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

Subject: Transmittal of Davis-Besse Nuclear Power Station Risk Assessment of Control Rod Drive Mechanism Nozzle Cracks

Ladies and Gentlemen:

As discussed at an October 24, 2001, public meeting between the Nuclear Regulatory Commission (NRC) and the Davis-Besse Nuclear Power Station (DBNPS) staffs, attached is the DBNPS Risk Assessment of Control Rod Drive Mechanism (CRDM) Nozzle Cracks as supplemental information to the DBNPS response to NRC Bulletin 2001-01, "Circumferential Cracking of Reactor Pressure Vessel Head Penetration Nozzles." This plant-specific assessment, as was discussed during the aforementioned meeting, was developed based on the Framatome ANP document 51-5012567-01, "RV Head Nozzle and Weld Safety Assessment." The Framatome ANP document was provided to the NRC staff by the FirstEnergy Nuclear Operating Company (FENOC) letter Serial Number 2735, dated October 17, 2001. The DBNPS assessment incorporates several refinements from the Framatome ANP document that make the analysis specific to the DBNPS.

The results of the Risk Assessment indicate that, using bounding analysis, the core damage frequency risk from CRDM nozzle cracks can be categorized as small using the guidelines in Regulatory Guide (RG) 1.174, "An Approach for Using Probabilistic Risk Assessment in Risk Informed Decisions on Plant Specific Changes to the Licensing Basis," (July 1998). The large early release frequency can be considered very small using the guidelines in Regulatory Guide 1.174 and the person-rem per year contribution from this event can be considered negligible.

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If you have any question or require further information, please contact Mr. David H. Lockwood, Manager-Regulatory Affairs, at (419) 321-8450.

Very truly yours,

David H. Lockwood
FOR Guy G. Campbell

Enclosure
Attachments

cc: J. E. Dyer, Regional Administrator, NRC Region III
S. P. Sands, DB-1 NRC/NRR Project Manager
D. Simpkins, DB-1 Acting Senior Resident Inspector
Utility Radiological Safety Board

SUPPLEMENTAL INFORMATION

IN RESPONSE TO

NRC BULLETIN 2001-01

FOR

DAVIS-BESSE NUCLEAR POWER STATION

UNIT NUMBER 1

This letter is submitted pursuant to 10 CFR 50.54(f) and contains supplemental information concerning the response (Serial 2371, dated September 4, 2001) to NRC Bulletin 2001-01, "Circumferential Cracking of Reactor Pressure Vessel Head Penetration Nozzles," for the Davis-Besse Nuclear Power Station, Unit Number 1.

I, Guy G. Campbell, state that (1) I am Vice President - Nuclear of the FirstEnergy Nuclear Operating Company, (2) I am duly authorized to execute and file this certification on behalf of the Toledo Edison Company and The Cleveland Electric Illuminating Company, and (3) the statements set forth herein are true and correct to the best of my knowledge, information and belief.

For: G. G. Campbell, Vice President - Nuclear

By: David H. Lockwood
D. H. Lockwood, Manager – Regulatory Affairs

Affirmed and subscribed before me

Lori J. Strauss
Notary Public, State of Ohio

LORI J. STRAUSS
Notary Public, State of Ohio
My Commission Expires 3/24/2003

Risk Assessment of CRDM Nozzle Cracks

1. Summary

The Davis-Besse Nuclear Power Station calculation C-NSA-99.16-46 evaluates the risk significance, during the current 13th operating cycle, of possible undetected control rod drive mechanism (CRDM) nozzle cracks that could lead to loss of coolant accident (LOCA) events. The approach used in this calculation is based on the method documented in Framatome document 51-5012567-01, "RV Head Nozzle and Weld Safety Assessment" (Reference 2). However, this calculation incorporates several refinements that make the analysis specific to Davis-Besse. The results of this analysis indicate that, using bounding analysis, the core damage frequency risk from CRDM nozzle cracks can be categorized as small using the guidelines in 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) for core damage frequency. The large early release frequency can be considered very small using the guidelines in Regulatory Guide 1.174 and the person-rem per year contribution from this event can be considered negligible.

2. Method

The approach used in the Davis-Besse plant specific calculation is similar to the method documented in Framatome document 51-5012567-01, "RV Head Nozzle and Weld Safety Assessment" (Reference 2). However, the Framatome analysis was developed generically for a B&W plant. Davis-Besse has several plant specific refinements that were incorporated in the analysis. Additionally, the initiating event frequency was modified to account for more recent inspection results from the Crystal River 3 (CR-3) plant.

3. Assumptions

The following are assumptions made in the Davis-Besse plant specific risk analysis of CRDM nozzle cracks. Other assumptions and sources of conservatism are discussed in reference 2.

- Probability of Leakage Detection – It was assumed that leakage could not be detected from any CRDM nozzles where boron deposits from flange leaks may have obscured indications of CRDM nozzle leakage. Visual inspections of the reactor vessel head have been performed during 10, 11, and 12 refueling outages (RFO). However, the number of CRDM nozzles that could be viewed for indications of nozzle leakage has depended on the extent of boron deposits from flange leaks that may have obscured indications of CRDM nozzle leakage. During 10RFO, in spring of 1996, the entire head was visible so 100% of the CRDM nozzles were inspected with the exception of four nozzles in the center of the head. The CRDM nozzles in the center of the head are not expected to show leakage as evaluated in reference 8. During subsequent outages in 1998 and 2000 a number of nozzles could not be inspected due to boron leakage. Based on review of the 1998 and 2000 videotapes it was determined that 45 CRDMs were inspected in 2000 (12 RFO) and 50 in 1998 (11 RFO).

- Consequence of CRDM Nozzle Failure - The failure of a CRDM nozzle is assumed, in this analysis, to result in the maximum effective break possible, a medium LOCA with a break area of 0.1 ft^2. In reality, it is considered to be much more likely that increased leakage would be detected before it developed into a LOCA. If a LOCA were to develop, it is likely that it would be a small LOCA due to less than complete failure of the CRDM nozzle. The conditional probability of core damage is higher for medium LOCAs than for small LOCAs and is much higher than for a situation in which the plant is shut down due to RCS leakage. Therefore, this assumption represents a very significant source of conservatism in this analysis.

- Crack Initiation - For this analysis it is assumed that the postulated CRDM nozzle leaks were initiated in 10^{th}, 11^{th}, 12^{th}, or 13^{th} operating cycles, with up to three opportunities to detect the leaks during 10 RFO, 11 RFO and 12 RFO. Therefore, this analysis includes cracks that could have initiated as early as November 1994 (after completion of 9 RFO). Although it is possible that leaks could have existed prior to this time, the risk contribution from cracks existing prior to 9 RFO is not expected to be a significant contribution to current risk due to the opportunities for detection and limited time at operating temperature. ?

- Crack Initiation Frequency - For the calculation of leak frequency, it is assumed that any CRDM nozzle leaks detected at Babcock & Wilcox (B&W) plants were initiated in the most recent two operating cycles. However, the Davis-Besse analysis applies this initiating event frequency over a time encompassing four operating cycles which is a very conservative approach and effectively "double counts" the observed leak occurrence rate. This very conservative assumption was applied to ensure the possibility of undetected leaks initiated in earlier operating cycles was considered.

4.0 Calculation

Risk from CRDM Nozzle Cracking

Reference 2 reviewed the risks associated with the possibility of an undiscovered CRDM nozzle crack. This report concluded that the risk from a LOCA caused by a CRDM nozzle failure during operation should be considered. Reactivity accidents were discounted as credible risk contributors because of the number of simultaneous CRDM nozzle failures that would be required. Missiles generated by CRDM nozzle failures are also not credible because even in the event of CRDM nozzle detachment the missile shields will prevent damage to safety systems. [redacted]

Probability of a Weld or Nozzle Leak

* Section 9.3.1 of reference 2 calculated an average frequency of 1.25 leaking CRDM nozzles per reactor year for the last two cycles. This was based on a total of 15 leaking nozzles identified at Oconee Nuclear Station (ONS)-1, ONS-2, ONS-3 and Arkansas Nuclear One (ANO)-1. One additional leaking nozzle was subsequently detected at CR-3. Therefore, this additional data was added to the calculation of the leak frequency. [redacted]

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For the estimation of leak frequency it is assumed that all 16 leaks appeared during the most recent two fuel cycles. This corresponds to 15 plant years of operation (conservatively assuming 18 months per fuel cycle - 1.5 yr./cycle x 5 plants x 2 cycles) The resulting average frequency is approximately 1.1 CRDM leaks per reactor year (16 leaks/15 reactor years).

[redacted] This generic B&W nozzle leak frequency does not include Davis-Besse plant-specific nozzle leakage experience. Plant specific experience can be included by calculating a generic distribution using the B&W data and updating with the Davis-Besse experience using Bayesian Updating techniques. To perform this calculation the generic distribution was calculated on a per nozzle basis as shown in Table 1 and Table 2.

Table 1 - Assessment of Generic Leak Frequency for B&W Plants

Unit	Number of Leaks	Leak Frequency
ANO-1	1	4.83×10^{-3}
CR-3	1	4.83×10^{-3}
ONS-1	1	4.83×10^{-3}
ONS-2	4	1.93×10^{-2}
ONS-3	9	4.35×10^{-2}
Total	16	1.55×10^{-2}

Table 2 – Parameters of the Generic Lognormal Distribution

Parameter of the Lognormal Distribution	Parameter Value
Mean	1.55×10^{-2}
Variance	2.28×10^{-4}
Median	1.11×10^{-2}
Error Factor	3.84

For the Davis-Besse experience the plant years of operation are calculated for the 11th, 12th and 13th operating cycles on a per inspected nozzle basis as follows:

11^{th} Cycle – 18 months (1year / 12 months) 65 nozzles = 97.5 nozzle years

12^{th} Cycle - 24 months (1year / 12 months) 50 nozzles = 100 nozzle years

13^{th} Cycle - 24 months (1year / 12 months) 45 nozzles = 90 nozzle years

[redacted]

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Total inspected nozzle years = 287.5

The plant-specific operating time is used to calculate a lognormal distribution which is then used to update the generic prior distribution using Bayesian Updating techniques. The resulting updated failure rate is as follows:

Updated nozzle leak rate = 5.03×10^{-3} / nozzle-year

The nozzle leak rate updated to include Davis-Besse experience is lower than the generic rate by a factor of three. However, the generic rate will be used in this calculation providing further assurance of a conservative bounding calculation.

Probability of Leakage Detection

Visual inspections of the reactor vessel head have been performed during 10, 11, and 12 RFO. However, the number of CRDM nozzles that were able to be viewed for indications of nozzle leakage has depended on the extent of boron deposits from flange leaks that may have masked indications of CRDM leakage. During 10 RFO, in spring of 1996, the entire head was visible so 100% of the CRDM nozzles were inspected with the exception of four nozzles in the center of the head. The CRDM nozzles in the center of the head are not expected to show leakage as evaluated in reference 8. During subsequent outages in 1998 and 2000 a number of nozzles could not be inspected due to CRDM flange leakage. Based on review of the 1998 and 2000 videotapes it was determined that 45 CRDMs were inspected in 2000 and 50 in 1998. Additionally, the 19 CRDMs that were obscured in 1998 were also obscured in 2000.

Reference 2 used a human event probability representing both the probabilities that the inspection would fail to detect leakage and the probability that the inspection would not be performed. The failure rate for failure of a visual inspection was determined to be 0.05 based on the methods in reference 2. Therefore, for CRDMs that were available for inspection a failure rate of 0.05 will be applied.

Probability of Core Damage

The consequence of a CRDM nozzle failure would be RCS leakage or in the worst case a medium LOCA. Due to the uncertainty about the probability of various size leaks for this analysis the bounding case will be assumed and all failures will be considered to be medium LOCAs. Medium LOCAs as defined for the Davis-Besse Probabilistic Safety Assessment (PSA) cover a range of breaks from 0.02 to 0.5 ft^2. For the lower end of the range (about 0.02 ft^2 to 0.1 ft^2) the success criteria are substantially less restrictive and only high pressure injection (HPI) is needed to provide makeup to the reactor coolant system (RCS). The break size for a CRDM nozzle is about 0.1 ft^2 so this break is in the range of medium LOCAs with less restrictive success criteria. Therefore, for this analysis, the medium LOCA top logic was modified to provide a more representative core damage probability. The following specific changes were made:

- The logic for a medium LOCA was modified by removing the requirement for injection from a low pressure injection pump and a core flood tank during the injection phase of the LOCA. However, since the low pressure injection pumps are required during the recirculation phase and since the core flood tanks are very reliable, this logic change has a small impact on the total core damage frequency.

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- The probability for human failure to initiate low pressure recirculation after a medium LOCA was revised to account for the much longer time that would be available if the LOCA is initiated by a CRDM failure. The human action used in the base line PSA analysis is based on the most limiting case for a medium LOCA. However, for a LOCA resulting from failure of a CRDM nozzle, the time available before borated water storage tank (BWST) depletion is much longer, allowing a longer time window to prepare for and perform this critical step.

Revision 3 of the Davis-Besse PSA (reference 5) was modified as described above and solved for medium LOCA with failure of safety injection and medium LOCA with failure of low pressure recirculation sequences. The resulting core damage frequency for the medium LOCA sequences was 1.07×10^{-7} / year. Based on this core damage frequency and the medium LOCA initiating event frequency of 4.0×10^{-5} / year the resulting conditional core damage probability is (1.07×10^{-7} / year) / (4.0×10^{-5} / year) or 2.7×10^{-3}.

Core Damage Frequency (CDF) Contribution from CRDM Nozzle LOCAs

The core damage frequency was calculated using the method shown in figure 5 of reference 2. However, for the Davis-Besse calculation the core damage frequency contribution was calculated considering the specific inspection history for the CRDM nozzles at Davis-Besse. The CRDM nozzles were placed into three groups based on the extent of visual inspection possible during 10, 11 and 12 RFO.

4 In addition to the three groups, four nozzles in the center of the head do not have demonstrable annular gaps at the operating conditions. The CRDM nozzles without a demonstrable gap, as calculated in the SIA analysis, at Davis-Besse are the center nozzle (No. 1) and three nozzles adjacent to the center nozzle (Nos. 2, 3, and 4). These particular nozzles are not prone to the circumferential cracking observed during recent

inspections of other plants in the B&W fleet and consequently are not risk significant. Reference 4 provides a detailed discussion of the stress on these nozzles and a justification for considering these nozzles not to be risk significant with respect to circumferential cracking.

Table 3 - CRDM Leak Frequency by CRDM Nozzle Group

CRDM Nozzle Group	Description	Number of Nozzles	Leak Frequency (/year)
1	Common nozzles not inspected during 11 and 12 RFO	15	2.54×10^{-1}
2	Additional nozzles not inspected during 12 RFO	5	8.46×10^{-2}
3	Nozzles inspected during all Outages	45	7.62×10^{-1}

The leak frequency was calculated for each group as follows:

Group 1: Leak Frequency = [redacted] / year (15 / 65)

Group 2: Leak Frequency = 1.1 / year (5 / 65)

Group 3: Leak Frequency = 1.1 / year (45 / 65)

[redacted]

Figures 1 through 3 show the calculation of core damage frequency for each of the three CRDM groups. The results of this calculation are shown in Table 4.

Table 4 – CDF Contribution by CRDM Nozzle Group

CRDM Nozzle Group	Core Damage Frequency (/year)
1	6.73×10^{-6}
2	1.47×10^{-7}
3	9.18×10^{-8}
Total	6.97×10^{-6}

This evaluation encompasses the period from the start of the 10th operating cycle, in November 1994, up to the present. Although it is possible that leaks could have existed prior to this time, the risk contribution from cracks existing prior to 9RFO is not significant due to the opportunities for detection and limited time at operating temperature. Additionally, the leak initiation frequency is based on only two operating cycles.

Large Early Release Frequency Contribution from CRDM Nozzle LOCAs

For the assessment of LERF it is conservatively assumed that all nozzle failures result in a medium LOCA. The conditional large early release probability (CLERP) for a medium LOCA is about 4.0×10^{-6} based on the LERF contribution from medium LOCAs documented in reference 6 and the medium LOCA frequency of 4.0×10^{-5} / year. Based on figures 1 through 3 the total nozzle failure frequency is 2.58×10^{-3} / year. Using this frequency and the medium LOCA conditional release probability the bounding LERF contribution from nozzle failures is 1.03×10^{-8} / year.

The conditional large early release probability for a medium LOCA is very small resulting in a LERF contribution from nozzle failures that is much less significant than the core damage frequency contribution. However, this small LERF contribution is expected for a medium LOCA due to unique plant features. Davis-Besse has a large dry containment with a large containment to core power ratio. Consequently, the most significant contributors to release from containment for Davis-Besse are containment bypass events such as steam generator tube ruptures or interfacing system LOCAs. Other significant contributors to LERF include sequences where core damage results from loss of support systems that also support containment cooling. These sequences include loss of cooling water systems or station blackout. Medium LOCAs are not representative of either of these dominant LERF sequences. Additionally, LOCAs depressurize the reactor coolant system prior to core damage, which reduces the possibility of high pressure melt ejection.

Public Health Risk

The public health risk associated with the CRDM nozzle cracking can be estimated by applying the Davis-Besse Level 3 PSA analysis (reference 7). To perform this analysis the Level 2 PSA (reference 6) was quantified for the medium LOCA sequences using a revised initiating event frequency. The initiating event frequency of 2.58×10^{-3} / year was used based on the results in figures 1 through 3.

The results of the Level 2 Quantification were input into the Level 3 Risk Calculation Spreadsheet. Inputs for all other initiating events were set to zero. Changes to other inputs to the Risk Calculation Spreadsheet including the Release Category Matrix and MACCS2 output file were not required. The results of the Level 3 analysis are shown in Table 5.

Uncertainty and Sensitivity Analysis

The analysis used conservative assumptions and bounding data where possible to address uncertainty. The Framatome analysis used bounding assumptions for crack initiation time, initial flaw distribution and multiple crack initiation sites. The Davis-Besse specific analysis used bounding estimates for leakage detection probability, the consequences of failure and leak initiation frequency.

5.0 Conclusions

Core Damage Frequency

The Davis-Besse plant specific core damage frequency contribution from LOCAs initiated by CRDM nozzle cracking was estimated to be 6.97×10^{-6} / year using conservative and bounding assumptions. The conservative assumptions used in the Framatome and EPRI analysis include bounding assumptions for crack initiation time, initial flaw distribution, multiple crack initiation sites and probable sizes of the subsequent RCS leak. The Davis-Besse specific analysis includes the following significant conservative or bounding assumptions:

- No credit for leakage detection was given for CRDM nozzles where boron deposits obscured the head to nozzle annulus area. Due to this assumption the risk is dominated by the CRDM nozzles that could not be inspected. This reduces the sensitivity to the failure rate of the visual inspection that is applied to the CRDM nozzles that were not obscured.

- All failures are assumed to result in the maximum 0.1 ft^2 medium break LOCA. In reality, it is considered to be much more likely that increased leakage would be detected before it developed into a LOCA. If a LOCA were to develop, it is likely that it would be a small LOCA due to less than complete failure of the CRDM nozzle. The conditional probability of core damage is higher for medium LOCAs than for small LOCAs and is much higher than for a situation in which the plant is shut down due to RCS leakage. Therefore, this assumption represents a very significant source of conservatism in this analysis.

- For the calculation of leak frequency, it is assumed that any CRDM nozzle leaks detected at B&W plants were initiated in the most recent two operating cycles. However, the Davis-Besse analysis applies this initiating event frequency over a time encompassing four operating cycles which is a very conservative approach and effectively "double counts" the observed leak occurrence rate. This very conservative assumption was applied to ensure the possibility of undetected leaks initiated in earlier operating cycles was considered.

- The nozzle leak frequency used in this calculation is a generic rate calculated using experience from ONS-1, ONS-2, ONS-3, ANO-1 and CR-3. This frequency neglects Davis-Besse plant specific experience, which if used to update the generic data, would reduce the leak frequency by a factor of three. If the updated leak frequency were used to calculate the CDF and LERF, both of these frequencies would be reduced by a factor of three.

Based on the risk acceptance guidelines in Regulatory Guide 1.174 (reference 3), the increase in core damage frequency estimated for CRDM nozzle cracking falls in Region 2 and is categorized as small.

Large Early Release Frequency

The Davis-Besse plant specific large early release frequency contribution was estimated to be 1.0×10^{-8} / yr based on the bounding case where all nozzle failures are assumed to cause the most severe medium LOCA. Based on the risk acceptance guidelines in Regulatory Guide 1.174 (reference 3), the large early release frequency estimated for CRDM nozzle cracking is categorized as very small.

The very small LERF contribution estimated for CRDM nozzle cracking is expected due to unique plant features. Davis-Besse has a large dry containment with a large containment to core power ratio. Consequently, the most significant contributors to release from containment for Davis-Besse are containment bypass events such as steam generator tube ruptures or interfacing system LOCAs. Other significant contributors to LERF include sequences where core damage results from loss of support systems that also support containment cooling. These sequences include loss of cooling water systems or station blackout. Medium LOCAs are not representative of either of these dominant LERF sequences. Additionally, LOCAs depressurize the reactor coolant system prior to core damage, which reduces the possibility of high pressure melt ejection.

Public Health Risk

The public health consequences of potential CRDM nozzle failures are also very small. For the bounding case where all nozzle failures are assumed to cause the most severe medium LOCA, the estimated person-rem per year is about 6.5×10^{-1}. This is a negligible contribution to public health risk.

Summary

This analysis specifically evaluates the risk for the current 13th operating cycle. The risk for subsequent cycles will be much lower after a 100% inspection of all 69 CRDM nozzles is completed during 13RFO. Therefore, this calculation represents the maximum risk expected from potential CRDM nozzle cracking.

In summary, using bounding analysis the risk of core damage from CRDM nozzle cracks can be categorized as small using the guidelines in Regulatory Guide 1.174 for core damage frequency. The large early release frequency risk is very small and the person-rem per year contribution from this event can be considered to be negligible.

References

1. C-NSA-99.16-46, Risk Assessment for CRDM Nozzle Cracks, Revision 0, November 2001.
2. Framatome Document 51-5012567-01, RV Head Nozzle and Weld Safety Assessment.
3. 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.
4. Responses to Requests for Additional Information Concerning NRC Bulletin 2001-01, "Circumferential Cracking of Reactor Pressure Vessel Head Penetration Nozzles", Serial 2741.
5. Davis-Besse Level 1 PSA, Revision 3, May 2001.
6. Davis-Besse Level 2 PSA, Revision 1, October 2001.
7. BAW-2382, Level 3 PRA for Davis-Besse, January 2001.
8. SIA Engineering Report for CRDM Nozzles, September 2001.

Figure 1 – Event Tree for Nozzle Failure Core Damage Frequency (Group 1)

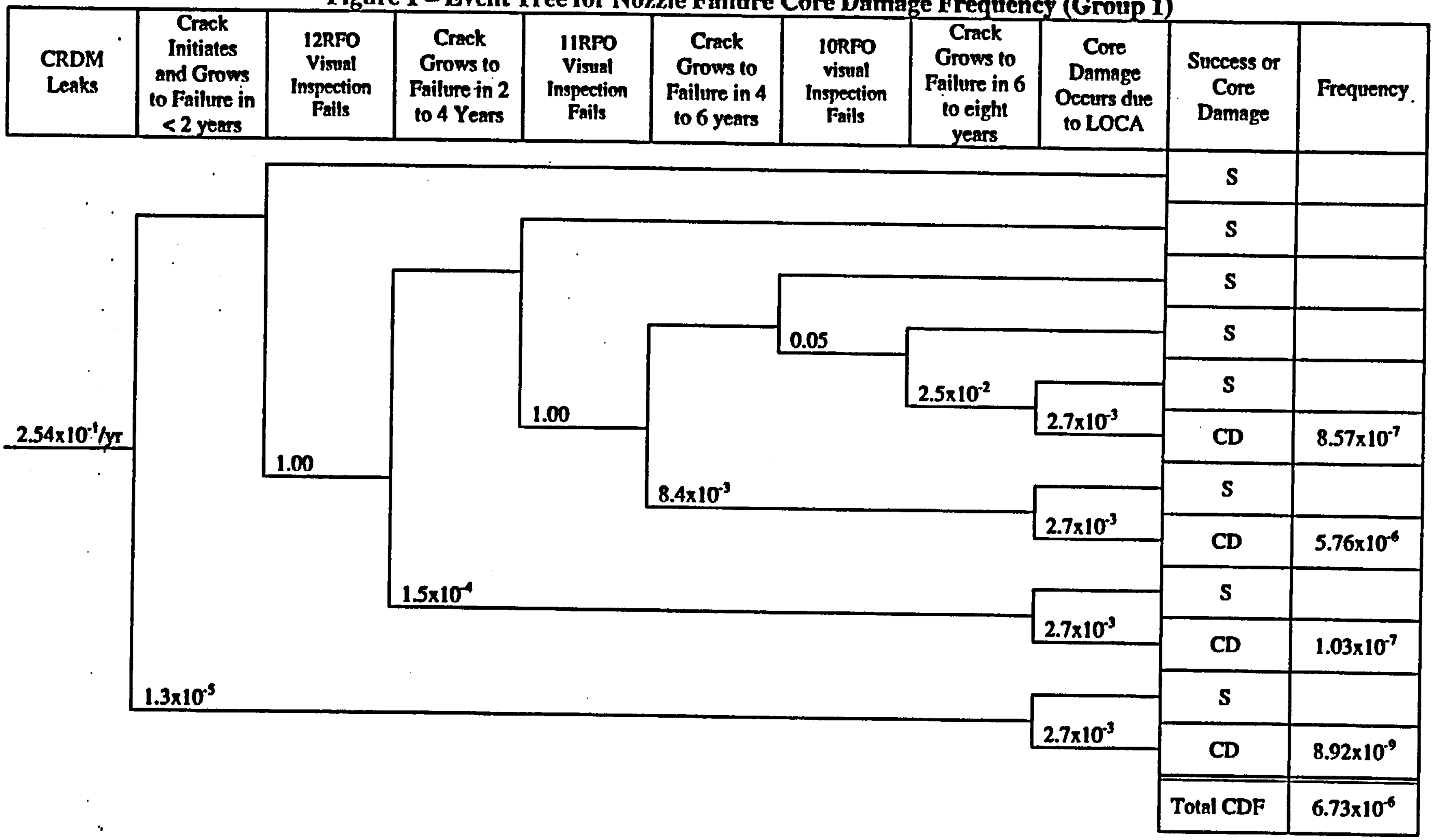

Figure 2 – Event Tree for Nozzle Failure Core Damage Frequency (Group 2)

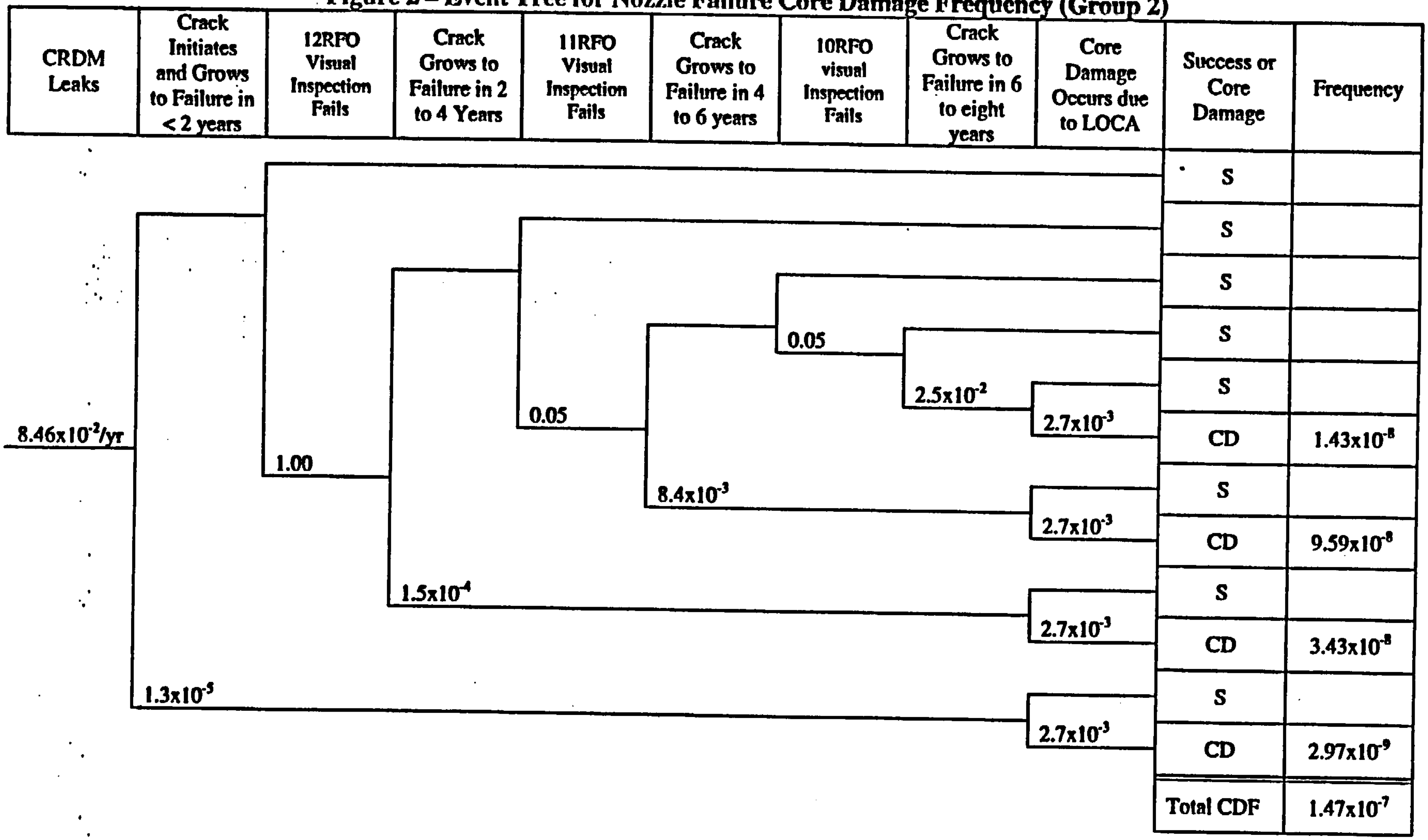

Figure 3 – Event Tree for Nozzle Failure Core Damage Frequency (Group 3)

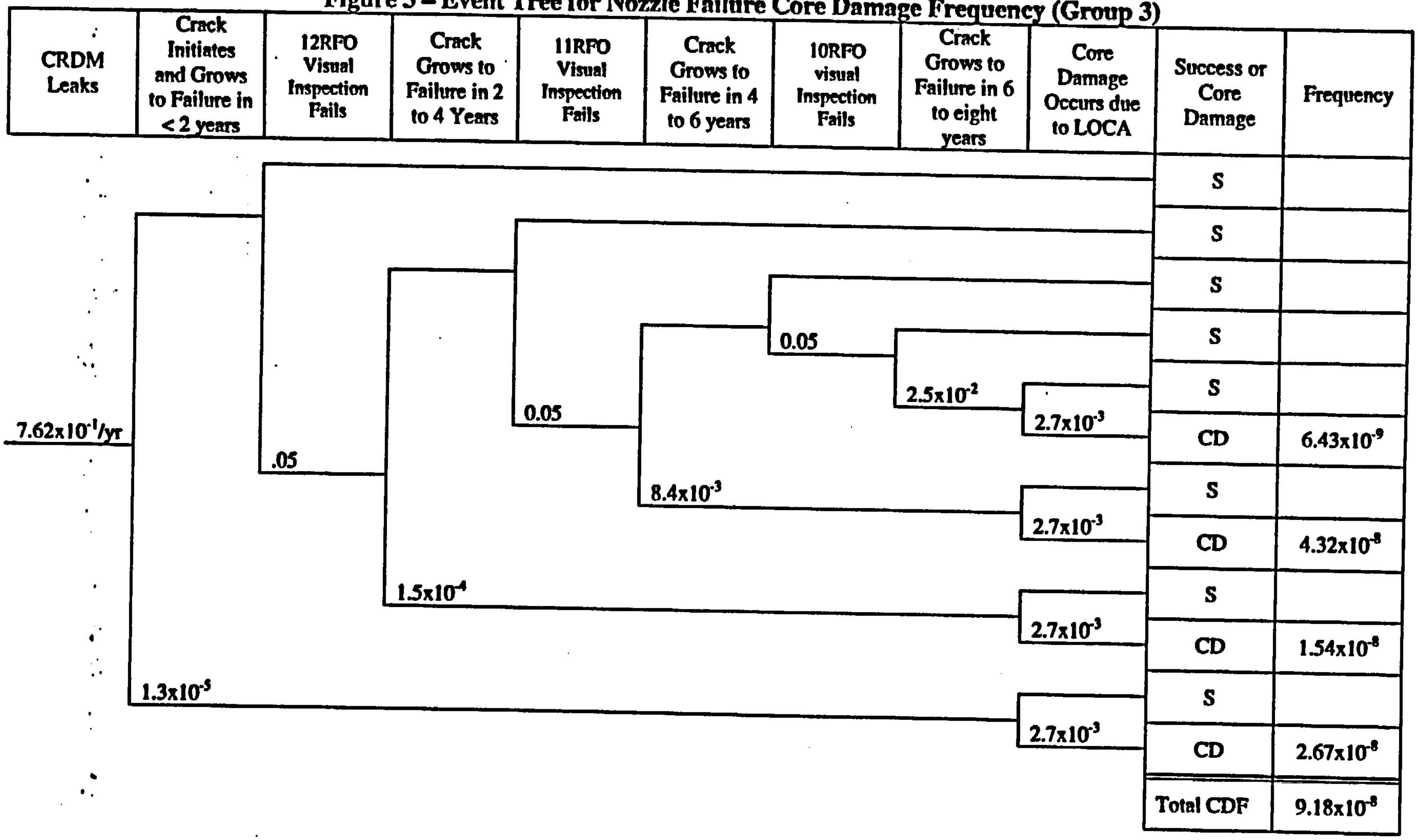

Table 5 – Summary of Risk Results Within 50 Miles

Containment End States	Early Fatalities /	Early Injuries /	Latent Fatalities /	Thyroid Cancers /	Whole Body Person
Steam Generator Tube Rupture	0.00 0.00%	0.00 0.00%	0.00 0.00%	0.00 0.00%	0.00 0.00%
ISLOCA	0.00 0.00%	0.00 0.00%	0.00 0.00%	0.00 0.00%	0.00 0.00%
Large Isolation Failure	6.91×10^{-11} 0.75%	6.45×10^{-10} 1.19%	7.44×10^{-07} 0.23%	1.41×10^{-08} 0.24%	1.02×10^{-03} 0.16%
Small Isolation Failure	3.66×10^{-09} 39.47%	2.63×10^{-08} 48.58%	1.03×10^{-04} 31.74%	1.88×10^{-06} 31.98%	2.06×10^{-01} 31.89%
Early Containment Failure	4.29×10^{-10} 4.63%	1.82×10^{-09} 3.36%	1.30×10^{-05} 3.99%	1.58×10^{-07} 2.70%	2.32×10^{-02} 3.59%
Late Containment Failure (Catastrophic)	4.43×10^{-09} 47.82%	1.73×10^{-08} 31.89%	4.18×10^{-05} 12.81%	1.19×10^{-06} 20.34%	5.67×10^{-02} 8.76%
Late Containment Failure (Benign)	6.61×10^{-11} 0.71%	1.86×10^{-10} 0.34%	9.42×10^{-07} 0.29%	1.61×10^{-08} 0.27%	2.01×10^{-03} 0.31%
Basemat Meltthrough	6.14×10^{-10} 6.62%	7.93×10^{-09} 14.64%	1.11×10^{-04} 33.90%	1.81×10^{-06} 30.78%	2.36×10^{-01} 36.43%
No Containment Failure	0.00 0.00%	0.00 0.00%	5.55×10^{-05} 17.03%	8.03×10^{-07} 13.68%	1.22×10^{-01} 18.86%
TOTAL	9.27×10^{-09}	5.42×10^{-08}	3.26×10^{-04}	5.87×10^{-06}	6.48×10^{-01}

Commitment List

The following list identifies those actions committed to by the Davis-Besse Nuclear Power Station (DBNPS) in this document. Any other actions discussed in the submittal represent intended or planned actions by the DBNPS. They are described only for information and are not regulatory commitments. Please notify the Manager - Regulatory Affairs (419-321-8450) at the DBNPS of any questions regarding this document or associated regulatory commitments.

COMMITMENT	DUE DATE
None	