Safety Balance for Elimination of Reactor Coolant System Main Loop Pipe Break Protective Devices

South Texas Project

Units 1 and 2

Prepared for

Houston Lighting and Power Company

by

Bechtel Energy Corporation

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W2B/COM2/b

Safety Balance for the Elimination Of Reactor Coolant System Main Loop Pipe Break Protective Devices

I. Introduction

This report presents a safety balance evaluation of the consequences of eliminating the protective devices currently employed in the design of the South Texas Project Units 1 and 2 (STP) to mitigate dynamic effects associated with postulated breaks in the reactor coolant system (RCS) main loop piping. This assessment uses methods suggested in the "Leak Before Break Value-Impact Analysis" attached to the Nuclear Regulatory Commission's (NRC) Generic Letter 84-04 (Reference 1). Plant specific data and the generic data developed in Reference 1, and other public documents are used to perform the safety balance evaluation for STP. 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 man-rem savings is presented in tabular form and listed as nominal, lower and upper values. These represent the range of values expected at STP; however, there are conservatisms 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. The man-rem savings associated with not installing jet impingement barriers independent of the pipe whip restraints are not included in this analysis. The elimination of jet impingement barriers and associated supporting structures will result in increased work efficiency due to improved access for maintenance. These factors are not considered in this analysis. Only man-rem savings associated with not installing pipe whip restraints are analyzed.

B. 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). It is assumed that it takes two persons, two shifts to remove each pipe whip restraint and another two shifts for reinstallation. The STP expected exposure rates in the vicinity of the reactor coolant piping are in the range of 0.02 to 0.2 rem/hr. This corresponds to an expected dose of between 1.3 and 12.8 man-rem per restraint per ISI. 40 man-rem per restraint per ISI is used as a maximum based on industry experience. The rounded-off values of 1.0, 10 and 40 man-rem per restraint per ISI used in this analysis represent low, expected, and upper bound estimates of the radiation exposure, respectively.

C. Increased work efficiency due to improved access for maintenance (based on fewer interferences with the pipe whip restraints and supporting structural members) was not considered in this evaluation. The reduction in interferences allows platform locations to be optimized to increase efficiency. Typical maintenance operations which are beneficially affected include steam generator sludge lancing and tube plugging, reactor coolant pump seal replacement and pipe whip restraint gap verification.

II. Safety Balance Assessment Summary and Conclusions

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

Value (man-rem)	Nominal Estimate	Lower Estimate	Upper Estimate
Public Health (Accidental)[(a)]	-1.0	0	-8
Occupational Exposure (Accidental)(a)	-0.3	0	-5
Occupational Exposure (Operational)			
Inservice Inspection	171	20	656
Total Quantified Value	170	20	643

(a) The estimates shown here use negative values to represent a decrease in man-rem savings. The upper and lower estimates are transposed from the values presented in section III.A.

III. Development of Safety Balance

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

1. Public Health

Dose estimates derived in Reference 1 are found to be conservative and bound the results calculated for STP for the following reasons:

a. Reference 1 assumed a uniform population density of 340 people per square-mile around the reactor site and a 50-mile release radius model. The expected average population density at the STP site is 45.6 people per square mile in the year 2000. A total of 97.7 percent of that population is expected to live between 10 to 50 miles away from the plant. The corresponding numbers for the year 2030 are 68.4 people per square mile and 97.3 percent (Reference 3, Section 2.2).

b. Based on the significantly lower population density around STP, the off-site doses calculated in Reference 1 are considered to envelope the STP doses. (The STP whole body population dose to 50 miles is 10.5 man-rem per Ref. 3). The increased population density at the end of plant life does not significantly change the population doses and is still well within the bounds of Reference 1. The nominal estimate of added risk to public health for plants that use a two-loop configuration was estimated to be 0.006 man-rem/plant year (py) in Reference 1. For STP this number is adjusted to account for the four loop design. This results in a nominal risk of:

$$\text{Risk} = \frac{4}{2} \times 0.006 = 0.012 \text{ man-rem/py.}$$

Upper estimate risk calculations are made using procedures similar to those of the nominal estimates. No corrections are necessary for the number calculated in reference 1 because this frequency is per plant year and not based on the number of loops. The upper estimate risk is:

$$\text{Risk} = 0.1 \text{ man-rem/py.}$$

The lower estimate is assumed to be 0.

Multiplying each of the risk calculations by the number of years of expected plant life (2 plants x 40 yr = 80 py) results in the STP public risk increase of:

	Total Added Risk (man-rem)
Nominal Estimate	1.0
Upper Estimate	8.0
Lower Estimate	0

The nominal estimate from Reference 1 of the total increase in core melt frequency for not providing protection against dynamic effects associated with pipe breaks is used and adjusted for the larger number of loops in the STP design. This results in a core melt frequency increase of:

Core melt frequency increase = $\frac{4}{2} \times 1 \times 10^{-7} = 2 \times 10^{-7}$

The upper estimate of core melt frequency increase of 2×10^{-6}/py (Reference 1) is applicable for the STP analysis. No correction for the number of loops is necessary because this number is per plant year. A lower estimate of 0 is used for STP. In summary, core melt frequency increase estimates are as follows:

	Increase in Core Melt Frequency (events/py)
Nominal Estimate	2×10^{-7}
Upper Estimate	2×10^{-6}
Lower Estimate	0

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 6 and 7), 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. Occupational Exposure Accidental

The increased occupational exposure from accidents is estimated to be 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 2×10^{-7} events/py. The occupational exposure in the event of a major accident has two components. The first is the "immediate" exposure to the personnel onsite during the span of the event and the time necessary to achieve short term control. The second is the longer term exposure associated with the cleanup and recovery from the accident.

The total avoided occupational exposure is calculated as follows:

$$D_{TOA} = NTD_{OA}$$

$$D_{OA} = P(D_{IO} + D_{LTO})$$

Where

D_{TOA} = Total avoided occupational exposure

N = Number of affected facilities = 2

T = Average plant lifetime = 40 yrs.

D_{OA} = Avoided occupational dose per reactor year

P = Change in core melt frequency

D_{IO} = "Immediate" occupational exposure

D_{LTO} = Long-term occupational exposure

Results of the calculations are shown below. Uncertainties are conservatively propagated by the use of extremes (e.g., upper bound D_{IO} + upper bound D_{LTO}).

	Increase in Core Melt Frequency (events/Plant-yr)	Immediate[(a)] Occupational Exposure (man-rem/event)	Long Term[(a)] Occupational Exposure (man-rem/event)	Total[(b)] Avoided occupational Exposure (man-rem)
Nominal Estimate	2×10^{-7}	1×10^{3}	2×10^{4}	0.3
Upper Estimate	2×10^{-6}	4×10^{3}	3×10^{4}	5
Lower Estimate	0	0	1×10^{4}	0

(a) Based on cleanup and decommissioning estimates contained in Reference 2.

(b) These values represent increases in exposure due to accident conditions.

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 - Operational

a. Inservice Inspection (ISI)

Review of the STP design indicates that the RCS pipe whip restraints are located such that there is sufficient access to the RCS piping welds for performing ISI with the exception of the crossover leg pipe whip restraints. Interferences posed by the crossover leg pipe whip restraints during ISI cause a minimum of 25% of the restraints to be removed to facilitate crossover leg piping weld ISI four times over the life of the plant (once every 10 years). Industry experience shows that the radiation exposure associated with removal and reinstallation of the crossover leg pipe whip restraints ranges from 1 man-rem to 40 man-rem per restraint per ISI with a nominal value of 10 man-rem per restraint per ISI. Since in the STP design there are eight pipe whip restraints per unit which require removal, the nominal reduction in ORE for not installing these pipe whip restraints is estimated as follows:

$$\text{Reduction in ORE} = 0.25 \times 2 \text{ units} \times 8 \frac{\text{restraints}}{\text{unit/ISI,}} \times 10 \frac{\text{man-rem}}{\text{restraint}} \times 4 \frac{\text{ISI}}{\text{Plant life}}$$

$$= 160 \text{ man-rem}$$

Upper estimate at 40 man-rem/restraint = 640 man-rem
Lower estimate at 1.0 man-rem/restraint = 16 man-rem

In addition, with all the RCS pipe whip restraints and supporting structural members removed, improved access is provided for ISI of the following:

1) Reactor coolant piping

2) Steam generator welds (lower shell)

3) Reactor coolant pumps

The annual radiation exposure for performing the above ISI is estimated to be 14.25 man-rem averaged over a 10 year period (Reference 4, Table 12.4-15). It is further estimated that removal of the pipe whip restraints will provide improved access and increase the inspection efficiency by 1 percent. Therefore, the nominal reduction in ORE due to improved access for ISI is:

Reduction in ORE = 2 units x 0.01 x 14.25 man-rem/yr x 40 yr.

= 11.4 man-rem

The upper and lower exposure estimates are assumed to be 20 man-rem/yr and 5 man-rem/yr, respectively, giving ORE reductions of 16 man-rem and 4 man-rem.

The total reduction in ORE for operational occupational exposure due to ISI if the pipe whip restraints are not installed is:

	Occupation Radiation Exposure (man-rem)
Nominal Estimate	171
Upper Estimate	656
Lower Estimate	20

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. NUREG/CR-2601, "Technology, Safety and Costs of Decommissioning Reference Light Water Reactors Following Postulated Accidents," November 1982.

3. STP Environmental Report, sections 2.2 and 7.1

4. STP FSAR, section 12.4

5. NUREG 0933, "A Prioritization of Generic Safety Issues," 3/31/83

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

7. German Risk Study, NRC Translation 729, May 1980

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Response to NRC Request for Additional Information Concerning Leak-Before-Break Analysis

South Texas Project

Units 1 and 2

Houston Lighting & Power Company

Enclosure A: WCAP 10559 - Technical Bases for Eliminatoring Large Primary Loop Pipe Rupture as the Structural Design Basis for the South Texas Project Units 1 and 2 (Proprietary)

Enclosure B: WCAP 10560 - Technical Bases for Eliminating Large Primary Loop Pipe Rupture as the Structural Design Basis for the South Texas Project Units 1 and 2 (Proprietary)

Response to Request for Additional Information
Concerning Leak-Before-Break Analysis
for South Texas Project Units 1 and 2

In response to your request for additional information concerning leak-before-break analysis contained in the letter from G. W. Knighton (NRC) to J. H. Goldberg dated April 20, 1984, Houston Lighting and Power Company (HL&P) is enclosing a Westinghouse Report, "Technical Bases for Eliminating Large Primary Loop Pipe Rupture as the Structural Design Basis for the South Texas Project", May 1984, as Enclosure A. Because of the proprietary nature of this report, Enclosure A has been provided only to the addressee, to Mr. J. T. Collins and to Mr. V. Nerses of the NRC. A non-proprietary version is included as Enclosure B and has been provided to others on the attached distribution list.

A cross reference, indicating where in Enclosure A the responses to the questions contained in the April 20, 1984 letter are located, follows:

<u>Question 251.8</u> Update Section 1.0 of References (1) and (2) (Enclosures (A) and (B) to the applicant's letter of Sept. 28, 1983) to include appropriate references to NRC Generic Letter 84-04 dated February 1, 1984, "Safety Evaluation of Westinghouse Topical Reports Dealing With Elimination of Postulated Pipe Breaks in PWR Primary Main Loops."

<u>Response</u> See Enclosure A, page 1-3

<u>Question 251.9</u> The first sentence of Section 2.0 of References (1) and (2) refers to an operating history of over 400 reactor years. For clarification, specify a number of facilities in service for various periods of time to indicate that 400 reactor years of operation includes units that have had long histories of operating experience.

<u>Response</u> See Enclosure A page 2-1

<u>Question 251.10</u> Based upon operating experience, provide a conclusion in Section 2.0 of References (1) and (2) regarding the susceptibility of the reactor coolant primary loop piping, or portions thereof, to failure from the effects of corrosion (e.g., intergranular stress corrosion cracking) water hammer or fatigue (low and high cycle).

<u>Response</u> See Enclosure A page 2-1 through 2-3

<u>Question 251.11</u> Incorporate by reference WCAP-10456 (proprietary) and WCAP-10457 (non-proprietary) entitled, "The Effects of Thermal Aging on the Structural Integrity of Cast Stainless Steel Piping for the Westinghouse Steam Supply

System," dated December 1983. Identify the contents of these reports that are directly applicable to References (1) and (2).

Response — See Enclosure A page 4-4

Question 251.12 — Although the information in References (1) and (2) generally comply with the staff criteria currently developed, a sensitivity study is requested to address the adequacy of certain aspects of the fracture mechanics analytical model. In regard to Section 3.3 of References (1) and (2), what is the critical crack size under Level D loads and for the 7.5 inch through wall flaw, identify the margin, in terms of load, to unstable propagation. Further, justify both the maximum J and maximum crack extension via J-R data and/or tests. With the objective to assure adequate material toughness under adverse loadings, an elastic-plastic fracture mechanics analysis of the pipe test in WCAP-10456 incorporating analyses of decrease in bending versus crack size may provide the information requested.

Response — See Enclosure A Section 7.0

The response to Question 251.12 provides plant specific information regarding margin to unstable crack propagation. Information addressing the second part of Question 251.12 regarding justification of maximum J and maximum crack extension is under development by Westinghouse and has been discussed by Westinghouse with the NRC staff. Upon completion of the Westinghouse work, HL&P will determine whether any additional submittal on the STP docket is required.

Because Enclosure A contains information proprietary to Westinghouse Electric Corporation, the attached affidavit signed by Westinghouse management sets forth the basis on which the information may be withheld from public disclosure by the NRC in accordance with the requirements of 10 CFR 2.790 (b)(1). The affidavit addresses with specificity the considerations of 10 CFR 2.790 (b)(4). Correspondence with respect to the proprietary aspects of the affidavit and Application for Withholding should reference CAW-84-49, and should be addressed to R. A. Wiesemann, Manager, Regulatory and Legislative Affairs, Westinghouse Electric Corporation, P. O. Box 355, Pittsburgh, Pennsylvania 15230.