Westinghouse Non-Proprietary Class 3

An Estimation of Plant Risk Due to Reactor Vessel Head Penetration Cracks

for D. C. Cook Unit 2

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1.0 INTRODUCTION

1.1 Introduction

This report provides a risk assessment of the events that can result from reactor vessel head penetration circumferential cracks that may occur in D.C. Cook Unit 2, if the upper head inspection is delayed after the year 2001. The probabilities of leakage occurring in a specified period of time are calculated in Reference 1 for all US plants for a six-year period. These probabilities are used to obtain yearly frequencies of the leakage events, which are modeled as the initiating event for a PRA model for D. C. Cook Unit 2. In doing so, it has been assumed that a leak may progress to larger leaks, up to and including a medium LOCA (full circumferential break of penetration tube). The model is discussed in Section 2 of this report in terms of an event tree. This event tree is quantified for different plants and different initiating event frequencies to estimate plant CDF from the initiating event. Plant LERF is also estimated. The incremental plant risk due to an upper head inspection delay of 30 days is compared against the acceptance criteria, which are discussed below.

A bounding sensitivity analysis where it is assumed that the leak will progress to a medium LOCA event with a 100% certainty is also made and is presented in Section 3.

The results are summarized in Section 4.

1.2 Risk Acceptance Criteria

In order to make a risk assessment of an event, an acceptance criteria for "small" risk must be defined. Furthermore, this definition must be acceptable to the NRC. Since a temporary status is sought after for a short time period, the plant configuration risk criteria used for maintenance rule A4 will be used.

CDP Acceptance Criteria:

Core damage risk is small if the incremental risk due to the temporary plant configuration, namely the incremental plant core damage probability (ΔCDP), is 1-E-6 or less. This is calculated as the additionally incurred plant CDF due to the temporary plant status times the time interval in which this status is to be maintained:

ΔCDP = CDF (incremental) * time period it applies.

LERP Acceptance Criteria:

Similarly, the incremental Large Early Release Probability (ΔLERP) risk is small if the contribution to plant LERP is 1-E-7 or less.

Once the ΔCDP criterion for small risk is met, the ΔLERP criterion can also be met automatically, if it can be demonstrated that the conditional LER probability, CLERP, (given that core damage has occurred) is 0.1 or smaller. For events that do not directly lead to containment bypass (events such as SGTR and interfacing systems LOCA have direct containment bypass potential), CLERP has been observed to be in the range of 0.01 to 0.10 for Westinghouse PWRs. This is especially true for the LOCA events since they are design basis events, and the realistic containment yield strength is a factor of two higher than the accepted design strength.

1.3 Summary of Risk Acceptance Criteria

The risk of an event is considered to be small if its contribution to plant CDP is 1-06 or less and its contribution to plant LERP is 1-07 or less during the time period for which it is applicable.

The acceptance criteria are presented as single values, taken as mean values, for decision making. A sensitivity analysis labeled as the bounding case is also made to understand upper bounds of the potentially incurred additional risk.

In demonstrating that the risk of the event studied meets the acceptance criteria for small, internal events at power and events at other plant conditions (such as external events, shutdown events, etc.) should be taken into consideration, either qualitatively, or preferably quantitatively (if feasible).

2.0 RISK MODEL

In this section, a risk model is defined for event sequences that may follow an axial crack initiated as defined in Reference 1 in the vessel head penetrations. First the initiating event and its frequency are introduced. Then, an event tree model to calculate the plant CDF due to the event is presented. The CDF for the event is calculated and is used to estimate the ΔCDP and ΔLERP. Finally, ΔCDP and ΔLERP are compared against the acceptance criteria.

2.1 Initiating Event

The initiating event is the first leak in the upper head penetrations as a result of PWSCC (primary water stress corrosion cracking). In Reference 1, for all 69 domestic PWR units the probabilities of first axial leak occurring by, respectively, 1.5, 3, 4.5, and 6 EFPY into the future (where "now" is defined as February 2001) are calculated. Reference 1 provides the data in yearly increments and is used to assign yearly frequencies to first leak. The data taken from Reference 1 is given in Table 2.1-1.

The cumulative probabilities in Table 2.1-1 are converted to yearly frequencies. For this purpose the following equation is used:

$f_i = (p_i - p_{(i-1)}) / (1-p_{(i-1)})$, for i = 1,2,3,4,5,6.

These frequencies are shown for D. C. Cook Unit 2 in Table 2.1-2. Note that there is an increasing trend in the frequencies. Thus, it is conceivable that the risk acceptance criteria may be met for early years, but may not be met for later years. An average risk over the years is not calculated, since the yearly frequency trend is increasing and the average may mask the unacceptable years.

Table 2.1-1 Cumulative Probability per Year (EFPY) of Leak for D. C. Cook Unit 2

Unit	Probability of First Leak by "now" (02/2001)	Probability of First Leak within 1 Added EFPY	Probability of First Leak within 2 Added EFPY	Probability of First Leak within 3 Added EFPY	Probability of First Leak within 4 Added EFPY	Probability of First Leak within 5 Added EFPY	Probability of First Leak within 6 Added EFPY
	p0	p1	p2	p3	p4	p5	p6
Cook 2	9.86E-03	2.05E-02	4.01E-02	7.46E-02	1.31E-01	2.19E-01	3.44E-01

Table 2.1-2 Probability per EFPY of Leak for D. C. Cook Unit 2

Unit	Probability of First Leak by "now" (02/2001)	First Year	Second Year	Third Year	Fourth Year	Fifth Year	Sixth Year
	p0	f1	f2	f3	f4	f5	f6
Cook 2	9.86E-03	1.07E-02	2.00E-02	3.59E-02	6.09E-02	1.01E-01	1.60E-01

2.2 Event Progression

After the initiation of the first through-the-wall axial crack (leak) which is defined as the initiating event, the following four categories of events may develop during the year (EFPY) in question:

1. The leak is not detectable by CVCS system (less than 1 gpm). It can be detected by visual inspection and presence of boron on the outside of the upper head. This event does not lead to RCS inventory loss of measurable amount and does not pose a safety challenge during the year.

   However, the model should consider the probability of this case to become one of the more consequential cases discussed below in the following years, if this crack is not detected and repaired.

   This event will be termed as a Leak Event in the risk model.

2. The leak is within the CVCS makeup capacity (less than 100 gpm). The CVCS makeup occurs; the plant is manually shutdown for repairs. This type of event is termed as small-small LOCA in PRA models. The risk from this category event is smaller than that of LOCAs of larger size.

3. The leak is large enough to be termed as a small LOCA, which is generally defined as a LOCA with an equivalent diameter of 2" or less, but is greater than that of a small-small LOCA). Such an event would cause an automatic reactor trip; actuation of auxiliary feedwater (AFW) and Safety Injection. The conditional core damage probabilities (CCDP) of small LOCAs are routinely calculated in nuclear power plant PRAs, and are available on a plant by plant basis.

4. The leak is large enough to be termed as a medium LOCA, which is generally defined as a LOCA with an equivalent diameter of greater than 2", but is less than a Large LOCA (6" equivalent diameter or greater). Moreover, the crack may be circumferential and the head penetration may separate from the upper head, causing a control rod ejection. Ejection of a single control rod is a design basis accident for which the plant is designed against. The risk from rod ejection accidents was never observed to be an important contributor to plant risk, based on past PRA experience. Thus, the dominant event in this case would be the medium LOCA. Due to the physical dimensions of the opening below the circumferential break, the medium LOCA would be limited by a 2.75" equivalent diameter event (tube inner diameter is 2.75"). After the medium LOCA and rod ejection, the reactor will automatically trip; AFW and safety injection will actuate, accumulators may inject in time. The conditional core damage probabilities (CCDP) of medium LOCAs are routinely calculated in nuclear power plant PRAs, and are available plant by plant basis. In general it is observed that the CCDP of a medium LOCA event is larger than that of a small LOCA event.

   The control rod ejection event is a design basis event and the plant layout is such that there is no expected significant risk due to interference of the damaged rod with the reactor rod insertion process. Moreover, the missile generated is not expected to damage in-containment equipment to be used in medium LOCA mitigation. The water flowing and spraying out of the broken penetration is not expected to render in-containment equipment to be used in LOCA mitigation. Thus, the medium LOCA event is the dominant risk contributor to the set of events that can be envisioned after a full circumferential break of an upper reactor head penetration.

Thus, it is possible that three of the four event sequences described above may lead to plant core damage during the year of interest. For each of the three categories, except for the leak event (category 1 above), the CCDP can be either obtained from a plant-specific PRA model, or can be generically estimated. The plant risk model representing the above discussion is given in Figure 2.2-1, and it can be used to estimate the CDF from the initiating event. Then the CDP for the time period of interest (ΔT) can be calculated by using the equation:

$\Delta CDP = CDF * \Delta T$ Equation 2.2-1.

If ΔCDP is smaller than 1.0E-06 (as discussed in Section 1), then the plant risk can be termed as acceptably small.

Note that plant risk outside of modes 1 and 2 is not modeled since it is deemed to be insignificant due to lower RCS pressures and temperatures during other plant modes.

Figure 2.2-1 Risk Model for Plant CDF

CDF Risk Assessment Model Event Tree

Leak Frequency	Type of Leak or LOCA	Conditional core damage Probability	Sequence	End State	Frequency
	q1 0		1	OK- seepage	
			2	OK - leak	
f1 1.07E-02	q2 0.9	p2 3.310E-05	3	CD - leak	3.20E-07
			4	OK - small LOCA	
	q3 0.08	p3 3.310E-03	5	CD - small LOCA	2.83E-06
			6	OK - Medium LOCA	
	q4 0.02	p4 4.520E-03	7	CD - Medium LOCA	9.67E-07
				Total CDF =	4.14E-06

Where:

q1 = fraction of leaks remain as "leak events";
q2 = fraction of events that develop into small-small LOCA;
q3 = fraction of events that develop into small LOCA;
q4 = fraction of events that develop into large LOCA.

Note that the sum of q1+q2+q3+q4 is one.

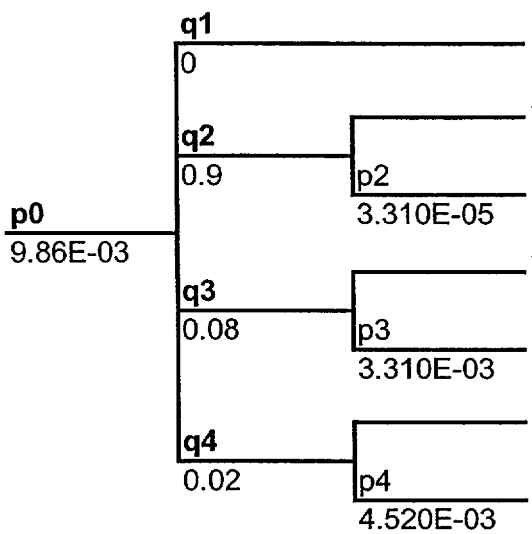
p2 = CCDP of small-small LOCA event;
p3 = CCDP of Small LOCA event;
p4 = CCDP of medium LOCA event.

In general p2 < p3 < p4.

CD = core damage.

Figure 2.2-2 Core Damage Probability for the Period up to "now"

Leak Probability up to "now"	Type of Leak or LOCA	Conditional core damage Probability	Sequence	End State	Frequency



Sequence	End State	Frequency
1	OK- seepage	
2	OK - leak	
3	CD - leak	2.94E-07
4	OK - small LOCA	
5	CD - small LOCA	2.61E-06
6	OK - Medium LOCA	
7	CD - Medium LOCA	8.91E-07
	Total CDF =	3.80E-06

2.3 Model Quantification

In order to quantify the risk model of Figure 2.1-1, the quantities f, q, and p must be known. The calculated values are expected values (mean values).

The initiating event frequencies (f) for the first six years (f1 through f6) are given in Table 2.1-2.

For the purposes of this phase of quantification, the fraction of leak events that may result in small-small LOCA, small LOCA, and medium LOCA are estimated as follows:

$q2 = 0.90$

$q3 = 0.08$

$q4 = 0.02$

Note that this implies $q1 = 0$, namely no first leak will stay undetected and will be transferred to the next year for a potential LOCA event in the next year.

The above fractions currently reflect somewhat conservative (but not bounding) estimates. A bounding estimate is presented separately in Section 3 with $q4 = 1.0$ (all leaks are assumed to lead to medium LOCA).

The CCDPs of D. C. Cook Unit 2 is taken from Reference 2. They are as follows:

$p2 = 0.01*p3 = 3.31E-05$

$p3 = 3.31E-03$

$p4 = 4.52E-03$.

Since p2 is not calculated in the PRA, it is estimated as follows: the failure probability of CVCS to deal with the leak is assigned 0.01 (actual failure probability may be lower). Upon failure of CVCS, a small LOCA plants response is expected. Thus, the p2 is calculated as shown above.

An example calculation for the first year is shown in Figure 2.2-1. The results are shown in Table 2.3-1 for up to six years. The CDFs are labeled as CDF-1 through CDF-6 for each of the six years. These CDFs are not cumulative, but apply to each year. The CDP acceptance criteria of less than 1.0E-06 must be satisfied for the time period of interest in which the inspection is to be delayed.

Note that, the total core damage probability up to "now" (2/2001) can also be calculated by using the probability p0 of first leak by "now". This calculation is shown in Figure 2.2-2. The resulting probability is also very small (3.8 E-06).

2.4 CDP Acceptance

Using Equation 2.2-1 and a time period of 30 days for the inspection delay, the incremental plant ΔCDP due to delay of the inspection can be calculated as:

$\Delta CDP = 4.14E-06$ events/year * 30 day /365(days/year) = 3.4 E-7 events.

This incremental CDP for a 30-day delay of the inspection is less than the acceptance criteria for D. C. Cook Unit 2. Thus, no immediate actions are necessary.

2.5 LERP Acceptance

PRA experience indicates that for design basis events that are not direct containment bypass events, the conditional large early release probability (CLERP) is at the order of a few percent. Since LOCA events meet the criteria mentioned in the last sentence, its CLERP is expected to be in the range of 0.01 to 0.10. Namely, if $\Delta CDP < 1.0E\text{-}06$, and $CLERP < 0.10$, then

$\Delta LERP = \Delta CDP * CLERP < 1.0E\text{-}07$.

In fact, the CLERP for D.C. Cook Unit 2 for LOCA events is given in Reference 2 as follows:

CLERP for Small LOCA = 0.0517

CLERP for Medium LOCA = 0.0457.

Using the largest CLERP of 0.0517 for all CDF events, if the plant CDF leads to an acceptable CDP, then the LERF associated with it would be 0.0517 * CDF. Thus, once the plant meets the ΔCDP acceptance criteria, it will also meet the ΔLERF acceptance criteria. Note that this is valid as long as CLERP of a plant is less than or equal to 0.10 for LOCA events.

Table 2.3-1 Plant CDF Results – Base Case

Unit	cdf-1	cdf-2	cdf-3	cdf-4	cdf-5	cdf-6
Cook 2	4.14E-06	7.70E-06	1.38E-05	2.35E-05	3.90E-05	6.16E-05

3.0 BOUNDING SENSITIVITY CASE

In this section, a sensitivity case for the model presented in Section 2 is made. In this sensitivity case, the leak event is assumed to lead directly to a medium LOCA, e.g. a full circumferential crack that results in a break of one of the control rod upper head penetrations. This sensitivity case intends to address the largest perceived uncertainty in the calculation, namely the uncertainty in the assigned probabilities of the event tree node labeled as "Type of leak or LOCA".

The model of Section 2 is quantified by setting q1 = q2 = q3 = 0, and q4 = 1.0. An example CDF quantification is given in Table 3-1. The results of the sensitivity analysis for D. C. Cook Unit 2 is given in Table 3-2. cdf-1, through cdf-6 refer to plant CDF due to the postulated medium LOCA event.

Using Equation 2.2-1 and a time period of 30 days for the inspection delay, the incremental plant ΔCDP due to delay of the inspection can be calculated as:

ΔCDP = 4.84E-05 events/year * 30 day /365(days/year) = 4.0 E-06 events.

ΔCDP for a 30-day delay of the inspection for the bounding is slightly above the acceptance criteria. This increases our confidence that the risk associated with this event is tolerable. Note that the upper bound probability need not meet the acceptance criterion.

Table 3-1 An Example Quantification for the Bounding Sensitivity Analysis

Leak Frequency	Type of Leak or LOCA	Conditional core damage Probability	Sequence	End State	Frequency
	q1 0		1	OK- seepage	
	q2 0		2	OK - leak	
f1 1.07E-02		p2 3.310E-05	3	CD - leak	0.00E+00
	q3 0		4	OK - small LOCA	
		p3 3.310E-03	5	CD - small LOCA	0.00E+00
	q4 1		6	OK - Medium LOCA	
		p4 4.520E-03	7	CD - Medium LOCA	4.84E-05
				Total CDF =	4.84E-05

Table 3-2. Plant CDF Results - Bounding Case

Unit	cdf-1	cdf-2	cdf-3	cdf-4	cdf-5	cdf-6
Cook 2	4.84E-05	9.04E-05	1.62E-04	2.75E-04	4.58E-04	7.23E-04

4.0 CONCLUSIONS

4.1 Conclusions

In this report, the plant risk stemming from reactor upper head axial cracks is studied for D. C. Cook Unit 2. A conservative model is generated to estimate the plant CDF and LERF as input to determine if the plant risk is acceptable if the plant continues to operate 30 days beyond the currently prescribed inspection date limit.

The incremental core damage probability for the base case is 3.4 E-7, which is well under the acceptance criteria. In fact the plant can have a 90-day inspection delay before exceeding the risk acceptance criteria.

The upper bound incremental core damage probability estimate is 4.0E-06, which is still in the range of a small probability. This increases our assurance that the incremental risk is tolerable and no immediate actions are needed.

Thus, not allowing the delay of the planned inspection from December 2001 to January 2002 for this plant would be needlessly overconservative. Even the bounding case results give a ΔCDP value at the order of 10-6, which, although is beyond the "small risk" threshold, is still very small for temporary risk tolerability.

Thus, for D. C. Cook Unit 2, a partial-year inspection delay may be sought with prudence.

4.2 Limitations

Although multiple leaks in the range of 1 to 9 have been observed in inspections of B&W plants, it is expected that the Westinghouse plants will have either no leaks or most likely one leak after an inspection. Thus, the expected number of leaks for the frequencies calculated in Table 2.1-2 is taken as 1. Otherwise, the progression of multiple leaks independently and in parallel should also be factored in by multiplying the calculated initiating event frequencies by the expected number of leaks.

Time dependence of a crack growing into a larger crack with a more potential for a larger break is simply modeled through the use of various leak/LOCA sizes (probabilities q1 through q4 in the event tree model). Analytic work on this subject is not done yet. Thus the risk results are based on these probabilities of various size leaks. It is deemed that the probabilities used are conservative. Work is underway to better quantify these probabilities and it is expected that the probability of a complete break will be found to be lower than the base case assumption. This may move the base case CDF frequencies lower.

5.0 REFERENCES

1. Westinghouse Calcnote: STD-01-0048. Progress Report on EPRI MRP 82/182 Project: "Reactor Vessel Head Penetration Data Analysis and Risk Assessment (rev.2)" (STD-0 1-0047), Robert K. Perdue, October 2001.

2. American Electric Power Letter (DCN-6280.8 NDM Correspondence No. 2001-3853) dated October 25, 2001, signed by J.T. Hawley and P.L. Appignani.

ATTACHMENT 2 TO C1101-05

WESTINGHOUSE ELECTRIC CORPORATION
"STD-01-0047, PROGRESS REPORT ON EPRI MRP 82/182 PROJECT,
'REACTOR VESSEL HEAD PENETRATION DATA ANALYSIS AND
RISK ASSESSMENT (REVISION 2): NON-PROPIETARY VERSION'"

STD-01-0047

Progress Report on EPRI MRP 82/182 Project, "Reactor Vessel Head Penetration Data Analysis and Risk Assessment (revision 2): Non-Proprietary Version

October 22, 2001

Prepared by: *Robert K Perdue* Robert K. Perdue[1]   Reviewed by: *Clark W. Mycoff 10/25/01* Clark W. Mycoff

Background & Objective: A recent EPRI safety assessment of the CRDM nozzle leaks detected in recent inspections at Oconee Units 1, 2, & 3 and ANO-1states that:

> "The CRDM nozzle leaks were traced to predominantly axial PWSCC cracks initiating on the outside surface of the nozzle wall below the J-groove weld. The crack at Oconee 1 appeared to initiate on the surface of the J-groove weld. Two of the leaking nozzles at Oconee 3 had circumferential cracks propagating from the OD of the nozzle above the J-groove weld." (p. vii)- *PWR Materials Reliability Program, Interim Alloy 600 Safety Assessments for US PWR Plants (MRP-44): Part 2: Reactor Vessel Top Head Penetrations, EPRI, Palo Alto, CA: 2001. TP-1001491, Part 2.*

In short, the axial leaks of concern are a result of PWSCC (primary water stress corrosion cracking). A conceptual simulation model for a crack at a specified location category that leads to a rod ejection event is illustrated in Figure 1 (see the **Appendix** for a more detailed description of Figure 1). The figure is actually an "influence diagram" (a tool borrowed from decision analysis) that shows the various stages of a crack as it progresses from the first appearance of a detectable axial crack to an Axial TW (leak) to a Circ Crack Initiation to a critical circ crack and rod ejection. Overlaid on this progression is the inspection strategy and the consequent probability that the crack will be detected at its various stages.

This report describes a statistical analysis of U.S. industry axial leak experience similar in approach to other stress corrosion cracking statistical studies in both related (e.g., steam generator tubes) and unrelated domains. Specifically, we use the industry experience to develop a statistical model of time to axial leak (thus by-passing the axial crack initiation - progression nodes in Figure 1) as a function of energy time (effective full power years) and head temperature. We next use the statistical model to calculate the probability of each U.S. nuclear unit having a leak now and at various times into the future. Finally, we discuss the implications of the results for an industry-wide sequential inspection and evaluate sensitivity of results to relaxation of assumptions.

Approach: The approach assumes that the cumulative fraction of relevant U.S. nuclear industry units that would experience the onset of PWSCC-related CRDM nozzle leaking by a specified time is governed by an energy time – temperature process that can be approximated by a two-parameter Weibull cumulative probability distribution,

$$(1) \quad F(x) = 1 - \exp(-(\frac{x}{\alpha})^{\beta}).$$

X is "equivalent degradation years" (EDY). Alpha = a scale parameter, and beta = a shape parameter. The cross-section of plants under study operate at different head temperatures. Since corrosion is known to be affected by temperature, a commonly-employed Arrhenius correction is assumed to account for the temperature effect:

(2) EDY of ith unit @ a reference temperature = EFPY of ith unit * $\exp[(Q/R)*(1/T_n - 1/T_o)]$ = EFPY of ith unit * TF_i.

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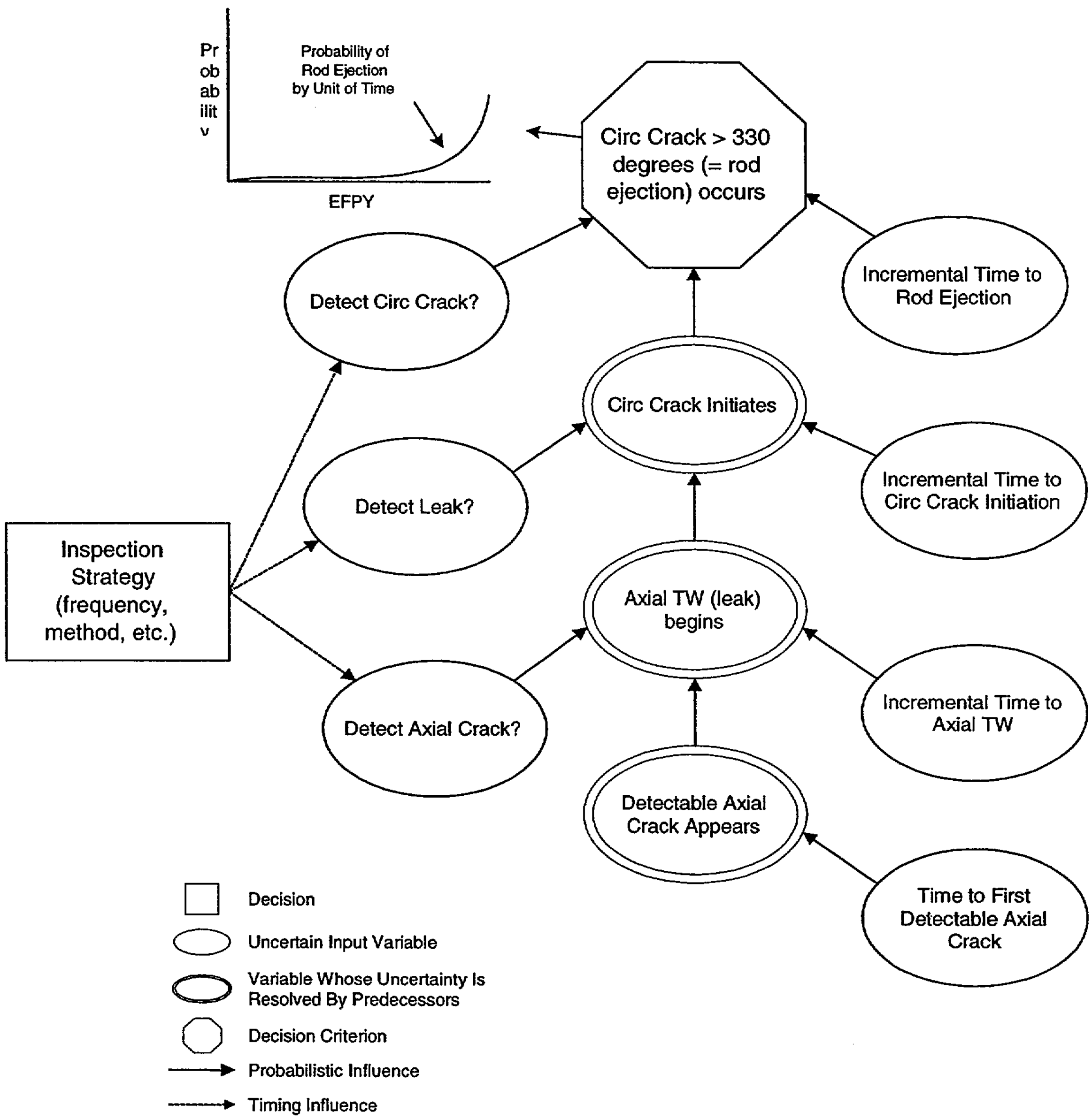
Where EFPY = effective full power year (= capacity factor*cumulative calendar years of operation), Q = apparent activation energy (Kcal/mole), R = 0.001103 Kcal/mole/°R, T_n = reference head temperature in absolute scale (Kelvin, K), T_o = ith plant's actual head temperature in absolute scale (K). TF_i = the temperature factor for the ith unit = $\exp[(Q/R)*(1/T_n - 1/T_o)]$. All calculations here assume Q=50 and T_n = 600 degrees F (these are actually assumptions made by Dominion Engineering, Inc.([4] and [6]) to calculate the data used in this study).

Figure 1: Conceptual Simulation Model of Rod Ejection for a Specified Location



It is assumed that the shape parameter is unaffected by temperature (Nelson, p.82). Only the scale parameter, which is also known as the "characteristic life" and is measured in the same units as X, is affected by the Arrhenius adjustment. Indeed, the basic idea is to fit (1) to industry data to get estimates of the scale and shape parameter appropriate for a single reference temperature, and then adjust the estimated scale parameter to each unit's actual head temperature (letting the shape parameter be the same for all units).

The model's parameters are estimated using data compiled by Dominion Engineering, Inc. to support the study in references [4] and [6] and updated for inspection results through the Fall 2001 outage season. Coming into the Fall, 2001 outage season, four units (Oconee Units 1, 2, and 3, plus ANO-1) had found leaks. In the Fall, 2001 outage season, more inspections took place, with the results as of October 22 being that 3 additional units (Beaver Valley 1, North Anna 1, and Farley 1) have found no leaks while two additional units (Crystal River 3, and TMI 1) have found leaks. Thus, since 1999 (inspections prior to 1999 are ignored), 19 units have inspected at least once, with 6 units (all with B&W nuclear steam supply systems) having found leaks. Table 1 shows how a "median rank analysis" has been employed to derive a cumulative distribution for the four units that have already experienced leaks. This is also standard practice (Abernethy, p. 16). Median ranks are estimates of the actual cumulative fraction failed based on the assumption that you have observed only a fraction of the total cumulative distribution of failures that will eventually occur if the items were allowed to run to failure.[2]

Table 1: Sample Data and Median Rank Analysis

Rank (by earliest EDY)	Unit	NSSS Supplier	Enhan. Visual or ID NDE	Date (Start of Outage)	Full / Partial	Equivalent Degradation Years (EDY) @ Start of Outage	Reverse Rank	Adjusted Rank	Median Rank Est. of Cum. Fraction
1	Prairie Is. 2	W	Visual	Apr-00	100%	9.2	19	Suspended	Suspended
2	Prairie Is. 1	W	Visual	Jan-01	100%	9.7	18	Suspended	Suspended
3	Salem 1	W	Visual	Apr-01	100%	10.3	17	Suspended	Suspended
4	Indian Pt. 3	W	Visual	Apr-01	60%	10.5	16	Suspended	Suspended
5	Beaver V. 1	W	Visual	Aug-01	100%	12.8	15	Suspended	Suspended
6	Ginna	W	NDE	Mar-99	100%	14.1	14	Suspended	Suspended
7	San Onofre 3	CE	Visual	Jan-01	34%	14.3	13	Suspended	Suspended
8	San Onofre 2	CE	Visual	Oct-00	34%	14.3	12	Suspended	Suspended
9	Farley 2	W	Visual	Feb-01	100%	14.4	11	Suspended	Suspended
10	Farley 1	W	Visual	Oct '01	100%	16.1	10	Suspended	Suspended
11	Crystal Rv. 3	B&W	Visual	Oct-01	100%	16.2	9	2	0.09
12	Davis-Besse	B&W	Visual	Mar-00	100%	16.9	8	Suspended	Suspended
13	TMI 1	B&W	Visual	Oct-01	100%	18.1	7	4.25	0.20
14	Robinson 2	W	Visual	Apr-01	100%	19.1	6	Suspended	Suspended
15	N. Anna 1	W	Visual	Sep-01	100%	19.4	5	Suspended	Suspended
16	ANO 1	B&W	Vis/NDE	Mar-01	100%	19.6	4	7.4	0.37
17	Oconee 3	B&W	Vis/NDE	Feb-01	100%	21.0	3	10.55	0.53
18	Oconee 1	B&W	Vis/NDE	Nov-00	100%	21.3	2	13.7	0.69
19	Oconee 2	B&W	Vis/NDE	Apr-01	100%	21.5	1	16.85	0.85

[2] Order the n failures from smallest to largest times to failure. Give the earliest failure rank 1, the second earliest rank 2 and so on. Calculate the median rank for each rank using the following Microsoft EXCEL function: Median Rank (i) = Betainv(0.5, rank(i), N-rank(i)+1), where N = failures plus non-failures. For more, see Abernethy (p.17, and Appendix I). If, as shown, there are non-failures (called suspensions) scattered among the failures, the non-failures are ordered along with failures and thereby affect the rank assigned to the failed items. Only the median ranks associated with *failures*, however, end up in the regression analysis. Equivalent Degradation Years in Table 1 are adjusted from data in the cited Dominion Engineering Report to the date at which the outage started using operating statistics from *World Nuclear Performance*, (1999, 2000), *Nucleonics Week*, The McGraw-Hill Companies Nuclear Publications.

Table 1 (last column) indicates that 9 percent of the fleet will experience first leak on or before they reach the current equivalent degradation age of Crystal River 3 (16.2 EDY) and 85% will leak by the time they reach 21.5 EDY.

Next, we need an empirical version of (1) to fit to the data. The classical Weibull regression model is obtained by taking the double log of (1) to obtain a version that is linear in the parameters:

(3) $\ln(-\ln(1-F(x))) = -\beta \ln \alpha + \beta \ln x$.

The median ranks in Table 1 are substituted for F(X) in (3). Letting the left term in (3) = Y = ln(-ln(1-Median Ranks))), the first term on the right in (3) = A, and ln(x) = ln(adjusted EFPY)=X, the resulting *linear regression model* is:

(4) $Y = A + \beta X + u$

Where u = the regression error term is distributed ~ normal with mean zero and standard deviation σ. Variability about the regression line is assumed to reflect the accumulated effects of random variations across plants in fabrication process, material heats, residual stress levels, et cetera after controlling for temperature differentials.

A least squares regression software package applied to (4) will generate an estimate of the shape parameter β while (2) can be used to solve for the Weibull *scale* (or characteristic life) parameter α (i.e., divide the estimated regression constant by the estimated slope and take the exponential of the result). The results of fitting (4) to the data in Table 1 are in Table 2 below.

Table 2: Regression Summary (standard errors in parentheses)		
A	β	R^2
-29.32	9.64	95.4 (%)
(3.16)	(1.06)	Std. Error of Regress.=0.2663

The derived scale value exp(29.32/9.64) = 20.94 EDY. Thus, using Table 2 and equations (1) and (2), the Weibull prediction model for the "mean" probability of an axial leak in ith unit's CRDMs before any specified time, measured in EFPY, is

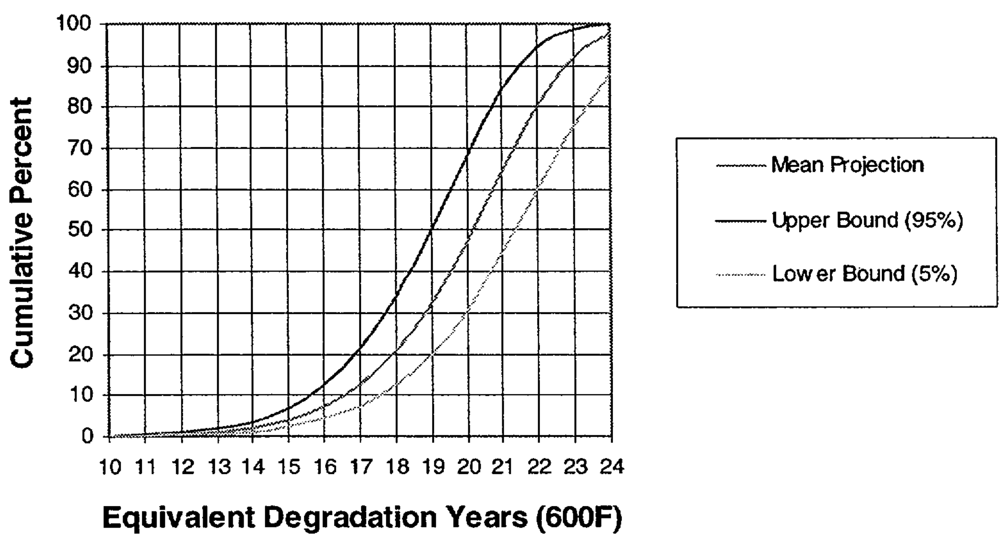
(5) $$F(EFPY) = 1 - \exp(-(\frac{EFPY}{20.94/TF_i})^{9.64}).$$

Figure 2 plots out Equation (3) for various values of EDY at the reference temperature (i.e., for a unit with the reference temperature of 600 degrees F). Upper and lower 90% confidence bounds are also shown.[3] Note that below a certain EDY (about 12 EDY for this temperature), the probability of a leak is virtually nil. After this age is reached, the probability increases very rapidly so that by 18 EDY, the mean probability forecast is above 20 percent. For lower temperatures

[3] Confidence bounds = (a complex transformation of) fitted Y +/- t-Statistic (alpha/2 = 5% for 4 degrees of freedom) times standard error of regression. The standard error of regression summarizes variability about the regression line. Note that the high R^2 (= percent of variation "explained" by the model) is consistent with the hypothesis that leaking is primary a function of age and head temperature, with other variables making only marginal contributions. Although biased upward by the median rank approach, the R^2 exceeds its "critical value" (Abernethy, p. 34, gives a graph of critical values at 90% confidence) for determining acceptability of the hypothesis that the underlying distribution is Weibull.

(not shown in Figure 2), the progression is less rapid and for the very lowest head temperatures, the growth is negligible over the life of the unit.

Figure 2: Weibull Model: Cumulative Distribution of Units with Leaks, by EDY (600F)



Actually, if a unit has recently (i.e., since Fall, 2000) inspected the CRDMs and found no leaks, then the unit is assumed to have zero leaks "now" (= February, 2001 for forecasting purposes). Thus, in calculating the probability of a leak by some future time defined by EFPY(now) + t, where t = the increment in EFPY to reach the future date, for a unit that has undergone a recent inspection a truncated version of (5) that recognizes this must be used:

$$(6) \quad F(EFPY) = \frac{F(now+t) - F(now)}{1 - F(now)}.$$

The term F(now) is Equation (5) set to the current EFPY and F(now+t) = Equation (5) set to current plus t EFPY ahead value. Equation (6) is, in other words, the conditional probability of a leak by the future date given <u>no</u> leak as of now. Equation 6 is applied to the units in Table 1 that have inspected since Fall, 2000, while Equation 5 is applied to all remaining units.

<u>Results</u>: Using Equation 5 or 6, as appropriate, Table 3 shows the calculated probability of first axial leak occurring by, respectively, 1, 2, 3, 4, 5 and 6 EFPY into the future (where, again, "now" is defined as February 2001) for the two DC Cook units.

Table 3: Predicted Probability of Axial Leak by Unit, from 1 to 6 EFPY Ahead

Unit (in descending order of Probability of Leak next 1.5 EFPY)	NSSS Supplier	Operating Time (EFPYs)[4]	Current Head Temp. (°F)	Head Temp. (°F)	Probability of First Leak by 02/2001	Probability of First Leak within 1 Added EFPY	Probability of First Leak within 2 Added EFPY	Probability of First Leak within 3 Added EFPY	Probability of First Leak within 4 Added EFPY	Probability of First Leak within 5 Added EFPY	Probability of First Leak within 6 Added EFPY
Cook 2	W	12.6	600.7	595.5	9.86E-03	2.05E-02	4.01E-02	7.46E-02	1.31E-01	2.19E-01	3.44E-01
Cook 1	W	15.1	578.0	591.5	7.04E-06	1.30E-05	2.33E-05	4.02E-05	6.74E-05	1.10E-04	1.76E-04

Cook 2 is orders of magnitude more likely to experience its first leak within 1 to 6 EFPY than is Cook 1. The probability of a leak for Cook 2 rises from about 2 percent in 1 EFPY to 34 percent in 6 EFPY. The comparable probabilities fro Cook 1 are 1.3E-5 and 1.8E-4.

Implications for Industry Response: This (and results for non-DC Cook units not shown) suggest that an industry-wide sequential inspection program, which focuses first on the units that have a significant probability of seeing a leak "soon," would be appropriate. The inspection results from these early inspections could be used to update and refine the statistical projections for remaining plants. Further, the model suggests that a large number of units could safely spread their inspections over (or postpone their inspections for) multiple refueling cycles because they simply have not accumulated sufficient "equivalent degradation years" to be at risk.

Note that this conclusion contradicts that of the "NRC Statistical Analysis" ([5]), which implies the need for all plants to perform 100% inspections. The NRC approach assumes that the same unknown probability (p) of a crack exists for each unit at a given point in time and then goes on to show that an acceptance sampling defense of anything less than 100% sampling would be hard to make. The critical assumption of a common binomial probability, however, is more appropriate to making inferences about a single unit (plant) based on a partial sample of CRDMs for that unit than for making inferences about the whole fleet of units based on a look at a sample of units. Our analysis makes the assumption that the likelihood of a crack or leak is an increasing function of each unit's energy time (EFPY) and head temperature. The latter is surely the appropriate assumption for an active corrosion mechanism and means that plants with a lower age measured in effective degradation years are less likely to experience the subject cracks or leaks than their older counterparts.

Table 4: Probability of Not Detecting at Least One Leak for Different Numbers of Leaks and Different PODs

	Number of Leaking CRDM Nozzles in the Unit								
POD	1	2	3	4	5	6	7	8	9
0.5	5.00E-01	2.50E-01	1.25E-01	6.25E-02	3.13E-02	1.56E-02	7.81E-03	3.91E-03	1.95E-03
0.6	4.00E-01	1.60E-01	6.40E-02	2.56E-02	1.02E-02	4.10E-03	1.64E-03	6.55E-04	2.62E-04
0.7	3.00E-01	9.00E-02	2.70E-02	8.10E-03	2.43E-03	7.29E-04	2.19E-04	6.56E-05	1.97E-05
0.8	2.00E-01	4.00E-02	8.00E-03	1.60E-03	3.20E-04	6.40E-05	1.28E-05	2.56E-06	5.12E-07
0.9	1.00E-01	1.00E-02	1.00E-03	1.00E-04	1.00E-05	1.00E-06	1.00E-07	1.00E-08	1.00E-09

Analysis of 100% POD Assumption :The foregoing analysis strives to predict a unit's time to the first observation of at least one leaking CRDM nozzles. The assumption is that if a unit inspects and does not find at least one leak, then no leaks are present. Sensitivity analysis indicates that up to two of the older units could have missed leaks without substantially altering regression results or forecast. [4]The likelihood of detecting a leak depends upon the number of nozzles inspected (all but three in the sample inspected 100%), the probability of detecting (POD) a leak if

[4] If, for example, Davis-Besse and Robinson 2 both had leaks (these are two older units in Table 1 that did not find leaks), then the regression estimate of the scale (characteristic life) falls from 21 EDY to 19, and the mean life drops from 20 EDY to 18 (a relatively smallchange).

in fact one is in the sample and the number of leaks to be found. The four original sample units that found leaks found an average of 5 (Oconee 1 = 6, Oconee 2 = 4, Oconee 3 = 9, ANO 1 = 1), suggesting that multiple leaks will be the rule. The probability of not finding at least one leak when n are present and 100% inspection is undertaken is $PND=(1\text{-}POD)^n$. Table 4 shows the probability of not detecting at least one leak for different hypothesized numbers of leaks present. Suppose (any) three units listed in Table 1 as not having found leaks did in fact have 5 leaks each and each inspected 100 percent with a POD of 50 percent. The probability that all three would fail to find at least one leak = $PND^3 = 3.13E\text{-}2^3 = 3E\text{-}5$. For another case, assume the three units each have 2 leaks with a POD remaining at 50 percent. In this case, the probability that all three would not find a leak is 1.6E-2. Thus, while we cannot rule out the possibility, if multiple leaks are the rule and the POD on each leak is at least 50%, then it would constitute a rare event for enough of the "suspended" units in Table 1 to have missed finding leaks to materially affect the results.

Analysis of Assumption Regarding Age of Leaks Found: The analysis also implicitly assumes that the leaks that were found occurred "recently." In fact, it is not known when they occurred, although both TMI1 and Crystal River 3 found leaks within 2 years of a prior inspection in which none had been found. If the leaks occurred at an age substantially younger than the EDY at which they were found, then the scale parameter in Equation 5 is underestimated and consequently the projections of probability of a leak are also underestimated. Sensitivity analysis shows, however, that if the scale parameter is reduced by up to 4 EDY, then the same units that were flagged as likely to see a leak soon are still (even more likely) to see a leak soon, the units in the middle ranks tend to move from "inspect later" to "inspect sooner" while the rest remain orders of magnitude less at risk during the next 1.5 to 3 EFPY. Plant X (a middle ranked unit), for example, shows probabilities of leak of 4.2E-3 and 2.4E-2 in 1 and 3 EFPY respectively (not shown in the non-proprietary version of this report). If the scale parameter in Equation 5 is reduced by 4 EDY, the corresponding probabilities rise to 3E-2 and 2E-1. Plant Y (a bottom quartile unit), on the other hand, shows no credible probability of a leak over the same period in either the estimated version or with the 4 EDY lower scale parameter. In general, the overall picture of which units should be looking sooner and which can delay inspections does not change.

Analysis of the Hypothesis That B&W Plants are More Susceptible: So far, all plants with leaks have B&W nuclear steam supply systems. This suggests the hypothesis that the B&W plants are more susceptible to the leaking CRDM nozzle phenomenon than their Westinghouse or CE counterparts. The extent to which field experience to date supports this hypothesis can be evaluated. First, we split the sample in Table 1 into B&W and Non-B&W units and run the regression model in Equation (4) on the B&W (only) units. We cannot run the regression on the non-B&W units because they have no failures. We can, however, employ a maximum likelihood technique that is sometimes called the *Weibayes estimator* (Abernethy, Appendix to Chapter 6). In this approach, we assume that the slope parameter is not affected by the NSSS supplier and is known to be equal to the B&W regression slope value. We do, however, allow differences in NSSS supplier to affect the shape parameter – that is, if the hypothesis is true, we expect non-B&W nozzles to exhibit a longer characteristic life. Let T_1, T_2, $\ldots T_n$ be both the failure and censured times (i.e., times for which no failures have yet occurred) to first failure observed across the non-B&W units. Let r be the number of failures in the data. Since, to date, r = 0, a conservative[5] lower v x 100% confidence interval for the maximum likelihood estimator of the Weibull scale parameter α is[6]

[5] "Conservative" means that the true confidence bound is unknown but it is at least the value shown.
[6] The Weibayes result is obtained as a natural extension of the observation that Weibull - distributed times to failure raised to the power of the shape parameter are distributed as an exponential distribution. This allows us to then use the simple formulas for maximum likelihood confidence bounds on an estimator of the exponential parameter (i.e., the scale parameter raised to the power β).

$$(7) \quad \alpha_l = \left[(\sum_{i=1}^{n} T_i^{\beta} / (-\ln(1-v)) \right]^{1/\beta}$$

In words, re-scale all of the observed times by raising them to the power of β = the assumed slope, sum, then divide by minus the natural log of 1minus the percentile of interest (v) and take the whole thing to the power of 1 over β. The median would be a logical choice for a point estimator. This is obtained by setting v = 0.50 in (4) to get

$$(8) \quad \alpha_{0.5} = \left[(\sum_{i=1}^{n} T_i^{\beta} / (-\ln(1-0.5)) \right]^{1/\beta} = \left[(\sum_{i=1}^{n} T_i^{\beta} / 0.69315) \right]^{1/\beta}.$$

Upper (95th percentile) and lower (5th percentile) confidence bounds can be obtained by an analogous process. Table 5 contains the results. The Non-B&W median scale parameter, as hypothesized, is above the B&W counterpart of 21.2. However, the latter scale estimate is within the plausible range of random variability defined by the 90% confidence bounds on the non-B&W scale estimate, suggesting that the null hypothesis of no difference between the two categories cannot be rejected. This, of course, is not to imply that we reject the proposition that the B&W units are more susceptible. Rather, we can only say that the evidence to date is not sufficient to support the proposition.

Table 5: Weibayes Result for Non-B&W Units

Shape (Slope)	Scale: Median	Scale: Lower CI	Scale: Upper CI
9.04	22.2	18.9	29.6

Note: "B&W Only" Scale = 21.2

References

[1] Abernethy, R. B., (1996), *The New Weibull Handbook* (available on Amazon.Com).

[2] EPRI MRP 82/182 Project, "Reactor Vessel Head Penetration Data Analysis and Risk Assessment: Technical Description," Westinghouse Electric Company

[3] Nelson, W., (1990), *Accelerated Testing: Statistical Models, Test Plans, and Data Analyses*, (available on Amazon.Com).

[4] Nuclear Energy Institute, Letter from Alexander Marion to Dr. Brian Sharon, Nuclear Regulatory Commission, entitled "NRC Staff Questions on EPRI Interim Report TP-1001491, Part 2, Section 4.0, Comment No. 2." July 31, 2001.

[5] Nuclear Regulatory Commission, Memo to William Bateman from Mark Cunningham, "Statistical Analysis of CRDM Sampling (Revised)," August 17, 2001.

[6] *PWR Materials Reliability Program, Interim Alloy 600 Safety Assessments for US PWR Plants (MRP-44): Part 2: Reactor Vessel Top Head Penetrations*, EPRI, Palo Alto, CA: 2001. TP-1001491, Part 2.

Appendix: Description of Figure 1 (Conceptual Simulation Model of Rod Ejection for a Specified Location)

Starting at the bottom right of Figure 1, if we know the "Time to First Detectable Axial Crack," then we know the date (e.g., EFPY) on which the first such crack will appear. Since the time to first crack is uncertain, we can represent the uncertainty with a probability distribution. Typically, for corrosion mechanisms, an "age-dependent" analytical probability distributions (e.g., the Weibull or the log-normal distribution) will be used to represent our uncertainty. These distributions have the property that the conditional probability of failure is not constant as a function of time but rather increases with age. The distribution's parameters may be estimated from historical data on observed crack initiations if available or from elicited "expert" distributions on time to failure or some combination of the two. Whether and when the detectable axial crack progresses to a leak is modeled as a function of the "Incremental Time to Axial TW" and whether the axial crack is detected at some point before leak. Note that the model as drawn assumes that a circumferential crack can only initiate when a leaking axial crack sets up a wet environment for the stress corrosion cracking mechanism.

Conceptually, the probabilistic time to leak for a specified "first" detectable axial crack can be modeled as a function of flaw characteristics weld thickness and environment by a "Structural Risk and Reliability Model" (SRRA). Whether a crack is detected (and presumably repaired) is a function of the "Inspection Strategy" chosen, including the method and the frequency with which it is applied. The uncertainty as to detection is represented by a probability of detection. Analogous comments are applicable to the "Circ Crack Initiates" and the "Circ Crack > 330 degrees (=rod ejection) occurs" nodes, where input from PRA models can be used. If implemented as a Monte Carlo simulation model, all uncertainties can be propagated through to produce a (conditional on age) "Probability of Rod Ejection by Unit of Time" for the location category of concern. The same simulation could be used to combine distributions across location categories.

In fact, this report provides a model that "leaps over" the nodes involving axial crack initiation and growth to leak and goes directly to the calculation of a probability of a leaking axial crack. Given the probability of a leak at future EFPY, the SRRA and PRA analyses can still be employed to model the remaining progression to the event of interest (i.e., initiation and growth of a circ crack and potential rod ejection).

ATTACHMENT 3 TO C1101-05

INDIANA MICHIGAN POWER COMPANY SUMMARY OF WESTINGHOUSE ELECTRIC CORPORATION SPECIFIC RISK ANALYSIS

Probabilistic Risk Assessment (PRA) of Vessel Head Penetration (VHP) Cracking

Scope of Assessment

A simplified Donald C. Cook Nuclear Plant (CNP)-specific risk analysis has been performed to provide insight into the potential additional risk due to VHP cracking (Attachment 1). This simplified risk analysis uses an event tree framework to relate the occurrence of an axial crack in a VHP to the potential for damaging the core and resulting in a large early release of radioactivity from the containment. The analysis estimates the increase in core damage frequency (CDF) and large early release frequency (LERF) for the year starting February 1, 2001, through January 31, 2002. These increased CDF and LERF values are then used to determine the incremental core damage probability and incremental large early release probabilities for extending operation of CNP Unit 2 from December 31, 2001, to January 30, 2002.

Description of Assessment Methodology

The steps performed for the CNP Unit 2 risk assessment for VHP cracking are as follows.

1. The starting point for this simplified risk analysis is an estimate of the leak frequency due to a VHP crack. This initiating event frequency is based on Attachment 2, which is a CNP-specific version of an industry analysis. This analysis provides the probability of a leak occurring during different time periods, including up to February 1, 2001, and then during the periods of 1 year, 2 years, 3 years, 4 years, 5 years, and 6 years following that date. This analysis includes the results of industry VHP inspections completed through October 24, 2001. The probability for the first year period is converted to an initiating event frequency for the year from February 1, 2001, until January 31, 2002.

2. The analysis considers four possible event progressions following the postulated leak initiation: the leak is not detectable by reactor coolant system leakage monitoring and does not pose a safety challenge; the leak is within the chemical volume control system make-up capability and is termed a small-small loss of coolant accident (LOCA); the leak is large enough to cause a reactor trip and safety injection signal and is termed a small LOCA; and the leak results in flow through the nominal VHP diameter and is termed a medium LOCA.

3. The risk analysis then makes the bounding assumption that the leak will grow to a size that requires mitigation within the next year. This assumption represents a significant conservatism in this risk evaluation since on-going industry fracture mechanics analyses indicate that the probability of a crack initiating and growing into such a large leak is on the order of 1E-02 to 1E-05 for operating time periods corresponding to CNP Unit 2.

4. The fractions of leaks leading to each of the possible event progressions are estimated based on engineering judgment of the relative likelihood of each leak size, except that all leaks are conservatively assumed to be large enough to require mitigation as stated in item 3 above. This means that the first assumed event progression described in item 2, which does not pose a safety challenge, is assigned a fraction of zero.

5. The event tree risk model then applies a conditional core damage probability that is appropriate for each of these potential leak sequence of events and determines the core damage frequency for each of these sequences.

6. The frequencies of all of the core damage sequences are summed to obtain an estimate of the change in CDF that may be associated with VHP cracking.

7. An estimate of the change in LERF that may be associated with VHP cracking is obtained by multiplying the overall CDF by the conditional large early release probability.

8. Since 19 days of operation beyond December 31, 2001, is under consideration, after which time the unit will be shut-down, inspected, and repaired (if necessary), this situation is viewed as a temporary condition. Accordingly, the incremental core damage probability (ICDP) and incremental large early release probability (ILERP) are determined, and compared to industry guidance for temporary risk conditions.

9. An additional potential risk impact of shutting down Unit 2 nineteen days earlier than is currently planned for entry into its refueling outage on January 19, 2002, would be the possibility that there is an increase in plant risk due to extended Mode 5 and Mode 6 shutdown time. Since the outage is not currently planned to start on December 31, 2001, some parts of the outage plan will occur less efficiently due to issues associated with purchase lead-times, contractor/equipment availability, etc. To remove the risk from VHP cracking, additional cold shut down time would likely accrue, and since these shutdown states have higher CDF than the hot modes, this risk may partially or completely off-set the higher risk assumed to exist in Mode 1 due to the potential for VHP cracking. The recently completed Shutdown Safety Monitor (SDSM) model was used to investigate this situation.

Results and Conclusions of Assessment

The results of this at-power event tree analysis, for the year of operation from February 1, 2001, to January 31, 2002, are as follows. For the more realistic case[1], the CDF increase is estimated as 4.14E-06/yr and the LERF increase is estimated as 2.14E-07/yr. For the bounding sensitivity case[2], the incremental CDF is estimated as 4.84E-05/yr and the incremental LERF is estimated as 2.50E-06/yr. The contributions to these results from the thirty-day period between December 31, 2001, and January 30, 2002, are an ICDP of 3.4E-07[1] (4.0E-06[2]), with a corresponding ILERP of 1.76E-08[3] (2.07E-07[3]) for the more realistic (bounding) estimates, respectively. Combining the numerical results of these cases, with the recognition that they contain an inherent conservatism that would reduce them by several orders of magnitude, provides confidence that the risk associated with operation of CNP Unit 2 through January 30, 2002, is well below the values of 1E-06 and 1E-07 for ICDP and ILERP, respectively, which are identified in NUMARC 93-01 for plant configurations not requiring any special risk management considerations.

In addition, this small increase in at-power risk is offset by additional unplanned Mode 5 shutdown time due to the earlier-than-planned shutdown. Based on the SDSM, the ICDP of an additional unplanned day in Mode 5 is about 6.1E-08. Consequently, all of the VHP cracking risk that is averted by early shutdown would be completely offset by the risk associated with five and one-half additional days in Mode 5 based on the more realistic case values above. This consideration provides additional support for operating CNP Unit 2 as planned.

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[1] Bounding More Realistic Case; See Attachment 1, Section 2.3.

[2] Bounding Sensitivity Case; See Attachment 1, Section 3.0.

[3] Obtained by multiplying the ICDP by the CLERP value of 0.0517 as stated in Attachment 1, Section 2.5, "LERP Acceptance."