Entergy
Operations

Entergy Operations, Inc.
Route 3, Box 137G
Russellville, AR 72801
Tel 501-964-3100

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U. S. Nuclear Regulatory Commission
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Subject: Arkansas Nuclear One - Units 1 and 2
Docket Nos. 50-313 & 50-368
License Nos. DPR-51 & NPF-6
Resolution of Unresolved Safety Issue A-47,
"Safety Implication of Control Systems in
LWR Nuclear Power Plants" Generic Letter 89-19
(TAC Nos. M74906 and M74907)

Gentlemen:

Generic Letter 89-19 was issued September 20, 1989 (0CNA098921), requesting action related to Unresolved Safety Issue A-47 concerning control systems in LWRs. The staff concluded that all PWR plants should provide automatic steam generator overfill protection and establish plant procedures and technical specifications to assure the system remains available. The licensee was to provide a response to the NRC outlining the intended compliance or justification for not implementing the recommendations.

On March 19, 1990 Entergy Operation responded in letters 1CAN039001 and 2CAN039001 that there were several issues that needed to be considered prior to installation of an automatic overfill protection system at Arkansas Nuclear One, Units 1 and 2 (ANO-1 & 2). Specifically, assumptions and information utilized in the NRC evaluation were outdated or unsupported, and an evaluation assessing the negative impact on safety of the proposed modification was not performed.

Due to the potential negative safety impact associated with the modification, Entergy Operations proposed to assess the issue in the Individual Plant Examination (IPE) process. The NRC responded in letters dated October 1, 1990 (1CNA109002 and 2CNA109002), that Entergy Operations' proposal was unacceptable and that implementation of Generic Letter 89-19's recommendations should progress.

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The Combustion Engineering Owners Group (CEOG), which ANO-2 is a member, then contacted the NRC and arranged for a meeting to discuss the generic letter. Entergy Operations notified the NRC in letter dated November 19, 1990 (0CAN119003), that a revised response to the generic letter for both units would be developed based on the outcome of the meeting. Based on the common position taken on the generic letter, for both ANO-1&2 and the generic nature of the CEOG presentation, Entergy Operations believed that the results of the meeting would be applicable to both ANO units.

On November 20, 1990, the NRC and representatives of the CEOG met. The CEOG presented their evaluation and demonstrated that steam generator overfill was not a significant safety concern and that the installation of an automatic steam generator overfill protection system could, in fact, degrade safety and was not cost justified. The group requested that the NRC staff consider the CEOG position as a basis for not implementing the generic letter's recommendations and accept the analysis as the justification for resolution. The NRC staff committed to assess the information and stated that implementation was to be delayed until the staff evaluation was completed. Based upon this NRC response, implementation of GL 89-19 for ANO-1&2 was delayed until NRC review of the CEOG data was complete.

Based on a recent discussion with the ANO-2 NRR Project Manager, ANO has been informed that the NRC Staff has completed their review of the CEOG data and are planning to close this issue for the CEOG in correspondence soon to be issued. During this discussion, the Project Manager requested that Entergy Operations provide a discussion of the applicability of the CEOG presentation to ANO-1. Attached is the requested information.

Entergy Operations believes that this submittal provides sufficient information for the Staff to close this issue for ANO-1. Should you have any questions regarding this issue, please contact me.

Very truly yours,

for James J. Fisicaro
Director, Licensing

JJF/RWC/sjf
Attachment

cc: Mr. Robert Martin
U. S. Nuclear Regulatory Commission
Region IV
611 Ryan Plaza Drive, Suite 400
Arlington, TX 76011-8064

NRC Senior Resident Inspector
Arkansas Nuclear One - ANO-1 & 2
Number 1, Nuclear Plant Road
Russellville, AR 72801

Mr. Thomas W. Alexion
NRR Project Manager, Region IV/ANO-1
U. S. Nuclear Regulatory Commission
NRR Mail Stop 13-H-3
One White Flint North
11555 Rockville Pike
Rockville, Maryland 20852

Ms. Sheri Peterson
NRR Project Manager, Region IV/ANO-2
U. S. Nuclear Regulatory Commission
NRR Mail Stop 13-H-3
One White Flint North
11555 Rockville Pike
Rockville, Maryland 20852

DISCUSSION OF THE APPLICABILITY OF THE
CEOG PRESENTATION TO ANO-1 WITH RESPECT TO
STEAM GENERATOR OVERFILL PROTECTION

BACKGROUND

Generic Letter (GL) 89-19 recommends that the Babcock and Wilcox (B&W) - designed PWR plants have an automatic steam generator overfill protection to mitigate main feedwater overfeed events. In addition, plant procedures and technical specifications are to include provisions to periodically verify the operability of the overfill protection. A third recommendation, which is not applicable to ANO-1, involves the installation of automatic protection to prevent steam generator dryout on loss of control system power. These recommendations are based in the probabilistic risk analysis (PRA) performed by Pacific Northwest Laboratory (PNL) and documented in NUREG/CR-4386 (Reference 1). A review of this and related documents in conjunction with the CEOG position presented to the NRC in November of 1990, reveals similar concerns and conclusions for ANO-1. Specifically, assumptions and information utilized in the NRC PRA evaluation are outdated or unsupported, and an evaluation assessing the negative impact on safety of the proposed modification was not performed. A review of the major core damage scenario, the assumptions utilized in the NRC PRA evaluation, and the safety benefit/value impact analysis demonstrates that a steam generator overfill protection system should not be installed at ANO-1.

DISCUSSION OF THE PUBLIC RISK ASSESSMENT

The major overfill transient scenario assessed in Reference 1 at Oconee dictated that there must be failures that would initiate a main feedwater (MFW) overfeed, a failure of the MFW trip signal, and a failure of the operator to isolate feedwater flow. As the steam generator overfills, water spills into the main steam line, eventually resulting in a main steam line break (MSLB) due to the static and dynamic water loads on the piping. The steam generator experiences a pressure transient upon blowdown of the secondary side following the postulated MSLB. The pressure differential across the steam generator tubes is then postulated to induce one or more steam generator tube ruptures (SGTR). High pressure injection into the primary system continues to maintain core cooling as long as a water source (reactor building sump or borated water storage tank (BWST)) is available. If the MSLB location is outside the Reactor Building but upstream of the main steam isolation valve (MSIV), sufficient primary water is lost through the ruptured tubes to eventually exhaust the BWST inventory, at which point core damage is postulated to occur. The public risk due to this scenario as described in Reference 1 dominates the total risk associated with control system failure scenarios.

As noted earlier the major core damage scenario considered in Reference 1 is a steam generator overfill with a resulting SGTR. NUREG/CR-4386 also discusses overfill scenarios in conjunction with two other accident sequences. In one sequence an overfill is experienced and a transient shutdown is effected with the power conversion system unavailable because of degrading conditions in the secondary side. This scenario is considered to account for the potential of a main turbine trip before the point of spillover in the main steam line. The other scenario assumed a MSLB after overfill led to core damage and included the sequence in the report to be conservative. These accident sequences are not considered in this discussion since their contribution to the public risk has been determined by Entergy Operations and the NRC to be acceptably low. It should be noted that the CEOG concerns are also applicable to these other cases considered in Reference 1, although only the overfill with a MSLB and SGTR will be explicitly considered here.

The public risk is assessed by first analyzing the frequency of an overfeeding event and the probability that the operator fails to terminate the overfeed to establish an initiating event (IE) frequency. This is then combined with the probabilities of a MSLB given overfill into the steam lines, break location (i.e., inboard of the MSIVs), SGTR given a MSLB, and core melt given primary system injection sources are exhausted out the ruptured SG tube(s). Each of these probabilities directly contribute to the resulting assessment of public risk. The assessment provided in the Reference 1 NUREG, however, made incorrect assumptions in deriving these probabilities which artificially increased the public risk calculation (Table 1).

A. STEAM GENERATOR OVERFILL INITIATING EVENT FREQUENCY

One area that was investigated by the CEOG was the assumed initiating event frequency in relation to the probability of an operator failing to terminate an overfill scenario. The probability of an operator failure to terminate the overfill was estimated at 0.7 (recognized by PNL as an upper bound estimate of operator error) for the B&W units. In plant-specific PRAs, such overfill scenarios would be assigned an operator failure probability an order of magnitude lower, resulting in an associated order of magnitude further reduction in public risk. It should be noted that NUREG/CR-4386 recognized that the A-47 issue deals with control systems routinely under operator control, and therefore interaction of the operator with failure diagnosis and recovery is an appropriate consideration; and also recognized that the average failure probability would be lower in plants with simulator programs stressing proper diagnosis of failures. For ANO-1, as well as other B&W plants, MFW overfeed due to control system malfunctions receives special attention in operator training due to the smaller secondary volume of the B&W once-through steam generator (OTSG) and its associated responsiveness. As a result, the probability that an operator fails to terminate an overfeed event can be readily reduced to 0.07 which produces an initiating event frequency of 0.0006/yr (0.006/yr * 0.07/0.7).

B. PROBABILITY OF A MSLB

The probability of a MSLB given a steam generator overfill event utilized in the assessment was also unrealistic and inconsistent with the NRC's own analysis. Reference 1, the NRC PRA evaluation for B&W plants, acknowledged that although several spillover events have occurred in U.S. commercial plants resulting in support damage, no steam line failures have occurred and that this was an area of likely conservatism. The basis for the probability cited in Reference 1 accounted for uncertainties associated with dynamic loading and waterhammer effects. NUREG/CR-3958 (Reference 2) provides a discussion on the studies of overfill in which the MSLB probability was based and this was cited as part of the CEOG position. One study indicated a low probability of MSL failure (1×10^{-3}) when static forces caused by the deadweight of water in filled steam lines were considered. Deadweight loading was also addressed in the draft version of NUREG-0844 which assigned a 1 x 10 ` probability. This NUREG, however, was silent in relation to dynamic loading and the MSLB probability. As a result, the NRC evaluation of the B&W units in Reference 1 incorporated a 0.95 value.

When the final version of NUREG-0844 (Reference 3) was issued there were several changes that effected the overfill evaluation. Although the draft NUREG, which was utilized as an input to Reference 1, did not address dynamic loading, the final report considered the issue. NUREG-0844 was updated as follows (Section 3.4.1, page 3-10):

> The staff has assessed the change in the probability of failure of the main steam line from the increased stress levels associated with the deadweight of water in the steam lines. Analyses have been performed of the increase in stress levels that would result from filling the steam lines in several plants. Information extracted from analyses on the Ginna, Zion 1, Waterford 3, and Oconee 3 plants indicates that, although in some cases the spring hangers may be loaded slightly beyond their operating range, they will not fail and that the stress levels in the main steam line will in all cases remain within the limits allowed by the ASME Code. In addition to the analyses available, the steam lines were inspected after overfilling events at Oconee and Ginna and no indications of failures or incipient failures were found. Therefore, the staff concludes that the probability of failure of the main steam line is not increased by the deadweight loading. Nor is there considered to be a significant potential for failure from waterhammer since the water in the steam lines will be essentially saturated. Accordingly, the estimates of risk in this report for event sequences that consider failure of the main steam lines are based on a conservatively determined conditional probability of main steam line failure of 1×10^{-3}/overfill event.

The effect of overfill on main steam line integrity continued to be examined under GI-135. On October 5, 1989, the NRC presented the results of the GI-135 program to the ACRS. Task 4 of GI-135 (Steam Generator Overfill) addressed steam line integrity concerns due to the steam generator being overfed or otherwise filled with water. The presentation showed that the NRC had resolved the steam generator overfill issue due to the small risk involved. This conclusion was based on analyses which indicate that some spring hangers may be loaded beyond specification due to deadweight loading, but they will not fail. In addition, because the water in the steam lines is at saturation temperature and pressure, the potential for failure due to condensation induced waterhammer is small. Overfills that have occurred under similar conditions have resulted in little or no damage to steam line piping. Therefore, based on the results of the NUREG-0844 Final Report and GI-135, a reduction in the probability of a MSLB due to an overfill from 0.95 to 1×10^{-3} is appropriate and justified.

C. PROBABILITY OF A MSLB LOCATED OUTSIDE THE REACTOR BUILDING AND INBOARD OF THE MSIVs GIVEN A MSLB

The third probability that was assessed in the CEOG position is the MSLB location. In the dominant scenario, core melt occurs as a result of a loss of RCS inventory through an unisolable steam line break (in conjunction with tube ruptures) which eventually exhausts the borated water storage tank. The steam line break location probability is based on the assumption that a MSLB has an equal probability of occurring upstream or downstream of the main steam isolation valve. A break upstream is assumed to result in core melt since all water exiting the break would be lost outside of the Reactor Building. In reality, the MSIV is located relatively close to the outside Reactor Building wall, which results in the majority of piping upstream of the MSIV being located inside the Reactor Building. If the MSLB occurred inside the Reactor Building, water lost through the break would be collected in the Reactor Building sump and be available for recirculation. Thus, core melt would not occur without additional failures. Reference 1 assumes a probability of 1.0 since Oconee has no MSIVs, although it acknowledges that valves are present in the general population of B&W PWRs. For these plants a 50 percent probability is to be utilized. The maximum probability, however, cannot exceed the product of 0.5 times the ratio of the main steam line piping length outside the Reactor Building to the MSIV to the total main steam line piping length to the MSIV. For ANO-1, this probability is 1.55×10^{-1} and should be utilized for its assessment.

D. PROBABILITY OF A SGTR GIVEN A MSLB

As noted earlier the final version of NUREG-0844 differed from the draft version which was utilized to prepare the overfill evaluation for the B&W units. The probability of SGTRs after a steam line break was taken from the draft report and adopted unchanged in the Reference 1 analysis (Section 2.3, page 2.8). The early draft of NUREG-0844 established the probability of tube rupture due to a MSLB as 0.034. This probability was broken down as follows:

p (1 SGTR)	=	0.017
p (2-10 SGTRs)	=	0.014
p (>10 SGTRs)	=	0.003

In preparing the final report for NUREG-0844, the NRC changed certain assumptions and approaches in calculating the probabilities associated with single and multiple SGTRs. The total probability of tube rupture due to an MSLB was revised to 0.0505. This probability was broken down as follows (Section 3.4.6, Sequences 8A, 8B, and 8C, of Reference 3):

p (1 SGTR)	=	0.025
p (2-10 SGTRs)	=	0.025
p (>10 SGTRs)	=	0.0005

Although the overall SGTR probability was increased, the probability of rupturing greater than 10 tubes was decreased by nearly an order of magnitude. The NUREG/CR-4386 analysis was particularly sensitive to the value assumed for p(>10 SGTRs) in the calculation of the core melt probability.

E. PROBABILITY OF CORE MELT GIVEN MSLB INBOARD OF THE MSIV

Utilizing the SGTR probability values from NUREG 0844 Final Report changes and reduces the probability of core melt given a MSLB inboard of the MSIV. Table 2 illustrates the differences by reproducing the table from Reference 1 page 2.10 which uses the draft NUREG 0844 probabilities and a revised table incorporating the appropriate final version values. By examining this table, it can be seen that the reduction in probability of greater than 10 SGTRs by more than an order of magnitude results in a notable change in the total core melt value. By merely substituting the final report values and making no other changes, the core melt probability given a MSLB inboard of the MSIV is modified to 5.25×10^{-4} from an original value of 1.66×10^{-3}.

Further reduction in the core melt probability can be justified by reassessing the probability of loss of BWST before a RCS depressurization for ruptures of more than 10 tubes. This probability dominates the core melt calculation (Table 2). It should be noted that the loss of the BWST supply probability utilized in Reference 1 was based upon the unrealistic estimate of time of 1 hour to empty the tank for Westinghouse plants with 20 ruptures, the assumptions of runout flow for low pressure safety injection, and no operator action to depressurize the RCS and stop leak flow. As noted earlier, operator training alone can reduce this value significantly. This is an area, however, that cannot be easily quantified and the 0.5 value utilized in Reference 1 will be included and considered conservative. Therefore, the probability of core melt given a MSLB inboard of the MSIV is 5.25×10^{-4} which is not only appropriate but is conservative.

F. PUBLIC RISK CALCULATION

A new public risk can be calculated utilizing the probabilities previously discussed. Table 1 illustrates the assessment using the Reference 1 analysis values and the new probabilities. NUREG/CR-4386 delineates a public risk of 45.4 man-rem/yr and the new calculation yields a value of 2.34×10^{-4} man-rem/yr. This is a significant reduction in the public risk.

This public risk, however, cannot be evaluated without considering the negative impact that the proposed steam generator overfill protection system could have on safety. The automatic feedwater pump trip function recommended by GL 89-19 can itself cause a loss of feedwater accident due to spurious actuation or testing failures. Adverse consequences can also result from spurious actuation during other events. Unfortunately, the analysis in NUREG/CR-4386 and the value/impact analysis do not address the negative impact to safety due to installation of an overfill protection system. Using the same approach as the PNL study including highly conservative failure assumptions, multiple failures, a high probability of operator failure to restore feedwater, etc., the public risk due to installation of the feedwater pump trip could be significant. At a minimum, it must be considered and will result in a reduction in the benefit attributed to such a system.

The CEOG presented the results of a scoping calculation using generic data and the noted conservative approach to estimate the core damage probability due to testing of the proposed system. The calculation result was a core damage probability of 1.4E-06/yr. This corresponds to a core melt probability from the potential overfeed of 4.88E-11/yr. Combining these probabilities indicates that the proposed modification would degrade safety (4.88E-11/yr - 1.4E-06/yr = - 1.4 E-06/yr).

G. COST BENEFIT OF GL 89-19 PROPOSED MODIFICATIONS

The final item that warrants consideration for ANO-1 is the safety benefit and value impact of the recommended steam generator overfill protection system. This, however, is difficult to assess since the assumptions made in the regulatory analysis were not applicable to ANO-1. The bases for the recommendations in GL 89-19 are discussed in NUREG-1218 (Reference 4) which used the calculations of NUREG/CR-4386 (Reference 1) to estimate the safety benefit and value impact of various proposed upgrades. The feedwater control system at ANO-1 is significantly different from Oconee, and as a result the values for both costs and benefits of the proposed upgrades which were used in the NRC s regulatory analysis do not apply to ANO-1. For example, ANO-1 has made major improvements in the MFW control system, and in the Integrated Control System (ICS) over the past several years which make the actual probability of a MFW overfeed due to control system failures significantly lower than that assumed for Oconee.

A further examination of the factors discussed above should lead to an estimated risk reduction for the applicable control system failure scenarios well below the point at which the NRC's value/impact guidelines would conclude that hardware changes are a viable option. More significantly, when plant specific factors are taken into account, the actual risk reduction due to an overfill protection system may actually be less than the risk increase due to spurious operation of the system. Based on the above concerns, Entergy Operations believes that, for ANO-1, the actual risk due to overfill scenarios is substantially lower than estimated in the basis NUREGs for GL 89-19. It should be noted that NUREG-1218 incorrectly assumed that all B&W plants other than Oconee either had in place or had committed to modify their designs to include a safety grade overfill protection system. The Emergency Feedwater Initiation and Control (EFIC) system at ANO-1, a safety grade system, was originally designed with the capability for MFW overfill protection. Due to the concerns related to adverse consequences resulting from spurious operation, questionable cost/benefit (cost estimated in excess of $1 million dollars), and expected (which have subsequently been implemented) improvements in the MFW and ICS control systems, Entergy Operations determined that overfill protection implementation was not appropriate. The MFW overfill issue was specifically addressed by Entergy Operations as part of the B&WOG Safety & Performance Improvement Program (SPIP).

Reference 4 specified a value of less than $200,000 for the installation of an automatic overfill protection system for the CE PWR plants which was used by the CEOG in their evaluation. Since ANO-1 cannot be assessed in relation to the alternatives discussed for the B&W PWR plants, this evaluation will utilize the low value of $200,000 as the cost to install the proposed system and the value which makes the option viable (Table 1). It should be reiterated that Entergy Operations estimated than an overfill system would cost in excess of one million dollars to install. Using $1000/man-rem reduction in public risk yields a seven dollar cost benefit over 30 years without considering the negative impact of the system on safety. A calculation such as the CEOG presented would yield a negative cost benefit to the plant (Table 1). If the seven dollar cost benefit, however, is compared to the NRC estimated $200,000 modification cost, the overfill protection modification is not warranted.

CONCLUSION

In conclusion, the installation of an automatic steam generator overfill protection system for ANO-1 is not justified from a public health and safety or cost benefit standpoint. Reviewing the data and input assumptions related to ANO-1, a B&W PWR plant, the same generic concerns presented to the NRC by the CEOG on November 20, 1990, for CE PWR plants can be applied. It has been shown that the public risk value can be reduced to 2.34×10^{-4} man-rem/yr from 45.4 man-rem/yr which is a difference of over five orders of magnitude. Deliberation of these values and a seven dollar cost benefit with a low value of $200,000 installation cost demonstrates that the generic letter recommendations are not justified for implementation at ANO-1.

REFERENCES

1. NUREG/CR-4386, Effects of Control System Failures on Transients, Accidents, and Core Melt Frequencies at a Babcock and Wilcox Pressurized Water Reactor, Pacific Northwest Laboratory, December 1985.

2. NUREG/CR-3958, Effects of Control System Failures on Transients, Accidents and Core-Melt Frequencies at a Combustion Engineering Pressurized Water Reactor, Pacific Northwest Laboratory, March 1986.

3. NUREG-0844, NRC Integrated Program for the Resolution of Unresolved Safety Issues A-3, A-4, and A-5 Regarding Steam Generator Tube Integrity, U. S. Nuclear Regulatory Commission, September 1988.

4. NUREG-[illegible]3, Regulatory Analysis for Resolution of USI A-47, U. S. Nuclear Regulatory Commission, July 1989.

5. CEOG-90-330, CEOG/NRC Meeting on S/G Overfill Protection (GL 89-19); November 20, Forwarding of Summary, November 27, 1990.

6. CEOG-91-143, NRC Summary of November 20, 1990 CEOG/NRC Meeting on Generic Letter 89-19, March 11, 1991.

TABLE 1

CALCULATION OF PUBLIC RISK

I. Definition of Terms

FOE: Frequency of an Overfeeding Event
POF: Probability of an Operator Failing to terminate the event
PEBWST: Probability of Loss of BWST before RCS Depressurization (Table 2)
IE: Steam Generator Initiating Event Frequency
PMSLB: Probability of a MSLB given overfill and spillover
PMSLBL: Probability of a MSLB Located outside the Reactor Building and inboard of the MSIVs given a MSLB
PSGTR: Probability of a SGTR given MSLB
PCM: Probability of Core Melt given MSLB inboard of the MSIV and outside the Reactor Building

II. Formula for Public Risk (PR)

$$IE = FOE * POF$$

$$PCM = \sum_{i=1}^{i} (PSGTR_i * PEBWST_i) \qquad \text{See Table 2}$$

$$PR = IE * PMSLB * PMSLBL * PSTGR * PCM/PSGTR * 4.8\times10^6 \ \frac{\text{man-rem}}{\text{core melt}}$$

III. Public Risk Utilizing Values from NUREG/CR-4386

POF = 0.7
IE = 0.006/yr
PMSLB = 0.95
PMSLBL = 1.0 (Oconee does not have MSIVs)
PSGTR = 0.034
PCM = 1.66×10^{-3}

$$PR = \frac{0.006}{\text{yr}} * 0.95 * 1.0 * 0.034 * \frac{1.66 \times 10^{-3}}{0.034} \text{ core melt} * 4.8 \times 10^6 \ \frac{\text{man-rem}}{\text{core melt}}$$

$$= 4.54 \times 10^1 \ \frac{\text{man-rem}}{\text{yr}}$$

Total Public Risk for 30 years = 1360 man-rem

TABLE 1 (Continued)

CALCULATION OF PUBLIC RISK

IV. Public Risk Utilizing the New Values Applicable for ANO-1

POF = 0.07
IE = 0.0006/yr
PMSLB = 1×10^{-3}
PMSLBL = 0.155
PSGTR = 0.0505
PCM = 5.25×10^{-4}

$$PR = \frac{0.0006}{yr} * 1 \times 10^{-3} * 0.155 * 0.0505 * \frac{5.25 \times 10^{-4} \text{ core melt}}{0.0505} * 4.8 \times 10^{6} \frac{\text{man-rem}}{\text{core melt}}$$

$$= 2.34 \times 10^{-4} \frac{\text{man-rem}}{\text{yr}}$$

Total Public Risk for 30 years = 7.03×10^{-3} man-rem

V. Cost Benefit of Overfill Protection

Assumption: \$1000/man-rem benefit and \$200,000[+] modification cost.

Given: Total Public Risk for 30 years = 7.03×10^{-3} man-rem

Conclusion: Benefit of \$7 over 30 years opposed to a \$200,000 cost that was assumed appropriate provides justification for not implementing the modification on the basis of the cost benefit.

[+]NUREG 1218 delineates cost projections between \$100,000 and \$1,100,000 to upgrade already existing overfill protection. In addition, the CEOG utilities admitted during the November 20, 1990 meeting that a system could not be installed for \$200,000. Therefore, this value is not representative of ANO-1 since it would cost far more to install the automatic overfill protection system.

TABLE 2

CORE MELT PROBABILITIES GIVEN MSLB CONSIDERING SGTR

NUREG/CR-4386 (REFERENCE 1) P. 2.10
Utilizes draft NUREG 0844

Case 1: Rupture of Main Steam Line Inboard of the MSIV

Number of SGTRs	Prob. of Rupture	Probability of Loss of RWST before RCS Depressurization	Prob. of Failure to Isolate SG	Net Core-Melt Prob.
1	0.017	1E-03	1	1.7E-05
2 to 10	0.014	1E-02	1	1.4E-04
>10	0.003	0.5	1	1.5E-03

Total Probability of Core Melt Given MSLB Inboard of MSIV 1.66E-03

Case 2: Rupture of Main Steam Line Downstream of the MSIV

Number of SGTRs	Prob. of Rupture	Probability of Loss of RWST before RCS Depressurization	Prob. of Failure to Isolate SG	Net Core-Melt Prob.
1	0.017	1E-04	1E-03	1.7E-09
2 to 10	0.014	1E-03	1E-03	1.4E-08
>10	0.003	1E-03	1E-03	3.0E-09

Total Probability of Core Melt Given MSLB Downstream of MSIV 1.87E-08

TABLE 2 (Continued)

CORE MELT PROBABILITIES GIVEN MSLB CONSIDERING SGTR

REVISED PROBABILITIES
Utilizes NUREG 0844 Final Report (Reference 3)

Case 1: Rupture of MSL Inboard of the MSIV

Number of SGTRs	Prob. of Rupture	Probability of Loss of RWST before RCS Depressurization	Prob. of Failure to Isolate SG	Net Core-Melt Prob.
1	0.025	1E-03	1	2.5E-05
2 to 10	0.025	1E-02	1	2.5E-04
>10	0.0005	5E-01	1	2.5E-04

Total Probability of Core Melt Given MSLB Inboard of MSIV 5.25E-04

Case 2: Rupture of MSL Downstream of MSIV

Number of SGTRs	Prob. of Rupture	Probability of Loss of RWST before RCS Depressurization	Prob. of Failure to Isolate SG	Net Core-Melt Prob.
1	0.025	1E-04	1E-03	2.5E-09
2 to 10	0.025	1E-03	1E-03	2.5E-08
>10	0.0005	1E-03	1E-03	5.0E-10

Total Probability of Core Melt Given MSLB Downstream of MSIV 2.8E-08