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INDIANA
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August 24, 2000

C0800-15

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50-316

U. S. Nuclear Regulatory Commission
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Donald C. Cook Nuclear Plant Units 1 and 2
REVIEW OF FOUR PRELIMINARY ACCIDENT
SEQUENCE PRECURSOR ANALYSES OF OPERATIONAL EVENTS AND
REVIEW OF DRAFT NUREG "ASSESSMENT OF RISK SIGNIFICANCE
ASSOCIATED WITH ISSUES IDENTIFIED AT D. C. COOK NUCLEAR
POWER PLANT"

References:
1. Letter, from John A. Grobe, Nuclear Regulatory Commission to R. P. Powers, Indiana Michigan Power Company, "Review of Four Preliminary Accident Sequence Precursor Analyses of Operational Events at D.C. Cook Nuclear Plant," dated May 9, 2000.
2. Letter from John A. Grobe, Nuclear Regulatory Commission to R. P. Powers, Indiana Michigan Power Company, "Donald C. Cook Plant, Units 1 and 2 – Issuance of Draft Report Entitled 'Assessments of Risk Significance Associated with Issues Identified at D.C. Cook Nuclear Plant'," dated May 17, 2000.

In References 1 and 2, the Nuclear Regulatory Commission (NRC) transmitted for review and comment two documents related to risk assessment for Donald C. Cook Nuclear Plant (CNP). Reference 1 transmitted four preliminary Accident Sequence Precursor (ASP) analyses of operational conditions that were identified at CNP during the 1999 calendar year. Reference 2 transmitted a draft NUREG assessing the risk of 141 issues identified at CNP between August 1997 and December 1999. The NRC requested that comments to both documents be provided within 60 calendar days of receipt of the letters. In a series of

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subsequent discussions with the NRC staff, Indiana Michigan Power Company (I&M) committed to provide the comments by August 24, 2000.

The attachment to this letter provides the requested comments. As documented in the attachment, I&M has identified instances where the methodology used was not consistent with published guidance. I&M has also provided additional information affecting the assumed frequency of some events. Changes to the affected analyses based on these considerations would result in reductions in the calculated core damage frequency.

No new commitments are identified in this letter.

Should you have any questions, please contact Mr. Wayne J. Kropp, Director of Regulatory Affairs at (616) 466-2447.

Sincerely,

MW Rencheck

M. W. Rencheck
Vice President Nuclear Engineering

/dmb

Attachment

c: J. E. Dyer
MDEQ - DW & RPD, w/o attachment
NRC Resident Inspector
R. Whale, w/o attachment

ATTACHMENT TO C0800-15

INDIANA MICHIGAN POWER COMPANY COMMENTS ON:

NRC LETTER "FOUR PRELIMINARY ACCIDENT SEQUENCE PRECURSOR ANALYSES OF OPERATIONAL EVENTS AT D.C. COOK NUCLEAR PLANT"

AND

DRAFT NUREG ENTITLED "ASSESSMENT OF RISK SIGNIFICANCE ASSOCIATED WITH ISSUES IDENTIFIED AT D.C. COOK NUCLEAR PLANT"

Background

The primary objective of the Nuclear Regulatory Commission's (NRC) Accident Sequence Precursor (ASP) Program is to identify and rank the risk significance of operational events and conditions at nuclear power plants. These events and conditions are typically documented in Licensee Event Reports (LERs), NRC Inspection Reports, and licensee corrective action documents. Accident sequences of interest to the ASP Program are those that would have resulted in inadequate core cooling, which could have caused severe core damage if additional failures had occurred. Events or conditions considered to be potential accident precursors are analyzed, and a conditional core damage probability (CCDP) is calculated by mapping failures observed during the event onto accident sequences in risk models.

In References 1 and 2, the NRC transmitted for review and comment two documents related to risk assessment for Donald C. Cook Nuclear Plant (CNP). Reference 1 transmitted four preliminary accident sequence precursor (ASP) analyses of operational conditions that were identified at CNP during the 1999 calendar year. Reference 2 transmitted a draft NUREG entitled "Assessment of Risk Significance Associated With Issues Identified at D. C. Cook Nuclear Plant" that assessed the risk of 141 issues identified at CNP between August 1997 and December 1999. The draft NUREG includes the four preliminary ASP analyses that were transmitted by Reference 1.

Indiana Michigan Power Company's (I&M) comments are presented in two sections. Section I consists of general comments to the draft NUREG transmitted by Reference 2. Section II consists of comments that are specific to the four preliminary ASP analyses that are common to Reference 1 and the draft NUREG transmitted by Reference 2.

Section I General Comments

Scope of Issues

As requested in Reference 2, I&M performed a review to identify any potentially risk significant issues at CNP that were not analyzed by the NRC in the draft NUREG. This review was performed by comparing the issues identified in the draft NUREG with the issues identified in the NRC Inspection Manual Chapter 0350 Restart Action Plan for CNP - Unit 2 (Reference 3). The list of issues contained in Reference 3 was compiled from issues identified in the NRC inspections and Licensee Event Reports, and issues identified via self assessment (i.e., I&M's Expanded System Readiness Reviews). I&M determined that the analyses in the draft NUREG included many of the issues contained in Reference 3. I&M did not identify any potentially risk-significant issues at CNP that the NRC had not analyzed in the draft NUREG. Many of the issues identified in Reference 3 were appropriately not included in the draft NUREG (e.g., issues that related to programmatic deficiencies).

Technical Adequacy of the Analysis

As requested by Reference 2, I&M has evaluated the technical adequacy of the analysis of risk significance of issues identified as potential precursors or non-precursors. I&M has identified several concerns regarding the analysis. The concerns involve apparent inconsistencies in the ASP threshold, the use of simplified models, the methodology used, the use of 1.0 as an estimation of postulated event frequency parameter, and the method of implementing Bayesian analysis. These concerns are discussed in the following paragraphs.

ASP Threshold

The draft NUREG uses a unique threshold consisting of a modified change in core damage frequency (ΔCDF) rather than the ASP threshold defined in NUREG CR-4674 (Reference 4) as an increase in core damage frequency (CDF) of greater than 1×10^{-6} events per year. I&M considers the ΔCDF threshold used in the draft NUREG to be modified because many aspects of a conventional quantitative model are not included, such as recovery and human reliability analysis. In I&M's opinion, the modified ΔCDF threshold used in the draft NUREG may result in misleading conclusions. The modified ΔCDF appears to be a screening value and is not representative of the threshold defined in NUREG CR-4674. Additionally, it is not apparent that the guidance in NUREG CR-4674, Section 2.2, for using different probabilities and their effects on the threshold was considered as part of the analysis.

Also, there appears to be an inconsistency in the draft NUREG regarding thresholds. In Paragraph 1.1 of the draft NUREG, the introduction states that the CCDP is calculated as the threshold. However, on Page 4 of Volume 1 of the draft NUREG, it is stated that the ΔCDF was calculated instead of using the CCDP.

Simplified Models

The draft NUREG uses simplified stand-alone models rather than the standard ASP technique used in NUREG CR-4674 of quantitative models. The approach of using lists of questions to represent simplified stand-alone models can be confusing. Additionally, these simplified models appear to be very sensitive to small changes in event frequencies.

Methodology

The methodology used for the issues identified as precursors do not appear to be explicitly identified. The executive summary of the draft NUREG states that the ASP methodology (which was defined in NUREG CR-4674) was applied to estimate the risk significance associated with the postulated events. That methodology includes the use of a standardized plant analysis risk (SPAR) model, which is based on the plant's individual plant examination (IPE) model (i.e., event trees and fault trees). Some event trees from the CNP specific SPAR models are provided in the report. However, only one of the cases identified as a potential ASP event uses a SPAR model fault tree. The other analyses provided in the draft NUREG appear to consist of specially constructed, simplified event trees.

Based on evaluation of the sequences, the practice of using segregated, simplified event trees may lead to discounting success paths, which could overestimate the significance of the postulated event. In addition, these simplified event trees appear to be overly sensitive to small changes in initiating event frequencies. The simplified event tree methodology used in the draft NUREG is inconsistent with the ASP methodology specified in NUREG CR-4674.

Use of 1.0 as an Estimate of a Postulated Event Frequency

The draft NUREG uses 1.0 as an estimation of a postulated frequency for many events. This presents several concerns. The most important of these is the assumption that the hypothetical event has occurred as described in the scenario. Although none of the events identified as potential precursors has occurred at CNP, the use of 1.0 inappropriately implies that the postulated event will occur with certainty. The use of 1.0 as an estimate of a postulated event frequency is clearly more deterministic than statistical.

Also, a postulated event frequency of 1.0 has been used in the draft NUREG for quantification of several events that are documentation discrepancies, such as design basis calculation errors. The documentation discrepancies do not, in and of themselves, render a structure, system or component (SSC) inoperable or otherwise incapable of performing its intended function for the purpose of establishing a postulated event frequency. I&M recommends that another Bayesian technique, such as a "non-informative prior," be used rather than using 1.0 as an estimate of a postulated event frequency. This is consistent with NUREG CR-4674 and the NRC PRA Procedures Guide, (Reference 5).

Bayesian Methodology

Page 6 of the discussion of Issue #122 in the draft NUREG indicates that half a failure was added to the 12 postulated High Energy Line Break (HELB) failures for use in determining a HELB initiating event frequency. The technique is not included in the NRC PRA Procedures Guide, Chapter 5, "Data-Base Development," as referenced in the ASP methodology, defined in Reference 4. The addition of failures is typically used only for cases of zero actual failures. Typically, Bayesian updating is performed by assuming 1/3, 1/2, or 1 failure for the zero actual failures in the exposure period under consideration to obtain a "posterior." The NRC PRA Procedures Guide suggests the use of a non-informative prior to obtain this posterior. A prior distribution is obtained using generic data. The estimates are then combined using one of the many published forms of Bayes' Theorem to derive an updated estimate of the event frequency.

Adding half to the denominator of the estimate is not a standard method of Bayesian analysis. An example of the addition of half to both the number of failures and to the denominator occurs in Paragraph D3 of Page 12 in the discussion of Issue #122. I&M considers that the methodologies used should be consistent with NUREG CR-4674 and consistent among the scenarios and sequences.

An alternative approach that would address the above discussed concerns regarding methodology, use of 1.0 as an event frequency, and the method of applying certain Bayesian statistics, would be to use CNP's IPE probabilistic risk assessment (PRA), especially when the actual IPE PRA model is available for analyzing these issues.

Other Comments on the Draft NUREG

The information pertaining to a given issue is dispersed among several appendices. A table of contents for each appendix that relates the issues under consideration to the issue numbers would facilitate use of the draft NUREG.

Using tables to define and describe the events and relate the probability estimates described in the text would make it easier to appreciate the analyst's assumptions and would facilitate understanding the quantification.

I&M suggests that the format used in Reference 4 be used in future alternate ASP analyses. This format presents the information in an integrated and structured manner and contains event trees, tables, system and component drawings, and other information important to the analysis.

Section II - Specific Comments

Of the 141 issues assessed in the draft NUREG, there were five issues for which there was an estimated increase in CDF which exceeds the NRC's ASP criterion of 1×10^{-6}/year. As noted in Reference 2, I&M has previously provided comments to the NRC on one of the five issues (failure of all component cooling water (CCW) pumps caused by a HELB). The analyses of the remaining four issues are the same as those identified in Reference 1. The remaining four issues that were estimated to exceed the ASP criterion are:

1) Loss of essential service water (ESW) as a result of a seismic event and the subsequent failure of the ESW strainer backwash capability. [Issue 61]

2) HELB programmatic inadequacies postulated to result in the following scenarios: [Issue 127]

    a) A HELB in the turbine building postulated to result in the loss of auxiliary feedwater (AFW).
    b) A HELB postulated to result in the loss of both safety-related and non safety-related 600 V and lower busses. This issue resulted in two scenarios exceeding the ASP threshold.

c) A HELB postulated to result in the loss of both emergency diesel generators (EDG).

3) Postulated pressure locking/thermal binding of emergency core cooling system (ECCS) and RHR valves. [Issue 135]

4) Failure of safety related block walls due to a postulated seismic event. [Issue 134]

I&M's comments on these four issues are provided below. It appears that simplified event trees were used in Reference 1 and the draft NUREG to analyze the precursor issues. However, the event trees were not provided in Reference 1 or the draft NUREG. As a review aid, I&M constructed event trees for some of the four issues based on the text provided in References 1 and 2. Those event trees are included in the section providing specific comments on the four ASP analyses. It is important to note that all five issues identified in the draft NUREG as exceeding the NRC's ASP criteria have been addressed satisfactorily by physical plant modifications, procedure revisions, equipment qualification, or training, or combinations thereof.

I&M has not provided specific comments with respect to the other postulated events that were analyzed in the draft NUREG because the analyses determined that the events were not risk significant even though very conservative risk analysis techniques were used.

Lack of Procedure for Manually Backwashing the ESW Pump Discharge Strainers (Issue 61)

Background

There is a duplex strainer installed at each ESW pump discharge. Each duplex strainer consists of two independent strainers with independent strainer baskets. Only one of the strainer baskets is required to support the ESW pump function. The other strainer basket is in standby. When the operating strainer basket differential pressure (Δp) exceeds a preset limit, the strainer baskets in the duplex strainer are automatically shifted. The standby strainer basket that was removed from service is then backwashed, via an automatic backwash cycle. After backwashing, the strainer is ready to be placed in service if the operating strainer basket reaches its Δp limit. The strainer baskets can also be backwashed and shifted manually. Shifting of the strainer baskets has no significant effect on ESW operation.

NRC inspection reports in 1998 and 1999 documented that there was no procedure at that time for manually backwashing the ESW pump discharge strainers, the evolution would require tools which were not readily available, and the operators had not been trained in how to perform a manual backwash of the strainers. In addition, the NRC inspectors concluded that there were degraded material conditions that decreased the automatic backwash capability during earthquakes and other events and therefore had the potential to impact CDF sequences relative to seismic and other events.

Comments

The following comments apply to Sequence 2, "Loss of Offsite Power and ESW due to Earthquake and AFW failure," on Pages 6 through 8 in the discussion of Issue #61 in References 1 and the draft NUREG.

In the postulated event, an earthquake occurs causing failure of offsite power and automatic ESW backwashing capability. The automatic backwashing function is demanded due to the earthquake stirring up sand and debris in the lake. Manual backwashing fails, leading to EDG unavailability and core damage occurs due to the station blackout.

Although there was no procedure to manually backwash the strainers, I&M has concluded that it would be unlikely that the ESW system would fail due to plugging of the strainer caused by a seismic event. The basis for this conclusion is provided below.

As stated on Page 8 in the discussion of Issue #61, it was assumed that the energy added by a seismic event would generate waves and turbulence in the lake similar to those induced during stormy weather conditions and this would result in a challenge to the automatic backwash capability of both ESW trains. However, the ESW system at CNP has never functionally failed despite the occasional very severe weather conditions that occur on Lake Michigan.

If sand were to be ingested into the ESW system, it would not be expected to plug the strainer. The openings in the strainer are approximately 1/8 inch in diameter. In a recent sand particle size evaluation, the largest dimension of the typical particle found was less than 1/8 inch. The sand would most likely pass through the strainer element and the ESW system with little or no effect on the system operation.

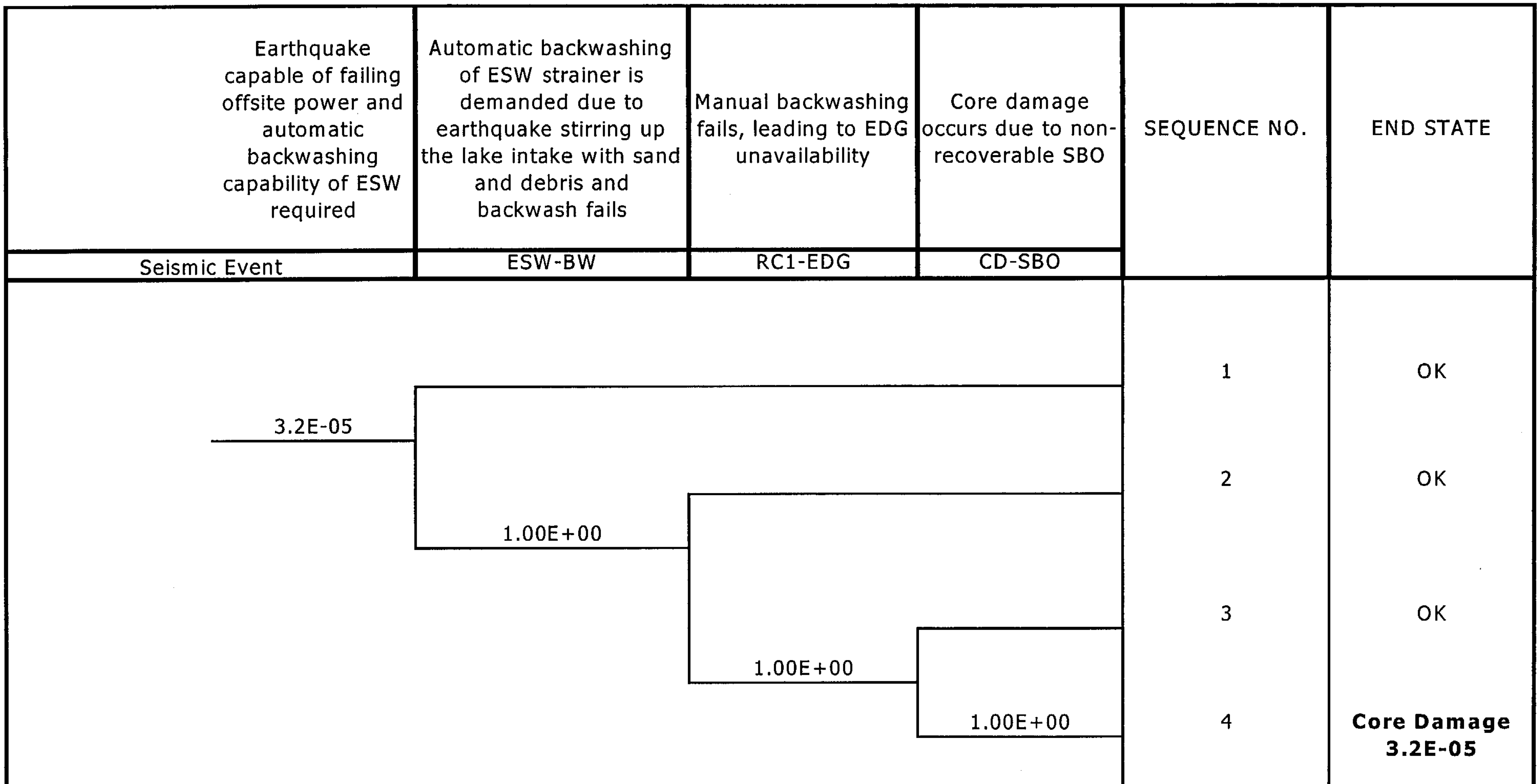
Even if the sand did plug one strainer, the clogged duplex strainer would have to fail to shift, or after the strainer shifts, the clean strainer basket would have to plug or otherwise fail. In addition, the standby ESW pump and its associated strainer baskets would have to fail. Even if all four operating strainer baskets were to plug and automatic backwashing were to fail, or if the strainers failed to shift, the operators and maintenance staff have more than two hours (based on the current CNP IPE) to restore at least one strainer and place an ESW pump in service. Restoration of ESW would allow operation of at least one EDG and provide ventilation cooling to any of the AFW pump rooms. Even without room cooling, the turbine driven AFW pump can operate 4 hours once the door to the room is open (per the station blackout analysis).

In summary, I&M's position is that a design basis earthquake would not be expected to cause a loss of the ESW system due to plugging of the strainers. Therefore, consideration should be given to reducing the failure estimate for this event.

Based on the preceding discussion, I&M believes that the analysis in Sequence 2 should include the frequency of failures of both the standby ESW pump and the turbine driven AFW pump. If each of these events is conservatively estimated to have a failure probability of 0.1 ($1.00 \cdot 10^{-1}$), then a CDF of 3.06×10^{-7} would be obtained. This value is below the ASP threshold.

With respect to the review criteria provided in Reference 1, Attachment 5, "Guidance for Licensee Review of Preliminary ASP Analysis," the above comments may be categorized as modeling assumptions that do not accurately describe the modeling of the event appropriate for the events that occurred or that had the potential to occur under the event conditions, including assumptions regarding the likelihood of equipment recovery.

Figures ESW1 and ESW2 are provided to compare the event trees described. Note that the small difference in the values of the initiating event frequency for "Earthquake capable of failing offsite power and automatic backwashing capability of ESW required" between figures ESW1 and ESW2 are due to rounding inaccuracies (i.e., the value was recalculated for ESW2).



Original NRC ASP Analysis

Figure ESW1 - Original NRC Analysis of Lack of Procedure for manually backwashing ESW pump discharge strainer

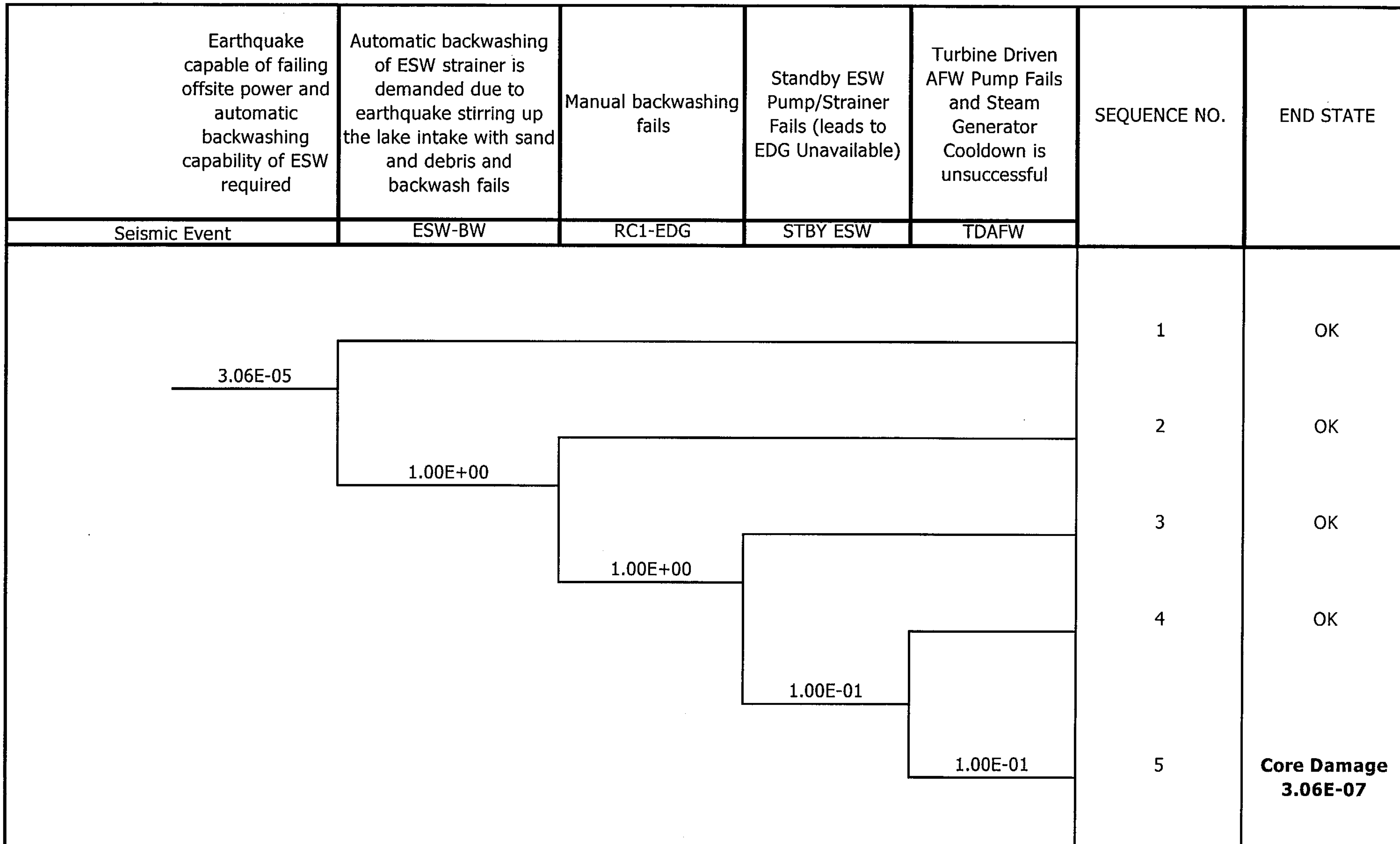


Figure ESW2 - Revised Analysis of Lack of Procedure for manually backwashing ESW pump discharge strainer

HELB Programmatic Inadequacies Result in Unanalyzed Conditions (Issue 122)

Background

A 1999 LER documented that a number of plant locations had to be considered unprotected from the effects of a HELB event. As indicated in Reference 1 and the draft NUREG, four of the HELB event sequences analyzed by the NRC were determined to exceed the threshold of 1.0×10^{-6} events per year. Comments to these sequences are provided below. An additional event sequence is discussed since an order of magnitude change in the probability would result from I&M's recommendation.

General Comments

As noted in Reference 2, I&M has provided comments to the NRC concerning a postulated HELB in a pipe chase affecting the CCW pumps. Those comments were provided by Reference 6. In Reference 6, I&M provided justification for reducing the assumed initiating event frequency at CNP based on the applicability of the historical HELB data used in the NRC analyses. I&M believes that similar justifications apply to reducing the initiating event frequencies for the five event sequences discussed below. However, since each of the conditions has been addressed by plant modifications, I&M has chosen not to develop these justifications.

For clarity in the following discussions, event trees have been developed to illustrate the NRC analyses and the effects of suggested minor changes in assumptions. These sensitivity studies use event frequencies of 0.5 in place of 1.0 to provide insight into the significance of the postulated event. An event frequency of 0.5 was chosen since it does not reduce the event frequency by an order of magnitude, but is still considered a conservative event frequency estimate. Since an event frequency of 1.0 is the maximum event frequency achievable, a sensitivity study with an increased event frequency was not possible. These small changes in assumed event frequencies were found to result in a significant change to CDF, which indicates the sensitivity of the analysis provided in the report. Additional comments applicable to a particular sequence are provided in the specific sections below.

Comments to HELB Sequence 1 Scenario B

Sequence 1 results from a HELB in high-energy equipment in the vicinity of the switchgear room door. The following comments apply to Pages 5-10 in the discussion of Issue #122 in References 1 and the draft NUREG

The sensitivity study resulted in the CDF changing from 1.3×10^{-4} to 1.71×10^{-5}. The event tree based on the NRC analysis and the event tree containing the revised event frequency estimates are included as figures H1 and H2, respectively.

The small difference in the values for "HELB occurs in the high-energy equipment in the vicinity of the switchgear room door" between figures H1 and H2 are due to rounding inaccuracies (i.e., the value was recalculated for H2).

In paragraph B1 on Page 6, the estimate of the frequency of HELB events is given as 1.0×10^{-3}. The correct value appears to be $12.5/1200 = 1.04 \times 10^{-2}$. The 1.0×10^{-3} value was not carried forward in the CDF calculation, leading to the conclusion that this is a typographical error.

The above comments are outside the licensee review guidance categories provided in Reference 1, Attachment 5.

Comments to HELB Sequence 2 Scenario B

Sequence 2 also results from a HELB in high-energy equipment in the vicinity of the switchgear room door. The following comments apply to Pages 5-10 in the discussion of Issue #122 in References 1 and the draft NUREG.

An event frequency estimate of 0.9 was used for the event sequence "MDAFP cross-tie from the unaffected unit success." This appears to be the frequency for the success path for the event. However, the other events in the sequence are assigned a frequency based on their failure path. I&M believes that, for consistency, the failure path for "MDAFP cross-tie from the unaffected unit fails" should be specified. For the failure path, this event frequency would be the compliment of 0.9, which is 0.1. This event can be difficult to understand, and an event tree, instead of a list of events, would have clarified the analysis. Figure H3 is an event tree based on the NRC analysis as I&M understands the sequences. The sensitivity study resulted in the CDF changing from 2.35×10^{-4} to 2.93×10^{-5}. This revised event tree is included as Figure H4.

With respect to the review criteria provided in Reference 1, Attachment 5, the above comments are categorized as modeling assumptions that do not accurately describe the modeling of the event appropriate for the events that occurred or that had the potential to occur under the event conditions.

Comments to HELB Sequence 3 Scenario C

Sequence 3 results from a HELB occurring in locations other than near the switchgear room doors. The following comments apply to Pages 10-12 in the discussion of Issue #122 in References 1 and the draft NUREG.

The first sequence of the event tree was modified by using a Bayesian methodology similar to the methodology described in Chapter 5 of Reference 5. The result of using this Bayesian methodology was a slight decrease in the event frequency. This change combined with the sensitivity study resulted in the CDF changing from 3×10^{-5} to 1.39×10^{-6}. The event tree based on the NRC analysis and the event tree containing the revised event frequency estimates are included as figures H5 and H6, respectively.

With respect to the review criteria provided in Reference 1, Attachment 5, the above comments are categorized as modeling assumptions that do not accurately describe the modeling of the event appropriate for the events that occurred or that had the potential to occur under the event conditions.

Comments to HELB Sequence 4 Scenario D

Sequence 4 results from a HELB that occurs in the vicinity of the EDG exhaust ducts in the Turbine Building. The following comments apply to Page 12-14 in the discussion of Issue #122 in References 1 and the draft NUREG.

The postulated event involved the EDG rooms becoming harsh environments due to backflow from the Turbine Building via the EDG ventilation exhaust ducts. The EDG ventilation system normally takes air from outside and discharges it into the Turbine Building. However, further analysis has determined that the Turbine Building HELB would not adversely impact operation of the EDG room exhaust system and the room would not be subjected to a harsh environment. Therefore, I&M requests that this scenario be withdrawn from the ASP and the draft NUREG.

The above comments are outside the licensee review guidance categories provided in Reference 1, Attachment 5.

Comments to HELB Sequence 5 Scenario E

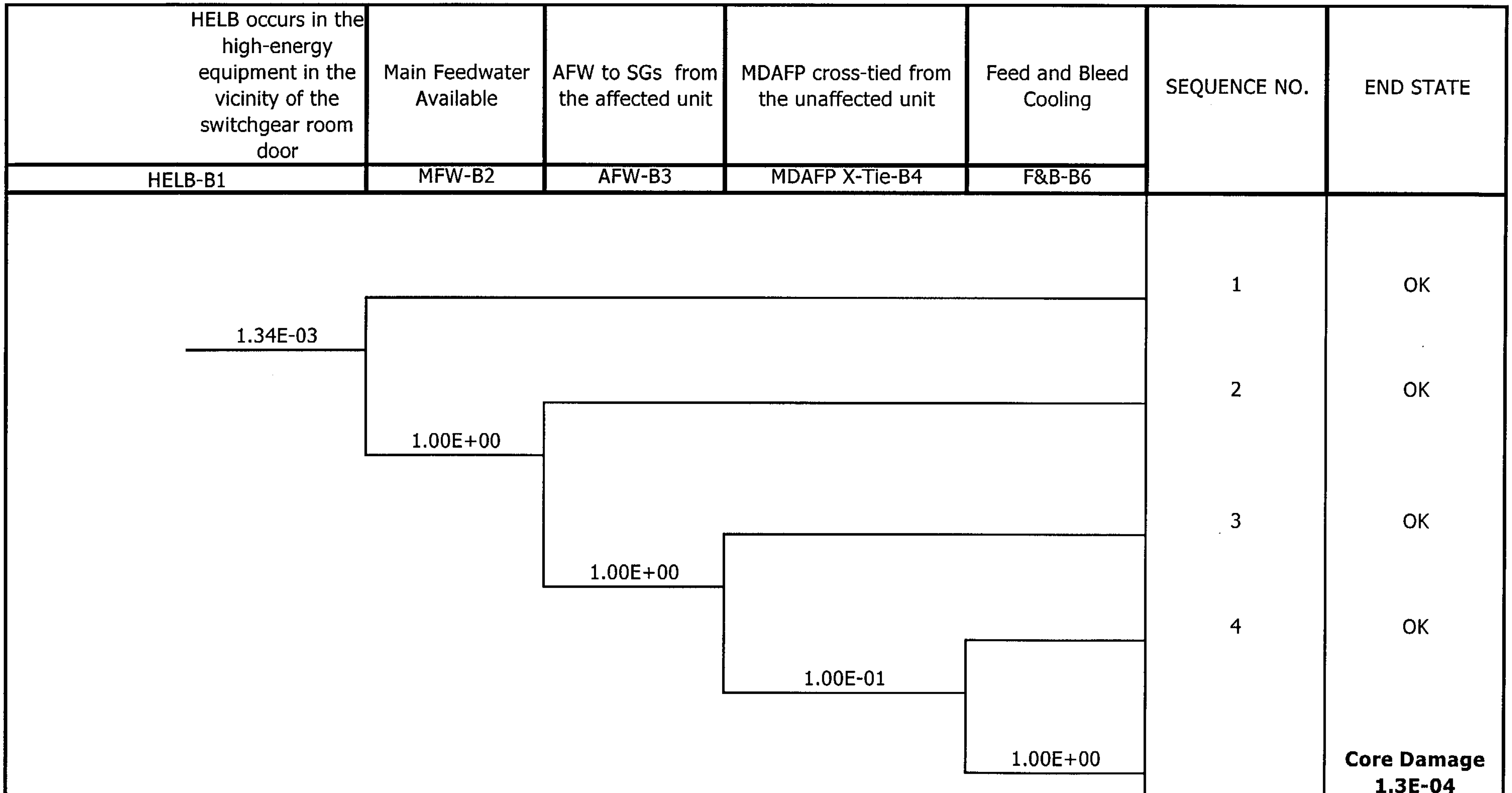
Sequence 5 describes a HELB that occurs in or near the steam generator blow-down flash tank. The following comments apply to Pages 14-15 in the discussion of Issue #122 in References 1 and the draft NUREG.

This sequence did not exceed the threshold; however, a potential error that could result in an order of magnitude change to the result is noted below.

The frequency of event E1 (HELB occurs in or near steam generator blowdown flash tank room) is estimated as 1.6×10^{-2}/year, and the frequency is identified on Page 14 as that which is used in Sequence 2. However, this event does not appear in Sequence 2. Also, in the calculation of the frequency of Sequence 5 on page 15, it appears that the value of $1.7 \times 10^{-2} \times 0.79 = 1.34 \times 10^{-2}$ is used. However, the Sequence 2 frequency of HELB is given on Page 6 as $(12.5/1200/6) \times 0.79 = 1.37 \times 10^{-3}$, which is a change in order of magnitude. If the actual Sequence 2 value is used to replace the value of 1.34×10^{-2}, the CDF changes from 1.29×10^{-8} to 1.31×10^{-9}.

In the initiating event frequency, Paragraph E.1, Page 14 of Issue #122, the value of the frequency used in Sequence 2 (1.6×10^{-2}) could not be validated. Also, a different value (1.7×10^{-2}) for this frequency was used in the calculation on the following page.

With respect to the review criteria provided in Reference 1, Attachment 5, the above comments are categorized as modeling assumptions that do not accurately describe the modeling of the event appropriate for the events that occurred or that had the potential to occur under the event conditions.



Original NRC ASP Analysis

Figure H1 - Event Tree of NRC Analysis of HELB Sequence 1 (HELB Scenario B)

HELB occurs in the high-energy equipment in the vicinity of the switchgear room door	Main Feedwater Available	AFW to SGs from the affected unit	MDAFP cross-tied from the unaffected unit	Feed and Bleed Cooling	SEQUENCE NO.	END STATE
HELB-B1	MFW-B2	AFW-B3	MDAFP X-Tie-B4	F&B-B6		

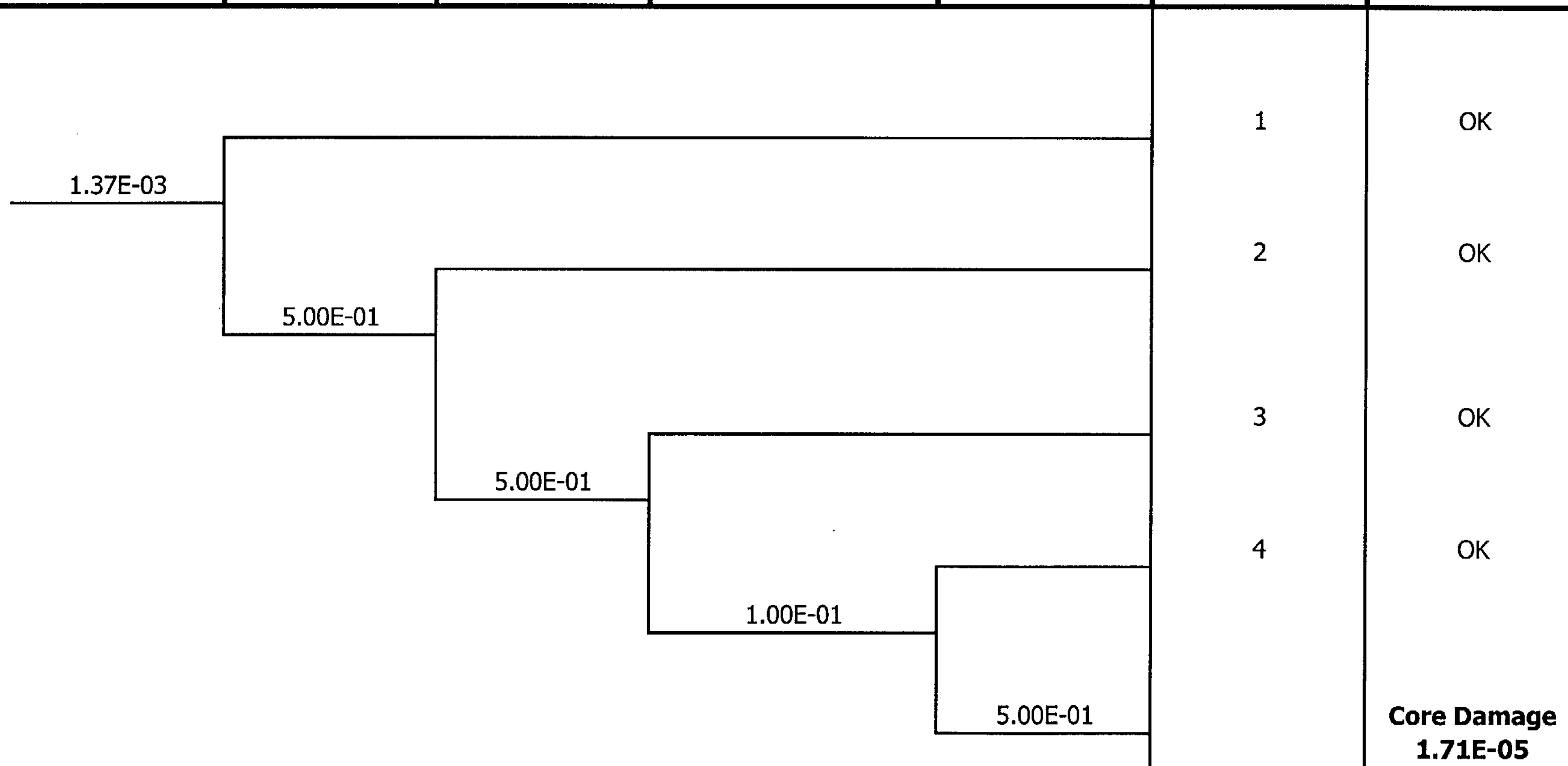
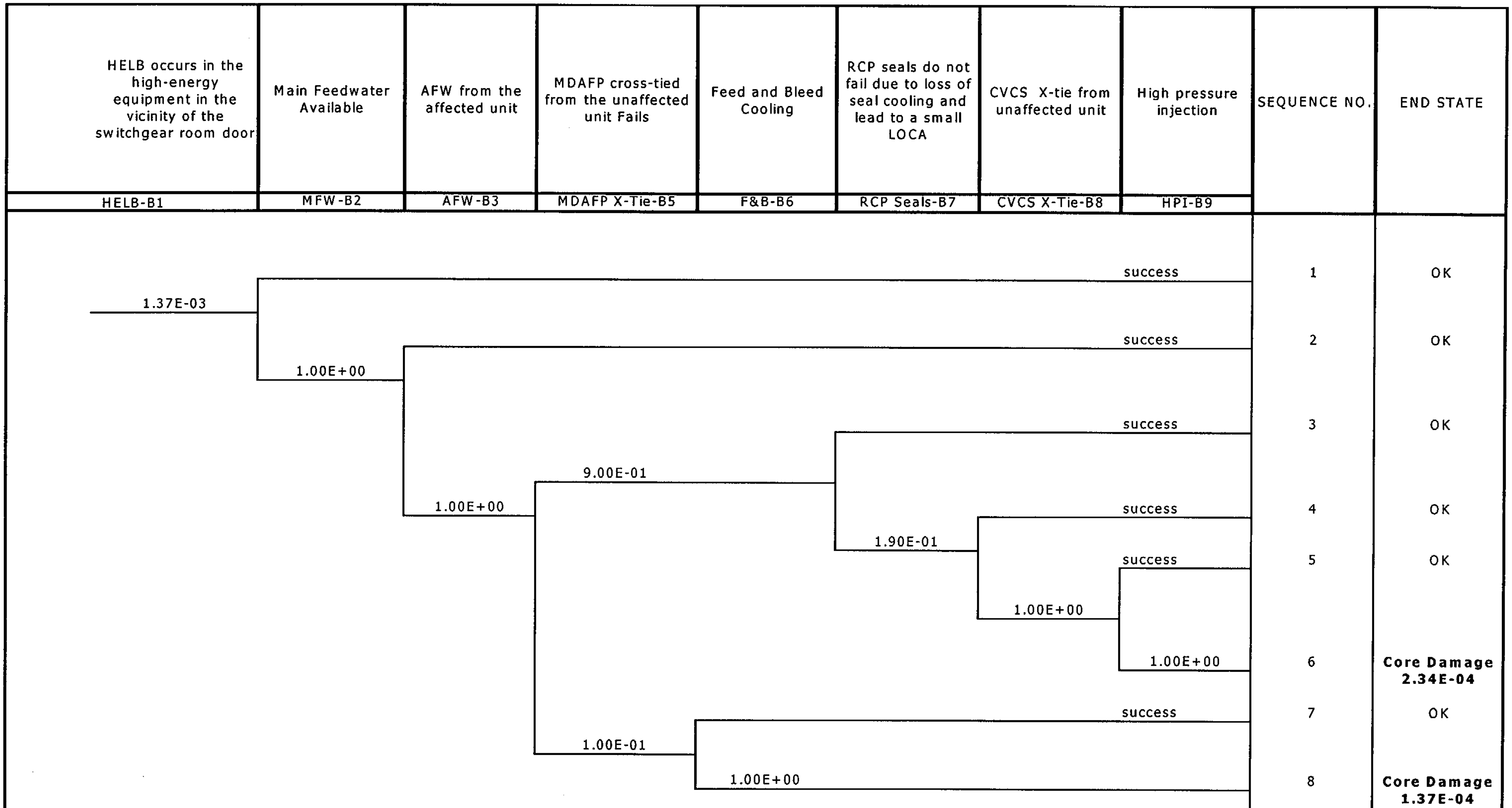


Figure H2 - Event Tree of Revised Analysis of HELB Sequence 1 Scenario B



Original NRC ASP Analysis

Figure H3 - Event Tree of NRC Analysis of HELB Sequence 2 (HELB Scenario B)

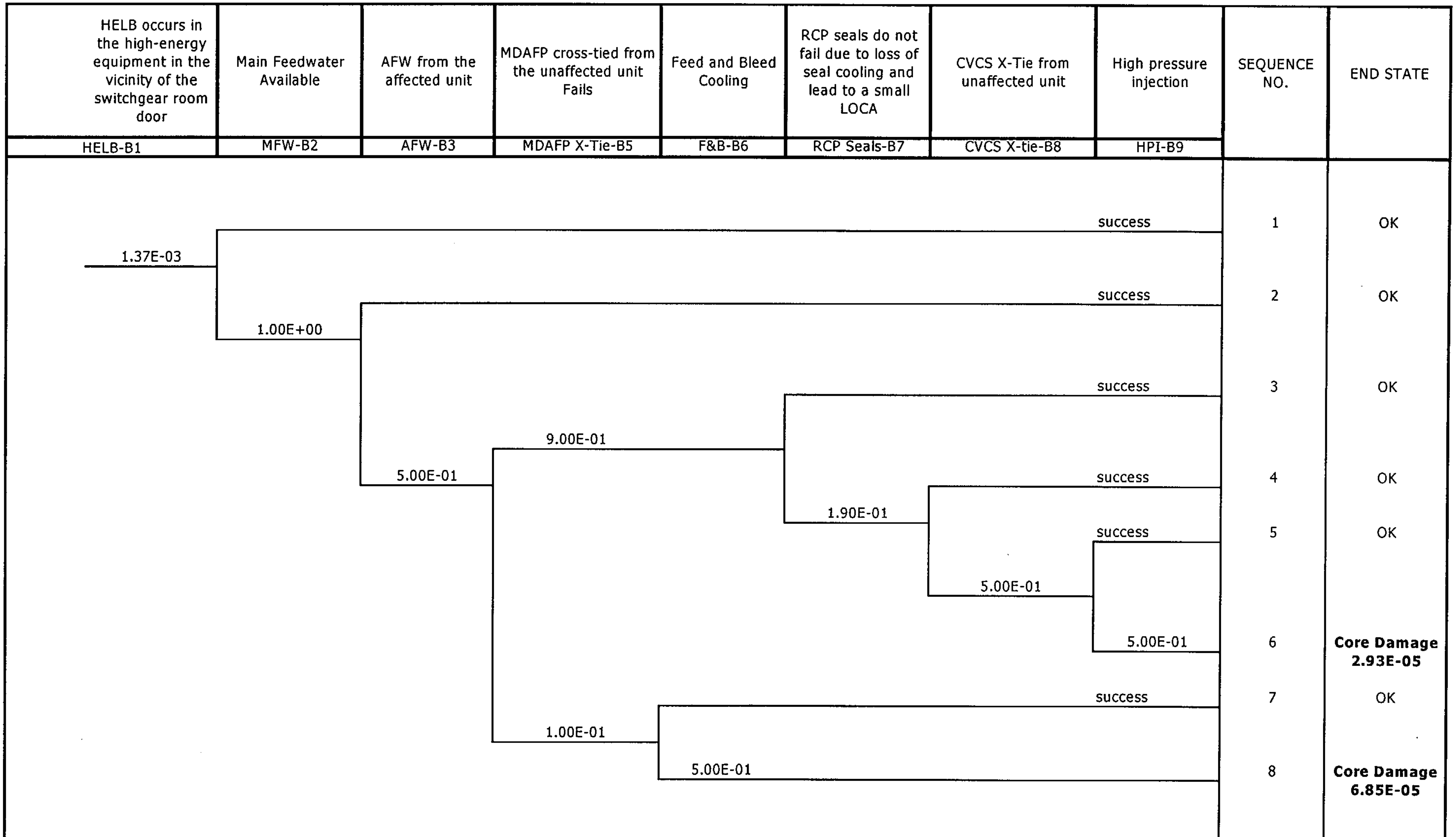
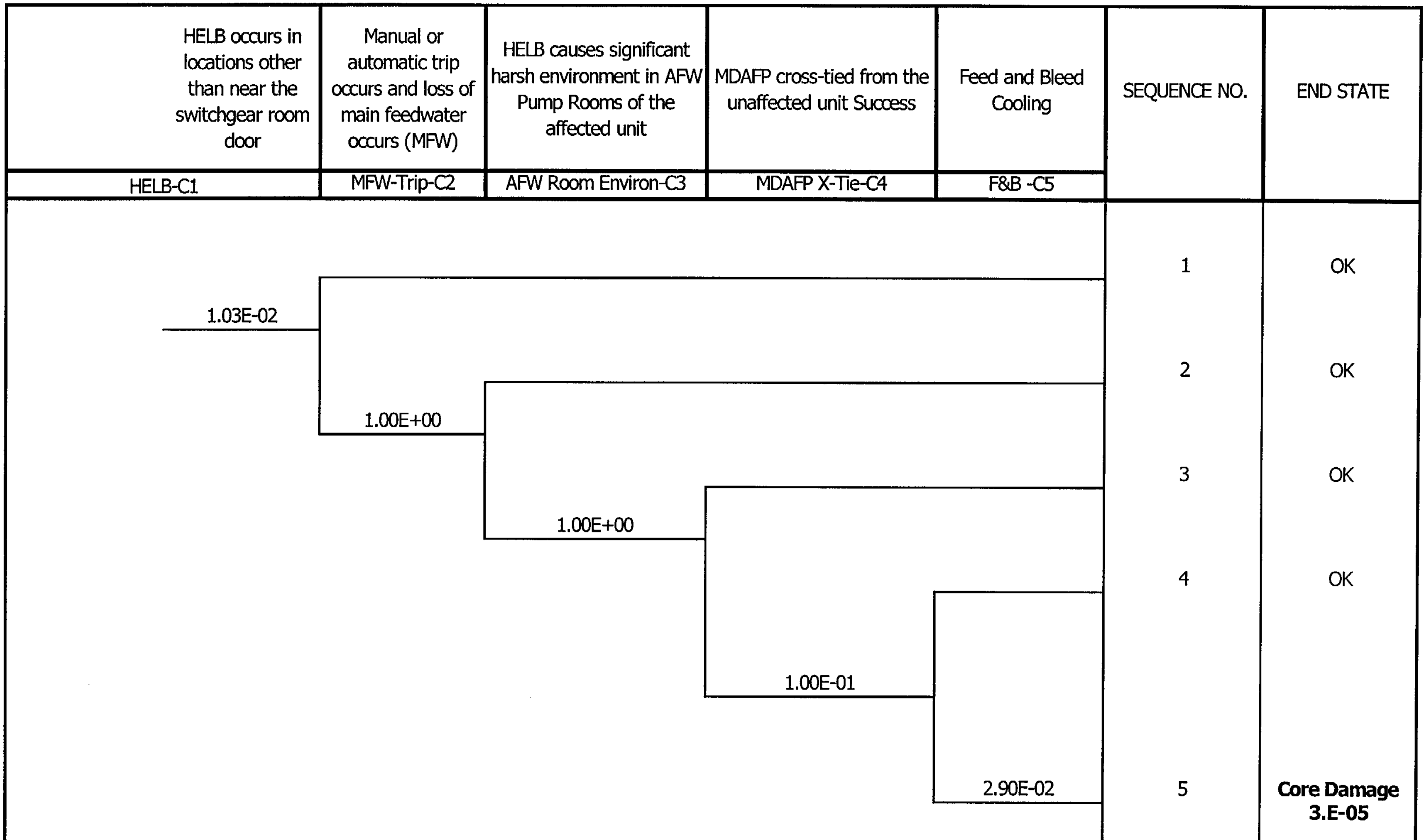


Figure H4 - Event Tree of Revised Analysis of HELB Sequence 2 Scenario B



Original NRC ASP Analysis

Figure H5 - Event Tree of NRC Analysis of HELB Sequence 3 (HELB Scenario C)

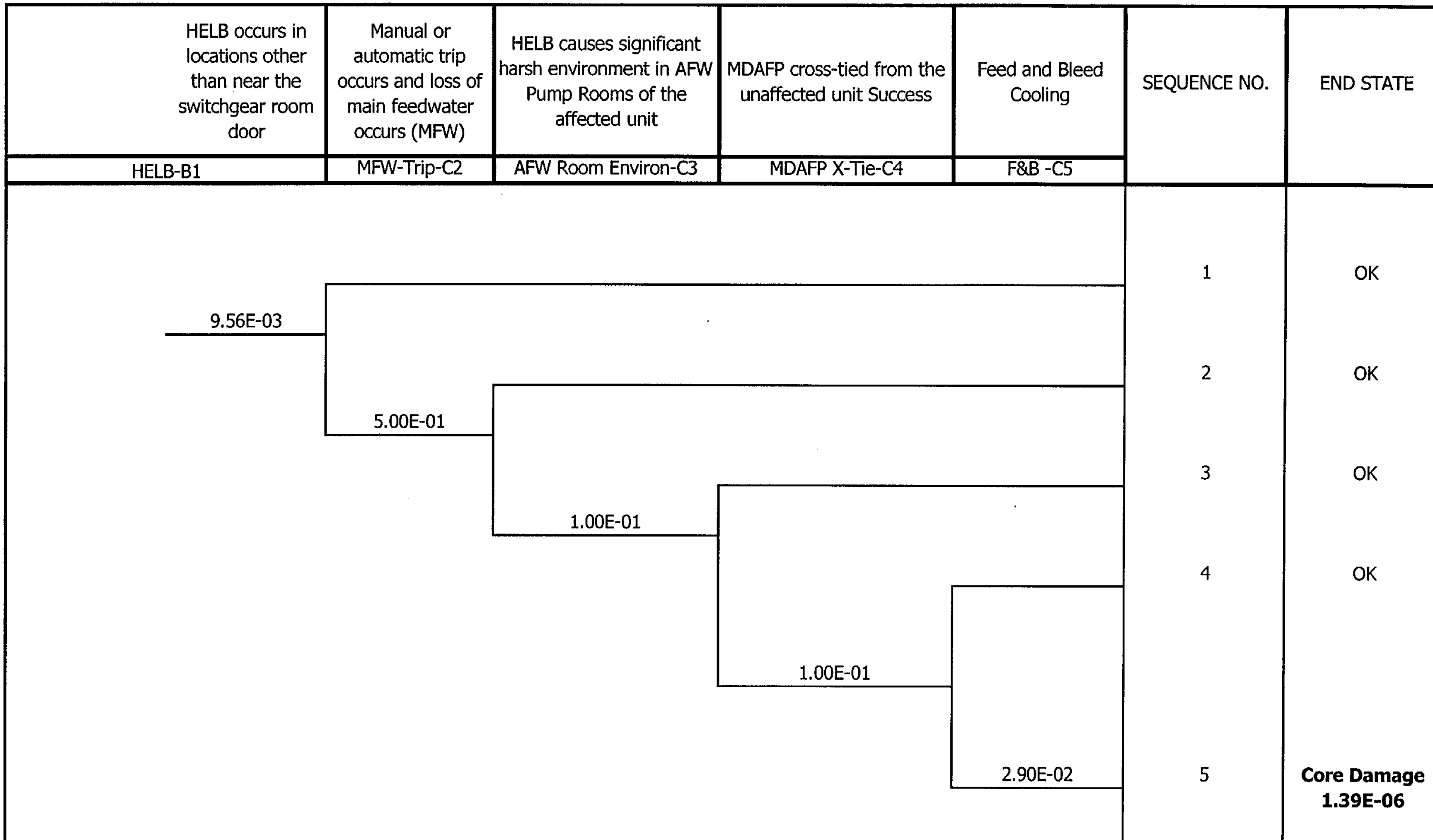


Figure H6 - Event Tree of Revised Analysis of HELB Sequence 3 Scenario C

Valves Required to Operate Post-Accident Could Fail to Open Due to Pressure Locking/Thermal Binding (Issue 135)

Background

In a 1999 LER, I&M reported that a preliminary calculation review determined that valves which provide a suction path from the containment sump to the emergency core cooling system pumps and the valves which align residual heat removal (RHR) to the upper containment spray header were susceptible to pressure locking following a postulated loss-of-coolant accident (LOCA).

Comments

The following comments apply to Sequence 5, "medium or large LOCA", on Pages 6-7 in the discussion of Issue #135 in References 1 and the draft NUREG.

In accordance with References 5 and 7, the frequency of a "medium or large pipe break" should be the Boolean sum and not the algebraic sum of the individual frequencies. Using the Boolean sum would change the estimate of occurrence for a "medium or large LOCA occurs" from 3.6×10^{-5} to 3.55×10^{-5}. Although the change is not significant, and of itself has no influence on the conclusion, use of a Boolean sum, consistent with References 5 and 7, would yield a significantly different result when applied to a large fault tree.

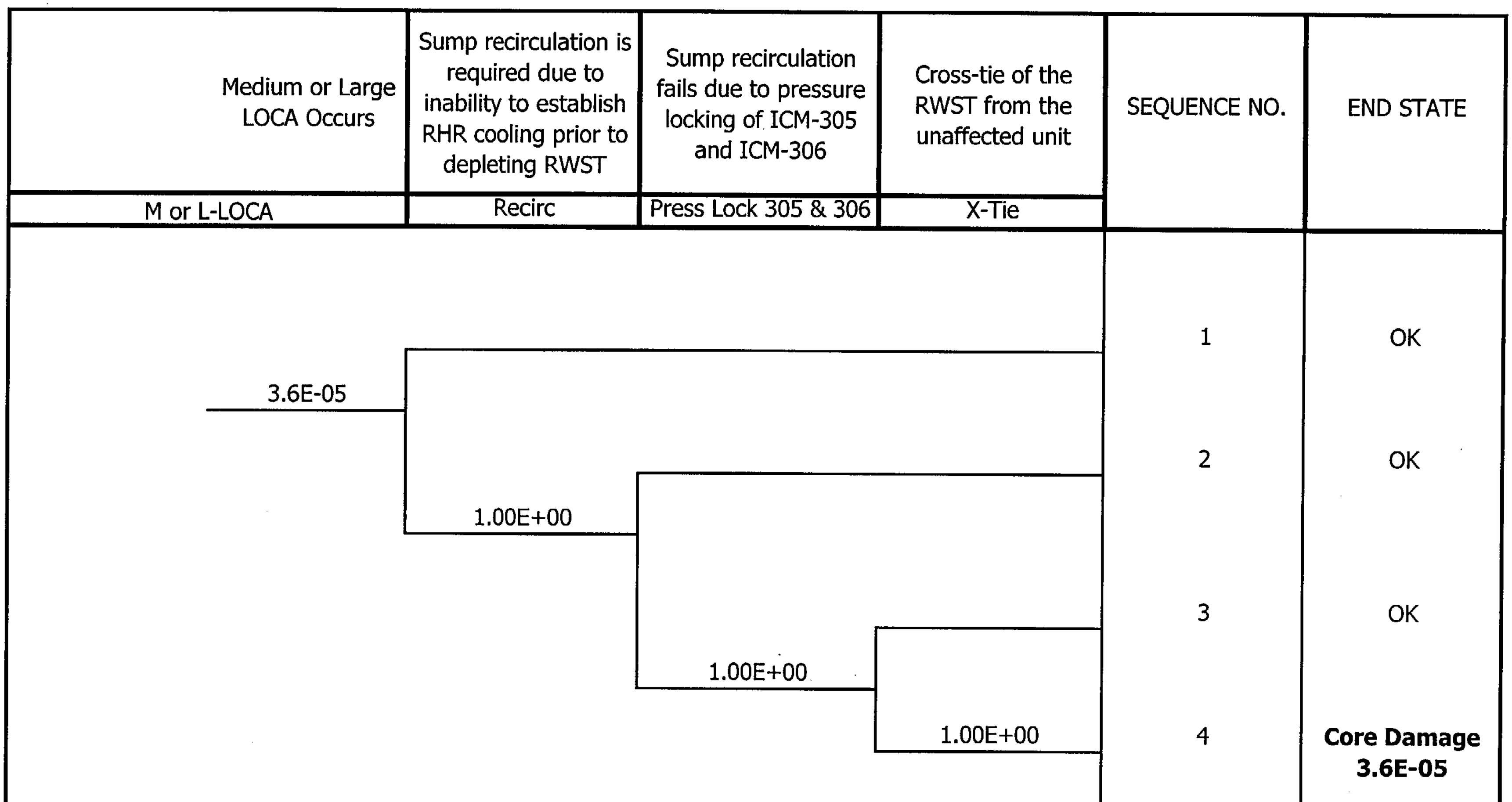
The assumption that sump recirculation fails with certainty (i.e., a frequency of 1.0) is overly conservative for the following reasons. Although the design bases parameters suggest the potential for pressure locking of the containment sump to RHR suction valves, ICM-305 and ICM-306, does not necessarily imply that a similar conclusion would be reached using a best-estimate analysis. I&M's review of maintenance records indicates that these valves have not previously failed. The CNP IPE has estimated the failure on demand of these valves individually to be 1.33×10^{-3}. Therefore, a conservative estimate that would be appropriate for the Reference 1 and the draft NUREG analysis would be 1.33×10^{-3}. Because both valves are required to fail, a Boolean "and" operation would result in an estimate of 1.77×10^{-6}. Therefore, I&M considers 1.33×10^{-3} to be a very conservative event frequency estimate for this sequence.

A change of the valve event frequency from 1.0 to 1.33×10^{-3} would change the CDF from 3.6×10^{-5} to 4.73×10^{-8}. Figure PL1 is an event tree representing the original analysis, while Figure PL2 is an event tree depicting this adjustment.

Similarly, the estimate that the crosstie of the refueling water storage tank from the affected unit to the unaffected unit fails with certainty (i.e., a frequency of 1.0) cannot be justified. However, this value was not changed in Figure PL 2 because the revised CDF is already below the threshold.

Again, a small change in the existing analysis, results in a significant change in the outcome, and indicates the inappropriate sensitivity of the draft NUREG analysis.

With respect to the review criteria provided in Reference 1, Attachment 5, the above comments are categorized as modeling assumptions that do not accurately describe the modeling of the event appropriate for the events that occurred or that had the potential to occur under the event conditions.



Original NRC ASP Analysis

Figure PL1 - Original ASP Analysis of Pressure Locking/Thermal Binding ASP Sequence 5 - Medium or Large LOCA

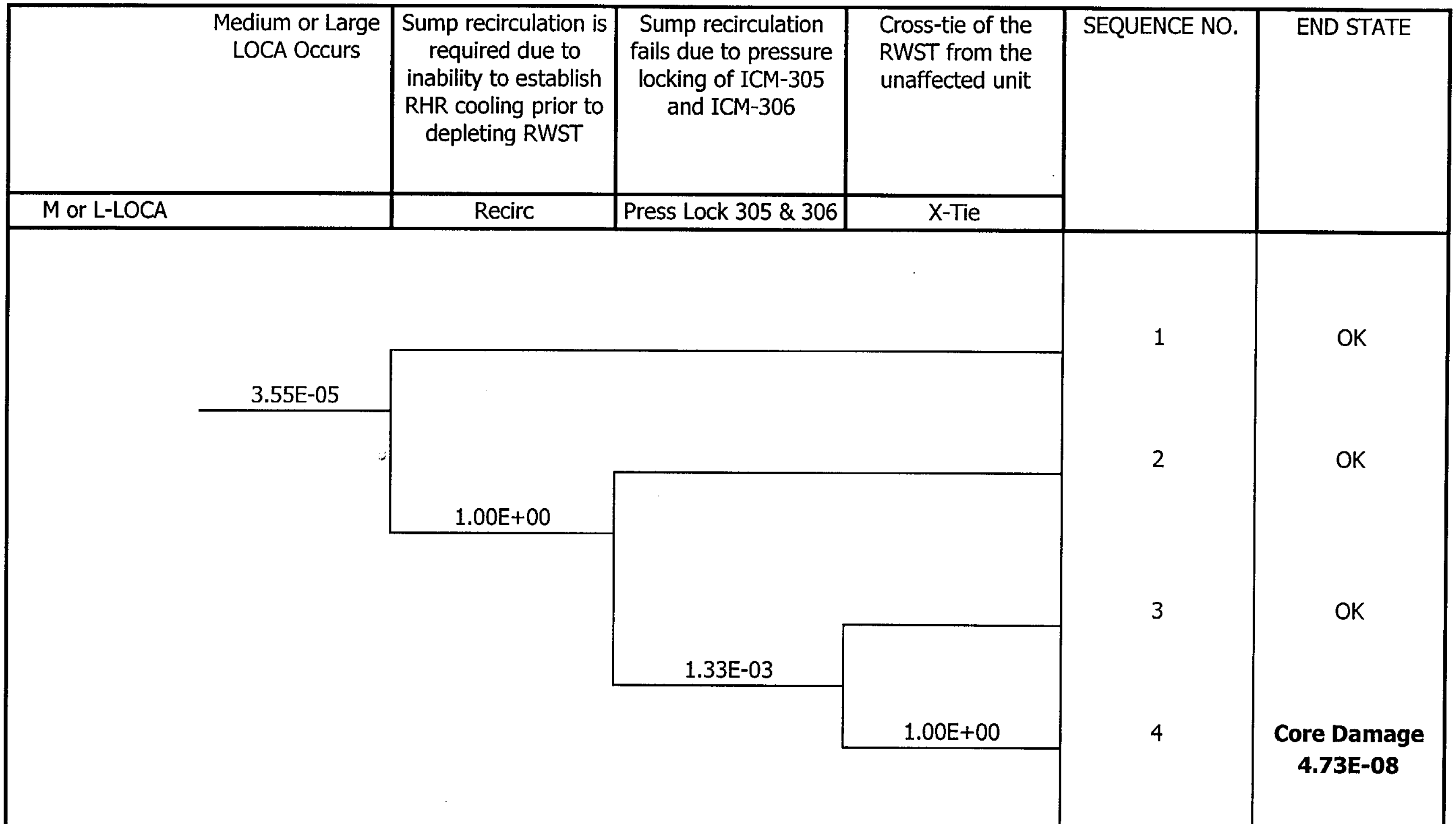


Figure PL2 - Revised Analysis of Pressure Locking/Thermal Binding ASP Sequence 5 - Medium or Large LOCA

Potential Seismic Deficiencies (Block Walls) (Issue 134)

Background

CNP Condition Report (CR) 99-14004 documents that reasonable assurance could not be provided that concrete block wall installations conform to the CNP design and licensing basis due to a lack of controlled configuration documentation. As a result, there were concerns that Seismic Class I components in proximity to these walls could be adversely affected. Approximately 123 such walls were identified. These walls were analyzed for applicable loading, including seismic loading, and the Unit 2 walls that did not meet the design basis acceptance criteria have been reinforced. There are 20 Unit 1 walls that were found not to meet design basis acceptance criteria. Design changes are in progress for these walls.

Comments

On page 1 of the discussion of Issue 134 in References 1 and the draft NUREG, the NRC stated that the CDF would be re-assessed if additional information was identified regarding the as-found strength of the walls or which components would fail if the walls were to fail. Accordingly, additional information is provided for reconsideration of the risk significance associated with the block walls. The supporting documentation is available for inspection at CNP.

I&M has determined that, with one exception, all of the Unit 2 walls have been operable in the past and would not have collapsed in the event of a design basis earthquake (DBE). The one exception was wall 4049-W2, which is an eight-inch hollow block wall separating the Unit 1 and 2 Turbine Buildings between elevations 609' and 633'. I&M has concluded that this wall could have potentially collapsed during a DBE and impacted the turbine driven auxiliary feed pump exhaust piping which is routed close to the wall. However, I&M has determined that the exhaust piping and turbine driven auxiliary feed pump would have remained operable. These determinations have been documented in CR 99-14004. The evaluation of the 20 Unit 1 walls for past operability is in progress.

With respect to the review criteria provided in Reference 1, Attachment 5, the above comments are categorized as event descriptions that do provide accurate additional information concerning the configuration of the plant.

References

1. Letter from John A. Grobe, Nuclear Regulatory Commission to R. P. Powers, Indiana Michigan Power Company, "Review of Four Preliminary Accident Sequence Precursor Analyses of Operational Events at D.C. Cook Nuclear Plant," dated May 9, 2000.

2. Letter from John A. Grobe, Nuclear Regulatory Commission to R. P. Powers, Indiana Michigan Power Company, "Donald C. Cook Plant, Units 1 and 2 – Issuance of Draft Report Entitled 'Assessments of Risk Significance Associated with Issues Identified at D.C. Cook Nuclear Plant'," dated May 17, 2000.

3. Letter from J. E. Dyer, Nuclear Regulatory Commission to R. P. Powers, Indiana Michigan Power Company, "Closure of NRC Inspection Manual Chapter 0350 Restart Action Plan for Restart of the Donald C. Cook Nuclear Plant - Unit 2," dated June 13, 2000.

4. "Precursors to Potential Severe Core Damage Accidents: 1997, A Status Report," NUREG CR-4674, ORNL/NOAC-232, Vol. 26, Prepared for Safety programs Division, Office for Analysis and Evaluation of Operational Data, US Nuclear Regulatory Commission, Washington, DC 20555-0001, NRC Job Code B0435, November 1998.

5. "PRA Procedures Guide, A Guide to the Performance of Probabilistic Risk Assessments for Nuclear Power Plants," NUREG/CR-2300, Volumes 1 and 2, Office of Nuclear Regulatory Research, US Nuclear Regulatory Commission, Washington, D.C. 20555, January 1983.

6. Letter from M. W. Rencheck, Indiana Michigan Power Company, to U. S. Nuclear Regulatory Commission, "Review of Preliminary Accident Sequence Precursor Analysis of Operational Condition," dated November 8, 1999.

7. "Fault Tree Handbook," NUREG-0492, Office of Nuclear Regulatory Research, U. S. Nuclear Regulatory commission, Washington, D. C. 20555, January 1981.