Safety Balance Assessment

for

Elimination of Reactor Coolant System

Main Loop Pipe Break Protective Devices

Crystal River 3 Generating Plant

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Tables

I. Introduction

This report presents a safety balance assessment of the consequences of eliminating LOCA protective devices currently employed in the design of the Crystal River 3 Electric Generating Plant (CR-3). These devices were designed to mitigate the dynamic effects associated with postulated breaks in the reactor coolant system (RCS) main loop piping. This assessment uses methods as described in the Lawrence Livermore National Laboratory (LLNL) PWR generic Value-Impact analysis, Reference 3. This safety balance evaluation is based on CR-3 plant specific data from Reference 2 and, where necessary, generic PWR data from Reference 3.

The CR-3 plant was included in the 'Probability of Pipe Failure' evaluation of the ten Babcock & Wilcox (B&W) plants in Reference 2. The results of the generic PWR evaluation (Reference 3) and the B&W plants evaluation (Reference 2) were extended to the CR-3 plant. The basic assumptions used in extending these evaluations to CR-3 are the same as those addressed in the LLNL evaluation (Appendix B of Reference 3).

As a result, this safety balance assessment includes the avoided costs developed in Reference 3. This safety balance assessment is intended to represent the CR-3 plant specific impact associated with the exclusion of pipe breaks from PWR primary reactor coolant loop piping.

The evaluation is performed in terms of public health and occupational accident risk avoidance attributable to the protection provided for dynamic effects associated with postulated breaks in the RCS primary loop versus the reduction in Occupational Radiation Exposure (ORE) resulting from a decision not to use such protection.

The following basic assumptions were made in extending the results of the Reference 2 and 3 evaluations to the CR-3 plant.

1. One plant is considered (CR-3), compared to 10 and 85 in the Reference 2 and 3 evaluations respectively.
2. The value-impact summary for the CR-3 plant is based on a total remaining lifetime of 23 years. The Reference 3 evaluation was based on a total remaining lifetime of 2922 plant-years (py) and an average remaining lifetime of 34.4 years per plant.
3. The CR-3 plant has four reactor coolant loops. All plants in the Reference 2 evaluation have four reactor coolant pumps. The Reference 3 evaluation accounts for variations in the number of reactor coolant loops among PWR plants. No corrections are necessary for the Reference 3 data.

The man-rem savings are presented in tabular form and listed as best, high and low values. These represent the range of values expected at CR-3; however, conservatisms are included in the analysis of the ORE which tend to lower the estimated man-rem savings over the entire range of values. These are explained as follows:

A. Conservatively low estimates of man-rem exposures are used when calculating the total exposure due to the removal and reinstallation of pipe whip restraints for access to perform inservice inspection (ISI). Only man-rem savings associated with not reinstalling pipe whip restraints are analyzed. The Reference 3 evaluation assumed that it takes two persons, two shifts to remove each pipe whip restraint and another two shifts for reinstallation. The Reference 3 evaluation assumed exposure rates in the vicinity of the reactor coolant piping in the range of 0.02 to 0.2 rem/hr. This corresponds to an expected dose of 1.3 to 12.8 rem per restraint per ISI. Twenty-five man-rem per restraint per ISI was used as a maximum based on industry experience. The 1, 10 and 25 man-rem per piping restraint per ISI values used in this

analysis are conservative when compared to the expected values based on expected dose rates near reactor coolant piping at CR-3.

B. Conservatively low estimates per Reference 3 of the increased work efficiency due to improved access for maintenance (based on fewer interferences with the pipe whip restraints and pump support structural members) are used. An increased work efficiency of 5 per cent is used for reactor coolant pump (RCP) piping and steam generator maintenance. Actual increases in work efficiency are expected to be higher when considering the instruments, equipment and scaffolding that must be manipulated during these activities and the difficulty of working in protective clothing.

II. Safety Balance Assessment Summary and Conclusions

A summary of the results of the safety balance is shown below. The dose best estimates support the request to not require consideration of the dynamic effects of RCS main loop pipe breaks in the CR-3 design basis.

Table 1

Safety Balance Value Summary

Value (man-rem)	Best Estimate	High Estimate	Low Estimate
Public Health	-2.2E-6	-1.4E-1	0.0
Occupational Exposure (Accidental)	-1.7E-5	-7.8E-2	0.0
Occupational Exposure (Routine)			
a) Inservice Inspection	30	75	3
b) Routine Maintenance	4	7	2
c) Special Maintenance	7	14	1
d) Removal of non-ISI Pipe Whip Restraints	-50	-125	-5
e) Pump Snubber Maintenance	2340	2340	2340
Total Quantified Value	2331	2311	2341

III. Development of Safety Balance

A. Risk Avoidance Atttributable to Protection from Dynamic Effects Associated with Pipe Breaks

The changes in public health risk and core melt frequency associated with this request are taken from a generic PWR probabilistic study performed by LLNL for the NRC (Reference 3). The changes in risk in Reference 3 were evaluated in a manner similar to that used in the value-impact analysis of Westinghouse A-2 Owners Group Plants addressed in NRC Generic Letter 84-04 (Reference 1). Dose estimates derived in Reference 3 are conservative and exceed the median probabilities of CR-3.

1. Core Melt Frequency Increase

The best estimate of the total increase in core melt frequency for not providing protection against dynamic effects associated with pipe breaks is calculated based on the methodogy used in the Reference 3 LLNL evaluations. The best estimate probability of a direct DEGB for the CR-3 plant has not been calculated. LLNL performed a review of reactor coolant loop stress information on ten Babcock & Wilcox plants and found by inference that the direct DEGB probability of B&W plants should be bounded by the Westinghouse A-2 evaluation. Therefore, the best-estimate probability (90-th percentile value) of a direct DEGB for Westinghouse plants east of the Rocky Mountains will be used for CR-3.

Summing the contributions of pipe breaks inside and outside the reactor cavity yields the best estimate increase in core melt frequency per Reference 3 as:

Best Estimate Increase = 3.6E-11/py

The median probability of an indirect DEGB for Westinghouse plants east of Rocky Mountains was taken as the upper bound increase in the core melt frequency for the CR-3 plant. This is conservative for several reasons as discussed in Reference 3 and based on the median probability of indirect DEGB calculated in Reference 2 for CR-3 (6.1E-11/py). However, the plant specific value has not been recalculated based on the proposed resupport scheme for the RCS reactor coolant pumps. Thus, the conservative PWR value from Reference 3 will be used. The high estimate of core melt frequency increase is:

High Estimate Increase = 1.0E-7/py

A low estimate increase of 0.0 is used for CR-3.

The resulting total core melt frequency increase estimates are as follows:

	Increase in Core Melt Frequency (events/py)
Best Estimate	3.6E-11
High Estimate	1.0E-7
Low Estimate	0.0

Note that this probabilistic analysis of the potential for increased risk to the public health due to the increase in core melt frequency demonstrates that there is no credible increase in the risk to public health. Because of the uncertainties in the core melt frequency estimates (References 4 and 6), the increase

in core melt frequency is not statistically significant enough to establish a credible difference in the core melt frequency and hence the estimated added risk to public health.

2. Public Health

The nominal estimate of added risk to public health for all PWR plants was estimated in Reference 3 to be 9.7E-8 man-rem/plant year (py). For CR-3 a best estimate risk is:

Best Estimate Risk = 9.7E-8 man-rem/py

The high estimate of added risk to public health for all PWR plants was estimated in Reference 3 to be 6.1E-3 man-rem/plant year (py). The CR-3 high estimate risk is:

High Estimate Risk = 6.1E-3 man-rem/py

The lower estimate risk is assumed to be 0.0.

Multiplying each of the risk calculations by the CR-3 remaining number of years of expected plant life, 23 py, results in a CR-3 public risk increase of:

	Total Added Risk (man-rem)
Best Estimate	2.2E-6
High Estimate	1.4E-1
Low Estimate	0.0

Note that this analysis results in an increase (albeit extremely small) in public risk. However, as stated in Reference 3, there are arguments that the proposed

request could conceivably reduce public risk at CR-3. Removal of unnecessary restraints would;

1) increase the reliability of the piping systems by increasing the effectiveness of in-service inspection
2) reduce the potential for restricted thermal expansion pipe movement (locked snubbers) during routine operation.

B. Reduction in Occupational Radiation Exposure (ORE) Resulting From a Decision Not to Use Protection Against Dynamic Effects Associated With Pipe Breaks

1. Occupational Exposure (Accidental)

The increased occupational exposure from accidents is estimated as the product of the change in total core melt frequency and the occupational exposure likely to occur in the event of a major accident. The nominal change in core melt frequency was estimated as 3.6E-11 events/py. The occupational exposure in the event of a major accident has two components. The first is the immediate exposure to the personnel on site during the span of the event and its short term control. The second is the long term exposure associated with cleanup and recovery from the accident. The incremental occupation exposure due to an accident is calculated as follows:

$$D_{TOA} = NTD_{OA}; \; D_{OA} = F(D_{IO} + D_{LTO})$$

where;

D_{TOA} = Total accidental occupational dose

N = Number of affected facilities = 1

T = Average remaining lifetime = 23 yrs.

D_{OA} = Accidental occupational dose per plant year

F = Change in core melt frequency

D_{IO} = Immediate occupational dose

D_{LTO} = Long term occupational dose

Table 2

Occupational Exposure Due to Accidents

	Change in Core Melt Frequency	Immediate Occupational Dose	Long Term Occupational Dose	Total Occupational Exposure
	(events/py)	(man-rem/ event)	(man-rem/ event)	(man-rem)
Best Estimate	3.6E-11	1.0E+3	2.0E+4	1.7E-5
High Estimate	1.0E-7	4.0E+3	3.0E+4	7.8E-2
Low Estimate	0.0	0.0	1.0E+4	0.0

2. Occupational Exposure - Routine

Reduction in routine Occupational Radiation Exposure (ORE) resulting from a decision not to use "Protection Against Dynamic Effects Associated with Pipe Breaks." This section gives reductions due to eliminating the need for RCS pipe whip restraints and unnecessary pump snubbers.

a. Inservice Inspection (ISI)

A review of the CR-3 design reveals that 63 per cent of the RCS pipe whip restraints are located such that there is sufficient access for performing ISI of the RCS piping welds. However, with all the RCS pipe whip restraints and supporting structural members removed, improved access is provided for ISI of the steam generator feedwater headers and nozzles including removal and reinstallation of the insulation from the steam generators.

However, since the nominal reduction in ORE due to improved access for ISI is minimal, its value is not calculated.

The avoided exposure for not reinstalling pipe whip restraints removed for ISI is assumed to be one-half that required for removal and reinstallation. This results in the following ORE estimates:

Best Estimate = 30 man-rem
High Estimate = 75 man-rem
Low Estimate = 3 man-rem

b. Routine Maintenance

During power operation there is no identifiable

routine maintenance activity which must be performed inside the steam generator compartment or primary shield wall where the RCS pipe whip restraints and RCS pump snubbers are located.

Routine maintenance will be required on the CR-3 reactor coolant pump (RCP) seals. As outlined in NUREG-0933 (Reference 5) each RCP is expected to require maintenance every outage, resulting in a total nominal exposure of 7 man-rem per maintenance activity. Reference 3 assumes that maintenance efficiency will increase by 5 percent. Thus, the nominal reduction in ORE due to improved access for four pumps is:

reduction in ORE = (0.05)(7 man-rem/py)(23py)(1/2)

= 4 man-rem (assumes 1 maintenance activity every two years)

The high and low estimates are based on engineering judgement that the total exposures will be 12 man-rem and 4 man-rem per RCP maintenance activity, respectively. The resulting estimates are:

Best Estimate = 4 man-rem
High Estimate = 7 man-rem
Low Estimate = 2 man-rem

c. Special Maintenance

Improvement in access may be realized for the following special maintenance activities if the pipe whip restraints and unnecessary pump snubbers are eliminated from CR-3:

1) Scaffolding near pumps
2) Scaffolding around steam generator

3) Hot leg insulation removal/replacement
4) RTD replacement
5) Steam generator main feedwater header and nozzles

The annual exposure from performing special maintenance activities was estimated to be 12.1 man-rem averaged over a 10 year period for a PWR per Reference 3. On this basis the reductions in ORE due to improved access for special maintenance is:

Best Estimate = 0.05(12.1)(23)(1/2)man-rem
= 7 man-rem
High Estimate = 14 man-rem
Low Estimate = 1 man-rem

A nominal increase in work efficiency of 5 per cent is assumed. The high and low estimates are based on 10 percent and 1 percent, respectively.

d. For Crystal River 3 the estimates must be reduced by the ORE incurred during removal of those restraints not normally removed for ISI. Assuming that the 5 additional restraints would be removed and that per-restraint exposure for removal would be one-half that for removal and reinstallation results in the following ORE estimates:

Best Estimate = 50 man-rem
High Estimate = 125 man-rem
Low Estimate = 5 man-rem

e. Reactor Coolant Pump Snubber Maintenance
The current FPC technical specification requires only visual inspections for the large bore pump

snubbers. Due to proposed changes in NRC mandated requirements, each snubber will now be removed a number of times during the life of the plant for inspection and maintenance due to service life of fluid and seals. The costs associated with snubber removal, inspection, testing, maintenance, refurbishing, and/or replacement is substantial.

A study was performed by Babcock & Wilcox to determine the exposure hours associated with snubber inspection & maintainance. To determine total exposure, a typical value of 15 mr/hr for time inside containment was used. To obtain exposure time, a factor of 0.6 was applied to the total site manhours to account for direct exposure time. FPC also performed a study to determine the total site manhours required. This resulted in a best estimate of:

Exposure = (0.6)(105,560 total site manhours)
(15 mr/hr)

Exposure = 950 man-rem

This exposure would occur for three outages (1987, 1994 and 2001) at CR-3. Thus, the total exposure is (3 x 950 man-rem) or 2850 man-rem.

The CR-3 study also determined the exposure resulting from removal of unnecessary pump restraints. This option would replace the eight large bore snubbers per pump with a support arrangement consisting of three restraints. The new restraints would be rigid link bars (where practical) and new smaller snubbers. These new restraints would be added during Refuel VI.

The site manhours associated with this task is 46,760 in 1987, 5000 in 1994 and 5000 in 2001 for a total of 56,760. The same methodology was again used to determine the best estimate of exposure time.

Exposure = 0.6 (56,760)(15 man-rem)

Exposure = 510 man-rem

Therefore, the nominal reduction in ORE due to not reinstalling and refurbishing the unneeded snubbers is a substantial 2340 man-rem. The estimated reduction resulted from a detailed evaluation of snubber maintenance costs and can be considered very reliable. The 2340 man-rem was, therefore, retained in both the high and low estimates.

IV. References

1. U.S. NRC Generic Letter 84-04 "Safety Evaluation of Westinghouse Topical Reports Dealing with Elimination of Postulated Pipe Breaks in PWR Primary Main Loops" dated February 1, 1984.

2. Ravindra, M. K. et. al., Probability of Pipe Failure in the Reactor Coolant Loop of Babcock and Wilcox PWR Plants, Volume 2: Guillotine Break Indirectly Induced by Earthquakes, Lawrence Livermore National Laboratory, UCRL-53644, NUREG/CR-4290, Vol. 2, (1985).

3. G. S. Holman and C. K. Chou, Assessment of Value Impact Associated with the Elimination of Postulated Pipe Ruptures from the Design Basis for Nuclear Power Plants, Lawrence Livermore National Laboratory, Report UCID 20397, dated March 29, 1985.

4. WASH 1400 (NUREG-75/014) "Reactor Safety Study," October 1975

5. Emrit, R., et. al., A Prioritization of Generic Safety Issues, NRC, DE84 900640, NUREG--0933 (1983).

6. Andrews, W. B., et al., Guidelines for Nuclear Power Plant Safety Issue Prioritization Information Development, Pacific Northwest Laboratory, Report PNL-4297, NUREG/CR-2800 (1983).