GI-156.6.1, “PIPE BREAK EFFECTS ON SYSTEMS AND COMPONENTS INSIDE CONTAINMENT”

Introduction

The staff of the U.S. Nuclear Regulatory Commission (NRC) raised Generic Issue (GI) 156.6.1, “Pipe Break Effects on Systems and Components Inside Containment,” in response to the Systematic Evaluation Program (SEP). Modern containment designs are built in conformance with general design criteria, which require that structures, systems, and components important to safety must be appropriately protected against the environmental and dynamic effects that may result from equipment failures, including the effects of pipe whipping and discharging fluids. However, the NRC licensed a number of older plants before these requirements were implemented in the Standard Review Plan (SRP) [Ref. 1]. Thus, GI-156.6.1 deals with the possibility that these older designs may be vulnerable to such interactions.

Historical Background

Overview

GI-156.6.1 is one of 27 distinct generic issues that arose from SEP, which examined a spectrum of concerns applicable to those plants that the NRC licensed before the SRP [Ref. 1] was finalized. Appendix A presents a more detailed history and a list of the affected plants.

GI-156.6.1 specifically deals with the effects of a pipe break within the containment. The SRP [Ref. 1] contains specific criteria for postulated pipe break locations, pipe whip restraints, and separation criteria for instrumentation and control systems, so that a single pipe break does not disable systems and components needed to respond to the event. However, the NRC licensed a number of reactors before those criteria were formally put in place. Although containment designs did not change significantly in response to the SRP criteria, the question arose as to whether the NRC should revisit the adequacy of the older containment designs.

Thus, GI-156.6.1 addresses whether there is a need to re-review the older nuclear power plant units — both pressurized-water reactors (PWRs) and boiling-water reactors (BWRs). Referred to as “Systematic Evaluation Program Phase II and Phase III” (SEP-II and SEP-III), these plants were licensed while the design criteria were still evolving, as specified in Appendix A to Title 10, Part 50, of the *Code of Federal Regulations* (10 CFR Part 50). Specifically, GI-156.6.1 considers whether these SEP-II and SEP-III plant designs adequately addressed the effects of pipe breaks inside containment. In that regard, it should be noted that 51 plants — approximately half of all domestic operating reactors — are included within the scope of this generic issue.

Previous Analyses

Draft NUREG/CR-6395, “Enhanced Prioritization of Generic Safety Issue 156.6.1 Pipe Break Effects on Systems and Components Inside Containment” [Ref. 2], which was prepared by the Idaho National Engineering and Environmental Laboratory (INEEL) and submitted in draft form in November 1999, resulted in GI-156.6.1 being assigned a high-priority rating. That rating was based on conservative outcomes of postulated high energy pipe break scenarios inside containment, which, for BWRs, resulted in drywell shell perforation and loss of all emergency core cooling systems (ECCSs). Nonetheless, Draft NUREG/CR-6395 offered little quantitative or qualitative bases for the origin of the high failure probabilities.

For the BWR case, the NRC benefited from some stakeholder interaction. Specifically, in August 2000, the NRC asked the Boiling-Water Reactor Owner’s Group (BWROG) to review INEEL’s enhanced prioritization [Ref. 2] and provide comments on five of the seven BWR cases. In response, the BWROG formed a committee to coordinate a report for all affected plants, and, in November 2001, the committee issued NEDC-33054, “Conservatism in NRC Prioritization of Pipe Break Effects on Systems and Components” [Ref. 3]. That report was highly critical of Draft NUREG/CR-6395, indicating that GI-156.6.1 should have been prioritized as “Low” or “Drop.”

In preparing Draft NUREG/CR-6395, INEEL assumed a containment impact probability of 0.25 for main steam (MS) and feedwater (FW) piping, and 0.5 for recirculation (RC) and residual heat removal (RHR) piping. However, the BWROG report indicated that the probability of a significant containment impact at a reasonably direct angle (high-energy impact) should be considerably lower than that assumed in Draft NUREG/CR-6395 because of the quantity of piping and structures in the drywell and various break locations.

Similarly, in preparing Draft NUREG/CR-6395, INEEL assumed a containment failure probability of 0.25 for MS and FW piping, 0.5 for RC piping, and 0.1 for RHR piping. However, the BWROG report referenced NRC safety evaluation reports, which indicated that there would be no containment rupture for an unrestrained pipe rupture (e.g., Nine Mile Point Unit 1, Dresden Unit 2), or that pipe ruptures in the cylindrical portion of the drywell would not result in impact energies sufficient to perforate the drywell shell (e.g., Pilgrim and Peach Bottom Units 2 and 3). The BWROG report also claimed that, for BWR Mark I containments, essentially the complete vertical section is backed by reinforced concrete, and greater than 80 percent of the spherical portion is backed by reinforced concrete or is equivalently protected (e.g., pipe penetrations and jet deflectors).

To address these disparate conclusions, Information Systems Laboratories (ISL) produced a report [Ref. 4], which compared the event probabilities stated in Draft NUREG/CR-6395 with those suggested in the BWROG report. In most instances, the ISL report agreed with the BWROG conclusions. However, all three reports (e.g., INEEL, BWROG, and ISL) tended to be subjective, and lacked firm technical bases for the stated conclusions.

Similarly, for PWRs, Draft NUREG/CR-6395 indicated that one event scenario (referred to as Event 9) had a mean core damage frequency (CDF) of 2.54 E-05 per reactor-year (RY). That scenario postulated a high-energy pipe break or jet impingement of the main steam (MS), feedwater (FW), or primary coolant system inside containment. That postulated scenario caused the failure of containment instrumentation and control systems, leading to the failure of accident-mitigating systems. However, Draft NUREG/CR-6395 did not identify other PWR events that had a CDF greater than 1 E-06/RY. Moreover, the analysis of Event 9 was thought to be overly conservative because the actual physical separation conditions of instrumentation and control systems were unknown and, therefore, the analysis was based on certain bounding assumptions.

Current Evaluation

Much of the difficulty experienced in the previous analyses of GI-156.6.1 arose from the generic nature of the safety question — whether the older plant designs adequately addressed the effects of pipe breaks inside containment. Although the issue is restricted to pipe break effects (i.e., pipe whip and jet impingement effects) within containment, the piping layout is highly plant-specific. There are many primary and secondary high-pressure pipes within the containment, and a broken pipe could affect other piping, instrumentation and control cabling, or the wall of the containment. For a specific pipe associated with a specific system, a break at any given location could be assessed for potential targets, the effects on other systems could be estimated, and a specific set of accident sequences could be assessed. However, with the break locations and target locations unknown, the system interactions are similarly unknown, and it is only practical to perform a bounding analysis, which will inevitably involve some degree of conservatism.

Consequently, the INEEL analysis used the following simple formula for each sequence:

$$CDF = (IE)(PIPETYPE)(TYPEFRAC)(RUPTPROB)(SYSTFAIL)$$

Where:

CDF	=	Core damage frequency from the pipe break event in question
IE	=	Pipe break initiating event frequency
PIPETYPE	=	fraction of piping considered in the initiating event that is from the system in question (i.e., RHR, SI, other)
TYPEFRAC	=	fraction of system piping that can cause another system failure from pipe whip or jet impingement
RUPTPROB	=	probability of pipe whip or jet impingement causing another system failure
SYSTFAIL	=	probability of additional systems failing randomly (not attributable to the pipe break) such that core damage occurs

The INEEL analysts examined final safety analysis reports (FSARs) and individual plant examination (IPE) submittals, and actually visited three BWRs and two PWRs to obtain information by direct observation of the relative locations of representative high- and medium-energy piping systems, equipment important to plant safety, and measures taken to mitigate the effects of pipe breaks. Based on the information gathered, the INEEL report narrowed the generic issue to 16 BWR pipe break scenarios and 17 PWR pipe break scenarios. INEEL then quantified those events in additional detail, and dropped all but seven BWR and three PWR scenarios based on low CDF.

The INEEL analysis was generally considered to be excessively conservative [Ref. 3]. However, there was general agreement that the scenarios and sequences that were dropped from consideration as a result of this conservative analysis need not be considered further. Thus, the INEEL analysis (which was, after all, intended as a screening tool) greatly narrowed the scope of GI-156.6.1 to a more tractable number of events. The nature of the surviving scenarios for BWRs differs from those for PWRs. Therefore, the two plant designs are examined separately herein.

BWR Scenarios

The following table lists the seven BWR scenarios identified for further evaluation.

BWR Scenario	Description	Comments
1	This scenario involves a rupture of the MS or FW piping inside the primary containment. Pipe whip causes failure of the containment metal shell. Resulting overpressure in the containment annulus fails all coolant injection systems.	
5	This scenario involves a rupture of the RC piping inside the containment. Pipe whip causes failure of a number of control rod drive (CRD) bundles by crimping the insert/withdraw lines. This results in a large-break loss-of-coolant accident (LOCA) with a failure to scram the reactor.	This event is evaluated under GI-80, and is not included in the scope of GI-156.6.1
9	This scenario involves a rupture of the RC piping inside the containment. Pipe whip causes failure of the containment metal shell. Resulting overpressure in the containment annulus fails all coolant injection systems required to respond to a large-break LOCA.	
10	This scenario involves a rupture of the RHR piping inside the containment. Pipe whip causes failure of a number of CRD bundles by crimping the insert/withdraw lines. This result in a large-break LOCA with a failure to scram the reactor.	This event is evaluated under GI-80, and is not included in the scope of GI-156.6.1
12	This scenario involves a rupture of the RHR piping inside the containment. Pipe whip causes failure of the containment metal shell. Resulting overpressure in the containment annulus fails all coolant injection systems required to respond to a large-break LOCA.	
14	This scenario involves a high-energy line break inside the containment. Pipe whip causes failure of containment instrumentation and control. This is assumed to lead to failure of accident-mitigating injection systems and eventual core damage.	
16	This scenario involves a high-energy line break inside containment. Pipe whip causes failure of the reactor building closed cooling water (RBCCW) system.	INEEL's conservative estimate of CDF for this sequence is 2.0 E-8/RY.

Scenarios 5 and 10 (above) are outside of the scope of GI-156.6.1 and, instead, are addressed under GI-80, “Pipe Break Effects on Control Rod Drive Hydraulic Lines in the Drywells of BWR Mark I and II Containments,” as noted in the preceding table.

Scenario 16 is within the scope of GI-156.6.1, but is of very low frequency. Under the guidelines listed in Appendix C to the Handbook for Management Directive 6.4, “Generic Issues Program,” an issue may be dropped from further consideration if the associated CDF is below 10^{-6}/RY, and the large early-release frequency (LERF) is below 10^{-7}/RY. The sequence involved in Scenario 16 does involve a high likelihood of containment bypass via the RBCCW system, but even if the LERF is equal to the CDF (i.e., containment always fails), the associated LERF and CDF are below the established thresholds.

Scenario 14 involves a high-energy line break within the primary containment that, by pipe whip or jet impact, causes failure of containment instrumentation and control (I&C). INEEL estimated a contribution to CDF of 3.8E-5/RY from this event. However, this estimate is highly conservative because INEEL based its analysis on the following assumptions:

TYPEFRAC	=	0.5	(half of all piping can affect I&C)
RUPTPROB	=	0.75	(there is a 75% probability that all ECCS will be lost as a consequence of pipe whip or jet impingement from the half of all large piping that can reach I&C lines)

Both the BWROG and ISL reviews were highly critical of this estimate, for the following reasons:

- In a BWR, the ECCS would actuate on high containment pressure. The transducers for this signal are connected to the primary containment volume, but are not located within it. This signal would not be adversely affected by pipe breaks within the primary containment volume.
- The level-sensing instrumentation could be affected. However, all BWRs with Mark I containments have two sets of level sensors, located on opposite sides of the reactor vessel. A pipe whip and associated jet would not be likely to affect the sensing lines on the opposite side.
- The only electrical power inside containment required for the functioning of low-pressure coolant injection (LPCI) is the recirculation discharge valve on the unbroken loop. Again, the two recirculation pumps and associated lines are located on opposite sides of the reactor vessel in all BWR/3 and later designs. Moreover, the core spray system would function independently of this valve.

The ISL report recalculated the CDF associated with this scenario to be on the order of 5 E-8/RY. Thus, under the guidelines listed in Appendix C to the Handbook for Management Directive 6.4, this scenario is below the threshold for consideration.

This leaves Scenarios 1, 9, and 12, which must be considered further.

BWR Primary Containment Penetrability

Scenarios 1, 9, and 12 all involve a pipe rupture, which penetrates the primary containment shell. The resulting overpressure in the containment annulus between the containment shell and the containment concrete structure fails all coolant injection systems (for which piping penetrates the containment shell) that are required to respond to a large-break LOCA. This sequence is best understood by referring to the following figure, which was taken from Reference 1.

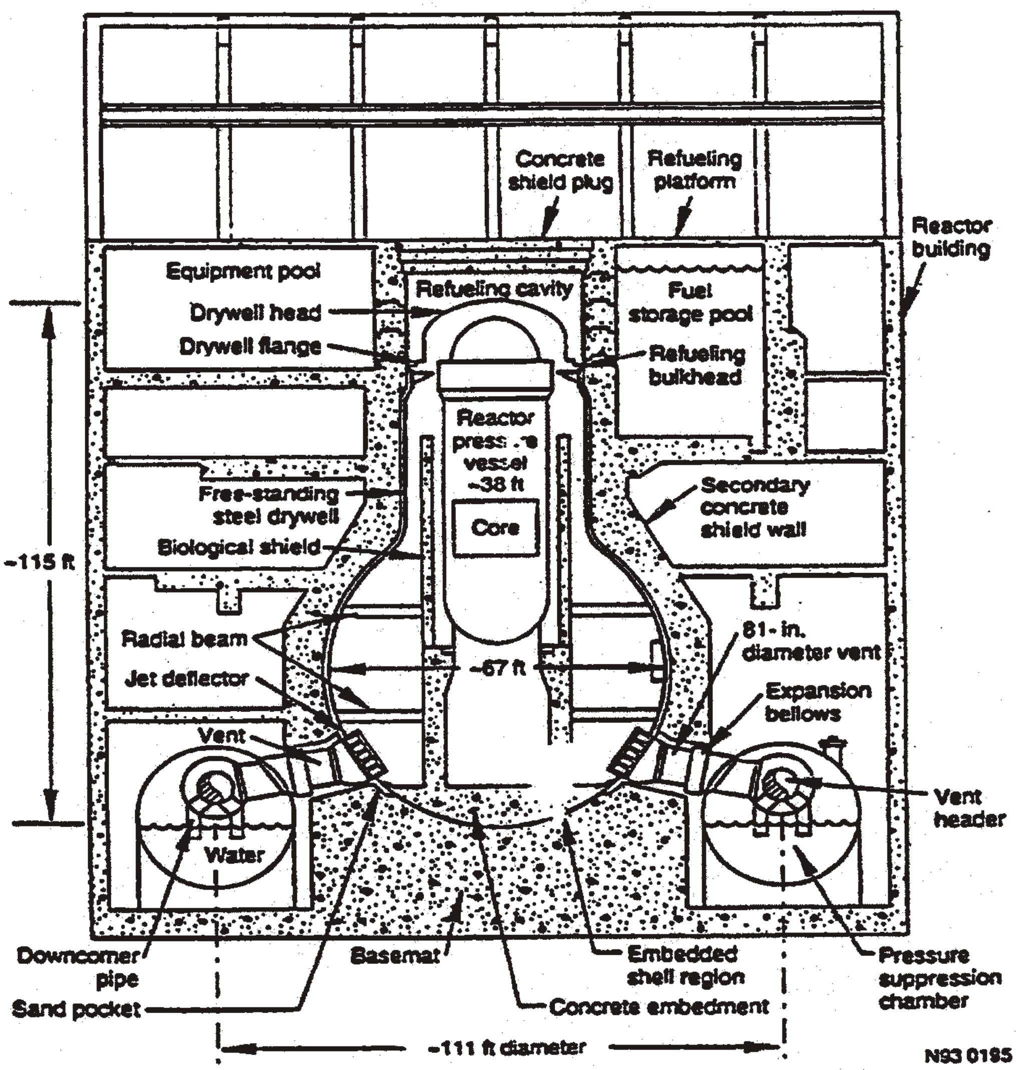

Figure 1. BWR Mark I Containment

With the exception of one plant [Ref. 1], the primary containment wall is a freestanding steel shell, called the "drywell," which is surrounded by a thick [1.2- to 1.8-m (4- to 6-ft.)] concrete secondary shield wall. Except for the basemat area, the areas surrounding the vents leading to the pressure suppression chamber, and the area around the drywell head, there is a gap of 5.1 to 7.6 cm (2 to 3 in.) between the steel drywell and the secondary shield wall to accommodate thermal expansion. Typically, this gap is filled with a compressible material (e.g., Styrofoam or polyurethane) to maintain proper spacing during construction. About half of the plants had the fill removed after construction, while the remaining plants left it in place.

The gap area is sealed at the refueling bulkhead, because the area above this bulkhead must be filled with water during refueling operations. Thus, if the steel drywell wall were penetrated, steam escaping into the gap area would eventually be driven into the gaps around the vents and would escape into the secondary containment area outside of the pressure suppression chamber (often called the "torus"). The ECCS equipment (core spray pumps, LPCI pumps, etc.) all take suction from the torus, and are located in the four corner areas outside of the torus. Thus, the escaping steam is likely to cause failure of the ECCS pumps by creating a harsh environment. Moreover, if the whipping pipe not only penetrates the drywell wall but discharges into the gap, the ECCS could fail as a result of high pressure in the gap area. causing displacement of the drywell and crimping or shearing of the ECCS piping.

References 3 and 4 note that a whipping pipe is unlikely to both penetrate the drywell and discharge into the gap, because the discharging fluid jet would have to be directed away from the drywell wall in order to drive the pipe into the wall to cause an opening. However, this does not completely answer the concern. The drywell pressure would rise to roughly 50 psig after a large LOCA or steam line break, which would cause some discharge into the gap area and presumably into the ECCS pump rooms, particularly in those plants with no filler material in the gap.

To address these concerns in a more deterministic fashion, the NRC staff performed a structural evaluation of the drywell and its ability to withstand an impact from a whipping pipe. In doing so, the staff reviewed several piping systems, including RC, MS, FW, and RHR, to determine where breaks could occur that could impact the drywell shell. Worst-case event conditions involving the largest pipe diameters having the largest energy reservoirs were the main focus of this technical assessment, which was performed using the ANSYS finite element analysis program. Appendix B to this paper provides a more detailed description of this assessment, which examined a break of an MS pipe adjacent to the pressure vessel, an FW line break at the pressure vessel, and a recirculation pump discharge riser pipe at the reactor vessel, all of which are similar designs for BWR/3 and BWR/4 plants. In addition, the staff examined several recirculation pump discharge breaks for the BWR/2 design.

Main Steam Line Break

The main steam line break for the BWR/2 design is illustrative. (Although the analysis was performed for the BWR/2, it is equally applicable to the BWR/3 and BWR/4 product lines with Mark I containments.) This event involves a rupture of an MS pipe with a nominal diameter of 0.6 m (24 in.), located adjacent to the reactor vessel. This break is postulated to occur at the upstream side of the elbow, which would produce a jet force that would accelerate the pipe toward the cylindrical portion of the drywell shell. The pipe is able to travel approximately 0.4 m (16 in.) before impacting the drywell shell. The angle of impact is approximately 3 degrees, and the length of the pipe rotation "arm" is approximately 14 m (46 ft), assuming that a plastic hinge does not develop. The minimum drywell shell thickness is approximately 1.63 cm (0.64 in.) at this location, and the drywell shell has an air gap of approximately 5.1 cm (2 in.) between the concrete drywell shield wall and the drywell shell. The thickness of the concrete at the point of impact is approximately 2.1 m (7 ft).

In conducting this assessment, the staff assumed that the motive force for the pipe to impact the drywell shell, following a double-ended guillotine break, is the blowdown force from the reactor vessel. This force is variable, and depends on the relative size of the pipe break end and the jet stream area as the pipe moves toward the drywell shell. Very little driving force would come from the downstream side of the break (toward the turbine generator) because of the lack of a fluid energy reservoir. The break area is 0.23 m^2 (360 $in.^2$), and the reservoir pressure is 1,050 psia.

In conducting this assessment, the staff used three pipe force methodologies to calculate forces:

(1) Method 1 (Bechtel) [Ref. 5.]

The average force on the main steam pipe from the reactor blowdown is 264,400 lbf. For this scenario, the blowdown coefficient ("C" in the equation $F = CP_0A$) is approximately 0.7.

(2) Method 2 (Moody) [Ref. 6.]

The average force on the main steam pipe from the reactor blowdown is 339,900 lbf. For this scenario, "C" in the equation $F = CP_0A$ is approximately 0.9.

(3) Method 3 (SRP)

The average force on the main steam pipe from the reactor blowdown is 453,000 lbf. For this scenario, "C" in the equation $F = CP_0A$ is approximately 1.2.

The following figure shows the maximum strain in the pipe and the drywell shell as a function of the blowdown coefficient "C." The drywell steel shell will deflect and move radially outward approximately 2.54 cm (1 in.) when impacted by the pipe impacts with the lower-bound force. Because the air gap between the steel drywell shell and the concrete shield wall is 5.1 cm (2 in.), the drywell steel will not come into contact with the concrete shield wall. Local yielding of the steel drywell shell will occur at the point of impact with a maximum strain of 4 percent.

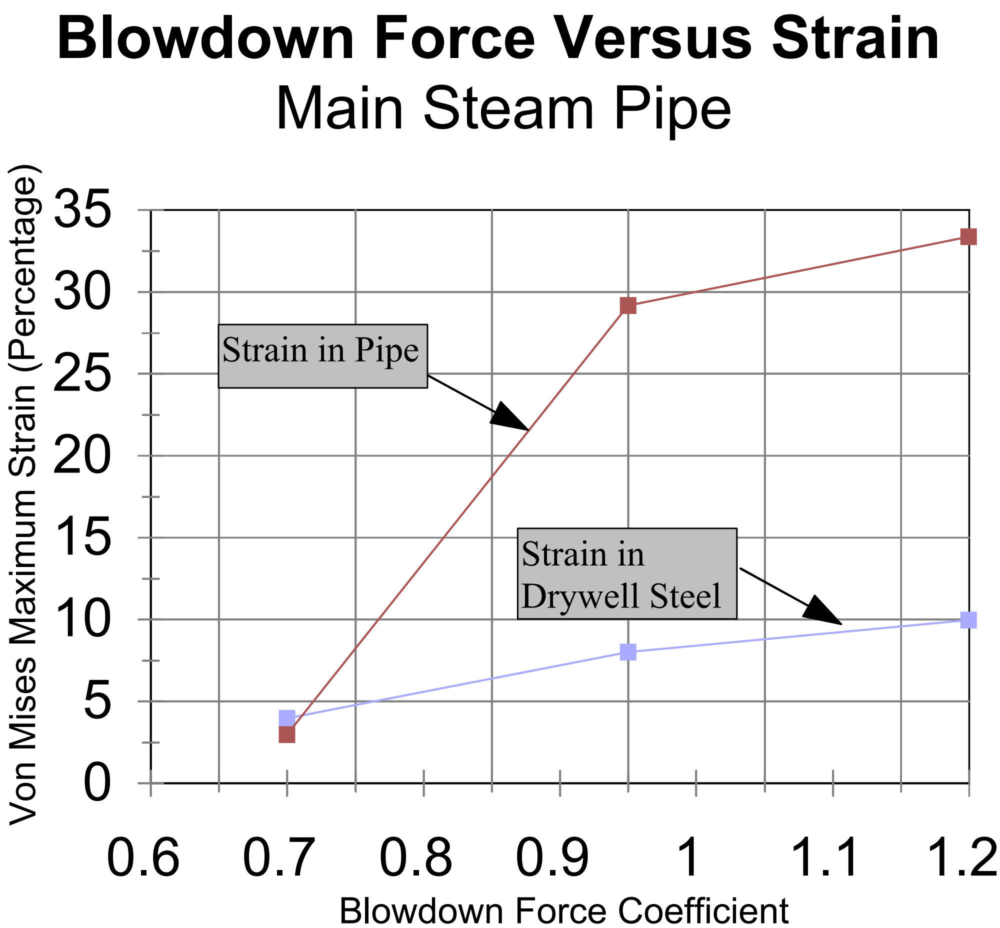

Figure 2. Blowdown Force Versus Strain

Alternatively, if the upper-bound blowdown coefficient is assumed, the steel drywell shell will deflect, move radially outward, and come into contact with the concrete shield wall. The maximum strain in the steel drywell shell is less than 10 percent, as compared to a minimum accepted tensile strain of 17 percent for the carbon steel drywell material. This will result in local yielding of the steel drywell shell. However, the steel drywell shell will not perforate. Moreover, the compressive stresses in the concrete shield wall will remain within allowable elastic limits and will not cause spalling or perforation. (The concrete shield wall will also be subjected to some local tensile stresses at the point of contact with the steel drywell shell. These tensile stresses may cause local cracking of the concrete.)

From the above, it can be concluded that the steel drywell shell will not perforate and the containment integrity will not be compromised due to a postulated break in a steam line at the reactor vessel nozzle.

Other Pipe Breaks

The following tables identifies the full set of break locations that the staff examined, which covers all BWRs with Mark I containments.

Break	Disposition
Main steam pipe adjacent to the reactor pressure vessel	The drywell wall will deflect 2.54 cm (1 in.) with the lower-bound force estimate, and will deflect and contact the concrete at two points with the upper-bound force estimate. The pipe will not cause perforation of the drywell in either instance.
Feedwater system pipe break adjacent to the reactor vessel	A plastic hinge will form after the FW pipe fails. The pipe will not impact the drywell steel shell, which is located 0.6 m (24 in.) away. However, the drywell steel shell would not be perforated even if it were impacted by the FW piping.
Recirculation pumps 11, 13, & 14 discharge pipe break at the reactor vessel (BWR/2)	There is a gap of 4.27 m (168 in.) between the pipe and the steel containment shell. After a guillotine break, the pipe will be propelled toward the containment shell. A plastic hinge will form, and the pipe will fail before it impacts the containment wall.
Recirculation pumps 12 & 15 discharge pipe break at the reactor vessel (BWR/2)	The discharge pipe will first strike the primary containment spray and FW pipes, but is unlikely to impact the drywell wall. Even if it did impact the drywell wall, the discharge pipe is unlikely to have sufficient energy to penetrate.
Recirculation pump discharge riser pipe at the reactor vessel (BWR/3&4)	This break is bounded by the BWR/2 recirculation pump discharge pipe break at the reactor vessel (above) because of its smaller pipe size and reduced system flow.

Likelihood of BWR Containment Perforation

On the basis of the calculations described above, if a whipping pipe strikes the drywell wall, the wall may bulge outward, but will be arrested by the secondary concrete shield wall and will maintain its integrity (i.e., not perforate). Thus, the likelihood of perforation is extremely low, so long as the drywell shell is backed up by the concrete secondary shield.

The cylindrical portion of the drywell is entirely surrounded by the secondary concrete shield wall. However, the BWROG has stated that “greater than 80% of the spherical portion is backed by reinforced concrete or equivalently protected (e.g., pipe penetrations and jet deflectors)” [Ref. 3]. Given that approximately half of the high-pressure piping is located in this spherical portion, this appears to imply that, given a pipe break, there would be about a 10% chance of impacting the containment wall at a location not backed up by concrete.

However, in reality, this likelihood is much less than 10%, because this figure does not include credit for pipe restraints, some intervening structures, and the direction of the jet force. Moreover, the areas not backed up by the concrete secondary shield, such as the pressure suppression vents, personnel and equipment hatches, and steam and feedwater penetrations, would generally direct a discharge into an area other than the gap between the drywell shell and the concrete secondary shield, even if they were penetrated by a whipping pipe. (The penetrations for pipes subject to thermal expansion, such as steam and feedwater pipes, incorporate guard pipes to direct the escaping fluid back into the containment in the event of a pipe break within the penetration.) Thus, it is improbable that a pipe striking one of these areas would lead to ECCS pump failure.

Conclusion — BWR Plants

On the basis of the assessment described above, the staff concluded that there is insufficient basis for any regulatory action regarding the BWR plants within the scope of GI-156.6.1.

PWR Scenarios

The following table lists the three PWR scenarios identified for further investigation.

PWR Scenario	Description	Comments
9	This scenario involves a high-energy line break within containment. Pipe whip or jet impingement causes failure of containment I&C, leading to failure of accident-mitigating systems.	
16	This scenario involves a break in the MS or FW lines, causing penetration of the containment shell as a result of pipe whip.	The only PWRs within the scope of this generic issue that have freestanding steel containments are Kewaunee and Prairie Island. The remaining PWRs have concrete containments with a steel liner.
17	This scenario involves a break in the MS or FW lines, causing failure of the component cooling water (CCW) system as a result of pipe whip.	In some plants, there may be only a single isolation valve at the containment entry and exit for the CCW system. If an isolation valve fails, the cooling water would drain into the containment, eventually leading to a containment-to-atmosphere leak path through the CCW surge tank vent.

Scenario 16

Scenario 16, which is only applicable to two sites, was estimated by INEEL to have a CDF of 1.4 E-9/RY. This scenario was included in the INEEL report primarily because it leads to containment failure and, thus, the event frequency is also a large early-release frequency. Even though the INEEL analysis contains some conservatism, the resulting value is in the range of 10^{-9} because, unlike the analogous BWR scenario, penetration of the steel containment wall does not directly threaten the emergency systems. Moreover, the primary coolant piping is entirely within the crane wall and, thus, the initiating event is a break in the secondary system, rather than a LOCA.

Because of the low estimated frequency, well below the 10^{-7} LERF threshold for generic issues, this scenario will not be considered further.

Scenario 17

For Scenario 17, INEEL estimated a frequency of 1.0 E-7 events per reactor-year, which is right at the LERF threshold. However, the INEEL estimate included some conservatism:

- The INEEL analysis assumed that rupture of the CCW with a failure to isolate would lead to severe core damage. In reality, a break in the MS or FW lines would require sufficient feedwater in the non-faulted steam generators to remove decay heat, but the auxiliary feedwater system would not generally require closed cooling water.
- Additionally, loss of the CCW within containment would imply the loss of cooling water to the reactor coolant pump thermal barrier heat exchangers. If seal injection water [supplied by the chemical and volume control system (CVCS)] was also lost, there would be some likelihood of excessive seal leakage or even seal failure, leading to a small primary system LOCA. Although the pumps in the CVCS may also need CCW, these pumps are part of the ECCS and will be supplied by redundant CCW trains that are isolatable from nonessential CCW loads. Inclusion of this consideration could lower the frequency by an order of magnitude or more.
- The INEEL analysis assumed that a failure in either the upstream or downstream CCW isolation valve would open a path to the environment through the surge tank vent. However, a pipe break in the CCW with a failure in the isolation valves does not automatically lead to a containment bypass. If the upstream isolation valve fails open and the system drains, the path to the atmosphere would involve flow going backward through the pumps, which surely would be equipped with check and shutoff valves. Moreover, there are likely to be maintenance valves even in the CCW return piping. Inclusion of these considerations could lower the estimated frequency by 25% or more.

For these reasons, the staff concluded that a more realistic estimate of the frequency of Scenario 17 would be well below the LERF threshold of 10^{-7} per reactor-year.

Scenario 9

Scenario 9 involves a high-energy line break inside the containment. In this scenario, pipe whip or jet impingement causes failure of containment I&C, leading to failure of accident-mitigating systems. INEEL estimated a CDF of up to 7.5 E-5/RY for this event, which is well above the generic issue screening thresholds.

The various control components [e.g., motor-operated valves (MOVs)] and instrumentation transducers (e.g., pressure sensors, temperature sensors, etc.) will be dispersed, because their locations are dictated by the parameter to be measured. Moreover, such components are generally protected by shields or concrete walls. A pipe break is not likely to cause simultaneous failure in more than one of these components. The concern comes from the fact that the I&C cables servicing these components were often field run in the older plants. Although these plants generally have pipe whip restraints, the fluid jet or spray from a broken high-energy pipe will be quite forceful, and it is quite likely that an electrical cable exposed to such an impingement will be rendered inoperable.

However, there are some mitigating design features for this pipe break scenario. First, an engineered safety features actuation signal will be generated on high containment pressure. This signal will have redundant channels, and will be generated by pressure sensors that are connected to (although not physically located within) the primary containment free volume. Thus, the ECCS, auxiliary feedwater system, etc., will be started regardless of damage to cabling within the containment. The safety-significance of the lost cables will be in long-term recovery, where the operator would monitor signals such as steam line pressure, steam generator level, reactor coolant system (RCS) pressure, pressurizer level, and primary coolant temperature to ensure core cooling and to identify and isolate a faulted secondary loop.

Second, because the signals described in the previous paragraph deal with system components that are separated by walls into their own compartments, it is unlikely that a single pipe break will disable enough of these signals (which also are multi-channel) to leave the operator "blind." Even field-run cables will not likely be located immediately adjacent to cabling associated with the other signals except at the containment electrical penetrations.

Third, the electrical penetrations are where a pipe whip or fluid jet could disable a significant amount of I&C functions. (Elsewhere, the cabling is more likely to be dispersed as it is routed to various locations within containment.) These penetrations and most of the cabling will be located in the annular region outside of the missile shield wall (or "crane wall"). In this area, the only large piping connected to the primary system would be the accumulator system pipes, which are equipped with check valves located inside of the missile shield. Thus, a LOCA is not likely to involve a pipe whip or fluid jet in the annular area. Instead, the initiating event of interest for this generic issue would be a break in the secondary system, primarily an MS or FW line break.

This is a credible scenario. If the electrical penetrations in the containment wall were located near a steam or feedwater line, and the line were to break in that area, the engineered safety features would still actuate on high containment pressure. However, if many signals were lost, it would be more difficult for the operator to identify and isolate the steam generator associated with the break and bring the plant to cold shutdown.

Because of the variation in containment designs for the early plants, a generic approach to this scenario is not possible. Instead, the staff individually examined the layout of each plant, using available drawings and information obtained from licensing project managers, resident inspection staff, and licensee personnel. The following guidelines were used in gathering information:

- If a plant had two electrical penetration areas not located in the same 90° quadrant, the staff did not further pursue the issue for that plant. For such configurations, it is not credible for one pipe break to disable cabling at both penetrations. (More modern designs, based on the SRP, are required to have such a separation.)
- If a plant had only one electrical penetration area (or two penetration areas within the same quadrant), the staff further examined the plant to determine whether a high-pressure pipe is within line-of-sight of the penetration area. If the pipes are far enough along the containment wall to be out of sight, or if there is an intervening wall, floor, or shield capable of stopping a fluid jet, the staff did not further pursue the issue for that plant.

The following table summarizes the results of this multi-plant investigation.

Plant	**Physical Arrangement**	**Conclusion**
ANO-1	The MS piping and containment penetrations are routed above the operating floor, away from the electrical penetration area. The FW containment penetrations are located below the electrical penetrations in the same quadrant, but are separated by a concrete floor slab.	Acceptable
Calvert Cliffs 1	Two distinct electrical penetration areas are located 180° apart and 90° out from the MS and FW piping penetrations.	Acceptable
DC Cook 1	There are two electrical penetrations, not in the same quadrant, and separated from the MS and FW piping by concrete floors and walls.	Acceptable
Fort Calhoun	MS and FW penetrations are located above the operating floor and are separated from electrical penetrations by the floor physical boundary.	Acceptable
Ginna	MS and FW piping are located above the operating floor and exit containment above the operating floor away from electrical penetrations, which are below the floor.	Acceptable
Indian Point 2&3	The MS and FW penetrations are located 45° from the electrical penetrations, but the MS and FW penetrations from two steam generators are directly above cable trays. However, the licensee has an analysis that shows the higher-stress areas of the pipes are at the containment penetration, and they will break there instead of elsewhere; therefore, the jet impingement will not impact the penetration area. From an analysis of plant layout diagrams, the likely spray pattern will not impact the cables.	Acceptable
Kewaunee	Two sets of electrical penetrations are located approximately 90° apart. MS and FW piping penetrations are located above and between the two electrical penetration areas. However, it is unlikely that fluid jets from midway between will disable both.	Acceptable

Plant	Physical Arrangement	Conclusion
Millstone 2	There are two electrical penetration areas 90° apart. MS and FW containment penetrations are located above the operating floor, and the electrical penetrations are located below the operating floor. Therefore, the MS and FW pipes are separated from the electrical cable by the operating floor structure.	Acceptable
Oconee 1,2,3	There are two containment building electrical penetration areas 180° apart.	Acceptable
Palisades	There are two containment penetration areas, separated by more than 90°. Also, one of the electrical penetration areas is shielded by the fuel transfer canal.	Acceptable
Point Beach 1&2	There are two electrical penetrations, but they are in the same quadrant and near the MS and FW penetrations. Most of the penetrations are shielded from the piping by a concrete floor, but one bank of pressurizer heaters and one channel of resistance temperature detectors and incore thermocouples could be disabled by a fluid jet. This could interfere with the plant operators' actions during recovery. However, a pipe break in the vulnerable location is of fairly low probability, and a random failure of the shielded channels would reduce the probability to a very low level. Nonetheless, the licensee agreed to examine its emergency operating procedures to ensure an appropriate response to such an accident.	Acceptable
Prairie Island 1&2	Electrical penetrations are in two groups which are in different quadrants, although one group is near a steam/feedwater penetration area.	Acceptable
Robinson 2	MS and FW piping penetrations are grouped in an area away from the electrical penetration area. There are intervening structures and floor slabs between the MS and FW penetrations and the electrical penetration area. There is a second piping penetration area which is close to the electrical penetration area; however, those pipes are not high-energy, and the adjacent control system cables are not required to mitigate a break in one of those pipes.	Acceptable
Surry 1&2	The units have one electrical penetration area. However, there is no high-energy line within line-of-sight of the penetration area.	Acceptable
TMI-1	High-energy lines are separated from cable penetrations by quadrant and elevation.	Acceptable
Turkey Point 3&4	There is only one electrical penetration area, but the FW lines are not in the same quadrant, and the MS lines are separated from the electrical penetration by a floor.	Acceptable

Conclusion — PWR Plants

Thus, based on this investigation, is the staff concluded that Scenario 9 does not pose a significant safety problem at any PWR within the scope of this generic issue.

Conclusion

On the basis described above, the staff concluded that further pursuit of this generic issue is not justified. Therefore, the staff recommends that GI-156.6.1 be closed out.

References

(1) NUREG-0800, “Standard Review Plan for the Review of Safety Analysis Reports for Nuclear Power Plants,” U.S. Nuclear Regulatory Commission, Washington, DC, 1st Edition: November 1975, 2nd Edition: March 1980, 3rd Edition: July 1981.

(2) Draft NUREG/CR-6395, “Enhanced Prioritization of Generic Safety Issue 156.6.1 Pipe Break Effects on Systems and Components Inside Containment,” U.S. Nuclear Regulatory Commission, Washington, DC, November 1999.

(3) NEDC-33054, “Conservatism in NRC Prioritization of Pipe Break Effects on Systems and Components,” Boiling-Water Reactor Owner’s Group (BWROG), San Jose, California, November 2001.

(4) “Review of Event Probabilities and Frequencies Used in NUREG/CR-6395 Enhanced Prioritization of Generic Safety Issue 156.6.1, Pipe Break Effects on Systems and Components Inside Containment,” Information Systems Laboratories, Rockville, Maryland, December 2002.

(5) BN-TOP-2 (Rev. 2), “Design for Pipe Break Effects,” Bechtel Power Corporation, San Francisco, California, May 1974.

(6) Moody, F. J., “Maximum Flow Rate of a Single Component, Two-Phase Mixture,” Journal of Heat Transfer, Trans American society of Mechanical Engineers, 87, No. 1, February 1965.

APPENDIX A

HISTORY OF GENERIC ISSUE 156.6.1

Systematic Evaluation Program

In 1977, the U.S. Nuclear Regulatory Commission (NRC) initiated the Systematic Evaluation Program (SEP) to review the designs of 51 older, operating nuclear power plants. The SEP was divided into two phases. In Phase I, the NRC staff defined 137 issues for which regulatory requirements had changed sufficiently over time to warrant evaluation of those plants that the NRC licensed before the issuance of the Standard Review Plan (SRP) [Ref. A-1]. In Phase II, the staff compared the designs of 10 of the 51 older plants to the criteria defined in the SRP, as it was issued in 1975. (The following table lists those 10 plants, which were referred to as the "SEP-II plants.") Then, on the basis of those reviews, the staff identified 27 of the original 137 issues that required some corrective action at one or more of the 10 plants that were reviewed. The staff referred to the issues on this smaller list as the SEP "lessons learned" issues and concluded that they would generally apply to operating plants that received operating licenses before the NRC issued the SRP in 1975.

SEP-II PLANTS

Plant	Vendor	Reactor	Current Status
Big Rock Point	General Electric (GE)	BWR	permanently shut-down
Dresden 2	GE	BWR	operating
Ginna	Westinghouse	PWR	operating
Haddam Neck	Westinghouse	PWR	permanently shut-down
LaCrosse	Allis-Chalmers	BWR	permanently shut-down
Millstone 1	GE	BWR	permanently shut-down
Oyster Creek	GE	BWR	operating
Palisades	Combustion Engineering (CE)	PWR	operating
San Onofre 1	Westinghouse	PWR	permanently shut-down
Yankee Rowe	Westinghouse	PWR	permanently shut-down

In SECY-84-133, "Integrated Safety Assessment Program (ISAP)," dated March 23, 1984 [Ref. A-2], the NRC staff presented the 27 SEP issues to the Commission as part of a proposal for an ISAP, the intent of which was to review safety issues for a specific plant in an integrated manner. Two SEP plants participated in the ISAP pilot efforts. Following the review of those two pilot plants, the staff discontinued the ISAP.

Then, in SECY-90-160, "Proposed Rule on Nuclear Power Plant License Renewal," dated May 3, 1990 [Ref. A-3], the staff forwarded for Commission approval a proposed License Renewal Rule and supporting regulatory documents. In that paper, the staff stated that certain unresolved safety issues could weaken the generic justification of the adequacy of the current licensing basis argument. Specifically, those issues included SEP topics for 41 older plants that had not been explicitly reviewed under Phase II of the SEP. In response, the Commission asked the staff to keep the Commissioners informed of the status of the program to enable them to understand how the SEP "lessons learned" issues had been factored into the licensing bases of operating plants.

Resolution of the 27 SEP issues was deemed by the staff to be important to the development of the license renewal rulemaking. The key regulatory principle underlying the License Renewal Rule is that, with the exception of age-related degradation, *the current licensing bases (CLBs) at all operating nuclear power plants provide adequate protection of public health and safety*. This principle is reflected in the provisions of the License Renewal Rule, which limit the renewal decision to whether age-related degradation has been adequately addressed to ensure continued compliance with a plant's CLB. In order to adopt this approach, the NRC must be able to provide a technical basis for the key principle of license renewal. Accordingly, the rulemaking included a technical discussion documenting the adequacy of the CLB for all nuclear power plants, in both the statement of considerations and NUREG-1412, “Foundation for the Adequacy of the Licensing Bases” [Ref. A-4]. However, as discussed in SECY-90-160 [Ref. A-3], the staff identified a potential weakness in the discussion of the adequacy of the CLB with regard to the 41 older plants that had not been included in the SEP-II list. (Those plants, referred to as the “SEP-III” plants, are identified in the following table. However, it should be noted that, although those 41 plants are commonly called the “SEP-III” plants, there never was a formal third phase of the SEP. Instead, as described later in this paper, the identified issues were incorporated into the Generic Issues Program.) To address this potential weakness, the staff undertook an effort to determine whether each SEP issue either had been or was being addressed by other regulatory programs and activities.

SEP-III PLANTS			
Plant	**Vendor**	**Reactor**	**Current Status**
Arkansas Nuclear One (ANO) 1	Babcock & Wilcox (B&W)	PWR	operating
Browns Ferry 1&2	GE	BWR	operating
Brunswick 2	GE	BWR	operating
Calvert Cliffs 1	CE	PWR	operating
Cooper	GE	BWR	operating
DC Cook 1	Westinghouse	PWR	operating
Dresden 3	GE	BWR	operating
Duane Arnold	GE	BWR	operating
Fitzpatrick	GE	BWR	operating
Fort Calhoun	CE	PWR	operating
H.B. Robinson 2	Westinghouse	PWR	operating

SEP-III PLANTS			
Plant	**Vendor**	**Reactor**	**Current Status**
Hatch 1	GE	BWR	operating
Indian Point 2&3	Westinghouse	PWR	operating
Kewaunee	Westinghouse	PWR	operating
Maine Yankee	CE	PWR	permanently shut-down
Millstone 2	CE	PWR	operating
Monticello	GE	BWR	operating
Nine Mile Point 1	GE	BWR	operating
Oconee 1, 2, & 3	B&W	PWR	operating
Peach Bottom 2&3	GE	BWR	operating
Pilgrim	GE	BWR	operating
Point Beach 1&2	Westinghouse	PWR	operating
Prairie Island 1&2	Westinghouse	PWR	operating
Quad Cities 1&2	GE	BWR	operating
Surry 1&2	Westinghouse	PWR	operating
Three Mile Island 1	B&W	PWR	operating
Trojan	Westinghouse	PWR	permanently shut-down
Turkey Point 3&4	Westinghouse	PWR	operating
Vermont Yankee	GE	BWR	operating
Zion 1&2	Westinghouse	PWR	permanently shut-down

The staff completed this effort and categorized the 27 SEP issues as (1) issues that had been completely resolved (i.e., necessary corrective actions had been identified by the staff, transmitted to licensees, and implemented by licensees); (2) issues that were of such low safety-significance that they required no further regulatory action; (3) issues that were unresolved, but for which the staff had identified existing regulatory programs, such as the Individual Plant Examination (IPE) and the Individual Plant Examination of External Events (IPEEE), that covered the scope of the technical concerns and would resolve the specific SEP issue once implemented; or (4) issues that were unresolved and for which regulatory actions to resolve the issues had not been identified. The 27 SEP issues and applicable regulatory programs were then summarized and presented in SECY-90-343, "Status of the Staff Program To Determine How the Lessons Learned from the Systematic Evaluation Program Have Been Factored into the Licensing Bases of Operating Plants," dated October 4, 1990 [Ref. A-5]. In that paper, the staff concluded that the 22 SEP issues in Categories 3 and 4 remained unresolved for purposes of justifying the adequacy of the CLB for some portion of the 41 older plants.

GI-156.6.1 History

In 1967, the Atomic Energy Commission (AEC), the predecessor of the NRC, published General Design Criteria (GDC) for Nuclear Power Plants for interim use. [The current version of those criteria are set forth in Appendix A to Title 10, Part 50, of the *Code of Federal Regulations* (10 CFR Part 50).] From that time through 1972, the AEC staff's implementation of the GDC required consideration of postulated pipe break effects inside containment. However, given the lack of documented review criteria, the AEC staff's review positions continually evolved.

Review uniformity was finally achieved in the early 1970s, as initiated by a note from L. Rogers to R. Fraley, dated November 9, 1972, which proposed a draft safety guide (the predecessor of today's regulatory guides), entitled "Protection Against Pipe Whip Inside Containment." That draft safety guide contained some of the first documented deterministic criteria, which the staff had used for several years (to varying degrees) as guidelines 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 the use of those deterministic criteria, the staff used non-deterministic guidelines on a plant-specific basis. The draft safety guide was subsequently revised and issued in May 1973 as Regulatory Guide 1.46 [Ref. A-6], which was intended for implementation on a forward-fit basis only.

The AEC also issued two generic letters, dated December 1972 [Ref. A-7] and July 1973 [Ref. A-8], to all licensees and all applicants for construction permits (CPs) or operating licenses (OLs), regarding pipe break effects outside containment. Known as the "Giambusso" and "O'Leary" letters, respectively, those generic letters extended pipe break concerns to locations outside containment, and provided deterministic criteria for break postulation and evaluation of the dynamic effects of postulated breaks. In addition, the generic letters asked all recipients to submit a report to the AEC summarizing the plant-specific analysis of the issue. All operating reactor licensees and license applicants submitted the requested analyses in separate correspondence or updated the safety analysis reports (SARs) for their proposed plants to include the analysis. The staff reviewed the submitted analyses and prepared safety evaluation reports for all plants. Then, in November 1975, the NRC staff published Sections 3.6.1 and 3.6.2 of the SRP [Ref. A-1], which slightly revised the two generic letters introduced above. Thus, after 1975, the staff considered the specific structural and environmental effects of pipe whip, jet impingement, flooding, etc., on systems and components relied on for safe reactor shutdown.

Given that the requirements set forth in the Giambusso and O'Leary letters and related SRP sections are applicable to all affected plants, pipe breaks outside of containment were deemed to be a compliance issue and were not considered in this analysis. Compliance matters are dealt with promptly, rather than awaiting 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 the NRC's Office of Nuclear Reactor Regulation (NRR) by separate correspondence [Ref. A-9], and the remainder of this analysis only addressed pipe breaks inside containment. Toward that end, as a part of its plant-specific reviews in 1975 – 1981, the staff used the guidelines in Regulatory Guide 1.46 [Ref. A-6] for postulated pipe breaks inside containment, and SRP Sections 3.6.1 and 3.6.2 [Ref. A-1] for postulated pipe breaks outside containment. Then, in July 1981, the staff revised SRP Sections 3.6.1 and 3.6.2 to apply both inside and outside containment, thus eliminating the need for Regulatory Guide 1.46, which was subsequently withdrawn.

In 1983–1987, the staff revisited the general issue of pipe breaks inside and outside containment in the SEP. The objective of the SEP was to determine the extent to which 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 that were adequate to convert provisional operating licenses (POLs) to full-term operating licenses (FTOLs). As a result of these SEP reviews, the staff required plants to perform engineering evaluations, technical specification or procedural changes, and physical modifications both inside and outside containment. With regard to in-containment modifications, of the two SEP-II plants evaluated in the original ISAP pilot program (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 that there was a wide spectrum of implementation associated with the original reviews of these early plants for pipe breaks inside containment.

As with the evolution of uniform pipe break criteria (described above), electrical system design criteria also developed over time. Prior to 1974, the staff generally reviewed electrical system designs in accordance with the guidelines provided in the Institute of Electrical and Electronics Engineers Standard IEEE-279 [Ref. A-10]; however, significant variations in interpretations of that document resulted in substantial design differences among plants. In particular, true physical separation of wiring to redundant components was not necessarily accomplished. Consequently, in February 1974, the NRC published Regulatory Guide 1.75, “Physical Independence of Electric Systems” [Ref. A-11], to clarify the requirements.

The remaining question for both the pipe break and electrical system design criteria was, of course, whether to reexamine the designs of the plants that the NRC licensed before those criteria were finalized. This question was turned over to the Generic Issues Program in a memorandum from T.E. Murley (NRR) to E.S. Beckjord [Office of Nuclear Regulatory Research (RES)], dated February 5, 1991 [Ref. A-12] and, in 1994, the issue was identified as a “Medium” priority. Then, in 1999, an enhanced prioritization, based on fairly extensive investigation by the Idaho National Engineering and Environmental Laboratory, concluded that there was a possibility that the issue could have greater safety-significance than estimated by the 1994 prioritization, and the issue went on to the technical assessment stage.

References

(A-1) NUREG-0800, “Standard Review Plan for the Review of Safety Analysis Reports for Nuclear Power Plants,” U.S. Nuclear Regulatory Commission, Washington, DC, 1st Edition: November 1975, 2nd Edition: March 1980, 3rd Edition: July 1981.

(A-2) SECY-84-133, “Integrated Safety Assessment Program (ISAP),” U.S. Nuclear Regulatory Commission, Washington, DC, March 23, 1984.

(A-3) SECY-90-160, “Proposed Rule on Nuclear Power Plant License Renewal,” U.S. Nuclear Regulatory Commission, Washington, DC, May 3, 1990.

(A-4) NUREG-1412, “Foundation for the Adequacy of the Licensing Bases,” U.S. Nuclear Regulatory Commission, Washington, DC, December 1991.

(A-5) SECY-90-343, “Status of the Staff Program to Determine How the Lessons Learned from the Systematic Evaluation Program Have Been Factored into the Licensing Bases of Operating Plants,” U.S. Nuclear Regulatory Commission, Washington, DC, October 4, 1990.

(A-6) Regulatory Guide 1.46, “Protection Against Pipe Whip Inside Containment,” U.S. Nuclear Regulatory Commission, Washington, DC, May 1973.

(A-7) Letter to W. Dickhoner, Cincinnati Gas & Electric Company, from A. Giambusso, U.S. Atomic Energy Commission, Washington, DC, December 18, 1972. (The text of this and similar letters is included in Branch Technical Position SPLB 3-1 in Appendix B to Standard Review Plan 3.6.1, Rev. 2, NUREG-0800, October 1990.)

(A-8) Letter to all applicants, reactor vendors, and architect-engineers, from J. F. O’Leary, U.S. Atomic Energy Commission, July 1973. (The text of this letter is included in Branch Technical Position SPLB 3-1 in Appendix C to Standard Review Plan 3.6.1, Rev. 2, NUREG-0800, October 1990.)

(A-9) Memorandum from E. Beckjord to A. Thadani, “Generic Issue 156-6.1, ‘Pipe Break Effects on Systems and Components” U.S. Nuclear Regulatory Commission, Washington, DC, October 31, 1994.

(A-10) IEEE-279, “Criteria for Protection Systems for Nuclear Power Generating Stations, Institute of Electrical and Electronics Engineers, New York, NY, 1971.

(A-11) Regulatory Guide 1.75, “Physical Independence of Electric Systems,” U.S. Nuclear Regulatory Commission, Washington, DC, February 1974.

(A-12) Memorandum from T.E. Murley (NRR) to E.S. Beckjord (RES), U.S. Nuclear Regulatory Commission, Washington, DC, February 5, 1991.

APPENDIX B

STRUCTURAL EVALUATION FOR GENERIC ISSUE GI-156.6.1

1.0 Purpose

This appendix describes the structural evaluation performed to resolve Generic Issue (GI) 156.6.1, "Pipe Break Effects on Systems and Components Inside Containment."

2.0 Postulated Pipe Breaks

This appendix evaluates the impact of the following postulated pipe breaks inside the drywell of a boiling-water reactor (BWR):

- main steam system pipe line break at the reactor vessel nozzle
- feedwater system line break at the reactor vessel nozzle
- discharge line break at the reactor vessel nozzles of recirculation system pumps 11, 13, and 14
- discharge line break at the reactor vessel nozzle of recirculation system pumps 12 and 15

3.0 Structural Evaluation

3.1 Main Steam Line Break at the Reactor Vessel Nozzle

A double-ended guillotine break in the main steam line is postulated at the reactor vessel nozzle, as shown in Figures B-1 and B-2. The vertical leg of the steam line is located 40.6 cm (16 in.) away from the drywell steel shell containment. The minimum thickness of the drywell steel containment for the various BWRs is 1.63 cm (0.64 in.). There is an air gap of 5.1 cm (2 in.) between the drywell steel containment and the concrete shield wall. The thickness of the shield wall adjacent to the steam line at the break location is approximately 2.1 m (7 ft). The main steam line is fabricated from A106-Grade B material, as defined by the American Society for Testing and Materials (ASTM), with an outside diameter of 61 cm (24 in.) and a wall thickness of 3.3 cm (1.3 in.). The operating pressure (P_o) of the steam line is 1,050 psi, and the pipe break area (A) is 0.23 m^2 (360 $in.^2$).

The main steam pipe at the postulated break location is subjected to a blowdown force, which propels the main steam pipe elbow toward the drywell and away from the reactor vessel. If a pipe whip restraint was not previously installed on the steam pipe line to prevent movement of the pipe, the main steam pipe will travel 40.6 cm (16 in.) until it impacts the drywell steel shell. The magnitude of this blowdown force varies with time and depends on the piping layout and configuration. Therefore, the impact of the main steam pipe on the drywell steel shell was evaluated for the upper- and lower-bound values of blowdown force equal to 1.20 times P_oA (453,000 pounds) and 0.70 times P_oA (264,400 pounds), respectively. In addition, an air gap between the drywell steel shell and the concrete shield wall was conservatively increased to 7.94 cm (3.125 in.), instead of 5.1 cm (2 in.), to account for construction tolerances.

This figure shows the ANSYS finite element analysis model. Only a half-section of pipe and the appropriate portion of the shield wall and drywell steel shell were modeled by using axis-of-symmetry boundary conditions.

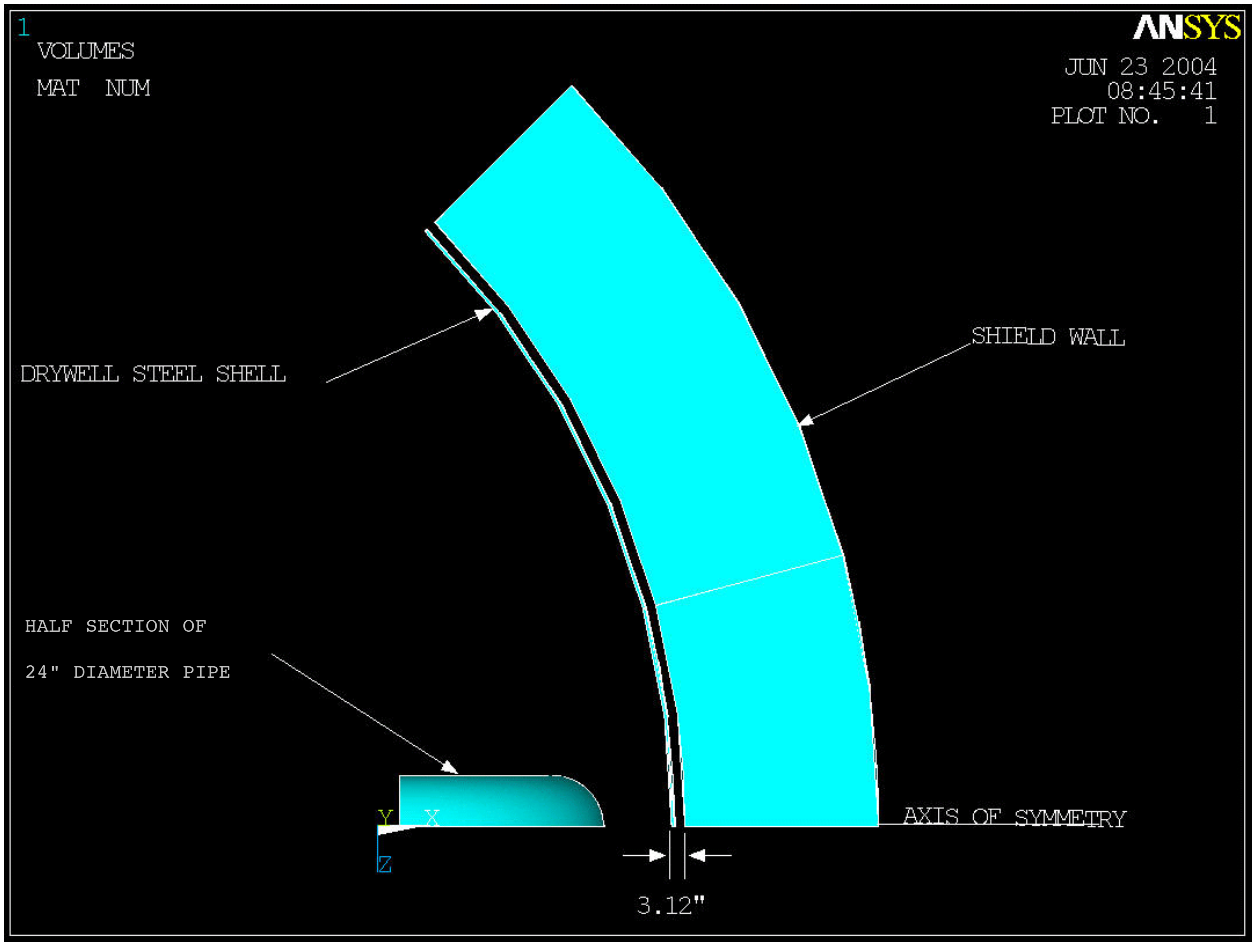

Figure B-1. ANSYS Model of Main Steam Line and Drywell Wall, Top View

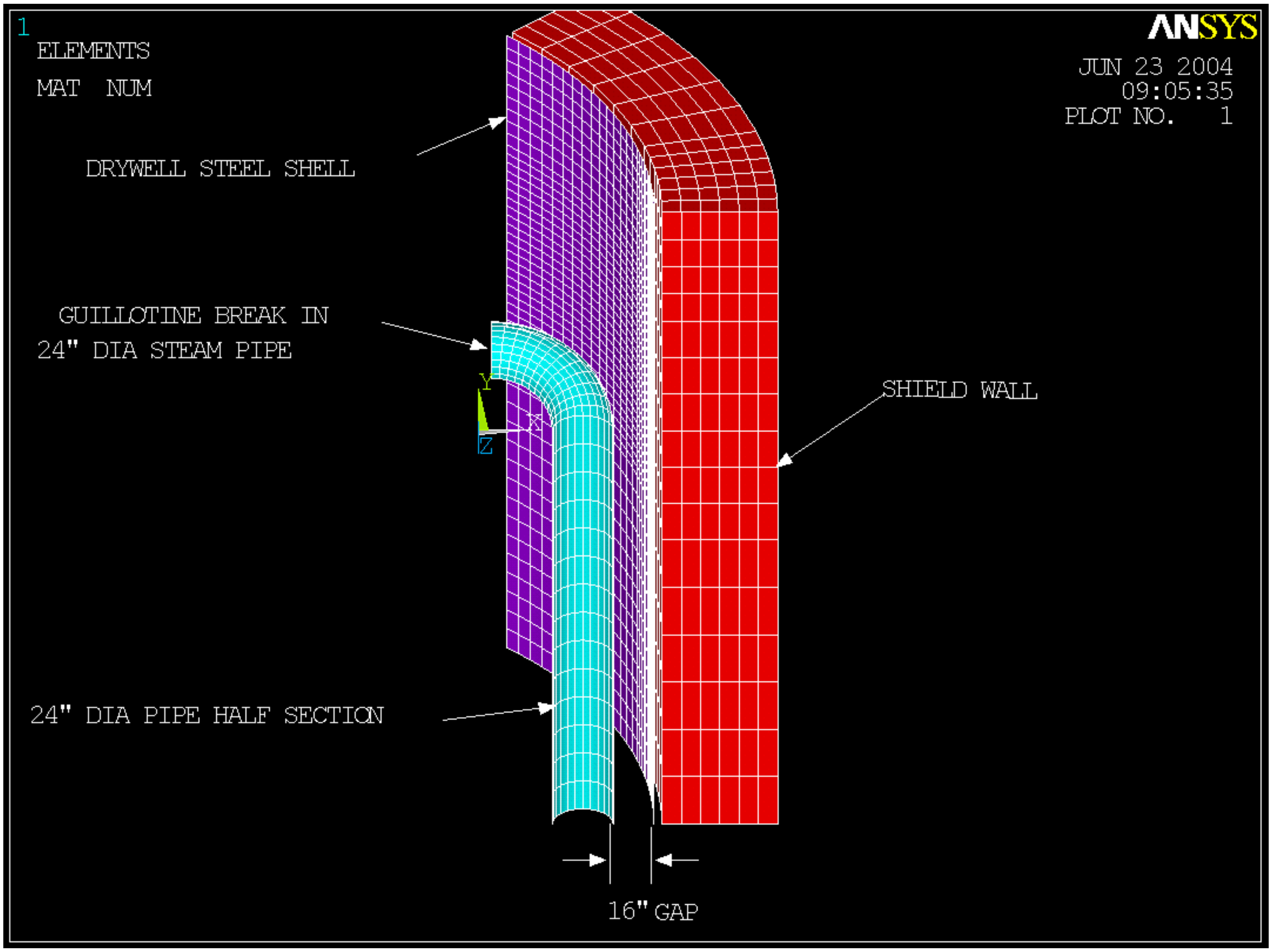

Figure B-2. ANSYS Model of Main Steam Line and Drywell Wall, Side View

Furthermore, the concrete shield wall thickness was also reduced to 114.3 cm (45 in.), instead of 213.4 cm (84 in.), to ensure a conservative structural analysis.

A nonlinear analysis of the main steam line impact on the drywell was performed using the ANSYS finite element computer modeling code. Figures B-3 through B-6 depict the impact of the main steam pipe on the steel drywell shell. The main steam pipe hits the steel drywell and deforms it locally. The maximum deflection in the pipe at the upper- and lower-bound values of the blowdown force was 96.5 cm (38 in.) and 53.3 cm (21 in.), respectively. The maximum Von Mises strain in the pipe resulting from the upper- and lower-bound blowdown force was 33 percent and 4 percent, respectively, compared to the minimum ultimate tensile strain of 22 percent for ASTM A106-Grade B pipe material. Therefore, such an impact causes the pipe to deform, with the possibility that it may fail locally. However, this local deformation or failure is not a concern because the pipe is already damaged before the impact.

The drywell steel shell deflects and moves radially outward about 2.54 cm (1 in.) when the pipe impacts it with the lower-bound blowdown force. However, because the air gap between the drywell shell steel and the concrete shield wall is 5.1 cm (2 in.), the drywell steel does not come into contact with the concrete shield wall. Nonetheless, local yielding of the drywell steel shell occurs at the point of impact with a maximum Von Mises total strain of 4 percent after the steam pipe impacts the lower-bound blowdown force.

The drywell steel shell also deflects, moves radially outward, and comes into contact with the concrete shield wall at two locations after the main steam pipe impacts it with the upper-bound blowdown force. In this case, the maximum Von Mises total strain in the steel drywell shell is less than 10 percent (compared to a minimum tensile strain of 17 percent for the carbon steel drywell material). This results in local yielding of the drywell steel shell; however, the drywell steel shell does not perforate. Figure B-7 illustrates the variation of strain in the steam pipe and drywell steel shell, as a function of blowdown force coefficient. Notably, the compressive stresses in the concrete shield wall remain within the allowable elastic limits and do not cause spalling or perforation. Nonetheless, the concrete shield wall is also subjected to some local tensile stresses at the point of contact with the drywell steel shell, and these tensile stresses may cause local cracking of the concrete.

On the basis of the above discussion, it is logical to conclude that the steel drywell shell will not perforate, and the containment integrity will not be compromised, as a result of a postulated break in a steam line at the reactor vessel nozzle. In addition, the annulus space between the drywell shell and concrete shield wall will not be subjected to any overpressure.

3.2 Feedwater Line Break at Reactor Vessel Nozzle

This scenario postulates a double-ended guillotine break in the feedwater line at the reactor nozzle of a BWR. The vertical leg of the feedwater line is located approximately 61 cm (24 in.) away from the drywell steel containment. The minimum thickness of the drywell steel containment for the various BWRs is 1.63 cm (0.64 in.), and there is an air gap of 5.1 cm (2 in.) between the drywell steel containment and the concrete shield wall. The thickness of the shield wall adjacent to the steam line at the break location is approximately 2.1 m (7 ft).

The FW pipe is fabricated from ASTM A106-Grade B material with an outside diameter of 27.31 cm (10.75 in.) and a wall thickness of 1.59 cm (0.625 in.). The design operating pressure

(P_o) of the feedwater line is 1050 psi, and the pipe break area (A) is 458 cm^2 (71 $in.^2$). The minimum ultimate tensile strain and strength for ASTM A106-Grade B steel pipe material are 22 percent and 60,000 psi, respectively.

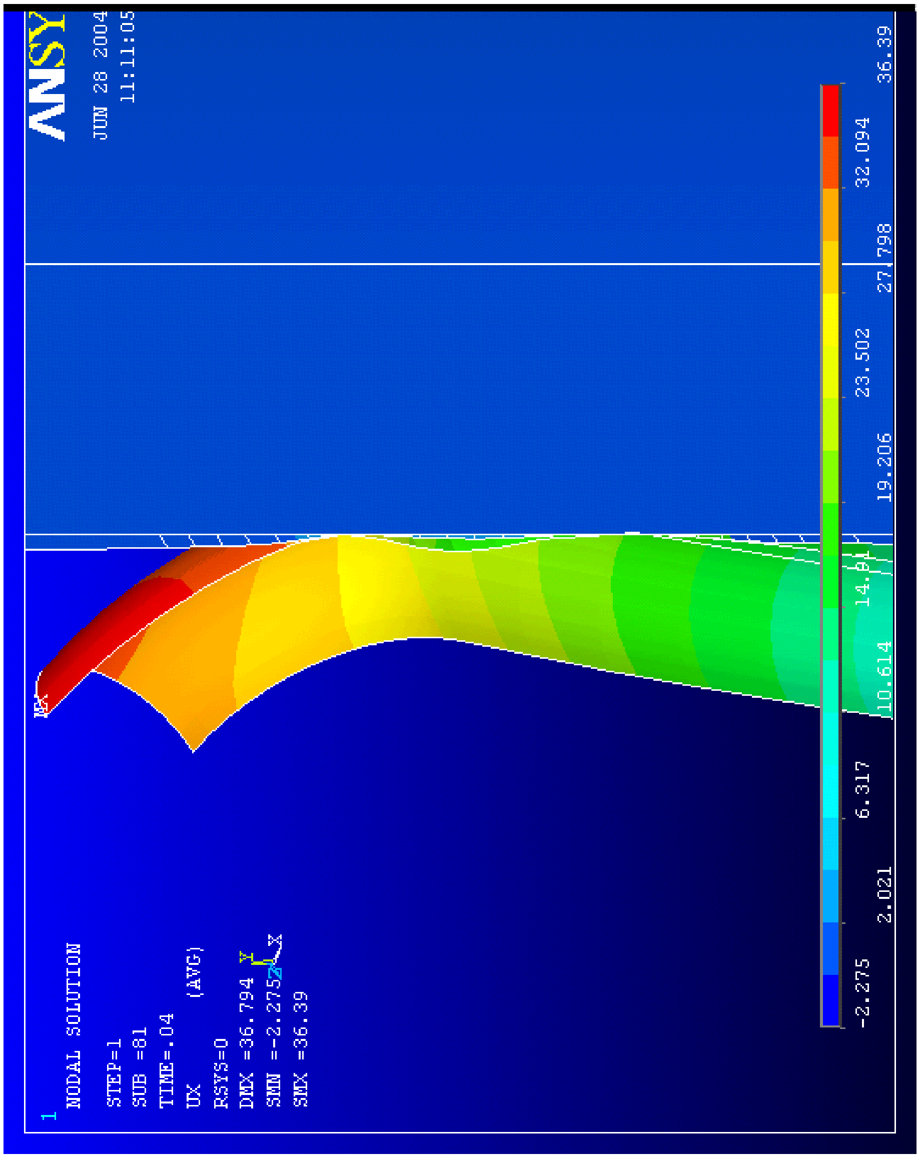

Figure B-3. Deflected Shape for Steam Line Impact
With Upper-Bound Blowdown Force

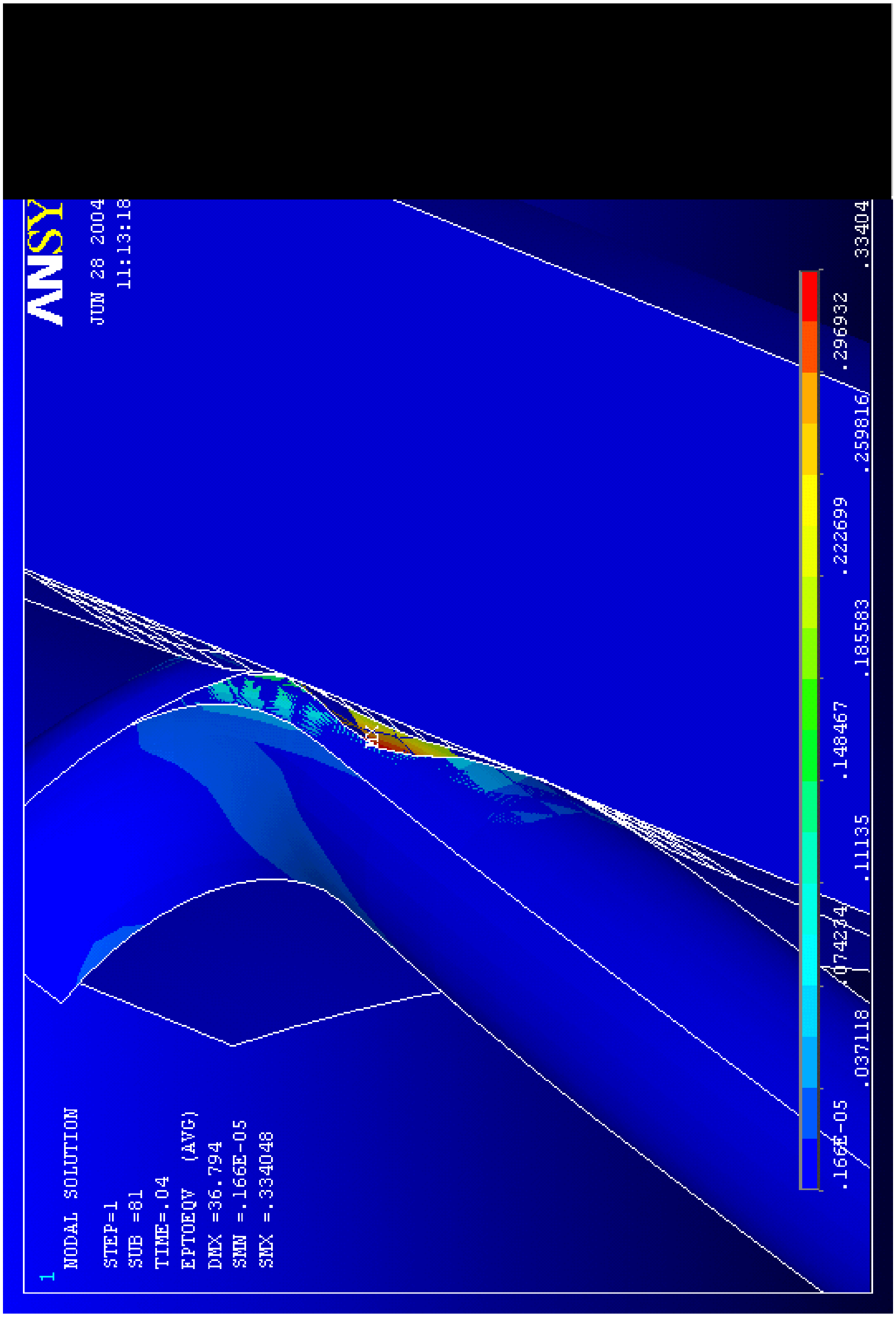

Figure B-4. Maximum Strain for Steam Line Impact
with Upper-Bound Blowdown Force

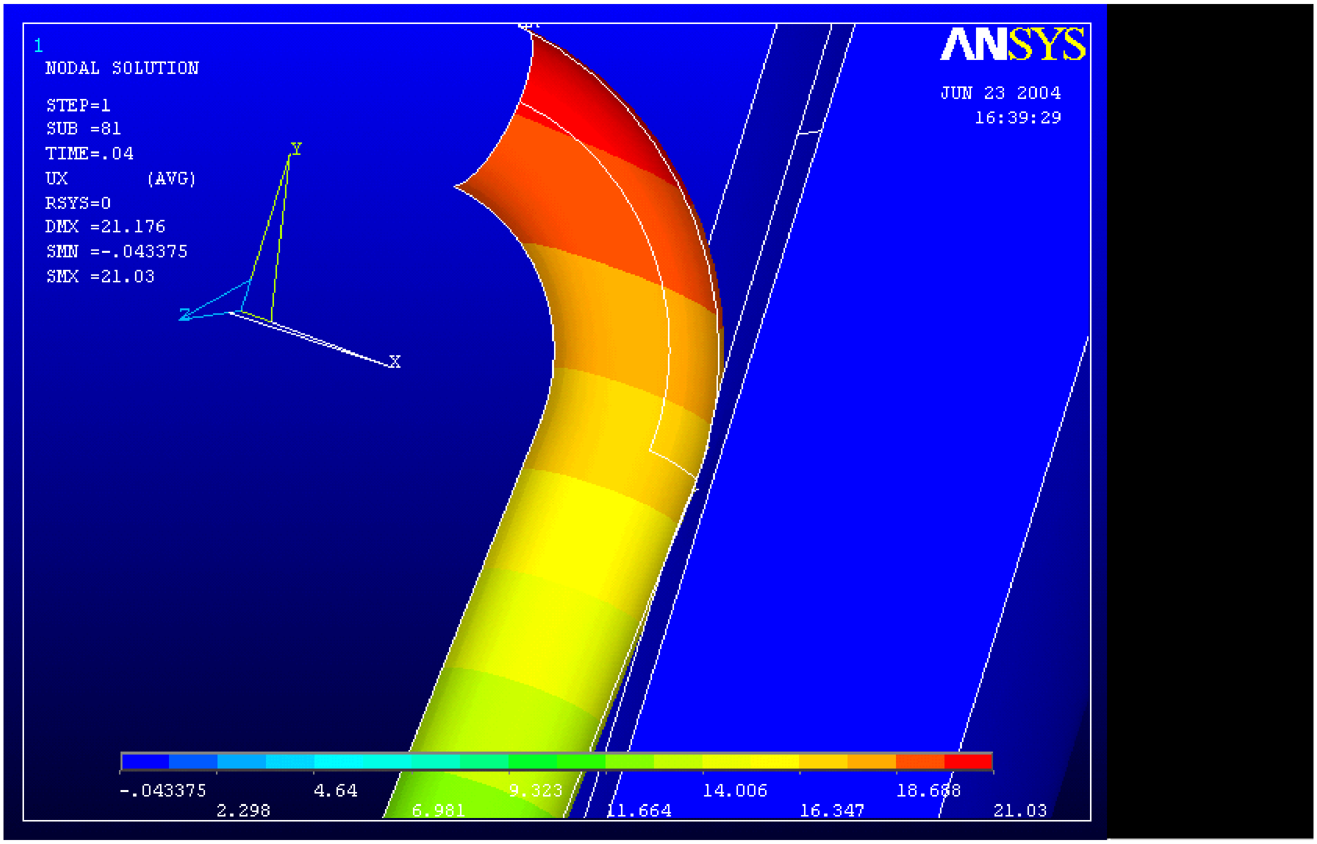

Figure B-5. Deflected Shape for Steam Line Impact
with Lower-Bound Blowdown Force

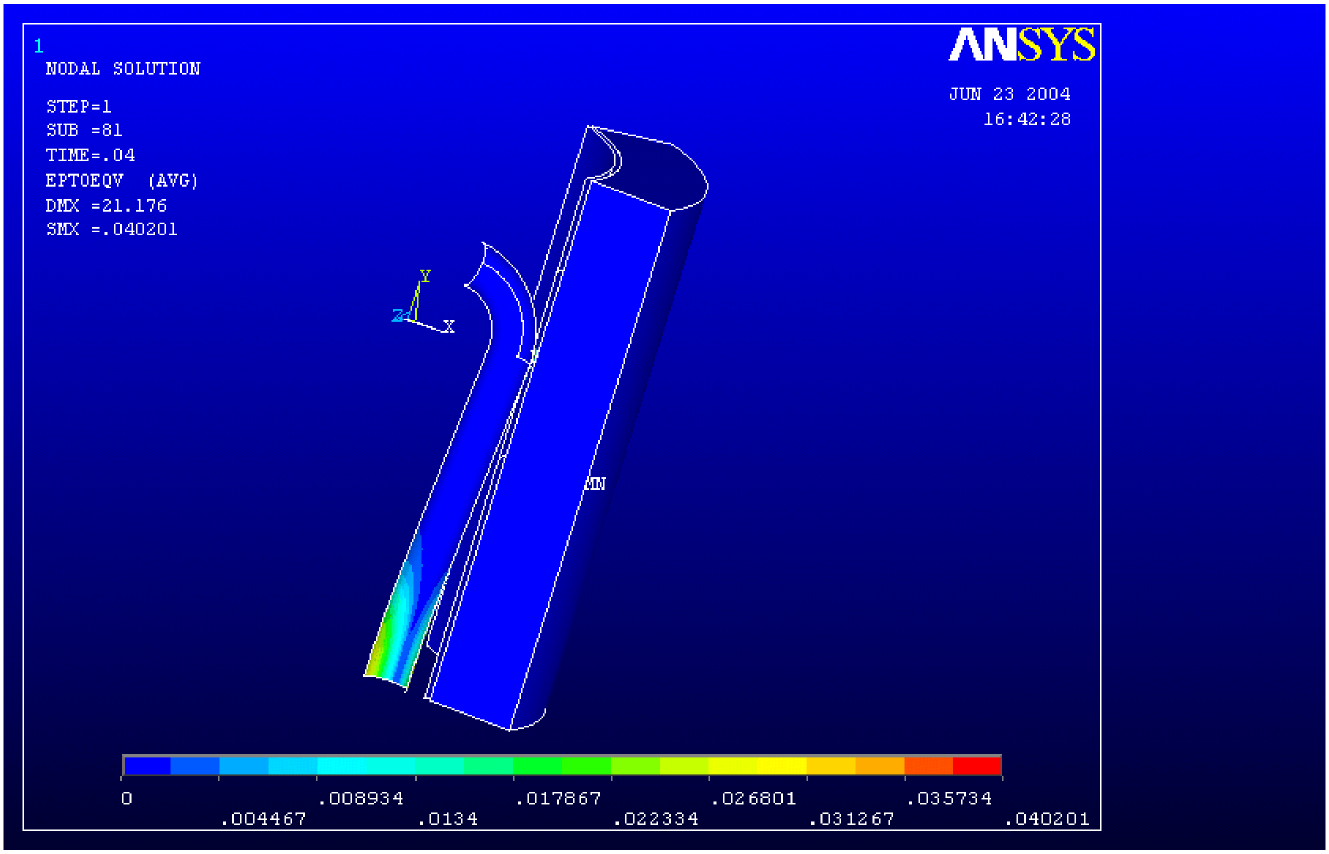

Figure B-6. Maximum Strain for Steam Line Impact
with Lower-Bound Blowdown Force

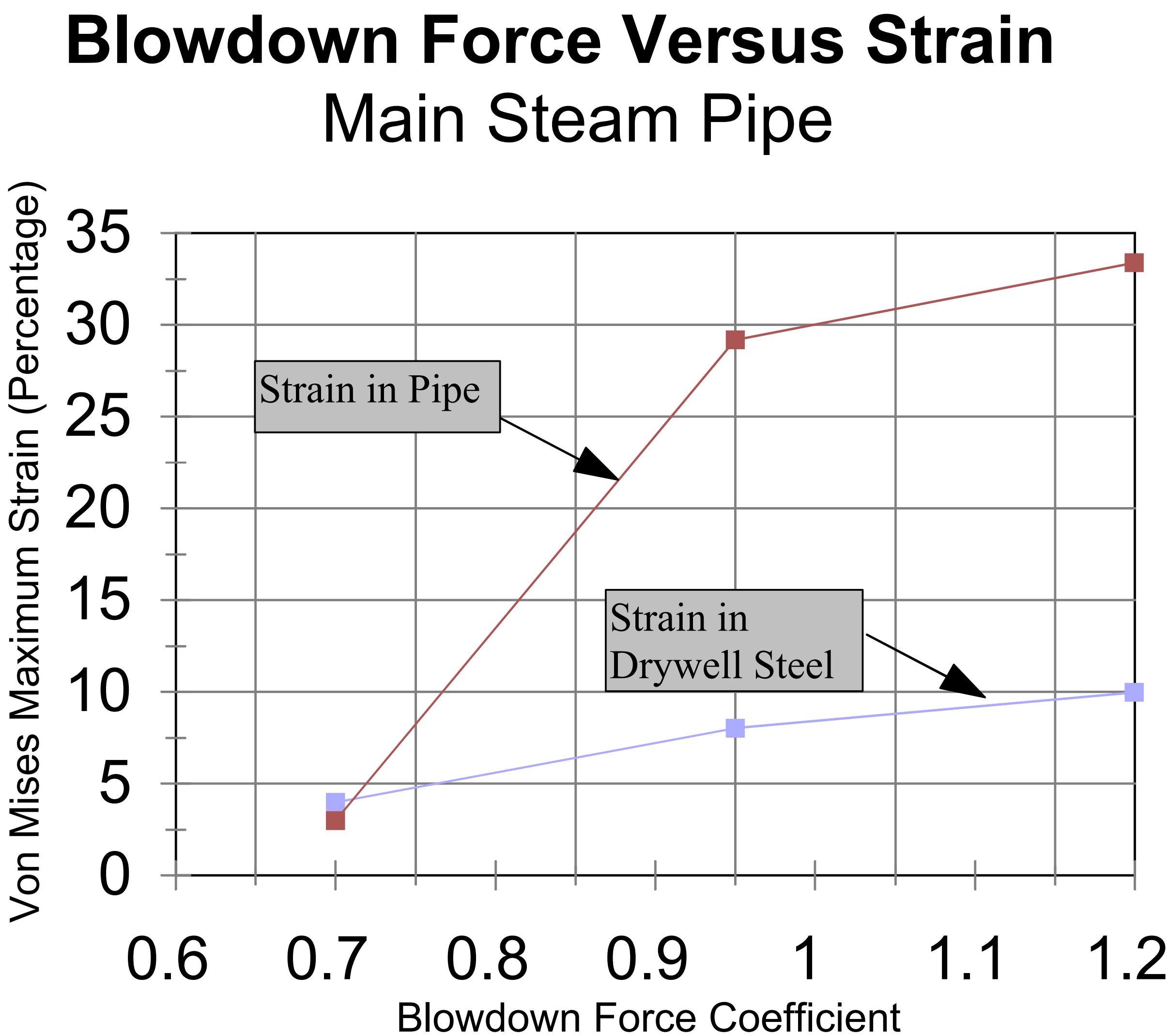

Figure B-7. Blowdown Force versus Strain

The FW pipe at the postulated break location is subjected to a blowdown force, which propels the FW pipe elbow toward the drywell and away from the reactor vessel. If a pipe whip restraint is not installed on the FW pipe line, the FW pipe must travel 61 cm (24 in.) before it impacts the drywell steel shell. The magnitude of this blowdown force varies with time and depends on piping layout and configuration. To ensure a conservative estimate of the total distance that the FW pipe can travel prior to failure, the structural evaluation considered an upper-bound blowdown force of 2.1 times P_oA.

The NRC staff conducted a detailed evaluation of the impact of the blowdown force on the FW pipe and drywell structure. For the upper- and lower-bound values of the postulated blowdown force (156,300 and 96,750 pounds, respectively), the maximum tensile strain of 1.75 times the minimum ultimate strain (38.5 percent), and the ultimate tensile strength of 1.33 times the minimum ultimate strength (80,000 psi) for ASTM A106-Grade B pipe material, a plastic hinge is formed at a distance of 99–163 cm (39–64 in.) below the centerline of the elbow, and the pipe fails after the plastic hinge is formed. However, the pipe does not impact the drywell steel shell because the horizontal deflection of the pipe prior to failure varies between 18 and 46 cm (7 and 18 in.), which is less than the initial gap of 61 cm (24 in.) between the FW pipe and the drywell steel shell.

The failed piece of FW pipe elbow falls inside the drywell and may impact other components. In addition, even if we conservatively consider that the FW pipe does not fail from the blowdown force, and instead travels radially 61 cm (24 in.) to impact the drywell steel shell, the FW pipe does not perforate the drywell steel shell. This is because the energy transferred at impact by the 25.4-cm (10-in.) diameter FW pipe traveling 61 cm (24 in.) is much less than the energy transferred by a 61-cm (24-in.) diameter steam pipe traveling 41 cm (16 in.). As described in Section 3.1 of this appendix, the steam pipe will not perforate the drywell steel shell on impact. Therefore, the structural integrity of the drywell steel containment is not be compromised by a pipe break in the FW pipe line at the reactor vessel nozzle.

3.3 Recirculation System Pumps 11, 13, and 14 Discharge Line Break at the Reactor Vessel Nozzles

This scenario postulates a double-ended guillotine break in the recirculation (RC) discharge lines for pumps 11, 13, and 14 at the reactor vessel nozzles. The vertical leg of the RC lines is located 4.27 m (168 in.) away from the drywell steel shell containment. The minimum thickness of the drywell steel shell containment for the various BWRs is 1.63 cm (0.64 in.), and there is an air gap of 5.1 cm (2 in.) between the drywell steel containment and the concrete shield wall. The thickness of the shield wall adjacent to the discharge line at the break location is approximately 2.1 m (7 ft).

The RC line is fabricated from Type 304/304L stainless steel material with an outside diameter of 71.1 cm (28.0 in.) and a wall thickness of 3.81 cm (1.50 in.). The operating pressure (P_o) of the RC line is 1050 psi, and the pipe break area (A) is 3,168 m^2 (491 $in.^2$). The minimum ultimate tensile strain and strength for Type 304/304L stainless steel are 40 percent and 75,000 psi, respectively.

The RC pipe at the postulated break location is subjected to a blowdown force, which propels the RC pipe elbow radially outward toward the drywell unless a pipe whip restraint has been installed on the RC pipe line to prevent this. However, the RC pipe elbow must travel 4.27 m (168 in.) before it can impact the drywell steel shell. The magnitude of this blowdown force varies with time and depends on the piping layout and configuration. To ensure a conservative approach, this structural evaluation considered upper- and lower-bound values of blowdown force equal to 2.2 times P_oA and 1.30 times P_oA, respectively.

The NRC staff conducted a detailed evaluation of the impact of blowdown force on the RC pipe and drywell structure. For the upper- and lower-bound values of postulated blowdown force (1,130,000 and 638,000 pounds, respectively), maximum tensile strain of 1.75 times the minimum ultimate strain (70 percent), and ultimate tensile strength of 1.2 times the minimum ultimate strength (90,000 psi) for Type 304/304L stainless steel, a plastic hinge is formed at a distance of 254–452 cm (100–178 in.) from the centerline of the elbow, and the pipe fails after the plastic hinge is formed. Nonetheless, the RC pipe does not impact the drywell steel shell because the maximum deflection in the pipe prior to failure varies between 86 and 272 cm (34 and 107 in.), and the initial gap between the RC pipe and drywell steel shell is 4.27 m (168 in.). The failed piece of RC pipe elbow falls inside the drywell and may impact other components, but does not compromise the structural integrity of the drywell steel containment.

3.4 Recirculation System Pump 12 Discharge Line Break at the Reactor Vessel Nozzle

This scenario postulates a double-ended guillotine break in the RC discharge lines for pump 12 at the reactor vessel nozzle. The vertical leg of the RC lines is located 4.27 m (168 in.) away from the drywell steel shell containment. The minimum thickness of drywell steel containment for the different BWRs is 1.63 cm (0.64 in.), and there is an air gap of 5.1 cm (2 in.) between the drywell steel containment and the concrete shield wall. The thickness of the shield wall adjacent to the discharge line at the break location is approximately 21 m (7 ft).

The RC line is fabricated from Type 304/304L stainless steel and has an outside diameter of 71.1 cm (28.0 in.) and a wall thickness of 3.81 cm (1.50 in.). The operating pressure (P_o) of the RC line is 1050 psi, and the pipe break area (A) is 3,168 m^2 (491 $in.^2$). The minimum ultimate tensile strain and strength for Type 304/304L stainless steel are 40 percent and 75,000 psi, respectively.

The RC pipe at the postulated break location is subjected to a blowdown force, which propels the RC pipe elbow radially outward toward the drywell unless a pipe whip restraint has been installed on the RC pipe line to prevent this movement. The RC pipe elbow travels radially and breaks the primary containment spray (PCS) and FW lines located in its path before it impacts the drywell steel shell located 4.27 m (168 in.) away.

The magnitude of this blowdown force varies with time and depends on the piping layout and configuration. To ensure a conservative approach, this structural evaluation considered upper- and lower-bound values of blowdown force equal to 2.2 times P_oA and 1.30 times P_oA, respectively.

The elbow of the RC pipe will strike the 15-cm (6-in.) diameter PCS line, which is located 23 cm (9 in.) away. There is a possibility that the RC pipe will break the PCS pipe line and continue to travel outward. The next obstruction in the outward travel path is the FW line, which is located approximately 183 cm (72 in.) away from the PCS line, so the RC pipe may also impact and break the FW pipe. However, the RC pipe does not impact the drywell steel shell because the maximum deflection in the pipe prior to failure varies between 86 and 272 cm (34 and 107 in.), and the initial gap between the RC pipe and drywell steel shell is 4.27 m (168 in.). The failed pieces of the feedwater, RC, and PCS pipe lines fall inside the drywell and may impact other components, but do not compromise the structural integrity of the drywell steel containment.

4.0 Conclusion

The structural integrity and leak-tightness of the drywell steel containment shell and shield wall will not be compromised as a result of postulated breaks in the MS, FW, and RC discharge pipe lines at the reactor vessel nozzles. The drywell steel shell will yield locally and may come into contact with the shield wall. However, the drywell steel shell will not perforate and cause overpressure in the annular space between the steel shell and shield wall. Small portions of the FW, RC, and PCS pipe lines may break and fall onto other components inside the drywell, but will not compromise the structural integrity of the drywell steel containment.