELEASE SEVERITY AND TIMING CLASSIFICATION SCHEME[1]

Release Severity		Release Timing	
Classification Category	Cs Iodide % in Release	Classification Category	Time of Initial Release[2] Relative to Time for General Emergency Declaration
High (H)	Greater than 10	Late (L)	Greater than 24 hours
Medium or Moderate (M)	1 to 10	Intermediate (I)	6 to 24 hours
Low (L)	0.1 to 1	Early (E)	Less than 6 hours
Low-low (LL)	Less than 0.1		
No iodine (OK)	<<0.1		

(1) The combinations of severity and timing classifications results in one OK release category and 12 other release categories of varying times and magnitudes.

(2) The accident initiation is used as the surrogate for the time when EALs are exceeded.

Time of Release	Magnitude of Release			
	H	M	L	LL
E	H/E[1] (LERF)	M/E	L/E	LL/E
I	H/I	M/I	L/I	LL/I
L	H/L	M/L	L/L	LL/L

(1) LERF is equated to H/E – "high" magnitude of radionuclide release at an "early" time.

Figure B-1 SUPPLEMENTARY CET NODES

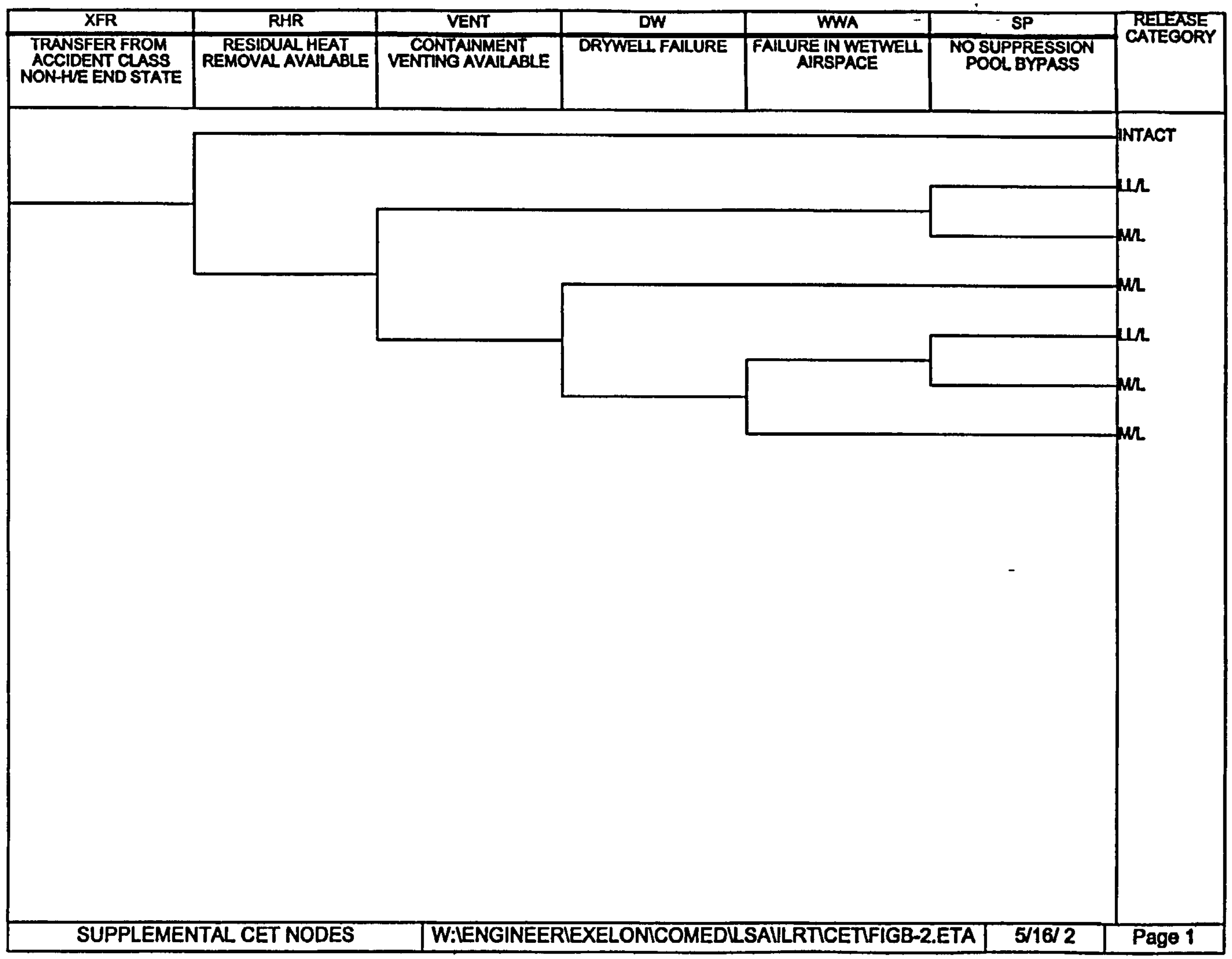


Table B-2

SUPPLEMENTARY CET NODAL DESCRIPTIONS

Node ID	Description
RHR	This node addresses the availability of the RHR system and the operator action to initiate the system for containment heat removal. The RHR system, operating in suppression pool cooling mode, can maintain long term containment integrity through adequate containment heat removal if other failure modes can also be mitigated. The upward branch at this node represents successful containment heat removal via the RHR system operating in the suppression pool cooling mode. Sequences with successful suppression pool cooling lead to an endstate with an intact containment. The downward branch models failure of RHR suppression pool cooling. Sequences with unsuccessful suppression pool cooling will lead to some containment release, either through use of the EOP-directed containment vent or through a containment breach caused by over-temperature and pressure failure.
VENT	This node models use of the wetwell vent to relieve containment pressure in the event of RHR suppression pool cooling failure. Containment venting provides the operator a means of removing decay heat and non-condensible gases, and maintaining containment integrity. The upward branch at this node represents successful use of the containment vent, and release of fission products. Subsequent node SP will determine whether or not the release of fission products is scrubbed by the suppression pool water. The downward branch at this node represents failure of the containment vent. Failure of RHR and VENT will eventually result in containment failure and release of fission products. Subsequent nodes will question whether the containment failure occurs in the drywell or the wetwell, and whether the release is scrubbed by the suppression pool water.
DW	The upward branch of this node indicates containment failure occurs in the drywell. Releases are characterized assuming the drywell failure is at the Drywell head and are in the Moderate magnitude range. The timing of the release is Late given the lengthy time required to overpressurize the primary containment. The downward branch of this node indicates containment failure occurs in the wetwell. Subsequent nodes question whether the wetwell failure occurs in the wetwell airspace or below the waterline, and whether the release is scrubbed by the suppression pool water.

Table B-2

SUPPLEMENTARY CET NODAL DESCRIPTIONS

Node ID	Description
WWA	If the containment failure does not occur in the drywell then it occurs in the wetwell, either in the wetwell airspace region or below the wetwell waterline. The upward branch of this node indicates containment failure occurs in the wetwell airspace. The subsequent SP node questions whether the radionuclide releases are scrubbed or not. The downward branch of this node indicates containment failure occurs below the wetwell waterline. The model assumes that the wetwell failure location is such that the containment breach is not submerged by the pool level. As such, the release associated with this pathway are similar to that of a drywell release.
SP	This node models potential bypass of the containment vapor suppression system (VSS) to determine whether or not releases through the containment vent or via a breach in the wetwell are scrubbed by the pool water. The vapor suppression system (VSS) is composed of the suppression pool, vent pipes, internal ring header, downcomers that connect the drywell to the torus, discharge lines from the relief valves to the suppression pool, the vacuum breakers between the wetwell and the drywell, and the overall boundary between the drywell and the wetwell. The principal function of the VSS is to control containment pressure by condensing steam. In severe accidents in which core damage has occurred, the system also directs potential radionuclide releases to be scrubbed in the suppression pool. The scrubbing of fission products in the suppression pool represents a significant removal mechanism for fission products. The suppression pool can act as an effective scrubber of fission products when it is maintained in the path of radionuclide releases. Possible ways that the suppression pool can be bypassed, and therefore, scrubbing effectiveness diminished, is if: (1) a breach is created between the drywell and the wetwell; (2) wetwell to drywell vacuum breakers fail open; or (3) suppression pool water level decreases below the bottom of the downcomers. If loss of the vapor suppression function (i.e., suppression pool bypass) occurs after the molten core has penetrated the reactor vessel, the effectiveness of continued fission product scrubbing could be compromised. This CET heading is used to estimate the split fraction

Table B-2

SUPPLEMENTARY CET NODAL DESCRIPTIONS

Node ID	Description
SP (Con't)	related to suppression pool bypass; and therefore, to characterize the magnitude of radionuclides that may escape the containment if wetwell failure or venting occurs. The downward branch of this node indicates that radionuclides bypass the suppression pool water due to one or more of the following failures: • Wetwell to drywell vacuum breaker stuck open • Suppression pool water level below the bottom of the downcomers • Vent pipes or downcomers breached during the core melt progression Releases associated with this pathway are similar to that of a drywell release. The upward branch of this node indicates that radionuclides are directed through the suppression pool (i.e., no suppression pool bypass), this requires that none of the above failures occurs. The magnitude of scrubbed releases are two magnitude classifications lower than that of unscrubbed releases.

These supplemental CET nodes are added to the non-LERF sequences of the "no initial containment failure" accident classes (i.e., Class I's, IIIB, and IIIC).

The supplemental CET nodes are not added to accidents in which the containment has already failed (i.e, Classes II, IIID, IV, and V). Sufficient information exists in the LERF CETs for these accident classes to enable assignment of release categories for the non-LERF sequences.

B.2 SUPPLEMENTARY CET NODAL QUANTIFICATION

The LaSalle Level 1 cutset results by accident class were reviewed to identify the dominant contributors to each accident class. Based on these cutsets, the supplemental CET nodes are quantified on a conditional basis. These conditional failure probabilities reflect the functional and support system failures that have occurred in the Level 1 PSA analysis and prior CET nodes. These conditional failure probabilities reflect the dependencies from the Level 1 cutsets and also account for degraded plant conditions and operating environment.

Table B-4 summarizes the quantification of the failure probabilities for the supplemental CET nodes.

B.2 RESULTS OF EXTENDED CETS

The quantified LaSalle extended CETs are provided in Attachment B-1. The results are summarized in Table B-7.

Table B-4

SUMMARY OF SUPPLEMENTAL NODAL QUANTIFICATION (DOWN BRANCHES)

Node ID	Quantification
RHR	The base RHR suppression pool cooling (SPC) failure probability with support systems intact is approximately 1E-3 (based on Level 1 PSA model gate SPC). The failure probability for a single train of RHR SPC is approximately 2E-2 (based on Level 1 PSA model gate RHR-TRAINA-SP). These two failure probabilities are used in most cases for the RHR node. Refer to Table B-5 for a detailed summary of the RHR conditional failure probabilities used in each supplemental CET.
VENT	The conditional failure probability of containment heat removal via venting is dependent on the availability of DC power and Instrument Air. The conditional failure probability of containment venting is negligibly impacted by previous failure of the RHR system. The failure probability for containment venting given SPC failure is approximately 4E-2 (based on Level 1 PSA model gates PCV and SPC). Estimation of the VENT conditional failure probability is based on review on the Level 1 cutsets. In all cases, the conditional failure probability of 1E-1 is used. The 1E-1 value is used instead of the base 4E-2 value to account for the potential increase in the containment venting HEP during post-core damage accident scenarios. Refer to Table B-6 for a detailed summary of the basis for the 1E-1 failure probability for each supplemental CET.
DW	The downward branch of the DW supplemental CET node indicates containment failure occurs in the wetwell. Based on the containment structural evaluation of the Level 2 PSA, the probability of failure in the wetwell (and not in the DW) is 2.47E-1 (0.1172 + 0.1111 + 0.0183) for accident Classes I and III given core melt progression, no containment heat removal but TD = S. (See Table 3.2-3 of the LaSalle Level 2 PSA.).
WWA	The downward branch of the WWA supplemental CET node indicates containment failure below the wetwell waterline. Based on the containment structural evaluation of the LaSalle Level 2 PSA, the conditional probability of failure in the wetwell waterspace (and not the wetwell airspace) is 7.42E-2 (0.0183/(0.1172+0.1111+0.0183)) for accident Classes I and III given core melt progression, no containment heat removal but TD = S. (See Table 3.2-3 of the Level 2 PSA.).

Table B-4

SUMMARY OF SUPPLEMENTAL NODAL QUANTIFICATION (DOWN BRANCHES)

Node ID	Quantification
SP	The following three suppression pool bypass conditional failure probabilities are used: • 2.1E-3 • 2.1E-2 • 1.0 The 2.1E-3 SP failure probability applies to non-LOCA scenarios in which core melt is successfully arrested in-vessel. This failure mode is derived from NRC modeling of fission product transport in the MARCH code in which Sandia postulated a potential bypass mechanism which can occur early in a scenario resulting in high concentration of volatile fission products in the wetwell airspace, and subsequent suppression pool bypass (dominated by the coincidental random failures of SRV discharge vacuum breakers and WW-DW vacuum breakers.) The 2.1E-2 SP failure probability applies to LOCA sequences where steam is discharged directly to the drywell, but where no core debris is discharged to the drywell. The 1.0 SP failure probability applies to scenarios in which the RPV is breached by the core damage progression (these scenarios are addressed in the Page 2 supplemental CETs). As discussed in Section C.6 of the LaSalle Level 2 PSA, the drywell sumps are adequate to hold approximately 30% of the core debris; however, it is estimated that eventually more than 80% of the core debris may be released from the RPV causing the sumps to overflow. The overflowing core debris is postulated to contact and fail (in under an hour following RPV breach) the drywell downcomers, thus leading to suppression pool bypass.

Table B-5

SUMMARY OF SUPPLEMENTAL CET NODE 'RHR' CONDITIONAL PROBABILITIES

Accident Class	Relevant Level 1 Failures	Relevant Prior CET Nodes	RHR Nodal Probability	Bases for Nodal Conditional Probability
IA	• RHR not asked in IA Level 1 accident sequences • Approximately 20% of IA cutsets involve loss of one DC division	An injection source eventually recovered, either: • RX=S: core melt arrested in-vessel, or • RX=F and TD=S: core damage progression melts through RPV, but water source aligned for containment sprays/injection	2E-2	Although RHR is not asked in the Level 1, a significant percentage of Class IA cutsets involve loss of a division of DC. Therefore, it is reasonably assumed that only 1 train of RHR may be available for use. The failure probability for 1 train of RHR is approximately 2E-2.
IBE	• RHR not asked in IBE Level 1 accident sequences • No AC power available in IBE Level 1 scenarios	An injection source eventually recovered, either: • RX=S: core melt arrested in-vessel, or • RX=F and TD=S: core damage progression melts through RPV, but water source aligned for containment sprays/injection	1E-3	Recovery of injection in the Level 2 for IB scenarios is dominated (100% contribution) by offsite AC power recovery. Therefore, the base RHR SPC failure probability (approximately 1E-3) is used.
IBL	• RHR not asked in IBL Level 1 accident sequences • No AC power available in IBL Level 1 scenarios	An injection source eventually recovered, either: • RX=S: core melt arrested in-vessel, or • RX=F and TD=S: core damage progression melts through RPV, but water source aligned for containment sprays/injection	1E-3	Recovery of injection in the Level 2 for IB scenarios is dominated (100% contribution) by offsite AC power recovery. Therefore, the base RHR SPC failure probability (approximately 1E-3) is used.

Table B-5

SUMMARY OF SUPPLEMENTAL CET NODE 'RHR' CONDITIONAL PROBABILITIES

Accident Class	Relevant Level 1 Failures	Relevant Prior CET Nodes	RHR Nodal Probability	Bases for Nodal Conditional Probability
IC	• RHR asked in some IC Level 1 accident sequences • IC cutsets dominated by operator failure to emergency depressurize and not by LP injection equipment failure	An injection source eventually recovered, either: • RX=S: core melt arrested in-vessel, or • RX=F and TD=S: core damage progression melts through RPV, but water source aligned for containment sprays/injection	2E-2	Although some Class IC sequences ask RHR, the majority of Class IC cutsets are due to operator failure to perform RPV emergency depressurization. This nodal probability assumes that at least 1 train of RHR may be available for use. The failure probability for 1 train of RHR is approximately 2E-2.
ID	• RHR asked in ID Level 1 accident sequences • LP ECCS failures present in most, if not all, ID cutsets	An injection source eventually recovered, either: • RX=S: core melt arrested in-vessel, or • RX=F and TD=S: core damage progression melts through RPV, but water source aligned for containment sprays/injection	0.5	RHR has been asked and has failed in the Level 1 Class ID sequences. Although an injection source has been recovered in the Level 2, this nodal probability assumes that the recovered system may not be an RHR train.
IE	• RHR asked in IE Level 1 accident sequences • 100% of IE cutsets involve failure of both divisions of DC	An injection source eventually recovered, either: • RX=S: core melt arrested in-vessel, or • RX=F and TD=S: core damage progression melts through RPV, but water source aligned for containment sprays/injection	2E-2	Recovery of injection in the Level 2 is most likely due to recovery of one division of DC power. Therefore, it is reasonably assumed that only 1 train of RHR may be available for use. The failure probability for 1 train of RHR is approximately 2E-2.

Table B-5

SUMMARY OF SUPPLEMENTAL CET NODE 'RHR' CONDITIONAL PROBABILITIES

Accident Class	Relevant Level 1 Failures	Relevant Prior CET Nodes	RHR Nodal Probability	Bases for Nodal Conditional Probability
IIIB	• RHR not asked in IIIB Level 1 accident sequences • IIIB cutset dominated by operator failure to emergency depressurize	An injection source eventually recovered, either: • RX=S: core melt arrested in-vessel, or • RX=F and TD=S: core damage progression melts through RPV, but water source aligned for containment sprays/injection	1E-3	RHR is not asked in the Level 1 and the Class IIIB cutsets are not dominated by support system failures. Therefore, the base RHR SPC failure probability (approximately 1E-3) is used.
IIIC	• RHR asked in IIIC Level 1 accident sequences • LP ECCS failures present in most, if not all, IIIC cutsets	An injection source eventually recovered, either: • RX=S: core melt arrested in-vessel, or • RX F and TD=S: core damage progression melts through RPV, but water source aligned for containment sprays/injection	0.5	RHR has been asked and has failed in the Level 1 Class IIIC sequences. Although an injection source has been recovered in the Level 2, this nodal probability assumes that the recovered system may not be an RHR train.

Table B-6

SUMMARY OF SUPPLEMENTAL CET NODE 'VENT' CONDITIONAL PROBABILITIES

Accident Class	Relevant Level 1 Failures	Relevant Prior CET Nodes	VENT Nodal Probability	Bases for Nodal Conditional Probability
IA	• Vent not asked in IA Level 1 accident sequences • Approximately 20% of IA cutsets involve loss of one DC division	• An injection source eventually recovered, either: -RX=S: core melt arrested in-vessel, or -RX=F and TD=S: core damage progression melts through RPV, but water source aligned for containment sprays/injection • RHR SPC failed	1E-1	The containment vent is dependent upon Div. I and II AC power and Instrument Air. Failure of RHR SPC has a negligible impact on the failure probability of containment venting. A nominal conditional vent failure probability of 1E-1 is used to account for the potential increase in the vent HEP for post-core damage scenarios (L1 PSA value for vent failure given RHR SPC failure ~4E-2).
IBE	• Vent not asked in IBE Level 1 accident sequences • No AC power available in IBE Level 1 scenarios	• An injection source eventually recovered, either: -RX=S: core melt arrested in-vessel, or -RX=F and TD=S: core damage progression melts through RPV, but water source aligned for containment sprays/injection • RHR SPC failed	1E-1	The containment vent is dependent upon Div. I and II AC power and Instrument Air. Recovery of injection in the Level 2 for IB scenarios is dominated (100% contribution) by offsite AC power recovery. Failure of RHR SPC has a negligible impact on the failure probability of containment venting. A nominal conditional vent failure probability of 1E-1 is used to account for the potential increase in the vent HEP for post-core damage scenarios (L1 PSA value for vent failure given RHR SPC failure ~4E-2).

Table B-6

SUMMARY OF SUPPLEMENTAL CET NODE 'VENT' CONDITIONAL PROBABILITIES

Accident Class	Relevant Level 1 Failures	Relevant Prior CET Nodes	VENT Nodal Probability	Bases for Nodal Conditional Probability
IBL	• Vent not asked in IBL Level 1 accident sequences • No AC power available in IBL Level 1 scenarios	• An injection source eventually recovered, either: -RX=S: core melt arrested in-vessel, or -RX=F and TD=S: core damage progression melts through RPV, but water source aligned for containment sprays/injection • RHR SPC failed	1E-1	The containment vent is dependent upon Div. I and II AC power and Instrument Air. Recovery of injection in the Level 2 for IB scenarios is dominated (100% contribution) by offsite AC power recovery. Failure of RHR SPC has a negligible impact on the failure probability of containment venting. A nominal conditional vent failure probability of 1E-1 is used to account for the potential increase in the vent HEP for post-core damage scenarios (L1 PSA value for vent failure given RHR SPC failure ~4E-2).
IC	• Vent asked in some IC Level 1 accident sequences • IC cutsets dominated by operator failure to emergency depressurize and not by LP injection equipment failure	• An injection source eventually recovered, either: -RX=S: core melt arrested in-vessel, or -RX=F and TD=S: core damage progression melts through RPV, but water source aligned for containment sprays/injection • RHR SPC failed	1E-1	The containment vent is dependent upon Div. I and II AC power and Instrument Air. The majority of Class IC cutsets are due to operator failure to emergency depressurize the RPV. Failure of RHR SPC has a negligible impact on the failure probability of containment venting. A nominal conditional vent failure probability of 1E-1 is used to account for the potential increase in the vent HEP for post-core damage scenarios (L1 PSA value for vent failure given RHR SPC failure ~4E-2).

Table B-6

SUMMARY OF SUPPLEMENTAL CET NODE 'VENT' CONDITIONAL PROBABILITIES

Accident Class	Relevant Level 1 Failures	Relevant Prior CET Nodes	VENT Nodal Probability	Bases for Nodal Conditional Probability
ID	• Vent asked in ID Level 1 accident sequences • LP ECCS failures present in most, if not all, cutsets	• An injection source eventually recovered, either: -RX=S: core melt arrested in-vessel, or -RX=F and TD=S: core damage progression melts through RPV, but water source aligned for containment sprays/injection • RHR SPC failed	1E-1	The containment vent is dependent upon Div. I and II AC power and Instrument Air. A minor percentage of Class ID cutsets contain AC or IA failures that would impact VENT. Failure of RHR SPC has a negligible impact on the failure probability of containment venting. A nominal conditional vent failure probability of 1E-1 is used to account for the potential increase in the vent HEP for post-core damage scenarios (L1 PSA value for vent failure given RHR SPC failure ~4E-2).
IE	• Vent asked in IE Level 1 accident sequences • 100% of IE cutsets involve failure of both divisions of DC	• An injection source eventually recovered, either: -RX=S: core melt arrested in-vessel, or -RX=F and TD=S: core damage progression melts through RPV, but water source aligned for containment sprays/injection • RHR SPC failed	1E-1	100% of the Class IE cutsets are loss of DC events; divisional DC failures have no impact on the VENT failure probability. Failure of RHR SPC has a negligible impact on the failure probability of containment venting. A nominal conditional vent failure probability of 1E-1 is used to account for the potential increase in the vent HEP for post-core damage scenarios (L1 PSA value for vent failure given RHR SPC failure ~4E-2).

Table B-6

SUMMARY OF SUPPLEMENTAL CET NODE 'VENT' CONDITIONAL PROBABILITIES

Accident Class	Relevant Level 1 Failures	Relevant Prior CET Nodes	VENT Nodal Probability	Bases for Nodal Conditional Probability
IIIB	• Vent not asked in IIIB Level 1 accident sequences • Cutset dominated by operator failure to emergency depressurize	• An injection source eventually recovered, either: -RX=S: core melt arrested in-vessel, or -RX=F and TD=S: core damage progression melts through RPV, but water source aligned for containment sprays/injection • RHR SPC failed	1E-1	The containment vent is dependent upon Div. I and II AC power and Instrument Air. Class IIIB cutsets are not dominated by support system failures. Failure of RHR SPC has a negligible impact on the failure probability of containment venting. A nominal conditional vent failure probability of 1E-1 is used to account for the potential increase in the vent HEP for post-core damage scenarios (L1 PSA value for vent failure given RHR SPC failure ~4E-2).
IIIC	• Vent asked in IIIC Level 1 accident sequences • LP ECCS failures present in most, if not all, cutsets	• An injection source eventually recovered, either: -RX=S: core melt arrested in-vessel, or -RX=F and TD=S: core damage progression melts through RPV, but water source aligned for containment sprays/injection • RHR SPC failed	1E-1	The containment vent is dependent upon Div. I and II AC power and Instrument Air. A minor percentage (~10%) of Class IIIB cutsets contain AC or IA failures that would impact VENT. Failure of RHR SPC has a negligible impact on the failure probability of containment venting. A nominal conditional vent failure probability of 1E-1 is used to account for the potential increase in the vent HEP for post-core damage scenarios (L1 PSA value for vent failure given RHR SPC failure ~4E-2).

Table B-7

SUMMARY OF LASALLE UNIT 2 LEVEL 2 PSA RESULTS

Level 1 CDF		LaSalle Level 2 PSA Release Bin Frequencies [1,2]													
Class	CDF	Intact (OK)	LL/E	LL/I	LL/L	L/E	L/I	L/L	M/E	M/I	M/L	H/E	H/I	H/L	Total
IA	3.68E-08	3.34E-08	0.00E+00	0.00E+00	5.73E-10	0.00E+00	0.00E+00	0.00E+00	0.00E+00	2.23E-09	1.08E-10	4.56E-10	0.00E+00	0.00E+00	3.37E-09
IBE	4.35E-07	2.85E-07	0.00E+00	0.00E+00	1.47E-10	0.00E+00	0.00E+00	0.00E+00	0.00E+00	1.43E-07	1.39E-10	6.19E-09	0.00E+00	0.00E+00	1.50E-07
IBL	9.87E-07	5.49E-07	0.00E+00	0.00E+00	2.52E-10	0.00E+00	0.00E+00	0.00E+00	0.00E+00	4.24E-07	2.97E-10	0.00E+00	1.47E-08	0.00E+00	4.39E-07
IC	6.47E-09	6.26E-09	0.00E+00	0.00E+00	1.18E-10	0.00E+00	0.00E+00	0.00E+00	0.00E+00	2.01E-12	1.02E-11	7.44E-11	0.00E+00	0.00E+00	2.04E-10
ID	1.87E-06	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	1.84E-06	0.00E+00	2.81E-08	0.00E+00	0.00E+00	1.87E-06
IE	6.50E-08	3.70E-08	0.00E+00	0.00E+00	5.02E-10	0.00E+00	0.00E+00	0.00E+00	0.00E+00	2.62E-08	2.53E-10	9.16E-10	0.00E+00	0.00E+00	2.79E-08
II [3]	1.84E-06	0.00E+00	0.00E+00	1.35E-07	0.00E+00	0.00E+00	1.21E-06	0.00E+00	0.00E+00	4.44E-08	0.00E+00	0.00E+00	4.50E-07	0.00E+00	1.84E-06
IIIB	4.39E-09	4.33E-09	0.00E+00	0.00E+00	3.92E-12	0.00E+00	0.00E+00	0.00E+00	0.00E+00	6.05E-13	4.19E-13	5.04E-11	0.00E+00	0.00E+00	5.54E-11
IIIC	9.09E-08	2.77E-08	0.00E+00	0.00E+00	1.30E-08	0.00E+00	0.00E+00	0.00E+00	0.00E+00	3.41E-08	1.47E-08	1.30E-09	0.00E+00	0.00E+00	6.31E-08
IIID	6.96E-08	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	6.96E-08	0.00E+00	0.00E+00	6.96E-08
IV [4]	1.81E-07	0.00E+00	1.23E-08	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	7.63E-08	0.00E+00	0.00E+00	9.24E-08	0.00E+00	0.00E+00	1.81E-07
V	7.12E-08	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	7.12E-08	0.00E+00	0.00E+00	7.12E-08
Total:	5.66E-06	9.43E-07	1.23E-08	1.35E-07	1.46E-08	0.00E+00	1.21E-06	0.00E+00	7.63E-08	2.52E-06	1.55E-08	2.70E-07	4.64E-07	0.00E+00	4.72E-06
% of Total CDF:		16.7	0.2	2.4	0.3	0.0	21.5	0.0	1.3	44.5	0.3	4.8	8.2	0.0	100.0
% of Total Release:		n/a	0.3	2.9	0.3	0.0	25.7	0.0	1.6	53.3	0.3	5.7	9.8	0.0	100.0

Notes to Table B-7:

(1) Release bin nomenclature is [Release Magnitude]/[Timing of Release], where:

LL:	Low-Low	E:	Early
L:	Low	I:	Intermediate
M:	Moderate	L:	Late
H:	High		

(2) The LaSalle Revision 2001A Level 2 PSA models internal transients, LOCAs, internal flooding scenarios, and seismic-induced accident sequences.

(3) Includes all Class II subcategories.

(4) Includes contributions from Class IVL.

REFERENCES

[B-1] *Peach Bottom Level 2 Individual Plant Examination (IPE)*, ERIN Engineering and Research, Inc. for Philadelphia Electric Company, February 1992.

[B-2] *Nine Mile Point Unit 1 Level 2 Individual Plant Examination (IPE)*, ERIN Engineering and Research, Inc. for Niagara Mohawk Power Corporation, March 1994.

[B-3] *Duane Arnold Energy Center Level 2 Individual Plant Examination (IPE)*, ERIN Engineering and Research, Inc. for Iowa Electric Light and Power Company, December 1992.

[B-4] *Cooper Nuclear Station Level 2 PRA*, NPPD, 1998.

[B-5] *Fermi 2 Level 2 Individual Plant Examination (IPE)*, ERIN Engineering and Research, Inc. for Detroit Edison Company, September 1991.

[B-6] *Limerick Level 2 Individual Plant Examination (IPE)*, ERIN Engineering and Research, Inc. for Philadelphia Electric Company, March 1992.

[B-7] *Quad Cities* Level *2/LERF Evaluation*, ERIN Engineering and Research, Inc. for ComEd, July 1999.

[B-8] *Nine Mile Point Unit 2 Level 2 Individual Plant Examination (IPE)*, ERIN Engineering and Research, Inc. for Niagara Mohawk Power Corporation, January 1992.

[B-9] *Brunswick Level 2/LERF Evaluation*, ERIN Report No. C1100001-4265, November 2000.

[B-10] *Performance-Based Containment Leak-Test Program*, NUREG-1493, September 1995.

[B-11] Letter from R.J. Barrett (Entergy) to U.S. Nuclear Regulatory Commission, IPN-01-007, dated January 18, 2001.

[B-12] *Risk Impact Assessment of Revised Containment Leak Rate Testing Intervals*, EPRI, Palo Alto, CA EPRI TR-104285, August 1994.

[B-13] 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.

[B-14] Regulatory Guide 1.174, *An Approach for Using Probabilistic Risk Assessment in Risk-Informed Decisions on Plant-Specific Changes to the Licensing Basis*, July 1998.

Attachment B1

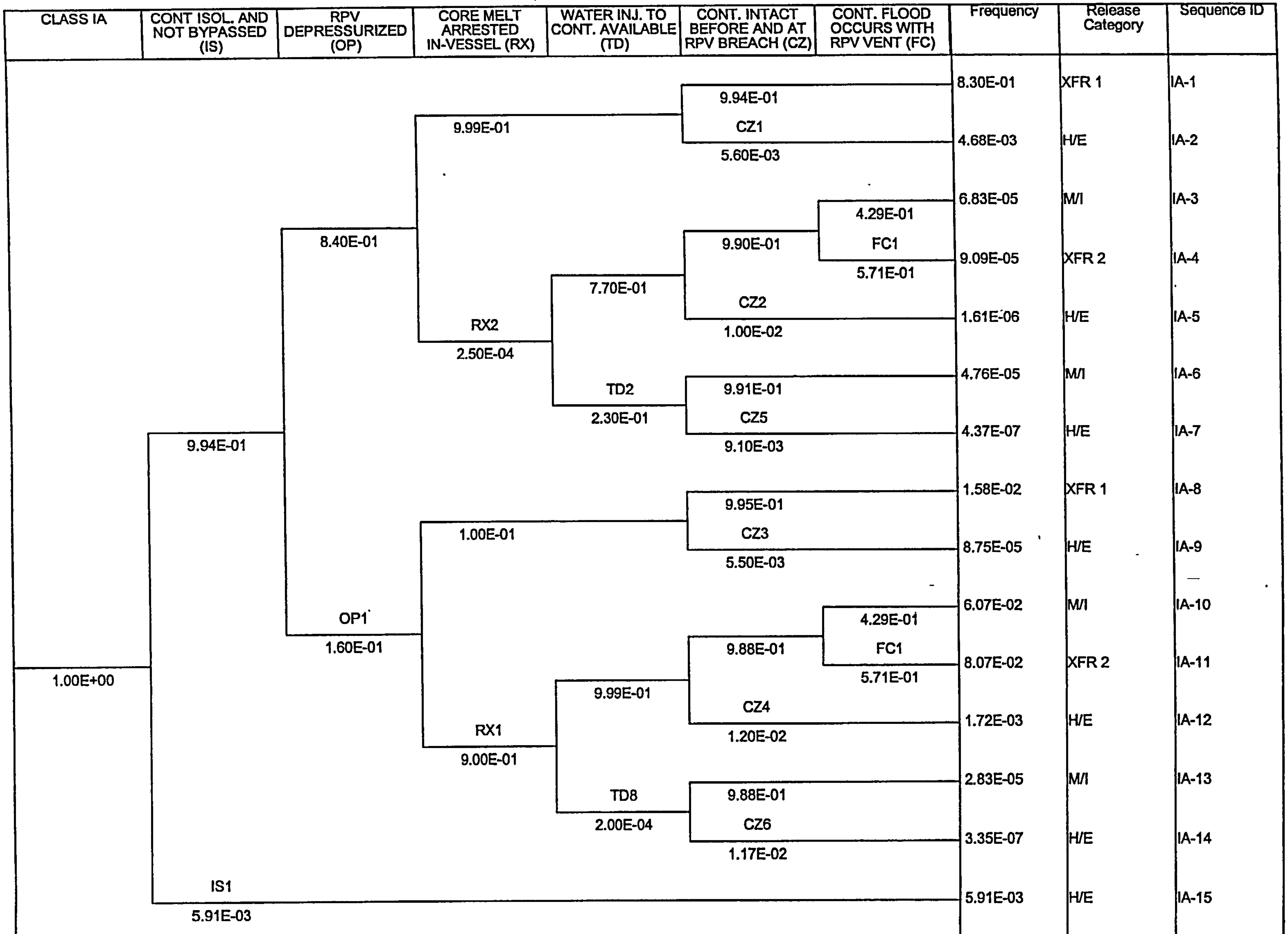
QUANTIFIED LASALLE EXTENDED CETs

This attachment provides the quantified LaSalle extended containment event trees. The following quantified CETs are included in this attachment:

- Class IA CET
- Supplemental CET for Class IA (Page 1)
- Supplemental CET for Class IA (Page 2)
- Class IBE CET
- Supplemental CET for Class IBE (Page 1)
- Supplemental CET for Class IBE (Page 2)
- Class IBL CET
- Supplemental CET for Class IBL (Page 1)
- Supplemental CET for Class IBL (Page 2)
- Class IC CET
- Supplemental CET for Class IC (Page 1)
- Supplemental CET for Class IC (Page 2)
- Class ID CET
- Supplemental CET for Class ID (Page 1)
- Supplemental CET for Class ID (Page 2)
- Class IE CET
- Supplemental CET for Class IE (Page 1)
- Supplemental CET for Class IE (Page 2)
- Class II CET
- Class IIIB CET
- Supplemental CET for Class IIIB (Page 1)
- Supplemental CET for Class IIIB (Page 2)
- Class IIIC CET

- Supplemental CET for Class IIIC (Page 1)
- Supplemental CET for Class IIIC (Page 2)
- Class IID CET
- Class IV CET
- Class V CET

As the CETs use only point estimates (i.e., no cutsets or fault tree logic are input into these CETs), the CETs are developed and quantified using the ETA event tree code. As can be seen from the attached quantified CETs, the incoming accident class information for each CET is entered as a 1.00 point estimate. As such, the CETs calculate conditional release categories. The individual sequences are summed according to release category and the totals are then multiplied in a spreadsheet by the individual accident class subtotals to determine the release category frequencies. The results are summarized in Table B-7.



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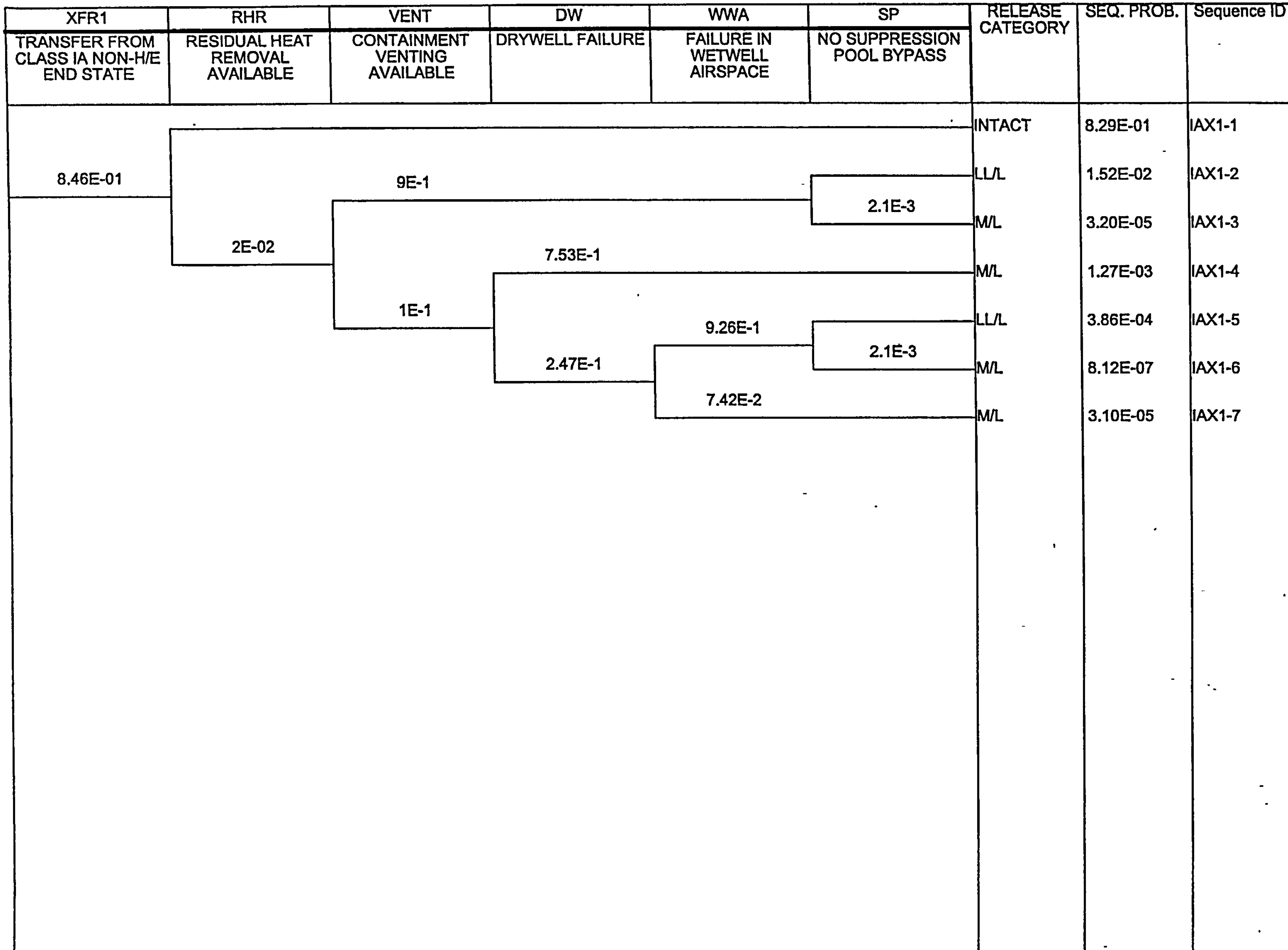

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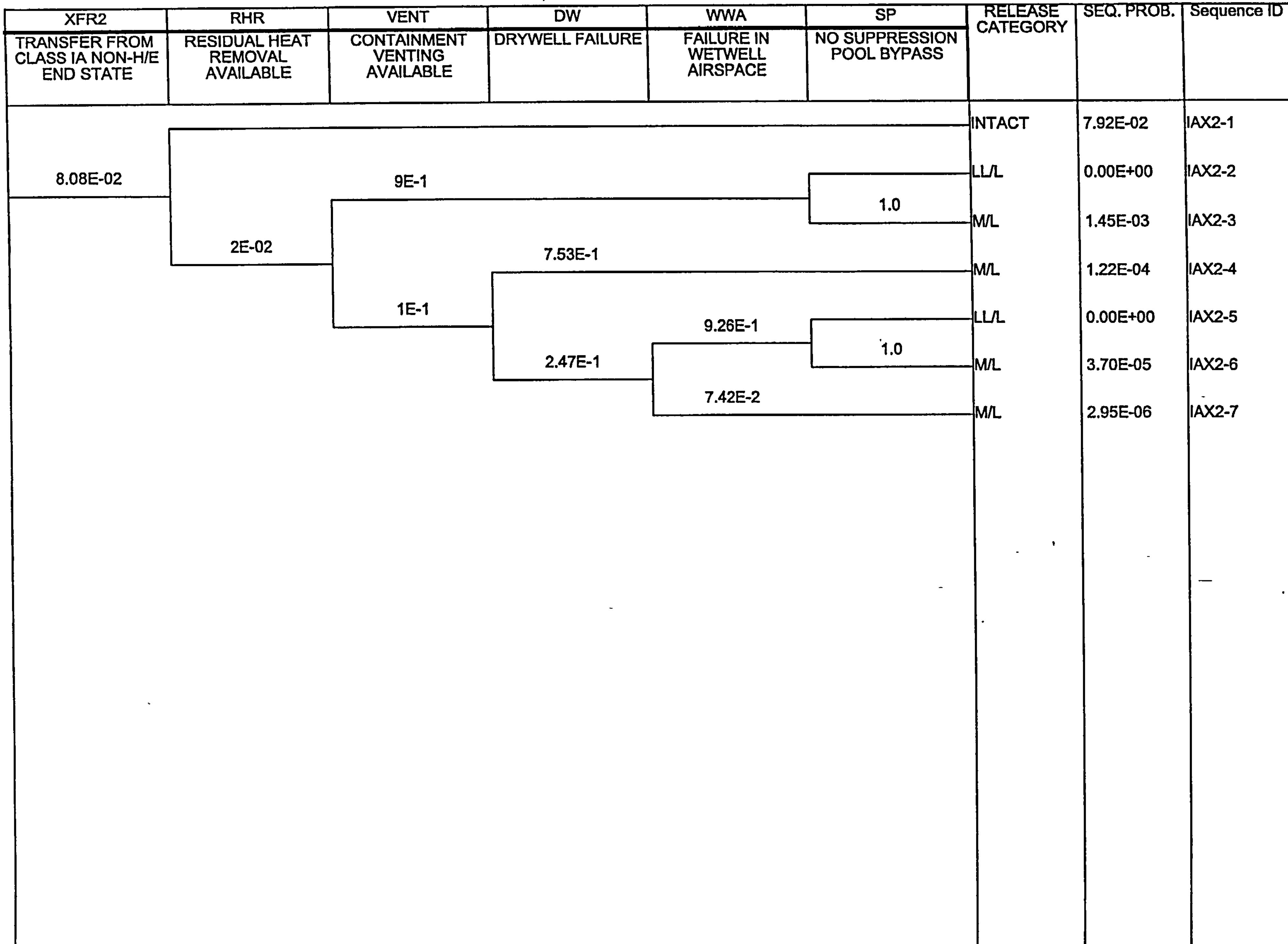
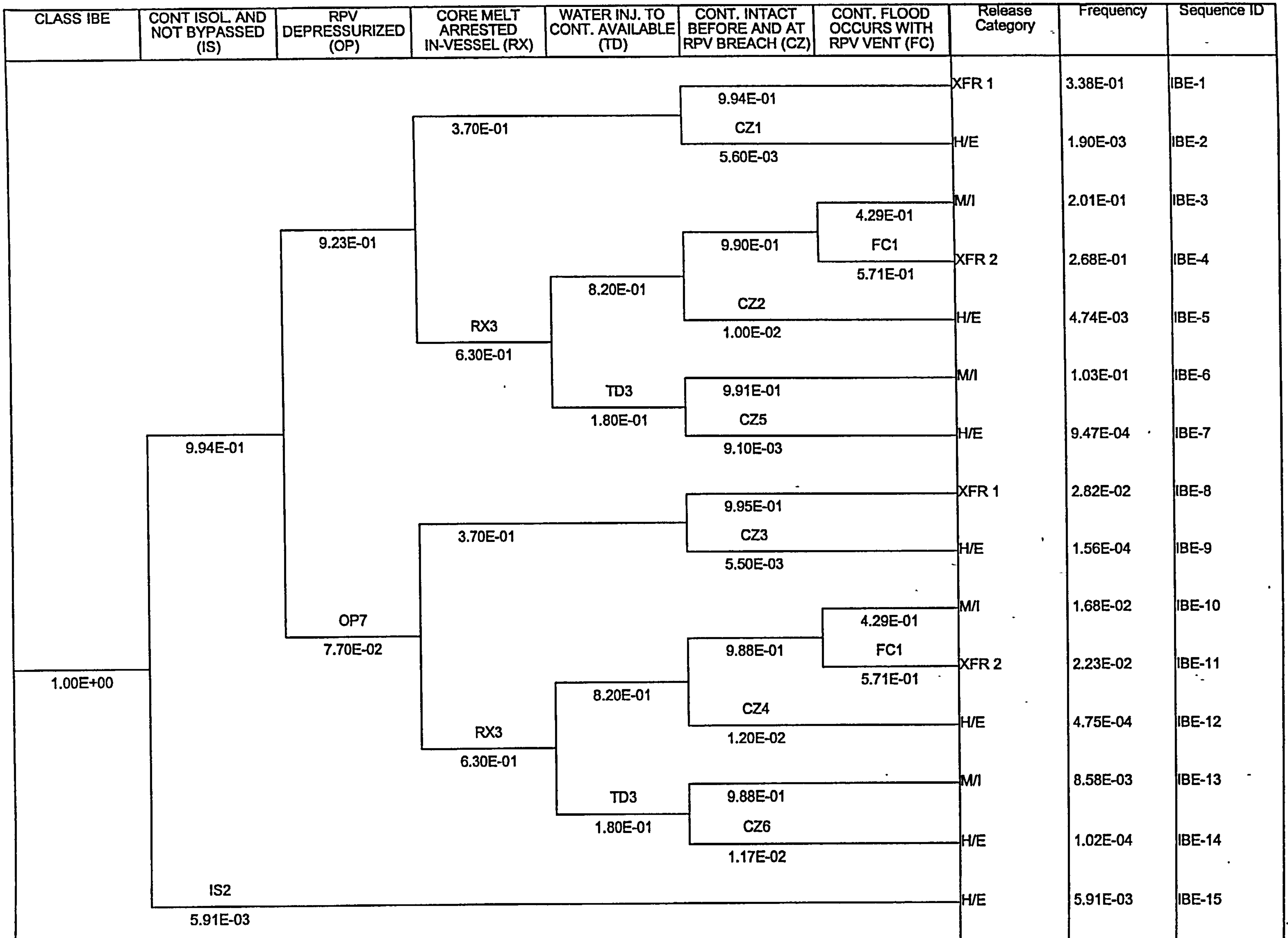


SUPPL. CET NODES FOR CLASS IA - Page 2	W:\ENGINEER\EXELON\COMED\LSA\ILRT\CET\IAX2.ETA	5/21/ 2	Page 1


CLASS IBE
CONT ISOL. AND NOT BYPASSED (IS)
RPV DEPRESSURIZED (OP)
CORE MELT ARRESTED IN-VESSEL (RX)
WATER INJ. TO CONT. AVAILABLE (TD)
CONT. INTACT BEFORE AND AT RPV BREACH (CZ)
CONT. FLOOD OCCURS WITH RPV VENT (FC)
Release Category
Frequency
Sequence ID
1.00E+00
9.94E-01
IS2
5.91E-03
9.23E-01
OP7
7.70E-02
3.70E-01
RX3
6.30E-01
8.20E-01
TD3
1.80E-01
9.94E-01
CZ1
5.60E-03
9.90E-01
CZ2
1.00E-02
9.91E-01
CZ5
9.10E-03
4.29E-01
FC1
5.71E-01
3.70E-01
RX3
6.30E-01
8.20E-01
TD3
1.80E-01
9.95E-01
CZ3
5.50E-03
9.88E-01
CZ4
1.20E-02
9.88E-01
CZ6
1.17E-02
4.29E-01
FC1
5.71E-01
XFR 1 3.38E-01 IBE-1
H/E 1.90E-03 IBE-2
M/I 2.01E-01 IBE-3
XFR 2 2.68E-01 IBE-4
H/E 4.74E-03 IBE-5
M/I 1.03E-01 IBE-6
H/E 9.47E-04 IBE-7
XFR 1 2.82E-02 IBE-8
H/E 1.56E-04 IBE-9
M/I 1.68E-02 IBE-10
XFR 2 2.23E-02 IBE-11
H/E 4.75E-04 IBE-12
M/I 8.58E-03 IBE-13
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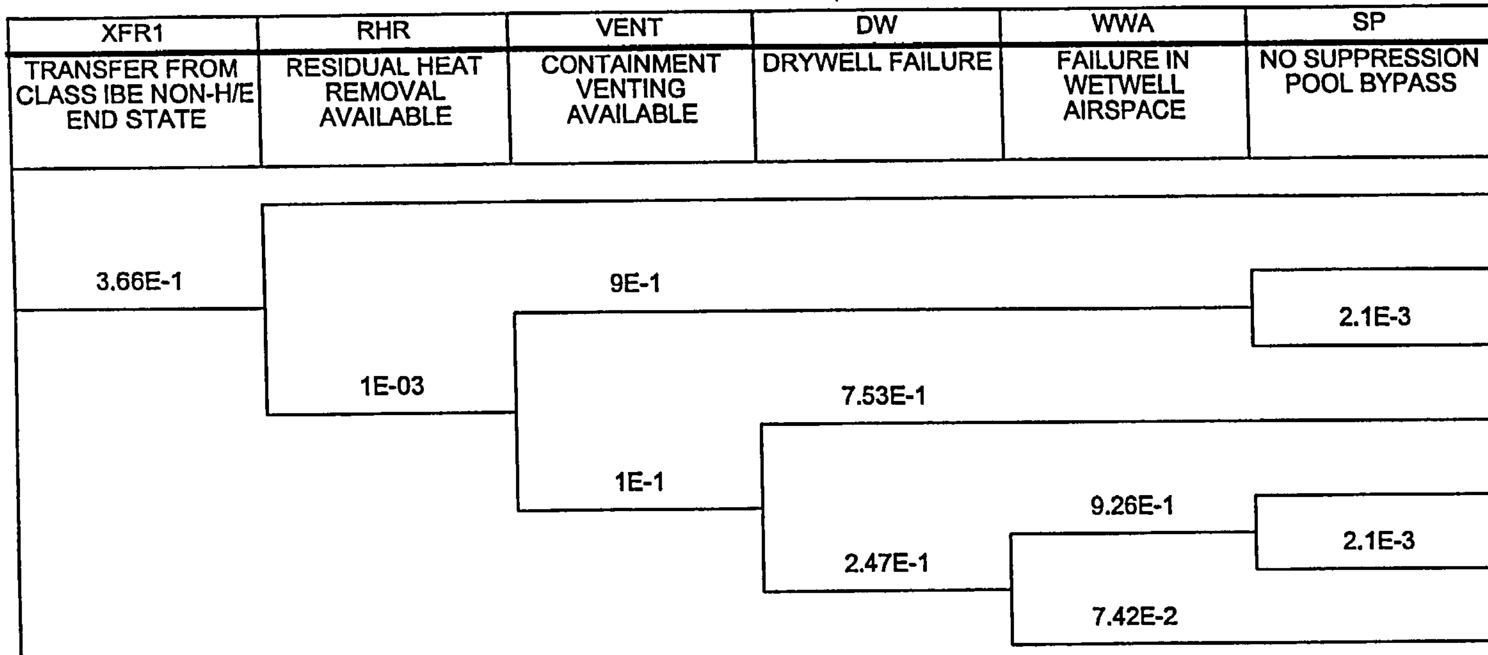
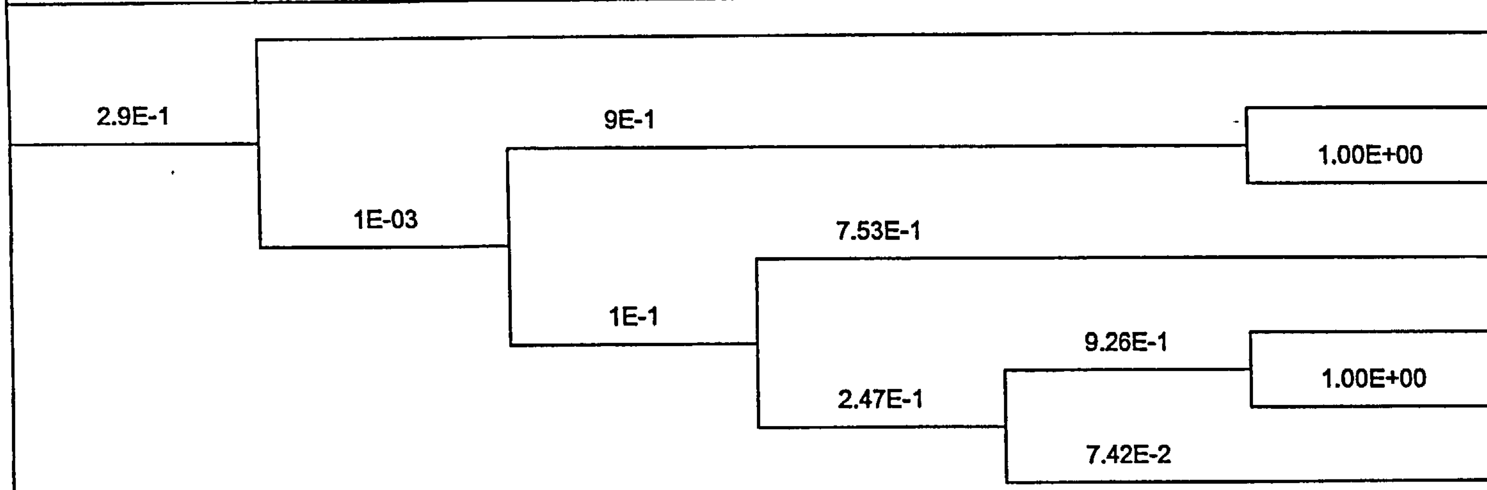
H/E 5.91E-03 IBE-15

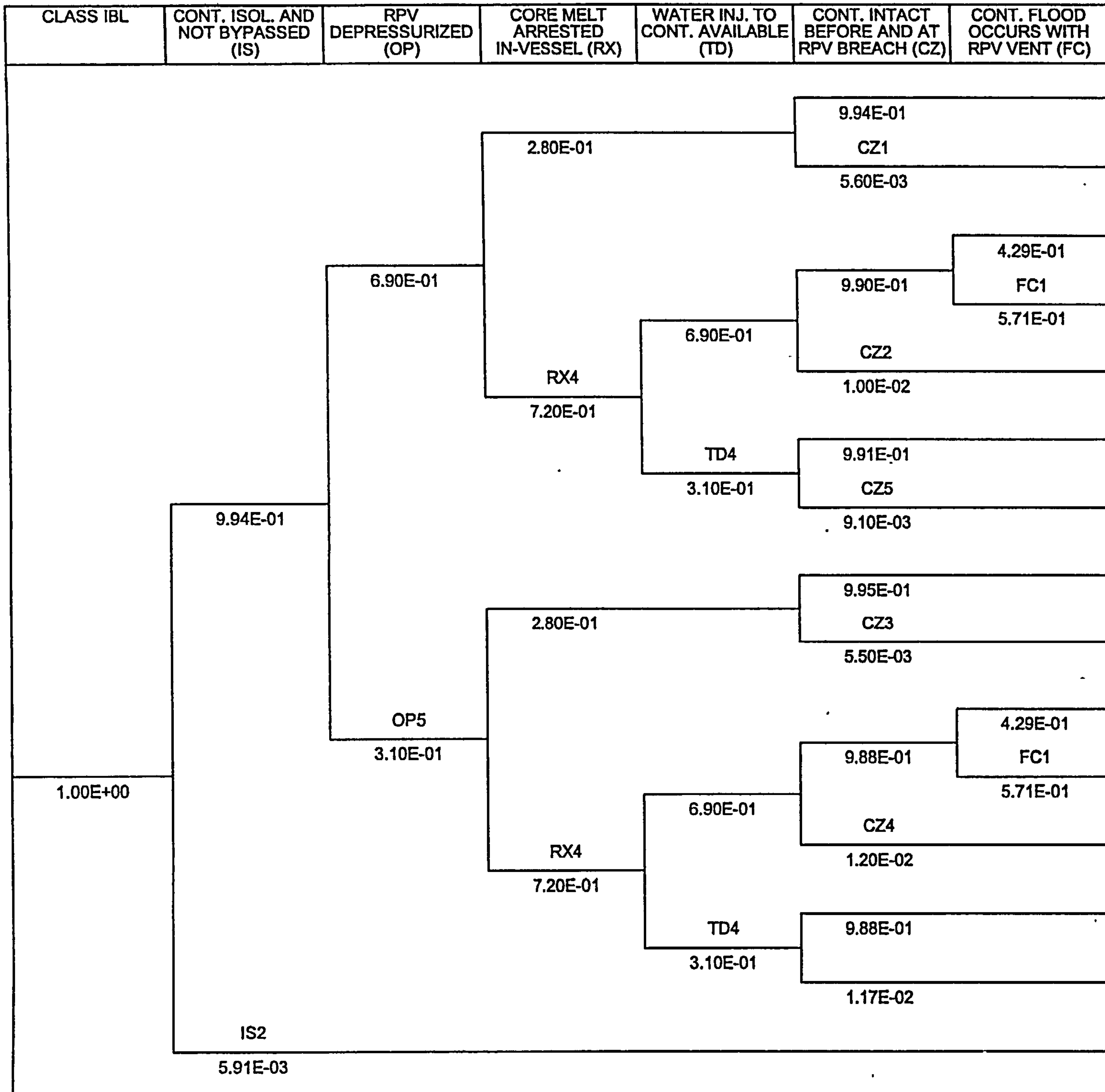


RELEASE CATEGORY	SEQ. PROB.	Sequence ID
INTACT	3.66E-01	IBEX1-1
LL/L	3.29E-04	IBEX1-2
M/L	6.92E-07	IBEX1-3
M/L	2.76E-05	IBEX1-4
LL/L	8.35E-06	IBEX1-5
M/L	1.76E-08	IBEX1-6
M/L	6.69E-07	IBEX1-7

XFR2	RHR	VENT	DW	WWA	SP
TRANSFER FROM CLASS IBE NON-H/E END STATE	RESIDUAL HEAT REMOVAL AVAILABLE	CONTAINMENT VENTING AVAILABLE	DRYWELL FAILURE	FAILURE IN WETWELL AIRSPACE	NO SUPPRESSION POOL BYPASS

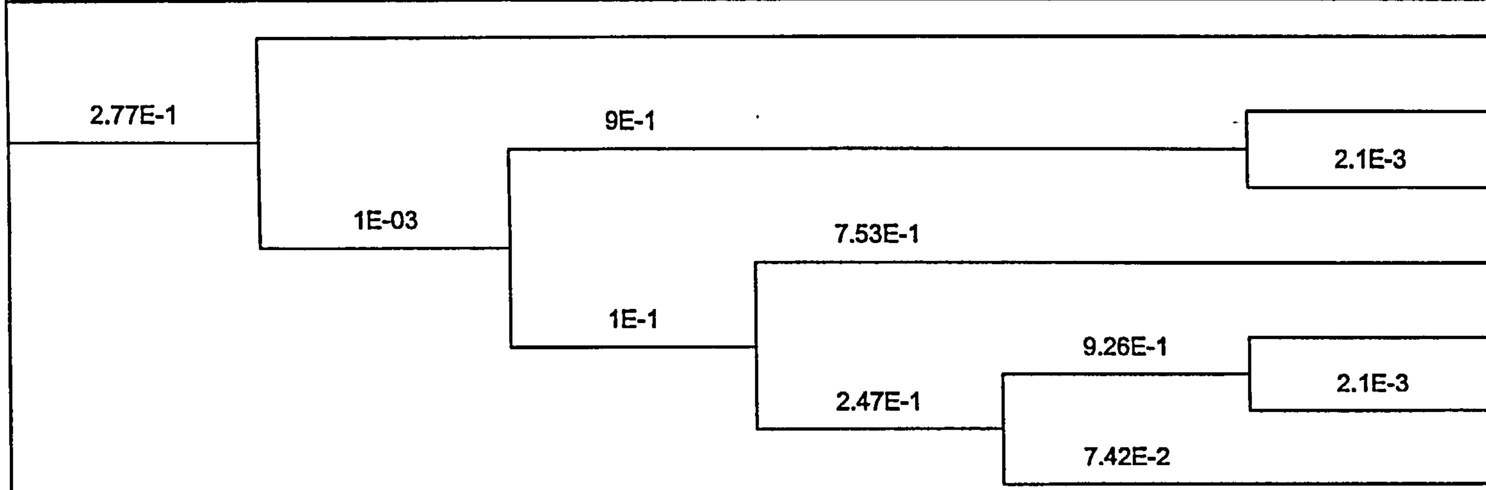


RELEASE CATEGORY	SEQ. PROB.	Sequence ID
INTACT	2.90E-01	IBEX2-1
LL/L	0.00E+00	IBEX2-2
M/L	2.61E-04	IBEX2-3
M/L	2.18E-05	IBEX2-4
LL/L	0.00E+00	IBEX2-5
M/L	6.63E-06	IBEX2-6
M/L	5.30E-07	IBEX2-7

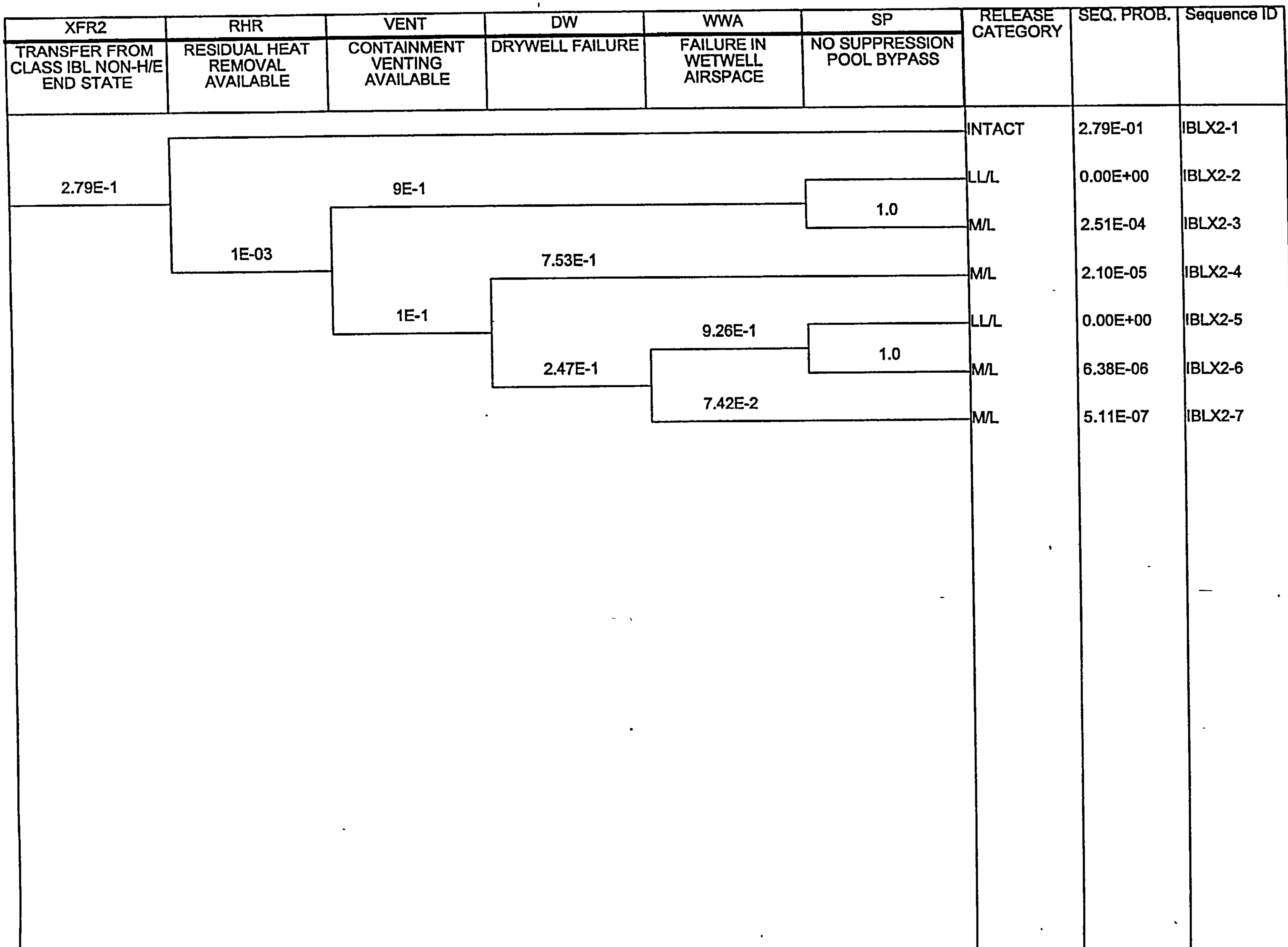


Release Category	Frequency	Sequence ID
XFR 1	1.91E-01	IBL-1
H/I	1.08E-03	IBL-2
M/I	1.45E-01	IBL-3
XFR 2	1.93E-01	IBL-4
H/I	3.41E-03	IBL-5
M/I	1.52E-01	IBL-6
H/I	1.39E-03	IBL-7
XFR 1	8.58E-02	IBL-8
H/I	4.75E-04	IBL-9
M/I	6.49E-02	IBL-10
XFR 2	8.64E-02	IBL-11
H/I	1.84E-03	IBL-12
M/I	6.80E-02	IBL-13
H/I	8.05E-04	IBL-14
H/I	5.91E-03	IBL-15

XFR1	RHR	VENT	DW	WWA	SP
TRANSFER FROM CLASS IBL NON-H/E END STATE	RESIDUAL HEAT REMOVAL AVAILABLE	CONTAINMENT VENTING AVAILABLE	DRYWELL FAILURE	FAILURE IN WETWELL AIRSPACE	NO SUPPRESSION POOL BYPASS



RELEASE CATEGORY	SEQ. PROB.	Sequence ID
INTACT	2.77E-01	IBLX1-1
LL/L	2.49E-04	IBLX1-2
M/L	5.24E-07	IBLX1-3
M/L	2.09E-05	IBLX1-4
LL/L	6.32E-06	IBLX1-5
M/L	1.33E-08	IBLX1-6
M/L	5.08E-07	IBLX1-7



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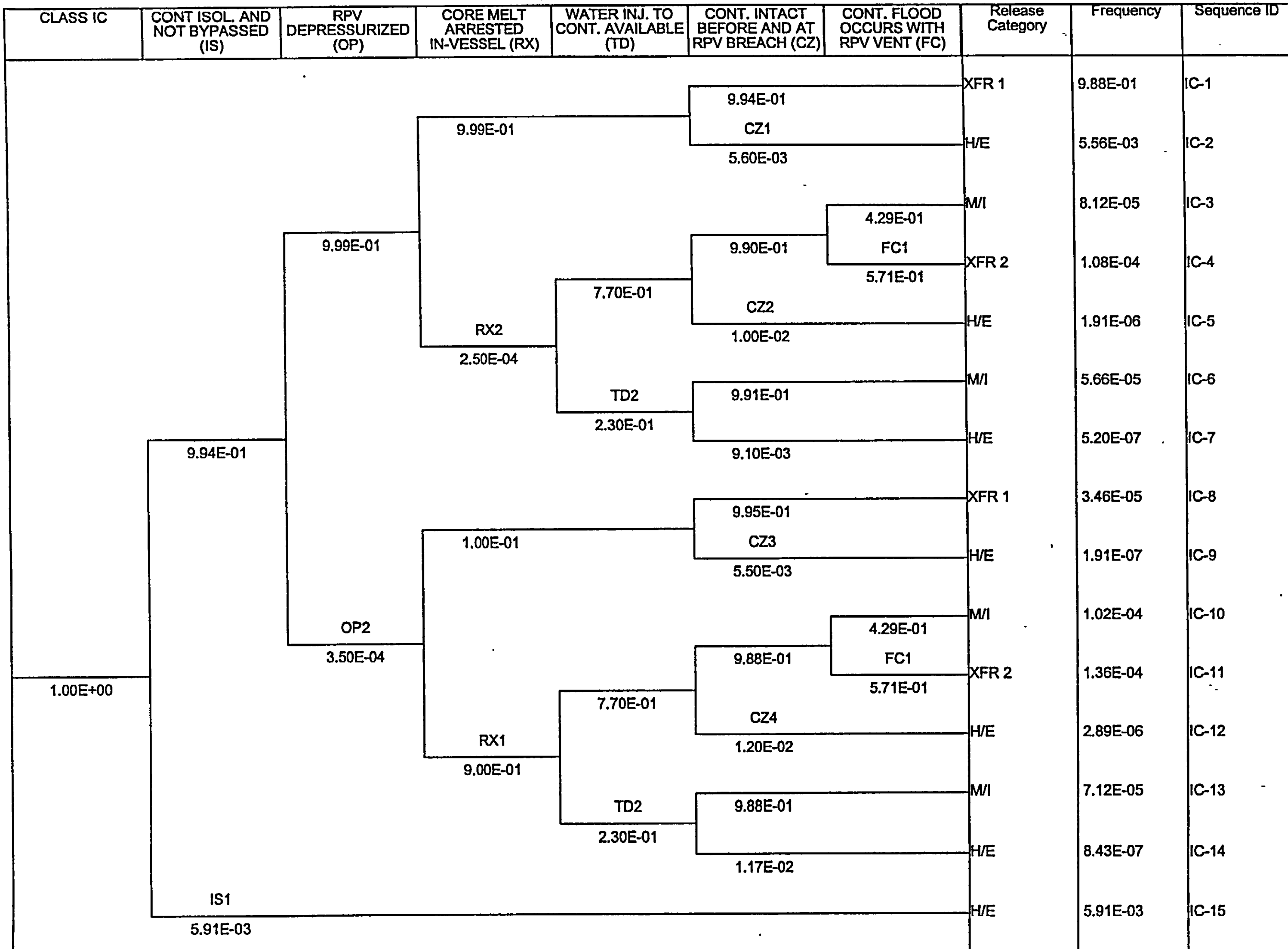
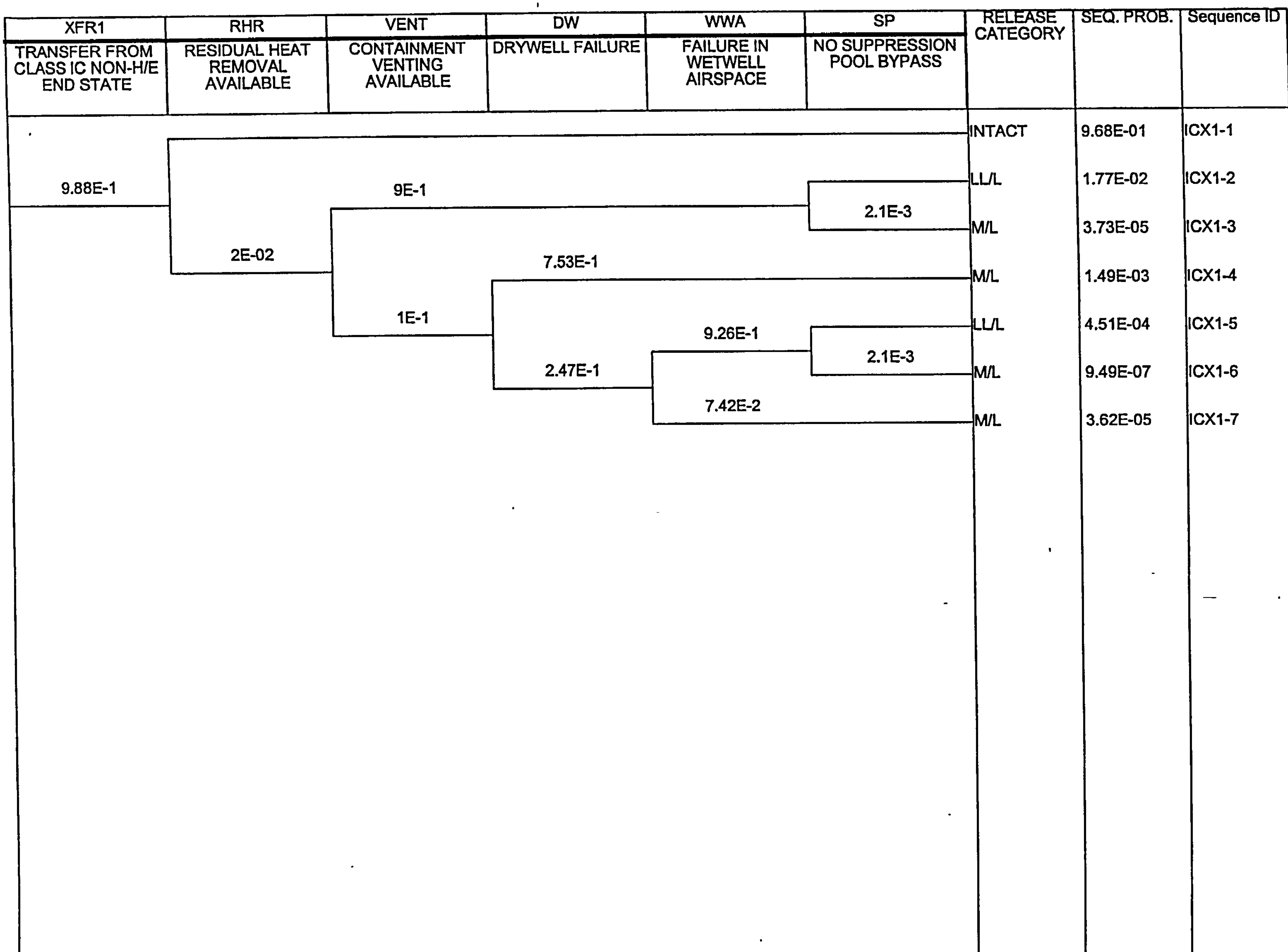


CLASS IC	W:\ENGINEER\EXELON\COMED\LSA\ILRT\CET\IC.ETA	5/21/ 2	Page 1



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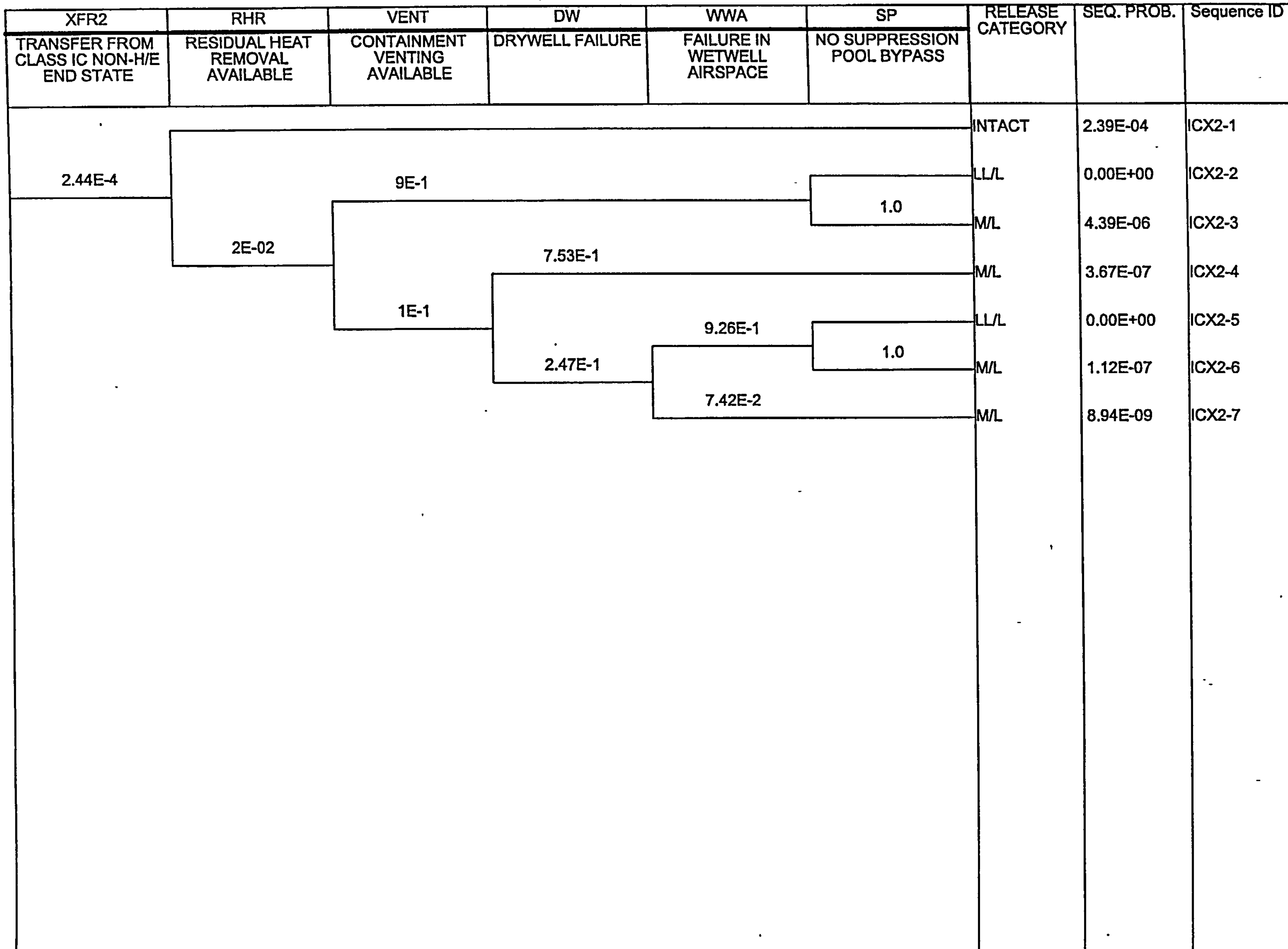

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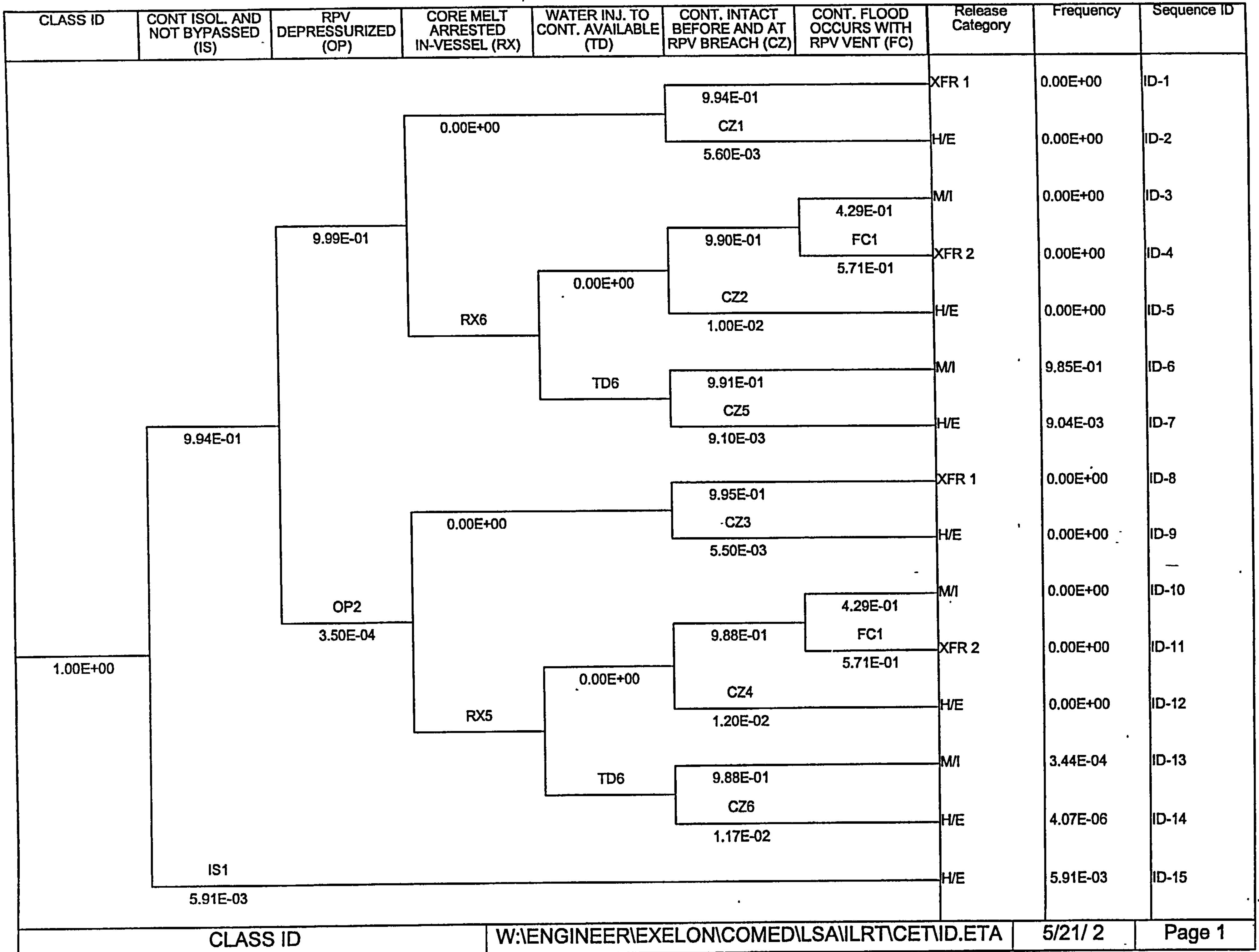
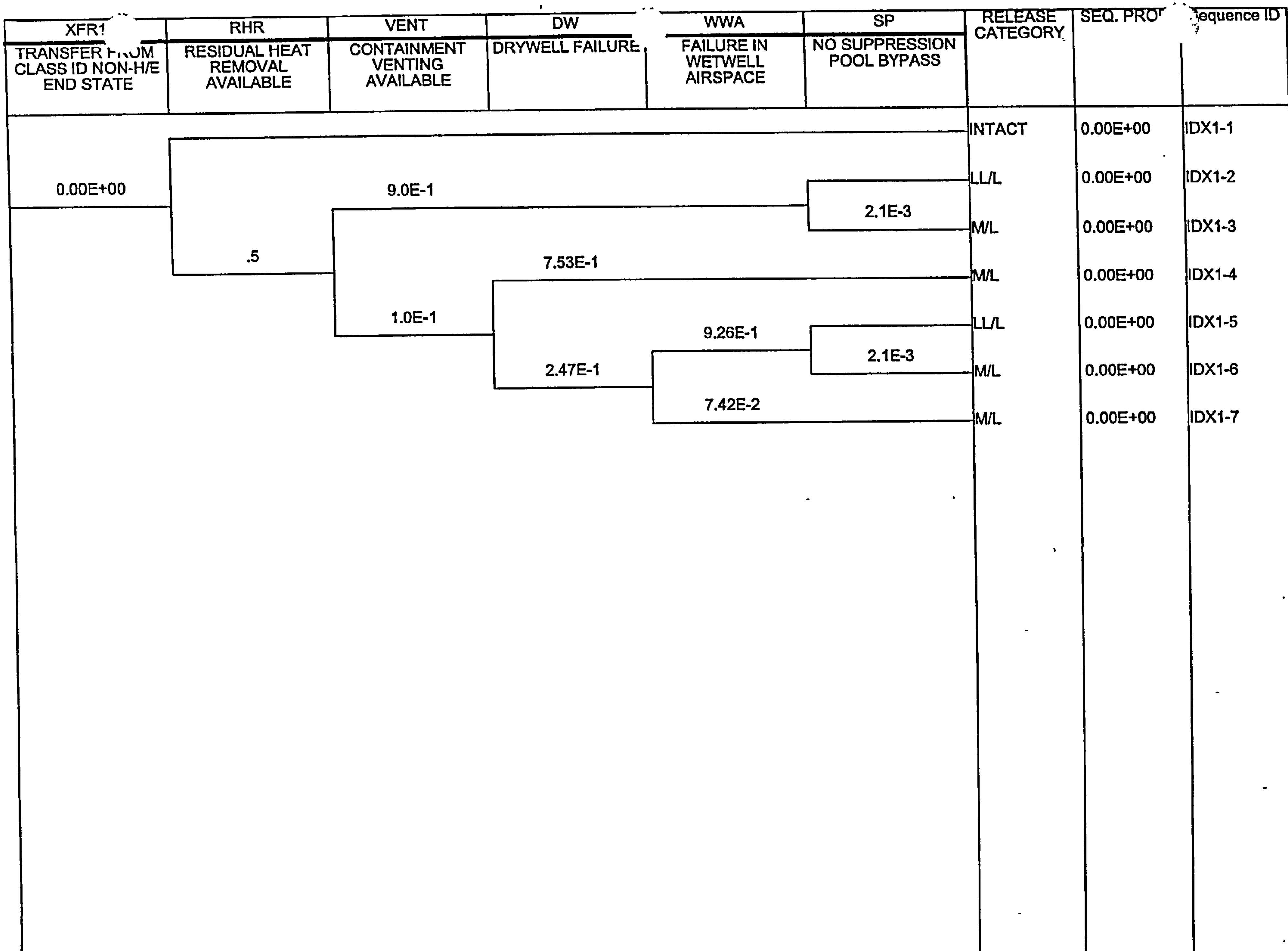
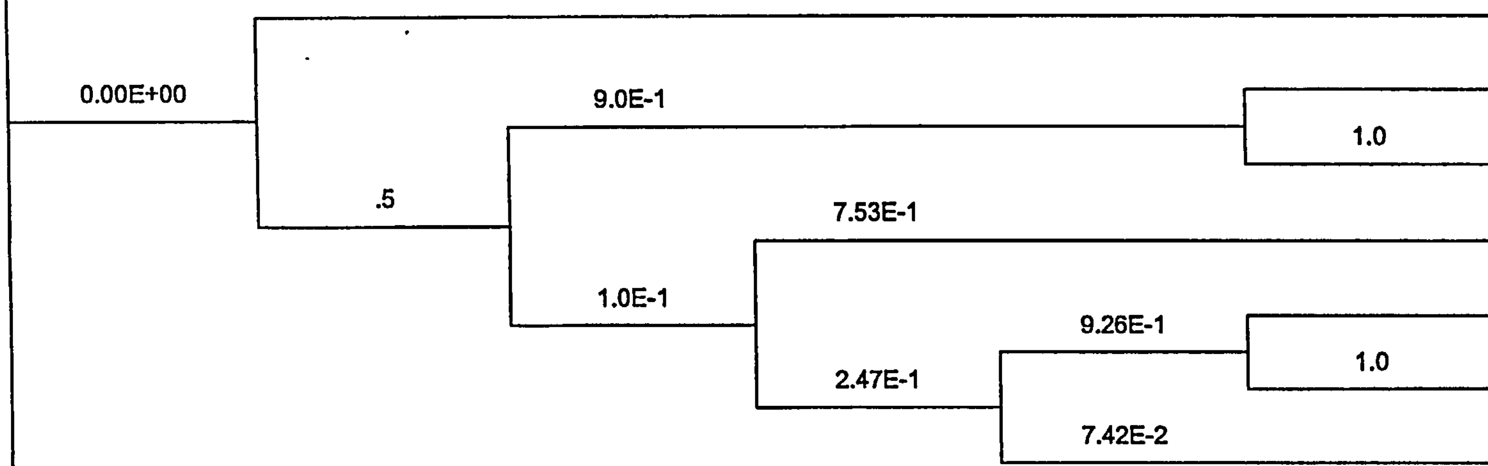

CLASS ID
CONT ISOL. AND NOT BYPASSED (IS)
RPV DEPRESSURIZED (OP)
CORE MELT ARRESTED IN-VESSEL (RX)
WATER INJ. TO CONT. AVAILABLE (TD)
CONT. INTACT BEFORE AND AT RPV BREACH (CZ)
CONT. FLOOD OCCURS WITH RPV VENT (FC)
Release Category
Frequency
Sequence ID
1.00E+00
9.94E-01
9.99E-01
0.00E+00
9.94E-01
CZ1
5.60E-03
XFR 1
0.00E+00
ID-1
H/E
0.00E+00
ID-2
RX6
0.00E+00
9.90E-01
CZ2
1.00E-02
4.29E-01
FC1
5.71E-01
M/I
0.00E+00
ID-3
XFR 2
0.00E+00
ID-4
H/E
0.00E+00
ID-5
TD6
9.91E-01
CZ5
9.10E-03
M/I
9.85E-01
ID-6
H/E
9.04E-03
ID-7
OP2
3.50E-04
0.00E+00
9.95E-01
CZ3
5.50E-03
XFR 1
0.00E+00
ID-8
H/E
0.00E+00
ID-9
RX5
0.00E+00
9.88E-01
CZ4
1.20E-02
4.29E-01
FC1
5.71E-01
M/I
0.00E+00
ID-10
XFR 2
0.00E+00
ID-11
H/E
0.00E+00
ID-12
TD6
9.88E-01
CZ6
1.17E-02
M/I
3.44E-04
ID-13
H/E
4.07E-06
ID-14
IS1
5.91E-03
H/E
5.91E-03
ID-15
CLASS ID
W:\ENGINEER\EXELON\COMED\LSA\ILRT\CET\ID.ETA
5/21/ 2
Page 1

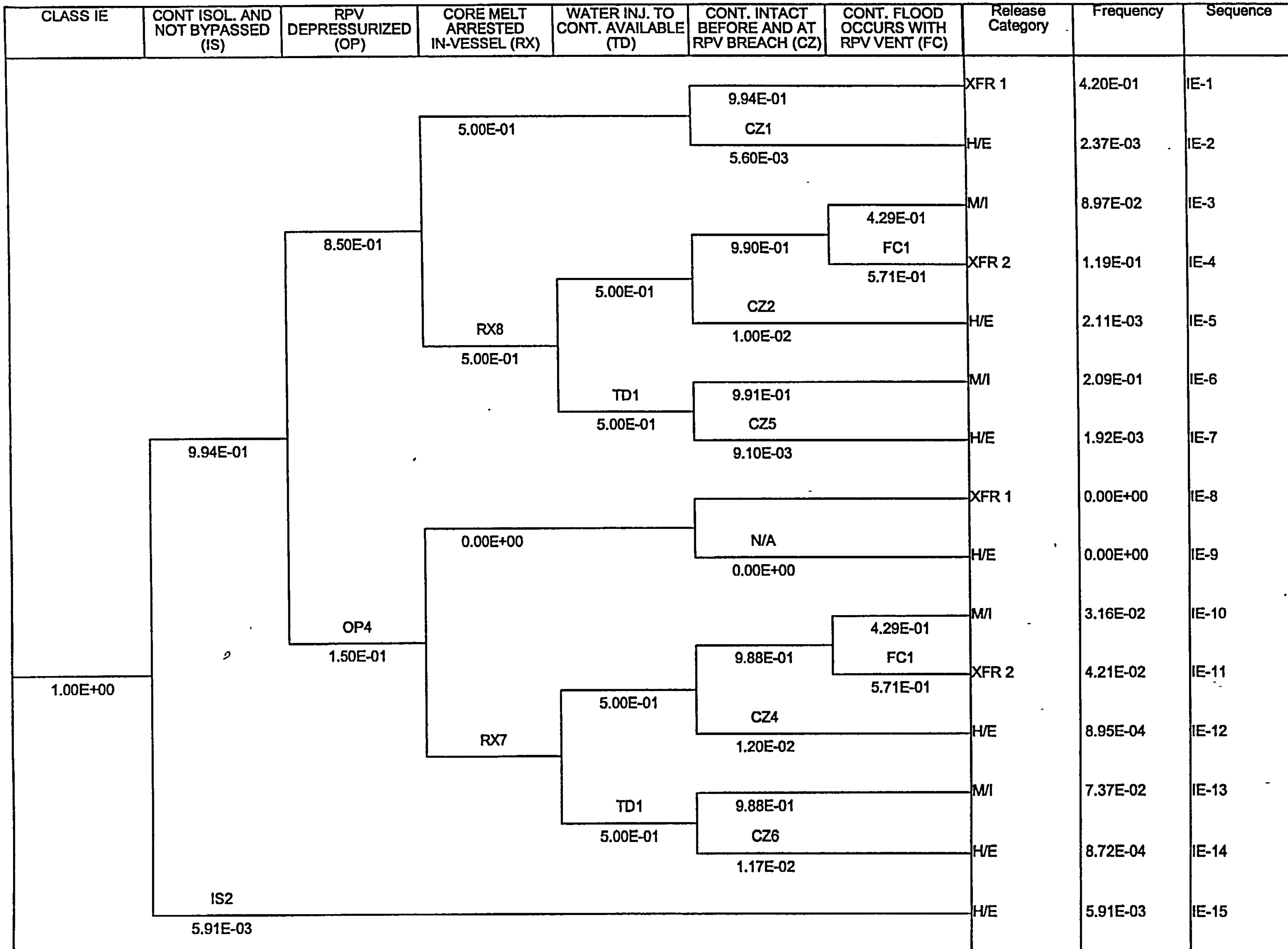


SUPPL. CET NODES FOR CLASS ID - Page 1	W:\ENGINEER\EXELON\COMED\LSA\ILRT\CET\IDX1.ETA	5/21/ 2	Page 1

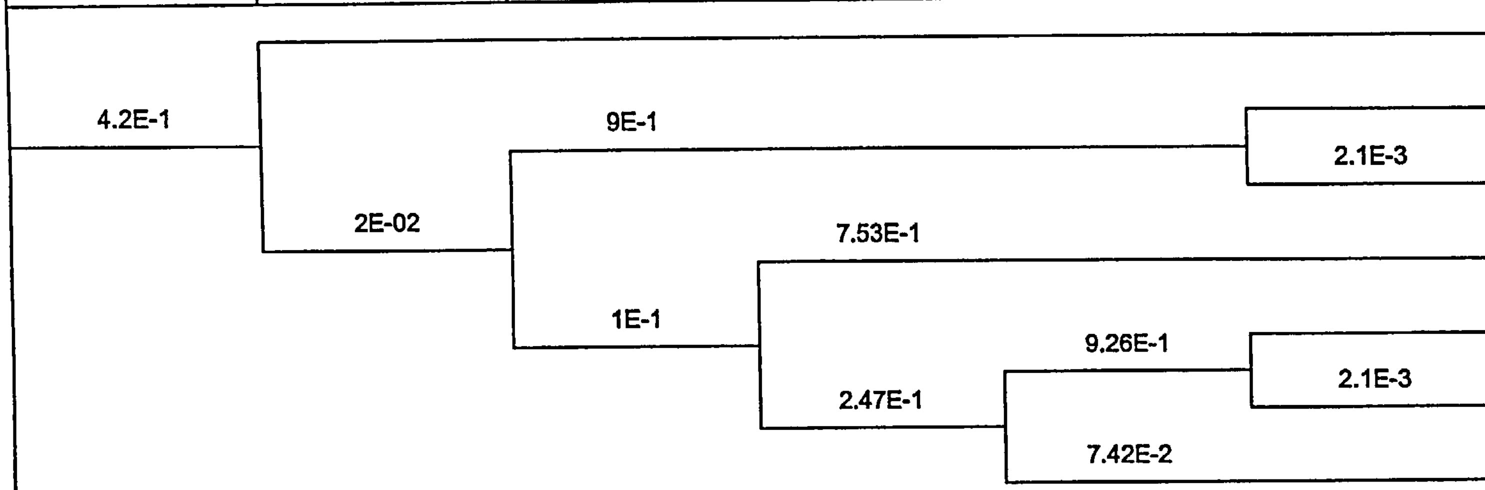
XFR2	RHR	VENT	DW	WWA	SP	RELEASE CATEGORY	SEQ. PROI	Sequence ID
TRANSFER FROM CLASS ID NON-H/E END STATE	RESIDUAL HEAT REMOVAL AVAILABLE	CONTAINMENT VENTING AVAILABLE	DRYWELL FAILURE	FAILURE IN WETWELL AIRSPACE	NO SUPPRESSION POOL BYPASS			



RELEASE CATEGORY	SEQ. PROI	Sequence ID
INTACT	0.00E+00	IDX2-1
LL/L	0.00E+00	IDX2-2
M/L	0.00E+00	IDX2-3
M/L	0.00E+00	IDX2-4
LL/L	0.00E+00	IDX2-5
M/L	0.00E+00	IDX2-6
M/L	0.00E+00	IDX2-7

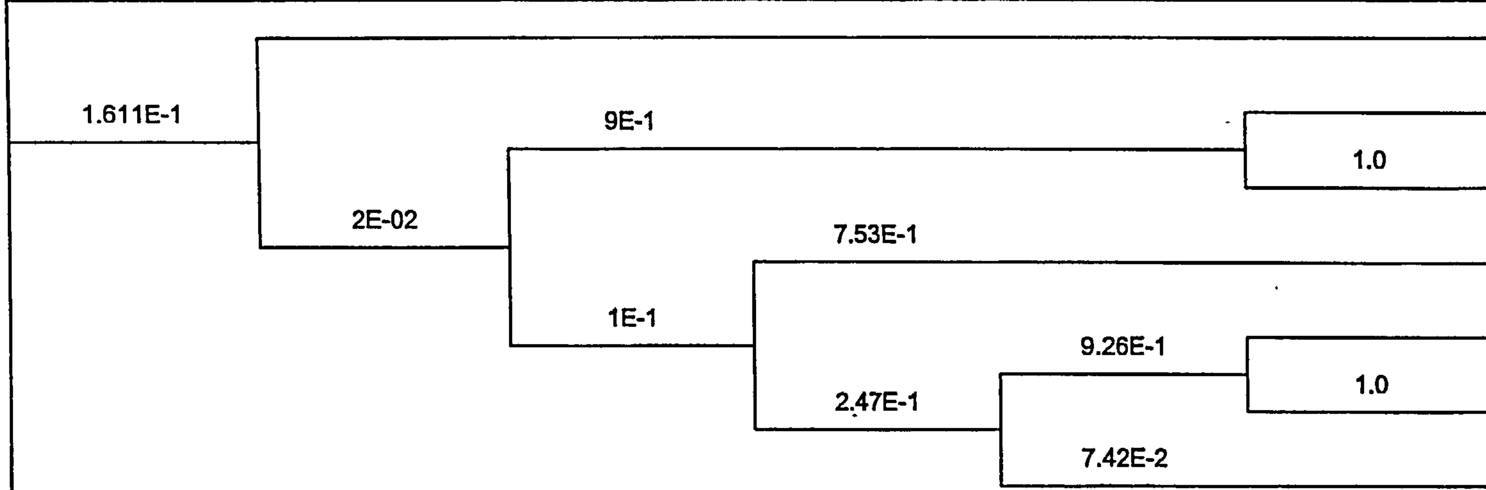

CLASS IE
CONT ISOL. AND NOT BYPASSED (IS)
RPV DEPRESSURIZED (OP)
CORE MELT ARRESTED IN-VESSEL (RX)
WATER INJ. TO CONT. AVAILABLE (TD)
CONT. INTACT BEFORE AND AT RPV BREACH (CZ)
CONT. FLOOD OCCURS WITH RPV VENT (FC)
Release Category
Frequency
Sequence
1.00E+00
9.94E-01
IS2
5.91E-03
8.50E-01
OP4
1.50E-01
5.00E-01
RX8
5.00E-01
0.00E+00
RX7
5.00E-01
TD1
5.00E-01
5.00E-01
TD1
5.00E-01
9.94E-01
CZ1
5.60E-03
9.90E-01
CZ2
1.00E-02
9.91E-01
CZ5
9.10E-03
N/A
0.00E+00
9.88E-01
CZ4
1.20E-02
9.88E-01
CZ6
1.17E-02
4.29E-01
FC1
5.71E-01
4.29E-01
FC1
5.71E-01
XFR 1 4.20E-01 IE-1
H/E 2.37E-03 IE-2
M/I 8.97E-02 IE-3
XFR 2 1.19E-01 IE-4
H/E 2.11E-03 IE-5
M/I 2.09E-01 IE-6
H/E 1.92E-03 IE-7
XFR 1 0.00E+00 IE-8
H/E 0.00E+00 IE-9
M/I 3.16E-02 IE-10
XFR 2 4.21E-02 IE-11
H/E 8.95E-04 IE-12
M/I 7.37E-02 IE-13
H/E 8.72E-04 IE-14
H/E 5.91E-03 IE-15

XFR1	RHR	VENT	DW	WWA	SP	RELEASE CATEGORY	SEQ. PROB.	Sequence ID
TRANSFER FROM CLASS IE NON-H/E END STATE	RESIDUAL HEAT REMOVAL AVAILABLE	CONTAINMENT VENTING AVAILABLE	DRYWELL FAILURE	FAILURE IN WETWELL AIRSPACE	NO SUPPRESSION POOL BYPASS			



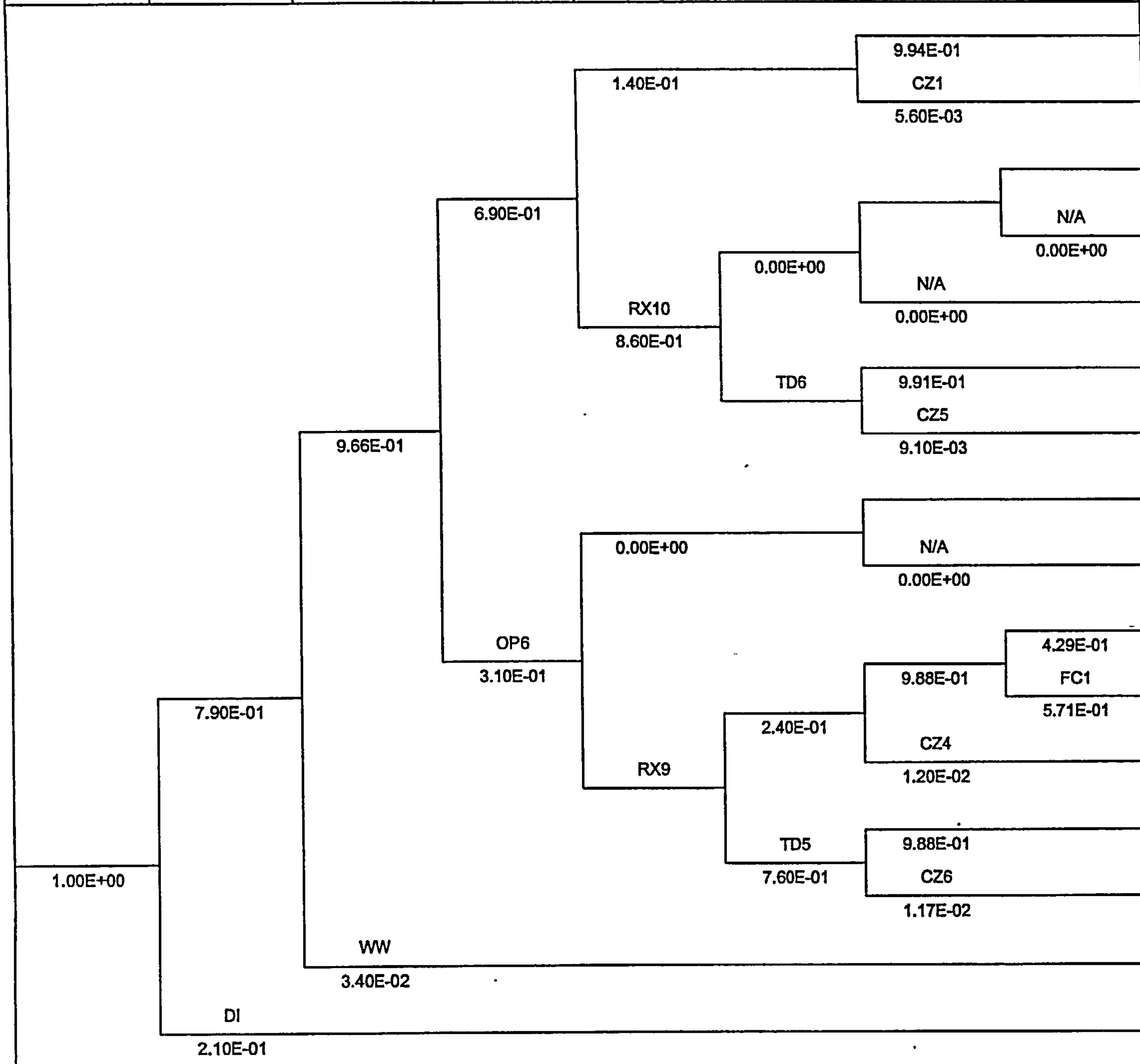
RELEASE CATEGORY	SEQ. PROB.	Sequence ID
INTACT	4.12E-01	IEX1-1
LL/L	7.54E-03	IEX1-2
M/L	1.59E-05	IEX1-3
M/L	6.33E-04	IEX1-4
LL/L	1.92E-04	IEX1-5
M/L	4.03E-07	IEX1-6
M/L	1.54E-05	IEX1-7

XFR2	RHR	VENT	DW	WWA	SP	RELEASE CATEGORY	SEQ. PROB.	Sequence ID
TRANSFER FROM CLASS IE NON-H/E END STATE	RESIDUAL HEAT REMOVAL AVAILABLE	CONTAINMENT VENTING AVAILABLE	DRYWELL FAILURE	FAILURE IN WETWELL AIRSPACE	NO SUPPRESSION POOL BYPASS			

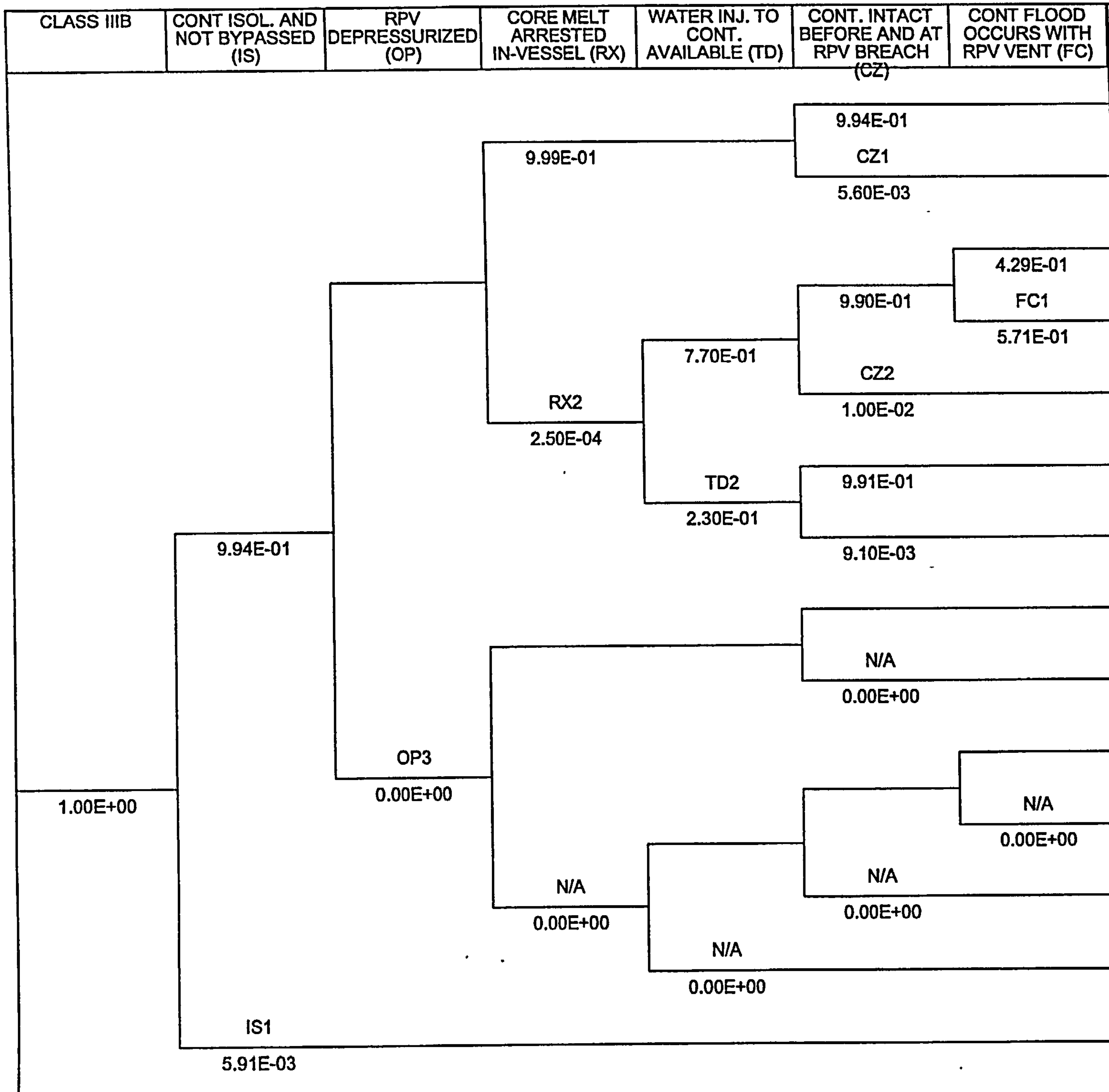


RELEASE CATEGORY	SEQ. PROB.	Sequence ID
INTACT	1.58E-01	IEX2-1
LL/L	0.00E+00	IEX2-2
M/L	2.90E-03	IEX2-3
M/L	2.43E-04	IEX2-4
LL/L	0.00E+00	IEX2-5
M/L	7.37E-05	IEX2-6
M/L	5.90E-06	IEX2-7

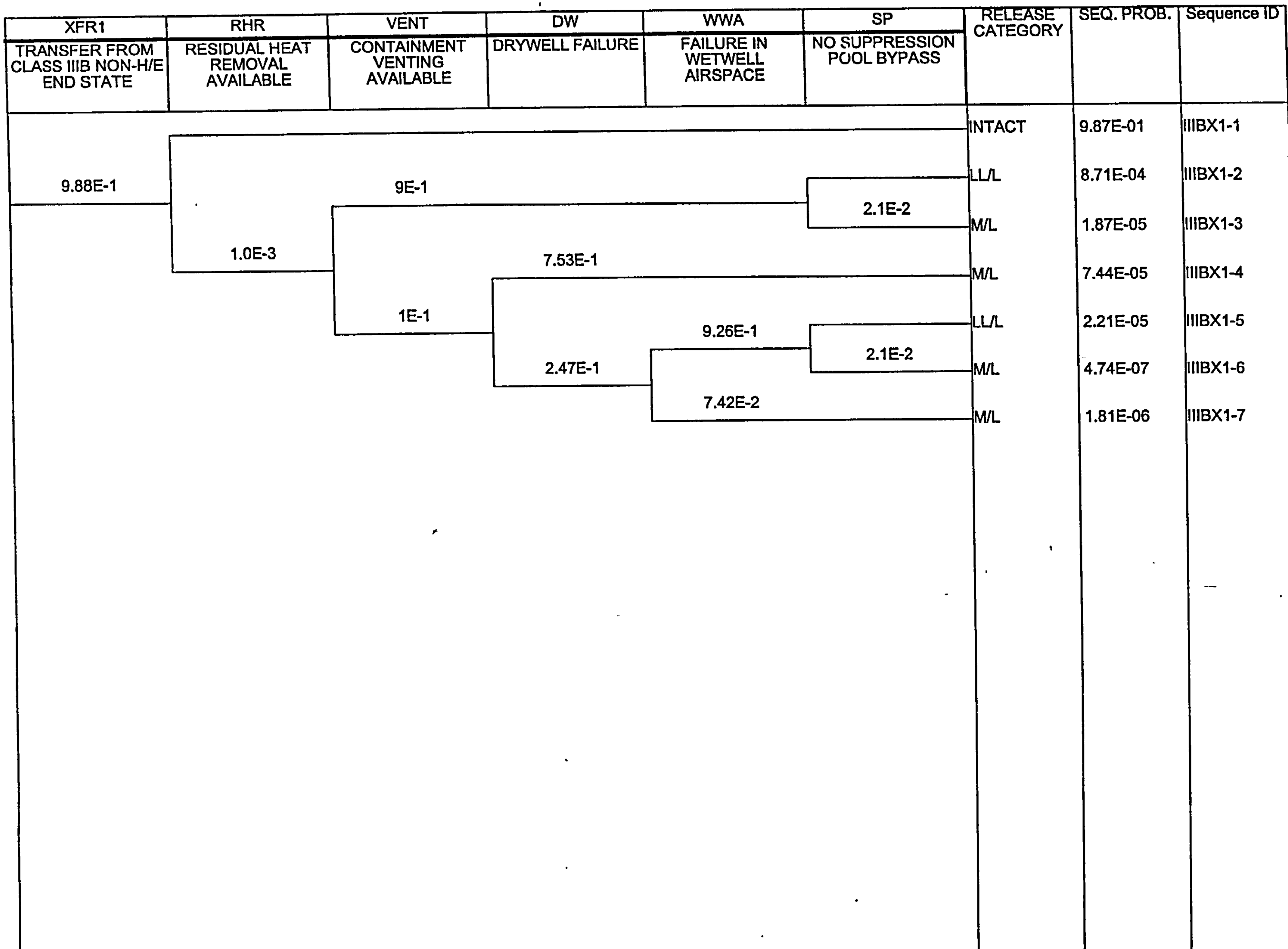
CLASS II	DW INTACT	WW AIRSPACE AND POOL NOT BYPASSED	RPV EPRESSURIZE (OP)	CORE MELT ARRESTED IN-VESSEL (RX)	WATER INJ. TO CONT. AVAILABLE (TD)	CONT. OK BEFORE AND AT RPV BREACH (CZ)	CONT. FLOOD OCCURS WITH RPV VENT (FC)



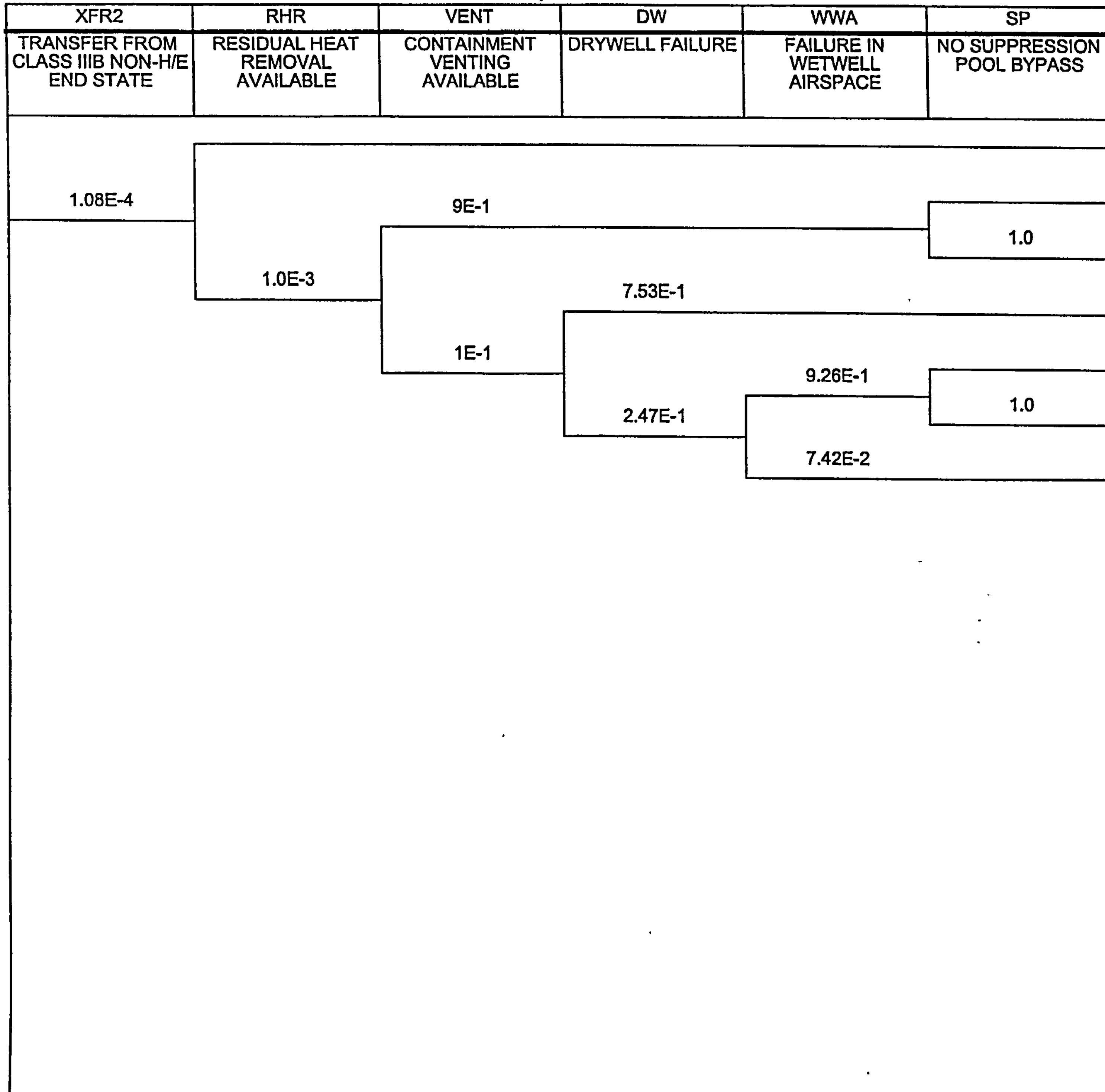
Release Category	Frequency	Sequence ID
LL/I	7.33E-02	II-1
H/I	4.13E-04	II-2
M/I	0.00E+00	II-3
L/I	0.00E+00	II-4
H/I	0.00E+00	II-5
L/I	4.49E-01	II-6
H/I	4.12E-03	II-7
LL/I	0.00E+00	II-8
H/I	0.00E+00	II-9
M/I	2.41E-02	II-10
L/I	3.20E-02	II-11
H/I	6.81E-04	II-12
L/I	1.78E-01	II-13
H/I	2.10E-03	II-14
H/I	2.69E-02	II-15
H/I	2.10E-01	II-16



Release Category	Frequency	Sequence ID
XFR 1	9.88E-01	IIIB-1
H/E	5.57E-03	IIIB-2
M/I	8.13E-05	IIIB-3
XFR 2	1.08E-04	IIIB-4
H/E	1.91E-06	IIIB-5
M/I	5.66E-05	IIIB-6
H/E	5.20E-07	IIIB-7
XFR 1	0.00E+00	IIIB-8
H/E	0.00E+00	IIIB-9
M/I	0.00E+00	IIIB-10
XFR 2	0.00E+00	IIIB-11
H/E	0.00E+00	IIIB-12
H/E	0.00E+00	IIIB-13
H/E	5.91E-03	IIIB-14

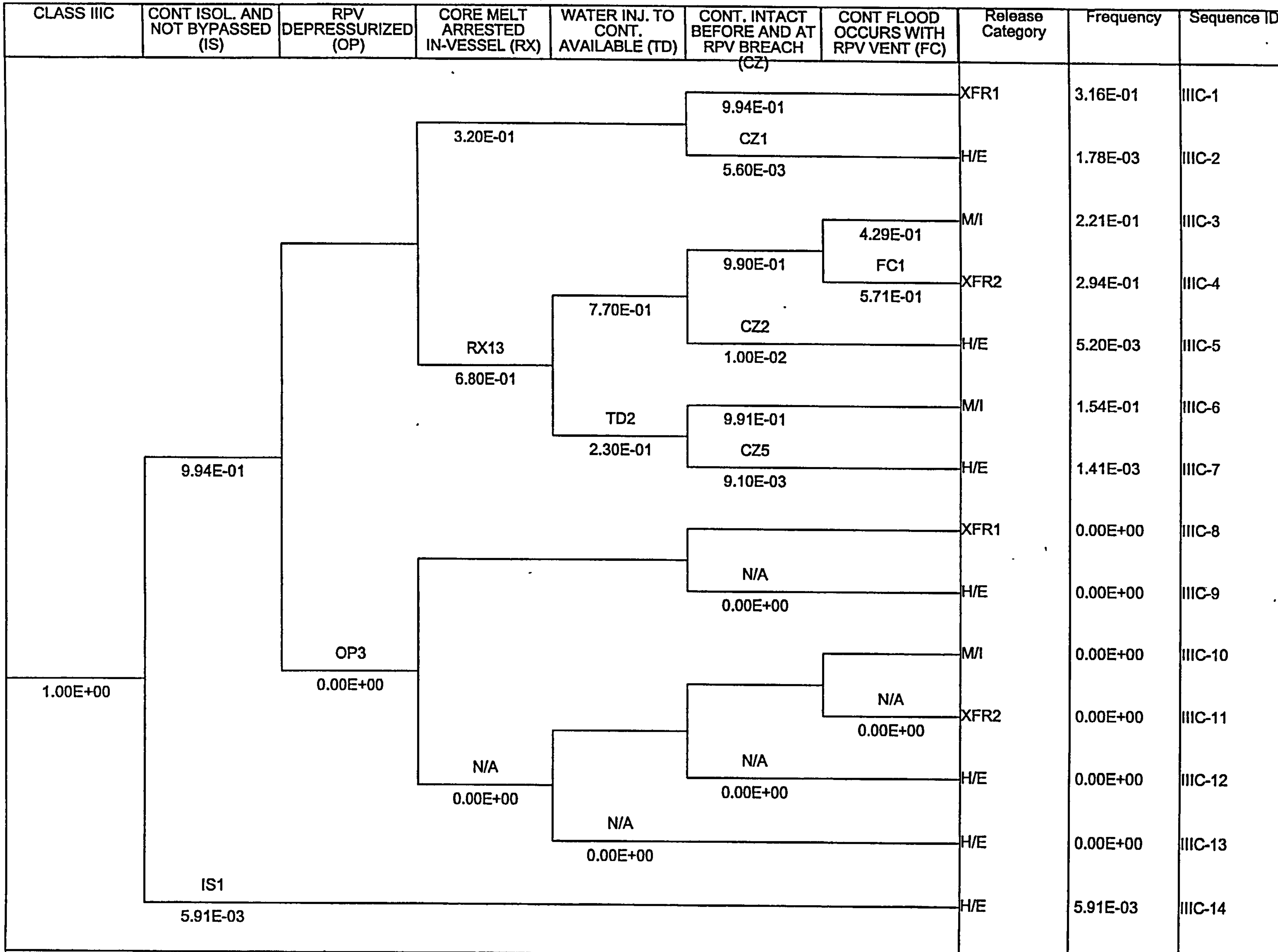


SUPPL. CET NODES FOR CLASS IIIB - Page 1	W:\ENGINEER\EXELON\COMED\LSA\ILRT\CET\IIIBX1.ETA	5/21/ 2	Page 1

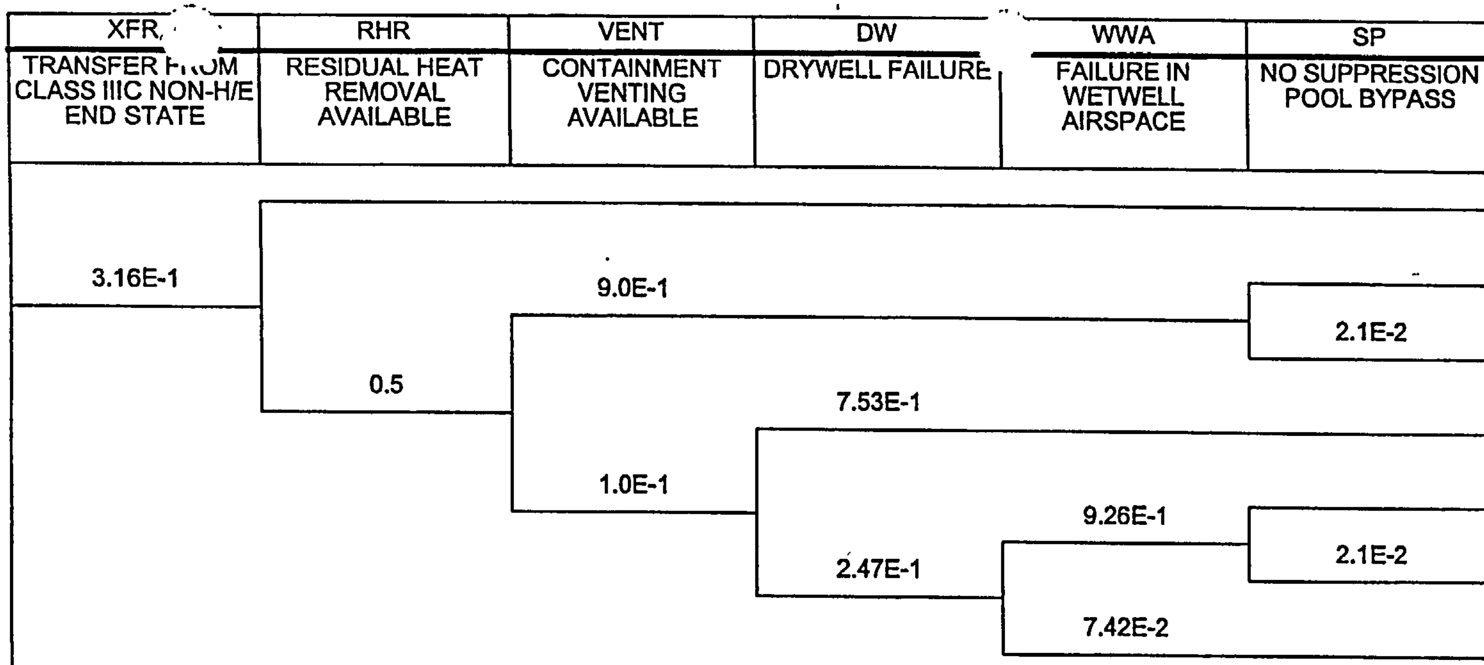


RELEASE CATEGORY	SEQ. PROB.	Sequence ID
INTACT	1.08E-04	IIIBX2-1
LL/L	0.00E+00	IIIBX2-2
M/L	9.72E-08	IIIBX2-3
M/L	8.13E-09	IIIBX2-4
LL/L	0.00E+00	IIIBX2-5
M/L	2.47E-09	IIIBX2-6
M/L	1.98E-10	IIIBX2-7

SUPPL. CET NODES FOR CLASS IIIB - Page 2 | W:\ENGINEER\EXELON\COMED\LSA\ILRT\CET\IIIBX2.ETA | 5/21/ 2 | Page 1

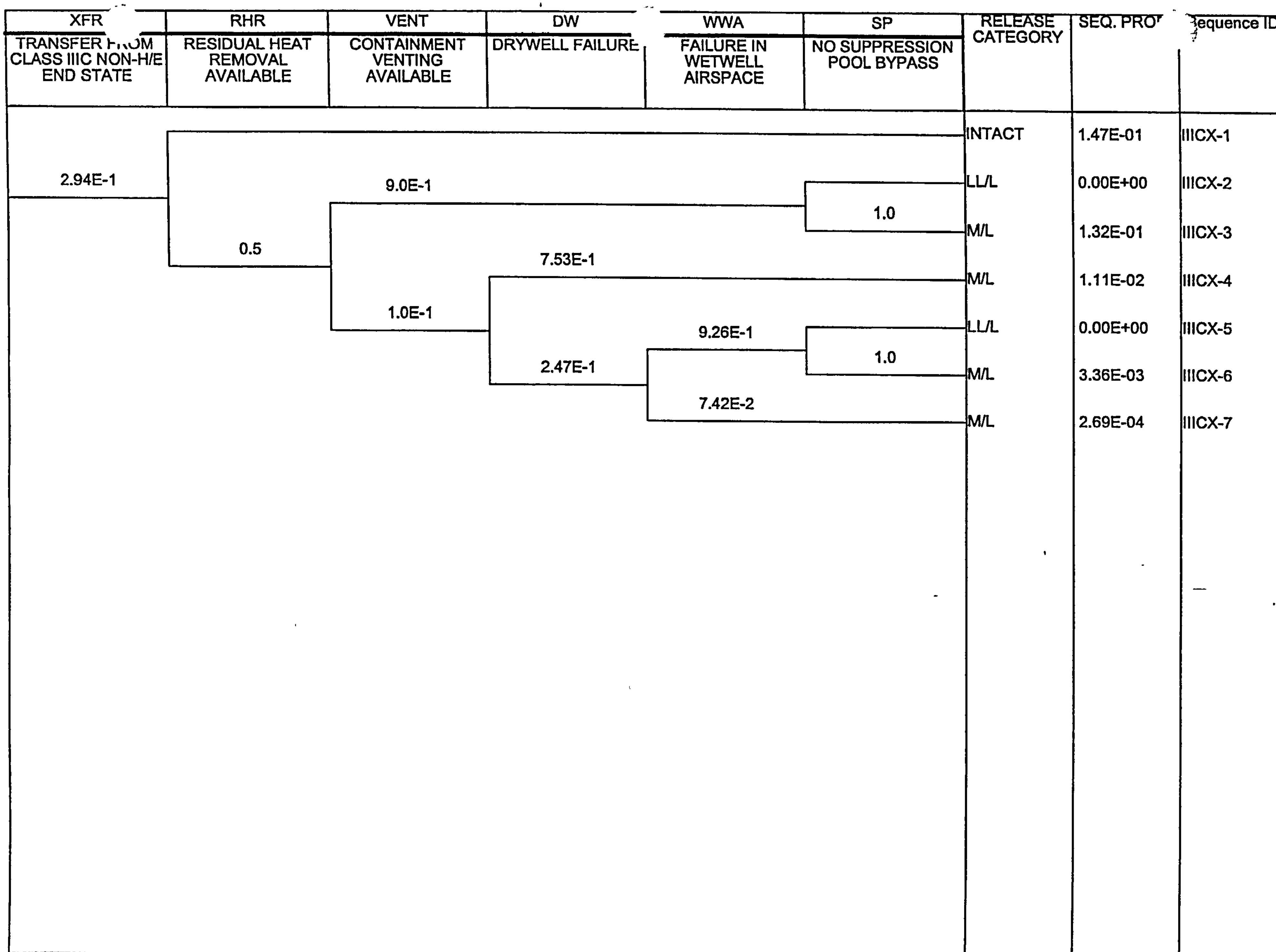


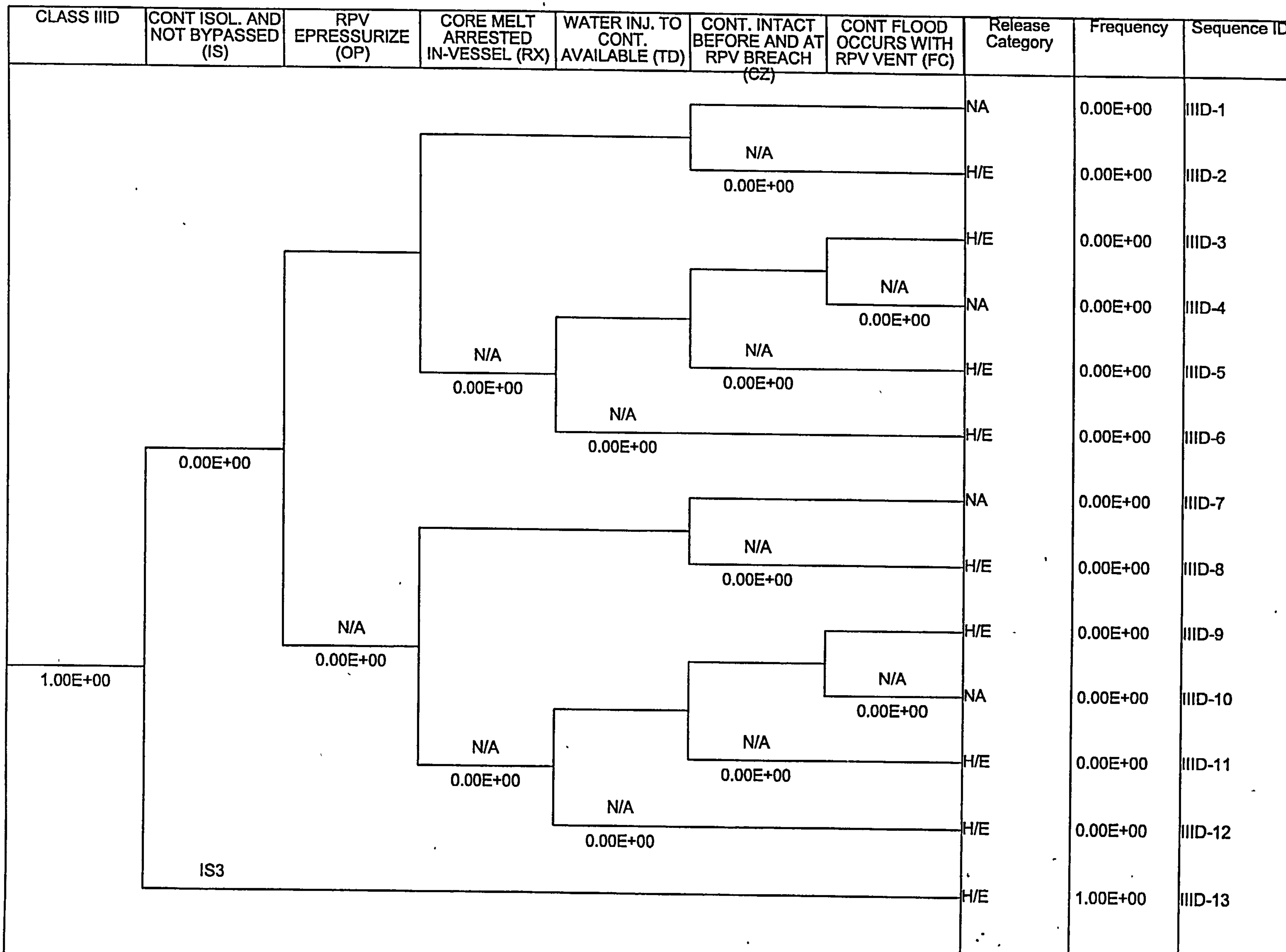
CLASS IIIC	W:\ENGINEER\EXELON\COMED\LSA\ILRT\CET\IIIC.ETA	5/21/ 2	Page 1



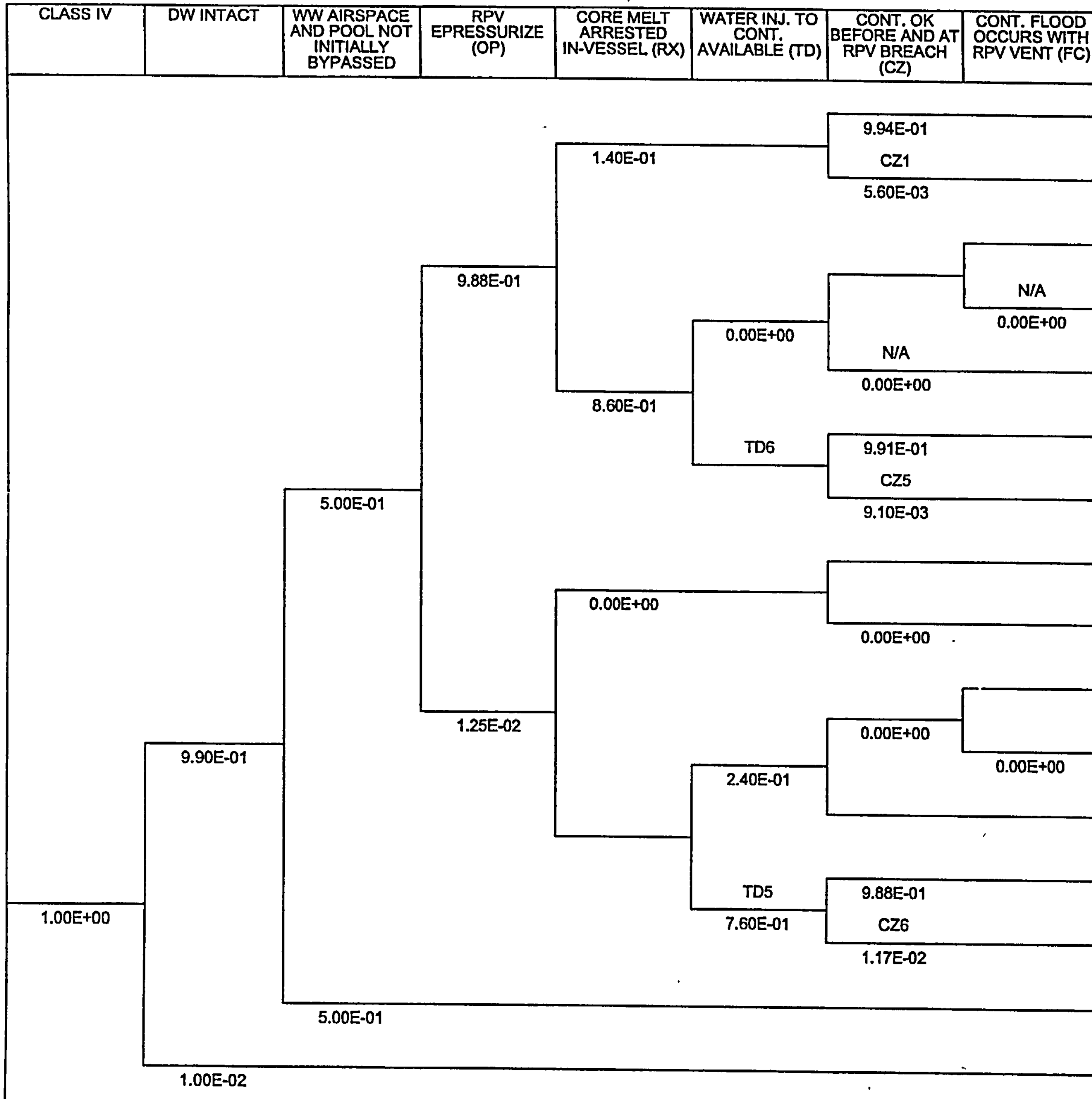
RELEASE CATEGORY	SEQ. PROB	Sequence ID
INTACT	1.58E-01	IIICX-1
LL/L	1.39E-01	IIICX-2
M/L	2.99E-03	IIICX-3
M/L	1.19E-02	IIICX-4
LL/L	3.54E-03	IIICX-5
M/L	7.59E-05	IIICX-6
M/L	2.90E-04	IIICX-7

SUPPL CET NODES FOR CLASS IIIC-Page 1	W:\ENGINEER\EXELON\COMED\LSA\ILRT\CET\IIICX1.ETA	5/21/ 2	Page 1

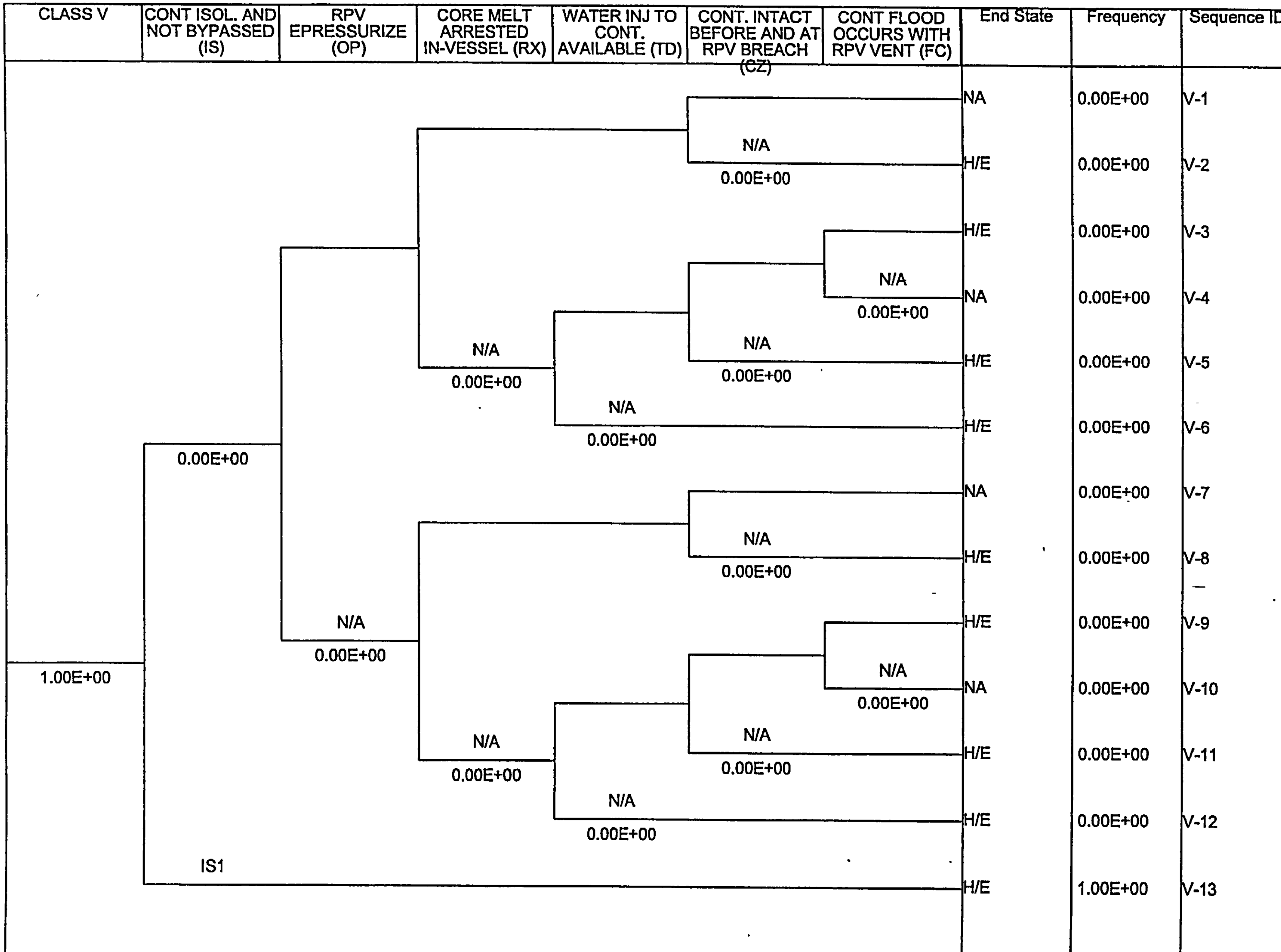

XFR
TRANSFER FROM CLASS IIIC NON-H/E END STATE
RHR
RESIDUAL HEAT REMOVAL AVAILABLE
VENT
CONTAINMENT VENTING AVAILABLE
DW
DRYWELL FAILURE
WWA
FAILURE IN WETWELL AIRSPACE
SP
NO SUPPRESSION POOL BYPASS
RELEASE CATEGORY
SEQ. PRO
equence ID
2.94E-1
9.0E-1
0.5
1.0
7.53E-1
1.0E-1
9.26E-1
1.0
2.47E-1
7.42E-2
INTACT 1.47E-01 IIICX-1
LL/L 0.00E+00 IIICX-2
M/L 1.32E-01 IIICX-3
M/L 1.11E-02 IIICX-4
LL/L 0.00E+00 IIICX-5
M/L 3.36E-03 IIICX-6
M/L 2.69E-04 IIICX-7



CLASS IIID
CONT ISOL. AND NOT BYPASSED (IS)
RPV EPRESSURIZE (OP)
CORE MELT ARRESTED IN-VESSEL (RX)
WATER INJ. TO CONT. AVAILABLE (TD)
CONT. INTACT BEFORE AND AT RPV BREACH (CZ)
CONT FLOOD OCCURS WITH RPV VENT (FC)
Release Category
Frequency
Sequence ID
1.00E+00
0.00E+00
N/A
0.00E+00
IS3
NA 0.00E+00 IIID-1
H/E 0.00E+00 IIID-2
H/E 0.00E+00 IIID-3
NA 0.00E+00 IIID-4
H/E 0.00E+00 IIID-5
H/E 0.00E+00 IIID-6
NA 0.00E+00 IIID-7
H/E 0.00E+00 IIID-8
H/E 0.00E+00 IIID-9
NA 0.00E+00 IIID-10
H/E 0.00E+00 IIID-11
H/E 0.00E+00 IIID-12
H/E 1.00E+00 IIID-13



End State	Frequency	Sequence ID
LL/E	6.81E-02	IV-1
H/E	3.83E-04	IV-2
H/E	0.00E+00	IV-3
LL/E	0.00E+00	IV-4
H/E	0.00E+00	IV-5
M/E	4.17E-01	IV-6
H/E	3.83E-03	IV-7
LL/E	0.00E+00	IV-8
H/E	0.00E+00	IV-9
H/E	0.00E+00	IV-10
M/E	0.00E+00	IV-11
H/E	1.49E-03	IV-12
M/E	4.65E-03	IV-13
H/E	5.50E-05	IV-14
H/E	4.95E-01	IV-15
H/E	1.00E-02	IV-16



CLASS V
CONT ISOL. AND NOT BYPASSED (IS)
RPV EPRESSURIZE (OP)
CORE MELT ARRESTED IN-VESSEL (RX)
WATER INJ TO CONT. AVAILABLE (TD)
CONT. INTACT BEFORE AND AT RPV BREACH (CZ)
CONT FLOOD OCCURS WITH RPV VENT (FC)
End State
Frequency
Sequence ID
1.00E+00
0.00E+00
IS1
N/A
0.00E+00
NA 0.00E+00 V-1
H/E 0.00E+00 V-2
H/E 0.00E+00 V-3
NA 0.00E+00 V-4
H/E 0.00E+00 V-5
H/E 0.00E+00 V-6
NA 0.00E+00 V-7
H/E 0.00E+00 V-8
H/E 0.00E+00 V-9
NA 0.00E+00 V-10
H/E 0.00E+00 V-11
H/E 0.00E+00 V-12
H/E 1.00E+00 V-13

Appendix C
EXTERNAL EVENT ASSESSMENT

C.1 INTRODUCTION

This appendix discusses the external events assessment in support of the LaSalle ILRT frequency extension risk assessment. This appendix uses as the starting point of this assessment the external event work documented in the LaSalle EDG Completion Time risk application. [C-1]

Background

Exelon[(1)] submitted the results of the RMIEP study (NUREG/CR-4832) to the NRC in 1994 as the basis for the LaSalle IPE/IPEEE Submittal. Each of the RMIEP external event evaluations were reviewed as part of the Submittal and compared to the requirements of NUREG-1407. The NRC transmitted to Exelon in 1996 their Staff Evaluation Report of the LaSalle IPE/IPEEE Submittal. No other LaSalle external event PSA models or analysis were developed by Exelon.

C.2 EXTERNAL EVENT SCREENING ASSESSMENT

The purpose of this portion of the assessment is to examine the spectrum of possible external event challenges to determine which external event hazards should be explicitly addressed as part of the LaSalle ILRT frequency extension risk assessment.

Volume 7 of NUREG/CR-4832 provides the LaSalle RMIEP external event screening analysis. The screening assessment appropriately begins with the comprehensive list of potential external event hazards provided in the PRA Procedures Guide, NUREG/CR-

2300. Consistent with NUREG/CR-2300, the screening assessment employed the following criteria to eliminate external event challenges from further consideration:

1. The event is of equal or lesser damage potential than the events for which the plant is designed, or
2. The event has a significantly lower mean frequency of occurrence than other events with similar uncertainties and could not result in worse consequences than those events, or
3. The event cannot occur close enough to the plant to affect it, or
4. The event is included in the definition of another event

Although not listed explicitly as one of the screening criteria, the RMIEP screening assessment does incorporate (as evidenced in the Table 3.2-1 of Volume 7) the following criterion employed in the NUREG/CR-4550 study: "The event is slow in developing and there is sufficient time to eliminate the source of the threat or to provide an adequate response." This criterion is also considered appropriate.

Aside from seismic and internal fires (which are identified specifically as part of Generic Letter 88-20, Supplement 4), the following external events were identified in the RMIEP screening assessment for further analysis:

- Aircraft Impact
- Extreme Winds and Tornadoes
- Transportation/Toxic Chemicals/Explosions
- Turbine Generated Missiles
- External Flooding

Further assessment of each of these hazards is discussed below.

Seismic

(1) Formerly ComEd.

Consistent with Generic Letter 88-20, the RMIEP study and the LaSalle IPEEE Submittal do not screen out this hazard but provide quantitative analyses. This is appropriate. This hazard is maintained in this assessment for further consideration.

Internal Fires

Consistent with Generic Letter 88-20, the RMIEP study and the LaSalle IPEEE Submittal do not screen out this hazard but provide quantitative analyses. This is appropriate. This hazard is maintained in this assessment for further consideration.

Aircraft Impact

Section 3.4.2 of Volume 7 of the RMIEP study provides a bounding assessment of the aircraft impact hazard. The assessment approach is consistent with the guidance provided in NUREG/CR-5042, Evaluation of External Hazards to Nuclear Power Plants in the United States, (identified in Generic Letter 88-20 as a source of acceptable methods to be used in the assessment of projected low frequency external events).

The LaSalle RMIEP bounding assessment conservatively assumes that any impact to a Category I structure sufficient to cause back face scabbing of an exterior wall results in a core damage probability of 1.0. The resulting bounding core damage frequency was estimated at 4.84E-7/yr.

The LaSalle RMIEP bounding assessment did not include the diesel generator building in the assessment because it is much smaller than the other key buildings and it is shielded on two sides by other buildings. Using the RMIEP-calculated reactor building aircraft impact CDF contribution of 3.93E-7/yr (obtained from Table 3.4-5 of NUREG/CR-4832 Volume 7), the contribution from an aircraft impact on the diesel generator building is estimated here as follows:

3.93E-7/yr x 0.20 x 0.50 x 1.00 = 3.93E-8/yr

where:

0.20 = DG Bldg. area / Rx Bldg. area (based on review of M dwgs)

0.50 = 2 of the 4 compass directions are protected by other buildings

1.00 = Per the RMIEP assumptions, the CCDP is 1.0

Incorporating the DG building into the RMIEP bounding assessment framework results in a conservative CDF estimate of 5.23E-7/yr due to aircraft impacts.

If it is assumed here that an aircraft impact sufficient to result in back face scabbing of building exterior walls does not conservatively result in a CCDP of 1.0 (as assumed in the RMIEP framework), but rather a more reasonable value on the order of 0.1 or less, the aircraft impact induced CDF is estimated in the mid to lower E-8/yr range. Such an estimate is less than 1% of the LaSalle Revision 2001A PSA CDF. Explicit quantification of such accidents would not provide any significant quantitative or qualitative information to this assessment; therefore, such sequences are appropriately excluded from further analysis.

Extreme Winds and Tornadoes

Section 3.4.3 of Volume 7 of the RMIEP study provides a bounding assessment of extreme wind and tornado hazards. The assessment considers the pressure loading of extreme winds and tornadoes on both seismic Category I and non-Category I structures, failure of non-Category I structures onto Category I structures, and the effects of tornado generated missiles . The LaSalle Category I structures are designed to the following Design Basis Tornado (DBT) loadings:

- maximum rotation velocity of 300 mph
- transnational velocity of 60 mph
- external pressure drop of 3 psi

- impacts from postulated tornado missiles (e.g., wood plank, automobile)

The non-Category I structures are designed to withstand 90 mph straight winds.

As the LaSalle Category I structures are designed to 300 mph winds, the RMIEP study determined the frequency of wind pressure induced failures of Category I buildings to be negligible (<1E-6/yr). With respect to tornado-generated missiles, the study concluded that deformable and non-deformable missiles are not significant contributors to plant risk (e.g., the contribution to plant risk due to the automobile missile impact on a Category I structure was estimated at less than 1E-8/yr). In addition, building air intakes and exhausts are protected from missiles by concrete barriers. Also, the ventilation stack is designed to withstand the effects of the DBT and therefore will collapse (onto the Auxiliary Bldg.) with a very low probability.

The plant risk contribution from extreme wind and tornado effects on non-Category I structures was estimated in the 1E-8/yr range. Although these buildings are more easily damaged, they do not contain equipment necessary for safe shutdown.

Due to the design of the LaSalle plant, the effect of extreme winds and tornadoes on plant safe shutdown is characteristic of LOOP and DLOOP initiator challenges.

The RMIEP study concluded that the median core damage frequency contribution from extreme wind and tornado hazards is 3E-8/yr. Although not specifically listed in the RMIEP study, the mean value is estimated here at 7.5E-8/yr (assuming a lognormal distribution and an error factor of 10). This estimate is approximately 1% of the LaSalle Revision 2001A PSA base CDF, and approximately 5% of the LOOP/DLOOP-initiated CDF. The tornado impact on LOOP/DLOOP accident sequences is already incorporated into the LOOP and DLOOP initiating frequencies and the LOOP and DLOOP offsite AC power recovery probabilities. Explicit quantification of such accidents would not provide any significant quantitative or qualitative information to the LaSalle ILRT frequency extension risk assessment. Such sequences are judged appropriately subsumed into the existing "internal events" analysis and are therefore excluded from further analysis.

Transportation

Section 3.4.4 of Volume 7 of the RMIEP study provides a bounding assessment of transportation hazards. The assessment addresses the frequency of occurrence of transportation accidents and the fragility of the plant to the associated effects (i.e., explosion forces, and toxic chemicals).

The maximum probable explosion hazard is a truck accident on nearby County Road 6 (6 miles south of the plant) involving an explosive force equivalent to a 50,000 lb. load of TNT. The walls of all LaSalle safety-related structures are designed to a minimum loading capacity of 3.0 psi. Using a conservative modeling approach documented in NUREG/CR-2462, the lower bound capacity of structural panels at LaSalle was conservatively estimated at 1.95 psi. Comparison of this calculated minimum wall capacity to the free-field incident overpressure of 0.66 psi due to the truck blast, shows that at least a factor of 3 capacity exists against the blast loading. The RMIEP study appropriately concluded that explosions due to transportation accidents are a negligible contributor to plant risk.

Regarding toxic chemical releases, the RMIEP study reviewed the types and amounts of chemicals typically stored and transported in and around the LaSalle site. Among the three transportation modes near the site, a barge accident in the Illinois River could result in the largest amount of chemical spill. The Illinois River is 3.5 miles away from the plant structures at its closest distance. Also, the river elevation is approximately 180 feet below the plant grade. Given that many toxic vapors are denser than air, the atmospheric dispersion of these chemicals towards the plant under favorable wind conditions is unlikely because of the difference in plant and river elevations. Also, for more turbulent wind conditions, it is highly unlikely that a toxic vapor would reach the control room air intakes at excessive concentrations. The RMIEP study appropriately concluded that toxic chemical releases are negligible contributors to plant risk.

Explicit quantification of such accidents would not provide any significant quantitative or qualitative information to the LaSalle ILRT Extension Submittal assessment.

<u>Turbine Missiles</u>

Section 3.4.5 of Volume 7 of the RMIEP study provides a bounding assessment of turbine missile hazards. The RMIEP assessment estimates the frequency of turbine missile induced core damage at less than 1E-7/yr and concludes that the hazard is not a significant contributor to risk. Explicit quantification of such accidents would not provide any significant quantitative or qualitative information to the LaSalle ILRT Extension Submittal assessment; therefore, such sequences are appropriately excluded from further analysis.

<u>External Flooding</u>

Section 3.4.6 of Volume 7 of the RMIEP study provides a bounding assessment of the external flooding hazard. The assessment appropriately considers the following three external flooding sources:

- Nearby Illinois River
- LaSalle cooling lake
- Local precipitation

The plant grade level is at 710' mean sea level (MSL). All safety-related structures at the LaSalle station have a ground floor surface elevation of at least 710.5' (MSL). An inspection of the plant was made as part of the RMIEP study. The inspection revealed that ground floor doors are leak tight; even if external water levels were to rise above plant grade, the buildings would not be flooded.

The probable maximum flood elevation of the Illinois River, including coincident wave effect, is 522.5'. This level is 188 feet below the 710.5' MSL ground floor elevation of all LaSalle site safety-related structures. Failures of low navigation dams existing upstream of the plant would also not affect the site.

The cooling lake is at a lower elevation, 700' MSL, than the 710.5' MSL ground floor elevation of all LaSalle site safety-related structures. Runoff from the lake (due to intense precipitation or breaching of the lake dikes) would flow away from the cooling lake into local creeks that meet the Illinois River.

The probable maximum precipitation (based on conservative assumptions) is calculated to result in a water level elevation at the LaSalle site of approximately 710.3' MSL, slightly lower than the 710.5' MSL ground floor elevation of all LaSalle site safety-related structures.

The RMIEP study appropriately excludes external flood hazards as negligible contributors to plant risk. Explicit quantification of such accidents would not provide any significant quantitative or qualitative information to the LaSalle ILRT Extension Submittal assessment; therefore, such sequences are appropriately excluded from further analysis.

Conclusions of Screening Assessment

Given the foregoing discussions, the following external event hazards are judged not screened out and are evaluated further in the LaSalle ILRT Extension Submittal:

- Seismic events
- Internal fires

The other external hazards are assessed to be negligible contributors to plant risk. Explicit treatment of these other external hazards is not necessary for most PSA applications (including the ILRT Extension Submittal) and would not provide additional risk-informed insights for decision making.

C.3 SEISMIC ASSESSMENT

Seismic induced accident sequences are included in the LaSalle PSA Revision 2001A (i.e. the current model of record, and the PSA models used in this ILRT risk assessment). The seismic sequences in the LaSalle model of record are based on rigorous seismic PRA work performed for the LaSalle RMIEP study.

This section discusses the seismic induced accident sequence assessment.

C.3.1 RMIEP Seismic Overview

The RMIEP study analyzed LaSalle seismic risk employing the methodology sponsored by the U.S. NRC under the Seismic Safety Margin Research Program (SSMRP) and developed by Lawrence Livermore National Laboratory (LLNL). The key elements of the LaSalle RMIEP seismic risk analysis are:

1. Development of the seismic hazard at the LaSalle site including the effect of local site conditions.

2. Comparisons of the best estimate seismic response of structures, components, and piping systems with design values for the purposes of specifying median responses in the seismic risk calculations.
3. Investigation of the effects of hydrodynamic loads on seismic risk.
4. Development of building and component fragilities for important structures and components.
5. Development of the system models (e.g., event and fault trees).
6. Estimation of the seismically induced core damage frequency.

This approach to seismic risk assessment is consistent with the requirements of the NRC IPEEE Program and current seismic risk assessment technology. Overviews of these elements are provided below.

RMIEP Seismic Hazard Frequency

The LaSalle seismic hazard curve used in the RMIEP study is based on the NRC sponsored Eastern United States Seismic Hazard Characterization study (NUREG/CR-5250) performed by Lawrence Livermore National Laboratory (LLNL) in the 1980's. The LaSalle RMIEP hazard curve is divided into seven discrete seismic magnitude ranges for final sequence quantification:

- LL1: magnitude 0.10-0.18g
- L1: magnitude 0.18-0.27g
- L2: magnitude 0.27-0.36g
- L3: magnitude 0.36-0.46g
- L4: magnitude 0.46-0.58g
- L5: magnitude 0.58-0.73g
- L6: magnitude >0.73g

The LLNL seismic hazard curves used in the RMIEP study are more conservative than the latest NRC estimates and the EPRI estimates. In conjunction with providing funding to LLNL in the 1980's to perform a probabilistic seismic hazard analysis (PSHA) study, the NRC recommended that the nuclear power industry perform an independent study to provide the NRC with comparative information. A consortium of nuclear power utilities funded EPRI to perform a seismic hazard study. EPRI developed its own PSHA methodology and PSHA estimates at 56 of the eastern United States sites (documented in EPRI NP-4726 and EPRI NP-6395D). The differences between the 1980's LLNL and the EPRI seismic hazard estimates (the EPRI curves were generally lower) are addressed in NUREG/CR-4885.

During 1992 and 1993, LLNL re-elicited input data from their seismicity and ground motion experts using a revised elicitation procedure. LLNL then revised their PSHA computer code and produced updated PSHA estimates at eastern United States sites. The updated LLNL methodology reduced the seismic hazard estimates below that of the 1980's study, thus reducing the differences between the LLNL and EPRI hazard estimates. According to NUREG-1488, the updated LLNL seismic hazard estimates will be considered by the NRC staff in future licensing actions such as safety evaluation reports, reviews of individual plant examination of external events (IPEEE) submittals, and early site reviews.

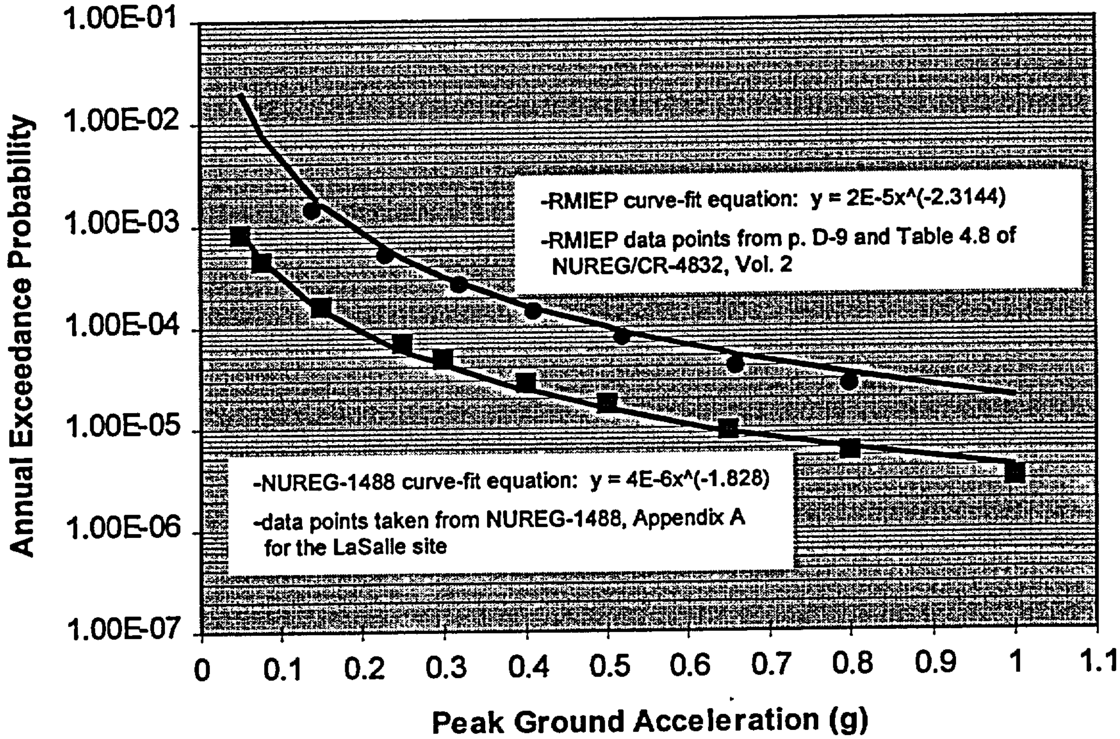
The seismic hazard curve used in the LaSalle RMIEP study is compared with the latest NRC estimates (taken from NUREG-1488) in Figure C.3-1. As can be seen from Figure C.3-1, the hazard frequencies used in the RMIEP study are approximately a factor of 5 higher than those assessed using the latest NRC estimates.

<u>RMIEP Seismic Response Analysis</u>

Seismic responses, together with fragilities, allow for the calculation of seismically induced failure probabilities. The seismic response task generated probabilistic seismic responses for all structures and equipment identified in the PSA models. The SMACS methodology (NUREG/CR-2015) of the SSMRP was used in the LaSalle RMIEP response analysis. SMACS analyses were performed on LaSalle structures, including

Figure C.3-1

COMPARISON OF LASALLE RMIEP AND NUREG-1488 SEISMIC HAZARD CURVES



NOTES:

1. RMIEP study seismic hazard curve: circle data points
   NUREG-1488 LaSalle site seismic hazard curve: square data points

2. RMIEP data points are plotted as the middle pga value of the discrete RMIEP seismic level range (the middle pga value for the >0.73g range is estimated here as 0.8g) with the mean frequency from Table 4.8 and page D-9 of NUREG/CR-4832, Volume 2.

3. Smooth curves are Microsoft Excel curve-fits to the RMIEP study and NUREG-1488 discrete data points (see chart text for equations).

the effects of soil-structure interaction (SSI). SMACS links together seismic input, SSI, structure response, and piping system and component response.

RMIEP Hydrodynamic Load Investigation and Load Combination Approach

The RMIEP study evaluated the probabilities of failure of a particular structure or equipment due to earthquake occurrence by including the effect of the hydrodynamic loads which may occur concurrently with the earthquake. The hydrodynamic loads identified and considered in the RMIEP analysis are: safety/relief valve discharge loads, LOCA-induced loads, jet forces, pool swell, condensation-oscillation (CO), and chugging. It was determined that hydrodynamic loads which may be experienced in BWRs during an earthquake are not significant at LaSalle.

RMIEP Fragility Analysis

The RMIEP structural fragility analysis followed the SSMRP structural fragility assessment methodology as documented in NUREG/CR-2320. Detailed fragility assessments were performed for various shear walls and diaphragms, the primary containment, and concrete members inside containment. Structural fragilities were assessed in terms of equivalent elastic capacities.

The RMIEP equipment fragility analysis followed the SSMRP subsystem fragility assessment methodology as documented in NUREG/CR-2405. Fragilities for selected LaSalle components were derived by extrapolating design information. The fragilities are defined as the conditional probability of failure given a specified structural response. The equipment fragilities are assumed to fit a lognormal distribution and are defined by a spectral acceleration capacity and two randomness and uncertainty variables. LaSalle specific fragilities were assessed for approximately three dozen key components, subsystems, and component types. Generic fragilities for other equipment were obtained from available industry studies.

The RMEIP general conclusion regarding this aspect of the seismic analysis is that the LaSalle plant is designed very well from a seismic point of view. Seismic induced structural and equipment failures, other than loss of offsite power (refer to Table C.3-1), do not contribute significantly to LaSalle seismic risk.

RMIEP Seismic PSA Models

The RMIEP study considers the following potential seismic induced accident sequence initiating events:

Seismic-Induced Initiator	Assessment
RPV Rupture	Not significant likelihood; no sequences explicitly modeled
ISLOCA/BOC	Not significant likelihood; no sequences explicitly modeled
LLOCA	3+ SORVs following transient, or seismic-induced piping failure (negligible contributor); sequences explicitly modeled
MLOCA	2 SORVs following transient, or seismic-induced piping failure (negligible contributor); sequences explicitly modeled
SLOCA	1 SORV following transient, or seismic-induced piping failure (negligible contributor); sequences explicitly modeled
Transient	Loss of Offsite Power likely for most seismic events. Loss of offsite power subsumes all other potential transients. Sequences explicitly modeled.

Table C.3-1

OFFSITE POWER FRAGILITIES (RMIEP)

RMIEP EVENT	DESCRIPTION	MEAN VALUE
LOSP-LL1	Loss of Offsite Power due to ceramic insulator failure in switchyard from LL1 seismic initiator	2.48E-01
LOSP-L1	Loss of Offsite Power due to ceramic insulator failure in switchyard from L1 seismic initiator	2.95E-01
LOSP-L2	Loss of Offsite Power due to ceramic insulator failure in switchyard from L2 seismic initiator	3.71E-01
LOSP-L3	Loss of Offsite Power due to ceramic insulator failure in switchyard from L3 seismic initiator	4.36E-01
LOSP-L4	Loss of Offsite Power due to ceramic insulator failure in switchyard from L4 seismic initiator	5.00E-01
LOSP-L5	Loss of Offsite Power due to ceramic insulator failure in switchyard from L5 seismic initiator	5.75E-01
LOSP-L6	Loss of Offsite Power due to ceramic insulator failure in switchyard from L6 seismic initiator	6.59E-01

The RMIEP event tree structure for seismic events is taken directly from the RMIEP internal event trees. Any event in the fault tree which could be the result of either a random failure or a seismically induced failure was modified by adding OR-gates with two basic event inputs. After the event trees and fault trees were developed, a detailed database providing the basic events, associated response fragility, and random failure data was generated to feed into the SEISIM code to yield the CDFs for all earthquake levels.

The following key assumptions and modeling issues are incorporated into the RMIEP seismic accident sequence structure:

- Seismic events that do not trigger seismic-induced loss of offsite power are not explicitly modeled, they are assessed as not risk significant.
- All modeled seismic sequences involve loss of offsite power, as such, systems dependent upon offsite power (e.g., Feedwater, Condensate, CRD, power conversion, etc.) are not modeled.
- Offsite AC power recovery is assigned a failure probability of 1.0 for all seismic levels.
- Onsite AC power recovery is credited, except in the case of common cause diesel generator failure.
- Primary containment venting is not credited.

RMIEP Seismic Quantification Results

The total seismic core damage frequency is estimated in the RMIEP study at a mean value of 7.58E-7/yr. More than 98% of the total seismic frequency is comprised of seismic induced station blackout sequences involving initial RCIC operation. Approximately 1% of the seismic CDF are seismic induced loss of offsite power sequences involving stuck open relief valves. The high percentage of station blackout core damage sequences is not surprising given that the RMIEP seismic sequences do not credit recovery of offsite power.

RMIEP Conclusions Regarding LaSalle Seismic Risk

The LaSalle seismic risk is dominated by seismic-induced loss of offsite power initiators followed by random equipment failures. The key conclusions of the RMIEP seismic analysis are best described by the following passages from NUREG/CR-4832, Volume 2, Section 4:

> *"The primary characteristic of the dominant sequences at LaSalle is that the only explicitly seismic events appearing in the final cut sets are the seismic initiating event frequencies for each level and the seismically induced loss of offsite power conditional probabilities at each level. No other seismic failures or seismic related events survived the initial and final quantifications. This is very different than the results for many other plants. The LaSalle plant is very well designed from a seismic view-point. The detailed structural analysis performed in Volume 8 did not find any structural failures where walls might fall and damage critical equipment, the cabinets and panels were bolted down correctly, and the piping penetrations were designed appropriately to handle any shifting as a result of the seismic event. The accident sequences, therefore, are equivalent to seismically induced transients.*
>
> *If a LOSP was not likely to occur as a result of the seismic event, there would be no dominant seismic sequences as LaSalle. No other seismically induced initiator has a significant conditional probability and compromises redundancy enough to result in accident sequences with a substantial frequency. The dominant sequences at LaSalle are, therefore, all seismically induced losses of offsite power except that no credit is given for recovering offsite power after the seismic failure."*

C.3.2 Seismic Modeling For LaSalle ILRT Extension Submittal

The LaSalle seismic analysis performed for the RMIEP study is a rigorous LaSalle specific analysis. The methodology used is consistent with the requirements of the NRC IPEEE Program and with current seismic risk assessment technology. The general conclusions regarding the seismic response of the LaSalle plant are judged still applicable. Specific dominant sequences and cutsets may currently differ due to plant procedural and PSA model changes. As the LaSalle seismic risk is sensitive to EDG availability and reliability, seismic sequences are explicitly included in the LaSalle PSA model of record. No additional seismic PSA effort other than this discussion has been performed in support of this ILRT risk assessment.

The seismic modeling approach used in the LaSalle PSA is based on the general conclusions of the RMIEP study and is as follows :

- The division of the LaSalle seismic hazard curve into seven discrete seismic magnitude ranges is maintained in this assessment (the same ranges used in the RMIEP study are maintained).
- Instead of the 1980's vintage seismic initiator frequencies used in the RMIEP study, this assessment uses the more current NUREG-1488 based frequencies (refer to Figure C.3-1). These are:

Seismic Magnitude Range		Exceedance Frequency
LL1:	Magnitude 0.10 – 0.18g	2.7E-4/yr
L1:	Magnitude 0.18 – 0.27g	9.2E-5/yr
L2:	Magnitude 0.27 – 0.36g	4.4E-5/yr
L3:	Magnitude 0.36 – 0.46g	2.6E-5/yr
L4:	Magnitude 0.46 – 0.58g	1.7E-5/yr
L5:	Magnitude 0.58 – 0.73g	1.1E-5/yr
L6:	Magnitude > 0.73g	7.1E-6/yr

  These frequencies are conservatively taken at the beginning point of each magnitude range (e.g., the 2.7E-4/yr frequency for the LL1 range is calculated based on a 0.10 pga seismic event).

- The RMIEP loss of offsite power fragilities (refer to Table C.3-1) are judged reasonable and are maintained in this assessment.

- The seismic hazard frequencies and associated offsite power fragilities are combined into the following seismic event tree initiating events:

Initiator ID	Description	Frequency
%SEIS-LL1	LL1 Seismic–Induced DLOOP Event	6.7E-5/yr
%SEIS-L1	L1 Seismic–Induced DLOOP Event	2.7E-5/yr
%SEIS-L2	L2 Seismic–Induced DLOOP Event	1.6E-5/yr
%SEIS-L3	L3 Seismic–Induced DLOOP Event	1.1E-5/yr
%SEIS-L4	L4 Seismic–Induced DLOOP Event	8.5E-6/yr
%SEIS-L5	L5 Seismic–Induced DLOOP Event	6.3E-6/yr
%SEIS-L6	L6 Seismic–Induced DLOOP Event	4.7E-6/yr

- Each of the above seismic initiators is propagated through the accident sequence quantification of the base LaSalle model. These seismic sequences are characterized as follows:

  - The sequences are dual-unit LOOPs and the base LaSalle DLOOP event tree structure is employed.
  - Consistent with the insights of the RMIEP seismic study, the only seismic-induced equipment or structural failures explicitly modeled in this assessment are the offsite power insulators.
  - Offsite AC recovery is not credited.
  - Emergency diesel generator recovery is not credited, consistent with the base LaSalle model.
  - As these sequences are DLOOPs and offsite power recovery is not credited, systems dependent upon offsite power (e.g., Feedwater, Condensate, Containment Venting, etc.) are not available to support accident mitigation.
  - Alternate injection using the diesel fire pump is credited for long term accidents (i.e., accidents with initial RPV injection via another system such as RCIC).

- Consistent with the insights of the RMIEP seismic study, seismic-induced RPV Rupture, ISLOCA, LOCA (SORVs following the seismic-induced DLOOP initiators are modeled) and BOC sequences are not explicitly quantified because they are assessed as not significant contributors to seismic risk.

Base Seismic Sequence Quantification Results

The LaSalle base seismic-induced core damage frequency is estimated at 6.6E-8/yr.

The numerical difference between this seismic-induced CDF and that estimated by RMIEP (7.58E-7/yr) is appropriately explained by the following two key factors:

- Use of the more current NUREG-1488 seismic initiator frequencies (the RMIEP frequencies are approximately a factor of 5 higher)
- Refinements to the LaSalle PSA since the RMIEP study (including key contributors such as the reduction in EDG failure rates to reflect plant Maintenance Rule Program data).

The dominant accident sequence types are station blackout scenarios, which represent approximately 80% of the seismic CDF. The dominant cutsets are seismic-initiated DLOOP events with successful RCIC operation and common cause failure of the emergency diesel generators (which result in core damage in approximately 8-9 hours due to battery depletion at 7 hours). These results are consistent with those of the RMIEP study (74% of the RMIEP seismic CDF is represented by such cutsets).

C.4 INTERNAL FIRES ASSESSMENT

This internal fires assessment is based on the extensive work performed for the LaSalle RMIEP study.

C.4.1 RMIEP Internal Fires Overview

The internal fires LaSalle RMIEP study is a detailed analysis that, like the seismic analysis, uses quantification and model elements (e.g., system fault trees, event tree structures, random failure rates, common cause failures, etc.) consistent with those employed in the internal events portion of the RMEIP study. The LaSalle RMIEP internal fires study was performed during the same time frame as the NUREG-1150 studies and The Fire Risk Scoping Study.

The RMIEP internal events study models were used to support sequence quantification. This ensured that the fire sequence quantifications included plant-specific line-up, reliability, and human pre-accident reliability data. Plant walkdowns were performed to document plant-specific combustible loading, suitability of fire severity factors, locations of critical equipment, locations of fire dampers, suitability of doors and other fire barriers, effectiveness of fire detection and suppression systems, and other component specific attributes. Plant-specific cable location data were used to spatially identify control and power cables passing through or powering components in the various fire areas.

The key elements of the LaSalle RMIEP internal fires assessment are consistent with current approaches and include:

1. Fire hazard analysis
2. Fire growth and propagation
3. Fire suppression.
4. Accident sequence development and quantification.

Overviews of these elements are provided below.

Fire Hazard Analysis

The LaSalle RMIEP fire hazard analysis is typical of fire PRA techniques and involves dividing the plant into discrete fire areas, estimating fire ignition frequencies for each fire area, and identifying critical fire areas for detailed quantitative assessment.

The RMIEP study uses the Appendix R fire areas and zones as a starting point for defining discrete fire areas. These areas are modified to account for barriers and equipment separation within fire areas. This partitioning is based on review of plant equipment location and arrangement drawings, plant Fire Hazards Analysis (FHA) discussions, and plant walkdowns. Fire area boundary definitions are based on the following:

- NRC Generic Letter 83-33 (10/19/83) definition of a fire area
- engineering judgment
- available level of detail of cable and component location information

A detailed list of the identified fire areas, descriptions of areas and barriers, and the bases for the boundary assessments are provided in Tables 3.3 and 3.4 of NUREG/CR-4832, Volume 9. Of the 160 LaSalle FHA defined fire zones, 54 PSA fire areas were identified.

The RMIEP fire ignition frequencies are estimated based primarily on the fire events database provided in NUREG/CR-4586, Users' Guide for a Personal-Computer-Based Nuclear Power Plant Fire Data Base (the database is compiled from information presented in NUREG/CR-5088, the Seabrook PSA, and the Limerick Severe Accident Risk Assessment). Fire area ignition frequencies are estimated for the following eight general plant buildings/areas: 1) Control Room; 2) cable spreading room; 3) diesel generator room; 4) electrical switchgear room; 5) battery room; 6) reactor building; 7) turbine building, and 8) auxiliary building. Estimation of specific fire area ignition frequencies is generally calculated as the ratio of the floor area in question to that of the larger building. In some cases, a specific fire area ignition frequency is based on the ratio of the foot-print area of the most probable ignition sources in a fire area (based on walkdown information) to that of the larger building.

To determine the fire areas warranting detailed quantification, the RMIEP study performs an initial screening quantification. The RMIEP internal events fault trees were used to identify all key components and cabling credited in the PSA. Plant schematics were used to map components to locations. Cables were identified from master electrical wiring diagrams. This information and Sargent and Lundy cable routing information for LaSalle were used to map fault tree basic events to associated equipment and cable locations.

The RMIEP internal event transient event tree structure is employed in the initial screening quantification of the fire areas. The fire ignition frequency of each fire area was set to 1.0 and all functions in the area were set to fail using the location information. In addition, a screening fire barrier failure rate of 0.1/demand was applied between fire areas in this initial screening quantification. The initial screening quantification resulted in identification of the following critical fire areas for further detailed quantitative analysis:

ID	Room Description
5C11-4	Diesel Generator Corridor
4D2	Cable Spreading Area
4D4	Electrical Equipment Room
4E2-1	Auxiliary Equipment Room (Main Area)
4E2-2	Auxiliary Equipment Room (Northwest Corner)
4F3	Aux. Bldg. Rad. Chemistry Offices
5B13-2	BOP Cable Area (North)

ID	Room Description
4E4-1	Cable Shaft Area of Div. 2 Ess. SWGR Room
4C1	Control Room
4E4	Div. 2 Ess. SWGR Room
4F2	Div. 1 Ess. SWGR Room

The details of the fire hazard analysis and initial screening quantification are discussed in Sections 3.1 - 3.5 of NUREG/CR-4832, Volume 9.

Fire Growth and Propagation

Discrete fire scenarios were modeled for the critical fire areas that survived the initial screening quantification. The COMPBRN fire growth code was used to model fire growth and fire-induced equipment damage. The RMIEP fire scenarios are generally modeled with two fire types:

- "Small fire", modeled as a 2 ft. diameter 1 gallon oil spill
- "Large fire", modeled as a 3 ft. diameter 10 gallon oil spill

This is a conservative treatment of fire modeling (i.e., compared with the techniques of the EPRI Fire PRA Implementation Guide) and may generally over estimate the fire-induced equipment damage in many areas (e.g., cable spreading room).

The cable damage threshold used was 662°F, and the cable insulation ignition temperature used was 932°F.

Fire propagation in cable trays and hot gas layer effects were treated where appropriate.

Zones of damage were then determined for each fire scenario. Dominant cutsets from the initial screening quantifications were used to identify dominant critical areas in each critical fire area. Using this information, the floor area in a given fire area in which fire-induced damage to equipment of interest to the PSA could occur was estimated.

In addition to the conservative selection of fire types, the RMIEP study employed the following conservative approaches when determining fire-induced equipment damage:

- Fire-induced failure of any Main Steam equipment is modeled as failure of MFW, Condensate, and the PCS
- Fire induced failure of any mode of RHR is modeled as failure of all modes of RHR
- Fire-induced failure of RHR and containment vent is modeled to also fail the PCS.

Fire Suppression

Automatic suppression, when present, and fire brigades were credited for fire scenarios during the time frame before the COMPBRN predicted time to fire-induced equipment damage.

A detailed analysis of manual fire suppression was performed in support of the RMIEP internal fire analysis. The RMIEP manual suppression analysis was supported by plant walkdowns, review of installed suppression system information, review of procedures and practices, and interviews with plant fire personnel. The manual suppression failure probabilities consider: time to detection, time to assemble and suit-up, time to respond to scene, time to set-up at scene, time to search for fire source location, time to control fire.

Credit for automatic suppression systems considered the detector and head spacing with respect to the fire location, as well as the time to fire-induced equipment damage. The

RMIEP automatic suppression failure probabilities are generic industry values taken from the NUREG-1150 external event guidelines (NUREG/CR-4840). The NUREG-1150 guidelines provide failure rates based on five different industry sources: Water (3.8E-2 failure probability); Halon (5.9E-2 failure probability); and CO2 (4.0E-2). The NUREG-1150 automatic suppression system failure probabilities are generally consistent with the values provided in the EPRI FIVE Methodology, these are: Preaction and Deluge Systems (5.0E-2); Sprinkler Systems (2.0E-2); Halon (5.0E-2); and CO2 (4.0E-2).

Accident Sequence Development and Quantification

Each fire scenario that indicated potential fire-induced damage to equipment of interest to the PSA was modeled probabilistically and addressed the following issues:

- building fire ignition frequency
- area ratio of fire area to that of building
- area ratio within fire area where fire scenario results in damage to equipment
- fire severity ratio
- failure of automatic suppression systems
- failure of manual suppression
- random and fire-induced equipment failures

Fire-induced equipment failures were modeled by failing appropriate basic events in the PSA. The fire scenarios were then modeled with the internal events transient accident sequences to quantify the fire-induced core damage frequency for each scenario.

RMIEP Internal Fires Quantification Results

The total fire-induced core damage frequency was estimated in the RMIEP study at a mean value of 3.21E-5/yr. A summary of the RMIEP internal fires modeling and quantification is provided in Table C.4-2.

Consistent with other BWR internal fire PSAs, the dominant fire areas are the Control Room and the Essential Switchgear Rooms.

In all fire areas, additional (i.e., in addition to fire-induced equipment failures) random failures and/or operator errors are necessary to result in a core damage accident. In the case of the Control Room, the dominant scenario (consistent with other fire PSAs) is smoke-induced abandonment of the Control Room and failure to successfully control the plant from the remote shutdown panel.

Excluding the Control Room fire scenario, the majority (99%) of the RMIEP fire-induced core damage accidents are long-term loss of containment heat removal scenarios (Class II). The Control Room fire scenario is conservatively assumed in the RMIEP study to result in a short term high-pressure loss of coolant injection accident (Class IA). Including the Control Room fire scenario, the breakdown is: 56% Class II, 43% Class IA, and 1% Class ID.

The fire-induced core damage frequency estimated for LaSalle in the RMIEP study is at the conservative end of the spectrum for the following reasons:

- The fire-induced damage indicated by the RMIEP fire scenario assessments are known to be conservative (i.e., the RMIEP assessment conservatively failed entire functions given fire induced failure of a portion of a system or of a related system).

- The RMIEP internal fire assessment conservatively assumes that each identified fire scenario represents 100% of the room ignition frequency.
- The Fire Severity factors used in RMIEP are generally conservative when compared to the EPRI Fire PRA Procedures Guide.

C.4.2 Application of RMIEP Internal Fire PSA to LaSalle ILRT Extension

As discussed in the previous section, the RMIEP calculated internal fires induced CDF is a conservative estimate. However, the qualitative conclusions of the RMIEP internal fires assessment are judged still applicable, though specific dominant sequences and cutsets may differ due to plant procedural and PSA model changes.

The LaSalle fire risk is dominated by long term core damage accidents. However, the LERF risk impact due to ILRT frequency changes is dominated by short term core damage accidents. As such, explicit inclusion of internal fires accident frequency information in this ILRT risk assessment would not significantly alter the quantitative results nor would it change the conclusions of this analysis (i.e., the risk impact of ILRT interval extension to 1/15 years is very small). Note that the total LERF remains approximately a factor of 100 below the LERF acceptance guideline in Reg Guide 1.174 (1E-5/yr total LERF). The change in LERF would likewise remain below 1E-7/yr.

Table C.4-2

SUMMARY OF RMIEP INTERNAL FIRE INDUCED CORE DAMAGE ACCIDENTS

(Fire Area) Room	Room Description	Equipment/Cable in Room	Auto Suppr Systems	Fire Scenario	Time to Target Damage (min)	Fire-Induced Equipment Failures Modeled in the Sequence Quant. (1)	Fire Area Ignition Freq (2)	Fire Room: Fire Area (3)	Fire Scen: Fire Room(4)	Fire Severity Ratio	Auto Suppr Failure Prob(5)	Manual Suppr Failure Prob(6)	Approx CCDP (7)	Fire Room CDF	% of Total Fire CDF
(E) 5C11-4	Diesel Generator Corridor	• 241X (CW pumps 2A & C; PSW pumps 2A & 0; SA comp. 2SA01C; MCCs 231X, 231Y, 237X, 237Y) • 232Y-2 (FW pump 2B valves; RHR A service water strainer) • 232B-1 (alt. feed RPS buses A&B) • 125VDC Battery 2A (train A systems) • Offsite Power	None	Very large floor fire (10)	8-9	RCIC, MFW, Condensate, PCS, all LPCS, all RHR	3.36E-2	0.0038	0.30	0.17	1.0	0.83	1E-1 to 2E-1	6.20E-7	1.9
(N) 4D2	Cable Spreading Area	• Cables for train B systems	Auto Sprinkler	Large floor fire (8)	3-5	All train B safety systems, MFW, Cond., PCS, venting	6.48E-3	1.0	0.15	0.30	0.038	0.99	8E-3 to 2E-2	1.63E-7	0.5
(P) 4D4	Electrical Equipment Room	• RPS 120VAC Bus A • MG Set A • RPS 120VAC Bus B • MG Set B • MSIV Closure signal • Train A system cables	None	Large floor fire	7-8	All train A safety systems, MFW, Condensate, PCS, and venting	4.90E-2	0.06	0.05	0.30	1.0	0.97	7E-3 to 2E-2	3.28E-7	1.0
				Small floor fire	3-4	Same fire-induced damage as for the large floor fire			0.016	0.70	1.0	0.97	7E-3 to 2E-2	2.45E-7	0.8
(S) 4E2-1	Auxiliary Equipment Room (Main Area)	• Cables for train B systems	None	Large floor fire #S-AA (8)	4-5	Same as for (AA) 5B13-2: Offsite power and venting	4.90E-2	0.028	0.11	0.30	1.0	0.97	1E-4 to 2E-4	5.94E-9	0.0
				Large floor fire #S-W (8)	4-5	Same as for (W) 4E4: all train B safety systems, MFW, Cond., PCS, venting			0.11	0.30	1.0	0.97	8E-3 to 2E-2	3.52E-7	1.1

Table C.4-2

SUMMARY OF RMIEP INTERNAL FIRE INDUCED CORE DAMAGE ACCIDENTS

(Fire Area) Room	Room Description	Equipment/Cable in Room	Auto Suppr Systems	Fire Scenario	Time to Target Damage (min)	Fire-Induced Equipment Failures Modeled in the Sequence Quant. (1)	Fire Area Ignition Freq (2)	Fire Room: Fire Area (3)	Fire Scen: Fire Room(4)	Fire Severity Ratio	Auto Suppr Failure Prob(5)	Manual Suppr Failure Prob(6)	Approx CCDP (7)	Fire Room CDF	% of Total Fire CDF
(T) 4E2-2	Auxiliary Equipment Room (Northwest Corner)	• Cables for train A systems	None	Large floor fire (8)	9-10	All train A safety systems, MFW, Cond., and PCS	4.90E-2	0.068	0.84	0.30	1.0	0.97	~3E-3	2.27E-6	7.1
(Z) 4F3	Aux. Bldg. Rad. Chem. Offices	• Cables for train A systems	Partial Sprinkler Coverage	Large floor fire (8)	5-6	All train A safety systems, MFW, Cond., PCS, venting	4.90E-2	0.082	0.005	0.30	1.0	0.91	7E-3 to 2E-2	3.58E-8	0.1
(AA) 5B13-2	BOP Cable Area (North)	• 242X (CW pump 2B; PSW pump 2B and jockey 0B; MCCs 232X, 232Y, 238) • Cables for train A systems	None	Large floor fire (8)	6-8	Offsite power and venting	4.90E-2	0.064	0.08	0.30	1.0	0.93	1E-4 to 2E-4	7.31E-9	0.0
(AC) 4E4-1	Cable Shaft Area of Div. 2 Ess. SWGR Room	• Cables for train B systems (11)	None	Small floor fire (9)	2-3	All train A safety systems (11), MFW, Cond., PCS, venting	4.90E-2	0.0016	1.0	1.0	1.0	0.99	7E-3 to 2E-2	5.42E-7	1.7
(G) 4C1	Control Room	• Cables for train A, B and C systems	None	(12)	(12)	(12)	(12)	(12)	(12)	(12)	(12)	(12)	(12)	1.39E-5	43.3
(W) 4E4	Div. 2 Ess. SWGR Room	• 252 (train B non-safety AC) • 242Y (train B safety AC) • 236X (DGCWP 2A; RHRSW pump 2C) • 236X-2 (WW vent; MG Set B) • 236X-3 (125VDC train B charging) • 236Y (RHRSW pump 2D; RHR train B and C; DW vent; RCIC & RBCCW isolations; FW turbines; SLC train B) • 125VDC 2B Battery, Bus and charger (train B systems)	None	Switchgear cubicle fire	4-5	All train B safety systems, MFW, Condensate, PCS, venting	7.97E-3	1.0	1.0	0.01 (13)	1.0	0.98	8E-3 to 2E-2	1.80E-6	5.6

Table C.4-2

SUMMARY OF RMIEP INTERNAL FIRE INDUCED CORE DAMAGE ACCIDENTS

(Fire Area) Room	Room Description	Equipment/Cable in Room	Auto Suppr Systems	Fire Scenario	Time to Target Damage (min)	Fire-Induced Equipment Failures Modeled in the Sequence Quant. (1)	Fire Area Ignition Freq (2)	Fire Room: Fire Area (3)	Fire Scen: Fire Room(4)	Fire Severity Ratio	Auto Suppr Failure Prob(5)	Manual Suppr Failure Prob(6)	Approx CCDP (7)	Fire Room CDF	% of Total Fire CDF
		• 125VDC 212X (FW pump 2B, DC to non-safety train B systems) • 125VDC 212Y (ADS train B, DC to train B safety systems)													
				Large floor fire (8)	4-5	Same fire-induced damage as for the SWGR cubicle fire			0.18	0.30	1.0	0.98	8E-3 to 2E-2	6.71E-6	20 9
(Y) 4F2	Div. 1 Ess. SWGR Room	• 251 (train A non-safety AC) • 241Y (train A safety AC) • 235X (DGCWP 0; RHRSW pump 2A; WW vent; RCIC & SDC isolations) • 235X-2 (MG Set A) • 235X-3 (125VDC train A and 250VDC charging) • 235Y (RHRSW pump 2B; RHR train A; LPCS; DW vent; SLC train A) • 125VDC 2A Bus and charger (train B systems) • 125VDC 211X (DC to non-safety train A systems) • 125VDC 211Y (DC to train A safety systems) • 250VDC 2 Battery, Bus and charger (RCIC, all 250VDC)	None	Switchgear cubicle fire	4-5	All train A safety systems, MFW, Condensate, PCS, venting	7.97E-3	1.0	1.0	0 01 (13)	1.0	0.95	7E-3 to 2E-2	1.76E-6	5.5
				Large floor fire (8)	4-5	Same fire-induced damage as for the SWGR cubicle fire			0.13	0.30	1.0	0.95	7E-3 to 2E-2	3.39E-6	10.6

Notes to Table C.4-2:

1) Deterministic fire modeling was performed using COMPBRN. The RMIEP study modeled fires with two general fire scenarios, a "small" 1 gallon oil fire and a "large" 10 gallon oil fire. This is a conservative treatment of fire modeling and may generally over estimate the fire-induced equipment damage in many areas (such as a cable spreading room). In addition, the RMIEP study made the following additional conservative assumptions when modeling fire-induced equipment failures: 1) fire induced failure of any main steam equipment was modeled as failure of MFW, Condensate and the PCS; 2) fire induced failure of one mode of RHR was modeled as failing all modes of RHR; and 3) modeling fire induced failure of RHR and Vent was extrapolated to also imply failure of the PCS. These lists of fire-induced equipment failures by fire scenario are based on review of cutsets and text discussions in the RMIEP internal fire analysis documentation (NUREG/CR-4832, Vol. 9).

2) The RMIEP fire ignition frequencies are based on the NUREG-1150 external event guidelines (NUREG/CR-4840). The NUREG-1150 guidelines provide a compilation of fire events by eight key plant buildings/areas. The data is complied from information presented in NUREG/CR-5088, the Seabrook PSA, and the Limerick Severe Accident Risk Assessment.

3) The Fire Room to Fire Area ratio is a ratio of the floor area of the fire room to that of the larger fire area, and is used to partition the fire area ignition frequency to apply to the fire room in question.

4) The Fire Scenario to Fire Room ratio is a ratio of the floor area within the fire room in question where the fire scenario in question may be located and cause the damage of interest.

5) The RMIEP automatic suppression failure probabilities are generic industry values taken from the NUREG-1150 external event guidelines (NUREG/CR-4840). The NUREG-1150 guidelines provide failure rates based on five different industry sources. The recommended generic values are: Water (3.8E-2 failure probability); Halon (5.9E-2 failure probability); and CO_2 (4.0E-2). The NUREG-1150 automatic suppression system failure probabilities are generally consistent with the values provided in the EPRI FIVE Methodology, these are: Preaction and Deluge Systems (5.0E-2); Sprinkler Systems (2.0E-2); Halon (5.0E-2); and CO_2 (4.0E-2).

6) The RMIEP manual suppression failure probabilities are based on LaSalle fire area specific analyses which consider: time to detection, time to assemble and suit-up, time to respond to scene, time to set-up at scene, time to search for fire source location, time to control fire. The RMIEP manual suppression analysis was supported by plant walkdowns, review of installed suppression system information, review of procedures and practices, and interviews with plant fire personnel.

Notes to Table C.4-2 (cont'd)

7) Review of the RMIEP fire core damage cutsets and back-calculation of the CCDPs produces slightly (in the factor of 2-3 range) varying CCDPs for the same fire-induced damage states. This variance is due to cutset truncation limits and potential minor mis-interpretations of the fire-induced equipment damage (as represented in the RMIEP cutsets). Provided here for information.

8) Per RMIEP, a small floor fire does not damage the cables of interest in this area.

9) Per RMIEP, a small floor fire is sufficient by itself to damage the cables of interest in this area (a large floor fire will also damage the cables of interest). However, the time to damage in either case is very similar and very quick (1-3 min.) for this small room (4E4-1), and the fire location area to room area ratio is the same in both the small and large fire scenarios (i.e., 1.0 - a small or large fire anywhere in the room is sufficient enough to damage the cables of interest), that RMIEP quantified an accident sequence for a single scenario (the small fire) rather than two scenarios. No large fire: small fire ratio was applied in the RMIEP frequency analysis for this fire area.

10) Per RMIEP, a large floor fire does not damage the cables of interest; however, due to the important cabling in the area, RMIEP assumes a <u>very</u> large fire (with a severity factor assumed to be half that of a large fire).

11) RMIEP documentation and/or quantification appears to be in error (although, the 4E4-1 fire scenario CDF is not significantly impacted given the similarity in train A and train B system importances). The documentation in Appendix B of the RMIEP fire analysis (NUREG/CR-4832, Vol. 9) states the following regarding equipment in fire location 4E4-1: "No equipment important to safety in this room. Train B cable spreading area." These two sentences appear conflicting; however, the quantification of this fire area, as documented on pp. F-51 thru F-56 of the RMIEP fire analysis, is an additional contradiction in that random failures of train B equipment are credited and train A equipment appears to be failed by the fire.

12) The RMIEP fire analysis modeled the Control Room with the following fire scenario: Fire starts in a Control Room panel/cabinet (1.85E-3/yr frequency), the fire is not suppressed before smoke requires abandonment of the Control Room (0.10 probability), and the operators do not successfully recover the plant from the Remote Shutdown Panel (6.4E-2 probability).

13) The RMIEP switchgear cubicle fire is assigned a probability of 0.01 that the fire exits the top of the switchgear due to an inadequate seal; no area or severity ratios are applied.

REFERENCES

[C-1] ERIN Engineering and Research, Inc., Technical Evaluation of Extending LaSalle Diesel Generator Completion Time (CT) Using Probabilistic Risk Assessment Models for LaSalle, ERIN doc. #C1349911-4029, Rev. 2, March 2000.

Appendix D

MODEL FILES AND MODIFICATIONS

This appendix summarizes the following aspects of the LaSalle ILRT extension risk assessment:

- Model files used in the analysis
- Modifications made to LaSalle PSA models in support of this analysis

Level 1 PSA Model

The Level 1 PSA information used in this risk assessment is based directly on the LaSalle 2001A CDF model. The individual LaSalle 2001A CDF model files are listed in Appendix B of Exelon RM Document No. 790, Rev. 0, "LaSalle 2001A Core Damage Frequency (CDF) Models", and are not repeated here.

A "sequence labeling" error in the LaSalle Revision 2001A CDF sequence model was identified as part of this risk assessment application. The error in question exists in the PRAQuant batch sequence quantification file L2PREQAS.qnt (date 8/22/01). In this file, the following ATWS sequences (all belonging to the MCSTAT.ETA event tree) were assigned incorrect accident class categories (note that the class assignments on the MCSTAT.ETA tree itself are correct):

Accident Sequence ID	Erroneous Accident Class	Correct Accident Class
ATW1-09	IC	IV
ATW1-11	IV	IC
ATW1-14	IC	IV
ATW1-16	IV	IC
ATW1-17	IC	IV

Accident Sequence ID	Erroneous Accident Class	Correct Accident Class
ATW1-20	IV	IC
ATW1-28	IC	IV
ATW1-30	IV	IC
ATW1-31	IC	IV
ATW1-34	IV	IC
ATW1-36	IC	IV
ATW1-43	IV	IC

As can be seen from the above list, the labeling errors involved mislabeling of a dozen Class IC and IV core damage sequences, such that the IC accident class CDF was totaling higher than the correct amount and the IV accident class CDF was totaling lower that the correct amount. These errors are "labeling" ones and do not involve any modeling logic or probabilities. Hand calculations were used in the LaSalle ILRT risk assessment to re-assign IC and IV accident frequencies to their proper location. No PSA model changes were necessary or made, but a URE was created to track this error for correction in the next LaSalle PSA update.

Level 2 PSA Model

As discussed in Appendix B of this risk assessment, the LaSalle Level 2 PSA LERF containment event trees (CETs) have been extended as part of this analysis to provide additional release category information (i.e., not just LERF). The CET extension effort required the following:

- Appending additional nodes onto the non-LERF accidents, or assigning a single specific non-LERF release category, as appropriate (see App. B)
- Development of a Containment Flooding (FC) fault tree
- Re-classification of Class II releases as Intermediate

The model files associated with the extended CETs of this risk assessment are summarized in Table D-1. The majority of the files are ETA (EPRI R&R Workstation Event Tree Analysis code) files, one for each of the CETs. The final four files listed In Table D-1 are related to the development and quantification of a Containment Flooding fault tree for use in the extended CETs. In the LaSalle Level 2 PSA LERF models, the LERF CET node (FC) for Containment Flooding is set to a probability of 1.0 to hard-wire transfer all accidents that reach the decision point of containment flooding to a non-LERF endstate (refer to Appendix C.7 of the LaSalle Level 2 PSA LERF documentation for details as to why containment flooding scenarios do not result in LERF). However, in this risk assessment application information beyond the LERF release risk measure is required; as such, a failure probability for containment flooding had to be estimated for this analysis (i.e., if containment flooding proceeds successfully then the release occurs in the Intermediate time frame, if containment flooding is not implemented then the release will be due to containment failure and will be Late).

In addition, the Class II release categories were modified. The current LaSalle Level 2 PSA models Class II accidents as proceeding to early releases, based on the assumption that a General Emergency would not be declared for such accidents until very late in the accident sequence. Based on a re-evaluation by Exelon, this assumption has been proven to be conservative. [D-1] As such, the LERF model modifications performed in support of this risk application include reclassifying releases for Class II accidents to the Intermediate time frame.

<u>LaSalle ILRT Risk Assessment Calculational Files</u>

The LaSalle ILRT risk assessment calculations performed per the NEI Interim Guidance were performed using an Excel spreadsheet file:

4900-467.xls	56kb	6/26/02

Table D-1

LIST OF LEVEL 2 PSA FILES USED IN LASALLE ILRT RISK ASSESSMENT

File Name	Bytes	Date	Description of File
IA.eta	8,733	5/13/02	Class IA CET
IAX1.eta	5,349	5/13/02	Supplemental CET for Class IA (Page 1)
IAX2.eta	5,337	5/13/02	Supplemental CET for Class IA (Page 2)
IBE.eta	8,727	5/13/02	Class IBE CET
IBEX1.eta	5,342	5/21/02	Supplemental CET for Class IBE (Page 1)
IBEX2.eta	5,342	5/21/02	Supplemental CET for Class IBE (Page 2)
IBL.eta	8,721	5/13/02	Class IBL CET
IBLX1.eta	5,342	5/21/02	Supplemental CET for Class IBL (Page 1)
IBLX2.eta	5,330	5/21/02	Supplemental CET for Class IBL (Page 2)
IC.eta	8,712	5/13/02	Class IC CET
ICX1.eta	5,341	5/13/02	Supplemental CET for Class IC (Page 1)
ICX2.eta	5,329	5/13/02	Supplemental CET for Class IC (Page 2)
ID.eta	8,726	5/13/02	Class ID CET
IDX1.eta	5,344	5/21/02	Supplemental CET for Class ID (Page 1)
IDX2.eta	5,332	5/21/02	Supplemental CET for Class ID (Page 2)
IE.eta	8,726	5/13/02	Class IE CET
IEX1.eta	5,333	5/13/02	Supplemental CET for Class IE (Page 1)
IEX2.eta	5,337	5/13/02	Supplemental CET for Class IE (Page 2)
II.eta	9,404	5/13/02	Class II CET
IIIB.eta	8,303	5/13/02	Class IIIB CET
IIIBX1.eta	5,350	5/21/02	Supplemental CET for Class IIIB (Page 1)
IIIBX2.eta	5,338	5/21/02	Supplemental CET for Class IIIB (Page 2)
IIIC.eta	8,312	5/21/02	Class IIIC CET
IIICX1.eta	5,345	5/21/02	Supplemental CET for Class IIIC (Page 1)
IIICX2.eta	5,333	5/21/02	Supplemental CET for Class IIIC (Page 2)
IIID.eta	7,885	5/13/02	Class IID CET
IV.eta	9,309	5/13/02	Class IV CET
V.eta	7,881	5/13/02	Class V CET
FC1-ILRT.caf	98,163	5/8/02	Containment Flooding fault tree file for extended CET
L2-FC1.be	662,528	5/8/02	LaSalle L2 PSA database (BE) file included added basic events for containment flooding fault tree
L2-FC1.gt	4,668,416	5/8/02	LaSalle L2 PSA database (GT) file included added gate information for containment flooding fault tree
L2-FC1.tc	4,096	10/16/99	LaSalle L2 PSA database (TC) file (unchanged from base model)

REFERENCES

[D-1] Letter from J.E. Steinmetz (Exelon) to E.T. Burns (ERIN), "Evaluation of Emergency Response Organization Declaration of a GSEP During a Loss of Decay heat Removal Sequence", Support Application: SA 827, February 6, 2002.