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ALLOWABLE OUTAGE TIME STUDY FOR RESIDUAL HEAT REMOVAL VALVES FOR FARLEY NUCLEAR PLANT UNITS 1 AND 2

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1.0 INTRODUCTION

Resolution of Generic Issue 94, "Additional Low-Temperature Overpressure Protection for Light-Water Reactors," Pursuant to 10 CFR 50.54(f) is discussed in Generic Letter 90-06 (Reference 1). The general objective of GI-94 was to evaluate the need for additional low-temperature overpressure protection (LTOP) and to examine alternatives to reduce the risk of core damage accidents associated with low-temperature overpressure events in PWRs. The basis for this was the need to ensure that there is a low likelihood of brittle reactor pressure vessel failure. Such a failure could result in the reactor pressure vessel being unavailable for either subsequent recovery of the reactor core or as an additional barrier for fission product retention, following a core damage event.

The NRC staff prepared a regulatory analysis for GI-94 based on the results reported in NUREG-1326 (Reference 2). The NRC staff determined that LTOP protection system unavailability is the dominant contributor to risk from low-temperature overpressure transients. Resolution of GI-94 requires a revision to the plant technical specification for overpressure protection to ensure that both LTOP channels are operable, especially in a water-solid condition; that is, to treat the operability of LTOP as a system that performs safety-related functions. The specific action specified by the NRC is to reduce the allowable outage time (AOT) for a single LTOP channel when operating in Mode 5 (cold shutdown) or Mode 6 (with the reactor pressure vessel head bolted down) from the current AOT of 7 days to an AOT of 24 hours before remedial actions to depressurize and to vent the reactor coolant system would be required.

The probabilistic analysis documented in this report evaluates core damage frequency as a function of LTOP channels, which are the Residual Heat Removal (RHR) Safety Relief Valves (SRVs), being out-of-service concurrent with an overpressurization of the Reactor Coolant System (RCS). The objective of this analysis is to provide a basis for establishing an AOT for the RHR SRVs in Modes 4, 5, and 6 for Farley Nuclear Plants Units 1 and 2 to justify lengthening the AOTs recommended in Generic Letter 90-06.

2.0 SUMMARY

This section summarizes the methodology used to perform the probabilistic analysis to justify changing the AOT for Farley Units 1 and 2 and the results and conclusions of this study.

Methodology

The following steps were performed to evaluate the impact of changing the AOT for Farley Units 1 and 2:

1. Review the background documents supporting the NRC's justification for changing the AOT from 7 days to 24 hours,

2. Develop the probabilistic model for Farley Nuclear Plant Units 1 and 2,

3. Evaluate the probabilistic model for AOTs of 7 days and 24 hours, and

4. Calculate the core damage frequency for these AOTs.

Review of the NRC background probabilistic models (References 2 and 3) indicates that only two AOTs are analyzed: either the LTOP system is not available over the entire shutdown time interval or the LTOP system is available over the entire shutdown time interval.

To be consistent with the analyses performed in References 2 and 3, the same basic probabilistic methods as documented in Reference 2 are used to evaluate the frequency of core damage. These analyses were modified to be specific to Farley Units 1 and 2. Equations were developed to determine the impact of changing the AOT on the unavailability of the LTOP protection.

As stated in Reference 1 (Enclosure B to Generic Letter 90-06), "The NRC staff considered the conditions under which a low-temperature overpressure transient

is most likely to occur. While LTOP protection is required for all shutdown modes, the most vulnerable period of time was found to be MODE 5 (cold shutdown) with the reactor coolant temperature less than or equal to 200°F, especially when water-solid, based on the detailed evaluation of operating reactor experiences performed in support of GI-94."

Because the most vulnerable time for overpressure events is when the RCS is water solid, the time spent in modes 5 and 6 is divided into three time periods: the time when the RCS is water solid, the time when the RCS is not water solid, and the time when the reactor vessel head is removed. The operational philosophy at Farley Units 1 and 2 is to minimize operation in a water-solid condition to prevent overpressure events (estimated to be about 100 hours per year). As this is a short time interval, a maximum AOT of 24 hours is assumed for the analysis of core damage frequency during the time interval of water-solid operation. The unavailability of LTOP is calculated for AOTs of 7 days and 24 hours for modes 5 and 6 when the RCS is not water solid.

The total core damage frequency is obtained over the time interval spent in modes 5 and 6 with the reactor vessel head not removed. The core damage frequency reduction and the percent change in core damage frequency by changing the 7 day AOT to a 24 hour AOT are calculated.

Results and Conclusions

Based on the results obtained from this probabilistic analysis of core damage frequency, it was concluded that the AOT for water-solid operation should be reduced from the current technical specification value of 7 days to 24 hours. The current technical specification value of 7 days for all other low temperature conditions will be retained at 7 days, since a reduction to a 24 hour AOT for the non-water-solid condition would result in an insignificant mean core damage frequency reduction of 6.21×10^{-8} per year.

In addition, the Final Safety Analysis Report Update for Farley Units 1 and 2, (Reference 4) details the administrative procedures and controls that are

employed to minimize the potential for the development of overpressure events. Normal operational procedures maximize the use of a steam bubble in the pressurizer during low pressure operation.

3.0 REVIEW OF NUREG DOCUMENTS

This section presents a review of the technical findings documented in NUREG 1326 (Reference 2) and NUREG/CR-5186 (Reference 3). NUREG-1326 presents the resolution of Generic Issue 94 and the detailed evaluation of the issue is found in NUREG/CR-5186.

Six specific alternatives were evaluated as proposed resolutions of GI-94:

1. No action (rejected),
2. Changes to Technical Specifications,
3. Require removal of all power to safety injection pumps and prohibit reactor coolant pump restart while in a water-solid condition,
4. Removal of RHR autoclosure interlock,
5. Require that the LTOP system be upgraded to a fully safety-grade system,
6. Require that water-solid operation be prohibited by providing for a steam or nitrogen bubble in the pressurizer at all times.

As long as the fracture resistance of the reactor pressure vessel material is relatively high, major overpressurization of the reactor coolant system while at low temperature is not expected to cause vessel failure. The fracture resistance of the reactor pressure vessel decreases with exposure to fast neutrons over the life of the vessel. If the fracture toughness of the vessel has been reduced sufficiently by neutron irradiation, low-temperature overpressure events could cause propagation of fairly small flaws that might exist near the inner surface. These assumed flaws might propagate into a crack through the vessel wall. This through wall crack (TWC) could lead to core damage with the failure to recover the reactor core or failure of the vessel to provide an additional barrier for fission product retention.

Reactor pressure vessel failure resulting from brittle fracture is defined as occurring on propagation of a through wall crack. Because overpressure events could lead to failure of the reactor pressure vessel, core damage frequency was assessed for each of these alternatives as the frequency of TWC.

A value/impact analysis was performed for each of these alternatives. The frequency of core damage (as the frequency of TWC) is calculated for each alternative. The reduction in core damage frequency (based on alternative 1 which is no changes), and the corresponding dose reduction is calculated for each alternative. A value/impact (V/I) ratio is calculated as the sum of industry plus NRC implementation costs divided by the dose reduction (person-rem) expected for each alternative.

The results of the value/impact (V/I) ratios reported in NUREG-1326 for the alternatives that would apply to Farley Units 1 and 2 are summarized as:

Alternative	TWC Frequency Reduction	Dose Reduction (person-rem)	V/I Ratio
1 - No Action	None		
2 - Tech Specs	2.89×10^{-6}	14,500	160
3 - SI lockout	1.07×10^{-6}	7,000	780
3 - RCP restart	0.21×10^{-6}	1,400	1,600
6 - Pressurizer Bubble			
(a)	3.25×10^{-6}	16,000	2,700
(b)	1.74×10^{-6}	9,300	4,600

Alternative 4 was not evaluated for plants that use the RHR relief valves for LTOP as no additional benefit would be obtained from this alternative.

Alternative 5 was evaluated only for plants that use the pressurizer PORVs for LTOP. The conclusion of the value impact analysis for these plants was that

upgrading LTOP is not expected to result in significant changes to hardware design or functioning.

Alternative 6 was evaluated for:

(a) pressurizer bubble with peak pressure less than 600 psi and
(b) pressurizer bubble, 10 percent chance of reaching 2500 psi.

Based on the NRC value/impact analyses, the specific action recommended in NUREG 1326 is to reduce the allowable outage time for a single LTOP channel when operating in either modes 5 or 6 from the current AOT of 7 days to an AOT of 8 hours[1] before remedial actions to depressurize and vent the reactor coolant system would be required. This would require a change in the current technical specifications.

The details of the NRC analyses performed to justify these changes are provided in the following section.

3.1 DETAILS OF NUREG ANALYSES

Table 1 lists the events used in the base case analyses reported in References 2 and 3. Although low pressure protection is required in modes 4, 5, and 6, the actual events mainly occurred in Mode 5 with reactor coolant temperature ranging from 80°F to 190°F. These events are categorized as either mass addition or energy addition events. The date of the event, pressure increase and cause of the event are given for each plant. Four main categories are listed for cause of the event: inadvertent SI (S), charging/letdown problems (C), reactor coolant pump (RCP) restart (P), and autoisolation of the RHR system (R). Table 2 presents the calculated peak (or hypothetical) pressures assuming failure of the LTOP system.

[1] Generic Letter 90-06 (Reference 1) specifies an AOT of 24 hours while NUREG 1326 recommends 8 hours.

TABLE 1

SUMMARY OF LER UPDATE FOR BASE CASE RISK ANALYSIS

Plant	Date	Pressure	Cause
		Westinghouse Plants	
Mass Addition			
Byron 1	3/18/85	Not Available	S - SI - operator error
Callaway 1	4/05/86	380 to 463	C - Excess charging, RCP seal injection valve failure
Farley 1	11/07/86	400 to 450	C - operator error on pressure control
Farley 2	10/15/83	700	C - Excess charging, loss of instrument air
Ginna	6/9/83	S.P.	C - Excess charging (safety injection testing)
North Anna 1	12/19/85	350 to 395	C - charging and letdown control
North Anna 1	3/29/81		S - safety injection
North Anna 1	9/14/84	350 to 410	P - RCP Restart
North Anna 2	5/23/83	387	S - safety injection (pressurizer bubble)
Salem 2	6/17/83	S.P.	S - safety Injection (small pressurizer bubble)
San Onofre 1	11/10/83	300 to 522	S - safety injection
Surry 1	7/2/81	S.P.	C - Inadvertent charging, FCV failure
Surry 1	6/1/84	325 to 412	C - Excess charging, operator error
Surry 1	5/12/85	350 to 410	C - Letdown decrease w/charging
Turkey Pt.4	11/28/81	310 to 1100	R - RCP restart, RHR autoisolation
Turkey Pt.4	11/29/81	310 to 1100	R - RCP restart, RHR autoisolation
Zion 1	9/11/84	450	C - Increased charging flow, operator error (pressurizer bubble)
Zion 2	1/3/86	400 to 435	R - RHR isolated letdown, with charging 190 gpm, electrical bus problem
Zion 2	1/3/86	400 to 435	C - Increased charging, operator error

KEY:

C = Charging/letdown events
P = RCP restart events
S = Inadvertent SI events
R = RHR autoisolation events
Q = Operator Error
S.P. = Upper bound pressure limited to LTOP setpoint, event mitigated

TABLE 1 (CONTINUED)

SUMMARY OF LER UPDATE FOR BASE CASE RISK ANALYSIS

Plant	Date	Pressure	Cause
		Westinghouse Plants	
Energy Addition			
Farley 1	11/15/86	400 to 450	P - RCP restart
Farley 2	10/15/83	480	P - RCP restart
North Anna 2	5/18/82	S.P.	P - RCP restart
North Anna 2	5/24/82	S.P.	P - RCP restart
Salem 2	2/15/84	325 to 350	P - RCP restart
Salem 2	3/29/85	325 to 380	P - RCP restart
Salem 2	3/30/85	325 to 380	P - RCP restart
Summer	5/6/85	450	P - RCP restart
		Combustion Engineering Plants	
Mass Addition			
Palisades	12/3/81	S.P.	S-safety injection
Calvert Cliff 1	4/26/83	425	Q-operator error, closes one PORV
Energy Addition			
Palisades	8/26/85	350 to 375	P-Reactor coolant pump restart

KEY:

C = Charging/letdown events
P = RCP restart events
S = Inadvertent SI events
R = RHR autoisolation events
Q = Operator Error
S.P. = Upper bound pressure limited to LTOP setpoint, event mitigated

TABLE 2

PEAK PRESSURES CALCULATED FOR EACH EVENT ASSUMING FAILURE OF LTOP

Plant	Date	Peak Pressure	Cause
	Westinghouse Plants		
Mass Addition			
Byron 1	3/18/85	450	S
Callaway 1	4/05/86	450	C
Farley 1	11/07/86	850	C
Farley 2	10/15/83	2500	C
Ginna	6/9/83	760	C
North Anna 1	12/19/85	467	C
North Anna 1	3/29/81	467	S
North Anna 1	9/14/84	467	P
North Anna 2	5/23/83	467	S
Salem 2	6/17/83	375	S
San Onofre 1	11/10/83	2500	S
Surry 1	7/2/81	600	C
Surry 1	6/1/84	600	C
Surry 1	5/12/85	600	C
Turkey Pt.4	11/28/81	1400	R
Turkey Pt.4	11/29/81	1400	R
Zion 1	9/11/84	450	C
Zion 2	1/3/86	450	C
Zion 2	1/3/86	2500	R
Energy Addition			
Farley 1	11/15/86	550	P
Farley 2	10/15/83	600	P
North Anna 2	5/18/82	467	P
North Anna 2	5/24/82	467	P
Salem 2	2/15/84	375	P
Salem 2	3/29/85	375	P
Salem 2	3/30/85	375	P
Summer	5/6/85	550	P
	Combustion Engineering Plants		
Mass Addition			
Palisades	12/3/81	850	S
Calvert Cliff 1	4/26/83	850	Q
Energy Addition			
Palisades	8/26/85	500	P

KEY
C = Charging/letdown events
P = RCP restart events
S = Inadvertent SI events
R = RHR autoisolation events
Q = Operator Error

Operating reactors w classified by overpressure protection system design as PORV plants or RHR SRV ,lants:

1. PORV plants: two PORVs, water-solid operation allowed (40 plants), and

2. RHR SRV plants: two SRVs in the RHR, water-solid operation allowed (15 plants).

Byron 1, Callaway 1, Farley Units 1 and 2, Kewaunee, Millstone 3, Summer, Wolf Creek, Yankee Rowe, ANO 2, Palo Verde Units 1 and 2, San Onofre Units 2 and 3 and Waterford 3 are classified as RHR SRV plants. The remaining Westinghouse and Combustion Engineering Plants were classified as PORV plants. Babcock and Wilcox plants were not specifically analyzed as these plants do not operate in a water-solid condition.

The mean core damage frequency per reactor year was determined for the two plant classifications (PORV plants and RHR SRV plants) as:

CDF = CFRQ x QLTOP x OPFR x MTWC.

where:

CDF = Mean core damage frequency

CFRQ = The challenge frequency per year (frequency of overpressure events per year).

QLTOP = The unavailability of LTOP (number of events with two channels out divided by number of events)

OPFR = The fraction of the overpressure spectrum determined for each of the plant classes.

MTWC = The mean through wall crack (TWC) probability over life for each of the overpressure spectrums.

The challenge frequency per year (CFRQ) was calculated as the number of overpressure events divided by the number of reactor years of operation from 1980 through 1986.

PORV plants = 23 events/244 reactor years = 0.094
RHR SRV plants = 7 events/56 reactor years = 0.125

The unavailability of LTOP (QLTOP) was calculated as the number of events with two channels out divided by the number of events.

PORV plants = 2 events with no channels available/23 demands = 0.087
RHR SRV plants = 1 event with no channels available/7 demands = 0.143
(The demands are assessed as the number of overpressure events)

The fraction of the overpressure spectrum (OPFR) was calculated from the data presented in Table 2. The overpressure events were categorized as mass addition or energy addition events for the two plant classifications:

	Number of Events			
	1400 to 2500 psi	850 to 1400 psi	600 to 850 psi	Up to 600 psi
PORV plants Mass Addition	2	2	3	10
Energy Addition	0	0	6	
Total	2	2	3	16
Fraction (of 23 demands)	0.09	0.09	0.13	0.69
RHR SRV plants Mass addition	1	0	1	2
Energy addition	0	0	0	3
Total	1	0	1	5
Fraction (of 7 demands)	0.14	0	0.14	0.72

The mean through wall crack (TWC) probability is calculated as an average vessel failure probability (VFP) over all plants in each group (PORV plants and RHR SRV plants). Plant-specific information on vessel chemistry and fluence was used to determine the reference temperature of the nil-ductility transition as a function of time between the present and end of life for each of the operating plants. The nil-ductility transition information was used to determine the fracture probability of each vessel as a function of time for peak pressures of 2500, 1500, and 700 psi with the VISA computer program (Reference 5). The plant-specific values of VPF as a function of time were determined, assuming a unit annual overpressurization frequency to a specific pressure. This normalized VPF is integrated from the present to the end of life for each plant. The resulting values represent the probability of vessel fracture between now and the end of life. The integrated vessel fracture probability over the remaining life of each plant is given in Table 7.2 of Reference 3 for peak pressure of 2500, 1500, and 700 psi.

The integrated fracture probabilities were summed over all plants and divided by the number of years remaining for those plants placed in each group:

PORV plants: 969 total years, 24 years/plant

RHR SRV plants: 452 total years, 30 years/plant.

The average vessel failure probability over all plants divided by the total number of years in each group is defined as the mean through wall crack (MTWC) or average vessel failure probability at each overpressure spectrum .

A total integrated fracture probability of 1.83×10^{-4} was determined and divided by 452 years to calculated a probability of 4.05×10^{-7} for pressures of 700 psi. The fracture probabilities at pressures of 850 and 600 psi were interpolated from a curve developed over all the RHR plants as 2.00×10^{-6} at 850 psi and 2.21×10^{-7} at 600 psi.

The MTWC as a function of the overpressure spectrum for RHR SRV plants was calculated as:

2500 psi	1400 psi	850 psi	600 psi
1.50×10^{-3}	-	2.00×10^{-6}	2.21×10^{-7}

Core damage frequency for the RHR SRV plants is calculated as:

	CFRQ	x	LTOP	x	OPFR	x	MTWC	=	CDF
@2500 psi	0.125	x	0.143	x	0.14	x	1.50×10^{-3}	=	3.75×10^{-6}
@1400 psi	0.125	x	0.143	x	0.0			=	0.0
@850psi	0.125	x	0.143	x	0.14	x	2.00×10^{-6}	=	5.01×10^{-9}
@600psi	0.125	x	0.143	x	0.72	x	2.21×10^{-7}	=	2.84×10^{-9}
Total									3.76×10^{-6}

This same calculation was performed for the PORV plants with a total core damage frequency of 3.04×10^{-6}. The average over both plant groups is 3.24×10^{-6}.

This calculation is for the base case, or alternative 1 (no changes). Each of the alternatives was re-evaluated by changing basic assumptions such as the number of challenges would decrease for alternatives 3 and the fraction of the overpressure spectrum would decrease for alternative 6.

Alternative 2 was evaluated by assuming that the unavailability of LTOP improved. The evaluation to support changing the technical specifications from 7 days to 8 hours, quoted from NUREG-1326, page 5-4 is:

> "The reduction in core damage frequency expected would be equivalent to the logical "and" of the estimated unavailability of the single channel system on which this evaluation is based. For the Group 1 (PORVs) plants the reduction is estimated to be from 0.087 to 0.087*0.087 or 0.0076. For the Group 2 (RHR SRVs) plants the reduction is estimated to

be from 0.143 to 0.143*0.143, or 0.02. The mean through wall crack frequency, or core damage frequency, is reduced from 3.24×10^{-6} per reactor year to 3.47×10^{-7} per reactor year."

This means that the unavailability of LTOP for the RHR SRV plants was changed from 0.143 to 0.143 x 0.143 = 0.02 and the core damage frequency recalculated (core damage at each overpressure spectrum was multiplied by 0.143).

This calculation does not address changing the time that an RHR valve could be out of service. This is addressed in more detail in the following section.

3.2 EVALUATION OF NRC ASSESSMENT

The conservatism identified in the assumptions used for the calculation of the mean core damage frequency are discussed in the following paragraphs.

1. The plants were placed in two categories: PORV plants and RHR SRV plants. The main difference in the two categories is the type of low pressure protection used. However, there is no indication that the major causes associated with most of the events (inadvertent SI, RCP start, letdown and charging control problems) are different for the two types of plants. Partitioning the plants so that the RHR SRV plants form a small sample size, increased the frequency of initiation events. There is variation in the number of events between plants but this variation does not depend on the overpressure protection systems. Therefore, the assumption that the frequency of challenges is greater for RHR SRV plants than for PORV plants is conservative.

2. The fraction of the overpressure spectrum causing peak pressures is calculated as only depending on the categorization of plants. However, the overpressure spectrum shows a strong correlation with the cause of the event as opposed to the type of overpressure protection.

3. The unavailability of the LTOP channels is highly conservative as this unavailability is calculated as the fraction of overpressure events with both RHR SRVs unavailable (1 overpressure event with the RHR SRVs unavailable divided by 7 overpressure events = 0.143). Such a conservative value can dominate the unavailability of a system so that the impact of changing an AOT is minimal.

   Define the unavailability of one RHR SRV as Q(SRV). The unavailability of both RHR SRVs is Q(SRV1) x Q(SRV2). The NUREG analysis then defines the unavailability (or failure) of one RHR SRV, Q(SRV), as 0.143 if the other valve is not allowed to be out of service for 7 days. The unavailability of both relief valves is then calculated as Q(SRV1) x Q(SRV2) = 0.143 x 0.143 under the assumption that the one valve could only be out of service for 8 hours. To create the unavailability of both RHR SRVs, assuming one valve is out of service for 7 days (the base case assumption), the unavailability (out for maintenance) of one valve Q(SRV1) must be assigned a value of 1.0 and the second valve assigned an unavailability (failed to operate) of Q(SRV2) = 0.143, or the unavailability of both valves is 1.0 x 0.143 = 0.143. This implies that the unavailability of the one RHR SRV must have been out of service over the entire time that the plant was at risk of the overpressure event.

4. The calculations given in the NUREG analyses implies that one RHR SRV is either out of service for the entire time period in modes 5 and 6 or that the valve is not out of service for the entire time period. Therefore, the NUREG analyses do not provide a probabilistic basis that the assumed reduction in core damage frequency is the result of changing the allowable outage time from 7 days to 8 hours.

4.0 PROBABILISTIC ANALYSIS

To be consistent with the analyses performed in References 2 and 3, the same basic probabilistic models as documented in Reference 2 are used for the probabilistic analysis. The data used to quantify these models are changed for this analysis:

1. The frequency of each initiator is calculated for Farley Units 1 and 2,

2. The overpressure spectrums as used in References 2 and 3 are changed so that these spectrums show the correlation with the initiating events,

3. The plant-specific integrated vessel fracture probabilities for Farley 1 and 2 are used for the through wall crack probability, instead of the average value over all plants,

4. The frequency of core damage (assessed as through wall cracking of the vessel) is calculated for each initiating event as a function of the overpressure spectrum, and

5. The unavailability of the LTOP channels is calculated so that the AOT time can be factored into the analysis.

To be consistent with the NUREG analyses, no credit is given for operator actions such as stopping the charging pump or opening pressurizer PORVs.

4.1 FREQUENCY OF INITIATING EVENTS

Only the overpressure events occurring at Westinghouse plants are included for the calculation of the frequency of the initiating events defined as challenges to the overpressure protection system. This is conservative as more overpressure events per year have occurred at Westinghouse plants than at Combustion Engineering plants.

The challenges to the overpressure protection system are subdivided into four major categories and are designated as the initiating events: charging/letdown events, RCP restart events, inadvertent SI events, and RHR isolation events. Table 3 lists the plant-specific challenge frequencies for each of the plants where:

Years = reactor years in data base for each plant (from 1980 through 1986)

No. = number of events in each category

Mean = mean frequency of each event.

The number of events (No.) and mean frequency are listed for each plant by the four challenge categories (which can be defined as the initiating events). The sum of all events and the mean over all events are listed under the heading All Events.

The sum of each challenge category (or each initiating event) over all plants is also listed. The mean frequency of each category is:

[ ] a,c

WCAP-11737 (Reference 6) lists the LTOP events for Westinghouse plants from April 13, 1979 (date when the NRC required LTOP protection) through October 26, 1987. These events were compared with the events listed for the base case risk study in NUREG-1326. [

.] a,c

TABLE 3
PLANT-SPECIFIC FREQUENCIES

a,c

Plant-specific events have occurred at Farley Units 1 and 2; 1 charging/letdown and 1 RCP restart event at each plant. These events occurred while the plants were operating in a water-solid condition. However, no additional events have occurred over the time period from the end of 1986 through December 31, 1990. Therefore, there are now 7 + 4 = 11 years of operation for Farley 1 and 5.6 + 4 = 9.6 years of operation for Farley 2, or a total of 20.6 years for both plants.

The frequency per year for each event is conservatively calculated as:

Frequency per year of a charging/letdown event is 2/20.6 = 9.7×10^{-2}/yr.

Frequency per year of a RCP restart event is also 2/20.6 = 9.7×10^{-2}/yr.

No overpressure events caused by either inadvertent SI or RHR isolation have occurred at either unit. Power is removed from the RHR isolation valves when the RCS temperature is below 180°F to prevent inadvertent isolation of the RHR SRVs. Therefore, RHR isolation is not expected and was not included in the NUREG analysis and is not included as an initiating event for this analysis.

Plant-specific frequencies are used for charging/letdown and RCP restart events and the generic frequency (over all plants) is used for inadvertent SI. The frequency of initiating events for this analysis are:

Charging/letdown events	= 9.7×10^{-2}/year
RCP restart	= 9.7×10^{-2}/year
Inadvertent SI	= 2.2×10^{-2}/year

These plant-specific frequencies and generic frequency are calculated for all overpressure events that occurred in modes 5 and 6. Two of the events listed in Table 3 occurred with a pressurizer bubble (SI at North Anna 2 on 5/23/83 and charging/letdown event at Zion 1 on 9/11/84). The remaining events occurred while the plants were operating in a water-solid condition. The frequency of an event occurring depends on the operating state (water solid or not water solid). The frequency of overpressure events is much higher when

the plant is operating in a water-solid condition. Therefore, each plant condition has a different frequency of occurring.

No overpressure events have occurred at either Farley Units 1 or 2 when the plants were not operating in a water-solid condition. Based on the data for all plants, one of five SI events (0.2) and one of the ten charging/letdown events (0.1) occurred while operating with a pressurizer bubble. Therefore, the frequency of overpressure events while operating with a pressurizer bubble can be adjusted for operating with a pressurizer bubble. The frequency of both charging/letdown and RCP restart events while operating in a water-solid condition are multiplied by 0.1 and the frequency of inadvertent SI is multiplied by 0.2. The frequency of these events while operating with a pressurizer bubble are estimated as:

[ ] a,c

4.2 PEAK OVERPRESSURE SPECTRUM

The peak (postulated) pressures calculated and listed in Table 2 can be partitioned by the type of initiating event. This is shown in Table 4 by each initiating event category. The fraction of each peak pressure calculated for each initiating event is shown in Table 5. This fraction is simply the number of events at each postulated peak pressure divided by the number of events in each initiating event category.

4.3 PROBABILITY OF TWC

The integrated vessel fracture probability [defined as the probability of through wall cracking (TWC) of the reactor vessel] over the remaining life of each plant is given in Table 7.1 of Reference 3. The total integrated fracture probabilities listed in this table for Farley Units 1 and 2 are:

TABLE 4

PEAK PRESSURES BY INITIATING EVENTS

Plant	Date	Peak Pressure	Cause
Byron 1	3/18/85	450	Inadvertent SI
North Anna 1	3/29/81	467	Inadvertent SI
North Anna 2	5/23/83	467	Inadvertent SI
Salem 2	6/17/83	375	Inadvertent SI
San Onofre 1	11/10/83	2500	Inadvertent SI
Callaway 1	4/05/86	450	Charging/letdown
Farley 1	11/07/86	850	Charging/letdown
Farley 2	10/15/83	2500	Charging/letdown
Ginna	6/9/83	750	Charging/letdown
North Anna 1	12/19/85	467	Charging/letdown
Surry 1	7/2/81	600	Charging/letdown
Surry 1	6/1/84	600	Charging/letdown
Surry 1	5/12/85	600	Charging/letdown
Zion 1	9/11/84	450	Charging/letdown
Zion 2	1/3/86	450	Charging/letdown
Turkey Pt.4	11/28/81	1400	RHR autoisolation
Turkey Pt.4	11/29/81	1400	RHR autoisolation
Zion 2	1/3/86	2500	RHR autoisolation
Farley 1	11/15/86	550	RCP restart
Farley 2	10/15/83	600	RCP restart
North Anna 2	5/18/82	467	RCP restart
North Anna 2	5/24/82	467	RCP restart
North Anna 1	9/14/84	467	RCP restart
Salem 2	2/15/84	375	RCP restart
Salem 2	3/29/85	375	RCP restart
Salem 2	3/30/85	375	RCP restart
Summer	5/6/85	550	RCP restart

TABLE 5

PEAK PRESSURE FRACTIONS BY INITIATING EVENT

a,c

[ ]a,c

The total integrated fracture probability for the two plants is divided by the remaining years of operation (65) to determine the yearly fracture probability (or mean through wall crack probability). Plant-specific values for pressure of 850 and 700 are interpolated from a curve constructed over all plants. The values estimated from this curve will be used in this analysis for pressures of 850 and 700 as they cannot be interpolated from this curve for Farley Units 1 and 2. However, the plant-specific probability for 2500 psi will be used for this analysis.

The integrated fracture probabilities per year listed above for each overpressure spectrum are used for the probabilities of TWC to calculate the frequency of core damage.

4.4 FREQUENCY OF CORE DAMAGE CALCULATION

The frequency of core damage is calculated by multiplying the frequency of the initiating event by the unavailability of LTOP, the fraction of the overpressure spectrum, and the probability of through wall cracking. The unavailability of the LTOP system does not depend on the type of initiating event and can be specified as X for this calculation. The frequency of each initiating event is given in Section 4.1, the overpressure spectrum as a function of each initiating event is given in Table 4, and the probability of TWC is given in Section 4.3.

The core damage frequency (CDF) is calculated by multiplying the frequency of the initiator (IE) times the unavailability of QLTOP (QLTOP = X) times the

fraction of the overpressure spectrum (OPFR) times the mean through wall crack probability (MTWC) for each initiating event:

$$CDF = IE \times X \times OPFR \times MTWC$$

The frequency of core damage is calculated for two plant conditions: the plant is water solid and the plant is not water solid.

<u>Frequency of Core Damage Calculation While the Plant is Water Solid</u>

The frequency of core damage is summed over all three initiating events as follows:

[ ] a,c

Frequency of Core Damage Calculation While the Plant is Not Water Solid

If the plants are not operating in a water-solid condition, the frequency of overpressure events is reduced. In addition, while operating with the pressurizer bubble established, reduced rates of pressurization are achieved under both mass and energy addition events. There is more time available for the operators to respond to such events. However, to be conservative, no other adjustments are made to the data (the same relative overpressure spectrum is used).

The frequency of core damage is summed over all three initiating events as follows:

a,c

The unavailability of LTOP (designated as X) is a variable in these equations and a range of core damage frequencies are calculated as a function of X (and AOTs for LTOP).

4.5 UNAVAILABILITY OF LTOP

The LTOP system consists of the RHR relief valves. There are two ways that this system could be unavailable during an overpressure event: the relief valves can fail to operate because of random failures or the relief valves may not be available because one system has been removed from service (either one train of RHR or its associated relief valve). The NUREG analysis considered the LTOP system to be unavailable during the overpressure event in 1983 at Farley Unit 2. The B train of RHR was valved out of service to work on the B RHR pump, therefore, train B RHR relief valve was not available to relieve pressure. A loss of instrument air caused the flow control valve to open and charging flow increased to maximum. Train A relief valve did not function at 450 psi. Two pins were missing from the disk assembly. The valve was repaired and returned to service. The failure of LTOP was a combination of a failure of one relief valve and the other valve out of service. Based on this event, the NUREG analysis calculated the unavailability of LTOP as 1/7 = 0.143. However, the unavailability of LTOP should be written as:

QLTOP = random failure of both valves or random failure of one valve and the other valve out of service.

The random failure of an RHR relief valve is not well documented. References 2 and 3 assumed that the random failure of both is 0.143. A review of a number of generic sources for failure of components is presented in Reference 7. A range of failures per demand are listed for failure of pressurizer safety relief valves to open on demand. The most conservative value (6.2×10^{-3}) is from NUREG/CR-1363 (Reference 8). Although these relief valves are not the same as the RHR relief valves, this value will be assigned as a lower bound value (essentially a 5 percent chance that the failure is this value).

A common method of assigning probabilities in probabilistic analyses is to assign an upper bound and lower bound value and calculate a mean (or average) value based on a log-normal distribution for the failures (this calculates a conservative average). An upper bound failure probability of 0.14 and a lower

bound failure probability of $6.2x10^{-3}$ are assigned for the failure of the relief valves. If the distribution of failures is assumed to be log-normal, then the mean value can be determined from the equation:

$$\text{Mean} = \text{median} \times \exp^{(\text{sigma})^2}/2$$

where:

Median = (u bound x lower bound)$^{.5}$ = (0.14 x $6.2x10^{-3}$)$^{.5}$ = 0.0295

Sigma = ln (upper bound/lower bound)/(2z) = 0.94744

z is obtained from the standard normal tables and z = 1.645 for a 90 percent confidence interval.

The mean is: 0.0295 x 1.5665 = 0.046

Therefore, a mean value of $5.0x10^{-2}$ is assigned for this study.

Without considering one RHR SRV to be out of service, the random failure of one RHR valve is assigned three values:

Q1 = 0.14 (upper bound)

Q1 = $5.0x10^{-2}$ (mean)

Q1 = $6.2x10^{-3}$ (lower bound).

The failure probability of two valves (with no valves out of service) can be due to either the random failure probability of both valves or the common cause failure probability of both valves. NUREG/CR-5186 estimates the common cause unavailability of two channels (PORVs and RHR SRVs) as 0.0024. However, they also estimate that a single channel failure was the result of 3 overpressurization events out of 30 challenges, or a failure of a single channel is 0.10. If there are two channels, channel A and channel B, with

independent random failure probabilities Q(A) and Q(B), respectively, then the common cause failure probability of the two channels is the joint failure probability of both which can be written as Q(A and B). In probability space, the failure of each valve can be calculated if the joint failure of both valves is known and the single failure of one of the valves is also known. The equation to determine the failure probability of one valve [say Q(A)] is:

$$Q(A) = Q(A \text{ and } B)/Q(B)$$

Using this equation, and the common mode unavailability of 0.0024 for the joint failure of both and the failure of Q(B) as 0.10, the failure of channel A would be 0.0024/0.10 = 0.024. If one channel fails with probability of 0.10, then the second channel would have a failure of 0.024. This is not logical as common mode failures are generally greater than multiple random failures of the valves. Therefore, a conservative factor of 0.1 is assigned for the common mode failure of the second valve, given that the first valve fails.

The total random failure probability of two valves is:

$$Q2 = (\text{random failure probability of one valve})^2 + \text{random failure probability of one valve} \times \text{common cause failure of second valve.}$$

Without considering one RHR SRV to be out of service, the failure of two RHR valves is assigned three values:

$Q2 = 3.36 \times 10^{-2}$ (upper bound)

$Q2 = 7.50 \times 10^{-3}$ (mean)

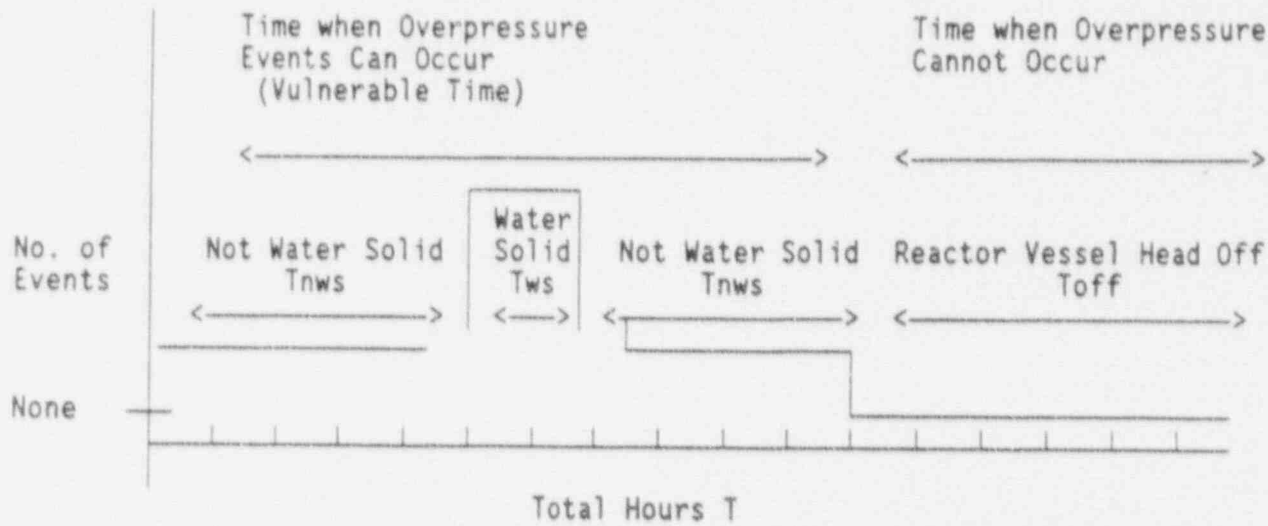
$Q2 = 6.58 \times 10^{-4}$ (lower bound).

These random failure probabilities calculated for the RHR safety relief valves are used in the equations developed to calculate the impact of AOTs on core damage frequency.

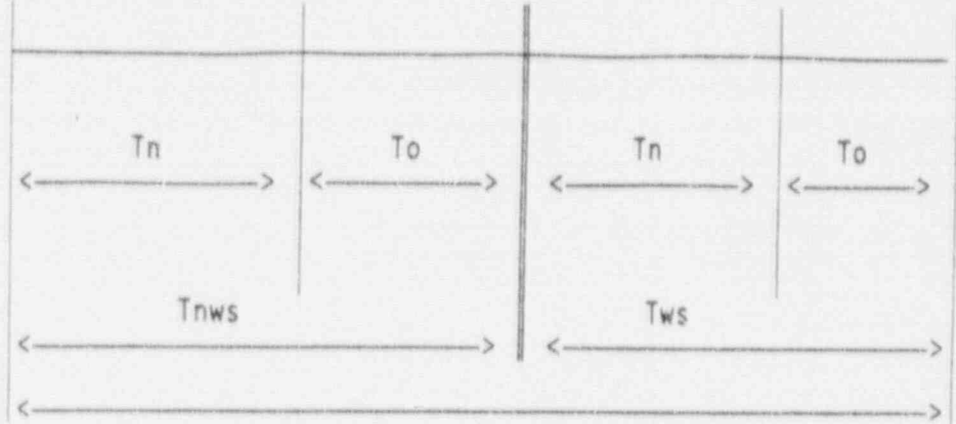
5.0 CALCULATION OF AOT TIMES

This section provides the methodology and equations to calculate the impact of changing AOT times.

Overpressure initiating events requiring the LTOP system are modeled for the time interval when the plants are operating in modes 5 and 6. Overpressure events are not possible while the reactor vessel head is removed. The majority of the overpressure events have occurred when the plants were operating in a water-solid condition. The frequency of overpressure events can be viewed as the probability per unit time. The vulnerable time for overpressure events is during that time period when plants are operating in a water-solid condition.



The total time during modes 5 and 6 is designated as T. This time can be partitioned into the vulnerable time for overpressure events and the time when the reactor vessel head is removed and overpressure events cannot occur. The vulnerable time for overpressure events can also be partitioned into the vulnerable time for overpressure events during water-solid operation (Tws) and the time when the plants are not operating in a water-solid condition (Tnws). Each of these times can be partitioned into the time when one RHR SRV (To) is out of service or none of the RHR SRVs (Tn) are out of service.



Total Shutdown Time T

For either time interval (Tnsw or Tws), while operating in the time interval Tn, both RHR SRVs would be available, and the unavailability Q2 of the LTOP system would be the random failure of SRVs (QRV1 and QRV2) or the common cause failure of both SRVs (QRV x CC):

Q2 = QRV1 x QRV2 + QRV x CC.

For either time interval (Tnsw or Tws), while operating in the time interval To, one SRV could be out of service for a time To. During the time interval To, one SRV would be available and the unavailability Q1 of the LTOP system would be the random failure of the remaining SRV or:

Q1 = QRV

The time when the plant is water solid and the time when the plant is not water solid can be partitioned into two independent time intervals (Tnws and Tws). In either operating state the times can be partition as:

Tnws = (Tn + To)nws or

Tws = (Tn + To)ws

If these equations are divided by the total time in each operating state:

$$\frac{Tn}{T} + \frac{To}{T}$$

Then:

$\frac{Tn}{T}$ = the fraction of time (or probability) when both SRVs are available (or that no SRVs are out of service). The unavailability of the SRVs is Q2.

$\frac{To}{T}$ = the fraction of time (or probability) when only one SRV is available (and one SRV is out of service). The unavailability of the SRVs is Q1.

In these equations, T is either Tnws or Tws, depending on the operating state. Substituting the appropriate time for each time interval, the unavailability (QLTOP) of the SRVs is:

Plant is not operating in water-solid condition:

$$QLTOP = \frac{Tn}{Tnws} \times Q2 + \frac{To}{Tnws} \times Q1 \qquad \text{(Equation 1)}$$

Plant is operating in water-solid condition:

$$QLTOP = \frac{Tn}{Tws} \times Q2 + \frac{To}{Tws} \times Q1 \qquad \text{(Equation 2)}$$

Before these equations can be evaluated, an estimate of the vulnerable time in the various modes of operation is required.

The estimated time spent in modes 1, 3, 5, and 6 (from January 1980 through September 1990) was obtained by combining information from References 9, 10, and 11. Following plant trips, the plants generally will remain in mode 3. Maintenance outages are times spent in mode 5 (not associated with refueling). Mode 4 is a transitional state and the time spent in this mode is combined with the time spent in mode 5. The refueling outage time includes time spent in modes 4, 5, and 6. The outage times are listed in the Appendix. A summary of the outage times are as follows:

a,c

The reactor vessel head is on for an average of three days, eight hours while shutting down and four days, two hours when returning to power (178 hours). The time that the plants spend in a water-solid condition is the vulnerable time period for overpressure events. Farley Units 1 and 2 are expected to be in a water-solid condition for 24 hours when the plant is either shutting down for a refueling outage or returning from a refueling outage (24 hours each way, Reference 11). An average of 50 hours is assigned for water-solid operation during refueling.

It is possible that water-solid operation would also be required for some mode 5 maintenance outages and the plants would only operate in a water-solid condition for 50 hours. Therefore, the vulnerable time (Tws) is 100 hours when the plants are operating in a water-solid operation. The plants do not operate in a water-solid condition for the remaining time of 327 hours (Tnws).

As discussed in Section 4.0, core damage frequency (CDF) was calculated for the two operating plant conditions as:

$$CDFnws \text{ (not water solid)} = X \times 3.32\times10^{-6}$$

$$CDFws \text{ (water solid)} = X \times 2.53\times10^{-5}$$

where X is the unavailability QLTOP evaluated with:

$$QLTOP = \frac{Tn}{Tnws} \times Q2 + \frac{To}{Tnws} \times Q1 \qquad \text{(Equation 1 not water solid)}$$

$$QLTOP = \frac{Tn}{Tws} \times Q2 + \frac{To}{Tws} \times Q1 \qquad \text{(Equation 2 water solid)}$$

$Q1 = 0.14$ (upper bound)
$Q2 = 3.36\times10^{-2}$ (upper bound)

$Q1 = 6.20\times10^{-3}$ (lower bound)
$Q2 = 6.58\times10^{-4}$ (lower bound)

Three random failure probabilities are used for the random failure of the RHR SRVs [unavailability of 1 RHR SRV (Q1) and unavailability of 2 RHR SRVs (Q2)]:

$Q1 = 5.0\times10^{-2}$ (mean)
$Q2 = 7.50\times10^{-3}$ (mean)

The total core damage frequency over the entire shutdown time period is the sum of each contribution for each case:

$$CDF = CDFnws + CDFws$$

These equations were evaluated for the following AOTs (in hours):

Case	Tnws	Tws	Reason
1	1300	100	This is upper bound case, equivalent to base case assumption in NUREG analyses (RHR SRVs are out over entire time period).
2	168	100	This is the case allowing RHR SRVs to be out of service for 7 days if not water solid or water solid (current technical specifications). The time Tws is 100 hours.
3	168	24	This is the case allowing RHR SRVs to be out of service for 7 days if not water solid and 24 hours while water solid.
4	24	24	This is the case to evaluate the technical specification change to allow RHR SRVs to be out for only 24 hours over entire shutdown period.
5	0	0	This is lower bound case, equivalent to assumption in NUREG analyses that RHR SRVs are never out of service (or assumed to be out 8 hours).

6.0 RESULTS

The results of these calculations detailed in Section 5 are shown in Table 6 for each of the AOT cases. The core damage frequency calculation is given for both conditions: not water solid and water solid. The total core CDF for each case is given as well as the CDF reduction and the percent reduction calculated by changing the AOT from the current technical specifications of 7 days (not water solid and water solid) to:

7 days (not water solid) and 24 hours (water solid),
24 hours (not water solid) and 24 hours (water solid), and
0 hours (RHR SRVs not out of service).

The results shown in these tables can be summarized to show the expected CDF reduction by reducing the allowable outage time from 7 days (168 hours) to 24 hours over the entire time in modes 5 and 6 when the reactor vessel head is not removed as follows:

Value	AOT Time	Total CDF	CDF Reduction	Percent Change
Mean	168 hours	5.45×10^{-7}		
Mean	24 hours	4.83×10^{-7}	6.21×10^{-8}	4.6
Upper Bound	168 hours	1.79×10^{-6}		
Upper Bound	24 hours	1.63×10^{-6}	1.55×10^{-7}	4.1
Lower Bound	168 hours	6.19×10^{-8}		
Lower Bound	24 hours	5.38×10^{-8}	8.09×10^{-9}	4.8

The largest reduction in CDF results by changing the AOT from 7 days (or out the entire time the plant is operating in a water-solid condition) to an AOT of 24 hours when the plant is operating in a water-solid condition. The reduction in the mean CDF frequency per year is 8.17×10^{-7} (a 60 percent reduction).

TABLE 6

SUMMARY OF RESULTS
AOT WITH ONE SRV OUT

RCS CONDITION				TOTAL	REDUCTION	
NOT WATER SOLID		WATER SOLID				
AOT	CDF	AOT	CDF	CDF	CDF	PERCENT
MEAN						
327	1.66×10^{-7}	100	1.26×10^{-6}	1.43×10^{-6}		
168	9.73×10^{-8}	100	1.26×10^{-6}	1.36×10^{-6}		
168	9.73×10^{-8}	24	4.48×10^{-7}	5.45×10^{-7}	8.17×10^{-7}	60.0
24	3.52×10^{-8}	24	4.48×10^{-7}	4.83×10^{-7}	8.79×10^{-7}	64.5
0	2.49×10^{-8}	0	1.90×10^{-7}	2.15×10^{-7}	1.15×10^{-6}	84.3
UPPER BOUND						
327	4.64×10^{-7}	100	3.54×10^{-6}	4.00×10^{-6}		
168	2.93×10^{-7}	100	3.54×10^{-6}	3.83×10^{-6}		
168	2.93×10^{-7}	24	1.50×10^{-6}	1.79×10^{-6}	2.04×10^{-6}	53.4
24	1.37×10^{-7}	24	1.50×10^{-6}	1.63×10^{-6}	2.20×10^{-6}	57.4
0	1.11×10^{-7}	0	8.50×10^{-7}	9.61×10^{-7}	2.87×10^{-6}	74.9
LOWER BOUND						
327	2.06×10^{-8}	100	1.57×10^{-7}	1.77×10^{-7}		
168	1.16×10^{-8}	100	1.57×10^{-7}	1.68×10^{-7}		
168	1.16×10^{-8}	24	5.03×10^{-8}	6.19×10^{-8}	1.06×10^{-7}	63.2
24	3.53×10^{-9}	24	5.03×10^{-8}	5.38×10^{-8}	1.15×10^{-7}	68.1
0	2.18×10^{-9}	24	1.66×10^{-8}	1.88×10^{-8}	1.50×10^{-7}	88.8

By changing the AOT from 7 days to 24 hours for the entire vulnerable time period (when the reactor vessel head is not removed) results in an additional reduction in mean CDF frequency per year of 6.21×10^{-8} (a 4.6 percent reduction). This is not a significant reduction in CDF.

The percent change in the mean CDF from 7 days (not water solid) and 24 hours (water solid) to 24 hours over the entire vulnerable time period is calculated as:

$$\text{CDF reduction} = 5.45 \times 10^{-7} - 4.83 \times 10^{-7} = 6.21 \times 10^{-8}$$

$$\text{Percent Change} = (5.45 \times 10^{-7} - 4.83 \times 10^{-7})/1.36 \times 10^{-6} \times 100 = 4.6\%$$

The results of this study can be compared to the results obtained in the NUREG studies. The calculated core damage frequency present in NUREG-1326 is 3.24×10^{-6} per year (over all plants). By implementing alternative 2 (Technical specifications to only allow an AOT time of 8 hours) the core damage frequency is estimated to be reduced by 2.89×10^{-6} per year. This is a 77 percent reduction in core damage frequency. If these results are compared to the core damage frequency for the upper bound case, the total CDF is 4.00×10^{-6}. This is assuming that the RHR SRVs are out of service over the entire shutdown mode of operation. By allowing a seven day AOT if the RCS is not water solid and an AOT of 24 hours if the RCS is water solid, the CDF is reduced to 1.79×10^{-6} which is a 53.4 percent reduction in CDF.

It should be noted that this analysis is conservative as the assumption is made that the RHR SRVs are out of service for 24 hours while the plants are operating in a water solid condition.

7.0 CONCLUSIONS

Based on the results obtained from this probabilistic analysis of core damage frequency, it was concluded that the AOT for water-solid operation should be reduced from the current technical specification value of 7 days to 24 hours. The current technical specification value of 7 days for all other low temperature conditions will be retained at 7 days, since a reduction to a 24 hour AOT for this non-water-solid condition would result in an insignificant mean core damage frequency reduction of 6.21×10^{-8} per year.

8.0 REFERENCES

1. U. S. Nuclear Regulatory Commission, Generic Letter 90-06, "Resolution of Generic Issue 70," June 25, 1990.

2. U. S. Nuclear Regulatory Commission, NUREG-1326, "Regulatory Analysis for the Resolution of Generic Issue 94, Additional Low-Temperature Overpressure Protection for Light-Water Reactors", December 1989.

3. U. S. Nuclear Regulatory Commission, NUREG/CR-5186, "Value/Impact Analysis of Generic Issue 94, Additional Low-Temperature Overpressure Protection for Light-Water Reactors", November 1988.

4. Joseph M. Farley Nuclear Plant Unit 1 and Unit 2, Final Safety Analysis Report Update, Volume 10, Chapter 5.2, Rev 4, July 1986.

5. U. S. Nuclear Regulatory Commission, NUREG/CR-4483, "VISA II--A Computer Code for Predicting the Probability of Reactor Pressure Vessel Failure," March 1986.

6. Westinghouse Electric Company, WCAP-11737 "Low Temperature Overpressuriztion", March 1989.

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8. U. S. Nuclear Regulatory Commission, NUREG/CR-1363, "Data Summaries of Licensee Event Reports of Valves at U. S. Commercial Nuclear Power Plants, June 1980.

9 Joseph M. Farley Nuclear Plant Unit 1 and Unit 2, Individual Plant Examinations Draft Data Notebook, (from Alabama Power letter NA-89-1567 - Plant Outage Histories, July 18, 1989).

10. Joseph M. Farley Nuclear Plant Unit 1 and Unit 2, Individual Plant Examinations Data Draft Initiating Events Notebook.

11. Alabama Power letter NL-90-3108, Attachment ES90-1892, December 13, 1990.

APPENDIX

This Appendix contains the plant-specific outage times for Farley Units 1 and 2.

OUTAGE TIMES FOR FARLEY UNITS 1 AND 2

The outage times for maintenance, test, refueling and plant changes were combined with the outage times following reactor trips. These outage times are from January 1, 1980 to September 30, 1990 for Unit 1 and from January 1, 1982 to September 30, 1990 for Unit 2. These outage times were obtained (and combined) from References 10 and 11. The outage times are shown in Table A-1.

The time spent in mode 4 is not specifically determined as this is a transitional mode but is combined with the time specified as mode 5; only the times in mode 3, mode 5 maintenance outages and refueling are shown on the table. The time spent in refueling includes time in modes 4, 5, and 6.

The RHR system is not in operation in mode 3 and the plant is operating with a pressurizer bubble. Therefore, no analysis is performed for mode 3 operation.

The sum of outages for each of the plants was determined and combined as one outage time in each of the modes. The percent of time (per year) was then calculated for each mode to determine the estimated time spent in each mode per year. These sums are shown in Table A-2.

TABLE A-1
OUTAGE AND POWER HOURS

a,c

TABLE A-1 (Continued)
OUTAGE AND POWER HOURS

a,c

TABLE A-1 (Continued)
OUTAGE AND POWER HOURS

a,c

TABLE A-1 (Continued)
OUTAGE AND POWER HOURS

a,c

TABLE A-1 (Continued)
OUTAGE AND POWER HOURS

a,c

TABLE A-1 (Continued)
OUTAGE AND POWER HOURS

[ ]a,c

TABLE A-2

SUMMARY OF OUTAGE TIMES

a,c