Generic Issue 156.6.1
"PIPE BREAK EFFECTS ON SYSTEMS AND COMPONENTS INSIDE CONTAINMENT"
Executive Summary

Generic Issue 156.6.1 is one of 27 separate generic issues that arose from the Systematic Evaluation Plan (SEP), a program initiated in 1977 which examined a spectrum of concerns applicable to those plants which were licensed before the Standard Review Plan was finalized.

GI-156.6.1 specifically deals with the effects of a pipe break within the containment. The Standard Review Plan (SRP) 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, a number of reactors were licensed before these criteria were formally put in place. Although containment designs did not change greatly in response to the SRP criteria, the question arose as to whether the adequacy of these older containment designs should be revisited.

Thus, this generic issue addresses whether there is a need to re-review the older nuclear power plant units (both PWRs and BWRs) referred to as the "Systematic Evaluation Program Phase II and Phase III" (SEP-II and SEP-III) plants that were licensed while 10 CFR 50 Appendix A design criteria were still evolving. Specifically, Generic Issue 156.6.1 addresses whether the effects of pipe breaks inside containment have been adequately addressed in these plant designs. It should be noted that 51 plants - approximately half of all domestic operating reactors - are included within the scope of this generic issue.

Given a pipe break within containment, with no other assumptions regarding system separation or piping restraints, it is prudent to investigate a failure due to pipe whip or fluid impingement of any system which has components located within the containment building. This would lead to a rather large number of possible accident scenarios. However, an assessment performed by the Idaho National Engineering and Environmental Laboratory in 1999, which was based on several site visits in addition to the examination of a large number of drawings, was able to reduce this possible set of scenarios to seven BWR events and three PWR events.

The BWR events all involve consideration of a pipe whip which may penetrate the steel primary containment wall, thereby discharging steam into the gap between this wall and the secondary concrete shield wall. Steam that is discharged into this gap will likely find its way to the rooms containing the ECCS equipment, which may fail due to the resulting harsh environment. A severe core damage event could result, with the integrity of the primary containment already lost.

To address these events, the staff performed a series of calculations using a nonlinear finite element program to estimate the effect of a pipe whip on the containment wall. The results of these calculations indicated that denting but not penetration of the containment wall would occur. In the more extreme cases, the dented steel containment wall would touch the concrete shield, but this contact would arrest any further displacement. Based on these calculations, it was concluded that penetration of the steel wall will not occur and the BWR events can be eliminated from further consideration.

Of the three PWR events, only one has an estimated frequency high enough to warrant investigation. The event investigated is initiated by a high energy pipe break within

containment. The possibility of pipe whip or jet impingement then causing failure of instrumentation or control cables within containment, leading to failure of accident-mitigating systems was investigated.

Because of the variation in containment designs for these early PWRs, a generic approach was not possible. Instead, the layout of each plant was examined individually. It was discovered that some plant designs anticipated the SRP requirements for channel separation and separate penetrations on opposite sides of the containment. In other plant designs, cables were separated from piping by walls, floors, or other structures, or were spatially separated by significant distances. No instance was found where a whipping pipe or fluid jet would directly disable both channels of any safety-significant system. It was concluded that the PWR events could also be eliminated from further consideration.

Based on the above, it is recommended that Generic Issue 156.6.1 be closed out.

Generic Issue 156.6.1
"PIPE BREAK EFFECTS ON SYSTEMS AND COMPONENTS INSIDE CONTAINMENT"
Technical Assessment

INTRODUCTION

Generic Issue 156.6.1, "Pipe Break Effects on Systems and Components Inside Containment," was raised in response to the Systematic Evaluation Program. Modern containment designs are built in conformance with general design criteria which require 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. However, a number of older plants were licensed before these requirements were implemented in the Standard Review Plan. Thus, this generic issue deals with the possibility that these older designs may be vulnerable to such an interaction.

HISTORICAL BACKGROUND

Overview

Generic Issue 156.6.1 is one of 27 separate generic issues that arose from the Systematic Evaluation Plan (SEP), which examined a spectrum of concerns applicable to those plants which were licensed before the Standard Review Plan was finalized. A more detailed history and a list of the affected plants are provided in Appendix A.

GI-156.6.1 specifically deals with the effects of a pipe break within the containment. The Standard Review Plan (SRP)(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, a number of reactors were licensed before these criteria were formally put in place. Although containment designs did not change greatly in response to the SRP criteria, the question arose as to whether the adequacy of these older containment designs should be revisited.

Thus, this generic issue addresses whether there is a need to re-review the older nuclear power plant units (both PWRs and BWRs) referred to as the "Systematic Evaluation Program Phase II and Phase III" (SEP-II and SEP-III) plants that were licensed while 10 CFR 50 Appendix A design criteria were still evolving. Specifically, Generic Issue 156.6.1 addresses whether the effects of pipe breaks inside containment have been adequately addressed in these plant designs. It should be noted that 51 plants - approximately half of all domestic operating reactors - are included within the scope of this generic issue.

Previous Assessments

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

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

For the BWR case, there was some stakeholder interaction. In August 2000, the NRC requested the Boiling Water Reactor Owner's Group (BWROG) to review INEEL's enhanced prioritization (2) and provide comments on five of the seven BWR cases. The BWROG formed a committee to coordinate a response for all affected plants, and, in November 2001, issued NEDC-33054, "Conservatism in NRC Prioritization of Pipe Break Effects on Systems and Components" (3). The BWROG report was highly critical of the INEEL report, indicating that the prioritization should have been prioritized as "Low" or "Drop."

The INEEL report assumed a containment impact probability of 0.25 for MS and FW piping and 0.5 for recirculation (RC) piping and residual heat removal (RHR) piping. 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 the value used in the INEEL report because of the quantity of piping and structures in the drywell and different break locations.

The INEEL report assumed a containment failure probability of 0.5 for RC piping, 0.25 for MS and FW piping, and 0.1 for RHR piping. In response, the BWROG report made references to NRC safety evaluation reports indicating 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 do 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 that greater than 80 percent of the spherical portion is backed by reinforced concrete or equivalently protected (e.g., pipe penetrations and jet deflectors).

Information Systems Laboratories (ISL) produced a report (4) that compared the event probabilities stated in the INEEL report with those suggested in the BWROG report. In most instances, the ISL report agreed with the statements made by the BWROG. All three reports (e.g., INEEL, BWROG, and ISL) tended to be subjective and lacked a firm technical basis for statements that were made.

EVALUATION

Much of the difficulty experienced in the earlier assessments of this generic issue arises from the fact that the safety question, i.e., whether the effects of pipe breaks inside containment have been adequately addressed in these older plant designs, is very general. 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 containment wall. 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 assessment, which will inevitably involve some degree of conservatism.

The INEEL assessment used a simple formula for each sequence:

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

Where:

CDF	=	Core damage frequency from the pipe rupture event in question
IE	=	Pipe rupture 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 caused by the pipe break) such that core damage occurs

The INEEL analysts examined FSARs and 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 report narrowed the generic issue down to 16 BWR pipe break events and 17 PWR pipe break events. These events were then quantified in more detail, and all but seven BWR and three PWR events were dropped based on low core damage frequency.

The INEEL assessment was generally considered to be excessively conservative [3, 4]. However, there was general agreement that the events and sequences that were dropped from consideration by this conservative assessment need not be considered further. Thus, the INEEL assessment, which was, after all, intended to be a screening tool, greatly narrowed the scope of GI-156.6.1 to a more tractable number of events. The nature of the surviving events for BWRs differs from those for PWRs. Therefore, the two plant designs will be examined separately.

BWR EVENTS

The seven BWR events designated for further evaluation are listed in the following table:

BWR event number	Description	Comments
1	This event involves a rupture of the main steam or feedwater piping inside primary containment. Pipe whip causes failure of the containment metal shell. Resulting overpressure in the containment annulus fails all coolant injection systems.	
5	This event involves rupture of the recirculation piping inside containment. Pipe whip causes failure of a number of CRD bundles by crimping the insert/withdraw lines. The result is a large 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 event involves rupture of the recirculation piping inside containment. Pipe whip causes failure of the containment metal shell. Resulting overpressure in the containment annulus fails all coolant injection systems required for a large LOCA response.	
10	This event involves the rupture of the RHR piping inside containment. Pipe whip causes failure of a number of CRD bundles by crimping the insert/withdraw lines. The result is a large 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 event involves the rupture of the RHR piping inside containment. Pipe whip causes failure of the containment metal shell. Resulting overpressure in the containment annulus fails all coolant injection systems required for a large LOCA response.	
14	This event involves a high-energy line break inside containment. Pipe whip causes failure of containment instrumentation and control. This was assumed to lead to failure of accident-mitigating injection systems and eventual core damage.	
16	This event involves a high-energy line break inside containment. Pipe whip causes failure of the RBCCW system.	INEEL's conservative estimate of core damage frequency for this sequence is 2.0E-8 per reactor-year.

Two of these events (event 5 and event 10) are outside of the scope of this generic issue, and instead are addressed under Generic Issue 80, as noted in the table.

Event 16 is within the scope of GI-156.6.1, but is of very low frequency. Under the guidelines for generic issues listed in Appendix C of the handbook for Management Directive 6.4, an issue may be dropped from further consideration if the associated core damage frequency is below 10^{-6}/reactor-year, and the large early release frequency (LERF) is below 10^{-7}/reactor-year. The sequence involved in Event 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 this threshold.

Event 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. INEEL estimated a contribution to CDF of 3.8E-5/reactor-year from this event. However, this estimate is highly conservative because the assessment assumed that:

TYPEFRAC	=	0.5	i.e., half of all piping can affect I&C, and
RUPTPROB	=	0.75	i.e., there is a 75% probability of all ECCS being lost as a consequence of pipe whip or jet impingement from that 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, in all BWRs with Mark I containments, there are 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 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.

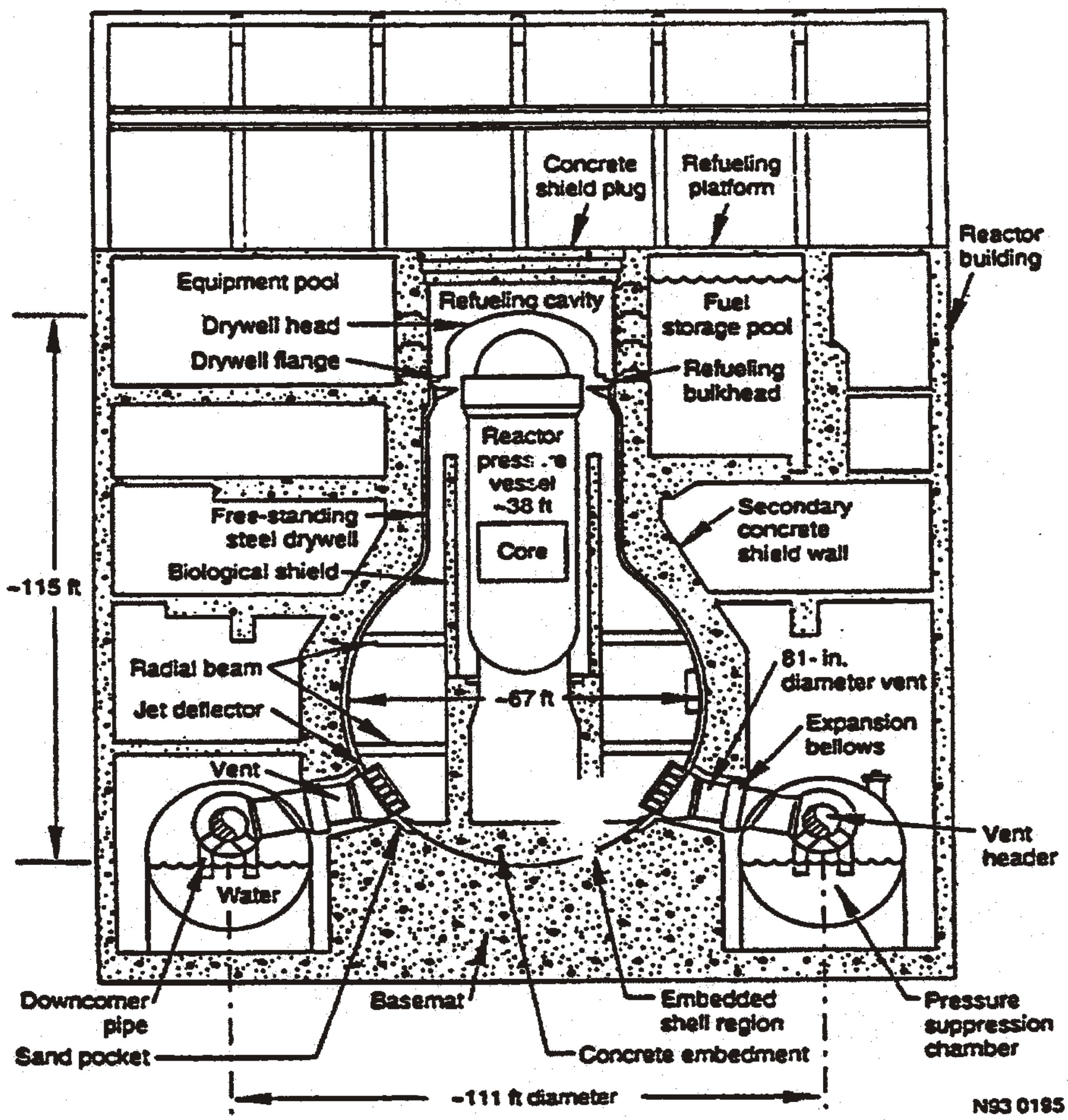
The ISL report recalculates the CDF associated with this event to be on the order of 5E-8 per reactor-year. Under the guidelines for generic issues listed in Appendix C of the handbook for Management Directive 6.4, an issue may be dropped from further consideration if the associated core damage frequency is below 10^{-6}/reactor-year, and the large early release frequency (LERF) is below 10^{-7}/reactor-year. Thus, this event is below the threshold for consideration.

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

BWR primary containment penetrability

Events 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 (whose piping penetrates the containment shell) which are required for a large LOCA response.

These events are best understood by referring to the figure below, which was taken from Reference 2:



The primary containment wall is, except for one plant[1], a freestanding steel shell, called the "drywell." The steel drywell is surrounded by a thick (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 2 to 3 inch gap between the steel drywell and the secondary shield wall to accommodate thermal expansion. Typically,

[1]The Brunswick 2 plant is a Mark I design, but uses a concrete rather than a steel drywell. This design does not have a gap analogous to that of the steel designs, and thus is not susceptible to this scenario.

this gap is filled with a compressible material (e.g., Styrofoam or polyurethane) to maintain proper spacing during construction. About half of the plants have had the fill removed after construction; the remainder have left it in place.

The gap area is sealed above 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 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 due high pressure in the gap area causing displacement of the drywell and crimping or shearing of the ECCS piping.

References 3 and 4 point out that a whipping pipe is unlikely to both penetrate the drywell and discharge into the gap, since 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. Several piping systems, including recirculation, main steam, feedwater, and RHR, were reviewed 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. The ANSYS finite element analysis program was used for this evaluation. A more detailed description of this analysis is provided in Appendix B.

The analysis examined a break of a main steam pipe adjacent to the pressure vessel, a feedwater line break at the pressure vessel, and a recirculation pump discharge riser pipe at the reactor vessel, all of which are similar designs for the BWR/3 and BWR/4 plants. In addition, several recirculation pump discharge breaks were examined 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 a Mark I containment.) This event involves a rupture of a main steam pipe (nominal diameter 24 inches) located adjacent to the reactor vessel. The break is postulated to occur at the upstream side of the elbow, which would produce a jet force that would accelerate the pipe towards the cylindrical portion of the drywell shell. The pipe is able to travel approximately 16 inches prior to impacting the drywell shell. The angle of impact is approximately 3 degrees. The length of the pipe rotation "arm" is approximately 46 feet, assuming that a plastic hinge does not develop. The minimum drywell shell thickness is approximately 0.64 inches at this location. The drywell shell has an air gap of approximately 2 inches between the concrete drywell shield wall and the drywell shell. The thickness of the concrete at the point of impact is approximately 7 feet.

It was 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 (towards the turbine generator) because of the lack of a fluid energy reservoir. The break area is 360 square inches, and the reservoir pressure is 1050 psia. Three pipe force methodologies were used to calculate forces:

(1) Method 1 estimate (Bechtel)

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

(2) Method 2 estimate (Moody)

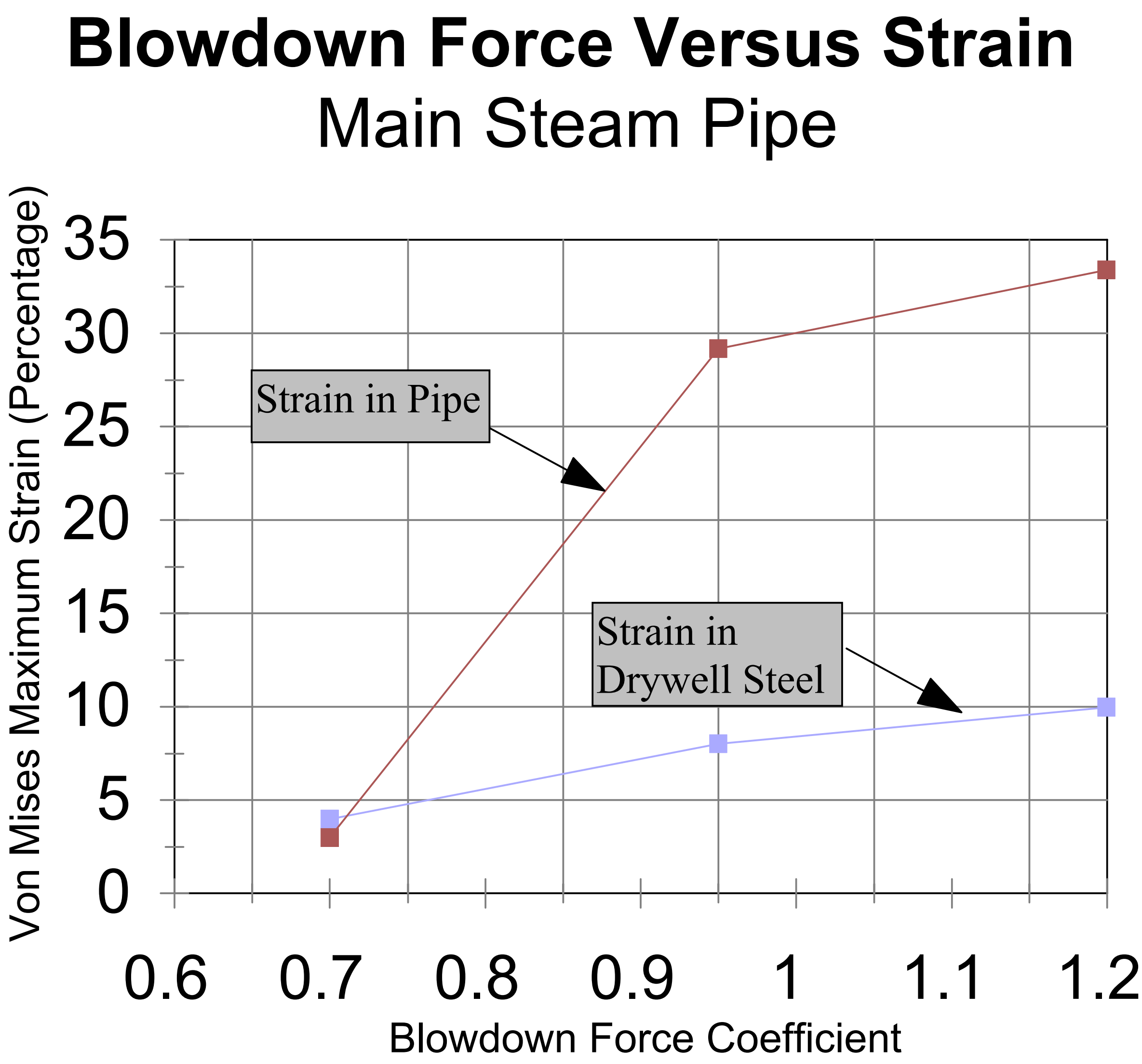
The average force on the main steam pipe from the reactor blowdown is 339,900 lbf. For this scenario, "C" in the equation $F = CP_0A$ is approximately 0.9.

(3) Method 3 estimate (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 outwards about one inch when the pipe impacts it with the lower bound force. Since the air gap between the drywell shell steel and the concrete shield wall is 2 inches, the drywell steel will not come into contact with the concrete shield wall. Local yielding of the drywell steel shell will occur at the point of impact with a maximum strain of 4 percent.

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 full set of break locations examined was as follows:

Break	Disposition
Main Steam Pipe adjacent to Reactor Pressure Vessel	The drywell wall will deflect one inch with the lower bound force estimate. It deflects & contacts the concrete at two points with the upper bound force estimate. The pipe does not cause perforation of the drywell in either instance.
Feedwater system pipe break adjacent to the reactor vessel	A plastic hinge will be formed after the feedwater pipe fails. The pipe will not impact the drywell steel shell located 24 inches away. However, the drywell steel shell would not be perforated even if the feedwater piping were to impact it.
Recirculation pumps 11, 13, & 14 discharge pipe break at the reactor vessel (BWR/2)	There is a gap of 168 inches between the pipe and the steel containment shell. After a guillotine break, the pipe will be propelled towards the containment shell. A plastic hinge will be formed and the pipe will fail before it impacts the containment wall.
Recirculation pumps 12 & 15 discharge pipe break at the reactor vessel (BWR/2)	Will first strike the primary containment spray and feedwater pipes. Not likely to impact the drywell wall, and even then, not likely to have enough energy to penetrate.
Recirculation pump discharge riser pipe at the reactor vessel (BWR/3&4)	Bounded by BWR/2 recirculation pump discharge pipe break at the reactor vessel because of smaller pipe size and reduced system flow.

This set of break locations covers all BWRs with a Mark I containment.

Likelihood of BWR containment perforation

Based on 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 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 BWR owners' group 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)."[3] Since approximately half of the high-pressure piping is located in this spherical portion, this would seem 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, this likelihood is, in reality, much smaller than 10%, since 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, the personnel and equipment hatches, and the steam and feedwater penetrations, generally would 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

Based on the above, it is concluded that there is insufficient basis for any regulatory action on the BWR plants within the scope of this generic issue.

PWR EVENTS

The three PWR events designated for further investigation are listed in the following table:

PWR event number	Description	Comments
9	A high energy line breaks within containment. Pipe whip or jet impingement causes failure of containment instrumentation and control, leading to failure of accident-mitigating systems.	
16	A break in the main steam or feedwater lines causes a penetration of the containment shell due to 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 event involves a break in the main steam or feedwater lines. Pipe whip causes failure of the component cooling water system.	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.

Event 16

Event 16, which is only applicable to two sites, was estimated by INEEL to have a core damage frequency of 1.4E-9 per reactor-year. The event 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 assessment contains some conservatism, the resulting number is down in the 10^{-9} range 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 not a LOCA but instead is a break in the secondary system.

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

Event 17

Event 17 was estimated by INEEL to have a frequency of 1.0E-7 events per reactor-year, which is right on the threshold for large early release frequencies. However, the INEEL estimate includes some conservatism:

- The INEEL assessment assumed that rupture of the CCW with a failure to isolate would lead to severe core damage. In reality, a break in the steam or feedwater 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 CVCS) were 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 water, 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 assessment assumed that a failure in either the upstream or the 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 backwards 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 frequency estimate by 25% or more.

For these reasons, it is concluded that a more realistic estimate of the frequency of event 17 would be well below the threshold of 10^{-7} per reactor-year for large early release frequency.

Event 9

This event involves a high energy line break inside containment. In this scenario, pipe whip or jet impingement causes failure of containment instrumentation and control, leading to failure of accident-mitigating systems. INEEL estimated a core damage frequency of up to 7.5E-5 per reactor-year for this event, which is well above the generic issue screening thresholds.

The various control components (e.g., MOVs) and instrumentation transducers (pressure sensors, temperature sensors, etc.) will be dispersed, since 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 instrumentation and control cables servicing these components were often field run in these older plants. Although there are generally 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 by high containment pressure. This signal will have redundant channels, and will be generated by pressure sensors that are connected to the primary containment free volume, but not located physically within it. Thus, the ECCS, auxiliary feedwater, etc., will be started regardless of damage to cabling within the containment. The safety significance of the lost cables will be in the long-term recovery, where the operator would monitor signals such as steam line pressure, steam generator level, 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, it is at the electrical penetrations where a pipe whip or fluid jet could disable a significant amount of instrumentation and control 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 loss-of-coolant accident 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 a main steam (MS) or feedwater (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 these early plants, a generic approach to this scenario is not possible. Instead, the layout of each plant was examined individually, 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 90º quadrant, the issue was not pursued further for that plant. For such a configuration, it is not credible for one pipe break to disable cabling at both penetrations. (More modern designs, based on the Standard Review Plan, 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 plant was examined further to see if a high pressure pipe were located within line-of-sight of the penetration area. If the pipes were far enough along the containment wall to be out of sight, or if there was an intervening wall, floor, or shield capable of stopping a fluid jet, the issue was not pursued further for that plant.

The results of this multi-plant investigation are summarized in the following table:

Plant	Physical arrangement	Conclusion
ANO-1	The main steam piping and containment penetrations are routed above the operating floor, away from the electrical penetration area. The feedwater 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 degrees apart and 90 degrees 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 main steam and feedwater 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 degrees from the electrical penetrations, but the MS & FW penetrations from two SGs are directly above cable trays. However, the licensee has an analysis that says the higher stress areas of the pipes are at the containment penetration and they will break there instead of elsewhere, and therefore the jet impingement will not impact the penetration area. From an examination of plant layout diagrams, the likely spray pattern will not impact the cables.	Acceptable
Kewaunee	Two sets of electrical penetrations are located approximately 90 degrees 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
Millstone 2	There are two electrical penetration areas 90 degrees 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 degrees apart.	Acceptable
Palisades	There are two containment penetration areas, separated by more than 90 degrees. 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 in the same quadrant and near the main steam and feedwater penetrations. Most of the penetrations are shielded from the piping by a concrete floor, but one bank of pressurizer heaters and one channel of the RTDs and the 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. Nevertheless, the licensee has agreed to examine the 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, these pipes are not high energy and the adjacent control system cables are not required to mitigate a break in one of these 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, it is concluded that Event 9 is not a significant safety problem at any PWR within the scope of this generic issue.

CONCLUSION

Based on the above, it is concluded that further pursuit of this generic issue is not justified. Therefore, it is recommended that Generic Issue 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, (1st Edition) November 1975, (2nd Edition) March 1980, (3rd Edition) July 1981.

(2) NUREG/CR-6395, "Enhanced Prioritization of Generic Safety Issue 156.6.1 Pipe Break Effects on Systems and Components Inside Containment," November 1999.

(3) BWROG report, NEDC-33054, "Conservatism in NRC Prioritization of Pipe Break Effects on Systems and Components," November 2001

(4) Information Systems Laboratories report, "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," December 2002

APPENDIX A

HISTORY OF GENERIC ISSUE 156.6.1

Systematic Evaluation Program

In 1977, the NRC initiated the Systematic Evaluation Program to review the designs of 51 older, operating nuclear power plants. The SEP was divided into two phases. In Phase I, the staff defined 137 issues for which regulatory requirements had changed enough over time to warrant an evaluation of those plants licensed before the issuance of the SRP. [A-1] In Phase II, the staff compared the design of 10 of the 51 older plants to the SRP [A-1] issued in 1975. (These plants were referred to as the “SEP-II plants.”) Based on these 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 SRP [A-1] was issued in 1975.

SEP-II PLANTS			
Plant	Vendor	Reactor	current status
Big Rock Point	GE	BWR	permanently shut down
Dresden 2	GE	BWR	operating
Ginna	Westinghouse	PWR	operating
Haddam Neck	Westinghouse	PWR	permanently shut down
LaCrosse	AC	BWR	permanently shut down
Millstone 1	GE	BWR	permanently shut down
Oyster Creek	GE	BWR	operating
Palisades	CE	PWR	operating
San Onofre 1	Westinghouse	PWR	permanently shut down
Yankee Rowe	Westinghouse	PWR	permanently shut down

In SECY-84-133, [A-2] the staff presented the 27 SEP issues to the Commission as part of a proposal for an Integrated Safety Assessment Program (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 these two pilot plants, ISAP was discontinued.

In SECY-90-160, [A-3] the staff forwarded for Commission approval a proposed license renewal rule and supporting regulatory documents. In this paper, the staff stated that certain unresolved safety issues could weaken the generic justification of the adequacy of the current licensing basis argument. These issues included SEP topics for 41 older plants that had not been explicitly reviewed under Phase II of the SEP. The Commission requested that the staff keep it informed of the status of the program to determine 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 the current licensing bases (CLBs) at all operating nuclear power plants, with the exception of age-related degradation, provide adequate protection to the 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 assure 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 in NUREG-1412, "Foundation for the Adequacy of the Licensing Bases." [A-4] However, as discussed in SECY-90-160, [A-3] the staff identified a potential weakness in the discussion of the adequacy of the CLB with regard to the 41 older plants which had not been included in the SEP-II list. (These plants were referred to as the "SEP-III" plants.) To address this potential weakness, the staff undertook an effort to determine whether or not each SEP issue either had been or was being addressed by other regulatory programs and activities.

SEP-III PLANTS			
Plant	Vendor	Reactor	current status
ANO-1	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
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 placed each SEP issue into one of the following categories: (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 so as to require no further regulatory action; (3) issues that were unresolved, but for which the staff had identified existing regulatory programs that cover the scope of the technical concerns and whose implementation would resolve the specific SEP issue, such as the Individual Plant Examination (IPE) and the Individual Plant Examination of External Events (IPEEE); and (4) issues that were unresolved and regulatory actions to resolve the issues had not been identified. The 27 SEP issues and applicable regulatory programs were summarized and presented in SECY-90-343. [A-5] 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. (However, it should be noted that, although these 41 plants are commonly called the "SEP-III" plants, there never was a formal third phase to the SEP. Instead, as will be described below, the issues were instead incorporated into the Generic Issues Program.)

GI-156.6.1 history

The Atomic Energy Commission (AEC) published the General Design Criteria (GDC) for interim use in 1967. Until 1972, the AEC staff's implementation of the GDC required consideration of postulated pipe break effects inside containment. However, due to the lack of documented review criteria, the AEC staff's review positions were continually evolving.

Review uniformity was finally developed in the early 1970s, initiated by a November 9, 1972, note from L. Rogers to R. Fraley, 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 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 use of these 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 [A-6] for implementation on a forward-fit basis only.

The AEC issued two generic letters to all licensees and CP or OL applicants regarding pipe break effects outside containment in December 1972 [A-7] and July 1973. These letters, known as the "Giambusso" and "O'Leary" letters, respectively, extended pipe break concerns to locations outside containment, and provided deterministic criteria for break postulation and evaluation of the dynamic effects of postulated breaks. The letters requested all recipients to submit a report to the staff summarizing each plant-specific assessment of the issue. All operating reactor licensees and license applicants submitted the requested assessments in separate correspondence or updated the SARs for their proposed plants to include the assessment. The staff reviewed the submitted assessments and prepared safety evaluations for all plants. In November 1975, the staff published SRP [A-1] 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 were judged to be a compliance issue and were not considered in this assessment. Compliance matters are dealt with promptly and do 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. [A-8] The remainder of this evaluation only addressed 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 [A-6] for postulated pipe breaks inside containment, and SRP [A-1] Sections 3.6.1 and 3.6.2 for outside containment. In July 1981, SRP [A-1] Sections 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, [A-6] which was subsequently withdrawn.

Between the period 1983-1987, the general issue of pipe breaks inside and outside containment was revisited in the SEP. 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 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 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 the Institute of Electrical and Electronics Engineers Standard 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.

The remaining question for both the pipe break criteria and the electrical systems design criteria was, of course, whether to reexamine the designs of the plants which were licensed before these criteria were finalized. This question was turned over to the Generic Issues Program in a memo from T. E. Murley (NRR) to E. S. Beckjord (RES) dated February 5, 1991. The issue was prioritized as “Medium” priority in 1994. In 1999, an enhanced prioritization, based on some 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, (1st Edition) November 1975, (2nd Edition) March 1980, (3rd Edition) July 1981.

(A-2) SECY-84-133, “Integrated Safety Assessment Program (ISAP),” March 23, 1984.

(A-3) SECY-90-160, “Proposed Rule on Nuclear Power Plant License Renewal,” May 3, 1990.

(A-4) NUREG-1412, "Foundation for the Adequacy of the Licensing Bases," U.S. Nuclear Regulatory Commission, 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,” October 4, 1990

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

(A-7) Letter to W. Dickhoner (The Cincinnati Gas & Electric Company) from A. Giambusso (NRC), December 18, 1972.

(A-8) Memorandum to A. Thadani from E. Beckjord, “Generic Issue 156-6.1, ‘Pipe Break Effects on Systems and Components,’” October 31, 1994.

APPENDIX B

STRUCTURAL EVALUATION
FOR
GENERIC ISSUE GI-156.6.1

1.0 PURPOSE

This document describes the details of the structural evaluation performed for resolution of Generic Issue (GI) 156.6.1, “Pipe Break Effects on Systems and Components Inside Containment.”

2.0 POSTULATED PIPE BREAKS

The impact of the following postulated pipe breaks inside a BWR drywell are evaluated in this report.

- Main steam system pipe line break at the reactor vessel nozzle.
- Feedwater system line break at the reactor vessel nozzle.
- Recirculation system pumps 11, 13, and 14 discharge line break at the reactor vessel nozzles.
- Recirculation system pump 12 and 15 discharge line break at the reactor vessel nozzle.

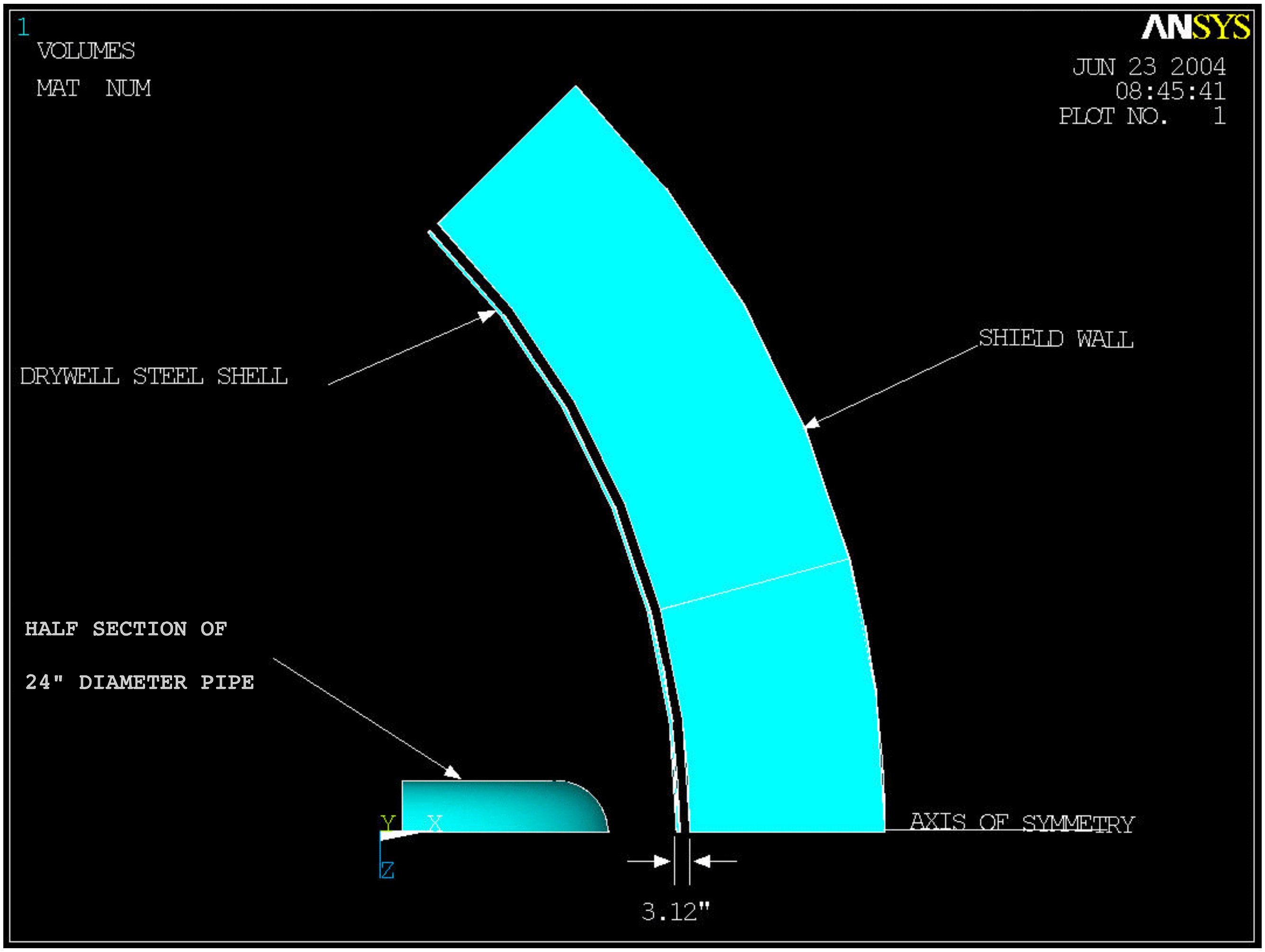
3.0 STRUCTURAL EVALUATION

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 1 and 2. The vertical leg of the steam line is located 16 inches away from the drywell steel shell containment. The minimum thickness of drywell steel containment for the different BWR’s is 0.64 inches. There is an air gap of 2 inches 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 7 feet.

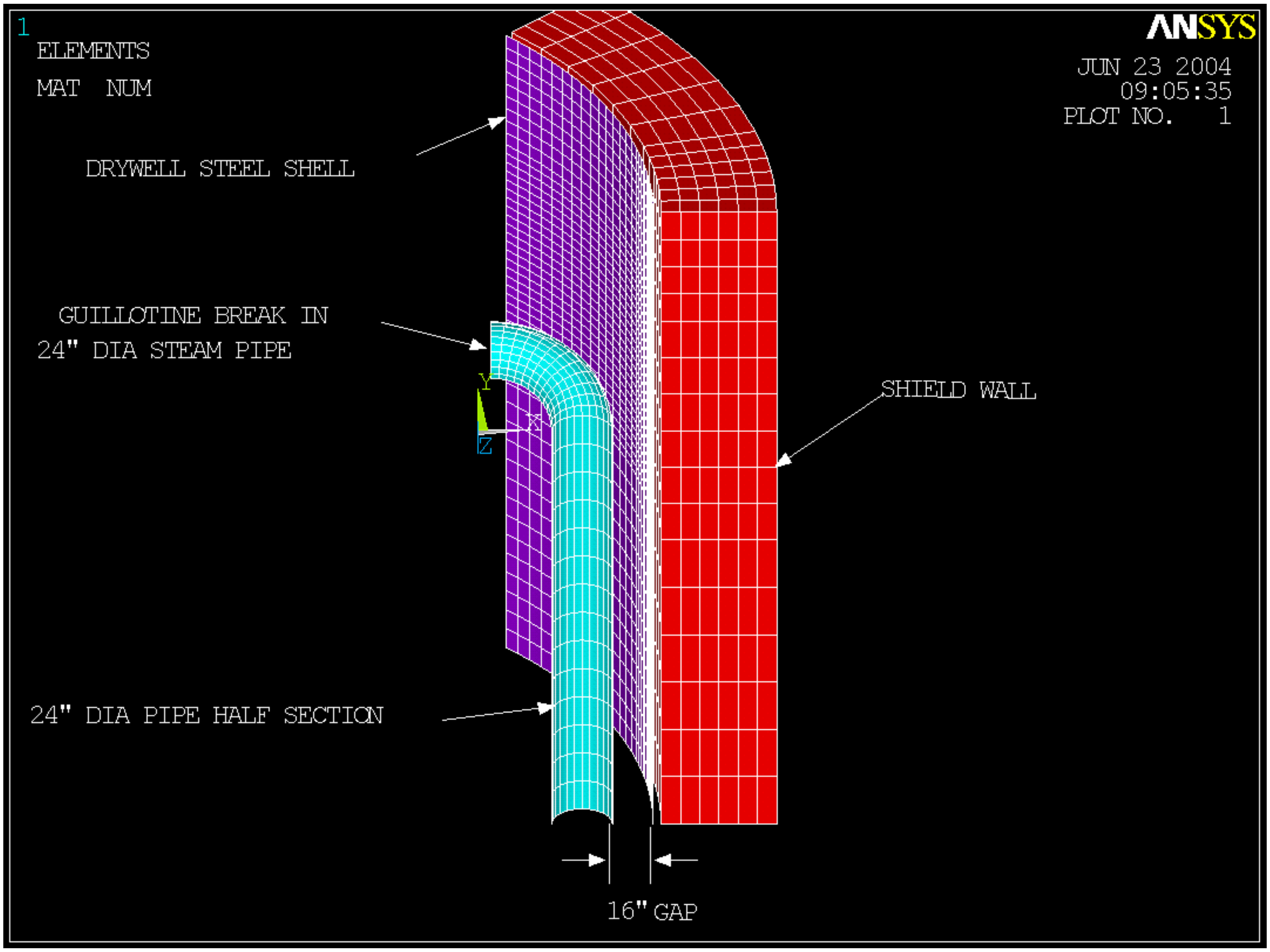
The main steam line is fabricated from ASTM A 106-Grade B material with an outside diameter of 24 inches and a wall thickness of 1.3 inches. The operating pressure (P_o) of the steam line is 1050 psi, and the pipe break area (A) is 360 square inches.

The main steam pipe at the postulated break location will be subjected to a blowdown force. This blowdown force will propel 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 16 inches until it impacts the drywell steel shell. The magnitude of this blowdown force varies with time and is dependent on the piping layout and configuration. Therefore, the impact of the main steam pipe on the drywell steel shell has been evaluated for the upper bound and lower bound values of blowdown force equal to 1.20 times P_oA (453000 pounds) and 0.70 times P_oA (264400 pounds), respectively. In addition, an air gap between the drywell steel shell and the concrete shield wall has been conservatively increased to 3.125 inches instead of 2 inches to account for construction tolerances.



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

<u>PLAN</u>
FIGURE 1



ELEVATION

FIGURE 2

Furthermore, the concrete shield wall thickness has also been reduced to 45 inches instead of 84 inches for structural analysis to ensure a conservative evaluation.

A nonlinear analysis of the main steam line impact on the drywell has been performed using the ANSYS finite element program. Figures 3-6 show 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 value of the blowdown force is 38 inches and 21 inches, respectively. The maximum Von Mises strain in the pipe due to the upper and lower bound blowdown force is 33 and 4 percent, respectively as compared to the minimum ultimate tensile strain of 22 percent for ASTM A106 grade B pipe material. Therefore, the pipe will deform and may fail locally. However, this local deformation or failure is not a concern since the pipe will already be damaged prior to impact.

The drywell steel shell will deflect and move radially outwards about one inch when the pipe impacts it with the lower bound blowdown force. Since the air gap between the drywell shell steel and the concrete shield wall is 2 inches, the drywell steel will not come into contact with the concrete shield wall. Local yielding of the drywell steel shell will occur at the point of impact with a maximum Von Mises total strain of 4 percent after steam pipe impact with the lower bound value of blowdown force.

The drywell steel shell will deflect, move radially outward, and come into contact with the concrete shield wall at two locations after the main steam pipe impacts it with the upper bound value of blowdown force. The maximum Von Mises total strain in the steel drywell shell is less than 10 percent as compared to a minimum tensile strain of 17 percent for the carbon steel drywell material. This will result in local yielding of the drywell steel shell. However, the drywell steel shell will not perforate. The variation of strain in the steam pipe and drywell steel shell as a function of blowdown force coefficient is shown in Figure 7. The compressive stresses in the concrete shield wall will remain within the 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 drywell steel 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. 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

A double-ended guillotine break in the feedwater line is postulated at the reactor nozzle. The vertical leg of the feedwater line is located 24 inches away from the drywell steel containment. The minimum thickness of drywell steel containment for the different BWR's is 0.64 inches. There is an air gap of 2 inches 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 7 feet.

The feedwater pipe is fabricated from ASTM A 106-Grade B material with an outside diameter of 10.75 inches and a wall thickness of 0.625 inches. The design operating pressure (P_o) of the feedwater line is 1050 psi, and the pipe break area (A) is 71 square inches. The minimum ultimate tensile strain and strength for ASTM A 106 Grade B steel pipe material are 22 percent and 60000 psi respectively.

The feedwater pipe at the postulated break location will be subjected to a blowdown force. This blowdown force will propel the feedwater pipe elbow toward the drywell and away from the reactor

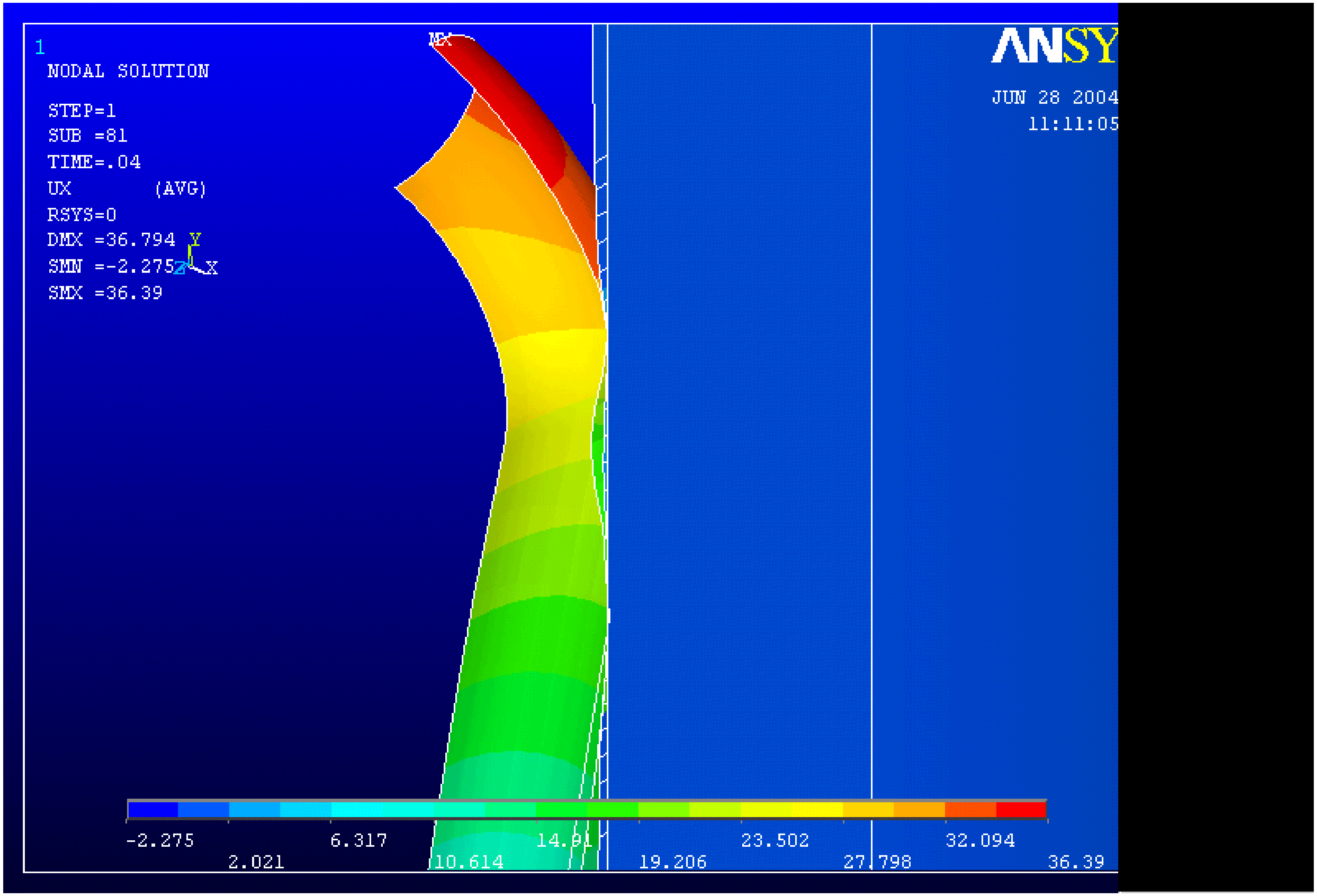
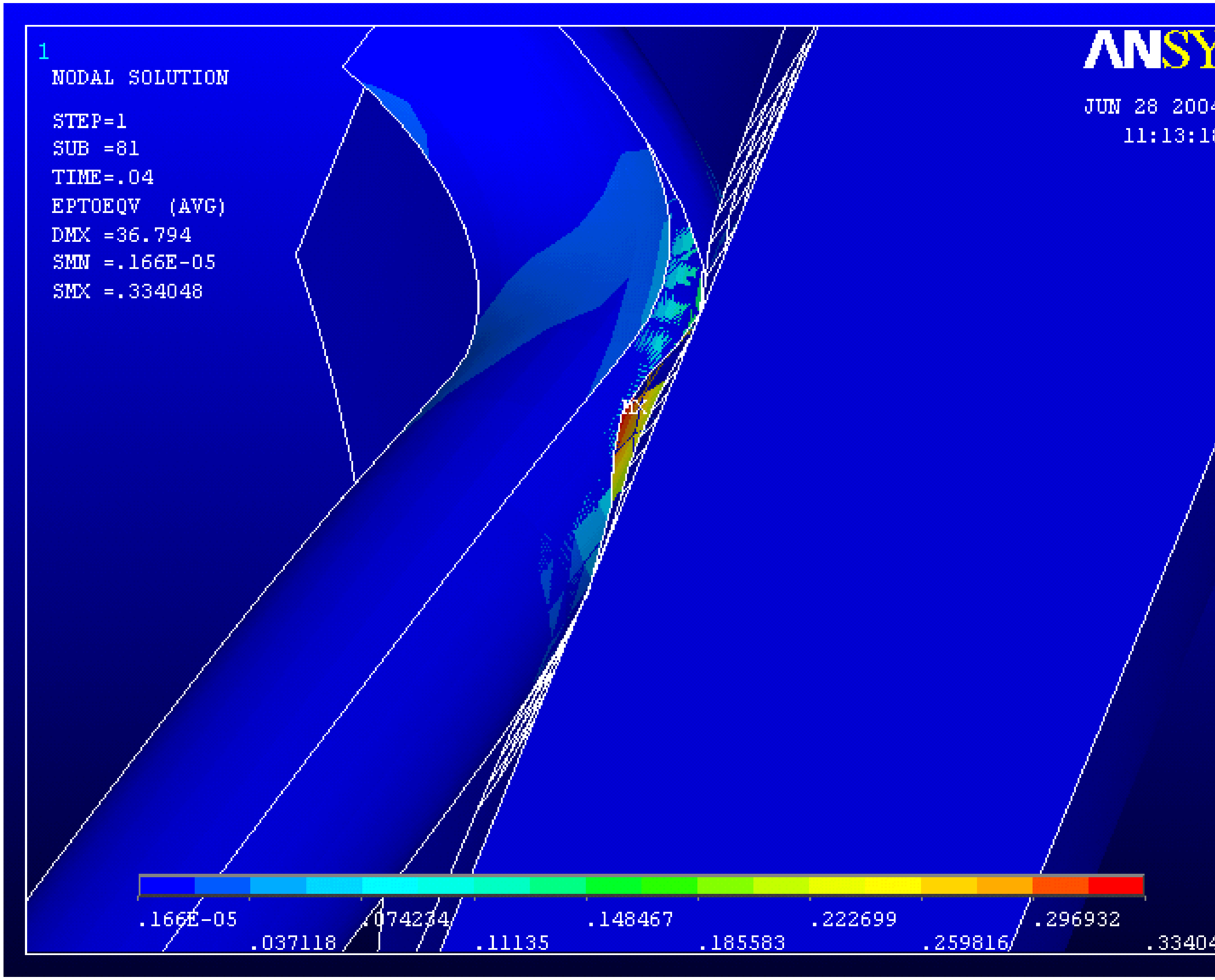
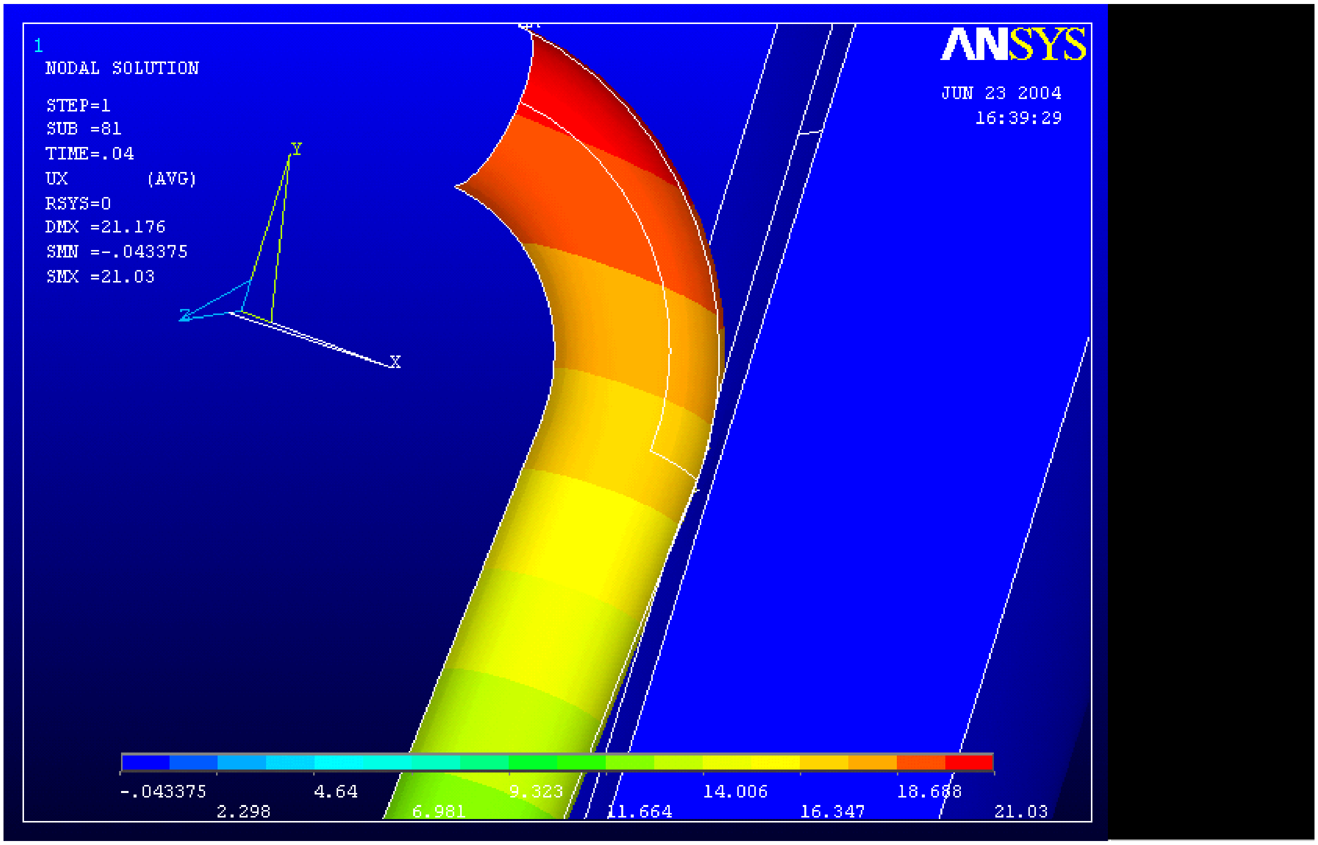


Figure 3 - DEFLECTED SHAPE FOR STEAM LINE IMPACT WITH UPPER BOUND BLOWDOWN FORCE



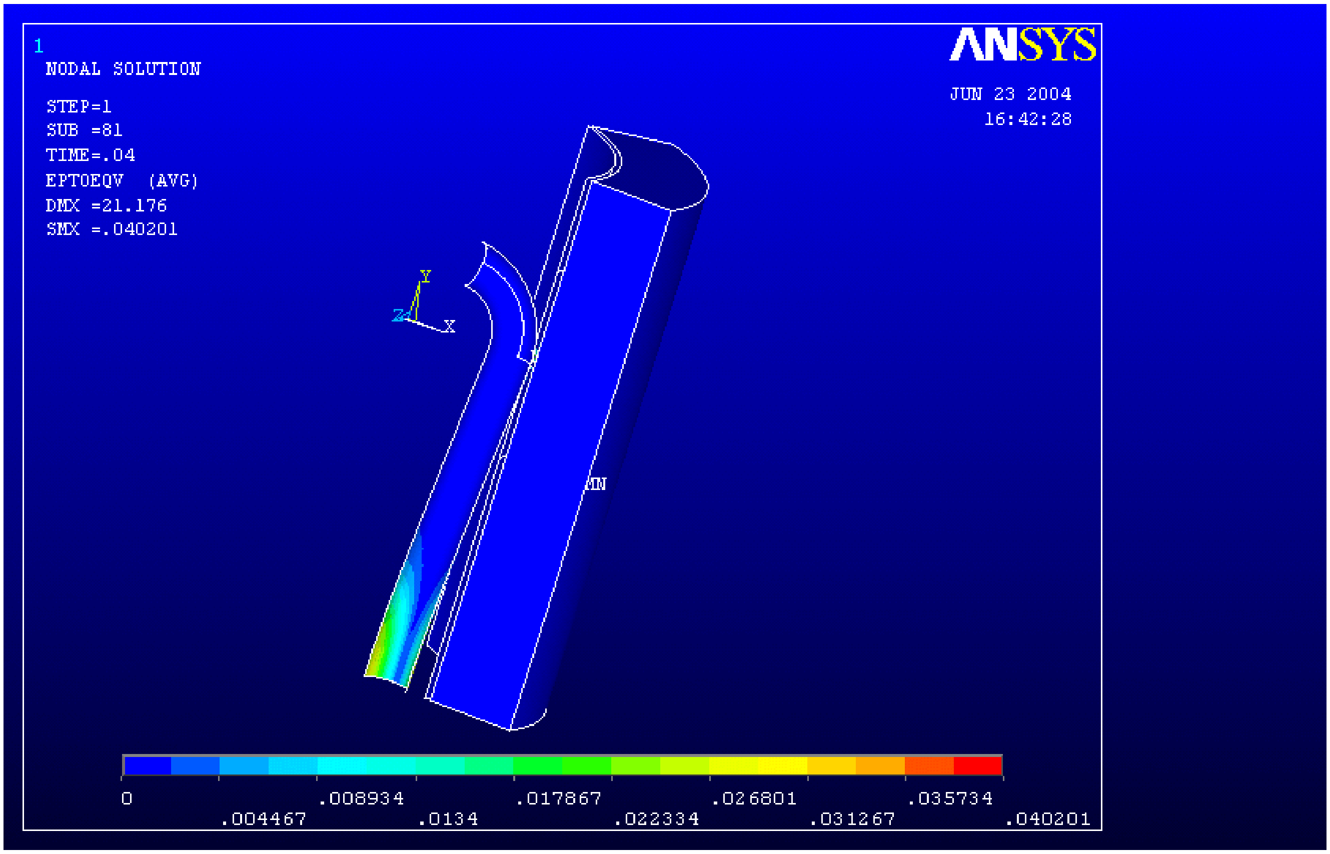
[redacted]

Figure 4 - MAXIMUM STRAIN FOR STEAM LINE IMPACT WITH UPPER BOUND BLOWDOWN FORCE



DEFLECTED SHAPE FOR
STEAM LINE IMPACT WITH LOWER BOUND BLOWDOWN FORCE

Figure 5



MAXIMUM STRAIN FOR
STEAM LINE IMPACT WITH LOWER BOUND BLOWDOWN FORCE

Figure 6

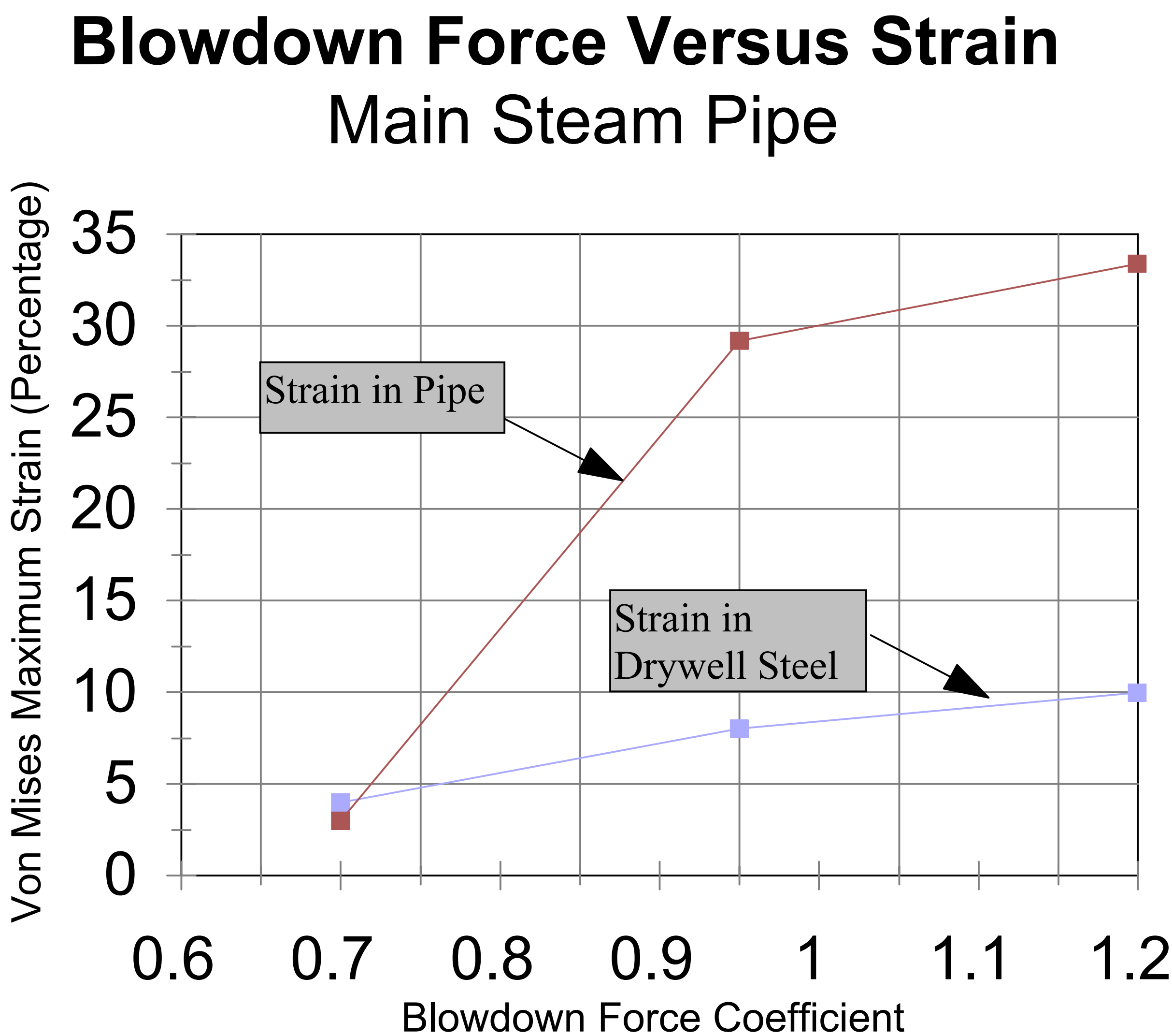

Blowdown Force Versus Strain
Main Steam Pipe
Von Mises Maximum Strain (Percentage)
35
30
25
20
15
10
5
0
Strain in Pipe
Strain in
Drywell Steel
0.6
0.7
0.8
0.9
1
1.1
1.2
Blowdown Force Coefficient

FIGURE 7

vessel. If a pipe whip restraint is not installed on the feedwater pipe line, the feedwater pipe will have to travel 24 inches before it impacts the drywell steel shell. The magnitude of this blowdown force varies with time and is dependent on piping layout and configuration. To ensure a conservative estimate of the total distance the feedwater pipe can travel prior to failure, an upper bound value of blowdown force equal to 2.1 times P_oA has been considered for the structural evaluation.

A detailed evaluation of the impact of the blowdown force on the feedwater pipe and on the drywell structure was performed. For the upper and lower bound values of the postulated blowdown force (156300 and 96750 pounds), and the maximum tensile strain equal to 1.75 times the minimum ultimate strain (38.5 percent) and ultimate tensile strength equal to 1.33 times the minimum ultimate strength (80000 psi) for ASTM A 106 pipe material, a plastic hinge will be formed at a distance of between 39 and 64 inches below the centerline of the elbow. The pipe will fail after the plastic hinge is formed. The pipe will not impact the drywell steel shell since the horizontal deflection in the pipe prior to failure will vary between 7 and 18 inches, which is less than the initial gap of 24 inches between the feedwater pipe and drywell steel shell.

The failed piece of feedwater pipe elbow will fall inside the drywell and may impact other components. In addition, even if we conservatively consider that the feedwater pipe will not fail from the blowdown force, and instead travel radially 24 inches to impact the drywell steel shell, it will not perforate it. The energy transferred at impact by the 10-inch diameter feedwater pipe traveling 24 inches is much less than the energy transferred by a 24-inch diameter steam pipe traveling 16 inches. As described in Section 3.1, the steam pipe will not perforate the drywell steel shell on impact. Therefore, the structural integrity of the drywell steel containment will not be compromised due to a pipe break in feedwater pipe line at the reactor vessel nozzle.

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

A double ended guillotine break in the recirculation (RC) discharge lines for pumps 11, 13, and 14 is postulated at the reactor vessel nozzles. The vertical leg of the RC lines is located 168 inches away from the drywell steel shell containment. The minimum thickness of the drywell steel containment for the different BWRs is 0.64 inches. There is an air gap of 2 inches 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 7 feet.

The RC line is fabricated from stainless steel type 304/304L material with an outside diameter of 28.0 inches and a wall thickness of 1.50 inches. The operating pressure (P_o) of the RC line is 1050 psi, and the pipe break area (A) is 491 square inches. The minimum ultimate tensile strain and strength for type 304/304L type stainless steel pipe material are 40 percent and 75000 psi, respectively.

The RC pipe at the postulated break location will be subjected to a blowdown force. This blowdown force will propel the RC line pipe elbow radially outward toward the drywell unless a pipe whip restraint has been installed on the RC pipe line to prevent this. The RC pipe elbow will have to travel 168 inches before it can impact the drywell steel shell. The magnitude of this blowdown force varies with time and is dependent on piping layout and configuration. To ensure a conservative approach, an upper and a lower bound value of blowdown force equal to 2.2 x P_oA and 1.30 x P_oA, respectively, have been considered for the structural evaluation.

A detailed evaluation of the impact of blowdown force on RC pipe and on drywell structure has been performed. For the upper and lower bound values of postulated blowdown force (1130000 and 638000 pounds), and the maximum tensile strain equal to 1.75 times the minimum ultimate strain (70 percent) and ultimate tensile strength equal to 1.2 times the minimum ultimate strength (90000 psi) for stainless steel type 304/304L material, a plastic hinge will be formed at a distance of between 100 and 178 inches from the centerline of the elbow. The pipe will fail after the plastic hinge is formed. The RC pipe will not impact the drywell steel shell since the maximum deflection in the pipe prior to failure will vary between 34 and 107 inches, and the initial gap between the RC pipe and drywell steel shell is 168 inches. The failed piece of RC pipe elbow will fall inside the drywell and may impact other components. However, the structural integrity of the drywell steel containment will not be compromised.

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

A double ended guillotine break in the RC discharge lines for pump No. 12 is postulated at the reactor vessel nozzles. The vertical leg of the RC lines is located 168 inches away from the drywell steel shell containment. The minimum thickness of drywell steel containment for the different BWRs is 0.64 inches. There is an air gap of 2 inches 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 7 feet.

The RC line is fabricated from stainless steel type 304/304L material with an outside diameter of 28.0 inches and a wall thickness of 1.50 inches. The operating pressure (P_o) of the RC line is 1050 psi, and the pipe break area (A) is 491 square inches. The upper bound values of the minimum ultimate tensile strain and strength for stainless steel type 304/304L pipe material are 40 percent and 75000 psi respectively.

The RC pipe at the postulated break location will be subjected to a blowdown force. This blowdown force will propel the RC line 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 will travel radially and break the primary containment spray (PCS) and feedwater lines located in its path before it can impact the drywell steel shell located 168 inches away.

The magnitude of this blowdown force varies with time and is dependent on piping layout and configuration. To ensure a conservative approach, an upper and a lower bound value of blowdown force equal to 2.2 x P_oA and 1.30 x P_oA respectively have been considered for the structural evaluation.

The elbow of the RC pipe will strike the 6-inch diameter Primary Containment Spray (PCS) line located 9 inches away. There is a possibility that the RC pipe will break the PCS pipe line and continue to travel outwards. The next obstruction in the outward travel is the feedwater line, which is located approximately 72 inches away from the PCS pipe line. The RC pipe may also impact and break the feedwater pipe. However, the RC pipe will not impact the drywell steel shell since the maximum deflection in the pipe prior to failure will vary between 34 and 107 inches, and the initial gap between the RC pipe and drywell steel shell is 168 inches. The failed pieces of the feedwater, RC and PCS pipe lines will fall inside the drywell and may impact other components. However, the structural integrity of the drywell steel containment will not be compromised.

4.0 CONCLUSION

The structural integrity and leak tightness of the drywell steel containment shell and the shield wall will not be compromised due to postulated breaks in the main steam, feedwater, 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 feedwater, RC, and PCS pipe lines may break and fall onto other components inside the drywell.