October 26, 2000

Mr. Michael A. Balduzzi
Vice President, Operations
Vermont Yankee Nuclear Power Corporation
185 Old Ferry Road
P.O. Box 7002
Brattleboro, VT 05301-7002

SUBJECT: VERMONT YANKEE NUCLEAR POWER STATION - REQUEST FOR LICENSEE COMMENTS ON PRIORITIZATION OF GENERIC SAFETY ISSUE 156.6.1: "PIPE BREAK EFFECTS ON SYSTEMS AND COMPONENTS INSIDE CONTAINMENT"

Dear Mr. Balduzzi:

The Office of Nuclear Regulatory Research (RES) is currently assessing whether the nuclear power plant units, referred to as the Systematic Evaluation Program Phase III (SEP-III) plants, will need to be individually reevaluated for the resolution of Generic Safety Issue (GSI) 156.6.1, "Pipe Break Effects on Systems and Components Inside Containment." GSI 156.6.1 deals with whether the effects of high energy pipe breaks inside containment have been adequately addressed in the respective designs of these units. The 41 SEP-III plants for which this GSI is applicable are listed in Enclosure 1. Vermont Yankee Nuclear Power Station (Vermont Yankee), is among the original 41 nuclear power plant units within the scope of GSI 156.6.1.

As background, in November 1975, the Nuclear Regulatory Commission (NRC) staff issued Standard Review Plan (SRP), Section 3.6.1, "Plant Design for Protection Against Postulated Piping Failures in Fluid Systems Outside Containment" and Section 3.6.2, "Determination of Rupture Locations and Dynamic Effects Associated with the Postulated Rupture of Piping." The Atomic Energy Commission (AEC)/NRC staff conducted the licensing safety evaluations for the SEP-III plants, including Vermont Yankee, before these SRPs were issued. Prior to issuance of these SRP sections, AEC/NRC staff positions were evolving for the licensing safety evaluation reviews for the effects of pipe breaks inside containment. Although the AEC/NRC licensed the SEP-III plants, the potential lack of uniformity in those reviews may have resulted in some of the units not being adequately analyzed and/or designed for postulated pipe breaks inside containment by the standards of these SRPs. For plants reviewed after November 1975, the specific structural and environmental effects of pipe whip and jet impingement on systems and components relied on for safe reactor shutdown were systematically and consistently considered by the staff in the licensing safety evaluation reviews.

In 1999, RES completed an "enhanced" prioritization of GSI 156.6.1 in accordance with the NRC's internal procedures. The prioritization of this GSI is contained in two documents. The first document, entitled: "Prioritization of Generic Issue 156.6.1, 'Pipe Break Effects on Systems and Components,'" is provided in Enclosure 2. It is a priority determination analysis by the RES staff. The second document, provided in Enclosure 3, is Draft NUREG/CR-6395, entitled: “Enhanced Prioritization of Generic Safety Issue 156.6.1: “Pipe Break Effects on Systems and Components Inside Containment.” The latter document was prepared by the Idaho National Engineering and Environmental Laboratory (INEEL) and provides extensive and detailed technical information and analysis information in support of the staff’s priority determination analysis. The prioritization resulted in the GSI being given a “high” priority for resolution. In conducting the prioritization study (i.e., Enclosure 3) several boiling water reactor (BWR) and pressurized water reactor (PWR) SEP III facilities were visited by INEEL. Vermont Yankee was one of the BWR facilities visited by INEEL.

The BWR and PWR SEP-III plant pipe break effect insights used in the enhanced prioritization will be included in the follow-on GSI technical evaluation, including the development of the recommended resolution actions. The objective of this request is to collect additional information on a voluntary basis which identifies sources of elevated conservatism in the scenarios used in the prioritization of probabilistic risk assessments. Comments could be based on information in the literature or knowledge of your individual plant design. For example, information on the plant-specific equipment arrangements of Vermont Yankee might show where and how the prioritization analysis for the BWR SEP-III plants is overly conservative or incorrect. Information might also be provided that shows that assuming a break, the model for the pipe break effects, or the model of the plant (or operator) response to the postulated break is incorrect or overly conservative for Vermont Yankee. We specifically invite your comments on whether pipe break locations and pipe break effects assumed in the staff’s prioritization analysis for the BWR SEP-III plants (Enclosure 2) are applicable to Vermont Yankee from a deterministic (i.e., engineering analysis) standpoint. For pipe break locations and effects which are considered not applicable, you may describe the technical basis for your conclusion. Comments received within 45 days of receipt of this letter will be considered. (Note: Pipe break scenarios for BWRs designated as Case 4 and Case 5 in Enclosure 2 will not be included in the technical evaluation of GSI 156.6.1 and, therefore, comments are not requested for these cases. These scenarios are being evaluated separately in connection with the resolution of GSI-80, “Pipe Break Effects on Control Rod Drive Hydraulic Lines in the Drywell of BWR Mark I and Mark II Containments.”)

If you or your staff have any questions on this request or the enclosures, please feel free to contact either Richard Croteau, E-mail: RXC2@NRC.GOV, 301-415-1475 of the Office of Nuclear Reactor Regulation or Mr. Stuart D. Rubin, E-mail: SDR1@NRC.GOV, 301-415-7480 of the Office of Nuclear Regulatory Research. Comments should also be forwarded to these individuals.

Sincerely,

/RA/

Richard P. Croteau, Sr. Project Manager, Section 2
Project Directorate I
Division of Licensing Project Management
Office of Nuclear Reactor Regulation

Docket No. 50-271

Enclosures: As Stated

cc w/encls: See next page

Idaho National Engineering and Environmental Laboratory (INEEL) and provides extensive and detailed technical information and analysis information in support of the staff's priority determination analysis. The prioritization resulted in the GSI being given a "high" priority for resolution. In conducting the prioritization study (i.e., Enclosure 3) several boiling water reactor (BWR) and pressurized water reactor (PWR) SEP III facilities were visited by INEEL. Vermont Yankee was one of the BWR facilities visited by INEEL.

The BWR and PWR SEP-III plant pipe break effect insights used in the enhanced prioritization will be included in the follow-on GSI technical evaluation, including the development of the recommended resolution actions. The objective of this request is to collect additional information on a voluntary basis which identifies sources of elevated conservatism in the scenarios used in the prioritization of probabilistic risk assessments. Comments could be based on information in the literature or knowledge of your individual plant design. For example, information on the plant-specific equipment arrangements of Vermont Yankee might show where and how the prioritization analysis for the BWR SEP-III plants is overly conservative or incorrect. Information might also be provided that shows that assuming a break, the model for the pipe break effects, or the model of the plant (or operator) response to the postulated break is incorrect or overly conservative for Vermont Yankee. We specifically invite your comments on whether pipe break locations and pipe break effects assumed in the staff's prioritization analysis for the BWR SEP-III plants (Enclosure 2) are applicable to Vermont Yankee from a deterministic (i.e., engineering analysis) standpoint. For pipe break locations and effects which are considered not applicable, you may describe the technical basis for your conclusion. Comments received within 45 days of receipt of this letter will be considered. (Note: Pipe break scenarios for BWRs designated as Case 4 and Case 5 in Enclosure 2 will not be included in the technical evaluation of GSI 156.6.1 and, therefore, comments are not requested for these cases. These scenarios are being evaluated separately in connection with the resolution of GSI-80, "Pipe Break Effects on Control Rod Drive Hydraulic Lines in the Drywell of BWR Mark I and Mark II Containments.")

If you or your staff have any questions on this request or the enclosures, please feel free to contact either Richard Croteau, E-mail: RXC2@NRC.GOV, 301-415-1475 of the Office of Nuclear Reactor Regulation or Mr. Stuart D. Rubin, E-mail: SDR1@NRC.GOV, 301-415-7480 of the Office of Nuclear Regulatory Research. Comments should also be forwarded to these individuals.

Sincerely,

/RA/

Richard P. Croteau, Sr. Project Manager, Section 2
Project Directorate I
Division of Licensing Project Management
Office of Nuclear Reactor Regulation

Docket No. 50-271
Enclosures: As Stated
cc w/encls: See next page

DISTRIBUTION:

PUBLIC	S. Rubin, RES
PDI-2 R/F	T. Clark
F. Eltawila, RES	R. Croteau
OGC	ACRS
G. Meyer, RI	E. Adensam
J. Clifford	

ACCESSION NO.: ML003742318

OFFICE	PDI-2/PM	PDI-2/LA		PDI-2/SC
NAME	RCroteau	TLClark		JClifford
DATE	10/19/00	10/19/00		10/24/00

OFFICIAL RECORD COPY

Systematic Evaluation Program Phase III Plants

Nine Mile Point 1	Vermont Yankee	Cooper
Robinson 2	Maine Yankee	Arkansas 1
Point Beach 1 & 2	Kewaunee	Calvert Cliffs 1
Monticello	Fort Calhoun	D. C. Cook 1
Dresden 3	Zion 1* & 2*	Hatch 1
Pilgrim	Browns Ferry 1 & 2	FitzPatrick
Quad Cities 1 & 2	Indian Point 2 & 3	Three Mile Island 1
Surry 1 & 2	Peach Bottom 2 & 3	Brunswick 2
Turkey Point 3 & 4	Prairie Island 1 &2	Trojan*
Oconee 1, 2, & 3	Duane Arnold	Millstone 2

* permanently shutdown

Enclosure 1

Prioritization of Generic Issue 156.6.1,
"Pipe Break Effects on Systems and Components"

DESCRIPTION

Historical Background

In 1967 the AEC published draft General Design Criteria (GDCs) for comment and interim use. Until 1972 the staff's implementation of the GDCs required consideration of pipe break effects inside containment. However, due to the lack of documented review criteria, NRC/AEC staff positions were continually evolving. Review uniformity was finally developed in the early 1970s; initiated by a note from L. Rogers to R. Fraley, "Safety Guides" dated November 9, 1972, in which a draft safety guide entitled "Protection Against Pipe Whip Inside Containment" was proposed. This draft guide contained some of the first documented deterministic criteria that the staff had been using (to varying degrees) for several years for selecting the locations and orientations of postulated pipe breaks inside containment, and for identifying the measures that should be taken to protect safety-related systems and equipment from the dynamic effects of such breaks. Prior to use of these deterministic criteria, the staff used non-deterministic guidelines on a plant-specific basis. This draft safety guide was subsequently revised and issued in May 1973 as Regulatory Guide 1.46 with the same title. The regulatory guide was implemented only on a forward-fit basis.

Regarding pipe break effects outside containment: in December 1972 and July 1973, the AEC issued two generic letters to all licensees and CP or OL applicants (References 1 and 2) ; known as the "Giambusso" and "O'Leary" letters, respectively. These letters extended the pipe break concerns to outside containment, and provided deterministic criteria for break postulation and evaluation of the dynamic effects of postulated breaks. The letters requested that all recipients submit a report to the staff which summarized each plant-specific analysis of this issue. All operating reactor licensees and license applicants submitted the requested analyses in separate correspondence or updated the safety analysis report for the proposed plant to include the analysis. The staff reviewed all of these submitted analyses and prepared safety evaluations for all plants. In November 1975, the staff published SRP Sections 3.6.1 and 3.6.2 that slightly revised the two generic letters discussed above. Thus, after 1975 the specific structural and environmental effects of pipe whip, jet impingement, flooding, etc. on systems and components relied on for safe reactor shutdown were considered.

As stated above, the AEC/NRC has provided requirements to the industry regarding pipe breaks outside of containment through the issuance of the "Giambusso" and "O'Leary" generic letters. Since these requirements are applicable to all the affected plants, pipe breaks outside of containment are considered a compliance issue and have been dropped from this prioritization. By EDO direction, compliance matters are to be dealt with promptly, and not await the generic issue resolution process. Therefore the issue of pipe breaks outside of containment for the 41 affected plants was brought to the attention of NRR by separate correspondence (Reference 3). The remainder of this prioritization discusses only pipe breaks inside containment.

As a part of its plant-specific reviews between 1975 and 1981, the staff used the guidelines in Regulatory Guide 1.46 for postulated pipe breaks inside containment and SRPs 3.6.1 and 3.6.2 for outside containment. In July 1981, SRPs 3.6.1 and 3.6.2 were revised to be applicable to both outside and inside containment; thus, eliminating the need for further use of Regulatory Guide 1.46.

Between the period 1983-1987, the NRC Systematic Evaluation Program (SEP) revisited the general issue of pipe breaks inside and outside containment. The objective of the SEP was to determine to what extent the earliest 10 plants (i.e., SEP-II) met the licensing criteria in existence at that time. This objective was later interpreted to ensure that the SEP also provided safety assessments adequate for conversion of provisional operating licenses (POLs) to full-term operating licenses (FTOLs). As a result of these reviews plants were required to perform engineering evaluations, technical specification or procedural changes, and physical modifications both inside and outside containment. Regarding inside containment modifications: of the two SEP-II plants evaluated for this prioritization (one BWR and one PWR), the BWR was required to modify four piping containment penetrations and the PWR was required to modify steam generator blowdown piping supports. This indicates there was a wide spectrum of implementation associated with the original reviews of these early plants for pipe breaks inside and outside containment.

As with the above-described evolution of uniform pipe break criteria, electrical systems design criteria were also in a state of development. Prior to 1974, electrical system designs were generally reviewed in accordance with the guidelines provided in IEEE-279; however, significant variations in interpretations of that document resulted in substantial design differences in plants. Specifically, true physical separation of wiring to redundant components was not necessarily accomplished. In 1974, Regulatory Guide 1.75 was published, clarifying the requirements.

A draft prioritization of this issue resulted in a MEDIUM determination and that the scope could be limited to pipe breaks inside containment since the NRC had already provided requirements regarding outside containment pipe breaks to the industry through the issuance of the previously mentioned "Giambusso" and "O'Leary" generic letters.

However, the uncertainty in the analysis was much wider than desired for a definitive priority ranking. Thus, the issue appeared to warrant additional analysis to enhance the prioritization. In July 1994 a contract was begun with the Idaho National Engineering Laboratory to:

1. Review of pipe failure rate data, pipe break methodologies, and related publications to determine recommended pipe failure rates (initiating events) applicable to the affected SEP-III plants.

2. Review of Updated Final Safety Analysis Reports and related Safety Evaluation Reports for SEP-II, SEP-III, and for representative non-SEP plants to identify and prioritize potential safety concerns (i.e., accident sequences). Several plant visits/walkdowns were included as part of this review.

3. Estimate changes to core damage frequencies for accident sequences that are determined to be of high or medium priority.

4. Identify potential corrective actions and their estimated costs.

Based on the results of the INEL research, the enhanced prioritization is presented below.

Safety Significance

GDC 4 is the primary regulatory requirement of concern. It requires, in part, that structures, systems and components important to safety be appropriately protected against the environmental and dynamic effects that may result from equipment failures, including the effects of pipe whipping and discharging fluids. Several possible scenarios for plants that do not have adequate protection against pipe whip were identified as a result of the research performed in support of the enhanced prioritization.

Related regulatory criteria include common cause failures, protection system independence, and the single failure criterion.

Recommended Solution

Issue Generic Letters to the affected plants requesting that they perform plant-specific reviews and walkdowns, identify vulnerable pipe break locations, and inform the NRC of proposed corrective actions.

PRIORITY DETERMINATIONS

Numerous scenarios of potential concern were evaluated. The following were considered important enough to be specifically identified for future consideration. All estimated frequencies and probabilities are mean values.

BWRs

Case 1 (INEEL BWR Event 1): Failure of Main Steam or Feedwater Piping Resulting in Pipe Whip and Containment Impact/Failure, with Resultant Failure of All Safety Injection Systems

This event involves a BWR with a Mark I steel containment; 15 of the 16 affected BWRs are of this design. A DEGB of an unprotected (i.e., no pipe whip restraint or containment liner impact absorber) large reactor coolant recirculation pipe inside containment and near the containment liner might result in puncturing the liner. The resulting unisolable LOCA steam environment would be introduced into the secondary containment building, possibly disabling the ECCS equipment located there. This scenario would greatly increase the probability of core damage and potential offsite doses.

All of the affected BWRs are more than 10 years old, and most use type 304 stainless steel in the primary system piping; a material that is susceptible to IGSCC degradation. It should be noted that piping of this material does not qualify for the extremely low rupture probability (Leak-Before-Break) provision of GDC 4. From NUREG-1150, the recirculation loop DEGB frequency for this material is estimated to be 1 E-4/Reactor-Year (Rx-Yr). The fraction of BWR primary piping inside containment that is either Main Steam (MS) or Feedwater (FW) is estimated to be 4.0 E-1. The fraction of MS or FW piping that can impact the containment metal shell is estimated to be 2.5 E-1.

The research performed indicates that there is considerable variation among the affected plants regarding the amount of pipe whip protection provided and the proximity of high energy lines to potential targets of concern, including redundant trains, (see Other Considerations). It was assumed that the probability of a MS or FW broken pipe rupturing the containment metal shell was 2.5 E-1.

The postulated event may also cause a common mode failure of the ECCS system since much of this equipment is located within the secondary containment and will be exposed to a harsh environment beyond its design basis, or that the ECCS piping will fail due to overpressurization of the containment annulus. In most of the affected plants, the ECCS is located in four different quadrants outside the suppression pool (torus). On the other hand, as stated above, redundant electrical power systems and initiating circuitry may not be physically separated in these older plants. Also, if the ECCS operates initially, the ECCS equipment rooms may not be fully protected from internal flooding as the water from the suppression pool flows out the broken pipe into the secondary containment. Based on these considerations the mean probability of loss of ECCS function was assumed to 8.0 E-1.

Based on the above assumptions, the mean value of change in CDF per reactor year is:

dCDF/Rx-Yr = 2.0 E-6

From WASH-1400, the nearest scenario to that described above is the large LOCA BWR-3 release category; involving a large LOCA and subsequent containment failure. However, in the WASH-1400 case, the containment failure results from overpressurization; not from pipe whip. Three of the four specific BWR-3 large LOCA accident sequences have an incidence frequency of 10 E-8/Rx-Yr, and the remaining one is 10 E-7/Rx-Yr; 10 E-8/Rx-Yr was chosen as the base case for this analysis.

<u>Case 2 (INEEL BWR Event 9): Failure of Recirculation Piping Resulting in Pipe Whip and Containment Impact/Failure, with Resultant Failure of All Emergency Core Cooling Systems</u>

This event is similar to Case 1 but involves the Recirculation System piping. From NUREG-1150, the recirculation loop DEGB mean frequency for this material is estimated to be 1 E-4/Rx-Yr. The fraction of BWR primary piping inside containment that is recirculation piping is estimated to be 2.0 E-1. The fraction of recirculation piping that can impact the containment metal shell is estimated to be 5.0 E-1. It was estimated that the mean probability of a recirculation system broken pipe rupturing the containment metal shell was 5.0 E-1. The mean probability of eventual failure of all ECCS by the same modes described for Case 1 is estimated to be 8.0 E-1.

Based on the above assumptions, the mean value of change in CDF per reactor year is:
dCDF/Rx-Yr = 4.0 E-6

<u>Case 3 (INEEL BWR Event 12): Failure of RHR Piping Resulting in Pipe Whip and Containment Impact/Failure, with Resultant Failure of All Emergency Core Cooling Systems</u>

This event is similar to Cases 1 and 2 but involves the RHR System piping. From NUREG-1150, the RHR DEGB frequency for this material is estimated to be 1 E-4/Rx-Yr. The fraction of BWR primary piping inside containment that is RHR piping is estimated to be

1.0 E-1. The fraction of RHR piping that can impact the containment metal shell is estimated to be 5.0 E-1. The mean probability of a recirculation system broken pipe rupturing the containment metal shell is 1.0 E-1. The mean probability of eventual failure of all ECCS by the same modes described for Cases 1 and 2 is estimated to be 8.0 E-1.

Based on the above assumptions, the mean value of change in CDF per reactor year is:

dCDF/Rx-Yr = 4.0 E-7

Case 4 (INEEL BWR Event 5): Failure of Recirculation Piping Resulting in Pipe Whip or Jet Impingement on Control Rod Drive Bundles, Causing Failure by Crimping of Enough Insert/Withdraw Lines to Result in Failure to Scram the Reactor

From NUREG-1150, the recirculation loop DEGB frequency for this material is estimated to be 1 E-4/Rx-Yr. The fraction of BWR primary piping inside containment that is recirculation piping is estimated to be 2.0 E-1. The fraction of recirculation piping that can impact or impinge on the CRD lines is estimated to be 2.5 E-1. It is estimated that the mean probability of a broken RHR pipe crimping enough CRD lines to prevent a scram (about 5 to 10 adjacent lines) is 1.0.

Based on the above assumptions, the mean value of change in CDF per reactor year is:

dCDF/Rx-Yr = 5.0 E-6

Case 5 (INEEL BWR Event 10): Failure of RHR Piping Resulting in Pipe Whip or Jet Impingement on Control Rod Drive Bundles, Causing Failure by Crimping of Enough Insert/Withdraw Lines to Result in Failure to Scram the Reactor

This event is similar to Case 3 but involves the RHR System piping. The research performed indicates that there is considerable variation among the affected plants regarding the amount of pipe whip protection provided and the proximity of high energy lines to potential targets of concern; walkdowns showed that in at least one case a large “unisolable from the R.C.S.” RHR line was routed directly between the two banks of CRD bundles. An RHR pipe break in this vicinity would surely impinge and/or impact on both banks simultaneously.

From NUREG-1150, the RHR DEGB frequency for this material is estimated to be 1 E-4/Rx-Yr. The fraction of BWR primary piping inside containment that is RHR piping is estimated to be 1.0 E-1. The fraction of RHR piping that can impact or impinge on the CRD lines is estimated to be 2.5 E-1. It is estimated that the mean probability of a broken RHR pipe crimping enough CRD lines to prevent a scram (about 5 to 10 adjacent lines) is 1.0.

Based on the above assumptions, the mean value of change in CDF per reactor year is:

dCDF/Rx-Yr = 2.5 E-6

Case 6 (INEEL BWR Event 14): Failure of High Energy Piping Resulting in Pipe Whip or Jet Impingement on Reactor Protection or Instrumentation & Control Electrical, Hydraulic or Pneumatic Lines or Components and Eventually Resulting in Failure of Mitigation Systems and Core Damage

From NUREG-1150, the Large LOCA frequency is 1.0 E-4/Rx-Yr. All high energy piping inside containment is considered. The fraction of high energy piping that can impact or impinge on these lines or components is estimated to be 5.0 E-1. The mean probability of a broken high energy line failing some of these lines or components to the extent that core damage results is estimated as 7.5 E-1.

Based on the above assumptions, the mean value of change in CDF per reactor year is:

dCDF/Rx-Yr = 3.8 E-5

Case 7 (INEEL BWR Event 16): Failure of High Energy Piping Resulting in Pipe Whip Impact on Reactor Building Component Cooling Water (RBCCW) System to the Extent That the RBCCW Pressure Boundary is Broken, Potentially Opening a Path to Outside Containment if Containment Isolation Fails to Occur; Also Possible Loss of RBCCW Outside Containment for Mitigation

From NUREG-1150, the Large LOCA frequency is 1.0 E-4/Rx-Yr. All high energy piping inside containment is considered. The fraction of high energy piping that can impact the RBCCW system is estimated as 1.0 E-1. The probability of an HELB broken pipe rupturing the RBCCW system is 5.0 E-1. The probability of failure to close of containment isolation check valve is 1.0 E-3; the probability of failure to close of a containment isolation motor operated valve is 3.0 E-3; this combines for a total of 4.0 E-3. Since the RBCCW surge tank in the secondary containment is vented to atmosphere and has a relatively small volume, it is assumed that its water inventory will drain quickly; for this reason the mean probability of opening a path to atmosphere outside containment is 1.0. Once this scenario proceeds to this point the RBCCW system in secondary containment will become unavailable, including the RHR heat exchanger; therefore, the probability of losing the RBCCW function outside containment to the extent that core damage occurs is 1.0.

Based on the above assumptions, the mean value of change in CDF per reactor year is:

dCDF/Rx-Yr = 2.0 E-8

The total change in core damage frequency for the above 7 BWR cases is:

dCDF/Rx-Yr = 5.2 E-5 (Ranks HIGH/MEDIUM in Figure 2 of NUREG-0933)

And, for all 16 affected BWRs:

dCDF/Yr = 8.3 E-4 (Ranks HIGH/MEDIUM in Figure 2 of NUREG-0933)

BWR Offsite Dose Table

GSI-156.6.1 Event Number per NUREG/CR-6395	GSI-156.6.1 dCDF (Events/Rx-Yr)	WASH-1400 Release Category	WASH-1400 Offsite Dose (Person-Rem/ Event)	Offsite Dose (OSD) (Person-Rem/ Reactor Year)
BWR Event 1	2.0 E-6	BWR-3	5.1 E+6	10.2
BWR Event 5	5.0 E-6	BWR-4	6.1 E+5	3.1
BWR Event 9	4.0 E-6	BWR-3	5.1 E+6	20.4
BWR Event 10	2.5 E-6	BWR-4	6.1 E+5	1.5
BWR Event 12	4.0 E-7	BWR-3	5.1 E+6	2.0
BWR Event 14	3.8 E-5	BWR-4	6.1 E+5	23.2
BWR Event 16	2.0 E-8	BWR-3	5.1 E+6	0.1
			Total	60.5

For the 17 affected BWRs, the estimated change in offsite dose per reactor (d Person-Rem/Reactor) is:

$$\frac{60.5 \text{ Person-Rem}}{\text{Reactor-Year}} \times 17 \text{ Average Remaining Years} = \frac{1029 \text{ Person-Rem}}{\text{Reactor}}^{*} \text{ (Offsite)}$$

*(Ranks HIGH/MEDIUM in Figure 2 of NUREG-0933)

For 20 years of life extension:

$$\frac{60.5 \text{ Person-Rem}}{\text{Reactor-Year}} \times 37 \text{ Average Remaining Years} = \frac{2239 \text{ Person-Rem}}{\text{Reactor}}^{*} \text{ (Offsite)}$$

*(Ranks HIGH/MEDIUM in Figure 2 of NUREG-0933)

And the estimated change in offsite dose for the 16 affected BWRs is:

$$\frac{1029 \text{ Person-Rem}}{\text{Reactor}} \times 16 \text{ Affected BWRs} = 16{,}464 \text{ Person-Rem}^{*} \text{ (Total Offsite, All Affected BWRs)}$$

*(Ranks MEDIUM/LOW in Figure 2 of NUREG-0933)

For 20 years of life extension:

$$\frac{2239 \text{ Person-Rem}}{\text{Reactor}} \times 16 \text{ Affected BWRs} = 35{,}824 \text{ Person-Rem}^{*} \text{ (Total Offsite, All Affected BWRs)}$$

*(Ranks HIGH/MEDIUM in Figure 2 of NUREG-0933)

PWRs

Case 1 (INEEL PWR Event 9): Failure of Non-Leak-Before-Break Reactor Coolant System, Feedwater, or Main Steam Piping Resulting in Pipe Whip or Jet Impingement on Reactor Protection or Instrumentation & Control Electrical, Hydraulic or Pneumatic Lines or Components and Eventually Resulting in Failure of Mitigation Systems and Core Damage

From NUREG-1150, the HELB frequency in the above listed systems is 1.5 E-3/Rx-Yr. All of the listed high energy piping inside containment is considered. The fraction of high energy piping that can impact or impinge on these lines or components is estimated to be 1.0 E-1. The mean probability of a broken high energy line failing some of these lines or components to the extent that core damage results is estimated as 5.0 E-1.

Based on the above assumptions, the mean value of change in CDF per reactor year is:

dCDF/Rx-Yr = 7.5 E-5

Case 2 (INEEL PWR Event 16): Failure of Main Steam or Feedwater Piping Resulting in Pipe Whip and Containment Impact/Failure, with Resultant Failure of All Emergency Core Cooling Systems

From NUREG-1150, the DEGB frequency in Feedwater (FW) piping is estimated to be 4 E-4/Rx-Yr; for Main Steam (MS) piping it is estimated as 1 E-4/Rx-Yr. The fraction of FW piping that can impact the containment shell is estimated as 1.0 E-1; the fraction of MS piping is also estimated as 1.0 E-1; this fraction remains 1.0 E-1. The mean probability of a FW or MS system broken pipe rupturing the containment metal shell was 5.0 E-1. The mean probability of additional I&C or ECCS systems failures to the extent that core damage results is estimated as 4.8 E-5 for the case involving FW piping breaks, and 9.8 E-5 for the case involving MS piping breaks.

Based on the above assumptions, the mean value of change in CDF per reactor year is:

dCDF/Rx-Yr = 1.4 E-9

Case 3 (INEEL PWR Event 17): Failure of Main Steam or Feedwater Piping Resulting in Pipe Whip Impact on Component Cooling Water (CCW) System to the Extent That the CCW Pressure Boundary is Broken, Potentially Opening a Path to Outside Containment if Containment Isolation Fails to Occur; Also Possible Loss of CCW Outside Containment for Mitigation

From NUREG-1150, the DEGB frequency in Feedwater (FW) piping is estimated to be 4 E-4/Rx-Yr; for Main Steam (MS) piping it is estimated as 1 E-4/Rx-Yr; this combines for a total of 5.0 E-4. The fraction of FW piping that can impact the CCW system is estimated as 1.0 E-1; the fraction of MS piping is also estimated as 1.0 E-1; this fraction remains 1.0 E-1. The probability of a FW or MS system broken pipe rupturing the CCW system is 5.0 E-1. The probability of failure to close of containment isolation check valve is 1.0 E-3; the probability of failure to close of a containment isolation motor operated valve is 3.0 E-3; this combines for a total of 4.0 E-3. Since the CCW surge tank is in the auxiliary building near where mitigation equipment is, is vented to atmosphere and has a relatively small volume, it is assumed that its water inventory will drain quickly; for this reason the mean probability of opening a path to atmosphere outside containment is 1.0. Once this scenario proceeds to this point the CCW system outside containment will become unavailable, including the RHR heat exchanger; therefore, the probability of losing the CCW function outside containment to the extent that core damage occurs is 1.0.

Based on the above assumptions, the mean value of change in CDF per reactor year is:

dCDF/Rx-Yr = 1.0 E-7

The total change in core damage frequency for the above 3 PWR cases is:

dCDF/Rx-Yr = 7.5 E-5 (Ranks HIGH/MEDIUM in Figure 2 of NUREG-0933)

And, for all 25 affected PWRs:

dCDF/Yr = 1.9 E-3 (Ranks HIGH/MEDIUM in Figure 2 of NUREG-0933)

PWR Offsite Dose Table

GSI-156.6.1 Event Number per NUREG/CR-6395	GSI-156.6.1 dCDF (Events/Rx-Yr)	WASH-1400 Release Category	WASH-1400 Offsite Dose (Person-Rem/ Event)	Offsite Dose (OSD) (Person-Rem/ Reactor Year)
PWR Event 9	7.5 E-5	PWR-6	1.5 E+5	11.3
PWR Event 16	1.4 E-9	PWR-4	2.7 E+6	0.004
PWR Event 17	1.0 E-7	PWR-4	2.7 E+6	0.3
			Total	11.6

For the 25 affected PWRs, the estimated change in offsite dose per reactor (d Person-Rem/Reactor) is:

$$\frac{11.6 \text{ Person-Rem}}{\text{Reactor-Year}} \times 17 \text{ Average Remaining Years} = \frac{197 \text{ Person-Rem}}{\text{Reactor}}{}^{*}$$

(Offsite)

* Ranks MEDIUM/LOW in Figure 2 of NUREG-0933

For 20 years of life extension:

$$\frac{11.6 \text{ Person-Rem}}{\text{Reactor-Year}} \times 37 \text{ Average Remaining Years} = \frac{429 \text{ Person-Rem}}{\text{Reactor}}{}^{*}$$

(Offsite)

*Ranks HIGH/MEDIUM in Figure 2 of NUREG-0933

And the estimated change in offsite dose for the 25 affected PWRs is:

$$\frac{197 \text{ Person-Rem}}{\text{Reactor}} \times 25 \text{ Affected PWRs} = 4{,}925 \text{ Person-Rem}^{*}$$

(Total Offsite, All Affected PWRs)

*Ranks HIGH/MEDIUM in Figure 2 of NUREG-0933

For 20 years of life extension:

$$\frac{\text{429 Person-Rem}}{\text{Reactor}} \times \text{25 Affected PWRs} \quad = \quad \text{10,725 Person-Rem* (Total Offsite, All Affected PWRs)}$$

*Ranks MEDIUM in Figure 2 of NUREG-0933

The estimated total offsite dose for the 41 affected plants (BWRs and PWRs) is:

16,464 + 4,925 = 21,389 Person-Rem* (Total Offsite, All Affected Reactors w/o life extension)

*Ranks MEDIUM in Figure 2 of NUREG-0933

35,824 + 10,725 = 46,549 Person-Rem* (Total Offsite, All Affected BWRs & PWRs w/ life extension)

*Ranks HIGH/MEDIUM in Figure 2 of NUREG-0933

Cost Estimate

Industry Cost: Implementation of the possible solution is assumed to require the performance of engineering analyses inside containment, perform system walkdowns, and provide a report to the NRC. Ultimately, it is expected that operating procedures and/or technical specifications will be modified, inservice inspections will be enhanced, or physical modifications will be done either to piping (probably addition of pipe whip restraints or jet shields) or to the inside containment leakage detection system. It is expected that the cost to each plant will be $1M. Therefore, for the 41 affected plants (16 BWRs and 25 PWRs) the total implementation cost is estimated to be $41M. This estimate was based on the presumption that the level of effort at the affected plants would be similar to that which resulted for this issue during the SEP program review of the 10 earliest SEP plants.

NRC Cost: Development and implementation of a resolution is estimated to cost $1M; primarily involving review of industry submittals and possible proposed changes to hardware.

Total Cost: The total industry and NRC cost associated with the possible solution is $42M.

Impact/Value Assessment

$$S = \frac{\text{Total Cost (\$)}}{\text{Person-Rem (All Reactors)}}$$

$$= \frac{\text{\$42M}}{\text{21,389 Person-Rem}}$$

= $1960/Person-Rem* w/o Life Extension

*Ranks HIGH in Figure 2 of NUREG-0933

$$S = \frac{\text{Total Cost (\$)}}{\text{Person-Rem (All Reactors)}}$$

$$= \frac{\$42M}{46{,}549 \text{ Person-Rem}}$$

$$= \$900/\text{Person-Rem}^* \quad \text{w/ 20 Years of Life Extension}$$

*Ranks HIGH in Figure 2 of NUREG-0933

OTHER CONSIDERATIONS

1. The Updated Safety Analysis Report for an SEP-III BWR (i.e., one of the 41 plants potentially affected by this issue) stated that, in the event of a DEGB, the broken pipe would strike the Mark I Containment and deform it significantly. However, another BWR of about the same vintage is known to have been required to add energy absorbing structures to protect the Mark I Containment from pipe whip, prior to receipt of an operating license. Therefore, it appears that there is considerable variation among the affected plants regarding the amount of pipe whip protection provided.

2. Pipe breaks have actually occurred in the industry. Examples include a Surry Feedwater line break, a WNP-2 Fire System valve structural pressure boundary failure, and a Ft. Calhoun 12" Steam line break.

3. Some suspect configurations were observed in the SEP-III walkdown plants; for example, at one BWR a very close proximity exists between a large RHR (unisolable from R.C.S.) pipe and both banks of the Control Rod Drive piping, and at one PWR it appeared that a large volume of piping penetrated the containment near where a large amount of electrical wiring also penetrated the containment. This demonstrates that even through modest efforts (i.e., sampling walkdowns of a sampling of plants) configurations of potential concern have been identified.

4. Readily available plant documentation provides very little insights regarding actual proximity of high energy piping and potential targets or concern. The potential lack of adequate separation of redundant system targets (e.g., I&C electrical wiring) is also a concern.

5. Uncertainty remains a significant factor because of the large scope of this issue. This is because of the large number and types of plants, and significant differences in the specific as-built details applicable to this issue.

6. Many of the affected plants are either currently applying for life extension or are expected to in the near future. Most of the lead life extension applications will be from the affected plants for many years to come.

7. Although there is a large apparent disparity between the BWR and PWR cases evaluated, it must be remembered that much of the background of this issue was based on sampling walkdowns; that is, only selected portions of selected plants were available for these walkdowns. Therefore, it is important to treat the BWR and PWR evaluations equally during the next phase of the evaluation. Also, some of the listed scenarios seem to have low probabilities but potentially high consequences. They should be further evaluated.

CONCLUSION

Several potential accident scenarios were identified; 7 for BWRs and 3 for PWRs. Mean values for core damage were estimated for each and the cumulative effect of each group was also estimated. When compared to Figure 2 of NUREG-0933, these values mostly showed that this issue is of HIGH/MEDIUM safety significance. Further evaluations which included estimates of offsite doses and costs for potential solutions showed that this issue is of HIGH priority.

REFERENCES

1. Giambusso, A., 1972, AEC Deputy Director for Reactor Projects, Directorate of Licensing, generic letter to applicable BWR and PWR plants regarding postulated pipe failures outside of the containment structure, issued December 1972.

2. O'Leary, J.F., 1973, AEC Director, Directorate of Licensing, generic letter to applicable BWR and PWR plants regarding clarification of A. Giambusso letter of 12/72, issued January through July 1973.

3. Memorandum from E. S. Beckjord (RES) to A. C. Thadani (NRR) dated October 31, 1994.

4. Memorandum from T. E. Murley (NRR) to E. S. Beckjord (RES) dated February 5, 1991. (User Need Letter).

5. NUREG/CR-6395, "Enhanced Prioritization of Generic Safety Issue 156.6.1, Pipe Break Effects on Systems and Components Inside Containment," draft dated June 1999.

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

7. G. S. Holman, C. K. Chou (LLNL), "Assessment of Value-Impact Associated with the Elimination of Postulated Pipe Ruptures from the Design Basis for Nuclear Power Plants," March 29, 1985.

8. NUREG/CR-2800, "Guidelines for Nuclear Power Plant Safety Issue Prioritization Information Development," U.S. Nuclear Regulatory Commission, February 1983, (Supplement 1) May 1983, (Supplement 2) December 1983, (Supplement 3) September 1985, (Supplement 4) July 1986.

9. NUREG/CR-4627, "Generic Cost Estimates," U.S. Nuclear Regulatory Commission, June 1986.

10. NUREG/CR-4550, "Analysis of Core Damage Frequency: Surry, Unit 1 Internal Events," April 1990.

11. NUREG-0933, "A Prioritization of Generic Safety Issues," July 1991.

12. NUREG/CR-4792, "Probability of Failure in BWR Reactor Coolant Piping," Volume 1, March 1989.

13. NUREG-1150, “Severe Accident Risks: An Assessment for Five U.S. Nuclear Power Plants,” December 1990.

Vermont Yankee Nuclear Power Station

cc:

Regional Administrator, Region I
U. S. Nuclear Regulatory Commission
475 Allendale Road
King of Prussia, PA 19406

Mr. David R. Lewis
Shaw, Pittman, Potts & Trowbridge
2300 N Street, N.W.
Washington, DC 20037-1128

Mr. Richard P. Sedano, Commissioner
Vermont Department of Public Service
112 State Street
Montpelier, VT 05620-2601

Mr. Michael H. Dworkin, Chairman
Public Service Board
State of Vermont
112 State Street
Montpelier, VT 05620-2701

Chairman, Board of Selectmen
Town of Vernon
P.O. Box 116
Vernon, VT 05354-0116

Mr. Richard E. McCullough
Operating Experience Coordinator
Vermont Yankee Nuclear Power Station
P.O. Box 157
Governor Hunt Road
Vernon, VT 05354

G. Dana Bisbee, Esq.
Deputy Attorney General
33 Capitol Street
Concord, NH 03301-6937

Chief, Safety Unit
Office of the Attorney General
One Ashburton Place, 19th Floor
Boston, MA 02108

Ms. Deborah B. Katz
Box 83
Shelburne Falls, MA 01370

Mr. Raymond N. McCandless
Vermont Department of Health
Division of Occupational
and Radiological Health
108 Cherry Street
Burlington, VT 05402

Mr. Gautam Sen
Licensing Manager
Vermont Yankee Nuclear Power
Corporation
185 Old Ferry Road
P.O. Box 7002
Brattleboro, VT 05302-7002

Resident Inspector
Vermont Yankee Nuclear Power Station
U. S. Nuclear Regulatory Commission
P.O. Box 176
Vernon, VT 05354

Director, Massachusetts Emergency
Management Agency
ATTN: James Muckerheide
400 Worcester Rd.
Framingham, MA 01702-5399

Jonathan M. Block, Esq.
Main Street
P. O. Box 566
Putney, VT 05346-0566