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NUCLEAR REGULATORY COMMISSION
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SEP 28 1992

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MEMORANDUM FOR: Frank P. Gillespie, Director
Program Management, Policy Development, and Analysis Branch
Office of Nuclear Reactor Regulation

FROM: C. J. Heltemes, Jr., Deputy Director
Office of Nuclear Regulatory Research

SUBJECT: GI-163, MULTIPLE STEAM GENERATOR TUBE LEAKAGE

Reference: NRC Integrated Program for the Resolution of Unresolved Safety Issues A-3, A-4, and A-5 Regarding Steam Generator Tube Integrity, Final Report, NUREG-0844, September 1988

License amendments have recently been granted to Westinghouse plant owners for operation during one refueling cycle with modified technical specifications for steam generator tube leakage limits and steam generator tube repair limits. One such amendment [No. 178, Facility Operating License No. NPF-1] prompted a DPO and the opening of the subject new generic issue. The prioritization for this issue is now being peer-reviewed. The preliminary priority ranking is HIGH. Although not normally done for HIGH priority issues, NRR is being provided a concurrent copy of this prioritization for review and comment because of recent NRR experience in evaluating licensee submittals with respect to steam generator technical specifications.

NRR is also invited to assist in defining the scope and priority of issue resolution efforts by identifying the steam generator tube integrity issues which would be most useful to address on a priority basis to best suit the needs of NRR. The reference contains previous discussion, conclusions, and recommendations with respect to many such issues. Issues not discussed in the reference might also be important in plant operation with revised steam generator tube repair and leakage limits (e.g. a PWR ATWS). Severe accident issues such as steam generator tube heat-up from hot degraded core gases might also need consideration. Finally, a number of regulations and supporting documents might be impacted and require revision if the technical specification revisions are to be permanent, rather than limited to the current single-cycle. NRR comments in this regard are also invited.

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Please assign an NRR contact to work with the task manager in coordinating activities bearing on this issue. The Task Manager for this issue is Dr. Gary Burdick, 492-3950.

C. J. Heltemes, Jr., Deputy Director
Office of Nuclear Regulatory Research

Enclosure: As stated

Enclosure 1

Prioritization Evaluation

Issue 163: Multiple Steam Generator Tube Leakage

Issue 163: Multiple Steam Generator Tube Leakage

DESCRIPTION

Background

This issue was opened (Reference 1) in response to a Differing Professional Opinion (DPO) with supplemental information contained in Reference 2. The NRC is currently considering changes in steam generator technical specifications and monitoring. The changes being considered include utility proposals to permit tubes with up to 100 percent through-wall cracks to remain in service (Ref. 3, Pg. 326). From the same reference, "The staff has approved higher depth-based limits, up to 64 percent, for specific types of flaws at specific plants." One utility proposal combines eddy current testing augmented by more restrictive leakage limits and inspection programs. The DPO states that, "While considerable research will be required to define a new plugging limit and change the SRP, the result will not increase plant safety. The basic problem is with the NDE procedures and their inability to predict tube degradation and leakage." At least one other NRC staff member has also expressed concern with industry proposals for alternate tube plugging criteria allowing operation of steam generators with known through-wall tube cracks (Reference 4). That reference, among other issues, also questions the capability of eddy current testing in steam generator tube flaw detection and sizing.

License amendments have been granted for plants to operate with eddy current limits on steam generator tube testing augmented by more stringent primary to secondary leak criteria. One such plant, the plant specifically mentioned in the DPO, was selected as the Base Case for this prioritization (a four-loop W PWR).

The concern stated in the DPO is "...a Main Steam Line Break outside containment could trigger multiple steam generator tube failures which would then result in a core melt because of depletion of coolant inventory." The concern applies to PWR operation with multiple steam generator tube through-wall-cracks or other tube degradations. This statement of concern defines the issue prioritized herein.

Safety Significance

A PWR main steam line break (MSLB) concurrent with steam generator tube rupture (SGTR) could result in a containment bypass loss of coolant accident. If the SGTR involved enough tubes and if the MSLB was not isolable, core damage would ensue upon refueling water storage tank (RWST) depletion. The ruptured tubes and open steam line would then provide a direct path to the atmosphere for fission products from the deteriorating reactor core. For prioritization purposes, the MSLB will be assumed to occur in the steam line segment between containment and the first MSIV. BWRs are not affected by this issue.

Possible Solutions

One solution is repair of tubes or replacement of large portions of seriously degraded steam generators.

A second solution is to install a feature to prevent depressurization, given a MSLB; i.e. the steam line between containment and the first main steam isolation valve (MSIV) should be provided with a guard pipe assembly to maintain secondary system pressure, given a MSLB. The MSIVs should be of sufficiently high availability for the case where MSLBs occur downstream of the MSIVs.

A third solution is to mitigate the seriousness of loss of RCS inventory by providing borated water sources sufficient to maintain core cooling long enough beyond RWST depletion to initiate and maintain cooling by the decay heat removal (DHR) system.

A fourth solution might be to add MSIVs inside containment. However, due to the potential for installation and maintainability problems for such valves, this option was not pursued.

PRIORITY DETERMINATION

Frequency Estimate

Reference 5 estimated the probability of a MSLB in the segment between containment and the MSIV as 1.0E-3 per reactor year (RY) with an assumed

worst-case peak differential of 2600 psid. An unpublished estimate has also been made, using an expert judgement approach, by a team assembled under the "Evaluation and Improvement of NDE Reliability for Inservice Inspection of Light Water Reactors" program sponsored by the NRC. The team median estimate (to be published) was 1.7E-4 per reactor year per segment, for a leak volume greater than 50 gallons per minute, which gives 6.8E-4 per reactor year for a four loop plant. Due to lack of data, Reference 6 could only estimate an upper bound of 7.0E-3 per RY for breaks in secondary system piping greater than three inches in diameter. For this exercise, 6.8E-4 per RY per four loop unit will be assumed as the initiating event frequency.

The Base Case W plant identified 3025 flawed tubes of which 2597 were repaired by either plugging or sleeving during 1991. The balance of 428 were approved to be left in service (Ref. 7) under new steam generator (SG) tube eddy current testing criteria and more stringent primary to secondary leakage limits. In the following evaluation, the 428 tubes are assumed to have flaw depth greater than 40% through-wall based on evidence from pulled tubes (Ref. 7).

From Reference 8, there are 87, 67, 101, and 173 flawed tubes in the Base Case SGs A, B, C, and D respectively. It is assumed herein that flaw depth is uniformly distributed through the tube walls. Under this assumption, there are approximately 15, 11, 17, and 29 flaws in the respective SGs with depths between N and N+10 percent through wall for N=40, 50, ---, 90 assuming only one flaw per tube. [There are probably several flaws per tube, particularly at the support plates. It should also be recognized that many more flawed

tubes could exist at the Base Case Plant than the 428 in question. These conditions are both due to the inherent weaknesses in eddy current testing. Reference 4 describes an exercise where several different techniques were used to detect laboratory produced flaws. Detection probability ranged from 0.2 to 0.75 with an average of 0.5. Flaw sizing ability was poor.]

Figure 1 is a plot of curves illustrating normalized burst pressure vs. flaw length for a family of flaw penetration curves [cf. Reference 9, Uniform Thinning Equation]. Po is the burst pressure of a virgin (unflawed) tube, P is the burst pressure of a flawed tube, t is virgin tube thickness, and a is flaw depth. Po is approximately 10E+4 psid, thus 0 26 on the P/Po scale corresponds to approximately 2600 psid. From the figure, tubes with an a/t ratio of 0.75 or more would leak or break at a pressure of 2600 psid or less depending upon flaw length. Again assuming uniform distribution of flaw depth, there are about 36, 28, 42, and 72 flaws, in the respective SGs with depths 0.75 or more. No attempt will be made to estimate flaw length. It is, however, noted from the figure that flaw length can be quite small for flaws greater than 0.9 through-wall to exhibit leakage at pressure much less than 2600 psid. Actual rupture of ten or more tubes at 2600 psid appears to be a reasonable assumption. A leak-before-break (LBB) concept is inherent in the Safety Evaluation (Ref. 7) for the Base Case plant licensing amendment which is the subject of the DPO. Reference 11 cautions that, "It is not recommended that the LBB concept be applied to piping systems that are susceptible to IGSCC [intergranular stress corrosion cracking] or to water hammer and to piping subject to erosion, such as portions of the steam extraction systems."

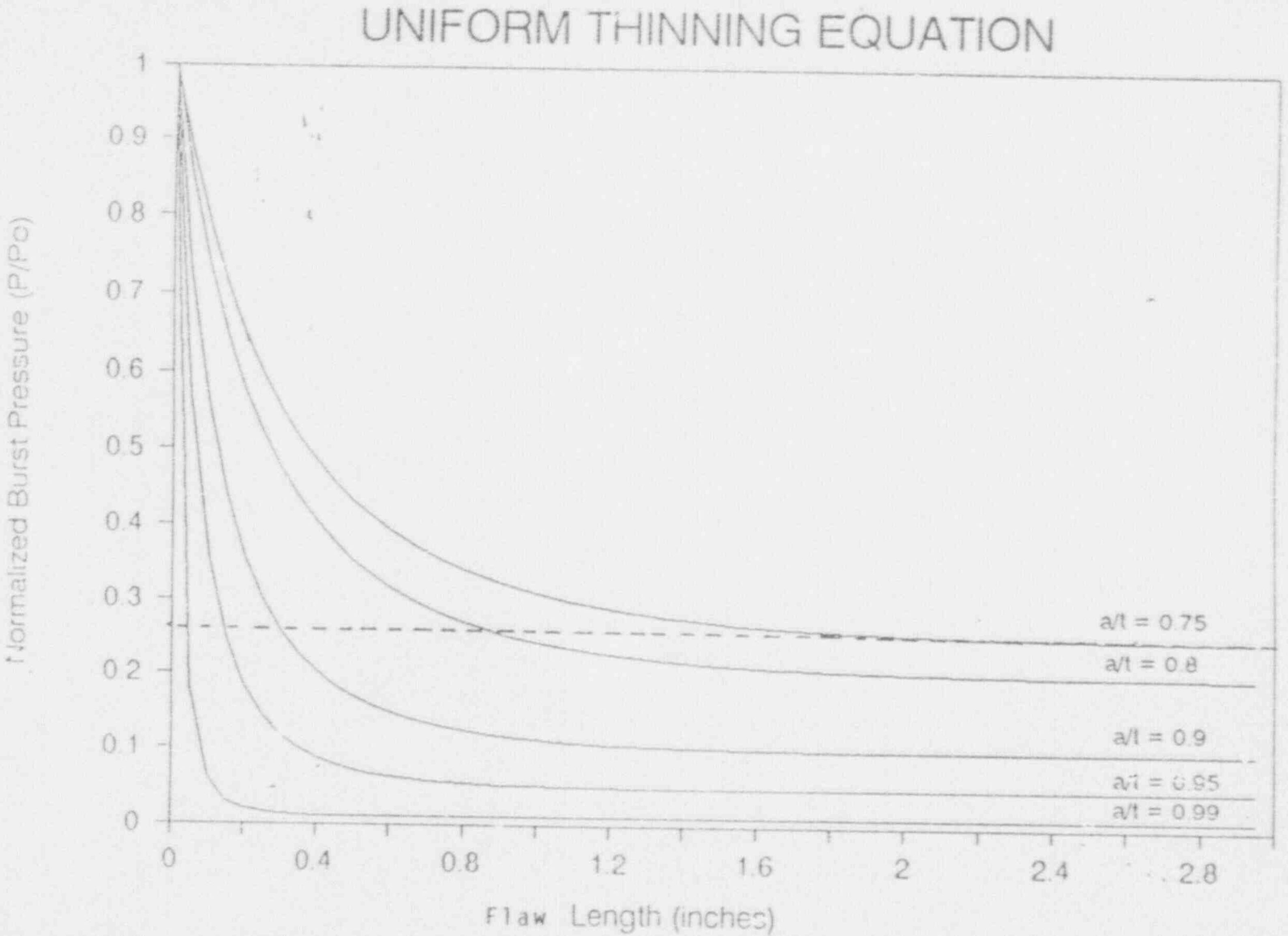
UNIFORM THINNING EQUATION
Normalized Burst Pressure (P/Po)
1
0.9
0.8
0.7
0.6
0.5
0.4
0.3
0.2
0.1
0
a/t = 0.75
a/t = 0.8
a/t = 0.9
a/t = 0.95
a/t = 0.99
0
0.4
0.8
1.2
1.6
2
2.4
2.8
Flaw Length (inches)

Figure 1.

For a large main steam line break with more than 10 steam generator tube ruptures, Sequence 8c of Reference 5 has an assigned probability of 0.5 for failure to depressurize the RCS to atmospheric pressure before RWST depletion. The estimated core damage frequency for the case under consideration is then F(MSLB) X P(more than 10 tube rupt s) X P(failure to depressurize) or (6.8E-4) X (1.0) X (0.5) =3.4E-4 events per reactor year. The 0.5 for failure to depressurize is not elaborated upon (other than being time-based) in Reference 5 and could be conservative.

Reference 12 states that, "Initiatives that are directed to prevention of core damage should also assess the potential for early failure or bypass of the containment in order not to exceed the large release frequency of 1.0E-6/reactor year." Although Reference 12 is not yet implemented, with a frequency goal of 1.0E-6 per reactor year for this containment bypass sequence, virtually all of the 3.4E-4 events per reactor year constitutes the frequency reduction potential [(3.4E-4) - (1.0E-6) =3.39E-4].

Consequence Estimate

Reference 13 (cf. Table 7, Page 3-24) assigns a PWR-1 release category to an MSLB with multiple (>10) SGTRs. The whole body dose per PWR-1 event is 5.4E+6 person-rem. The offsite public dose consequence is thus (3.4E-4) x (5.4E+6) =1.8E+3 person-rem per reactor year. Although the license amendment granted for the Base Case unit was for cycle 14, it is assumed that extensions will be granted through the remaining four years of unit operation (not yet approved by the NRC) currently desired by the owners. With an assumed 4 years of

remaining unit operation, the total risk reduction per four loop unit is 7.2E+3 person-rem, with 5.4E+3 and 3.6E+3 person-rem for three and two loop units respectively. Currently, two four loop units and one three loop unit have been granted license amendments allowing operation with flawed SG tubes. The total risk reduction potential from these plants is 2.0E+4 person-rem, assuming 4 years remaining unit life for each.

Assuming all currently operating PWRs will be in similar circumstances to the case considered herein, and using number of loops per unit information from Reference 14, the total risk reduction available is: (31) (7.2E+3) + (14) (5.4E+3) + (30) (3.6E+3) =4.1E+5 person-rem.

Cost Estimate

Industry Cost:

From information in Reference 15, replacing the steam generator would cost $150M with other estimates exceeding $200M. Plugging and sleeving the 428 flawed tubes at the same ratios experienced at the Base Case plant up to Refueling cycle 14 (71% plugging, 29% sleeving) would cost (Ref. 15) about $2M. This action would probably cause the plant power rating to be lowered over the remaining plant life. Assuming a 10% reduction in power over four years with a 0.65 industry average plant availability and approximate per day Base Case plant replacement power cost of $774K (Information from Ref. 16) the net cost, including plugging and sleeving, is about $76M. Thus, neither of these options appear to be economically attractive.

Another possible solution, involves installing main steam line guard pipes. This appears to be economically more attractive due to short main steam line segments between the containment wall and the first MSIV. This is also a preventive action. Which would meet the Reference 12 CDF goal for a containment bypass sequence since the calculation of F(MSLB) x P(>10 tube ruptures) x P (failure to depressurize) yields $(6.8E-4)^2 \times (1.0) \times (0.5) = 2.3E-7 < 1.0E-6$ events per reactor year.

It was assumed that four MSIVs would require replacement in addition to guard pipe installation because of new large flanges required on each MSIV. Cost estimates were taken from Tables 4.1 and 4.2, Abstract 2.1.9, Reference 16, after noting from Reference 17 that guard pipes need not be designed for greater pressure and temperature than the enclosed pipe. Allowing room for the enclosed pipe and insulation, 36 inch diameter carbon steel pipe was selected. The nearest satisfactory pipe thickness tabulated was 2 inch. A 4% inflation factor of 1.22 was used for the years 1988 - 1992 inclusive.

o Carbon Steel Pipe (36 in diam., 2 inch thick)

Number of Linear Feet:	44
Unit Installation Labor:	316 MH/LF (man-hours/linear feet)
Labor Rate X Overhead Factor:	\$24.90/MH X 1.59 =\$39.60/MH
Factory Cost:	\$2914/LF
Site Materials:	\$818/LF

Carbon Steel Pipe Cost:

1.22 X 44 X [(316 X \$39.60) + \$818 + \$2914] =\$872,064

- Carbon Steel Stop Valves (30 inch)

  Number of Valves: 4

  Unit Installation Labor 333MH/Valve

  Labor Rate X Overhead Factor \$29.90/MH X 1.59 =\$39.60/MH

  Factory Cost: \$34897/Valve

  Carbon Steel Stop Valves Cost:

  1.22 X 4 [(333 X \$39.60) + \$34897] = \$234,649

  Total Guard Pipe Installation Costs =\$1,106,713 per four loop plant.

NRC Cost

Development of a solution is estimated to cost three and one half person-years of contractor labor at a cost of \$125K/person-year to investigate solutions to this issue and related problems [cf. Other Considerations]. NRC Project Manager cost would be about \$41K or 1000 hours (Ref. 16). Assuming the results indicate a rulemaking activity for operation with degraded SG tubes, an additional 1000 hours of staff time would be required. Tech. Spec. changes, supporting Reg. guides, and other regulatory costs for the 75 reactors involved total \$1.6M (also Ref. 16). Total NRC cost is estimated to be \$2.1M.

Total Cost

Assuming the current population of 75 PWRs for eventual operation similar to the Base Case unit wi、 modified SG tube plugging requirements and

primary/secondary leakag. riteria, total industry and NRC costs are about \$65M for the total number steam lines involved. That figure is obtained by scaling the \$1.11M per plant appropriately for 2 and 3-loop plants, totalling over the numbers of plants, and adding the NRC costs.

Value/Impact Assessment

Based on an estimated total industry risk reduction of 4.1E+5 person-rem and total cost of \$65M for the solution considered, the value/impact score is:

$$S = \frac{4.1E+5 \text{ person.rem}}{\$65M}$$

$$= 6.3E+3 \text{ person.rem/\$M}$$

Other Considerations

The PWR issue considered herein was a core damage scenario initiated by a main steam line break (MSLB) in the pipe segment between containment and the MSIV. Loss of pressure in the secondary side resulted in a primary to secondary side pressure differential sufficient to rupture flawed steam generator tubes. It must be recognized that a MSLB is not the only initiating event for a high primary/secondary pressure differential. Stuck open relief valve, break in the main feed line, or MSLB downstream of a stuck open MSIV should also be considered as initiators of similar scenarios. Other issues may be affected by multiple steam generator tube ruptures, such as Pressurized Thermal Shock. From Figure 1., note that, depending upon flaw length and depth, such SGTR initiators can occur at pressures less than or equal to PWR operating

pressures. It should also be noted that if operating years for the Base Case were more than the four assumed, e.g. 24 years (to achieve a 40 year plant life), the individual plant and total plant risks would be correspondingly larger.

Severe accident considerations should also be made during resolution activities. A PWR ATWS under degraded SG tube conditions, for example, would likely result in rupture of significant numbers of SG tubes with possible MSLB due to overpressure of the secondary system. In other severe accident scenarios, hot gases from a damaged core could heat degraded tubes, including tubes with through-wall cracks, causing tubes to slump, resulting in open pathways, to the secondary system and environment, for radionuclides.

Consideration should also be given to any new permanent leak rate criteria vis-a-vis existing regulations such as 10 CFR 50.34a (ALARA) and Part 50, Appendix I, as well as GDCs 14, 15, 30, 31, and 32.

CONCLUSION

Based on the potential public risk reduction associated with this issue and additional potential concerns under Other Considerations, this issue is assigned a HIGH priority ranking.

References

1. E. S. Beckjord memorandum to Thomas E. Murley, "A New Generic Issue: Multiple Steam Generator Tube Leakage," June 16, 1992.

2. J. Hopenfeld memorandum to E. Beckjord, "A new Generic Issue: Multiple Steam Generator Leakage," March 27, 1992.

3. W. T. Russell and J. T. Wiggins, "Staff Initiatives in Steam Generator Integrity," presented at the 1992 NRC Regulatory Information Conference, Washington, D.C., July 22, 1992.

4. Joseph Muscara memorandum to Charles Z. Serpan, "Steam generator Tube Inspection, Integrity and Plugging," March 16, 1992.

5. NRC Integrated Program for the Resolution of Unresolved Safety Issues A-3, A-4, and A-5 Regarding Steam Generator Tube Integrity, Final Report, NUREG-0844, September 1988.

6. "Reactor Safety Study, an Assessment of Accident Risks in U.S. Commercial Nuclear Power Plants WASH-1400/NUREG-75/014, October 1975.

7. Letter from Lawrence E. Kokajko, NRC to James E. Cross, Portland General Electric Company, February 5. 1992. Docket No. 50-344.

8. J. E. Cross, PGE, letter to U.S. Nuclear Regulatory Commission, January 16, 1992, Docket 50-344.

9. Alzheimer, J. M. et al., "Steam Generator Tube Integrity Program - Phase I Report," NUREG/CR-0718, Pacific Northwest Laboratory, Richland, Washington, 1979.

10. E. C. Rodabough, "Comments on Leak-Before-Break Concept for Nuclear Power Plant Piping System," NUREG/CR-4305, Oak Ridge National Laboratory, Oak Ridge, TN, August 1985.

11. Report of the U.S. Nuclear Regulatory Commission Piping Review Committee, NUREG-1061, Vol. 5, P-17, April 1985.

12. SECY-91-270, "Interim Guidance on Staff Implementation of the Commission's Safety Goal Policy," August 27, 1991.

13. W. B. Andrews, et al., "Guidelines for Nuclear Power Plant Safety Issue Prioritization - Information Development," NUREG/CR-2800, Pacific Northwest Laboratory, Richland, WA, February 1983.

14. Information Digest, NUREG-1350, U.S. Nuclear Regulatory Commission, March 1992.

15. Wayne L. Miller and Lloyd C. Brown, "Generic Cost Analysis for Steam Generator Repairs and Replacements," EGG-PE-6670, EG&G Idaho, Inc., Idaho Falls, ID, August 1984.

16. E. Claiborne, et al., "Generic Cost Estimates," NUREG/CR-4627, Science and Engineering Associates, Inc., Albuquerque, NM, February 1989.

17. Section 3.6.2, "Determination of Rupture Locations and Dynamic Effects Associated with Postulated Rupture of Piping, "Standard Review Plan, NUREG-0800, USNRC, Rev. 1, July 1981.